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SOUTHERN COMPANY
Energy to Serve Your World℠

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

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

Ladies and Gentlemen:

On January 7, 2004, Southern Nuclear Operating Company (SNC) and NRC held a teleconference to discuss the SNC Technical Specifications (TS) submittal for the inclusion of Linear Heat Generation Rate (LHGR) monitoring, dated October 3, 2003. In the call, NRC indicated the need for additional information concerning the methodology used for the calculation of the LHGR low flow and low power multipliers. NRC further suggested that the information be included in the Technical Specifications Bases. Subsequent to that call, the NRC requested more information concerning the use of total peaking factors and the definition of limiting control rod pattern.

Another teleconference between SNC and NRC personnel was held on January 30, 2004, to discuss the SNC responses to those questions. This letter serves to formally document those responses.

Enclosure 1 provides the response to the NRC questions on the LHGR multiplier methodology, total peaking factor, and limiting control rod pattern. Enclosure 2 contains the 10 CFR 50.92 evaluation and the justification for the categorical exclusion from performing an environmental assessment. Enclosure 3 contains proposed TS pages which add a definition of LHGR to section 1.1 of the Units 1 and 2 TS. These pages are an addition to the TS pages already provided in the October 3, 2003 letter. Enclosure 4 provides the entire Bases changes which support the LHGR TS addition with the proposed modifications. These Bases pages supersede the previously provided Bases pages from the October 3, 2003 submittal.

A001

Finally, attached is a letter from Global Nuclear Fuel to Southern Nuclear (Enclosure 5), detailing the requested information about the LHGR multipliers. The technical justification provided in the October 3, 2003 letter remains valid for this change. A new 10 CFR 50.92 and environmental evaluation has been performed.

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 me and subscribed before me this 9th *day of* Feb., *2004.*

Valerie A O'Brien

Notary Public

My commission expires: 4/28/07

HLS/OCV/daj

Enclosures:
1 – Answers to NRC Questions
2 – No Significant Hazards and Environmental Assessment
3 – Marked up and clean typed TS pages
4 – Marked up and clean typed Bases pages
5 – Global Nuclear Fuel Letter

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., 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

Enclosure 1
Edwin I. Hatch Nuclear Plant
Supplement to Technical Specifications Revision to Include LHGR Monitoring

Responses to NRC Questions

In a teleconference between Southern Nuclear (SNC) and NRC personnel held on January 7, 2004, NRC requested information on the LHGR multipliers for adjusting the LHGR limits to low power and flow conditions. These multipliers will be used as part of the new proposed LHGR Technical Specifications (TS) limit, and serve to adjust the limit under low flow and low power conditions. Specifically, information was requested on the methodology used to calculate the multipliers, and where such methodology is documented. Subsequent to the call, SNC requested clarification on these issues from Global Nuclear Fuel (GNF). Accordingly, a letter has since been provided to SNC by GNF with these answers, and is provided as an attachment to this submittal. The proposed Bases changes provided in this letter contain clarification and elaboration on the MAPLHGR and LHGR multipliers beyond those given in the Bases pages from the October 3, 2003 submittal. Those changes are found primarily on pages B 3.2-1 and B 3.2-10 of the proposed TS Bases. The GNF letter is also being added as a reference to these Bases sections, and will also be added as a reference to the Core Operating Limits Report (COLR).

NRC also requested information on why the concepts of total peaking factor and limiting control rod pattern were eliminated from the Hatch TS. That information is provided below:

The use of total peaking factor was eliminated from the Hatch TS with the Average Power Range Monitor, Rod Block Monitor, Technical Specifications Improvement (ARTS) program (Ref. 1) which predates Amendment 19 of GESTAR (Ref. 2). Reference 1 discusses in Section 3.1 why the APRM set down, based on a derivative of total peaking factor, is no longer needed with ARTS implementation. Since the use of a total peaking factor was deleted from the Hatch Technical Specifications prior to the Amendment 19 deletion of LHGR, it does not need to be reinstated now.

