CP&L
A Progress Energy Company

John S Keenan
Vice President
Brunswick Nuclear Plant

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10 CFR 50.90

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
Core Flow Operating Range Expansion

REFERENCES:

1. GE Nuclear Energy Report NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 1, dated August 2002.

2. GE Nuclear Energy Report NEDC-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002.

3. GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002.

4. GE Nuclear Energy Report NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, "GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," dated March 2002.

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company 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 TSs, as necessary, to support an expansion of the core flow operating range (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)).

The basis for the request was prepared following the guidelines contained in GE Nuclear Energy Report NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 1, dated August 2002. Enclosure 2 contains GE

PO Box 10429
Southport, NC 28461

T> 910 457 2496
F> 910 457 2803

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Nuclear Energy Report NEDC-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002 (i.e., M+SAR). The M+SAR is a summary of the results of the safety analyses performed for the BSEP core flow operating range expansion. The M+SAR contains information which GE Nuclear Energy considers to be proprietary. GE Nuclear Energy requests that the proprietary information in this report be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1). An affidavit supporting this request is included in the M+SAR. The NRC may duplicate this submittal, including the M+SAR, for the purpose of internal review. Enclosure 3 contains a non-proprietary version of the M+SAR.

As part of MELLLA+ implementation for BSEP, CP&L will implement the Detect and Suppress Solution - Confirmation Density (DSS-CD) approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities (THI). DSS-CD represents an evolutionary step from the Boiling Water Reactor Owners' Group (BWROG) Option III Reactor Stability Long-Term Solution, currently approved for use at BSEP. Section 2.4, "Stability," of the M+SAR discusses this change and verifies the applicability of GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002, to BSEP Units 1 and 2. Upon issuance of MELLLA+ for BSEP, the Backup Stability Protection (BSP) described in NEDC-33075P will be the preferred alternate method to detect and suppress THI oscillations as allowed by TS 3.3.1.1 Action I. CP&L intends to operate for up to one cycle, for each unit, with the new DSS-CD trip function bypassed. The will allow sufficient time to collect operational data necessary to validate trip setpoints and, thereby, avoid unnecessary reactor scrams. The proposed license amendments revise TSs, as necessary, to support implementation of DSS-CD at BSEP.

The M+SAR also documents the applicability of GE Nuclear Energy Report NEDE-23785P-A, "GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," to BSEP Units 1 and 2. As a result, the 1600 degree F, upper bound peak clad temperature NRC limit is removed. Further discussion of this subject can be found in Section 4.3, "Emergency Core Cooling System Performance," of the M+SAR.

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 plans to implement MELLLA+ during the Unit 1 Cycle 15 Refueling Outage (i.e., B115R1, currently scheduled to begin in February 2004) and the Unit 2 Cycle 17 Refueling Outage (i.e., B217R1, currently scheduled to begin in March 2005). However, to support required reload core analyses for Unit 1, CP&L requests that these amendments be issued by October 1, 2003.

CP&L requests that the amendments, once approved, be issued effective immediately, to be implemented prior to startup from the B115R1 refueling outage for Unit 1 and prior to the startup from the B217R1 refueling outage for Unit 2.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of North Carolina a copy of the proposed license amendments.

Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager - Support Services, at (910) 457-3512.

Sincerely,

John S. Keenan

MAT/mat

Enclosures:

1. Evaluation of Proposed License Amendment Request
2. NEDC-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002 **Proprietary**
3. NEDO-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002 **Non-Proprietary**
4. Marked-up Technical Specification Pages - Unit 1
5. Marked-up Technical Specification Pages - Unit 2
6. Marked-up Technical Specification Bases Pages - Unit 1 (For Information Only)
7. List of Regulatory Commitments

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Notary (Seal)

My commission expires: August 29, 2004

cc (with enclosures except as noted):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Luis A. Reyes, Regional Administrator
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303-8931

U. S. Nuclear Regulatory Commission
ATTN: Mr. Theodore A. Easlick, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission
ATTN: Ms. Brenda L. Mozafari (Mail Stop OWFN 8G9) **(Electronic Copy Only)**
11555 Rockville Pike
Rockville, MD 20852-2738

U. S. Nuclear Regulatory Commission
ATTN: Leonard N. Olshan (Mail Stop OWFN 8H12)
11555 Rockville Pike
Rockville, MD 20852-2738

