Progress Energy

Cornelius J. Gannon
Vice President
Brunswick Nuclear Plant
Progress Energy Carolinas, Inc.

MAR 0 5 2004

SERIAL: BSEP 04-0015
TSC-2002-09

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
Supplement to Request for License Amendments
Core Flow Operating Range Expansion
(NRC TAC No. MB6692 and MB6693)

References:

1. Letter from John S. Keenan to the U. S. Nuclear Regulatory Commission (Serial: BSEP 02-0169), "Request for License Amendments - Core Flow Operating Range Expansion," dated November 12, 2002 (ML023240227)
2. GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3, dated January 2004.

Ladies and Gentlemen:

On November 12, 2002, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC) requested a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Units 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+)).

As part of MELLLA+ implementation for BSEP, PEC will implement the Detect and Suppress Solution - Confirmation Density (DSS-CD) approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities (THI). The November 12, 2002, submittal was based, in part, on General Electric (GE) Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002. On January 23, 2004, GE submitted Revision 3 to NEDC-33075P, which includes a new Automated Backup Stability Protection (ABSP) function associated with implementation of DSS-CD. The ABSP is an automatic reactor scram region, implemented by the Average Power Range Monitor (APRM) flow-biased scram setpoint. It may be used if the Oscillation Power Range Monitoring (OPRM) system is inoperable to allow continued operation within the MELLLA+ operating domain. The purpose of this submittal is to update the BSEP TSs supporting MELLLA+/DSS-CD

P.O. Box 10429
Southport, NC 28461

T> 910.457.3698
F> 910.457.2803

A001

implementation to include the new ABSP function. Enclosure 1 provides a more detailed discussion of the changes being made by this submittal. Enclosures 2 and 3 provide marked-up TS pages for Units 1 and 2, respectively. For convenience, the entire set of marked-up TS and TS Bases pages are being submitted (i.e., those pages not affected by this supplement are also included).

PEC plans to install the modifications necessary to implement MELLLA+/DSS-CD during the Unit 1 Cycle 15 Refueling Outage (i.e., B115R1, which began in February 2004) and the Unit 2 Cycle 17 Refueling Outage (i.e., B217R1, currently scheduled to begin in March 2005). However, it is recognized that approval of the requested amendment to support Unit 1 startup in March 2004 is not achievable. PEC has determined, in accordance with the provisions of 10 CFR 50.59, that installation of the required modifications and operation in accordance with the current TSs (i.e., using the BWROG Option III Reactor Stability Long-Term Solution as implemented by the DSS-CD defense in depth algorithms which are essentially the same as the existing Option III algorithms) does not require prior NRC approval. To establish operability of the modified OPRM system, the system will not be armed during the initial startup from the B115R1 refueling outage. This will allow for data collection and verification of proper algorithm performance and to confirm the plant does not exhibit increased sensitivity to spurious noise not previously observed. The system will be armed and operable once proper operation is confirmed.

To support the above implementation plans, PEC requests that the amendments, once approved, be issued effective immediately, to be implemented within 180 days from approval for Unit 1 and prior to the startup from the B217R1 refueling outage for Unit 2.

PEC has evaluated the proposed changes 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.

In accordance with 10 CFR 50.91(b), PEC is providing the State of North Carolina a copy of the supplement to 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,

Cornelius J. Gannon

MAT/mat

Enclosures:

1. Evaluation of Supplemental Amendment Request
2. Marked-up Technical Specification Pages - Unit 1
3. Marked-up Technical Specification Pages - Unit 2
4. Marked-up Technical Specification Bases Pages - Unit 1 (For Information Only)
5. List of Regulatory Commitments

Cornelius J. Gannon, 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.

Dean S. Mason

Notary (Seal)

My commission expires: August 29, 2004

cc:

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. Eugene M. DiPaolo, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

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

Ms. Jo A. Sanford
Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Ms. Beverly O. Hall, Section Chief
Radiation Protection Section, Division of Environmental Health
North Carolina Department of Environment and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609-7221

Evaluation of Supplemental Amendment Request

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

1.0 Description

On November 12, 2002, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc. (PEC) requested a revision to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2 (i.e., Reference 1). 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+)).

