Progress Energy

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JAN 2 2 2007

SERIAL: BSEP 06-0129 10 CFR 50.90
TSC-2006-06

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62
Request for License Amendments Regarding Linear Heat Generation Rate and Core Operating Limits Report References for AREVA NP Fuel

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., is requesting a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments revise: (1) Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," to add references to AREVA NP analytical methods that will be used to determine core operating limits, and (2) add a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," add a new definition to Technical Specification 1.1 for LHGR, and revise Technical Specifications 3.4.1 and 3.7.6 to incorporate restrictions on LHGR when in single recirculation loop operation or with an inoperable Turbine Bypass System. An evaluation of the proposed license amendments is provided in Enclosure 1.

CP&L has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change involves no significant hazards considerations.

CP&L is providing, in accordance with 10 CFR 50.91(b), a copy of the proposed license amendment to the designated representative for the State of North Carolina.

CP&L requests approval of the proposed amendments by January 30, 2008. Once approved, the Unit 1 amendment shall be implemented prior to start-up from the 2008 Unit 1 refueling outage and the Unit 2 amendment shall be implemented prior to start-up from the 2009 Unit 2 refueling outage.

There are no regulatory commitments associated with this submittal. Please refer any questions regarding this submittal to Mr. Randy C. Ivey, Manager - Support Services, at (910) 457-2447.

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I declare, under penalty of perjury, that the foregoing is true and correct. Executed on January 22, 2007.

Sincerely,

James Scarola

James Scarola

WRM/wrm

Enclosures:

1. Evaluation of Proposed License Amendment Requests
2. AREVA NP BWR Approved Topical Reports for Brunswick Units 1 and 2 Technical Specification COLR References
3. GNF BWR Approved Topical Reports for Brunswick Units 1 and 2 Technical Specification COLR References
4. Marked-up Technical Specification Pages - Unit 1
5. Typed Technical Specification Pages - Unit 1
6. Typed Technical Specification Pages - Unit 2
7. Marked-up Technical Specification Bases Pages - Unit 1 (For Information Only)

cc (with enclosures):

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Evaluation of Proposed License Amendment Requests

Subject: Request for License Amendments Regarding Linear Heat Generation Rate Requirements and Core Operating Limits Report References for AREVA NP Fuel

1.0 Description

This letter is a request by Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., to amend the Technical Specifications of Renewed Operating Licenses DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

The proposed license amendments will revise: (1) Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," to add references to AREVA NP analytical methods that will be used to determine core operating limits, and (2) add a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," add a new definition to Technical Specification 1.1 for LHGR, and revise Technical Specifications 3.4.1 and 3.7.6 to incorporate restrictions on LHGR when in single recirculation loop operation or with an inoperable Turbine Bypass System.

These proposed Technical Specification changes are needed to support the transition to AREVA NP fuel, and to AREVA NP core design and analysis services. CP&L is planning to begin using AREVA NP fuel starting with the BSEP, Unit 1 refueling outage scheduled for March 2008 and the BSEP, Unit 2 refueling outage scheduled for February 2009.

2.0 Proposed Changes

Core Operating Limits Report Methodologies

Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," item b, is being revised as shown below. The specific changes to Technical Specification 5.6.5, item b, are shown in bold typeface.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

2. **XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.**

3. **XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.**

4. **EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.**

5. **ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.**

6. **XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.**

7. **XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.**

8. **EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.**

9. **XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.**

10. **XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.**

11. **ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.**

12. **ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.**

13. **ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors**

14. **EMF-2209(P)(A), SPCB Critical Power Correlation.**

15. **EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.**

16. **EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.**

17. **EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.**

18. **EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2.**

19. **NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.**

Linear Heat Generation Rate

First, a new definition for LHGR is being added to Technical Specification 1.1, "Definitions."

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

Second, a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," is being added, as shown below.

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any LHGR not within limits.	A.1	Restore LHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP AND 24 hours thereafter

Third, Technical Specification 3.4.1, "Recirculation Loops Operating," is being revised as shown below. The specific changes to Technical Specification 3.4.1 are shown in bold typeface.

3.4.1 Recirculation Loops Operating

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

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

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

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

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

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

Fourth, Technical Specification 3.7.6, "The Main Turbine Bypass System," is being revised as shown below. The specific changes to Technical Specification 3.7.6 are shown in bold typeface.

3.7.6 The Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; ~~**and**~~

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR**; and**

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

Fifth, Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," item a, is being revised as shown below. The specific changes to Technical Specification 5.6.5, item a, are shown in bold typeface.

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

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

4. The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

Enclosure 4 contains a marked-up version of the Unit 1 Technical Specifications showing the proposed changes. Since the affected Technical Specification Sections (i.e., 1.1, 3.2.3, 3.4.1, 3.7.6, and 5.6.5 for Unit 1 and Unit 2 are identical, only the mark-up for Unit 1 is provided. Enclosures 5 and 6 provide typed versions of the affected Unit 1 and Unit 2 Technical Specification pages, respectively. These typed Technical Specification pages are to be used for issuance of the proposed amendments.

CP&L will make supporting changes to the Technical Specification Bases in accordance with Technical Specification 5.5.10, "Technical Specifications (TS) Bases Control Program." Enclosure 7 provides a markup of the affected Technical Specification Bases pages for Unit 1. These pages are being submitted for information only and do not require issuance by the NRC.

3.0 Background

On March 10, 2006, CP&L held a public meeting with the NRC to discuss plans to change the Brunswick Unit 1 and 2 reload fuel supplier from Global Nuclear Fuels – Americas (GNF-A) to AREVA NP. A summary of this meeting was issued by the NRC on April 27, 2006 (i.e., Reference 1). As presented during the public meeting, CP&L plans included submittal of at least two separate license amendment applications to request the Technical Specification changes needed to support loading and use of the ATRIUM™-10 fuel assemblies.

This letter provides the proposed Technical Specification changes necessary to incorporate the AREVA NP analytical methods, previously approved by the NRC, that will be used to establish core operating limits. Under a separate submittal (i.e., Reference 2), CP&L has submitted proposed Technical Specification changes needed to support the receipt and storage of new AREVA NP fuel bundles. If changes to the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) values are required to support operation with the ATRIUM™-10 fuel, CP&L anticipates submittal of those MCPR SL changes in July 2007.

CP&L plans to begin using AREVA NP fuel beginning with the March 2008 Unit 1 refueling outage; therefore, CP&L requests that these Technical Specification changes be issued by January 30, 2008. Once approved, the Unit 1 amendment shall be implemented prior to start-up from the 2008 Unit 1 refueling outage and the Unit 2 amendment shall be implemented prior to start-up from the 2009 Unit 2 refueling outage.

4.0 Technical Analysis

Core Operating Limits Report Methodologies

Currently, Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," item b, identifies the NRC reviewed and approved analytical methods to be used to determine core operating limits. A revision to Technical Specification 5.6.5.b is needed to reference the AREVA NP analytical methods that will be used in upcoming fuel cycles to determine core operating limits.