U. S. Nuclear Regulatory Commission
ATTN: Allen G. Howe (Mail Stop OWFN 8G9)
11555 Rockville Pike
Rockville, MD 20852-2738

Ms. Jo A. Sanford **(w/o Enclosure 2)**
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Ms. Beverly O. Hall, Section Chief **(w/o Enclosure 2)**
Radiation Protection Section, Division of Radiation Protection
North Carolina Department of Environment and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609-7221

Evaluation of Proposed License Amendment Request

Subject: **Request For License Amendments**
Core Flow Operating Range Expansion

1.0 Description

This letter is a request to amend Operating Licenses DPR-71 and DPR-62 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

The proposed license amendments revise Technical Specifications (TSs), as necessary, to (1) support an expansion of the core flow operating range and (2) implement the Detect and Suppress Solution - Confirmation Density (DSS-CD) approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities (THI). These changes will support operation of Units 1 and 2 at 2923 megawatts thermal (MWt) with core flow as low as 85 percent of rated core flow (i.e., Maximum Extended Load Line Limit Analysis Plus (MELLLA+)).

2.0 Proposed Change

To support an expansion of the core flow operating range for Units 1 and 2, Carolina Power & Light (CP&L) Company requests the following TS changes.

Proposed Technical Specification Changes - MELLLA+		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"		
Table 3.3.1.1-1, Function 2.b (Average Power Range Monitors, Simulated Thermal Power - High) Allowable Value	≤ 0 55W + 62.6% RTP and ≤ 117.1% RTP	≤ 0 61W + 65.2% RTP and ≤ 117.1% RTP
Table 3.3.1.1-1, Note b	≤ [0.55 (W - ΔW) + 62 6% RTP] when reset for single loop operation per LCO 3.4 1, "Recirculation Loops Operating." The Value of ΔW is defined in plant procedures.	≤ [0.61 (W - ΔW) + 65.2% RTP] when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The Value of ΔW is defined in plant procedures. **Justification:** **M+SAR Section 5.3.1**

Proposed Technical Specification Changes - MELLLA+		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 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:	Two recirculation loops with matched recirculation pump speeds shall be in operation, OR One recirculation loop may be in operation provided the plant is not operating in the MELLLA+ region defined in the COLR and the following limits are applied when the associated LCO is applicable **Justification:** **M+SAR Section 3.6**
TS 3.4.3, "Safety/Relief Valves (SRVs)" *LCO 3.4.3*	The safety function of 10 SRVs shall be OPERABLE.	The safety function of 11 SRVs shall be OPERABLE.
Actions	A. One or more required SRVs inoperable. A.1 Be in MODE 3 12 hours AND A.2 Be in MODE 4 36 hours	A. One SRV inoperable. A.1 Exit the MELLLA+ operating region defined in the COLR. 12 hours B. Two or more SRVs inoperable. OR Required Action and associated Completion Time of Condition A not met. B.1 Be in MODE 3 12 hours AND B.2 Be in MODE 4 36 hours
SR 3.4.3.1	Verify the safety function lift setpoints of the required 10 SRVs are as follows:	Verify the safety function lift setpoints of the 11 SRVs are as follows: **Justification:** **M+SAR Section 9.3.1**

Proposed Technical Specification Changes - MELLLA+		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 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:*	No current requirement.	4. The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating region for Specifications 3.4.1 and 3.4.3. **Justification:** **Consistency with M+SAR Sections 3.6 and 9.3.1.**

To support implementation of the DSS-CD approach to automatically detect and suppress THI, CP&L requests the following TS changes.

Proposed Technical Specification Changes - DSS-CD		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" *SR 3.3.1.1.19*	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥25% and recirculation drive flow is ≤60%	Deleted. **Justification:** **M+SAR Section 2.4 and NEDC-33075P (Reference 4)**
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" *Table 3.3.1.1-1, Function 2.f (Average Power Range Monitors, OPRM - Upscale)* *Surveillance Requirements* *Note (d)*	SR 3.3.1.1.19 See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.	Delete Delete **Justification:** **M+SAR Section 2.4 and NEDC-33075P (Reference 4)**

Proposed Technical Specification Changes - DSS-CD		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 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:*	3. The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and	Delete **Justification:** **M+SAR Section 2.4 and NEDC-33075P (Reference 4)**

