UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 5, 1998

Mr. C. S. Hinnant
Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: ISSUANCE OF AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING CONVERSION TO IMPROVED STANDARD TECHNICAL SPECIFICATIONS AND IMPLEMENTATION OF REACTOR STABILITY SOLUTION - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. M97243 AND M97244)

Dear Mr. Hinnant:

The Commission has issued the enclosed Amendment No. 203 to Facility Operating License No. DPR-71 and Amendment No. 233 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 1&2), respectively. The amendments consist of the changes described below.

The amendments reflect full conversion from your current Technical Specifications (TS) to a set of TS based on NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 1, dated April 1995. This is in response to your application dated November 1, 1996, as supplemented on October 13, 1997, February 26, 1998, March 13, 1998, April 24, 1998, and May 22, 1998.

Secondly, the amendments modify the Technical Specifications (TS) for BSEP 1&2 to allow for transition to the long-term reactor stability solution known as Enhanced Option 1-A.

Thirdly, the amendment removes a condition from the BSEP 2 operating license that currently does not allow single recirculation loop operation.

In addition, the NRC staff has reviewed and approved the following changes which, although included in your improved TS conversion submittal, were identified as being beyond the scope of the conversion.

- Extension of the surveillance frequency for reactivity anomalies as requested in Carolina Power & Light (CP&L) letter dated November 1, 1996.

- Insertion of a manual scram in lieu of placing the reactor mode switch in the shutdown position if certain actions are not completed for an inoperable control rod scram accumulator. This change was requested in CP&L letter dated November 1, 1996.

- Revision of certain instrumentation allowable values based upon the application of CP&L's setpoint methodology. These changes were requested in CP&L letter dated November 1, 1996.

- Extension of 18-month refueling interval surveillance requirements to 24 months. These changes were requested in CP&L letter dated November 1, 1996, as supplemented on October 13, 1997, April 24, 1998, and May 22, 1998.

- Allowance of short duration out-of-service times for various combinations of inoperable Emergency Core Cooling System (ECCS) subsystems instead of requiring an immediate shutdown. These changes were requested in CP&L letter dated November 1, 1996.

- Reduction in the number of automatic depressurization system (ADS) valves required to be operable from seven to six. This change was requested in CP&L letter dated November 1, 1996.

- Increase in the minimum pressure at which the ADS is required to be operable to 150 psig. This change was requested in CP&L letter dated November 1, 1996.

- A relaxation in the low pressure ECCS pump flow acceptance criteria. This change was requested in CP&L letter dated November 1, 1996.

- A relaxation in the core spray pump flow acceptance criterion. This change was requested in CP&L letter dated November 1, 1996.

- Deletion of the Limiting Condition for Operation (LCO) and associated surveillance requirement (SR) for primary containment internal pressure. These changes were requested in CP&L letter dated November 1, 1996.

- Increase in the allowed out-of-service time (AOT) for various combinations of inoperable service water pumps. These changes were requested in CP&L letter dated November 1, 1996, as supplemented on February 26, 1998, and May 22, 1998.

- Allowance for reactor coolant system hydrostatic testing to be performed with average reactor coolant temperature above 212°F. This change was requested in CP&L letter dated November 1, 1996.

- Approval of exceptions in the primary containment leakage rate testing program. These changes were requested in CP&L letter dated November 1, 1996, and October 13, 1997.

- Modification to the rod block monitor operability requirements. This change was requested in CP&L letter dated November 1, 1996.

- Relaxation in the surveillance frequency for performing a rod worth minimizer channel functional test. This change was requested in CP&L letter dated November 1, 1996.

- Relocation of Main Steam Tunnel High Temperature Isolation Instrumentation. This change was requested in CP&L letter dated November 1, 1996.

Under the subject amendments, an additional condition is added to Appendix B of the BSEP 1&2 facility licenses. Previously Appendix B to the licenses contained Environmental Technical Specifications; however, those specifications were deleted on November 2, 1995, by Amendment 179 for BSEP 1 and Amendment 210 for BSEP 2. The additional condition authorizes relocation of certain requirements from the TS to licensee-controlled documents. The relocated requirements are not required to be in the TS under 10 CFR 50.36 or do not meet any of the four criteria in the "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," published on July 22, 1993 (58 FR 39132). The NRC staff has concluded that appropriate controls have been established for all of the current specifications, information, and requirements that are being moved to licensee-controlled documents. The license condition is adopted herewith, to require incorporation of these matters in the appropriate licensee-controlled document(s). Following implementation, the NRC will audit the removed provisions to ensure that an appropriate level of control has been achieved. Accordingly, these TS, information, and requirements, as described in detail in the enclosed Safety Evaluation, may be relocated from current TS and placed in the Updated Final Safety Analysis Report or other licensee-controlled documents as specified in Carolina Power & Light Company letters dated November 1, 1996, October 13, 1997, February 26, 1998, April 24, 1998, and May 22, 1998.

A second license condition is added to Appendix B for BSEP 2 only to ensure that the End-Of-Cycle Recirculation Pump Trip (EOC-RPT) system will not be used without prior NRC approval. This system is installed only on BSEP 2 and is currently abandoned in place. This license condition is adopted herewith, as specified in CP&L letter dated March 13, 1998, to require that the EOC-RPT system be maintained inoperable (i.e. manually bypassed) during Mode 1 operation, when thermal power is greater than or equal to 30% rated thermal power. Following implementation, the NRC will conduct an audit to ensure that an appropriate level of control in this area has been achieved.

A copy of the related Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

David C. Trimble

David C. Trimble, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-325
and 50-324

Enclosures:
1. Amendment No. 203 to
License No. DPR-71
2. Amendment No. 233 to
License No. DPR-62
3. Notice
4. Safety Evaluation

cc w/enclosures:
See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 203
License No. DPR-71

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated November 1, 1996, as supplemented on October 13, 1997, February 26, 1998, March 13, 1998, April 24, 1998, and May 22, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Final Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 203.

In addition, the license is amended to add paragraph 3 to the Facility Operating License No. DPR-71 as follows.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 203, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

3. License conditions are also added with this amendment to Appendix B, to read as follows:

Amendment Number	Additional Conditions	Implementation Date
203	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated November 1, 1996, October 13, 1997, February 26, 1998, April 24, 1998, and May 22, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

Appendix B to Facility Operating License No. DPR-71, is replaced by this amendment with Attachment 3.

4. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

H E Edison for

Pao-Tsin Kuo, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3 and 5b to the license and page 1 to Appendix B of the License
2. Changes to the Technical Specifications

Date of Issuance: June 5, 1998

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2.

(6) Carolina Power and Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report, dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, July 28, 1993, and February 10, 1994, respectively, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2558 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 203, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

> For Surveillance Requirements (SRs) that are new in Amendment 203 to Final Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 203.

(a) Refined radiation calculations (location specific), and/or
(b) Slightly reducing the qualified life, and/or
(c) Assessing the qualification bases by demonstrating qualification based on actual test and materials threshold data while maintaining the regulatory margin, and/or
(d) Assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function

(5) Human Factors

(a) Classroom Training

Power Uprate Operator Training, including the plant operating parameter changes resulting from power uprate, shall be performed as part of License Operator Retraining (LOR) prior to Unit 1 start-up for Cycle 11 operation.

(b) Simulator Training

Simulator training for power uprate shall be completed prior to Unit 1 start-up for Cycle 11 operation. The simulator training will include the following:

(i) A demonstration of selected transients at the uprated power compared to the non-uprated power, including changes in time to achieve critical points for operator actions.

(ii) The time to meet the conditions to inject boron for a high power ATWS.

(iii) The time to depressurize the reactor on a loss of all high pressure injection (time to achieve conditions requiring emergency depressurization at TAF; non-ATWS).

(c) Simulator Modification

Prior to Unit 1 start-up for Cycle 11, the simulator shall be modified to match the uprated control room, as close as possible, with the exception of the zone coding for the High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System speed indication meters. The HPCI and RCIC speed indication zone coding shall be completed prior to Unit 2 start-up following the implementation of the power uprate license amendment.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 203, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by R. C. DeYoung

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
Appendices A, Technical Specifications, and B - Additional Conditions

Date of Issuance:
September 8, 1976

APPENDIX B

Additional Conditions

Amendment Number	Additional Conditions	Implementation Date
203	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated November 1, 1996, October 13, 1997, February 26, 1998, April 24, 1998, and May 22, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 233
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), dated November 1, 1996, as supplemented on October 13, 1997, February 26, 1998, March 13, 1998, April 24, 1998, and May 22, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Final Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233 . For SRs that existed prior to Amendment 233 , including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 233 .

In addition, the license is amended to add paragraph 3 to the Facility Operating License No. DPR-62 as follows:

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 233 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

3. License conditions are also added with this amendment to Appendix B, to read as follows:

Amendment Number	Additional Conditions	Implementation Date
233	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated November 1, 1996, October 13, 1997, February 26, 1998, April 24, 1998, and May 22, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
233	The End-Of-Cycle Recirculation Pump Trip system instrumentation shall be maintained inoperable (i.e. manually bypassed) during Mode 1, when thermal power is greater than or equal to 30% rated thermal power.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

Implementation of this amendment shall include this condition, as described in the licensee's letter dated March 13, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.

Appendix B to Facility Operating License No. DPR-62, is replaced by this amendment with Attachment 3.

4. License condition 2.C.(5) restricting single recirculation system operation is deleted by this amendment.

5. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

N E Edison for

Pao-Tsin Kuo, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3a and 4b to the license and page 1 to Appendix B of the License
2. Changes to the Technical Specifications

Date of Issuance: June 5, 1998

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 233 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Final Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233 . For SRs that existed prior to Amendment 233 , including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 233 .

(a) The end of the current surveillance period for the surveillance requirements listed below may be extended beyond the time limit specified by Technical Specification 4.0.2a. After May 1, 1982, the plant shall not be operated in Conditions 1, 2, or 3 until the surveillance requirements listed below have been completed. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2a shall apply.

Specification 4.3.1.1; Table 4.3.1-1, items 9 & 10
4.3.1.2
4.3.1.3; Table 3.3.1-2, item 10
4.3.2.1; Table 4.3.2-1, items 1.d & 1.f
4.3.2.3; Table 3.3.2-3, items 1.a.1
4.3.3.2, Table 4.3.3-1, items 4.c & 4.f
4.5.2.a
4.8.1.1.2.d.2
4.8.1.1.2.d.3
4.8.1.1.2.d.6
4.8.1.1.2.d.7

(b) Effective June 30, 1982, the surveillance requirements listed below need not be completed until restart for Cycle 5 or July 15, 1982, whichever occurs first. The unit shall not be operated in Conditions 1, 2 or 3 until the surveillance requirements listed below have been completed. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2 shall apply.

Specification 4.3.3.1, Table 4.3.3-1, Items 5.a and 5.b.

2.C.(2)(c) Effective July 1, 1982, through July 8, 1982, Action statement "a" of Technical Specification 3.8.1.1 shall read as follows:

ACTION: a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within two hours and at least once per 12 hours thereafter; restore at least two offsite circuits and four diesel generators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(a) Refined radiation calculations (location specific), and/or
(b) Slightly reducing the qualified life, and/or
(c) Assessing the qualification bases by demonstrating qualification based on actual test and materials threshold data while maintaining the regulatory margin, and/or
(d) Assessing the impact of the radiation test and/or published threshold data on the material properties and its safety function.