The limiting control rod pattern concept for LHGR was deleted with Amendment 19 of GESTAR (Ref. 2), since it allowed LHGR to be removed entirely from the TS. However, Plant Hatch elected to keep the LHGR TS and its surveillance requirement related to limiting control rod pattern until the change to the Standard Technical Specification (STS) (Ref. 3). At that time, the entire LHGR section was removed from the TS altogether as were the additional surveillance requirements for a limiting control rod pattern on MAPLHGR and MCPR. Note that if SNC had elected to keep the optional STS LHGR specification, the definition of limiting control rod pattern and the associated surveillance requirement would have been deleted anyway because they are not contained in the STS (NUREG 1433). Since the concept of limiting control rod pattern for these limits and associated surveillance requirements were deleted with the STS, there is no need to reinstate this concept with the re-addition of the LHGR TS.

Response to NRC questions (continued)

References:

1. NEDC-30474-P, General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specifications Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Unit 1 and 2, December 1983.

2. Letter from A.C. Thadani (NRC) to J.S. Charnley (GE), Acceptance for Referencing of Amendment 19 to General Electric Licensing Topical Report NEDE-24011-P-A (GESTAR – II), "General Electric Standard Application for Reactor Fuel, " April 7, 1987, November 17, 1987.

3. Letter from K.N. Jabbour (NRC) to J.T. Beckham, Jr. (GPC), "Issuance of Amendments – Edwin I. Hatch Units 1 and 2 (TAC nos. M87310 and M87311)", March 3, 1995.

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

No Significant Hazards Evaluation and Environmental Assessment

Proposed Change

A definition of Linear Heat Generation Rate (LHGR) is proposed to be added to Definitions, Section 1.1 of the Technical Specifications (TS).

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 the LHGR definition to Section 1.1 is needed to support the new TS and Bases sections for LHGR. 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 Bases changes contain clarification and elaboration on the MAPLHGR and LHGR multipliers and are shown in APLHGR Section Bases B 3.2.1 and LHGR B 3.2.3. This information was requested by the NRC, who also suggested that the information be included in the TS Bases. The Bases pages also reflect the changes proposed in the October 3, 2003 submittal to the NRC (NL-03-1842).

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

No Significant Hazards Evaluation and Environmental Assessment

10 CFR 50.92 Evaluation (continued)

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 Section 1.1, 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 not defined in Section 1.1.

The margin of safety is not reduced since this is an administrative change only.

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.

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

No Significant Hazards Evaluation and Environmental Assessment

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
Supplement to Technical Specifications Revision to Include LHGR Monitoring

Marked up and Clean Typed TS Pages

1.1 Definitions (continued)

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

add definition of LINEAR HEAT GENERATION RATE from next page

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

(continued)

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually six inches. It is the integral of the heat flux over the heat transfer area associated with the unit length.

1.1 Definitions

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

(continued)

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually six inches. It is the integral of the heat flux over the heat transfer area associated with the unit length.

1.1 Definitions (continued)

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE	LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually six inches. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

(continued)

1.1 Definitions

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

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Enclosure 4
Edwin I. Hatch Nuclear Plant
Supplement to Technical Specifications Revision to Include LHGR Monitoring

Marked up and Clean Typed Bases Pages

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, 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 fuel design limits~~ during the limiting ~~AOOs~~ LOCA (Refs. ~~2~~, 6, and ~~7~~). 1, 3 & 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.~~

to 10 CFR 50.46

~~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~~.

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

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$_f$ factor~~s~~ 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).~~

For single recirculation loop operation, the MAPFAC$_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR).

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Some off-rated conditions require the reduction or set down of the rated APLHGR limit through multiplier factors (MAPFACs). A flow dependent multiplier, $MAPFAC_f$, is necessary at core flows below 61 % to provide protection for LOCA events (Ref. 12).

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

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 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

INSERT Ref. 12 from next page

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

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant 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 pellet and cladding.

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

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

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

Writer's note: This Bases section is to be included in both Units 1 and 2 TS Bases

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

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

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

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

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

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

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

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR limit calculated to cause a 1 % fuel cladding plastic strain as well as the other design limits described in 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.