In summary, the proposed changes:

(1) delete SR 3.3.1.1.19, reference to SR 3.3.1.1.19 in TS Table 3.3.1.1-1, and Note d of TS Table 3.3.1.1-1, to account for implementation of the DSS-CD approach to THI,

(2) modify the Average Power Range Monitor (APRM) scram flow-biased setpoint allowable values for two and single recirculation loop operation, in TS Table TS 3.3.1.1-1, to accommodate the expanded operating range,

(3) revise TS 3.4.1 to restrict single loop operation to areas outside of the MELLLA+ operating region,

(4) revise the SRV operability and surveillance requirements, of TS 3.4.3, to reflect the need to maintain 11 operable SRVs while in the MELLLA+ operating region,

(5) delete the existing TS 5.6.5.a.3 which requires the COLR to include the period based detection algorithm (PBDA) made obsolete with implementation of the DSS-CD approach to THI, and

(6) add a new requirement (i.e., TS 5.6.5.a.4) to established the MELLLA+ operating region in the COLR.

These changes support the MELLLA+ expansion of the core flow operating range and implement the DSS-CD approach to automatically detect and suppress THI for BSEP Units 1 and 2

CP&L will make supporting changes to the TS Bases in accordance with TS 5.5.10, "Technical Specifications (TS) Bases Control Program." Enclosure 5 provides marked-up TS 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 May 31, 2002, the NRC issued License Amendments 222 and 247 to the BSEP Unit 1 and 2 TSs (i.e., Reference 1). The amendments allowed an increase in the maximum power level of each unit from 2558 MWt to 2923 MWt. In support of the power uprate, the power/flow operating map (i.e., a licensing region which defines boundaries, inside of which operation of the plant has been analyzed and demonstrated to meet all applicable fuel and system design criteria) was revised. The revised power/flow operating map provides only a narrow operating band, with respect to core flow (i.e., approximately 99 percent to 102.2 percent core flow), when operating at 2923 MWt. This narrow operating band unduly restricts plant operation. In order to restore operating flexibility, CP&L is requesting an expansion of the core flow operating range.

4.0 Technical Analysis

As stated above, CP&L is requesting a revision to power/flow operating map to restore operating flexibility at the CLTP of 2923 MWt. This is accomplished by permitting operation at the CLTP with core flow as low as 85 percent of rated core flow versus approximately 99 percent as limited by the existing power/flow operating map. The expanded operating range is identified as MELLLA+. As a result of this change there will be:

- No increase in the current maximum normal operating reactor dome pressure,
- No increase in core power,
- No increase in the maximum licensed core flow,
- No change to source term methodology,
- No new fuel product line,
- No change in fuel cycle length, and
- No additions to currently licensed operational enhancements.

The basis for the request was prepared following the guidelines contained in GE Nuclear Energy Report NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 1, dated August 2002 (i.e., Reference 2). Enclosure 2 contains GE Nuclear Energy Report NEDC-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002 (i.e., Reference 3, referred to as the M+SAR). The M+SAR is a summary of the results of the safety analyses performed for the BSEP core flow operating range expansion. In preparation of the M+SAR, evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. All safety aspects of the plant, affected by MELLLA+, were evaluated using previously NRC-approved or industry-accepted analytical methods.

As part of MELLLA+ implementation for BSEP, CP&L will also implement the DSS-CD approach to automatically detect and suppress THI. DSS-CD represents an evolutionary step

from the BWROG Option III Reactor Stability Long-Term Solution, currently approved for use at BSEP. The DSS-CD approach uses the same hardware design as Option III; it introduces an enhanced detection algorithm (i.e., the Confirmation Density Algorithm (CDA)) to detect the inception of power oscillations and generate an earlier power suppression trip signal based on successive period confirmation recognition. The existing Option III algorithms are retained to provide defense-in-depth protection. The DSS-CD approach will continue to provide reliable, automatic detection and suppression of stability related power oscillations and provide protection against violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for anticipated oscillations.