(5) Human Factors

(a) Simulator Modification

Prior to the initial start-up following implementation of the power uprate license amendment, the simulator shall be modified to match the uprated Unit 2 control room, as close as possible.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 233 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

Original signed by:

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachments:
Appendices A, Technical Specifications, and B - Additional Conditions

Date of Issuance: Dec. 27, 1974

APPENDIX B

Additional Conditions

Amendment Number	Additional Conditions	Implementation Date
233	The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated November 1, 1996, October 13, 1997, February 26, 1998, April 24, 1998, and May 22, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.
233	The End-Of-Cycle Recirculation Pump Trip system instrumentation shall be maintained inoperable (i.e. manually bypassed) during Mode 1, when thermal power is greater than or equal to 30% rated thermal power. Implementation of this amendment shall include this condition, as described in the licensee's letter dated March 13, 1998, evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 90 days of the date of this amendment.

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSEE

DOCKET NOS. 50-325 AND 50-324

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 203 to Facility Operating License No. DPR-71 and No. 233 to Facility Operating License No. DPR-62 issued to Carolina Power & Light Company (the licensee), which revised the operating license and Appendices A and B to the operating license for the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP), located in Brunswick County, North Carolina. The amendments are effective as of the date of issuance.

The amendment implements a full conversion of the BSEP Technical Specifications (TS) to a set of TS based upon NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 1, dated April 1995. The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on January 24, 1997 (62 FR 3719). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (63 FR 29039).

For further details with respect to the action see (1) the application for amendment dated November 1, 1996, as supplemented by letters dated August 8, September 11, September 17, October 13, November 6, December 19, 1997, February 26, March 13, April 24, 1998 and May 22, 1998, (2) Amendment No. 203 to Facility Operating License No. DPR-71 and Amendment No. 233 to Facility Operating License No. DPR-62, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Dated at Rockville, Maryland, this 5th day of June 1998.

FOR THE NUCLEAR REGULATORY COMMISSION

David C Trimble

David C. Trimble, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 5, 1998

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE DPR-71

AND AMENDMENT NO. 233 TO FACILITY OPERATING LICENSE DPR-62

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS.1 AND 2

CAROLINA POWER AND LIGHT COMPANY

DOCKET NOS. 50-325 AND 50-324

I. INTRODUCTION

Brunswick Steam Electric Plant, Unit Nos. 1 and 2 (BSEP) has been operating with Technical Specifications (TS) issued with the original operating licenses on September 8, 1976 for Unit 1 and December 27, 1974 for Unit 2, as amended from time to time. By letter dated November 1, 1996, as supplemented by letters dated August 8, September 11, September 17, October 13, November 6, December 19, 1997, February 26, March 13, April 24, 1998 and May 22, 1998, Carolina Power and Light Company (the licensee) proposed to amend Appendix A of Operating License Nos. DPR-71 and DRP-62 to completely revise the BSEP TS. The proposed amendment was based upon NUREG-1433, "Standard Technical Specifications for General Electric Plants, BWR/4," Revision 1, dated April 1995, and upon guidance in the "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (Final Policy Statement), published on July 22, 1993 (58 FR 39132), and 10 CFR 50.36, as amended July 19, 1995 (60 FR 36953). The overall objective of the proposed amendment, consistent with the Final Policy Statement, was to rewrite, reformat, and streamline completely the existing TS for BSEP.

Hereinafter, the proposed TS are referred to as the improved TS (ITS), the existing BSEP TS are referred to as the current TS (CTS), and the TS in NUREG-1433 are referred to as the standard TS (STS). The corresponding TS Bases are ITS Bases, CTS Bases, and STS Bases, respectively.

In addition to basing ITS on STS, the Final Policy Statement, and 10 CFR 50.36, the licensee retained portions of the CTS as a basis for the ITS. Plant-specific issues, including design features, requirements, and operating practices, were discussed with the licensee during a series of conference calls and meetings that concluded on May 18, 1998. Based on these discussions, the licensee proposed matters of a generic nature that were not in the STS. The NRC staff requested that the licensee submit such generic issues as a proposed change to STS through the Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These generic issues were considered for specific applications in the BSEP ITS. Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The Commission's proposed action on the BSEP application for an amendment dated November 1, 1996, was published in the *Federal Register* on January 24, 1997 (62 FR 3719). The Staff's evaluation of the application, including supplements to the licensee's ITS proposal, submitted by letters dated August 8, September 11, September 17, October 13, November 6, December 19, 1997, February 26, March 13, April 24, 1998 and May 22, 1998, that resulted from NRC requests for information and discussions with the licensee during the NRC staff review, is presented in this Safety Evaluation (SE). These plant-specific changes serve to clarify the ITS with respect to the guidance in the Final Policy Statement and STS. Therefore, the changes are within the scope of the action described in the *Federal Register* notice.

During its review, the NRC staff relied on the Final Policy Statement and the STS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff conclusion that BSEP can develop ITS based on STS, as modified by plant-specific changes, and that the use of the ITS is acceptable for continued operation. The NRC staff also acknowledges that, as indicated in the Final Policy Statement, the conversion to STS is a voluntary process. Therefore, it is acceptable that the ITS differs from STS, reflecting the current licensing basis. The NRC staff approves the licensee's changes to the CTS with modifications documented in the revised submittals.

For the reasons stated *infra* in this SE, the NRC staff finds that the TS issued with this license amendment comply with Section 182a of the Atomic Energy Act, 10 CFR 50.36, and the guidance in the Final Policy Statement, and that they are in accord with the common defense and security and provide adequate protection of the health and safety of the public.

II. BACKGROUND

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection

to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968). Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SR); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility Owners Groups and the NRC staff developed improved STS that would establish models of the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and Owners Groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing technical specifications, which gives greater consideration to human factors principles and was used throughout the development of licensee-specific ITS.

In September 1992, the Commission issued NUREG-1433, which was developed using the guidance and criteria contained in the Commission's interim policy statement. STS were established as a model for developing improved TS for General Electric plants in general. STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the Nuclear Steam System Supplier (NSSS) Owners Groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all NSSS designs. As such, the generic Bases presented in NUREG-1433 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the improved STS, and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments, and for complete conversions to improved STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated surveillances should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing Limiting Condition for Operation (LCO) requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The Final Policy Statement criteria are as follows:

Criterion 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Part III of this SE explains the NRC staff conclusion that the conversion of the BSEP CTS to those based on STS, as modified by plant-specific changes, is consistent with the BSEP current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

III. EVALUATION

The NRC staff's ITS review evaluates changes to the CTS that fall into five categories defined by the licensee and includes an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements removed from the CTS and placed in licensee-controlled documents. This evaluation also discusses the NRC staff's plans for monitoring the licensee's implementation of these controls at BSEP.

In addition to the initial submittal of November 1, 1996, as supplemented, the NRC staff review identified the need for clarifications and additions to the submittal in order to establish an appropriate regulatory basis for translation of current TS requirements into ITS. Each change proposed in the amendment request is identified as either a discussion of change (DOC) to CTS or a justification for deviation from STS. The NRC staff comments were documented as requests for additional information (RAIs) and forwarded to the licensee for response by letters dated July 2, August 8, October 1, November 7, November 17, and November 19, 1997. The licensee provided written responses to the NRC staff requests in letters dated August 8, September 11, November 6, and December 19, 1997. The docketed letters clarified and revised the licensee basis for translating CTS requirements into ITS. The NRC staff finds that the licensee's submittals provide sufficient detail to allow the staff to reach a conclusion regarding the adequacy of the licensee's proposed changes.

The license amendment application was organized such that changes were included in each of the following CTS change categories, as appropriate: administrative changes, technical changes - less restrictive (specific), technical changes - less restrictive (generic), technical changes - more restrictive, and relocated specifications.

(1) Administrative Changes, (A), i.e., non-technical changes in the presentation of existing requirements;

(2) Technical Changes - More Restrictive, (M), i.e., new or additional CTS requirements;

(3) Technical Changes - Less Restrictive (specific), (L), i.e., changes, deletions and relaxations of existing TS requirements;

(4) Technical Changes - Less Restrictive (generic), (LA), i.e., deletion of existing TS requirements by movement of information and requirements from existing specifications (that are otherwise being retained) to licensee-controlled documents, including TS Bases, and

(5) Relocated Specifications, (R1), i.e., relaxations in which whole specifications (the LCO and associated actions and SRs) are removed from the existing TS (an NRC-controlled document) and placed in licensee-controlled documents.

These general categories of changes to the licensee's current TS requirements and STS differences may be better understood as follows:

A. Administrative Changes

Administrative (non-technical) changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily; making the TS more easily understood through editorial changes, clarifications of TS requirements, and format changes, without changing the technical content. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used STS as guidance to reformat and make other administrative changes. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

(1) providing the appropriate numbers, etc., for STS bracketed information (information that must be supplied on a plant-specific basis and that may change from plant to plant)

(2) identifying plant-specific wording for system names, etc.

(3) changing the wording of specification titles in STS to conform to existing plant practices

(4) splitting up requirements currently grouped under a single current specification to more appropriate locations in two or more specifications of ITS

(5) combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS.

(6) presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TS) that do not involve a change in requirements;

(7) wording changes and additions that are consistent with current interpretation and practice, and that more clearly or explicitly state existing requirements; and

(8) deletion of redundancies that are unnecessary since the requirements exist elsewhere in the TS.

Table A lists the administrative changes proposed in ITS. Table A is organized by the corresponding ITS section discussior of change, and provides a summary description of the administrative change that was made, and CTS and ITS LCO references. The NRC staff reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable, because they are compatible with the Writer's Guide and STS, do not result in any substantive change in operating requirements and are consistent with the Commission's regulations.

B. Technical Changes - More Restrictive

The licensee, in electing to implement the specifications of STS proposed a number of requirements more restrictive than those in the CTS. ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or that have additional restrictions that are not in the CTS but are in STS. Examples of more restrictive requirements are placing an LCO on plant equipment which is not required by the CTS to be operable, more restrictive requirements to restore inoperable equipment, and more restrictive SRs. Table M lists all the more restrictive changes proposed in ITS. Table M is organized by the corresponding ITS section discussion of change and provides a summary description of the more restrictive change that was adopted, and CTS and ITS LCO references. These changes are additional restrictions on plant operation that enhance safety and are acceptable.

C. Technical Changes - Less Restrictive (Specific)

Less restrictive requirements include changes, deletions, and relaxations to portions of current TS requirements that are not being retained in ITS. When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The BSEP design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in STS, and thus provide a basis for ITS.