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 ~~trancients~~ (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 LHGR 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

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.

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

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-~~31376~~ 31720P, "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 fuel design limits~~ during the limiting ~~AOOs~~ LOCA (Refs. ~~2~~, 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.~~

to 10 CFR 50.46

~~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

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For single recirculation loop operation, the MAPFAC$_f$ 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.

a maximum value specified in the Core Operating Limits Report (COLR)

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~~ LOCA ~~design, DBA, and transient~~ analyses. ~~For two recirculation loops operating,~~ the limit is determined by multiplying the ~~smaller of the MAPFAC$_p$ and~~ MAPFAC$_f$ factor~~s~~ 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).~~

For single recirculation loop operation, the MAPFAC$_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR).

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Some off-rated conditions require the reduction or set down of the rated APLHGR limit through multiplier factors (MAPFACs). A flow dependent multiplier, $MAPFAC_f$, is necessary at core flows below 61 % to provide protection for LOCA events (Ref. 12).

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.

INSERT Ref. 12 from next page

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

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

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

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

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

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 Reference 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

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.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

The APLHGR limits specified in the COLR are the result of the LOCA analyses. The limit is determined by multiplying the $MAPFAC_f$ factor times the exposure dependent APLHGR limits. For single recirculation loop operation, the $MAPFAC_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

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.

(continued)

BASES

ACTIONS (continued)

B.1

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

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

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

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. (Not used)
3. FSAR, Chapter 6.
4. FSAR, Chapter 15, 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.

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

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

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

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

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

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

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel."

2. FSAR, Chapter 15 (Unit 2).

3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.

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

5. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E. I. Hatch Nuclear Plant, Units 1 and 2," December 1983.

6. NRC approval of "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II"—Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.

7. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," August 1986, and NEDE-24154-P-A, Supplement 1, Volume 4, Revision 1, February 2000.

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transient analyses of Chapter 15 of the 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 MCPR setpoints for single loop operation are specified in the COLR. The APRM Simulated Thermal Power - High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

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

LCO

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

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

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

A.1

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

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

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

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

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

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

SR 3.4.1.2

(Not used.)

REFERENCES

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

2. FSAR, Section 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.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

The APLHGR limits specified in the COLR are the result of the LOCA analyses. The limit is determined by multiplying the $MAPFAC_f$ factor times the exposure dependent APLHGR limits. For single recirculation loop operations, the $MAPFAC_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

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.

(continued)

BASES

ACTIONS (continued)

B.1

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

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

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

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

REFERENCES

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

(continued)

BASES

REFERENCES (continued)

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

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

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

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

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

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

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

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

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

(continued)

BASES (continued)

LCO

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

APPLICABILITY

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

ACTIONS

A.1

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

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

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

2. FSAR, Chapter 15 (Unit 2).

3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.

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

5. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E. I. Hatch Nuclear Plant, Units 1 and 2," December 1983.

6. NRC approval of "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II"—Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.

7. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," August 1986, and NEDE-24154-P-A, Supplement 1, Volume 4, Revision 1, February 2000.

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

LCO

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

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

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

A.1

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

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

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

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

B.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

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

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

SR 3.4.1.2

(Not used.)

REFERENCES

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

2. FSAR, Section 5.5.1.4.

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

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

Enclosure 5
Edwin I. Hatch Nuclear Plant
Supplement to Technical Specifications Revision to Include LHGR Monitoring

Attachment

Letter from Global Nuclear Fuel To Southern Nuclear

GNF
Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

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January 22, 2004

E. B. Gibson
Hatch Core Analysis
Southern Nuclear Operating Company
P.O. Box 1295
Birmingham, AL 35201-1295

Subject: Plant Hatch Technical Specification Modification to include LHGR

Reference:

1) NEDC-30474-P, General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Unit 1 and 2, December 1983.

2) NEDE-24011-P-A-14, General Electric Company "General Electric Standard Application for Reactor Fuel, GESTAR II", and NEDE-24011-P-A-14-US, "General Electric Standard Application for Reactor Fuel (Supplement for US)", June 2000.