Section 2.4, "Stability," of the M+SAR discusses this change and verifies the applicability of GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002 (i.e., Reference 4), to BSEP Units 1 and 2.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

Section 11.3.3, "Assessment of 10 CFR 50.92 Criteria," of the M+SAR evaluates 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." Based on this evaluation, 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

Implementation of MELLLA+ does not require (1) an increase in the current maximum normal operating reactor dome pressure, (2) an increase in core power, (3) an increase in the maximum licensed core flow, (4) a change to source term methodology, (5) a new fuel product line, or (6) a change in fuel cycle length. As such, the impact on plant operation is minimal and, as demonstrated in the M+SAR, the MELLLA+ range expansion can be accomplished without exceeding any existing regulatory limits or design allowable limits applicable to BSEP.

As part of MELLLA+ implementation for BSEP, CP&L will also implement the DSS-CD approach to automatically detect and suppress THI. Since the DSS-CD approach will continue to provide reliable, automatic detection and suppression of stability related power oscillations and provide protection against violation of the SLMCPR for anticipated oscillations, compliance with General Design Crieria (GDC) 10 and 12 of 10 CFR 50, Appendix A is maintained.

This request is not being submitted as a risk informed licensing action, as defined by Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998 (i.e., Reference 5). However, it was evaluated from a risk perspective and, as demonstrated in Section 10.5,

"Individual Plant Evaluation," of the M+SAR, the net increase in core damage frequency (CDF) and large early release frequency (LERF) are not significant.

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 amendment does 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 amendment meets 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. Letter from the U. S. Nuclear Regulatory Commission to Mr. John S. Keenan, "Issuance Of Amendment Re: Extended Power Uprate," dated May 31, 2002. (ML021430551).
2. GE Nuclear Energy Report NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 1, dated August 2002.
3. GE Nuclear Energy Report NEDC-33063, "Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus," dated November 2002.
4. GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002.
5. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998.

Precedents

BSEP is the lead plant for MELLLA+ and DSS-CD implementation.

Marked-up Technical Specification Pages - Unit 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.3.1.1.19~~	~~Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 25% and recirculation drive flow is ≤ 60%.~~	~~24 months~~

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux —High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux —High (Setdown)	2	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 22.7% RTP
b. Simulated Thermal Power —High	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	~~≤ 0.55W + 62.6% RTP~~ (b) and ≤ 117.1% RTP

≤0.61W+65.2% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~≤ [0.55 (W − ΔW) + 62.6% RTP]~~ when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.

(c) Each APRM channel provides inputs to both trip systems.

≤[0.61(W-ΔW)+ 65.2% RTP]

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux —High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3[c]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 20% RTP	3[c]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 ~~SR 3.3.1.1.19~~	NA ~~[d]~~
3. Reactor Vessel Steam Dome Pressure —High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level —Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve —Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure —High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.

~~(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.~~

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

the plant is not operating in the MELLAPLUS region defined in the COLR and

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 3.3.1.1, "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)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs)

LCO 3.4.3 The safety function of ~~10~~ 11 SRVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

Exit the MELLLA+ operating region defined in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more required~~ SRV~~s~~ inoperable.	A.1 ~~Be in MODE 3.~~	12 hours
	~~AND~~	
	~~A.2 Be in MODE 4.~~	~~36 hours~~
B. Two or more SRVs inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND	12 hours
	B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.1 Verify the safety function lift setpoints of the ~~required 10~~ 11 SRVs are as follows:	In accordance with the Inservice Testing Program

Number of SRVs	Setpoint (psig)
4	1130 ± 33.9
4	1140 ± 34.2
3	1150 ± 34.5

(continued)

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

3. → 4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1; and

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).

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)

4. The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating region for Specifications 3.4.1 and 3.4.3.

Marked-up Technical Specification Pages - Unit 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
~~SR 3.3.1.1.19 Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 25% and recirculation drive flow is ≤ 60%.~~	~~24 months~~

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux —High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux —High (Setdown)	2	3[c]	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 22.7% RTP
b. Simulated Thermal Power —High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	~~≤ 0.55W + 62.6% RTP~~ [b] and ≤ 117.1% RTP

≤ 0.61W + 65.2% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~≤ [0.55(W − ΔW) + 62.6% RTP]~~ when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.

(c) Each APRM channel provides inputs to both trip systems.

≤ [0.61(W − ΔW) + 65.2% RTP]

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux —High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3[c]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 20% RTP	3[c]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 ~~SR 3.3.1.1.19~~	NA ~~[d]~~
3. Reactor Vessel Steam Dome Pressure —High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level —Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve —Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure —High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.