A significant number of changes to the CTS involved changes, deletions and relaxations to portions of current TS requirements evaluated as Categories I through XII that follow:

Category I - Relaxation of Applicability

Category II - Relaxation of Surveillance Frequency

Category III - Revision of Allowable Values Based on Instrument Setpoint Methodology

Category IV - Deletion of Post Maintenance Testing Requirements

Category V - Relaxation of Action Requirements for Exiting LCOs

Category VI - Relaxation of Surveillance Requirement Acceptance Criteria

Category VII - Relaxation of Allowed Outage Time

Category VIII - Deletion of Requirement for 30-day Special Report to NRC

Category IX - Relaxation of LCO

Category X - Deletion of Mode Restrictions on Surveillance Performance

Category XI - Deletion of Requirement to Lock the Mode Switch in Position

Category XII - Relaxation of Action Requirements to Reflect Safety Function

The following discussions address why various technical specifications within each of the twelve categories of information or specific requirements are not required to be included in ITS.

<u>Relaxation of Applicability</u> *(Category I)*

Reactor operating conditions are used in CTS to define when the LCO features are required to be operable. CTS applicabilities can be specific defined terms of reactor conditions: refueling, hot shutdown, cold shutdown, startup, or power operating condition. Applicabilities can also be more general. In this type of change, CTS requirements may be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS are satisfied by exiting the applicability, thus, taking the plant out of the conditions that require the safety system to be Operable. These changes are consistent with STS and changes specified as Category I are acceptable.

<u>Relaxation of Surveillance Frequency</u> *(Category II)*

CTS and ITS surveillance frequencies specify time interval requirements for performing surveillance requirement testing. Increasing the time interval between surveillance tests in the ITS results in decreased equipment unavailability due to test which also increases equipment availability. In general, the STS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the STS is acceptable based on similar design, like-component testing for the system application and the availability of other TS requirements which provide regular checks to ensure limits are met.

Reduced testing can result in a safety enhancement because the unavailability due to test is reduced; in turn, reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance frequency changes to incorporate alternate division testing have been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance, e.g., a 31 day air flow test is an indicator that positive pressure in a controlled space will be maintained because this test would use the same fans as the less frequent ITS 48 month pressurization test and industry experience shows that components usually pass the pressurization test.

Additionally, surveillance frequency extensions can be based on staff-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are consistent with STS and changes specified as Category II are acceptable.

Revision of Allowable Values Based on Instrument Setpoint Methodology *(Category III)*

Allowable Values for the trip settings of instrumentation establish operability limits for that instrumentation. These Allowable Values have been established consistent with the methods described in CP&L's Instrument Setpoint Methodology (Design Guide DG-VIII.0050 "Instrument Setpoints" Rev. 5). The Allowable Value determinations were done using vendor documented performance specifications, where available and applicable. Where vendor documented performance specifications for drift were not available or applicable, the Allowable Value was determined using plant specific operating and surveillance trend data or an allowance as provided for by the CP&L's instrument Setpoint Methodology. The Allowable Value verification used actual operating and surveillance trend information to ensure the validity of the developed Allowable Value. All changes to safety analysis limits applied in the methodologies were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits applied in the methodologies were confirmed as ensuring that applicable design requirements of the associated systems and equipment are maintained. Plant calibration procedures ensure that the assumptions regarding calibration accuracy, measurement and test equipment accuracy, and setting tolerance are maintained. Setpoints for each design or safety analysis limit have been established by accounting for the applicable instrument accuracy, calibration and drift uncertainties, environmental effects, power supply fluctuations, as well as uncertainties related to process and primary element measurement accuracy using the CP&L Instrument Setpoint Methodology. The Allowable Values have been established from each design or safety analysis limit by combining the errors associated with channel/instrument calibration (e.g., device accuracy, setting tolerance, and drift) with the calculated Nominal Trip Setpoint also using the CP&L Instrument Setpoint Methodology. Additionally, each channel/instrument has been evaluated and analyzed to support a fuel cycle extension to a 24 month interval. As a result, the revised Allowable Values ensure that the design basis and associated safety limits will not be exceeded during plant operation. The application of instrument setpoint methodology for the development of Allowable Values is consistent with the STS and changes specified as Category III are acceptable.

Deletion of Post Maintenance Testing Requirements *(Category IV)*

Any time the Operability of a system or component has been affected by repair, maintenance, or replacement of a component, post maintenance testing is required to demonstrate Operability of the system or component. After restoration of a component that caused a required surveillance requirement not to be met, ITS SR 3.0.1 requires the appropriate surveillance requirements to be performed to demonstrate Operability of the affected components. Therefore, explicit post maintenance surveillance requirements are not included in the ITS. These changes are consistent with the STS and changes specified as Category IV are acceptable.

Relaxation of Action Requirements for Exiting LCOs *(Category V)*

CTS require that in the event specified LCOs are not met, power or Mode reductions shall be initiated as the method to reestablish the appropriate limits. The ITS are constructed to specify actions for conditions of required features made inoperable. Adopting ITS action requirements for exiting LCO applicabilities is acceptable because the plant remains within analyzed parameters by performance of required actions, or the actions are constructed to minimize risks associated with continued operation while providing time to repair inoperable features. Such actions add margin to safety, thereby providing assurance that the plant is configured appropriately or operations that could result in a challenge to safety systems are exited in a time period that is commensurate with the safety importance of the system. Additionally, other changes to TS actions include placing the reactor in a Mode where the specification no longer applies, usually resulting in an extension to the time period for taking the plant into shutdown conditions. These actions are commensurate with industry standards for reductions in thermal power in an orderly fashion without compromising safe operation of the plant. These changes are consistent with STS and changes specified as Category V are acceptable.

Relaxation of Surveillance Requirement Acceptance Criteria *(Category VI)*

CTS require safety systems to be tested and verified Operable prior to entering applicable conditions. ITS provide the additional requirement to verify Operability by actual or test conditions. Adopting the STS allowance for "actual" conditions is acceptable because TS required features cannot distinguish between an "actual" signal or a "test" signal. Category VI also includes changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which when combined include Operability verification of all TS required components for the features specified in the CTS. Adopting this format preference in the STS is acceptable because TS SRs that remain include testing of all previous features required to be verified operable. These changes are consistent with STS and changes specified as Category VI are acceptable.

Relaxation of Allowed Outage Time *(Category VII)*

Upon discovery of a failure to meet an LCO, STS specify times for completing required actions of the associated TS conditions. Required actions of the associated conditions are used to establish remedial measures that must be taken within specified completion times (allowed outage times). These times define limits during which operation in a degraded condition is permitted.

Adopting completion times from the STS is acceptable because completion times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a design basis accident (DBA) occurring during the repair period. These changes are consistent with STS and allowed outage time extensions specified as Category VII are acceptable.

Deletion of Requirement for 30-Day Special Report to NRC *(Category VIII)*

CTS include requirements to submit Special Reports when specified limits are not met. Typically, the time period for the report to be issued is within 30 days. However, the STS eliminates the TS administrative control requirements for Special Reports and instead relies on the reporting requirements of 10 CFR 50.73. ITS changes to reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation. Therefore, this change has no impact on the safe operation of the plant. Additionally, deletion of TS reporting requirements reduces the administrative burden on the plant and allows efforts to be concentrated on restoring TS required limits. These are consistent with STS and changes specified as Category VIII are acceptable.

Relaxation of LCO *(Category IX)*

CTS provides lists of acceptable devices that may be used to satisfy LCO requirements. The ITS reflect the STS approach to provide LCO requirements that specify the protective limit that is required to meet safety analysis assumptions for required features. The protective limits replace the lists of specific devices previously found to be acceptable to the NRC staff for meeting the LCO. The ITS changes provide the same degree of protection required by the safety analysis and provide flexibility for meeting limits without adversely affecting operations since equivalent features are required to be operable. These changes are consistent with STS and changes specified as Category IX are acceptable.

Deletion of Mode Restrictions on Surveillance Performance *(Category X)*

Requirements that the certain surveillance testing be performed "during shutdown" are removed from TS in accordance with the guidance of Generic Letter 91-04. These requirements are prerequisites for performance of the surveillances and are not necessary for ensuring the requirements of the affected surveillance requirements are satisfied. These surveillances may be able to be performed in other than shutdown conditions without jeopardizing safe plant operations. The control of plant conditions appropriate to perform tests is an issue for procedures and scheduling and has been determined by the NRC Staff to be unnecessary as TS restrictions. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other TS surveillances that do not dictate plant conditions for performance of the surveillances. These changes are consistent with the STS and changes specified as Category X are acceptable.

Deletion of Requirement to Lock the Mode Switch in Position *(Category XI)*

Requirements to "lock" the reactor mode switch in a certain position (e.g., shutdown or refuel) are not included in the ITS. The required position of the reactor mode switch is adequately controlled by the Modes definition Table (ITS Table 1.1-1) or the special operations LCOs. Reactor mode switch positions other than those specified in ITS Table 1.1-1, or the special operations LCOs, result in the unit entering some other Mode or condition; with the associated Technical Specification compliance requirements of that Mode or condition and ITS LCO 3.0.4. In general, ITS LCO 3.0.4 restricts Mode or condition entry if LCO requirements of the Mode or condition are not met. Therefore, movement of the reactor mode switch from the required position is adequately controlled by the requirements of the ITS without the need to "lock" the reactor mode switch in position. These changes are consistent with the STS and changes specified as Category XI are acceptable.

Relaxation of Action Requirements to Reflect Safety Function *(Category XII)*

In the event actuation instrumentation channels are inoperable, the CTS actions typically require the channels to be tripped or the unit to be placed in a Mode or condition outside the Applicability of the LCO. For certain actuation instrumentation channels, in lieu of tripping the channels or requiring the unit to be placed in a Mode or condition outside the Applicability, these CTS actions are changed to allow the component(s) actuated by the instrumentation to be placed in a condition that satisfies the safety function of the instrumentation. For example, in event of inoperable instrumentation that transfers the high pressure coolant injection (HPCI) pump suction from the condensate storage tank to the suppression pool, manually aligning the HPCI pump suction to the suppression pool, in lieu of tripping the instrument channel, results in the same conditions as if a channel associated with the inoperable instrumentation were tripped (tripping one channel results in the HPCI pump suction being aligned to the suppression pool). Since manually placing the actuated components in a configuration where the safety function is satisfied accomplishes the same actions as the actuation instrumentation, the added actions are functionally equivalent to the actuation instrumentation. In addition, for the actuation instrumentation for which this change is provided, the ITS allows continued operation with the actuated components in this configuration, i.e., operation with the actuated component placed in a condition that satisfies the safety functions is an approved configuration. These changes are consistent with the STS and changes specified as Category XII are acceptable.

Table L lists all the less restrictive changes proposed in the ITS. Table L is organized by the corresponding ITS specification discussion of change and provides a summary description of the less restrictive change that was adopted, CTS and ITS reference, and category of change. Additionally, in electing to implement the specifications of STS, the licensee also proposed a number of less restrictive changes to the CTS which do not apply to the above Categories of changes, deletions and relaxations of CTS requirements. These changes are discussed below. The associated discussion of change identifier (e.g., L1) is provided for these unique less restrictive changes.

Section 1.0 - Less Restrictive

L1 The ITS Channel Functional Test definition combines analog and bistable channel requirements which results in an allowance for injecting the bistable channel test signal "as close to the sensor as practicable" in lieu of "into the sensor." The ITS Logic System Functional Test definition allows injecting the signal "as close to the sensor as practicable" in lieu of "from sensor output." Injecting a signal at the sensor involves significant increased probabilities of initiating undesired circuits during the test since several logic channels are often associated with a particular sensor. Performing the test by injecting a signal at the sensor requires jumpering of the other logic channels to prevent their initiation during the test, or increases the scope of the test to include multiple tests of the other logic channels. Either method increases the difficulty of performing the surveillance. Allowing initiating the signal close to the sensor provides a complete test of the logic channel while reducing the probability of undesired initiation.