As part of MELLLA+ implementation for BSEP, PEC will implement the Detect and Suppress Solution - Confirmation Density (DSS-CD) approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities (THI). The November 12, 2002, submittal was based, in part, on General Electric (GE) Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 2, dated November 2002. On January 23, 2004, GE submitted Revision 3 to NEDC-33075P (i.e., Reference 2), which includes a new Automated Backup Stability Protection (ABSP) function associated with implementation of DSS-CD. The ABSP is an automatic reactor scram region, implemented by the Average Power Range Monitor (APRM) flow-biased scram setpoint. It may be used if the Oscillation Power Range Monitoring (OPRM) system is inoperable to allow continued operation within the MELLLA+ operating domain. As a result, the BSEP TSs supporting MELLLA+/DSS-CD implementation are updated to include the new ABSP function. In addition to the inclusion of ABSP, other minor administrative changes have been included.

2.0 Changes Proposed by Supplement / Analysis

The following table provides a discussion of the TS changes proposed by this supplement. Only the changes associated with the supplement are discussed. Other changes supporting MELLLA+/DSS-CD implementation were provided in the November 12, 2002, request.

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" *Required Action I (to be entered when OPRM Upscale – TS Table 3.3.1.1-1, Function 2.f, is inoperable)*	I.1. Initiate Alternate method to detect and suppress thermal hydraulic instability oscillations. (12 hours)	I.1 Initiate action to implement the Manual BSP Regions defined in the COLR. (Immediately) <u>AND</u> I.2.1 Implement the Automated BSP Scram Region using the modified APRM flow-biased

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
		scram setpoints defined in the COLR. (12 hours) OR I.2.2 Reduce operation to below the BSP Boundary defined in the COLR. (12 hours) AND I.3 Initiate action in accordance with Specification 5.6.7. (120 days) **Justification:** **NEDC-33075P (Reference 2)** This change is more restrictive than the existing Required Action. The time allowed for implementing the alternate method to detect and suppress THI is revised from 12 hours to immediately. Additionally, a new scram function or restricted operation is required for continued operation and is required to be implemented within 12 hours. These restrictions serve to lessen the potential for and effects of THI, should it occur. The new Required Action I.3 imposes a reporting requirement should the OPRM system be inoperable for 120 days. This ensures that the NRC is made aware of long term issues and the plans and schedules for the resolution of such issues.
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" *Required Action J.1*	J.1 Reduce THERMAL POWER to < 20% RTP. (4 hours)	J.1 Reduce THERMAL POWER to < 18% RTP. (4 hours) **Justification:** **NEDC-33075P (Reference 2)** This change is more restrictive than the existing Required Action J.1. The OPRM Upscale Function is required to be operable at ≥ 18% rated thermal power (RTP). This provides a 5% RTP

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
		margin to the lower boundary of the Armed Region, thereby encompassing the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. This 5% margin is consistent with and maintains the existing 5% margin operability requirements for the Option III OPRM Upscale operability requirements.
TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" *Table 3.3.1.1-1, Function 2.b, Simulated Thermal Power - High*	None	Added Note d to Allowable Value (d) With OPRM Upscale (function 2.f) inoperable, the modified APRM flow-biased setpoints defined by the COLR may be required to implement the Automated BSP Scram Region in accordance with Action I of this Specification. **Justification:** **NEDC-33075P (Reference 2)** This change provides TS instructions for implementing the Automated BSP Scram Region, consistent with the new Action I.2.1 discussed above.
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)* *Applicable Modes or Other Specified Conditions*	≥ 20% RTP	≥ 18% RTP **Justification:** **NEDC-33075P (Reference 2)** The OPRM Upscale Function is required to be operable at ≥ 18% RTP. This provides a 5% RTP margin to the lower boundary of the Armed Region, thereby encompassing the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
		neutron flux oscillations. This 5% margin is consistent with and maintains the existing 5% margin operability requirements for the Option III OPRM Upscale operability requirements.
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)* *Note (e) – Added to "Applicable Modes or Other Specified Conditions" for Function 2.f*	 None	 (e) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD shall be OPERABLE and capable of arming for operation at recirculation drive flow rates above the DSS-CD Armed Region. **Justification:** This supplement adds a new Note (e) to allow the plant to pass through the newly established DSS-CD Armed Region for one controlled startup and controlled shutdown without the DSS-CD being armed. This is a one-time only allowance, the purpose of which is to ensure that no unexpected scrams occur as a result of the new DSS-CD algorithm.
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 when in the MELLLA+ Operating Region. OR The safety function of 10 SRVs shall