Each methodology reference being added is being cited consistent with the format established in TSTF-363, *Revise Topical Report References in ITS 5.6.5, COLR*, which was approved by the NRC on April 13, 2000. TSTF-363 has subsequently been incorporated into the latest NRC-approved version of the Standard Technical Specifications for BWR/4 plants (i.e., Revision 3.1 of NUREG-1433). NUREG-1433, Revision 3.1, states that: (1) Technical Specification 5.6.5.b should identify the topical report(s) by number and title or identify the staff Safety Evaluation Report for a plant-specific methodology by NRC letter and date, and (2) the COLR will contain the complete identification for each Technical Specification referenced topical report(s) used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

Brunswick Technical Specification 5.6.5.b contains an existing reference to the GNF-A topical report NEDE-24011-P-A, *General Electric Standard Application for Reactor Fuel.* This analytical methodology reference is being retained because some GNF-A-derived core operating limits will continue to be applicable to the co-resident GNF-A fuel. However, to maintain consistency with the format established in Revision 3 of NUREG-1433, the existing phrase "(latest approved version)" is being removed from the Technical Specification 5.6.5.b reference to NEDE-24011-P-A.

Brunswick Technical Specification 5.6.5.b also contains an existing reference to the GNF-A topical report NEDE-32906P-A, *TRACG Application for Anticipated Abnormal Operational Occurrences (AOO) Transient Analyses.* The TRACG analytical methodology was added to the Brunswick Technical Specifications through Amendments 231 (i.e., Reference 3) and 262 (i.e., Reference 4) for Units 1 and 2, respectively. At the time the TRACG analytical methodology was added to the Brunswick Technical Specifications, the TRACG method had not yet been incorporated into the GNF-A topical report NEDE-24011-P-A. The TRACG analytical methodology has subsequently been added to GNF-A topical report NEDE-24011-P-A; therefore, the separate Technical Specification 5.6.5.b reference to GNF topical report NEDE-32906P-A is no longer necessary and is being removed.

Finally, a reference to the GNF-A topical report NEDO-32465-A, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications*, is being added to Technical Specification 5.6.5.b. Although NEDO-32465-A has been incorporated into the compendium of methods contained in GNF-A topical report NEDE-24011-P-A, the NEDO-32465-A topical report will be used by AREVA NP as part of their detection and suppression analyses.

Enclosure 2 provides a list of the NRC-approved AREVA NP topical report references being added to Technical Specification 5.6.5.b and the associated Technical Specification Limiting Condition for Operations for the core limits or setpoints for which the methodology is used. Enclosure 3 provides a list showing the applicability of the retained GNF-A topical report NEDE-24011-P-A reference to Technical Specification 5.6.5.a core operating limits. Details regarding the GNF-A methodologies for calculating core limits are available in NEDE-24011-P-A-15.

Linear Heat Generation Rate

The Brunswick Technical Specifications currently do not include LHGR as a required power distribution limit. The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location and is an important fuel design parameter. Limits are calculated for LHGR to ensure that the cladding circumferential plastic strain will not exceed one percent and that the fuel centerline will not melt. Thus, limits on the LHGR in the COLR are specified to ensure the fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences.

For GNF-A fuel, the average planar linear heat generation rate (APLHGR) limits, in addition to protecting against excess peak clad temperature, include the exposure dependent thermal-mechanical limits for the fuel rods and thereby ensure that the LHGR limits are not exceeded. During the upcoming BSEP mixed core cycles, the APLHGR limits that were developed by GNF-A for their fuel types will continue to be the applicable limits and will ensure that the LHGR limits for GNF-A fuel are not exceeded.

To support the use of AREVA NP fuel, a new Technical Specification 3.2.3 is being proposed to add the LHGR limit as a separate limit. For AREVA NP fuel, both the APLHGR and the LHGR limits will be specified in the COLR.

Currently, Technical Specification 3.4.1, Recirculation Loop Operating, and Technical Specification 3.7.6, Main Turbine Bypass System, contain equipment out-of-service provisions for adjusting APLHGR and MCPR limits to ensure thermal margins are maintained if certain equipment is not in service. The specific adjusted APLHGR and MCPR limits for these conditions are currently documented in the COLR. As part of the proposed amendments, a similar provision is being added to Technical Specification 3.4.1 and Technical Specification 3.7.6 to apply an adjustment to the LHGR limit when a single reactor recirculation loop or the main turbine bypass system is inoperable. The specific adjusted LHGR limits for these conditions will also be documented in the COLR.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., is submitting proposed amendments to support transition to AREVA NP fuel and core design and analysis services for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed amendments: (1) revise Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," to add references to AREVA NP analytical methods that will be used to determine core operating limits, and (2) add a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," add a new definition to Technical Specification 1.1 for LHGR, and revise Technical Specifications 3.4.1 and 3.7.6 to incorporate restrictions on LHGR when in single recirculation loop operation or with an inoperable Turbine Bypass System. CP&L has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed amendments revise the list of NRC-approved analytical methods used to establish core operating limits. Core operating limits are established to ensure that fuel design limits are not exceeded during operating transients or accidents. The analytical methods used to determine core operating limits are those methods that have previously been found acceptable by the NRC and are required to be listed in the Technical Specification section governing the Core Operating Limits Report. The application of these NRC-approved analytical methods will continue to ensure that acceptable operating limits are established and applied to operation of the reactor core.

   The proposed amendments will add a new Technical Specification 3.2.3, "Linear Heat Generation Rate (LHGR)," for fuel bundles, add a new definition to Technical Specification 1.1 for LHGR, and revise Technical Specifications 3.4.1 and 3.7.6 to incorporate restrictions on LHGR when in single recirculation loop operation or with an inoperable Turbine Bypass System. These LHGR limits will be established using NRC-approved analytical methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable.

   Based on the above, the proposed amendments do not involve an increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   As previously stated, the proposed amendments support transition from Global Nuclear Fuels - Americas (GNF-A) fuel and core design and analysis services to AREVA NP fuel and core design and analysis services. The AREVA NP fuel assemblies which will be used in the BSEP Unit 1 and 2 cores will be similar in design to the GNF-A fuel that will be co-resident in the cores. The BSEP, Unit 1 and 2 cores in which this fuel will operate will be designed to meet all applicable design and licensing criteria. Adherence to these design and licensing criteria will not introduce any new modes of operation or introduce any new accident precursors, and thus will preclude the introduction of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed amendments will continue to require that core operating limits be determined using NRC-approved analytical methods. Acceptable fuel performance is obtained by ensuring that the peak cladding temperature (PCT) during a postulated design basis loss-of-coolant accident (LOCA) is maintained less than the limits specified in 10 CFR 50.46, and that the core remains in a coolable geometry following a postulated design basis LOCA. The proposed amendments ensure that adequate margin will continue to be maintained to the 2200 degree PCT limit of 10 CFR 50.46, and the use of NRC-approved analytical methods will continue to ensure acceptable fuel performance during normal operations, as well as during transient and accident conditions. Therefore, the proposed amendments do not involve a significant reduction in a margin of safety.

Based on the above, CP&L concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements for the content required in a licensee's Technical Specifications. Criterion 2 of 10 CFR 50.36(c)(2)(ii) requires a limiting condition for operation to be established for a process variable, design feature, or operating restriction that is an initial condition of a design

basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water power reactors," establishes the acceptance criteria for the design basis LOCA. Paragraph (b)(1) requires the calculated maximum fuel element cladding temperature (i.e., PCT) to not exceed 2200°F. The use of NRC-approved analytical methods to determine core operating limits will continue to ensure that fuel performance during normal, transient, and accident conditions complies with these requirements. Specific APLHGR limits will be determined and documented in the COLR to ensure compliance with 10 CFR 50.46(b)(1). LHGR is being added to the Technical Specifications to support Criterion 2 of 10 CFR 50.36(c)(2)(ii).