L2 The ITS Core Alteration definition deletes CTS reference to "or other components affecting reactivity." The change maintains Core Alterations as movement activities affecting core reactivity. The basis for deleting the CTS words is evident in that the CTS requirements applicable during Core Alterations are those that prevent or mitigate a reactivity event. The CTS and ITS provide that movement of any fuel, sources, or reactivity control components is a Core Alteration. Since only these components can have a more than negligible affect on core reactivity, it is not necessary to include the additional wording "or other components affecting reactivity" in the Core Alteration definition. This change is consistent with the CTS definition of Core Alteration which excludes movement of source range monitors (SRMs), intermediate range monitors (IRMs), local power range monitors (LPRMs), traversing incore probe (TIPs), and a special movable detectors (i.e., incore instruments) as a Core Alteration since incore instruments have a negligible (if any) affect on core reactivity. Therefore, the ITS definition deletes the wording "or other components affecting core reactivity" since these components are addressed by the remaining requirements of the definition.

L3 The ITS changes the definition of Core Alteration and no longer considers control rod movement with the normal control rod drive in a defueled core cell a Core Alteration. The negative reactivity inserted by removing the adjacent four fuel assemblies is significantly more than any minimal positive reactivity inserted during the movement of the control rod. Appropriate ITS controls are applied during the fuel movements (i.e., Core Alterations) preceding the control rod movement to protect from or mitigate a reactivity excursion event. After the fuel is removed, sufficient margin and administrative controls allow removing the Core Alterations CTS controls during the control rod movement. This change focuses the Core Alteration definition on activities that affect the core reactivity. Maintaining Core Alterations definition as movement reactivity components which can affect core reactivity is consistent with applying the CTS requirements. Core Alterations Applicabilities involve ITS requirements that prevent or mitigate a reactivity excursion event.

Section 2.0 - Less Restrictive

L1 The CTS 6.7.1 requirement for the Safety Limit Violation report submittal is "within 14 days of the violation." The ITS does not specify a Safety Limit Violation report submittal, thus the reporting requirements for a Safety Limit Violation are governed by the reporting requirements for a licensee event report (LER). This change increases the report submittal time to "within 30 days of discovery of the violation, " and is consistent with the requirements of 10 CFR 50.73 and 10 CFR 50.36(c)(1)(I)(A). The Safety Limit Violation Report is an after-the-fact report that remains a report requirement submitted to the NRC as a LER. In accordance with 10 CFR 50.36, resumed operation of the reactor is not allowed until authorization is received from the NRC; therefore, the time frame for report submittal is not necessary to assure safe reactor operation. This is a less restrictive change, consistent with the STS.

L.2 The Action of CTS 2.1.4 is made less specific to allow operator flexibility in determining the best method to restore the reactor vessel water level. Directions for the methods of restoring reactor vessel water level (manually initiate the low pressure emergency core cooling system (ECCS) to restore the water level, after depressurizing the reactor vessel, if required) are removed from the TS. This detail of how to restore the reactor vessel water level is not necessary to ensure restoration of the reactor vessel water level in a timely manner. The action to restore compliance with the Safety Limit is maintained in ITS SL 2.2.1, which provides a 2 hour Completion Time for restoration of the limit. The time frame for completion of the action is consistent with the allowed time to restore other Safety Limit violations and allows appropriate actions to be evaluated by the operator and completed in a timely manner. In addition, restoration of reactor vessel water level is part of a coordinated response to an unplanned transient governed by Emergency Operating Procedures.

Specification 3.1.1 - Less Restrictive

L3 ITS 3.1.1 Required Action E.2 only requires action to be initiated to fully insert control rods in core cells containing one or more fuel assemblies. If all fuel assemblies are removed from a core cell, inserting the associated control rod has a negligible impact on core reactivity. During Mode 5, refueling procedures could have cells emptied and the control rod withdrawn, but "insertable." However, due to a variety of considerations (i.e., location of blade guides, ongoing instrumentation maintenance, water chemistry), insertion of these control rods as required by CTS 3.1.1 Action c may not be desirable. Since there is negligible impact on Shutdown Margin (SDM) should the control rod be inserted with no fuel in the cell, it is acceptable to allow the control rod to remain withdrawn.

L6 The CTS 3.1.1 Action c requirement to suspend all Core Alterations precludes off-loading fuel and inserting control rods. ITS 3.1.1 Required Action E.1 modifies the requirement to suspend Core Alterations "except for control rod insertion and fuel assembly removal." This exception allows continuation of activities that may have caused the loss of adequate SDM and may have a potential to restore the SDM. This additional operational flexibility does not require new or different actions, but allows corrective actions which would have otherwise been precluded (except under the provisions of 10 CFR 50.54(x)). Additionally, the corrective actions can only be pursued in accordance with approved procedures.

Specification 3.1.3 - Less Restrictive

L2 If an inoperable stuck control rod is not separated from all other inoperable control rods by at least two control cells in all directions, a plant shutdown is required after 1 hour in accordance with CTS 3.1.3.1 Action a.2 and CTS 3.0.3. ITS 3.1.3 Required Actions D.1 and D.2 are added to provide compensatory measures (in lieu of a plant shutdown) which ensure that the highest incremental control rod worth is maintained low enough to prevent exceeding a peak fuel enthalpy of 280cal/gm during a control rod drop accident (CRDA) by the distribution of the inoperable control rods. Three distinct changes are addressed by the addition of these actions:

(1) ITS 3.1.3 Condition D is modified by a Note excluding its applicability above 10% power. The existing separation requirements for a stuck control rod in CTS 3.1.3.1 Action a.1.a, in part, account for allowing withdrawn inoperable control rods in accordance with CTS 3.1.3.1 Action b. To preserve scram reactivity, ITS 3.1.3 Required Action A.1 requires a stuck control rod be separated from "slow" control rods to ensure the reactor can be made subcritical during a postulated accident or transient. In accordance with ITS 3.1.3, all inoperable control rods except one stuck control rod are required to be fully inserted and disarmed. As a result, scram reactivity is preserved at power levels > 10% RTP and is unaffected by this change.

ITS 3.1.3 Condition D is applicable ≤ 10% RTP since separation requirements are necessary for inoperable control rods (even if inserted) due to CRDA assumptions related to the banked position withdrawal sequence (BPWS) and control rod worth at low power described in NEDO-21231, "Banked Position Withdrawal Sequence."

(2) ITS 3.1.3 Required Action D.1 is satisfied if the inoperable control rods are in conformance with BPWS constraints. Appropriate core reactivity and power distribution limits below 10% power are controlled by maintaining control rod positions within the limits of BPWS and maintaining scram times within limits. If two inoperable control rods are both stuck, ITS 3.1.3 Required Action B.1 requires an immediate plant shutdown, regardless of their proximity to each other. Therefore, since only one inoperable control rod is allowed to remain withdrawn, the separation limitation imposed on inoperable control rods that comply with the BPWS is adequate to ensure the reactor can be made subcritical in the event of a CRDA.

(3) Finally, ITS 3.1.3 Required Actions D.1 and D.2 allow 4 hours to correct non-compliance with the BPWS or to restore the inoperable control rods to Operable status prior to commencing a plant shutdown per ITS 3.1.3 Required Action E.1 (CTS 3.1.3.1 Action a.1 and b.1 allow 1 hour). This increased Completion Time recognizes the operational steps involved upon discovery of inoperable control rod(s). Time is required to attempt identification and correction of the inoperability and additional time is necessary to fully insert (some operational considerations may be necessary to adjust control rod patterns and/or power levels), and then disarm the affected control rod(s). After these high priority steps are accomplished, attention must be given to correcting localized distribution of inoperable control rods that deviate from the BPWS. This change is acceptable given the low probability of a CRDA during this brief period and the desire not to impose excessive time constraints on operator actions that could lead to hasty corrective actions. As such, the Completion Time extension from 1 hour to 4 hours does not represent a significant safety concern.

L4 CTS 3.1.3.1 Action a.3, which requires restoring a stuck control rod to Operable status within 48 hours, is not included in ITS 3.1.3. Instead, ITS 3.1.3 allows continued operation with a stuck control rod. With a single withdrawn control rod stuck, the remaining Operable control rods are capable of providing the required scram and shutdown reactivity. During a transient, a single stuck control rod in addition to an assumed single failure has no significant impact on the established operating limits. This change is acceptable since SDM must still be met in accordance with ITS 3.1.1, accounting for the loss of negative reactivity due to the stuck control rod (refer to the ITS definition of SDM and ITS 3.1.3 Required Action A.4). Additionally, ITS 3.1.3 Required Action A.1 verifies separation criteria is met between the stuck control rod and all "slow" control rods. Also, prompt action is required to confirm no additional stuck control rods exist by the performance of ITS SR 3.1.3.2 and SR 3.1.3.3 (ITS 3.1.3 Required Action A.3). As a result, continued operation is allowed provided the Required Actions of ITS 3.1.3 Condition A are completed in the specified time period.

L6 CTS 4.1.3.1.2.b requires the 7 day control rod notch surveillance be performed daily in the event power operation continues concurrent with an immovable control rod and the plant is operating greater than the low power setpoint (LPSP) of the RWM. In this condition, ITS 3.1.3 Required Action A.3 requires the control rod notch test to be performed only once within 24 hours when operating greater than the LPSP of the RWM. The purpose of the control rod notch test on each withdrawn Operable control rod when a control rod is stuck is to ensure that a generic problem does not exist and that control rod insertion capability is maintained. This change is acceptable since a single performance of the control rod notch test is adequate to verify that a generic problem has not occurred with the remaining withdrawn control rods and the normal periodic surveillance is adequate to ensure control rod insertion capability is maintained.

L9 In addition to requiring that two or more inoperable control rods be separated by two control cells in all directions (CTS 3.1.3.1 Required Action a.1.a and b.1.a), CTS 3.1.4.1 Action d.1 prohibits continued operation if more than 3 rods in any one BPWS group are inoperable when the RWM is required to be Operable. ITS 3.1.3 Condition D is only required to be entered and associated actions taken if two or more inoperable control rods are not in compliance with the BPWS and are not separated by two or more Operable control rods when THERMAL POWER is ≤ 10% RTP. The elimination of the requirement to have no more than 3 inoperable control rods in any one BPWS group is considered to be acceptable based on the analyses in NEDO-21231, "Banked Position Withdrawal Sequence." Section 7 of NEDO-21231 describes the effects that operation with inoperable control rods have on a control rod drop accident (CRDA). Two cases were analyzed to determine the peak fuel enthalpy for various patterns of inoperable control rods. These cases were analyzed using GE fuel (BSEP uses GE fuel exclusively).