~~(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.~~

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

the plant is not operating in the MELLLA+ region defined in the COLR and

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 3.3.1.1, "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)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs)

LCO 3.4.3 The safety function of ~~10~~ 11 SRVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

Exit the MELLLA+ operating region defined in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more required~~ SRV~~s~~ inoperable.	A.1 ~~Be in MODE 3.~~	12 hours
	~~AND~~	
	~~A.2 Be in MODE 4.~~	~~36 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.1 Verify the safety function lift setpoints of the ~~required 10~~ 11 SRVs are as follows: Number of SRVs — Setpoint (psig) 4 — 1130 ± 33.9 4 — 1140 ± 34.2 3 — 1150 ± 34.5	In accordance with the Inservice Testing Program

(continued)

B. Two or more SRVs inoperable OR Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 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 period-based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and~~

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

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).

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)

4. The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating region for Specifications 3.4.1 and 3.4.3.

Marked-up Technical Specification Bases Pages - Unit 1
(For Information Only)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (APRM) (continued)

channel is assigned to one, two or four OPRM "cells," forming a total of 24 separate OPRM cells per APRM channel, each with either three or four detectors. LPRMs near the edge of the core are assigned to either one or two OPRM cells. ~~A minimum of 18 OPRM cells in an APRM channel must have at least two OPERABLE LPRMs for the OPRM Upscale Function 2.f to be OPERABLE (Ref. 22)~~.

The minimum number of OPRM cells in an APRM channel will be determined by a function in the DSS-CD algorithm. The same number of OPRM cells, each with a minimum of two LPRMs, must be OPERABLE for the OPRM Upscale Function 2.f to be OPERABLE (Ref. 20).

2.a. Average Power Range Monitor Neutron Flux—High (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High (Setdown) Function will provide the primary trip signal for a core-wide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High (Setdown) Function. However, this Function is credited in calculations used to eliminate the need to perform the spatial analysis required for the Intermediate Range Monitor Neutron Flux—High Function (Ref. 6). In addition, the Average Power Range Monitor Neutron Flux—High (Setdown) Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 23% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 23% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 23% RTP.

The Average Power Range Monitor Neutron Flux—High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High (Setdown) (continued)

In MODE 1, the Average Power Range Monitor Simulated Thermal Power—High and Neutron Flux—High Functions provide protection against reactivity transients and the RWM and Rod Block Monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power—High

The Average Power Range Monitor Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant, nominally 6 seconds, representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of rated recirculation drive flow (W) in percent and is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Simulated Thermal Power—High Function provides a general definition of the licensed core power/core flow operating domain.

A note is included, applicable when the plant is in single recirculation loop operation per LCO 3.4.1, which requires reducing by ΔW the flow value used in the Allowable Value equation. The value of ΔW is defined in plant procedures. The value of ΔW is established to adjust the SLO limit down in power approximatley ~~8.5%~~ 17.3% RTP to reflect the difference between the analyzed limits for two-recirculation loop operation (TLO) and SLO. The adjustment maintains the SLO limits at approximately the same absolute thermal power level as was established prior to extended power uprate. The ~~8.5%~~ 17.3% RTP has been converted to an equivalent "ΔW" value for convenience of representation and to reflect the way the adjustment is actually made in the APRM equipment. In addition to this adjustment, the actual ΔW value entered into the equipment includes an allowance for additional flow measurement uncertainties that may occur in SLO. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.e. 2-Out-Of-4 Voter (continued)

independently voted sets of Functions, each of which is redundant (four total outputs). The analysis in Reference 15 took credit for this redundancy in the justification of the 12-hour Completion Time for Condition A, so the voter Function 2.e must be declared inoperable if any of its functionality is inoperable. The voter Function 2.e does not need to be declared inoperable due to any failure affecting only the APRM Interface hardware portion of the Two-Out-Of-Four Logic Module.

There is no Allowable Value for this Function.

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

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

Replace with Insert 1

~~References 17, 18 and 19 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period-based detection algorithm, the amplitude-based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense-in-depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period-based detection algorithm.~~

The OPRM Upscale Function receives input signals from the LPRMs within the reactor core, which are combined into "cells" for evaluation by the OPRM algorithms. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. Three of the four channels are required to be OPERABLE.