6.0 Environmental Considerations

A review has determined that the proposed amendments would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendments do not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

7.0 References

1. "Summary of March 10, 2006, Meeting with Carolina Power & Light Company (CP&L), Regarding the Fuel Transition Process and Licensing Submittals for Brunswick Steam Electric Plant, Units 1 and 2," dated April 27, 2006, ADAMS Accession Number ML061000574.

2. Letter from Mr. James Scarola (CP&L) to the Document Control Desk (NRC), "Request for License Amendments Regarding Fuel Design and Storage Requirements for AREVA NP Fuel," dated January 22, 2007.

3. Letter from Ms. Brenda L. Mozafari (NRC) to Mr. C. J. Gannon (CP&L), "Issuance of Amendment Re: Minimum Critical Power Ratio Safety Limit (TAC No. MC1249)," dated March 26, 2004, ADAMS Accession Number ML040900042.

4. Letter from Ms. Brenda L. Mozafari (NRC) to Mr. C. J. Gannon (CP&L), "Issuance of Amendment on Addition of TRACG Methodology for Determining Core Operating Limits (TAC No. MC4385)," dated March 4, 2005, ADAMS Accession Number ML050470131.

AREVA NP BWR Approved Topical Reports for
Brunswick Units 1 and 2 Technical Specification
COLR References

Report	Applicable LCO	Justification
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*.	3.2.1 / ATRIUM™-10 3.2.2 / ATRIUM™-10, GE14 3.2.3 / ATRIUM™-10	Provides an analytical capability to predict BWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.
XN-NF-85-67(P)(A) Revision 1, *Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.*	3.2.3 / ATRIUM™-10	Describes the process used to develop linear heat generation rates for fuel designs.
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.*	3.2.3 / ATRIUM™-10	Extends the exposure limit of the RODEX2A code which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.
ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs.*	3.2.3 / ATRIUM™-10	Establishes a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.

Report	Applicable LCO	Justification
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, *Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.*	3.2.1 / ATRIUM™-10 3.2.2 / ATRIUM™-10, GE14 3.2.3 / ATRIUM™-10 3.3.2.1, Table 3.3.2.1-1 / ATRIUM™-10, GE14	Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transients. Subsequently approved codes or methodologies have superseded portions of this report. Applicable portions include control rod drop accident, and methodology to determine neutronic reactivity parameters, void reactivity, Doppler reactivity, scram reactivity, delayed neutron fraction, and prompt neutron lifetime.
XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.*	3.2.1 / ATRIUM™-10 3.2.2 / ATRIUM™-10, GE14 3.2.3 / ATRIUM™-10	Summarizes the types of BWR licensing analyses performed, identifies the methodologies used.
EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.*	3.2.2 / ATRIUM™-10, GE14 3.2.3 / ATRIUM™-10 3.3.2.1, Table 3.3.2.1-1 / ATRIUM™-10, GE14	Describes the reactor core simulator code MICROBURN-B2 and the lattice physics code CASMO-4.

Report	Applicable LCO	Justification
XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.*	3.2.2 / ATRIUM™-10, GE14	Provides overall methodology for determining a MCPR operating limit.
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.*	3.2.2 / ATRIUM™-10, GE14	Provides a capability to perform analyses of transient heat transfer behavior in BWR assemblies.
ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors.*	3.2.2 / ATRIUM™-10, GE14	Provides a methodology for the determination of thermal margins, specifically the MCPR safety limit.
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1, Supplements 2, 3, and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.*	3.2.2 / ATRIUM™-10, GE14	Provides a computer program for analyzing BWR system transients.
ANF-1358(P)(A) Revision 3, *The Loss of Feedwater Heating Transient in Boiling Water Reactors.*	3.2.2 / ATRIUM™-10, GE14	Presents a generic methodology for evaluating the loss of feedwater heating event.
EMF-2209(P)(A) Revision 2, *SPCB Critical Power Correlation.*	3.2.2 / ATRIUM™-10, GE14	Presents the critical power correlation for use with the ATRIUM™-10 and GE14 fuel designs.

Report	Applicable LCO	Justification
EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.*	3.2.2 / GE14	Provides direct and indirect approaches to develop parameters necessary to appropriately model co-resident fuel with an approved critical power correlation.
EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model.*	3.2.1 / ATRIUM™-10	Describes an evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 and 10 CFR 50 Appendix K criteria.
EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.*	3.2.1 / ATRIUM™-10	Provides measured cladding temperatures from spray heat transfer tests to justify the use of Appendix K coefficients for ATRIUM™-10 fuel LOCA analyses.
EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2.*	3.3.1.1 / ATRIUM™-10, GE14	Describes methodology for stability analysis with input from the MICROBURN-B2 reactor core simulator.
NEDO-32465-A, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.*	3.3.1.1 / ATRIUM™-10, GE14	Describes the bases for period-based detection algorithm setpoint for the OPRMs.

GNF BWR Approved Topical Reports for
Brunswick Units 1 and 2 Technical Specification
COLR References

Applicable LCO / Assembly Types	COLR Specified Limit	NEDE-24011-P-A Applicability
3.2.1 / GE14	Average Planar Linear Heat Generation Rate (APLHGR)	The NEDE-24011-P-A APLHGR limits are applied to off-rated conditions using power- and flow-dependent factors, $MAPFAC_p$ and $MAPFAC_f$. NEDE-24011-P-A APLHGR limits are applied to the following Equipment Out-of-Service (EOOS) conditions: • SLO (TS 3.4.1) • Turbine Bypass OOS (TS 3.7.6)
3.2.2 / GE14	Minimum Critical Power Ratio (MCPR)	NEDE-24011-P-A will not be applied to establish MCPR operating limits. Operating Limit MCPR for rated and off-rated power and flow conditions and EOOS operation will be established for GE 14 fuel using AREVA-approved methods identified in Enclosure 2.
3.2.3 / GE14	Linear Heat Generation Rate (LHGR)	NEDE-24011-P-A will not be applied to establish off-rated or flow-dependent or EOOS corrections to LHGR limits.

Applicable LCO / Assembly Types	COLR Specified Limit	NEDE-24011-P-A Applicability
3.3.2.1, Table 3.3.2.1-1 / GE14	Rod Block Monitor (RBM) setpoints and applicable reactor thermal power ranges	NEDE-24011-P-A will not be applied to establish RBM setpoints and applicable reactor thermal power ranges. The RBM setpoints and applicable thermal power ranges will be established using the AREVA approved methods identified in Enclosure 2.
3.3.1.1, Table 3.3.1.1-1 / GE14	Oscillation Power Range Monitor (OPRM) period-based detection algorithm (PBDA) setpoint limits	NEDE-24011-P-A will not be applied to establish the OPRM setpoints. The OPRM setpoints will be established using AREVA NP approved methods identified in Enclosure 2.

Markup of Technical Specification Pages - Unit 1

1.1 Definitions

LEAKAGE (continued)	b. Unidentified LEAKAGE All LEAKAGE into the drywell that is not identified LEAKAGE; c. Total LEAKAGE Sum of the identified and unidentified LEAKAGE; and d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
INSERT	
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. 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.

(continued)

INSERT 1.1 DEFINITIONS

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP AND 24 hours thereafter

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

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

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

INSERT

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours

(continued)

INSERT LCO 3.4.1

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

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; ~~and~~

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR. ; and

INSERT

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

INSERT LCO 3.7.6

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

INSERT

~~3.~~ 4. The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

~~4.~~ 5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" ~~(latest approved version)~~.