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
		be OPERABLE when outside the MELLLA+ Operating Region. **Justification:** The changes to TS 3.4.3 proposed in this supplement only affect the Limiting Conditions for Operation (LCO) format. This supplement revises the wording of LCO 3.4.3 to specify 11 SRVs are required to be operable when in the MELLLA+ Operating Region and 10 SRVs are required to be operable when outside the MELLLA+ Operating Region. This change was made for consistency with other TS LCOs (e.g., TS 3.4.1, "Recirculation Loops Operating"), where LCO requirements vary. The proposed format will assist operators in identifying operability requirements.
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	3. The Manual Backup Stability Protection (BSP) Scram Region (Region 1), the Manual BSP Controlled Entry Region (Region 2), the modified APRM flow-biased setpoints used in the Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1; **Justification:** **NEDC-33075P (Reference 2)** The November 12, 2002, amendment request (i.e., Reference 1) deleted the existing Item 3 which dealt with the PBDA, associated with the BWROG Option III Reactor Stability Long-Term Solution. This supplement adds a new Item 3 which requires the Core Operating Limits Report (COLR) to include the BSP regions necessary to support Required Action I of TS 3.3.1.1.

Changes Proposed by Supplement		
Specification	**Existing Requirement**	**Proposed Requirement**
TS 5.6.5 Core Operating Limits Report (COLR) *a. 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).*	None	2. NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density" (latest approved version). 3. NEDE-23785P-A, 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," March 2002. 4. NEDE-32906P-A, Revision 1, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," April 2003. **Justification:** The latest approved version of NEDE-24011-P, "General Electric Standard Application for Reactor Fuel," does not include the topical reports used in support of MELLLA+ / DSS-CD implementation. This is an administrative change to include them for BSEP specific use, until such time as NEDE-24011-P is updated and approved by the NRC.
TS 5.6.7 Oscillation Power Range Monitor (OPRM) Report	None	When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status. **Justification:** **NEDC-33075P (Reference 2)** This is a new reporting requirement which is added by this supplement. The report supports Required Action I.3 of TS 3.3.1.1.

In summary, the follow TS changes are proposed by this supplement.

(1) TS 3.3.1.1, Required Action I is revised to be consistent with NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3, dated January 2004. The existing requirement to initiate an alternate (i.e., manual) method to detect and suppress thermal hydraulic instability oscillations is expanded to include a requirement to either implement an automated backup stability protection (i.e., Required Action I.2.1) or exit the operating region most susceptible to rapid onset of THI (i.e., Required Action I.2.2). Additionally, a new Required Action I.3 is included. Required Action I.3 ensures that a report is made to the NRC, if DSS-CD is inoperable for 120 days. This change results in more conservative plant operation than proposed by the original November 12, 2002, request.

(2) TS 3.3.1.1, Required Action J.1 is revised to require the plant to be < 18% RTP versus < 20% RTP in the event that the OPRM Upscale Function is inoperable and the Required Actions associated with Action I are not completed. The OPRM Upscale Function is required to be operable at ≥ 18% RTP. This provides a 5% RTP margin to the lower boundary of the Armed Region, thereby encompassing the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. This 5% margin is consistent with and maintains the existing 5% margin operability requirements for the Option III OPRM Upscale operability requirements. This change is also reflected in TS 3.3.1.1, Table Function 2.f, "Average Power Range Monitors, OPRM - Upscale," Applicable Modes or Other Specified Conditions, where the operability requirement for Function 2.f is changed from ≥ 20% RTP to ≥ 18% RTP.

(3) A note is added to TS 3.3.1.1, Table 3.3.1.1-1, Function 2.b, "Simulated Thermal Power - High," to provide TS instructions for implementing the Automated BSP Scram Region, consistent with the new Action I.2.1.

(4) A note is added to TS 3.3.1.1, Table 3.3.1.1-1, Function 2.f, "Average Power Range Monitors, OPRM - Upscale," to allow the plant to pass through the newly established DSS-CD Armed Region for one controlled startup and controlled shutdown without the DSS-CD being armed. This is a one-time only allowance, the purpose of which is to ensure that no unexpected scrams occur as a result of the new DSS-CD algorithm.