The first case (referred to here as Case 1) analyzed the effects on control rod worth using a control rod geometry with 6 control rods bypassed and fully inserted in the same BPWS group. As noted in NEDO-21231, this pattern violates the maximum of 3 inoperable control rods in a single control rod group criterion. However, Case 1 maintained the separation criterion required by ITS 3.1.3. Case 1 examined BPWS Rod Group 7 (Sequence A) since this group was determined to have the highest incremental control rod worth. In addition, only 6 bypassed and fully inserted inoperable control rods were analyzed in Case 1 because more than 6 control rods in this condition would have resulted in not meeting the separation criterion. The results of Case 1 showed that peak fuel enthalpy would reach approximately 162 cal/gm (NEDO-21231, page 7-2).

Another case (referred to here as Case 2) analyzed the effects on control rod worth using a control rod geometry with 8 inoperable control rods bypassed and fully inserted (maximum number of inoperable control rods allowed by TS) in the lower right portion of the core. As noted in NEDO-21231, this pattern maintained separation criteria and the maximum of 3 inoperable control rods in a single BPWS group criterion. Case 2 examined this control rod worth geometry (all inoperable rods in the same region of the core) to show the effects the reactivity shift would have on the highest incremental control rod worth. The results of Case 2 showed that peak fuel enthalpy would reach approximately 232 cal/gm (NEDO-21231, page 7-3).

As stated in NEDO-21231, both cases showed that the peak fuel enthalpy is "... well below the 280 cal/gm design basis." As a result, these analyses, which bound all other cases, indicate that CRDA design basis assumptions are maintained if more than 3 control rods are inoperable in the same BPWS group provided there are ≤ 8 inoperable control rods and that each inoperable control rod is separated by at least two Operable control rods. Therefore, this change, which deletes the restriction on continued operation if more than 3 control rods in any one BPWS group are inoperable, is acceptable since the inoperable control rod criteria are maintained in the TS; ≤ 8 inoperable control rods (ITS 3.1.3 Condition E) and each inoperable control rod is required to be separated by at least two Operable control rods (ITS 3.1.3 Condition D). These criteria have been demonstrated to be adequate for ensuring CRDA design basis assumptions are maintained.

L11 The requirement for additional scram time surveillance testing (CTS 3.1.3.2 Action d), when three or more control rods exceed the maximum scram time, is deleted. During normal power operating conditions, scram testing is a significant perturbation to steady state operation, involving significant power reductions, abnormal control rod patterns and abnormal control rod drive hydraulic system configurations. Therefore, requiring more frequent scram time surveillance testing is not desirable. The 7 day and 31 day periodic Frequencies of ITS SR 3.1.3.2 and SR 3.1.3.3, respectively (control rod notch tests), and the 7 day periodic Frequency of ITS SR 3.1.5.1 (accumulator Operability test) are considered adequate to maintain control rod Operability. In addition, the more frequent scram time testing is not necessary to assure safe plant operations.

L12 SRs associated with the control rod position indication system (CTS 4.1.3.7.b and 4.1.3.7.c) require that the control rod position indication system be determined Operable by the performance of the control rod movement test and the control rod coupling verification. To perform control rod movement tests required by CTS 4.1.3.1.2 (ITS SRs 3.1.3.2 and 3.1.3.3) and control rod coupling verifications required by CTS 4.1.3.6.b (ITS SR 3.1.3.5), position indication must be available. If position indication is not available, these tests cannot be satisfied and appropriate actions are taken for inoperable control rods in accordance with ITS 3.1.3 Condition C. As a result, the requirements for the control rod position indication system are adequately addressed by the requirements of ITS SR 3.1.3.2, SR 3.1.3.3, and SR 3.1.3.5; and CTS 4.1.3.7.b and 4.1.3.7.c are deleted.

Specification 3.1.8 - Less Restrictive

L2 CTS 4.1.3.1.3.b requires verifying that the scram discharge volume (SDV) vent and drain valves open when the scram signal is reset or open when the SDV trip is bypassed. The Applicability associated with CTS 4.1.3.1.3.b is Operational Conditions 1 and 2. ITS SR 3.1.8.3 includes the verification that the SDV vent and drain valves open when the scram signal is reset. The Applicability associated with ITS SR 3.1.8.3 is Modes 1 and 2. The CTS requirement to verify the SDV vent and drain valves open when the scram discharge volume trip is bypassed is not included in the BSEP ITS. This bypass allows the operator to reset the Reactor Protection System, so that the system is restored to operation while the operator drains the scram discharge volume. However, in Modes 1 and 2, when the SDV vent and drain valves are required to be Operable, TS do not allow the scram discharge volume trip to be bypassed. As such, this feature cannot be used in Modes 1 and 2 and is not required to ensure the SDV vent and drain valves are capable of fulfilling their required functions in Modes 1 and 2 (closing during the scram to limit the amount of reactor coolant discharged and opening on scram reset for draining to maintain sufficient volume to allow a complete scram). The requirements of ITS 3.1.8 are adequate for maintaining SDV vent and drain valves Operable (ITS SR 3.1.8.3 verifies that the SDV vent and drain valves close on a scram signal and open when the scram signal is reset. In Modes 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate control to ensure control rods can not be withdrawn. In Mode 5, only a single control rod can be withdrawn from core cells containing fuel assemblies. Therefore, the SDV vent and drain valves (and associated opening features) are not required to be Operable in Modes 3, 4, and 5 since the reactor is subcritical and no more than one control rod may be withdrawn and subject to scram.

Specification 3.3.1.1 - Less Restrictive

L10 CTS Table 4.3.1-1 requires an LPRM channel check to be performed daily. This requirement is deleted from CTS Table 4.3.1-1. The LPRMs do not directly initiate a reactor scram, rather, the LPRMs input into the average power range monitor (APRM) channels. The APRM channel check (required by ITS SR 3.3.1.1.1) performed every 12 hours is adequate to detect gross failure of LPRM detectors. Also, if < 11 LPRMs are available to an APRM channel, the channel automatically becomes inoperable and an alarm is annunciated in the control room. Additionally, individual LPRM channel checks may not detect a gross error in an LPRM instrument since local power varies core-wide. As such, a specific channel check SR to establish Operability of LPRM channels is not necessary because the instrumentation SRs that remain are sufficient to establish Operability of the reactor protection system.

Specification 3.3.1.1 - Less Restrictive

L13 The requirement to verify proper overlap between IRMs and APRMs during each startup (CTS Table 4.3.1-1 Note (c)) is deleted in ITS 3.3.1.1. In ITS, the IRM and APRM high flux scrams are required to be operable prior to placing the reactor mode switch in the run position. If the IRM overlap with the APRMs is inadequate at the time the reactor mode switch is placed in the run position, then a reactor scram will be initiated. The reactor mode switch also bypasses the IRM high flux scram when placed in the run position provided the APRMs are not downscale. The APRM downscale function is required to be operable in Mode 1 to ensure that adequate neutron monitoring system protection is available if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. In this condition, (mode switch in run and APRMs downscale), the IRM high flux instrumentation will initiate a reactor trip when the setpoint is exceeded. As such, verifying proper overlap between the IRMs and the APRMs during startup is not necessary because a scram is initiated by the neutron monitoring system functions in TS to prevent excessive power increases if adequate overlap is not maintained. In addition, the ITS SRs (which includes an instrument channel check requirement) associated with the IRM and APRM instruments are adequate to maintain Operability of these instruments and ensure the instruments will perform their associated safety function.

Specification 3.3.1.2 - Less Restrictive

L5 CTS 3.9.2.e and CTS 4.9.2.b require verifying SRM count rate is at least 3 cps prior to a core spiral reload. A specific performance of ITS SR 3.3.1.2.4 prior to a core spiral reload is not required. ITS SR 3.3.1.2.4 Note 1 allows SRM count rate to be below 3 cps with less than or equal to four fuel assemblies adjacent to the SRM, regardless of the type of core reload that will be conducted, provided no other fuel assemblies are located in the associated core quadrant. This change is acceptable since in this condition, even with a control rod withdrawn, the configuration will not be critical.

L6 CTS 3.9.2 Actions for refueling operations include suspending operations involving positive reactivity changes when the minimum required SRMs are inoperable. This action is deleted. Positive reactivity changes are not defined in the CTS. In the ITS, positive reactivity changes are captured by the definition of Core Alteration. This change is acceptable because the ITS provides actions to suspend Core Alterations except for rod insertion and to initiate actions to insert control rods in core cells containing fuel assemblies. These requirements and the requirements of ITS LCO 3.1.1, Shutdown Margin, are adequate to ensure the core is maintained subcritical.

L9 In Mode 5, the CTS 3.9.2 Action requires fully inserting all insertable control rods if one or more required SRM is inoperable. In this condition, ITS 3.3.1.2 only requires inserting all insertable control rods in core cell containing one or more fuel assemblies (ITS 3.3.1.2 Required Action E.2). This is acceptable because control rods withdrawn from or inserted into a core cell containing no fuel assemblies have a negligible impact on core reactivity and therefore are not required to be inserted to maintain the reactor subcritical.

L10 CTS 4.9.2.a.5 for refueling operations source range monitoring instrumentation requires verification that the fuel movement sheet is being followed during a core spiral unload and reload. This requirement is not required to verify the Operability of the SRM instrumentation and is deleted in ITS. ITS specify instead that shutdown margin (LCO 3.1.1) requirements are to be maintained and two source range monitors are required to be operable in Mode 5. As such, during spiral unload and reload sequences (which can only be performed in Mode 5), SRMs are required to be operable and SDM is maintained and ITS SR 3.1.1.1 (which requires verification that SDM limits are met) is satisfied. Therefore, the explicit requirement to periodically verify that the fuel movement sheet is being followed during a core spiral unload and reload is considered to be unnecessary for ensuring compliance with the applicable TS requirements. The Surveillance Requirements (SRs) of ITS 3.3.1.2 are considered adequate to maintain Operability of the SRMs.

Specification 3.3.2.1 - Less Restrictive

L3 CTS 3.1.4.1 requires control rod movement to be verified in compliance with the BPWS by a second licensed operator or a qualified member of the technical staff when the RWM is inoperable during a reactor startup. For this condition, ITS 3.3.2.1 Required Action C.1 provides an alternative action; to suspend control rod movement except by scram. This alternate action is acceptable since it is a more conservative action than compliance verification in that the reactor startup is suspended which minimizes the possibility of control rod mispositioning which is the design function of the RWM. The only way control rods may be moved, using this action, is a reactor scram which fully inserts all control rods, thereby preventing control rod mispositioning.

Specification 3.3.5.1 - Less Restrictive

L2 CTS Table 3.3.3-1 Action 31 requires the associated low pressure coolant injection (LPCI) subsystem to be declared inoperable if one or more channels of the LPCI reactor vessel shroud level function (CTS 3.3.3-1 Function 2.c) is inoperable. For the same condition, ITS 3.3.5.1 Required Action B.3 allows the inoperable channel(s) to be placed in a tripped condition within 24 hours. Tripping the affected channels conservatively compensates for the inoperable status by restoring the single failure capability of the function and by accomplishing the safety function of the channel (preventing LPCI flow from being diverted from the reactor vessel when the reactor vessel water level is not at least 2/3 core height). Allowing continued operation for 24 hours with an untripped channel is acceptable since the additional time allowed to continue operation with an ECCS subsystem inoperable (as a result of the inoperable channel being untripped) is relatively small (8 days versus 7 days), the probability of an accident occurring during the additional time period is low, and the 24 hour repair time is consistent with the overall reliability of LPCI instrumentation.