INSERT

~~2. NEDE-32906P-A, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," approved version as specified in the COLR.~~

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

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

(continued)

INSERT TS 5.6.5.a

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

INSERT TS 5.6.5.b

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

INSERT TS 5.6.5.b (Continued)

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

Typed Technical Specification Pages - Unit 1

1.1 Definitions

LEAKAGE (continued)	b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. Total LEAKAGE Sum of the identified and unidentified LEAKAGE; and d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. 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. 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.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any LHGR not within limits.	A.1	Restore LHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

<u>OR</u>

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

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

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

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

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours

(continued)

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

4. The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall:

1. Possess a radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

2. Possess a radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

3. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. Possess a self-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area; or

5. Be under the surveillance, as specified in the RWP or equivalent, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall:

1. Possess an alarming dosimeter with an appropriate alarm setpoint; or

2. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. Possess a direct-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

5. Possess a radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2, 5.7.2.d.3, and 5.7.2.d.4. above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

Typed Technical Specification Pages - Unit 2

1.1 Definitions

LEAKAGE (continued)	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; and 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 (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. 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. 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.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

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

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any LHGR not within limits.	A.1	Restore LHGR(s) to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched recirculation pump speeds shall be in operation,

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

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

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

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

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours

(continued)

3.7 PLANT SYSTEMS

3.7.6 The Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

4. The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 <u>High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation)</u>

a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall:

1. Possess a radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"); or

2. Possess a radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint; or

3. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates not exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation) (continued)

4. Possess a self-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area; or

5. Be under the surveillance, as specified in the RWP or equivalent, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift superintendent or the radiation control supervisor or designated representative; and

2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall:

1. Possess an alarming dosimeter with an appropriate alarm setpoint; or

2. Possess a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. Possess a direct-reading dosimeter and be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area; or

4. Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(continued)

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

5. Possess a radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2, 5.7.2.d.3, and 5.7.2.d.4, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.

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BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin to ensure the safety limit will not be reached or exceeded such that fuel damage would occur.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

INSERT

~~GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow.~~ For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

(continued)

INSERT TO B 2.1.1.1 (Fuel Cladding Integrity) APPLICABLE SAFETY ANALYSIS

The SPCB critical power correlation is valid for critical power calculations at pressures $\geq$ 571.4 psia and bundle mass fluxes $\geq$ 0.087E+6 lbm/hr-ft^2 (Reference 1).

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 46% RTP. Thus, a THERMAL POWER limit of 23% RTP for reactor pressure < 785 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved ~~General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 1. Reference 1 also includes, by reference, a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 1, 2, and 3 describe the uncertainties and methodology used to determine the MCPR SL.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. In conjunction with LCOs, the limiting safety system settings, defined in LCO 3.3.1.1 as the Allowable Values, establish the threshold for protective system action to prevent exceeding acceptable limits, including this reactor vessel water level SL, during Design Basis Accidents. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 2). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The

4

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

~~1. NEDE-24011-P-A (latest approved revision).~~

4. ~~2.~~ 10 CFR 50.67.

1. EMF-2209(P)(A), "SPCB Critical Power Correlation," (as identified in the COLR).
2. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," (as identified in the COLR).
3. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (as identified in the COLR).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)", and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)", which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 5) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 5).

To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 137 x 7% ≈ 10) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in

(continued)

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod independent of any other source of energy. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, and 3. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 8.75% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2 and 3.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 2 and 3. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have shown that sudden fuel pin rupture requires a fuel energy deposition of approximately 425 cal/gm (Ref. ~~4~~ 6), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Ref. ~~5~~ 7). Generic evaluations (Refs. ~~2 and 6~~ 4 and 8) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. ~~7~~ 9) and the calculated offsite doses will be well within the required limits (Ref. ~~8~~ 10).

(continued)

with AREVA methods (Refs. 2 and 3)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Control rod patterns analyzed ~~in Reference 2~~ follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 8.75% RTP (Ref. ~~2~~ 5). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS with GNF methods has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS during a plant startup or shutdown. The generic BPWS analysis (Ref. ~~3~~ 11) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and a required distribution of fully inserted, inoperable control rods. Cycle-specific analysis with AREVA methods (Refs. 2 and 3) demonstrates that postulated CRDA events do not exceed the acceptance criteria for CRDA events.
	Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. ~~10~~ 12).
LCO	Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.
APPLICABILITY	In MODES 1 and 2, when THERMAL POWER is ≤ 8.75% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 8.75% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. ~~2~~ 5). In MODES 3, 4, and 5, since the reactor is shut down and interlocks allow only a single control rod to be withdrawn from a core cell containing fuel assemblies in MODE 5, adequate SDM ensures that the consequences of a CRDA are acceptable. This is due to the fact that the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

to correct the position of control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows an individual control rod to be bypassed in the RWM or the entire RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator or other qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor must be manually scrammed within 1 hour. This ensures the reactor is shut down and, as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 8.75% RTP.

REFERENCES

1. UFSAR, Section 15.4.

~~2.~~ 4. NEDE-24011-P-A-11-US, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section 2.2.3.1, November 1995.

(continued)

2. XN-NF-80-19(P)(A), Volume 1 "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

3. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO/MICROBURN-B2," (as identified in the COLR).

BASES

REFERENCES (continued)

5. → ~~3.~~ NRC Safety Evaluation Report, Acceptance For Referencing of Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17; December 27, 1987.

6. → ~~4.~~ UFSAR, Section 4.3.2.5.

7. → ~~5.~~ NUREG-0800, Section 15.4.9, Revision 2, July 1981.

8. → ~~6.~~ NEDO-21778-A, Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors, December 1978.

9. → ~~7.~~ ASME, Boiler and Pressure Vessel Code.

10. → ~~8.~~ 10 CFR 50.67.

11. → ~~9.~~ NEDO-21231, Banked Position Withdrawal Sequence, January 1977.

12. → ~~10.~~ 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

INSERT

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 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.~~ 1, 2, and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, and 6 for the GNF fuel and References 11 and 12 for the AREVA fuel.

GNF Fuel

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and 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.

Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 7) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the

(continued)

INSERT B 3.2.1 BACKGROUND

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. For GNF fuel, the APLHGR limits also ensure that the fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 23% RTP and the previously mentioned bypass power level. 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 8.

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 9. 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.

INSERT

For single recirculation loop operation, Reference 5 shows that no APLHGR reduction is required.

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

GNF Fuel

The APLHGR limits ~~for each type of fuel as a function of axial location and average planar exposure~~ specified ~~by reference~~ in the COLR are the result of the fuel design, DBA, and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. The APLHGR limits have been approved for the respective fuel and lattice type and determined by the approved methodology described in Reference 1. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value, adjusted for core flow

(continued)

INSERT B 3.2.1 APPLICABLE SAFETY ANALYSIS

AREVA Fuel

The analytical methods and assumptions used in evaluating the DBA that determine the APLHGR limit are presented in Reference 11. APLHGR limits are developed as a function of planar exposure.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using analysis models that are consistent with the requirements of 10 CFR 50, Appendix K. The analysis models and methodology are described in Reference 11.

The AREVA fuel APLHGR limits are presented in the COLR. For single recirculation loop operation, a multiplier is applied to the two recirculation loop operation exposure dependent APLHGR limits. This multiplier is due to an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

BASES

LCO (continued)

and core power, for the most limiting lattice (excluding natural uranium) for each type of fuel shown in the applicable figures of the COLR. Limits have been provided in the COLR for two recirculation loop operation and single recirculation loop operation. The limits on single recirculation loop operation are provided to allow operation in this condition in conformance with the requirements of LCO3.4.1, "Recirculation Loops Operating."