(5) The TS 3.4.3 change, proposed in this supplement, only affects the LCO format. Specifically, the LCO is revised from requiring 11 SRVs to be operable to specifying that 11 SRVs are required to be operable when in the MELLLA+ Operating Region and 10 SRVs are required to be operable when outside the MELLLA+ Operating Region. This change was made for consistency with other TS LCOs (e.g., TS 3.4.1, "Recirculation Loops Operating"), where LCO requirements vary. The proposed format will assist operators in identifying operability requirements.

(6) The November 12, 2002, amendment request deleted the existing Item 3 which dealt with the PBDA, associated with the BWROG Option III Reactor Stability Long-Term Solution. This supplement adds a new Item 3 which requires the COLR to include the BSP regions necessary to support Required Action I of TS 3.3.1.1.

(7) NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density," (latest approved version); NEDE-23785P-A, 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," March 2002; and NEDE-32906P-A, Revision 1, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," April 2003, are added as to the list of analytical methods used to determine the core operating limits contained in TS 5.6.5.b. The latest approved version of NEDE-24011-P, "General Electric Standard Application for Reactor Fuel," does not include the topical reports used in support of MELLLA+ / DSS-CD implementation. This is an administrative change to include them for BSEP specific use, until such time as NEDE-24011-P is updated and approved by the NRC.

(8) A new report, "Oscillation Power Range Monitor (OPRM) Report," is added as TS 5.6.7. This new reporting requirement is added to support the new Required Action I.3 of TS 3.3.1.1.

The changes made in this supplement 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 and primarily result from changes made to the generic technical specifications included with NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3. The changes are considered to be either more restrictive in nature than those proposed in the original November 12, 2002, request or are administrative in nature.

PEC will make supporting changes to the TS Bases in accordance with TS 5.5.10, "Technical Specifications (TS) Bases Control Program." Enclosure 4 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 Regulatory Safety Analysis

3.1 No Significant Hazards Consideration

Section 11.3.3, of the 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., the M+SAR, Reference 3), evaluates whether or not a significant hazards consideration is involved with the DSS-CD portion of the proposed amendments. This basis for proposed no significant hazards consideration determination was published in the Federal Register on February 18, 2003 (i.e., 68 FR 7813). While the changes proposed in this supplement do not affect the conclusion 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, the scope of the request has expanded. This expansion is primarily a result of addition of the Automated Backup Stability Protection feature and the new actions proposed for TS 3.3.1.1, Required Action I, consistent with NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3, dated January 2004. As such, PEC has updated the 10 CFR 50.92 criteria assessment provided in the original November 12, 2002, request (i.e., Reference 1).

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

   Response: No

   The proposed change will implement DSS-CD as the long-term stability solution. The DSS-CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion to terminate the oscillations prior to any significant amplitude growth. The DSS-CD provides protection against violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for anticipated oscillations. Compliance with General Design Criteria (GDC) 10 and 12 of 10 CFR 50, Appendix A is accomplished via an automatic action. The DSS-CD introduces an enhanced detection algorithm that detects the inception of power oscillations and generates an earlier power suppression trip signal exclusively based on successive period confirmation recognition. The existing Option III algorithms are retained, with generic setpoints, to provide defense-in-depth protection for unanticipated reactor instability events.

   A developing instability event is suppressed by the DSS-CD system with substantial margin to the SLMCPR and no clad damage, with the event terminating in a scram and never developing into an accident. In addition, the DSS-CD solution defense-in-depth features incorporate all the backup scram algorithms plus the licensed scram feature of the existing Option III system. The DSS-CD system does not interact with equipment whose failure could cause an accident. Scram setpoints in the DSS-CD will be established so that analytical limits are met. The reliability of the DSS-CD will meet or exceed that of the existing system. No new challenges to safety-related equipment will result from the DSS-CD solution. Because an instability event would reliably terminate in an early scram without impact on other safety systems, there is no significant increase in the probability of an accident.

   The existing requirement to initiate an alternate (i.e., manual) method to detect and suppress thermal hydraulic instability oscillations is expanded to include a requirement to either implement an Automated Backup Stability Protection (ABSP) (i.e., Required Action I.2.1) or exit the operating region most susceptible to rapid onset of Thermal Hydraulic Instability (THI) (i.e., Required Action I.2.2). The ABSP is an automatic reactor scram region, implemented by the Average Power Range Monitor (APRM) flow-biased scram setpoint. It may be used if the Oscillation Power Range Monitoring (OPRM) system is inoperable to allow continued operation within the MELLLA+ operating domain. Additionally, a new Required Action I.3 is included. Required Action I.3 ensures that a report is made to the NRC, if DSS-CD is inoperable for 120 days.