Specification 3.3.6.1 - Less Restrictive

L9 CTS Table 3.3.2-1, Isolation Actuation Instrumentation, Function 4.a.4, HPCI steam line tunnel temperature — high, requires 2 channels per trip system to be Operable; Function 4.a.9, HPCI equipment area temperature — high, requires 1 channel per trip system to be Operable; Function 4.b.4, reactor core isolation cooling (RCIC) steam line tunnel temperature — high, requires 2 channels per trip system to be Operable; and Function 4.b.9, RCIC equipment room ambient temperature — high, requires 1 channel per trip system to be Operable. ITS Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 3.f, HPCI steam line area temperature — high (CTS Table 3.3.2-1, Function 4.a.4; the name of this function is changed in ITS by comment A.13), requires 1 channel per trip system to be Operable; Function 3.i, HPCI equipment area temperature — high, requires 2 channels per trip system to be Operable; Function 4.f, RCIC steam line area temperature — high (CTS Table 3.3.2-1, Function 4.b.4; the name of this function is changed in ITS by comment A.13), requires 1 channel per trip system to be Operable, and Function 4.j, RCIC equipment area ambient temperature — high (CTS Table 3.3.2-1, Function 4.b.9; the name of this function is changed in ITS by comment A.13), requires 2 channels per trip system to be Operable. As a result, the number of channels per trip system required to be Operable for ITS Table 3.3.6.1-1 Functions 3.f and 4.f are decreased from "2" to "1" and the number of channels per trip system required to be Operable for ITS Table 3.3.6.1-1 Functions 3.i and 4.j are increased from "1" to "2."

The HPCI and RCIC ambient temperature isolation instrumentation are provided to monitor each of the following areas that contain high energy HPCI or RCIC piping: the HPCI and RCIC steam line tunnel; the HPCI and RCIC steam line area; the HPCI equipment area; and the RCIC equipment area. The HPCI and RCIC steam line tunnel is the enclosed HPCI and RCIC mini-steam tunnel; the HPCI and RCIC steam line area is the pipe chase area between the HPCI and RCIC mini-steam tunnel and the HPCI and RCIC equipment areas; the HPCI equipment area is the area, below the elevation of the HPCI room roof, that contains the HPCI pump unit and turbine; and the RCIC equipment area is the area, below the elevation of the HPCI room roof, that contains the RCIC pump and turbine.

A discrepancy has been identified in the location of these instruments relative to the areas that they actually monitor. It has been determined that some of the instruments are located in the HPCI and RCIC pipe chase and some of the instruments are located in the HPCI and RCIC equipment areas. Since instruments located in the equipment areas may not detect a break in the pipe chase, it is inappropriate to group them together as is currently done in CTS Table 3.3.2-1, Functions 4.a.4, 4.a.9, 4.b.4, and 4.b.9. Regrouping the instruments based on actual instrument locations is necessary to ensure that adequate protection is provided in the event of a line break or leak in any of the affected areas. The HPCI and RCIC steam line area only contains four instrument channels (two associated with HPCI isolation and two associated with RCIC isolation), one in each trip system associated with HPCI isolation and one in each trip system associated with RCIC isolation. The other four instruments (two associated with HPCI isolation and two associated with RCIC isolation) are actually located in the HPCI and RCIC equipment areas. Therefore, the required number of channels per trip system for ITS Table 3.3.6.1-1 Functions 3.f and 4.f are decreased from "2" to "1" and the required number of channels per trip system for ITS Table 3.3.6.1-1 Functions 3.i and 4.j are increased from "1" to "2."

The HPCI and RCIC ambient temperature isolation instrumentation functions are assumed to provide HPCI and RCIC isolation for certain high energy line break analyses. These analyses are not impacted since the change continues to ensure that at least one high temperature leak detection instrument channel per trip system is maintained Operable in each of the areas that contain high energy HPCI and RCIC piping. The HPCI and RCIC leak detection logic is designed such that the trip of a single instrument channel will close either the inboard or outboard system isolation valve and isolate HPCI and RCIC, as applicable. Requiring at least one Operable channel per trip system ensures that, in the event of a single failure of one of the instrument channels, the instrumentation will still perform its safety function to isolate HPCI and RCIC, as applicable, in the event of a high energy line break. Diversity is provided by the HPCI and RCIC high steam flow isolation instrumentation (ITS Table 3.3.6.1-1 Functions 3.a and 4.a) which will also actuate to provide HPCI or RCIC isolation in the event of excessive HPCI or RCIC steam flow that is indicative of a high energy line break.

L10 ITS 3.3.6.1 Required Action H.1 is added which allows the standby liquid control (SLC) subsystems to be declared inoperable if reactor water cleanup system (RWCU) isolation is not desired. The purpose of isolating the RWCU on a SLC system initiation signal (ITS Table 3.3.6.1-1 Function 5.f) is to ensure that following an accident the SLC subsystems function properly and the RWCU system does not remove boron from the reactor coolant system. By declaring the affected supported equipment inoperable (i.e., the SLC system), and as a result taking the TS actions of the affected supported equipment, unit operation is maintained within the bounds of the TS approved remedial actions. Since the SLC system initiation isolation of the RWCU system supports the Operability of the associated SLC subsystems, it is appropriate that the proper action, in this condition, would be to declare that affected supported equipment inoperable. CTS Table 3.3.2-1 Action 24 is overly restrictive, in that if the associated supported equipment were inoperable for other reasons, a restoration time is provided in the CTS system specification. The time period provided for declaring the associated SLC subsystems inoperable is consistent with the existing time allowed in CTS Table 3.3.2-1 Action 24 to isolate the RWCU system. The additional time to operate without isolating the SLC system allowed by this change is acceptable because the probability of an event occurring requiring SLC system isolation during the time period when the capability to isolate the RWCU system is lost is low.

Specification 3.3.7.1 - Less Restrictive

L4 CTS 3.3.5.5 requirements are modified in ITS 3.3.7.1 with a note to SRs which states that when a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated function maintains control room emergency ventilation (CREV) system initiation capability. This change is acceptable because the delay allowed by the SR note only applies when the CREV system initiation function is maintained by the redundant Control Building Air Intake Radiation — High channel, and because the 6 hour period is based on a staff approved topical report GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992. The staff documented the applicability of GENE-770-06-1-A to BNP for the CREV System instrumentation in for Unit 1 in Amendment 175 and for Unit 2 in Amendment 206. This change adopts the allowances for surveillance allowed outage times and is consistent with the STS.

Specification 3.3.8.1 - Less Restrictive

L3 CTS 3/4.3.3 requirements are modified in ITS 3.3.8.1 with a note to SRs which allows a 2 hour delay to entering into the associated Conditions and Required Actions for a channel placed in an inoperable status solely for performing required Surveillances provided the 4.16 kV Emergency Bus Undervoltage Loss of Voltage instrument function (Table 3.3.8.1-1, Function 1) maintains initiation capability for three of the four diesel generators (DGs) and the 4.16 kV Emergency Bus Undervoltage Degraded Voltage instrument function (Table 3.3.8.1-1, Function 2) maintains DG initiation capability for all four diesel generators. For Function 1, the loss of function condition is justified in this case since only three of the four DGs are required to start within the required time and energize their associated three of the four 4.16 kV buses to meet accident analysis assumptions. For Function 2, the provisions of Note 2 are only applicable if a loss of function has not occurred (i.e., DG initiation capability is maintained). The short period of time (2 hours) in this condition has no appreciable impact on risk based on the systems that remain operable. In addition, upon completion of the surveillance test or upon expiration of the 2 hour allowance TS require that the channel be returned to operable status or the applicable Condition must be entered and Required Actions taken.

Specification 3.4.5 - Less Restrictive

L1 CTS 3.4.3.1 requires that both the containment particulate and gaseous radioactivity monitors to be Operable along with the drywell floor drain sump flow monitoring system. Consistent with the STS, ITS 3.4.5 requires that either one of the monitors along with the drywell floor drain sump flow monitoring system be Operable. This less restrictive requirement is acceptable because the two radioactivity monitors are not diverse as both rely on measuring the radioactivity that would result from a leak in order to monitor leakage. From a safety standpoint, it is much more important that two diverse means of monitoring leakage are available than it is to have two similar monitors. Therefore, as long as one of the two radioactivity monitors is Operable along with the sump flow monitoring method, adequate capabilities are available to monitor leakage.

Additionally, the small change from the allowed outage time of CTS 3.4.3.1 (31 days) to the Completion Time of ITS 3.4.5 (30 days) will not affect plant safety as the diverse method of leakage monitoring will still be available.

L2 CTS 3.4.3.1 does not permit entry into applicable Modes or conditions with any method of leakage monitoring inoperable. With the addition of two notes, ITS 3.4.5 allows entry into applicable Modes or conditions with one of the two diverse methods of leakage monitoring inoperable. This change is acceptable because there will always be a method by which leakage is being monitored and the ITS still requires the inoperable method to be restored within the specified Completion Time.

Specification 3.4.10 - Less Restrictive

L1 CTS 3.4.6.2 requires that the reactor steam dome pressure to be less than 1045 psig. ITS 3.4.10 allows the reactor steam dome pressure to be less than or equal to 1045 psig. This change is acceptable because even with steam dome pressure at 1045 psig, steam dome pressure is within the assumptions of the reactor safety analysis.

Specification 3.5.1 - Less Restrictive
Specification 3.5.2 - Less Restrictive

L10 ITS 3.5.1 allows one or both LPCI subsystems to be considered Operable when aligned for
L6 shutdown cooling in Mode 3 and ITS 3.5.2 allows a LPCI subsystem to be considered Operable when that subsystem is aligned for shutdown cooling. Given that LPCI is a multi-function system, without the Notes, the subsystem(s) would have to be considered inoperable in Mode 3 or 4 for either low pressure ECCS or shutdown cooling because of alignment to the other function. Considering one or both LPCI subsystems Operable for ECCS purposes when they are in the shutdown cooling alignment is acceptable because in

Mode 3 or 4 the potential consequences of an event requiring ECCS actuation are considerably less significant than they would be in Mode 1 or 2. Therefore, manual realignment of the subsystem(s) to restore the ECCS function could be done quickly enough to assure plant safety and that being the case, the subsystem(s) can be considered Operable even though they are initially in the shutdown cooling alignment.

Specification 3.5.2 - Less Restrictive

L7 With the plant in Mode 4 or 5, the low pressure ECCS independence required to mitigate a design basis loss of coolant accident is no longer required. The availability of two ECCS subsystems in those Modes, whether independent or cross-connected, provides sufficient equipment to handle any anticipated reactor coolant flow or level problems. Therefore, this less restrictive change is acceptable.

L9 The CTS 3.5.4 requirement to suspend all operations in the reactor vessel, including all positive reactivity changes, when no CS subsystem is available and the suppression pool is below the minimum required level is not retained in the ITS. This less restrictive change is acceptable. In the Modes in which this TS applies, the CS subsystems are required only to mitigate a loss of reactor vessel inventory. Therefore, the ITS retaining the requirement that all operations that have a potential for draining the reactor vessel be suspended when no CS subsystem is available assures plant safety. Further, ITS Section 3.9 adequately addresses the situations removed from the TS (suspending all positive reactivity changes).