INSERT

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.~~ Studies 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. At THERMAL POWER levels ≤ 23% RTP, the reactor is operating with substantial margin to the APLHGR limits. For consistency with the 2.1.1.1 SL, this power level was selected for LCO applicability.

INSERT

ACTIONS

A.1

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 and continued 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 4 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 occurring simultaneously with the APLHGR out of specification.

B.1

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

(continued)

INSERT B 3.2.1 LCO:

AREVA Fuel

The APLHGR limits specified in the COLR are the result of the DBA analyses. For two recirculation loops operating, the limit is dependent on exposure. With only one recirculation loop in operation, in conformance with LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a multiplier determined by a specific single recirculation loop analysis.

Additional APLHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

INSERT B 3.2.1 APPLICABILITY:

The GNF fuel APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. The AREVA fuel APLHGR limits are derived from LOCA analyses that are assumed to occur at high power levels.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref.1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

INSERT

~~The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, and 8. To ensure that the 99.9% of the fuel rods avoid boiling transition during any transient that occurs with moderate frequency, limiting transients are analyzed either with TRACG or other methodologies. The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The TRACG methodology calculates an operating limit MCPR (OLMCPR) for the transient initial condition that will yield the largest change in CPR (ΔCPR) resulting from the limiting transient. When the largest ΔCPR is added to the MCPR SL, an OLMCPR is obtained. The most limiting of the OLMCPR calculated by either the TRACG or other methodology sets the core operating limits.~~

~~The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and~~

(continued)

INSERT B 3.2.2 APPLICABLE SAFETY ANALYSIS:

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 7, 8, 9, and 10. To ensure that 99.9% of the fuel rods avoid boiling transition during any transient that occurs with moderate frequency, limiting transients are analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR (OLMCPR) is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$ respectively) to ensure adherence to fuel design limit during the worst transient that occurs with moderate frequency as identified in the UFSAR Chapter 15 (Reference 5).

Flow dependent MCPR limits are determined using steady state thermal hydraulic methods (Reference 7) to analyze slow flow runout transients. The $MCPR_f$ limits are dependent on the maximum core flow runout capability of the Recirculation System.

Power dependent MCPR limits ($MCPR_p$) are determined on a cycle-specific basis using the methodologies presented in References 8, 9, and 10. The $MCPR_p$ limits are established for a set of exposure intervals. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve and/or turbine control valve fast closure and the associated scrams occur, high and low flow $MCPR_p$ operating limits may be provided for the following core power level ranges: 23 to 26% RTP and 26 to 40% RTP. The 26% RTP is the previously mentioned bypass power level and 40% RTP is the power level below which the power load unbalance unit (PLU) may not generate a turbine control valve fast closure on a generator loss of load event.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~$MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Ref. 7).~~

~~Flow dependent MCPR limits are determined using the methodology described in Reference 2 to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.~~

~~Power dependent MCPR limits ($MCPR_p$) are determined using the methodology described in Reference 2. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 23% RTP and the previously mentioned bypass power level.~~

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8). 11

LCO

The MCPR operating limits, as a function of core flow, core power, and cycle exposure, specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% RTP is unnecessary due to the large inherent margin that ensures that the MCPRSL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 23% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. ~~SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The MCPR operating limits for the Option A and Option B analyses are specified in the COLR. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.~~

INSERT

REFERENCES

1. UFSAR Section 4.4.2.1.
2. ~~NEDO-24011-P-A, General Electric Standard Application for Reactor Fuel (latest approved version).~~
3. UFSAR, Chapter 4.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. ~~NEDC-31776P, Brunswick Steam Electric Plant Units 1 and 2 Single-Loop Operation, December 1989.~~

(Deleted.)

INSERT

(continued)

ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.

INSERT B 3.2.2 SURVEILLANCE REQUIREMENTS:

SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic (nominal) scram times. The scram time dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

INSERT B 3.2.2 REFERENCES:

7. XN-NF-80-19(P)(A) Volume 3, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," (as identified in the COLR).
8. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (as identified in the COLR).
9. ANF-913(P)(A) Volume 1, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," (as identified in the COLR).
10. XN-NF-84-105(P)(A), XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," (as identified in the COLR).
11. 10 CFR 50.36(c)(2)(ii).

BASES

REFERENCES (continued)	7.	NEDC-31654P, Maximum Extended Operating Domain Analysis for Brunswick Steam Electric Plant, February 1989.
	8.	NEDE-32906P-A, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," approved version as specified in the COLR.
	9.	10 CFR 50.36(c)(2)(ii).

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 the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. 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 References 1 and 2.

For GNF fuel, LCO 3.2.1 "AVEARGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)" ensures that the fuel design limits are not exceeded during normal operation and anticipated operational occurrences.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that the fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 50.67. The mechanisms that could cause fuel damage during normal operations and operational transients and that are considered in fuel evaluations are:

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

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 is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The LHGR limits are multiplied by the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power state to ensure adherence to the fuel mechanical design bases during the limiting transient. Flow-dependent LHGR factors are determined using the three dimensional BWR simulator code (Reference 4) to analyze slow flow runout transients. The $LHGRFAC_f$ is dependent on the maximum core flow runout capability of the Recirculation System. Based on analyses of limiting plant transients (other than the slow flow runout event) over a range of power and flow conditions, power-dependent LHGR factors 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 and/or turbine control valve fast closure and the associated scrams occur, high and low flow $LHGRFAC_p$ multipliers may be provided for the following core power level ranges: 23 to 26% RTP and 26 to 40% RTP. The 26% RTP is the previously mentioned bypass power level and 40% RTP is the power level below which the power load unbalance unit (PLU) may not generate a turbine control valve fast closure on a generator loss of load event. For GE fuel, no power- or flow-dependent LHGR factors are applied to the LHGR limits. The LHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Reference 5)
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 calculated LHGR which would cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR. Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 23% RTP, the reactor is operating with margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 23% RTP.

(continued)

BASES

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis in 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. The 4 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 Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored 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 < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is > 23% RTP and then every 24 hours thereafter. They are compared with the LHGR 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 ≥ 23% RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. UFSAR, Chapter 4.
2. UFSAR, Chapter 15.
3. NUREG-0800, Section 4.2.II A.2(g), Revision 2, July 1981.
4. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (as identified in the COLR).
5. 10 CFR 50.36(c)(2)(ii)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.f. Oscillation Power Range Monitor (OPRM) Upscale (continued)

necessary to satisfy a Required Action, the APRM equipment is conservatively designed to force an OPRM Upscale trip output from the APRM channel if an APRM Inop condition occurs, such as when the APRM chassis keylock switch is placed in the Inop position.)

There are four "sets" of OPRM related setpoints or adjustment parameters: (a) OPRM trip auto-enable setpoints for Simulated Thermal Power (STP) (25%) and drive flow (60%); (b) period based detection algorithm (PBDA) confirmation count and amplitude setpoints; (c) period based detection algorithm tuning parameters; and (d) growth rate algorithm (GRA) and amplitude based algorithm (ABA) setpoints.