To maintain the existing margin between equipment operability requirements and the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability, (1) TS 3.3.1.1, Required Action J.1 is revised to require the plant to be < 18% RTP versus < 20% RTP in the event that the OPRM Upscale Function is inoperable and the Required Actions associated with Action I are not completed, and (2) the operability requirement for the OPRM Upscale Function (i.e., TS 3.3.1.1, Table Function 2.f) is changed from ≥ 20% RTP to ≥ 18% RTP. This 5% margin is consistent with and maintains the existing 5% margin operability requirements for the Option III OPRM Upscale operability requirements.

Overall, these changes result in more conservative plant operation. Other changes proposed in this supplement are either in direct support of ABSP or are administrative in nature.

Proper operation of the DSS-CD system does not affect any fission product barrier or Engineered Safety Feature. Thus, the proposed change cannot change the consequences of any accident previously evaluated. As stated above, the DSS-CD solution meets the requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel SLMCPR.

Based on the above, the operation of the DSS-CD solution within the framework of the Option III OPRM hardware will not increase 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

   The DSS-CD solution operates within the existing Option III OPRM hardware. No new operating mode, safety-related equipment lineup, accident scenario, system interaction, or equipment failure mode was identified. The ABSP automatic reactor scram region is implemented by adjusting the existing APRM flow-biased scram setpoint. Therefore, the DSS-CD solution will not adversely affect plant equipment.

   Because there are no hardware changes, there is no change in the possibility or consequences of a failure. The worst case failure of the equipment is a failure to initiate mitigating action (i.e., scram), but no failure can cause an accident of a new or different kind than any previously evaluated.

Based on the above, the proposed change to the DSS-CD solution will not create the possibility 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 DSS-CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion to terminate the oscillations prior to any significant amplitude growth. The DSS-CD solution algorithm will maintain or increase the margin to the SLMCPR for anticipated instability events. The safety analyses in "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3 demonstrate the margin to the SLMCPR for postulated bounding stability events. Existing margin between equipment operability requirements and the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability are maintained. As a result, there is no impact on the SLMCPR identified for an instability event.

The existing requirement to initiate an alternate method to detect and suppress thermal hydraulic instability oscillations is expanded to include a requirement to either implement an ABSP (i.e., Required Action I.2.1) or exit the operating region most susceptible to rapid onset of THI (i.e., Required Action I.2.2). Additionally, a new Required Action I.3 is included. Required Action I.3 ensures that a report is made to the NRC, if DSS-CD is inoperable for 120 days. These change results in more conservative plant operation. Other changes proposed in this supplement are either in direct support of ABSP or are administrative in nature.

The current Option III algorithms (i.e., Period Based Detection, Amplitude Based, and Growth Rate) are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

Based on the above, the proposed change will not involve a significant reduction in the margin of safety.

Based on the above, PEC concludes that the proposed amendments, including the changes proposed in this supplement, 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.

3.2 Applicable Regulatory Requirements/Criteria

The changes proposed in this supplement do not affect the discussion of Applicable Regulatory Requirements/Criteria which was provided in the November 12, 2002, request (i.e., Reference 1).

4.0 Environmental Considerations

The changes proposed in this supplement do not affect the conclusion that, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments, which was provided in the November 12, 2002, request (i.e., Reference 1).

5.0 References

1. Letter from John S. Keenan to the U. S. Nuclear Regulatory Commission (Serial: BSEP 02-0169), "Request for License Amendments - Core Flow Operating Range Expansion," dated November 12, 2002 (ML023240227).
2. GE Nuclear Energy Report NEDC-33075P, "Detect And Suppress Solution - Confirmation Density Licensing Topical Report," Revision 3, dated January 2004.
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.