23%

The OPRM Upscale trip is automatically enabled (bypass removed) when THERMAL POWER is ≥ ~~25%~~ RTP, as indicated by the APRM Simulated Thermal Power, and reactor core flow is ≤ ~~60%~~ of rated flow, as indicated by APRM measured

75%

(continued)

Insert 1

Reference 20 describes the primary algorithm used in the OPRM for detecting thermal-hydraulic instability related neutron flux oscillations: the confirmation density algorithm. References 17, 18, and 19 describe three additional algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. All four algorithms are implemented in the OPRM Upscale Functions, but the safety analysis only takes credit for the confirmation density algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the confirmation density algorithm.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

recirculation drive flow. This is the operating region where actual thermal-hydraulic instability and related neutron flux oscillations may occur. See Reference ~~21~~ 20 for additional discussion of OPRM Upscale trip enable region limits. ~~The 25% RTP lower boundary of the enabled region was established by scaling the 30% value in Reference 21 for uprated power to correspond to 30% of original plant RTP. This scaling is not required by Reference 21, but has been done for conservatism.~~

Replace with Insert 2

These setpoints, which are sometimes referred to as the "auto-bypass" setpoints, establish the boundaries of the OPRM Upscale trip enabled region. The APRM Simulated Thermal Power auto-enable setpoint has 1% deadband while the drive flow setpoint has a 2% deadband. The deadband for these setpoints ~~is~~ was established so that it increased the enabled region.

The OPRM Upscale Function is required to be OPERABLE when the plant is at ≥ 20% RPT. The 20% RTP level is selected to provide margin in the unlikely event that a reactor power increase transient occurring while the plant is operating below ~~25%~~ 23% RTP causes a power increase to or beyond the ~~25%~~ 23% APRM Simulated Thermal Power OPRM Upscale trip auto-enable setpoint without operator action. This OPERABILITY requirement assures that the OPRM Upscale trip auto-enable function will be OPERABLE when required.

Replace with Insert 3

~~An OPRM Upscale trip is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with period confirmations and relative cell amplitude exceeding specified setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel trip. An OPRM Upscale trip is also issued from the channel if either the growth rate or amplitude based algorithms detect growing oscillatory changes in the neutron flux for one or more cells in that channel.~~ (Note: To facilitate placing the OPRM Upscale Function 2.f in one APRM channel in a

(continued)

Insert 2

The 23% RTP lower boundary of the enabled region was established at the lowest value at which APRM calibration to core thermal power is performed.

Insert 3

An OPRM Upscale trip is issued from an APRM channel when the confirmation density algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with period confirmation confirmed in a number of OPRM cells exceeding a preset value. An OPRM Upscale trip is also issued from the channel if any of the other three OPRM algorithms detect oscillatory changes in the neutron flux for one or more cells in that channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

"tripped" state, if 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.)

Replace with Insert 4

~~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, 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

(continued)

Insert 4

There are four "sets" of OPRM related setpoints or adjustment parameters: (a) OPRM trip auto-enable setpoints for STP (23%) and drive flow (75%), (b) confirmation density algorithm (CDA) setpoints, (c) algorithm tuning parameters, and (d) period based detection algorithm (PBDA), growth rate algorithm (GRA), and amplitude based algorithm (ABA) setpoints.

The first set, the OPRM auto-enable region setpoints, are treated as nominal setpoints with no additional margins added. The second set, the OPRM CDA trip setpoints, are established in accordance with methodologies defined in Reference 20, and are documented in plant procedures. There are no allowable values for these setpoints. The third set, the OPRM algorithm tuning parameters, are established or adjusted in accordance with and controlled by plant procedures. The fourth set, the PBDA, GRA, and ABA setpoints, are established in accordance with Reference 20 as nominal values only and are controlled by plant procedures.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing - applicable to Function 2.b and the auto-enable portion of Function 2.f only), the 2-Out-Of-4 Voter channels, and the interface connections into the RPS trip systems from the voter channels.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184-day Frequency of SR 3.3.1.1.11 is based on the reliability analyses of References 15 and 16. (NOTE: The actual voting logic of the 2-Out-Of-4 Voter Function is tested as part of SR 3.3.1.1.15.) ~~The auto-enable setpoints for the OPRM Upscale trip are confirmed by SR 3.3.1.1.19.)~~

A Note is provided for Function 2.a that requires this SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

A second Note is provided for Functions 2.b and 2.f that clarifies that the CHANNEL FUNCTIONAL TEST for Functions 2.b and 2.f includes testing of the recirculation flow processing electronics, excluding the flow transmitters.

SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The CHANNEL CALIBRATION for Functions 5 and 8 should consist of a physical inspection and actuation of the associated position switches.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.17 (continued)

The 24 month Frequency is consistent with the BNP refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.1.1.18

The APRM Simulated Thermal Power—High Function (Function 2.b) uses drive flow to vary the trip setpoint. The OPRM Upscale Function (Function 2.f) uses drive flow to automatically enable or bypass the OPRM Upscale trip output to RPS. Both of these Functions use drive flow as a representation of reactor core flow. SR 3.3.1.1.13 assures that the drive flow transmitters and processing electronics are calibrated. This SR adjusts the recirculation drive flow scaling factors in each APRM channel to provide the appropriate drive flow/core flow alignment.

The Frequency of once within 7 days after reaching equilibrium conditions following a refueling outage is based on the expectation that any change in the core flow to drive flow functional relationship during power operation would be gradual and the maintenance on the Recirculation System and core components which may impact the relationship is expected to be performed during refueling outages. The 7 day time period after reaching equilibrium conditions is based on plant conditions required to perform the test, engineering judgment of the time required to collect and analyze the necessary flow data, and engineering judgment of the time required to enter and check the applicable scaling factors in each of the APRM channels. The 7-day time period after reaching equilibrium conditions is acceptable based on the relatively small alignment errors expected, and the margins already included in the APRM Simulated Thermal Power—High and OPRM Upscale Function trip-enable setpoints.

~~SR 3.3.1.1.19~~

~~This surveillance involves confirming the OPRM Upscale trip auto-enable setpoints. The auto-enable setpoint values are considered to be nominal values as discussed in Reference 21. This surveillance ensures that the OPRM~~

(continued)

BASES

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.3.1.1.19 (continued)~~

~~Upscale trip is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.3) and core flow (SR 3.3.1.1.18), respectively.~~

~~In any auto enable setpoint is nonconservative (i.e, the OPRM Upscale trip is bypassed when APRM Simulated Thermal Power ≥ 25% and recirculation drive flow ≤ 60%), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the OPRM Upscale trip auto enable setpoint(s) may be adjusted to place the channel in a conservative condition (not bypassed). If the OPRM Upscale trip is placed in the not bypassed condition, this SR is met and the channel is considered OPERABLE.~~

~~The Frequency of 24 months is based on engineering judgment and reliability of the components.~~

REFERENCES

1. UFSAR, Section 7.2.

2. UFSAR, Chapter 15.0.

3. UFSAR, Section 7.2.2.

4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.

5. 10 CFR 50.36(c)(2)(ii).

6. NEDO-23842, Continuous Control Rod Withdrawal in the Startup Range, April 18, 1978.

7. UFSAR, Section 5.2.2.

8. UFSAR, Appendix 5.2A.

9. UFSAR, Section 6.3.1.

(continued)

BASES

REFERENCES (continued)

10. P. Check (NRC) letter to G. Lainas (NRC), BWR Scram Discharge System Safety Evaluation, December 1, 1980.

11. NEDC-30851-P-A, Technical Specification Improvement Analyses for BWR Reactor Protection System, March 1988.

12. MDE-81-0485, Technical Specification Improvement Analysis for the Reactor Protection System for Brunswick Steam Electric Plant, Units 1 and 2, April 1985.

13. UFSAR, Table 7.2.1-3.

14. NEDO-32291-A, System Analyses for the Elimination of Selected Response Time Testing Requirements, October 1995.

15. NEDC-32410P-A, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, October 1995.

16. 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.

17. NEDO-31960-A, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, November 1995.

18. NEDO-31960-A, Supplement 1, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, November 1995.

19. NEDO-32465-A, BWR Owners' Group Long-Term Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications, August 1996.

20. NEDC-33075P, "Detect And Suppress Solution-Confirmation Density Licensing Topical Report," Revision 2, dated November 2002.