The first set, the OPRM auto-enable region setpoints, as discussed in the SR 3.3.1.1.19 Bases, are treated as nominal setpoints with no additional margins added. The settings, 25% APRM Simulated Thermal Power and 60% drive flow, are defined (limit values) in and confirmed by SR 3.3.1.1.19. The second set, the OPRM PBDA trip setpoints, are established in accordance with methodologies defined in Reference 23 19 and are documented in the COLR. There are no allowable values for these setpoints. The third set, the OPRM PBDA "tuning" parameters, are established, adjusted, and controlled by plant procedures. The fourth set, the GRA and ABA setpoints, in accordance with References 15 and 16, are established as nominal values only, and controlled by plant procedures.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analyses of References 4,

(continued)

BASES

REFERENCES (continued)

22. General Electric Nuclear Energy Letter NSA 01-212, DRF C51-00251-00, A. Chung (GE) to S. Chakraborty (GE), "Minimum Number of Operable OPRM Cells for Option III Stability at Brunswick 1 and 2," June 8, 2001.

23. ~~NEDE-24011-P-A, General Electric Standard Application for Reload Fuel, (latest approved version).~~

BASES

BACKGROUND (continued)

A rod block signal is also generated if an RBM inoperable trip occurs, since this could indicate a problem with the RBM channel. The inoperable trip will occur if, during the nulling (normalization) sequence, the RBM channel fails to null or too few LPRM inputs are available, if a critical self-test fault has been detected, or the RBM instrument mode switch is moved to any position other than "Operate."

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 8.75% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed. The RWM is a single channel system that provides input into the RMCS rod withdraw permissive circuit.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in ~~Reference 2~~. A statistical analysis of RWE events was performed to

References 11 and 12.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established. INSERT

The RBM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

Trip setpoints are specified in the setpoint calculations. The setpoints are selected to ensure that the trip settings do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setting less conservative than the trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setting is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter, the calculated RBM flux (RBM channel signal). When the RBM flux value exceeds the applicable setpoint, the RBM provides a trip output. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint value, by accounting for calibration based errors. These calibration based instrument errors are limited to instrument drift, errors associated with measurement and test equipment, and calibration tolerance of LPRM input processing in the average power range monitor (APRM) equipment. The RBM performs only digital calculations on digitized LPRM signals received from the APRM equipment. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation

(continued)

INSERT B 3.3.2.1 APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY:

With AREVA methods (References 11 and 12), cycle specific analyses are performed to establish MCPR limits and RBM setpoints considering allowed combinations of out of service LPRM instruments feeding the RBM channels, and assuming one RBM channel out of service.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

uncertainties, process effects, calibration tolerances, instrument drift, and environment errors are accounted for and appropriately applied for the instrumentation.

INSERT

~~The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 2). When operating < 90% RTP, analyses (Ref. 2) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR greater than or equal to the limit specified in the COLR, no RWE event will result in exceeding the MCPR SL (Ref. 2). Therefore, under these conditions, the RBM is also not required to be OPERABLE.~~

The RBM selects one of three different RBM flux trip setpoints to be applied based on the current value of THERMAL POWER. THERMAL POWER is indicated to each RBM channel by a simulated thermal power (STP) reference signal input from an associated reference APRM channel. The OPERABLE range is divided into three "power ranges," a "low power range," an "intermediate power range," and a "high power range." The RBM flux trip setpoint applied within each of these three power ranges is, respectively, the "low trip setpoint," the "intermediate trip setpoint," and the "high trip setpoint" (Allowable Values for which are defined in the COLR). To determine the current power range, each RBM channel compares its current STP input value to three power setpoints, the "low power setpoint" (29%), the "intermediate power setpoint" (current value defined in the COLR), and the "high power setpoint" (current value defined in the COLR), which define, respectively, the lower limit of the low power range, the lower limit of the intermediate power range, and the lower limit of the high power range. The trip setpoint applicable for each power range is more restrictive than the corresponding setpoint for the lower power range(s). When STP is below the low power setpoint, the RBM flux trip outputs are automatically bypassed but the low trip setpoint continues to be applied to indicate the RBM flux setpoint on the NUMAC RBM displays.

(continued)

INSERT B 3.3.2.1 APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY:

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Analyses demonstrate that below this power level the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE. Cycle specific analyses with Reference 11 and 12 methods are performed to establish an initial MCPR value above which the occurrence of an RWE event will not result in exceeding the MCPR SL while operating < 90% RTP. Cycle specific analyses are also performed to demonstrate that with MCPR greater than or equal to a specified value, no RWE event will result in exceeding the MCPR SL when operating ≥ 90% RTP. The MCPR values for both < 90% RTP and ≥ 90% RTP are provided in the COLR. Under the conditions of RTP and MCPR specified in the COLR, the RBM is not required be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The calculated (required) setpoints and applicable power ranges are bounding values. In the equipment implementation, it is necessary to apply a "deadband" to each setpoint. The deadband is applied to the RBM trip setpoint selection logic and the RBM trip automatic bypass logic such that the setpoint being applied is always equal to or more conservative than the required setpoint. Since the RBM flux trip setpoint applicable to the higher power ranges are more conservative than the corresponding trip setpoints for lower power ranges, the trip setpoint applicable to the higher power range (high power range or intermediate power range) continues to be applied when STP decreases below the lower limit of that range until STP is below the power range setpoint by a value exceeding the deadband. Similarly, when STP decreases below the low power setpoint, the automatic bypass of RBM flux trip outputs will not be applied until STP decreases below the trip setpoint by a value exceeding the deadband.

the values specified in the COLR.

The RBM channel uses THERMAL POWER, as represented by the STP input value from its reference APRM channel, to automatically enable RBM flux trip outputs (remove the automatic bypass) and to select the RBM flux trip setpoint to be applied. However, the RBM Upscale function is only required to be OPERABLE when the MCPR values are less than ~~either 1.40 or 1.70, depending on the THERMAL POWER level.~~ Therefore, even though the RBM Upscale Function is implemented in each RBM channel as a single trip function with a selected trip setpoint, it is characterized in Table 3.3.2.1-1 as three Functions, the Low Power Range—Upscale Function, the Intermediate Power Range—Upscale Function, and the High Power Range—Upscale Function, to facilitate correct definition of the OPERABILITY requirements for the functions. Each Function corresponds to one of the RBM power ranges. Due to the deadband effects on the determination of the current power range, the transition between these three Functions will occur at slightly different THERMAL POWER levels for increasing power versus decreasing power. Since the RBM flux trip setpoints applied for the higher power ranges are more conservative, the OPERABILITY requirement for the Low Power Range—Upscale Function is satisfied if the Intermediate Power Range—Upscale Function or the High Power Range—Upscale Function is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

OPERABLE. Similarly, the OPERABILITY requirement for the Intermediate Power Range—Upscale Function is satisfied if the High Power Range—Upscale Function is OPERABLE.

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References ~~4, 5, and 6.~~ 11 and 12. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

The RWM is a microprocessor-based system with the principle task to reinforce procedural control to limit the reactivity worth of control rods under lower power conditions. Only one channel of the RWM is available and required to be OPERABLE. Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. As required by these conditions, one or more control rods may be bypassed in the RWM or the RWM may be bypassed. However, the RWM must be considered inoperable and the Required Actions of this LCO followed since the RWM can no longer enforce compliance with the BPWS.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 8.75% RTP. When THERMAL POWER is > 8.75% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Refs. 5 and 6). In MODES 3 and 4, all control rods are required to be inserted into the core;

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 7.6.1.1.5.
2. NEDC-31654P, Maximum Extended Operating Domain Analysis For Brunswick Steam Electric Plant, February 1989.
3. 10 CFR 50.36(c)(2)(ii).
4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Unit 1 and 2, September 1995.
5. UFSAR Section 15.4.
6. NRC SER, Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A; General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17, December 27, 1987.
7. GENE-770-06-1-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.
8. NEDC-30851P-A, Supplement 1, Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation, October 1988.
9. NEDC-32410P-A, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, October 1995.
10. NEDC-32410P-A Supplement 1, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, November 1997.