Marked-up Technical Specification Pages - Unit 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	~~I.1~~	~~Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.~~	~~12 hours~~
J. Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to < ~~20%~~ 18% RTP.	4 hours

I.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
AND		
I.2.1	Implement the Automated BSP Scram Region using the Modified APRM flow-biased scram setpoints defined in the COLR.	12 hours
OR		
I.2.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
AND		
I.3	Initiate action in accordance with specification 5.6.7.	120 days

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%.~~ (Not Used.)	~~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) ≤0.61W+65.2% RTP and ≤ 117.1% RTP (d)

(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]~~ ≤ [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.

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

(d) With OPRM Upscale (function 2.f) inoperable, the modified APRM flow-biased setpoints defined by the COLR may be required to implement the Automated BSP Scram Region in accordance with Action I of this specification.

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%~~ ≥ 18% RTP[(e)]	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[(e)]
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.

(e) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, the DSS-CD shall be OPERABLE and capable of arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

~~(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+ Operating 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, when in the MELLLA+ Operating Region.

OR

The safety function of 10 SRVs shall be OPERABLE when outside the MELLLA+ Operating Region.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more required~~ SRV~~s~~ inoperable.	A.1 ~~Be in MODE 3.~~ Exit the MELLLA+ Operating Region defined in the COLR. ~~AND~~ ~~A.2 Be in MODE 4.~~	12 hours ~~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 B.2 Be in Mode 4.	12 hours 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)

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:

Replace with Insert 1 from next page

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~~

Insert 2

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:

Insert 3

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)

Insert 1

The Manual Backup Stability Protection (BSP) Scram Region (Region 1), the Manual BSP Controlled Entry Region (Region 2), the modified APRM flow-biased setpoints used in the Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1;

Insert 2

5. The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Operating Region for Specifications 3.4.1 and 3.4.3.

Insert 3

2. NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density" (latest approved version).

3. NEDE-23785P-A, 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," March 2002.

4. NEDE-32906P-A, Revision 1, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," April 2003.

5.6 Reporting Requirements (continued)

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.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

Marked-up Technical Specification Pages - Unit 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	~~I.1~~	~~Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.~~	~~12 hours~~
J. Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to < ~~20%~~ 18% RTP.	4 hours

I.1 Initiate action to implement the Manual BSP Regions defined in the COLR. — Immediately

AND

I.2.1 Implement the Automated BSP Scram Region using the Modified APRM flow-biased scram setpoints defined in the COLR. — 12 hours

OR

I.2.2 Reduce operation to below the BSP Boundary defined in the COLR. — 12 hours

AND

I.3 Initiate action in accordance with Specification 5.6.7. — 120 days

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%.~~ (Not Used.)	~~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	~~≤ 0.66W + 62.6% RTP~~(b) ≤ 0.61W + 65.2% RTP and ≤ 117.1% RTP (d)

(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~~ ≤ [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.

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

(d) With OPRM Upscale (function 2.f) inoperable, the modified APRM flow-biased setpoints defined by the COLR may be required to implement the Automated BSP Scram Region in accordance with Action I of this specification.

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%~~ ≥ 18% RTP (e)	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.~~

(e) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, the DSS-CD shall be OPERABLE and capable of arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

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>

the plant is not operating in the MELLLA+ Operating 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, when in the MELLLA+ Operating Region.

OR

The safety function of 10 SRVs shall be OPERABLE when outside the MELLLA+ Operating Region.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~or more required~~ SRV~~s~~ inoperable.	A.1 ~~Be in MODE 3.~~ Exit the MELLLA+ Operating Region defined in the COLR. ~~AND~~ ~~A.2 Be in MODE 4.~~	12 hours ~~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 B.2 Be in Mode 4.	12 hours 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)

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;

Replace with Insert 1 from next page

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~~

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

Insert 2

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:

Insert 3

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)

Insert 1

The Manual Backup Stability Protection (BSP) Scram Region (Region 1), the Manual BSP Controlled Entry Region (Region 2), the modified APRM flow-biased setpoints used in the Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1;

Insert 2

5. The Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Operating Region for Specifications 3.4.1 and 3.4.3.

Insert 3

2. NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density" (latest approved version).

3. NEDE-23785P-A, 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," March 2002.

4. NEDE-32906P-A, Revision 1, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," April 2003.

5.6 Reporting Requirements

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.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Average Power Range Monitor (APRM) (continued)

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

<u>2.a. Average Power Range Monitor Neutron Flux—High (Setdown)</u> (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.

<u>2.b. Average Power Range Monitor Simulated Thermal Power—High</u>

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 approximately ~~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)

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

Replace entire discussion with Insert 1

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.