Specification 3.6.1.2 - Less Restrictive

L1 ITS 3.6.1.2 Actions Note 1, is added to the ITS to allow entry through a closed or locked air lock door for the purpose of making repairs. This note was not in the CTS. If the outer door is inoperable, it may be easily accessed for repair. However, if the inner door is the one that is inoperable, it is necessary to enter through the Operable outer door, which means there is a short time during which the primary containment boundary is not intact (during access through the outer door). Repairs are directed towards reestablishing two Operable doors in the air lock. Two Operable doors closed is clearly the most desirable plant condition for air locks. CTS 3.6.1.3 Actions, in some circumstances, allow indefinite operation with only one Operable door locked closed provided the Operable air lock door is locked and checked locked every 31 days. Two Operable doors closed is clearly an improvement in safety over one Operable door locked closed. By not allowing access to make repairs, the existing CTS Actions could result in an inability of the plant to establish and maintain this highest level of safety (two Operable doors closed), without a forced plant shutdown. Therefore, allowing entry and exit, while temporarily allowing primary containment inoperability, is based on the expected result of restoring two Operable doors to the air lock. Restricting this access to make repairs of an inoperable door or air lock ensures this allowance applies only towards meeting this goal. This change is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the containment Operability is compromised, and the increased safety attained by completing repairs such that two Operable doors can be closed. This less restrictive change is acceptable and is consistent with the STS.

L2 ITS 3.6.1.2, Action A Note 2 allows entry and exit for 7 days under administrative controls and ITS 3.6.1.2 Action B Note 2 allows entry into and exit from primary containment under the control of a dedicated individual. These Notes are not in the CTS. These Notes are added to the ITS to allow entry through a closed and/or locked Operable air lock door (for reasons other than repairs). Although one Operable air lock door closed is sufficient to maintain containment Operability and allows continued operation, entry and exit during operation may be necessary to perform maintenance and inspections as well as allowing access for operational considerations, such as preventative maintenance, etc. Should the air lock become inoperable and access not be allowed, a plant shutdown could be forced in a short period of time due to failure to attend to these activities. The allowance has strict administrative controls, which are detailed in the ITS 3.6.1.2 Bases. A dedicated (i.e., not involved with any repair or other maintenance effort) individual will be assigned to ensure: (1) the door is opened only for the period of time required to gain entry or exit from the air lock, and (2) the Operable door is re-locked prior to the departure of the dedicated individual. Therefore, allowing the Operable door to be opened (temporarily allowing primary containment inoperability) for brief moments, is an acceptable exchange in risk; the risk of an event during the brief period an Operable door is open for access, versus the risk associated with the transient of the plant shutdown that would follow from not attending to required activities within the primary containment. This is a less restrictive change which is acceptable and is consistent with the STS.

Specification 3.6.1.2 - Less Restrictive
Specification 3.6.1.3 - Less Restrictive
Specification 3.6.4.2 - Less Restrictive

L4 Notes are added to ITS 3.6.1.2 Required Action A.3 and Required Action B.3,
L7 ITS 3.6.1.3 Required Actions A.2 and Required Action C.2, ITS SR 3.6.1.3.1 Note 1
L9 and SR 3.6.1.3.2 Note 1, and ITS 3.6.4.2 Required Action A.2. The Notes added to ITS 3.6.1.2 Required Action A.3 and Required Action B.3 allow verification of the Operable locked closed door by administrative means when the door is located in high radiation areas or areas with limited access due to inerting. ITS 3.6.1.3 Required Action A.2, ITS 3.6.1.2 Required Action C.2, ITS SR 3.6.1.3.1 Note 1, and ITS SR 3.6.1.3.2 Note 1 allow verifying valves and blind flanges by administrative means when the isolation device is located in a high radiation area. ITS 3.6.4.2 Required Action A.2 Note allows verifying isolation devices in high radiation area by administrative means. These allowances are not included in the CTS. These allowances are acceptable since access to these areas is restricted and the probability of door or valve/isolation device misalignment, once it is verified in the proper position, is small. Eliminating the physical door and valve/isolation device verification in areas of high radiation and inerting removes a risk to personnel safety. Also, not requiring access to areas of high radiation to verify proper containment air lock door and valve/isolation device alignment reduces exposure to plant personnel is

consistent with the As-Low-As-Reasonably-Achievable (ALARA) concept. This less restrictive change is acceptable and is consistent with the STS.

Specification 3.6.1.2 - Less Restrictive

L5 CTS 3.6.1.3 Action b requires locking the inner door closed when the primary containment air lock door interlock inoperable. ITS 3.6.1.2 Required Action B.2 requires locking an Operable air lock door closed when the primary containment air lock door interlock is inoperable. Either primary containment air lock door is sufficient to maintain primary containment integrity during a DBA. By closing and locking one Operable air lock door, the interlock function is fulfilled. Therefore, operation can continue. Per CTS 3.6.1.3 Action b if the inner door is inoperable concurrent with the air lock door interlock inoperable, a plant shutdown is required since the inner door is not closed. By deleting the inner door requirement in CTS 3.6.1.3 Action b and allowing the Operable door closed, the change provides the benefit of allowing plant operation to continue indefinitely with an inoperable air lock door interlock and avoids the risk associated with the potential transient of a plant shutdown. This less restrictive change is acceptable and is consistent with the STS.

Specification 3.6.1.3 - Less Restrictive
Specification 3.6.4.2 - Less Restrictive

L1
L2 ITS 3.6.1.3 Actions Note 1, and ITS 3.6.4.2 Actions Note 1 allow penetration flow paths to be unisolated intermittently under administrative controls. ITS SR 3.6.1.3.1, Note 2 and ITS SR 3.6.1.3.2 Note 2 are "not required to be met for PCIVs that are open under administrative controls." These Notes contain an allowance for intermittently opening, under administrative control, closed primary and secondary containment isolation valves/dampers. Intermittently opening closed PCIVs and SCIDs is acceptable due to the low probability of an event that could pressurize the primary and secondary containment during the short time in which the PCIV or SCID is respectively is open and the administrative controls ensuring the affected penetration is isolated when a need for primary containment and secondary isolation is indicated. This less restrictive change is acceptable and is consistent with the STS.

Specification 3.6.1.3 - Less Restrictive

L2 CTS 3.6.3 Action a requires each affected penetration line be isolated within 8 hours by using at least one deactivated automatic valve secured in the isolation position or at least one closed manual valve or blind flange. CTS 3.4.7 Action 2, and CTS 4.6.1.1.a list some, but not all, possible acceptable isolation devices that are used to satisfy the need to isolate a penetration with an inoperable PCIV. ITS 3.6.1.3 Action A requires isolating the affected penetration flow path by using of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. ITS 3.6.1.3 Actions B and C require isolating the affected penetration flow path by using of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.

ITS 3.6.1.3 Action C (ITS 3.6.1.3 Action C Bases) does not allow a check valve to isolate an affected penetration. The ITS 3.6.1.3 Required Actions provide a complete list of acceptable isolation devices. The ITS Actions continue to require isolation of the penetration to allow continued operation. The change does not affect safe operation. Many penetrations are designed with check valves as acceptable isolation barriers. With a higher pressure in the line downstream of the check valve, the check valve is seated and is equivalent to a closed manual valve. ITS 3.6.1.3 Action A provides safety by allowing the use of check valves as isolating devices. ITS Actions allow closed manual valves or check valves with flow secured to satisfy the requirement of isolating an inoperable penetration. This change to CTS 3.4.7 Action 2 provides a consistent presentation for all penetrations. This flexibility does not result in any actual technical change in the CTS. This less restrictive change is acceptable and is consistent with the STS.

Specification 3.6.1.5 - Less Restrictive

L4 CTS 3.6.4.2 Action b requires that "with one Nitrogen Backup System subsystem inoperable, verify the remaining subsystem is Operable and..." The requirement to verify the other subsystem Operable is not included in ITS 3.6.1.5. CTS 3.6.4.2.b requires an Operable nitrogen backup system consisting of two independent subsystems (one subsystem for each pneumatic vacuum breaker). This change acknowledges that the inoperability of a subsystem is not automatically indicative of a similar condition in the redundant subsystem unless a generic failure is suspected and that the periodic Frequencies specified to demonstrate Operability are adequate to ensure equipment Operability. Therefore, this change allows credit to be taken for normal periodic Surveillance as a demonstration of Operability and availability of the remaining components and reduces unnecessary challenges and wear to the redundant subsystem components. This deletion is acceptable.

Specification 3.6.1.6 - Less Restrictive

L2 CTS 3.6.4.1.a requires the suppression chamber-to-drywell vacuum breaker position indicators be Operable. There is no reference to suppression chamber-to-drywell vacuum breaker position indicators in the ITS. CTS requirements for suppression chamber-to-drywell vacuum breaker position indication instrumentation (CTS 3.6.4.1.a, CTS 3.6.4.1 Action c, CTS 4.6.4.1.d.2, and references using vacuum breaker position indication) do not perform any safety function. The indicating instrumentation is used to determine Operability of the suppression chamber-to-drywell vacuum breakers. The indicating instrumentation is not required to support Operability of these vacuum breakers. The STS does not specify indication-only equipment to be Operable to support Operability of a system or component. Control of the availability of, and compensatory activities if not available, for indications and monitoring instruments are addressed by plant operational procedures and policies.

Vacuum breaker position satisfies the ITS SRs for the vacuum breakers. If position indication is not available and vacuum breaker position is not determined, then the SRs are not satisfied and the appropriate ITS Actions are taken for inoperable vacuum breakers in accordance with ITS 3.6.1.6 Actions. The requirements for the vacuum breaker position indication are addressed by the requirements of ITS 3.6.1.6 and associated SRs. CTS 3.6.4.1.a, CTS 3.6.4.1 Action c, CTS 4.6.4.1.d.2, and references using vacuum breaker position indication are not included in the ITS. Deleting these requirements reduces the risk of a transient associated with a required plant shutdown when only the loss of position indication occurs. This deletion is acceptable.

Specification 3.6.1.6 - Less Restrictive

L4 CTS 3.6.4.1 Action a (one or two vacuum breakers inoperable for opening) requires the Operable vacuum breakers to be cycled within 4 hours and once per 15 days thereafter until the inoperable vacuum breakers are restored to Operable status. CTS 3.6.4.1 Action b (an open vacuum breaker) requires the Operable vacuum breakers to be cycled within 8 hours and once per 72 hours thereafter until the vacuum breaker is closed. These requirements to verify the other vacuum breakers are Operable are not included in ITS 3.6.1.6. The change to ITS 3.6.1.6 avoids unnecessary testing of the vacuum breakers, thereby enhancing overall drywell vacuum relief capability. This change acknowledges that the inoperability of a component is not automatically indicative of a similar condition in the redundant components unless a generic failure is suspected. The periodic Frequencies specified in the Surveillances to demonstrate Operability have been shown to be adequate to ensure equipment Operability. Therefore, this change allows credit to be taken for normal periodic Surveillance as a demonstration of Operability and availability of the remaining components and reduces unnecessary challenges and wear to the redundant components. This less restrictive change is acceptable.