INSERT

INSERT B 3.3.2.1 REFERENCES:

11. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).
12. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (as identified in the COLR).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The high water level trip indirectly initiates a reactor scram from the main turbine trip (above 26% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR and increase in LHGR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

The LCO requires three channels of the reactor vessel high water level instrumentation to be OPERABLE to ensure that the feedwater pump turbines and main turbine trip on a valid high water level signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.2. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Trip setpoints are specified in the setpoint calculations. The setpoints are selected to ensure that the trip settings do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setting less conservative than the trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setting is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are determined from the analytic limits corrected for defined process, calibration, and instrument errors. The Allowable Values are then determined, based on the trip setpoint values, by accounting for calibration based errors. These calibration based instrument errors are limited to instrument drift, errors associated with measurement and test equipment, and calibration tolerance of loop components. The trip setpoints and Allowable Values determined in this manner provide adequate protection because

(continued)

and LCO 3.2.3, "LINEAR HEAT GENERATION RATE,"

BASES

LCO (continued)

instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for and appropriately applied for the instrumentation.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," ~~and~~ LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 23% RTP; therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

A.1

With one channel inoperable, the remaining two OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand

(continued)

and LCO 3.2.3 for Required Action A.1, since this instrument's purpose is to preclude MCPR and LHGR violations.

BASES

ACTIONS

B.1 (continued)

4 hour Completion Time provided in LCO 3.2.2 ~~for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.~~

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 23% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 23% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 23% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pump turbines and main turbine will trip when necessary.

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level—Low Level 3

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Reactor Vessel Water Level—Low Level 3 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level—Low Level 3 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS and associated DGs during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level—Low Level 3 Function is directly assumed in the analysis of the recirculation line break (Ref. 5). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level—Low Level 3 Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Four channels of Reactor Vessel Water Level—Low Level 3 Function are only required to be OPERABLE when the ECCS or DG(s) are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS and DG initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS—Shutdown," for Applicability Bases for the low pressure ECCS subsystems; and LCO 3.8.1 and LCO 3.8.2, "AC Sources—Shutdown," for Applicability Bases for the DGs.

1.b, 2.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated

(continued)

The Drywell Pressure - High function is not explicitly used in the AREVA analysis of the recirculation line break since it does not impact the limiting LOCA.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b, 2.b. Drywell Pressure—High (continued)

DGs are initiated upon receipt of the Drywell Pressure—High Function coincident with Reactor Steam Dome Pressure—Low Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function is directly assumed in the GNF analysis of the recirculation line break ~~(Ref. 5)~~. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

The Drywell Pressure—High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure—High Functions are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS and DG initiation. In MODES 4 and 5, the Drywell Pressure—High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure—High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the DGs.

1.c, 2.c. Reactor Steam Dome Pressure—Low

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The low reactor steam dome pressure signals are also used in the Drywell Pressure—High logic circuits to distinguish high drywell pressure caused by a LOCA from that caused by loss of drywell cooling. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS and associated DGs during the transients analyzed in References 2 and 3. In addition, the Reactor Steam Dome

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c, 2.c. Reactor Steam Dome Pressure—Low (continued)

Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 5). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure—Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is low enough to prevent overpressuring the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

Four channels of Reactor Steam Dome Pressure—Low Function are only required to be OPERABLE when the ECCS or DG(s) are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS and DG initiation. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems; and LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

1.d, 2.f. Core Spray and RHR Pump Start—Time Delay Relays

The purpose of these time delays is to stagger the start of the CS and RHR pumps that are in each of Divisions I and II, thus limiting the starting transients on the 4.16 kV emergency buses. These Functions are necessary when power is being supplied from either the normal power sources (offsite power) or the standby power sources (DGs). The Core Spray Pump Start—Time Delay Relays and the RHR Pump Start—Time Delay Relays are assumed to be OPERABLE in the accident and transient analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the power sources.

There are eight RHR Pump Start—Time Delay Relays, two channels in each of the RHR pump start logic circuits. There are six CS pump start timers arranged such that there are four separate channels of the Core Spray Pump Start Time—Delay Relay Function, two channels in each of the CS pump start logic circuits. Each channel consists of an individual

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

2.d. Reactor Steam Dome Pressure—Low (Recirculation Pump Discharge Valve Permissive) (continued)

be OPERABLE and capable of closing the valve(s) during the transients analyzed in References 2 and 3. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 5).

The Reactor Steam Dome Pressure—Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

Four channels of the Reactor Steam Dome Pressure—Low Function are only required to be OPERABLE in MODES 1, 2, and 3 with the associated recirculation pump discharge valve open or the associated recirculation pump discharge bypass valve open. With the valve(s) closed, the function of instrumentation has been performed; thus, the Function is not required. In MODES 4 and 5, the loop injection location is not critical since LPCI injection through the recirculation loop in either direction will still ensure that LPCI flow reaches the core (i.e., there is no significant reactor steam dome back pressure).

2.e. Reactor Vessel Shroud Level

The Reactor Vessel Shroud Level Function is provided as a permissive to allow the RHR System to be manually aligned from the LPCI mode to the suppression pool cooling/spray or drywell spray modes. The permissive ensures that water in the vessel is at least two thirds core height before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage. This function may be overridden during accident conditions as allowed by plant procedures. The Reactor Vessel Shroud Level Function is implicitly assumed in the analysis of the recirculation line break (Ref. 5) since the analysis assumes that no LPCI

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3.e. Suppression Chamber Water Level—High (continued)

Two channels of Suppression Chamber Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

Automatic Depressurization System (ADS)

4.a, 5.a. Reactor Vessel Water Level—Low Level 3

events analyzed in Reference 2.

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level—Low Level 3 is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the ~~accident analyzed in References 2 and 5.~~ The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Level 3 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level—Low Level 3 Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4.b, 5.b. ADS Timer

The purpose of the ADS Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The ADS Timer Function is assumed to be OPERABLE for the accident analyses of References 2 and 6 that require ECCS initiation and assume unavailability of the HPCI System.

There are two ADS Timer relays, one in each of the two ADS trip systems. The Allowable Value for the ADS Timer is chosen to be long enough to allow HPCI to start and avoid an inadvertent blowdown yet short enough so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the ADS Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Reactor Vessel Water Level—Low Level 1

The Reactor Vessel Water Level—Low Level 1 Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level—Low Level 3 signals. In order to prevent spurious initiation of the ADS due to spurious Level 3 signals, a Level 1 signal must also be received before ADS initiation commences.

Reactor Vessel Water Level—Low Level 1 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Allowable Value for

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Reactor Vessel Water Level—Low Level 1 (continued)

Reactor Vessel Water Level—Low Level 1 is selected at the RPS Level 1 scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases discussion of this Function. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of Reactor Vessel Water Level—Low Level 1 Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and RHR (LPCI Mode) Pump Discharge Pressure—High

The Pump Discharge Pressure—High signals from the CS and RHR pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure—High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in Reference~~s~~ 2 ~~and 5~~ with an assumed HPCI unavailability. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one CS pump (both channels for the pump) indicate the high discharge pressure condition or two RHR pumps in one LPCI loop (one channel for each pump) indicate a high discharge pressure condition. The Pump Discharge Pressure—High Allowable Value is less than the pump discharge pressure when the pump is operating at all flow ranges and high enough to avoid any condition that results in a discharge

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic and simulated automatic operation for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

Based on minimal assumed risk in performing this Surveillance with the reactor at power, the surveillance is not required to be performed during a refueling outage. Operating experience has demonstrated that these components will usually pass the SR when performed at the 24 month Frequency (originally based on the refueling cycle). Therefore, the Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36(c)(2)(ii).
5. (Deleted.) ~~NEDC-31624P, Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2), July 1990.~~
6. UFSAR, Section 9.2.6.2.
7. NEDC-30936-P-A, BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Parts 1 and 2, December 1988.