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 power range monitors (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.

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 recirculation drive flow. This is the operating region

(continued)

Insert 1

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.

Reference 20 describes the primary algorithm used in the OPRM for detecting thermal-hydraulic instability related neutron flux oscillations: the confirmation density algorithm. Reference 20 also describes 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.

The OPRM Upscale Function receives input signals from the local power range monitors (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. DSS-CD operability requires at least 8 responsive OPRM cells per channel.

The OPRM Upscale Function is required to be OPERABLE at ≥ 18% RTP. This provides a 5% RTP margin to the lower boundary of the Armed Region; thereby encompassing the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP corresponding to the MCPR monitoring threshold and reactor recirculation drive flow is ≤ 75% of rated flow, as indicated by APRM measured recirculation drive flow. This is the operating region. This OPERABILITY requirement assures that the OPRM Upscale trip auto-enable function will be OPERABLE when required.

Note (e) allows for entry into the OPRM Armed Region without automatic arming of the DSS-CD during both the first startup and the first controlled shutdown, which pass completely through the DSS-CD Armed Region, following DSS-CD implementation. Note (e) reflects the need for plant data collection in order to test the DSS-CD equipment. Testing DSS-CD equipment ensures its proper operation and prevents spurious reactor trips. While passing through the OPRM Armed Region, without arming the DSS-CD, during both the first startup and the first controlled shutdown following DSS-CD implementation, it is not necessary to enter Condition I. However, DSS-CD operability and capability to automatically arm shall be maintained or the appropriate Required Action entered, at recirculation drive flow rates above the DSS-CD Armed Region.

An OPRM Upscale trip is issued from an OPRM channel when the confirmation density algorithm in that channel detects oscillatory changes in the neutron flux, indicated by period confirmations exceeding a specified setpoint for a specified number of OPRM cells in the channel. An OPRM Upscale trip is also issued from the channel if any of the defense-in-depth algorithms (PBDA, ABA, GRA) exceed its trip condition for one or more cells in that channel. To facilitate placing the OPRM Upscale Function 2.f in one APRM channel in a "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.

Three of the four channels are required to be operable. 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.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

~~where actual thermal-hydraulic instability and related neutron flux~~ oscillations may occur. See Reference 21 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.

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 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% RTP causes a power increase to or beyond the 25% 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.

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 "tripped" state, if~~

Continued replacement with Insert 1

(continued)

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, 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.

Continued replacement with Insert 1

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

ACTIONS

D.1 (continued)

Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, G.1, and J.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Actions E.1 and J.1 are consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

Replace entire Action I discussion with Insert 2

I.1

Condition I exists when the OPRM Upscale Trip cabability has been lost for all APRM channels due to unanticipated equipment design or instability detection algorithm problems. References 15 and 16 justified use of alternate methods to detect and suppress oscillations under limited conditions. The alternate methods are procedurally established consistent with the guidelines identified in Reference 20. The alternate

(continued)

Insert 2

I.1

If OPRM Upscale trip capability is not maintained, Condition I exists and Backup Stability Protection (BSP) is required. The manual BSP Regions are described in Reference 20. The Manual BSP Regions are procedurally established consistent with the guidelines identified in Reference 20 and require specified manual operator actions if certain predefined operational conditions occur. The immediate Completion Time for Required Action I.1 is based on the importance of limiting the period of time during which no automatic or alternate detect and suppress trip capability is in place.

I.2.1 and I.2.2

Action I.2.1 and I.2.2 are alternative actions, either of which is taken in conjunction with Action I.1 when the OPRM Upscale trip capability is not maintained.

The Automated BSP Scram Region is designed to avoid reactor instability by automatically tripping the reactor, thereby preventing entry into the region of the power-flow map that is susceptible to reactor instability. The reactor trip is initiated by the modified APRM flow-biased scram setpoints for flow reduction events that would have terminated in the Manual BSP Region I. The Automated BSP Scram Region ensures an early scram and SLMCPR protection (Ref. 20).

The BSP Boundary defines an operating domain where potential instability events can be effectively addressed by the specified BSP manual operator actions. The BSP Boundary is constructed such that the immediate final statepoint for a flow reduction event initiated from this boundary and terminated at the core natural circulation line (NCL) would not exceed the Manual BSP Region I stability criterion. Potential instabilities would develop slowly as a result of the feedwater temperature transient (Ref. 20).