~~20. Letter, L. A. England (BWROG) to M. J. Virgilio, BWR Owners' Group Guidelines for Stability Interim Corrective Action, June 6, 1994.~~

~~21. BWROG Letter 96113, K. P. Donovan (BWROG) to L. E. Phillips (NRC), Guidelines for Stability Option III "Enable Region" (TAC M92882), September 17, 1996.~~

(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

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.

However, single loop operation is not allowed in the MELLLA+ operating region defined in the COLR.

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

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (SRVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The SRVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the relief mode (However, for the purposes of this LCO, only the safety mode is required). In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each SRV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The SRVs that provide the relief mode are the Automatic Depressurization System (ADS) valves. The ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

Design Basis Event

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe ~~transient~~ is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 9 SRVs are assumed to operate in the safety mode. The analysis results demonstrate that the design SRV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). ~~This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event~~.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Replace with Inserts

~~For overpressurization associated with an ATWS event, 10 SRVs are assumed to operate in the safety mode. The analysis (Ref. 2) results demonstrate that the design capacity is capable of maintaining reactor pressure below the ASME Section III Code Service Level C limits (1500 psig).~~

From an overpressure standpoint, the design basis events are bounded by the overpressurization associated with the ATWS event described above. Reference 3 discusses additional events that are expected to actuate the SRVs.

Insert 6

SRVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The safety function of ~~10~~ 11 SRVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1, 2, ~~and 3~~ 3, and 6). The requirements of this LCO are applicable only to the capability of the SRVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The SRV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. These setpoints also ensure that in the event of an ATWS, the reactor pressure remains below the ATWS limit of 1500 psig. The transient evaluations in the UFSAR involving the safety mode are based on these setpoints, but also include the additional uncertainties of ± 3% of the nominal setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, all required SRVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in

(continued)

Insert 5

For overpressuization associated with an ATWS event initiated from within the MELLLA operating region, 10 SRVs operated in the safety mode will maintain reactor pressure below the ASME Section III Code Service Level C limits of 1500 psig (Ref. 2). For overpressuization associated with an ATWS event initiated from within the MELLLA+ operating region, 11 SRVs operated in the safety mode will maintain reactor pressure below the ASME Section III Code Service Level C limits (Ref. 6). However, to bound potential surveillance test results, this MELLLA+ case assumed a 10% setpoint drift on a limiting SRV. The results showed that the vessel overpressure criterion was met with this configuration.

Insert 6

This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event and the acceptance limit of 1500 psig is met during an ATWS Event.

BASES

APPLICABILITY (continued)

these MODES. The SRVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The SRV function is not needed during these conditions.

ACTIONS

A.1 ~~and A.2~~

Replace with Insert 7

~~With less than the minimum number of required SRVs OPERABLE, a transient may result in the violation of the ASME Code limits on reactor pressure. If the safety function of one or more required SRVs is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

Insert 8

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the required ~~10~~ 11 SRVs will open at the pressures assumed in the safety analysis of References 1, 2, ~~and 3~~ 3, and 6. The demonstration of the SRV safety function lift settings must be performed during shutdown, since this is a bench test, in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

SR 3.4.3.2

A manual actuation of each required SRV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This

(continued)

Insert 7

In order to ensure that the reactor pressure remains below the ATWS transient limit of 1500 psig, for events initiated while in the MELLLA+ operating region, 11 SRVs must be OPERABLE. If the safety function of one SRV is inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must exit the MELLLA+ operating region, as defined in the COLR, within 12 hours. The allowed Completion Time is reasonable to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Insert 8

B.1 and B.2

If the safety function of two or more SRVs is inoperable, a transient may result in the violation of reactor vessel pressure ASME Code limits regardless of the operating region from which the event initiated. As such, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test. Adequate pressure at which this test is to be performed, to avoid damaging the valve, is 945 psig. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is considered OPERABLE.

The 24 month Frequency was developed based on the SRV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.2.2.2.

2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, Supplement 1, March 1996.

3. UFSAR, Chapter 15.

4. 10 CFR 50.36(c)(2)(ii).

5. ASME, Boiler and Pressure Vessel Code, Section XI.

6. NEDC-33063P, Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus, November 2002.

List Of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light (CP&L) Company in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to the Manager - Regulatory Affairs at the Brunswick Steam Electric Plant.

Commitment	Schedule
1. None	N/A