Dear Mr. Gibson

This letter is intended to provide Southern Nuclear Operating Company with information to support the modification of Plant Hatch Technical Specifications. GNF was asked to provide some additional information regarding the application of off-rated limits curves from the ARTS report (reference 1) to Linear Heat Generation Rate (LHGR) and to provide clarification regarding the application of Single Loop Operation (SLO) and the low flow ECCS limit multipliers to LHGR and Maximum Average Planar LHGR (MAPLHGR). These issues can be somewhat intertwined, so let us examine each one to provide clarification.

1) The need for this Technical Specification modification arises out of the change in the plant process computer from PANAC10 monitoring methodology to PANAC11 methodology. Currently, the thermal-mechanical limits curve is included in the MAPLHGR limits at Hatch. As a part of upgrading the nuclear methods, capability to explicitly monitor the thermal-mechanical limits curve separately from MAPLHGR was included. By separating the two limits, more accurate reporting and evaluation of limiting conditions will be provided. GNF submitted Amendment 26 to GESTAR II (reference 2) to provide for the use of LHGR to monitor thermal-mechanical limits in lieu of MAPLHGR (see section 2.2 of reference 2). MAPLHGR would continue to be used to monitor LOCA-related limits.

   In addition to monitoring thermal-mechanical limits, the LHGR limit is still used 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. This is because the LOCA analysis uses the thermal-mechanical limits curve as a bounding assumption on peak LHGR.

2) For AOO applications, the purpose of the power and flow dependent multipliers (MAPFAC or LHGRFAC) is to protect the fuel thermal-mechanical limits at off-rated conditions. Reference 1, Section 3.3.2 explicitly documents that the MAPLHGR factor assures compliance with the fuel thermal-mechanical design bases. At the time Reference 1 was implemented at Plant Hatch, the thermal-mechanical limits were part of the MAPLHGR. Therefore, the multipliers were presented as MAPLHGR factors (or MAPFAC) to protect the fuel thermal-mechanical limits. As stated in 1) above, with the implementation of PANAC11 methodology as the core monitoring program, the thermal-mechanical limits will no longer be monitored by MAPLHGR, instead they will by monitored exclusively by the LHGR.

   Therefore, to protect fuel thermal-mechanical limits at off-rated conditions, the multipliers must be applied to the LHGR in the same way as MAPFACs were applied to the MAPLHGR to protect the thermal-mechanical bases. These multipliers are called LHGRFAC to identify that they are factors on the LHGR.

   Except as described in item 4 below, the LHGRFAC is exactly the same as the MAPFAC described in Reference 1. The LHGRFAC (or MAPFAC) is calculated in the same way with the transient analysis methods documented in GESTAR II. The methodology (including the formulas) for calculating LHGRFAC(p) and LHGRFAC(f) is the same as it is for calculating MAPFAC(p) and MAPFAC(f), respectively, for all operating conditions. The difference in the name is to indicate to which limit the multiplier applies to be consistent with the thermal-mechanical basis.

3) Single Loop Operation (SLO) has a separately calculated multiplier. This multiplier is intended to protect the plant from potential ECCS-LOCA consequences during SLO. As long as the thermal-mechanical limits curve is more bounding, the multiplier must be applied to LHGR. If MAPLHGR is more limiting, then it would be necessary to apply this limit to MAPLHGR as well. It is our recommendation at this time to apply the multiplier to both.

4) With the changes identified above, the need for MAPFAC (p) has been eliminated. However, section 3.2.2.1 of reference 1 indicates there is still a need for a conservative, low flow MAPFAC(f) multiplier at flows below 61% to provide protection for ECCS-LOCA events. Future ECCS-LOCA analyses could eliminate this requirement.

If you have any questions, or need further clarification, please feel free to call me at 910.675.5762 or on my cell phone at 910.547.1042.

Sincerely,

Margaret E. Harding

Margaret E. Harding
Manager, Fuel Engineering Services
Global Nuclear Fuel