The Completion Time of 12 hours to complete the specified actions is reasonable, based on operational experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

I.3

Based on engineering judgment, the likelihood of an instability event that could not be adequately handled by the alternate methods during a 120-day period is negligible. The 120-day period is intended to allow for the case where limited design changes or extensive analysis might be required to understand or correct some unanticipated characteristic of the instability detection algorithms or equipment. Correction of the instability detection algorithms or equipment can include post-modification testing, during a reactor startup and/or shutdown, to establish operability. This action is not intended and was not evaluated as a routine alternative to returning failed or inoperable equipment to OPERABLE status. Correction of routine equipment failure or inoperability is expected to normally be accomplished with the Completion Times allowed for Conditions A and B.

If the required channels are not returned to an OPERABLE status within 120 days, the requirements of Specification 5.6.7 are followed. Since Backup Stability Protection is intended as a temporary means to protect against thermal-hydraulic instability events, the reporting requirements of Specification 5.6.7 document the corrective actions and schedule to restore the required channels to an OPERABLE status.

BASES

ACTIONS

I.1 (continued)

methods procedures require operating outside a "restricted zone" in the power-flow map and manual operator action to scram the plant if certain predefined events occur. The 12-hour allowed Completion Time for Required Action I.1 is based on engineering judgment to allow orderly transition to the alternate methods while limiting the period of time during which no automatic or alternate detect and suppress trip capability is formally in place. Based on the small probability of an instability event occurring at all, the 12 hours is judged to be reasonable.

This Required Action is intended to allow continued plant operation under limited conditions when an unanticipated equipment design or instability detection algorithm problem causes OPRM Upscale Function inoperability in all APRM channels. This Required Action is not intended and was not evaluated as a routine alternative to return failed or inoperable equipment to OPERABLE status. Correction of routine equipment failure or inoperability is expected to be accomplished within the completion times allowed for Required Actions for Condition A. The alternate method to detect and suppress oscillations implemented in accordance with I.1 is intended to be applied only as long as is necessary to implement corrective action to resolve the unanticipated equipment design or instability detection algorithm problem.

Continued replacement with Insert 2

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

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 RPS 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. 11, 15, and 16) assumption of the average time required to perform channel Surveillance.

(continued)

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 are compensated for by performing the 7 day

(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

(Not used.)

~~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 Upscale trip is enabled (not bypassed) for the correct values of~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.1.1.19 (continued)~~

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.

If 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. ~~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.~~

NEDC-33075P, Revision 3, "General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density," January 2004.

(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.

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.

However, single loop operation is not allowed in the MELLLA+ Operating Region defined in the COLR (Ref. 5).

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, operation in the MELLLA+ Operating Region is not permitted and 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)

the MELLLA+ Operating Region and

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.

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)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

speed conservatively equates to the 10% match criterion in terms of recirculation loop flow. The generator speed associated with the recirculation pump motor-generator set may be used to measure recirculation pump speed.

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

REFERENCES

1. UFSAR, Section 5.4.1.3.
2. UFSAR, Chapter 15.
3. NEDC-31776P, Brunswick Steam Electric Plant Units 1 and 2 Single Loop Operation, February 1990.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDC-33063P, Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus, November 2002.

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

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe ~~transient~~ Design Basis Event 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 Insert 3

~~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 4

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

LCO

Replace with Insert 5

~~The safety function of 10 SRVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1, 2, and 3).~~ 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 these MODES. The SRVs may

(continued)

Insert 3

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 4

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.

Insert 5

The safety function of 11 SRVs are required to be OPERABLE to satisfy the assumptions of the safety analysis when operating within the MELLLA+ Operating Region; the safety function of 10 SRVs are required to be OPERABLE to satisfy the assumptions of the safety analysis when operating within the MELLLA Operating Region (Refs. 1, 2, 3, and 6).

BASES

APPLICABILITY (continued)

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 6

~~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 7

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the required 10 SRVs will open at the pressures assumed in the safety analysis of References 1, 2, and 3. 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 can be demonstrated by the response of

(continued)

Insert 6

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 7

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)

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 Company, now doing business as Progress Energy Carolinas, Inc. (PEC) 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 - Support Services at the Brunswick Steam Electric Plant.

Commitment	Schedule
1. None	N/A