The AREVA LOCA analyses assume mismatched flow in the recirculation loops. In the GNF analyses, the analyses assume that both loops are operating at the same flow prior to the accident. The GNF

BASES

BACKGROUND (continued)

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 which increases the overall core heat transfer. As a result, core voiding is reduced thereby overcoming the negative reactivity void 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., approximately 60 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

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 Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA). 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. ~~The analyses assume that both loops are operating at the same flow prior to the accident. However, the~~ LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. ~~The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 1), which are analyzed in Chapter 15 of the UFSAR (Ref. 2).~~

(continued)

Mismatched flow has an insignificant impact on the fuel thermal margin during abnormal operational transients which are analyzed in Chapter 15 of the UFSAR (Ref. 2).

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. The GNF analysis

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~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, without the requirement to modify the APLHGR requirements (Ref. 3). However, the COLR may require APLHGR limits to restrict the peak clad temperature for a LOCA with a single recirculation loop operating below the corresponding temperature for both loops operating.

INSERT

~~The transient analyses of Chapter 15 of the UFSAR 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 without the requirement to modify the MCPR requirements.~~ During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) Simulated Thermal Power—High Allowable Value is required to account for the different analyzed limits between two-recirculation drive flow loop operation and operation with only one loop. The APRM channel subtracts the ΔW value from the measured recirculation drive flow to effectively shift the limits and uses the adjusted recirculation drive flow value to determine the APRM Simulated Thermal Power—High Function trip setpoint.

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

Two recirculation loops are normally required to be in operation with their recirculation pump speeds 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 Allowable Value (LCO 3.3.1.1), as

(continued)

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

INSERT B 3.4.1 APPLICABLE SAFETY ANALYSES:

For AREVA fuel, the COLR presents single loop operation APLHGR limits in the form of a multiplier that is applied to the two loop operation APLHGR limits.

The transient analyses of Chapter 15 of the UFSAR have also been evaluated for single recirculation loop operation. The evaluation concludes that results of the transient analyses are not significantly affected by the single recirculation loop operation. There is, however, an impact on the fuel cladding integrity SL since some of the uncertainties for the parameters used in the critical power determination are higher in single loop operation. The net result is an increase in the MCPR operating limit.

BASES

LCO (continued)

applicable, must be applied to allow continued operation. The COLR defines adjustments or modifications required for the APLHGR ~~and~~ , MCPR limits for the current operating cycle.

, and LHGR

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.

ACTIONS

A.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched recirculation pump speeds within 6 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the difference in pump speeds of the two recirculation pumps is greater than the match criteria. The loop with the lower recirculation pump speed 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.

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

The 6 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 (i.e., reset the applicable limits or setpoints for single recirculation loop operation), and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure

BASES

BACKGROUND

The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of design basis accidents and transients.

APPLICABLE SAFETY ANALYSES

The reactor steam dome pressure of ≤ 1045 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analyses of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% fuel cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)").

Reactor steam dome pressure satisfies the requirements of Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

The specified reactor steam dome pressure limit of ≤ 1045 psig ensures the plant is operated within the assumptions of the vessel overpressure protection analysis. Operation above the limit may result in a transient response more severe than analyzed.

APPLICABILITY

In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these MODES, the reactor may be generating significant steam and events which may challenge the overpressure limits are possible.

(continued)

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

BASES

BACKGROUND (continued)

The ADS (Ref. 4) consists of 7 of the 11 SRVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems so that these subsystems can provide coolant inventory makeup. Each of the SRVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5 and 6. The required analyses and assumptions are defined in Reference 7. The results of these analyses are also described in Reference 8.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

8

The limiting single failures are discussed in Reference 10. ~~For a large recirculation loop suction pipe break LOCA, failure of one 250 VDC power supply is considered the most severe failure. For a small break LOCA, one 250 VDC power supply failure was combined with a conservative~~

No mitigation credit for HPCI activation is assumed in any LOCA analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~out of service assumption to bound the single failure combinations in a single analysis. In addition to failing HPCI, one CS pump and one LPCI pump (due to the power supply failure); two ADS valves were assumed out of service (Ref. 10). This combination results in an allowance for a single ADS valve failure with no additional analysis and it results in no accident mitigation credit being assumed for HPCI in any LOCA analysis.~~ The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 11).

LCO

Each ECCS injection/spray subsystem and six of seven ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 9 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 9.

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR shutdown cooling isolation pressure in MODE 3, if they are capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes the period when the required RHR pump is not operating and the period when the system is being realigned to or from the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.12 (continued)

TIME testing. This exception is allowed since the ECCS instrumentation response time is a small part of the ECCS RESPONSE TIME (e.g., sufficient margin exists in the emergency diesel generator start time when compared to the instrumentation response time) (Ref. 14).

ECCS RESPONSE TIME tests are conducted every 24 months. The 24 month Frequency is consistent with the Brunswick refueling cycle.

REFERENCES

1. UFSAR, Section 6.3.2.2.3.
2. UFSAR, Section 6.3.2.2.4.
3. UFSAR, Section 6.3.2.2.1.
4. UFSAR, Section 6.3.2.2.2.
5. UFSAR, Section 15.2.
6. UFSAR, Section 15.6.
7. 10 CFR 50, Appendix K.
8. UFSAR, Section 6.3.3.
9. 10 CFR 50.46.
10. ~~NEDC-31624P, Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis, Revision 2, July 1990.~~ (Deleted.)
11. 10 CFR 50.36(c)(2)(ii).
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.
13. UFSAR, Section 6.3.3.7.
14. NEDO-32291-A, System Analyses for the Elimination of Selected Response Time Testing Requirements, October 1995.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 20.6% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of four valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.1.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the Speed Control System or load limit restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows through connecting piping and bypass valve pressure reducers to the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the generator load rejection transient, the turbine trip transient, and the feedwater controller failure maximum demand transient, as described in the UFSAR, Section 15.2.1 (Ref. 2), Section 15.2.2 (Ref. 3), and Section 15.1.2 (Ref. 4). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR, ~~and~~ MCPR, and LHGR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

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

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), ~~and~~ the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR, ~~and~~ MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the minimum number of bypass valves, specified in the COLR, to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Refs. 2, 3, and 4).

, and LHGR

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the turbine generator load rejection transient. As discussed in the Bases for LCO 3.2.1, ~~and~~ LCO 3.2.2, sufficient margin to these limits exists at < 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

and LCO 3.2.3,

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves as specified in the COLR inoperable), and the APLHGR ~~and~~, MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, ~~and~~ MCPR limits accordingly. The 4 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

and LHGR

(continued)

, and LHGR

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the APLHGR, ~~and~~ MCPR and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 23% RTP. As discussed in the Applicability section, operation at < 23% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on manufacturer's recommendations (Ref. 6), is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of10 CFR 50.36(c)(2)(ii) (Ref. 2) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator or other qualified member of the technical staff. These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."

APPLICABILITY

Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than 8.75% RTP, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to 8.75% RTP, the provisions of this Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6.

(continued)

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