Specification 3.6.2.1 - Less Restrictive

L2 CTS 4.6.2.1.c requires performance of an external examination of emergency core cooling system suction line penetrations of the suppression chamber enclosure prior to taking the reactor from cold shutdown after safety relief valves (SRV) operation with the suppression chamber water temperature greater than or equal to 160°F and reactor coolant system pressure greater than 200 psig. The maximum tested suppression pool bulk temperature showing no high loads due to unstable condensation for a high mass flux steam discharge into the suppression pool through an open-ended pipe (i.e., no quencher device at the discharge line exit) was 160°F. The CTS requirement ensures that the postulated high loads due to an SRV discharge through an open-ended pipe with suppression pool bulk temperature greater than or equal to 160°F does not degrade the suction line penetrations of the suppression chamber. This requirement is deleted per NEDO-30832, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," dated December, 1984. The basis for eliminating the requirement is NEDO-30832 demonstrating that there are no undue loads on the suppression chamber or its components from SRV discharges through the quenchers at elevated pressures and temperatures. The Units 1 and 2 SRV discharge lines include 2 SRV discharge line include quencher devices. The CTS 3.6.2.1 and ITS 3.6.2.1 Actions require manually scramming the reactor if suppression pool temperature is > 110°F and depressurizing the reactor to ≤ 200 psig if suppression pool temperature is > 120°F. These Actions ensure depressurization of the reactor to avoid the previously postulated high loads. Thus, there is no need to perform this visual examination ensuring suction line penetrations of the suppression chamber are not degraded due to SRV operation. This deletion is acceptable.

L3 CTS 3.6.2.1 Action b.2 details how to reduce suppression pool temperature to within the limits (by operating at least one residual heat removal loop in the suppression pool cooling mode). These details are not included in ITS 3.6.2.1. Methods for reducing suppression pool temperature to within limits are part of a coordinated response to an unplanned event governed by plant procedures. This detail of how to reduce suppression pool temperature to within limits is not necessary to ensure restoration of suppression pool temperature in a timely manner. The Required Actions of Condition D of ITS 3.6.2.1 ensure the unit is placed in a non-applicable Mode if the suppression pool temperature is not reduced to within limits. In addition, with the unit in a non-applicable Mode, the requirements of ITS LCO 3.0.4 ensure that suppression pool temperature is reduced to within limits prior to entering an applicable Mode. The deletion of these details is acceptable.

Specification 3.6.4.1 - Less Restrictive
Specification 3.6.4.2 - Less Restrictive
Specification 3.6.4.3 - Less Restrictive

L2
L5
L3
CTS 3.6.5.1 Action b, requires suspending irradiated fuel handling in CONDITION 5 or when handling irradiated fuel in the secondary containment, Core Alterations, and activities which could reduce the Shutdown Margin when Secondary Containment Integrity is not restored to Operable status within the required time. CTS 3.6.5.2 Actions, require suspending activities that could decrease Shutdown Margin when secondary containment isolation dampers are not restored to Operable status or the affected penetration is not isolated within the required time. CTS 3.6.6.1 Actions, require suspending activities that decrease Shutdown Margin when a standby gas treatment subsystem is not restored to Operable status within the required time or when two standby gas treatment subsystems are inoperable. These requirements are not retained in ITS 3.6.4.1, 3.6.4.2 and 3.6.4.3 respectively. This change is acceptable because most operations that could decrease Shutdown Margin are included in the definition of Core Alteration which is also suspended by CTS 3.6.5.1 Action b (ITS 3.6.4.1 Required Action C.2), CTS 3.6.5.2 Actions (ITS 3.6.4.2 Required Action D.2), and CTS 3.6.6.1 Actions (ITS 3.6.4.3 Required Actions D.2.2 and E.2), respectively. These requirements and the requirements of ITS 3.1.1, Shutdown Margin, ensure the core is maintained subcritical. Moving a control rod while in Mode 5 (Refueling) is allowed provided there are no fuel assemblies in the associated core cell. Moving source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable

detectors (including undervessel replacement) is allowed by this change. The reactivity change associated with moving these components is insignificant. During operation in Mode 4, the reactor vessel head is tensioned and the Reactor Mode Switch is in the "Shutdown" position. Control rods are prevented from withdrawing by reactor manual control system interlocks when the mode switch is in "shutdown" and there is no other viable method of decreasing Shutdown Margin. This change is acceptable.

Specification 3.6.4.2 - Less Restrictive

L3 CTS 3.6.5.2 Action b requires isolating the affected penetration using a closed damper. ITS 3.6.4.2 Required Action A.1 also allows isolating the affected penetration using a blind flange. This method of isolation ensures that the isolation barrier is not affected by a single active failure and provides an isolation barrier equivalent to that provided by CTS 3.6.5.2 Action b. This less restrictive change is acceptable and is consistent with the STS.

Specification 3.6.4.3 - Less Restrictive

L2 CTS 3.6.6.1 Action a.2 requires, when in Operational Condition 5 or when handling irradiated fuel, restoration of the inoperable SGT subsystem to Operable status within 31 days or suspension of irradiated fuel handling in the secondary containment, Core Alterations, and activities which could reduce the Shutdown Margin. An alternative requirement is provided, in ITS 3.6.4.3 Required Action D.1, to suspend operations if an SGT subsystem is not returned to Operable status within 31 days, and moving irradiated fuel assemblies, Core Alterations, or operations with a potential for draining the reactor bessel (OPDRVs) are conducted. The ITS requirement (ITS 3.6.4.3 Required Action D.1) is placing the Operable SGT subsystem in operation and continuing conducting operations (e.g., OPDRVs). Since one subsystem is sufficient for any accident, the risk of failure of the subsystem to perform its intended function is reduced if it is operating. This less restrictive requirement is acceptable and is consistent with the STS.

Specification 3.8.1 - Less Restrictive

L12 CTS 4.8.1.1.2.a.6 is deleted. The CTS verifies that each DG is aligned to provide standby power to the associated emergency buses once per 31 days. ITS 3.8.1 reflects the intent of the deleted CTS and requires that the DGs to be Operable, the ITS associated SRs for the DGs are adequate to ensure the DGs are maintained Operable. In addition, the definition of DG Operability and plant procedural controls on DG standby alignment are sufficient to ensure that the DG remains aligned to provide the required standby power. In general, these types of controls are addressed by plant specific processes which continuously monitor plant conditions and changes in the status of plant equipment that require entry into Actions (as a result of failure to maintain equipment Operable). This continuous monitoring process includes the timely re-evaluation of the status of compliance with TS requirements when TS equipment becomes inoperable so that the appropriate Conditions are entered and Actions are taken in the event of inoperability of TS equipment.

Therefore, the explicit requirement to periodically verify that each DG is aligned to provide standby power to the associated emergency buses is considered to be unnecessary for ensuring compliance with the applicable TS Operability requirements. This change implements the intent of the CTS requirements.

Specification 3.8.2 - Less Restrictive
Specification 3.8.5 - Less Restrictive
Specification 3.8.8 - Less Restrictive

L1 CTS 3.8.2.1, CTS 3.8.2.2, and CTS 3.8.2.4.2 Actions are deleted. CTS
L1 3.8.2.1 and CTS 3.8.2.2 Actions require suspending operations involving
L1,L2 positive reactivity changes when the minimum required AC sources and AC buses are inoperable. CTS 3.8.2.4.2 requires suspending operations that could decrease Shutdown Margin when the minimum required DC sources and associated DC distribution systems are inoperable. The changes are acceptable because most positive reactivity changes are already included in the definition of Core Alteration which is also required to be suspended by CTS 3.8.2.1 Action (ITS 3.8.2 and 3.8.5) and by CTS 3.8.2.2 and CTS 3.8.2.4.2 Actions (ITS 3.8.8). This requirement and the requirements of ITS LCO 3.1.1, Shutdown Margin, are adequate to ensure that the core is maintained subcritical. Movement of a control rod while in Mode 5 (Refueling) is allowed provided there are no fuel assemblies in the associated core cell. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including under vessel replacement) is also allowed; however, the reactivity change associated with movement of these components is insignificant. Additionally, during operation in Mode 4, the reactor vessel head is tensioned and the reactor mode switch is in the "shutdown" position and when the mode switch is in "shutdown", control rods are prevented from being withdrawn by reactor manual control system interlocks and thus while in Mode 4, there is no other viable method of adding positive reactivity to the core. The changes implement the intent of the CTS and are consistent with the STS.

Specification 3.8.4 - Less Restrictive

L1 CTS 4.8.2.3.2.c.2 is deleted. This CTS requires the verification that cell-to-cell and terminal connections are "clean and tight." The confirmation that the connection is "tight" is typically performed by application of a torque which results in unnecessary stress being applied to the bolted connection. ITS SR 3.8.4.2 has a requirement to verify battery connection resistance and if the connection satisfies the resistance requirements of the ITS, then it can be assumed to be sufficiently "tight". As a result, it is not necessary to verify the connections are "tight". The "clean" requirement in CTS 4.8.2.3.2.c.2 is deleted since it is redundant to the "free of corrosion" requirement in ITS 3.8.4.2. In addition, the requirement to verify that connections are "clean and tight" is only applicable to nickel cadmium batteries and the DC electrical power subsystem batteries are lead calcium batteries. The change implements the intent of the CTS and is consistent with the STS.

Specification 3.10.7 - Less Restrictive

L1 ITS 3.10.7 allows ITS 3.1.6 to be suspended for performance of the SDM demonstrations, control rod scram time testing, and control rod friction testing provided certain requirements are met. This is a less restrictive change because the CTS did not provide for this exception. The change is acceptable as adequate assurance is provided that the test sequence will be followed thereby ensuring that the assumptions of the special safety analysis for the test sequence is satisfied. Either the test sequence will be followed by programming that test sequence into the rod worth minimizer (with conformance verified by ITS SR 3.3.2.1.8) or, with the rod worth minimizer bypassed, the test sequence will be verified by a second licensed operator or qualified member of the technical staff.

Section 4.0 - Less Restrictive

L.1 ITS 4.2.1 adds an allowance to CTS 5.3.1 requirements for using lead test assemblies and an allowance to substitute a limited number of filler rods for fuel rods in a fuel assembly. This change precludes an ITS change if this allowance is used. The requirements of 10 CFR 50.59 for special tests or modifications remain applicable, and ensure that a limited number lead test assemblies placed in nonlimiting core regions, or substituting a limited number of zirconium or stainless steel filler rods for fuel rods in a fuel assembly, does not have an affect on safety. This change is consistent with guidance in Supplement 1 of Generic Letter 90-02, which was issued by the NRC. This less restrictive change is consistent with the STS.

L.2 CTS 5.6.1.1 and 5.6.1.2 include requirements for the K_{eff} of the new fuel storage racks and the spent fuel storage racks. The K_{eff} limits are specified as less than (<). The analytical values described in the Updated Final Safety Analysis Report (UFSAR) are specified as less than or equal to (≤). As such, less than or equal to (≤) is specified for the K_{eff} limits in ITS 4.3.1.1 and 4.3.1.2. This change is implemented because the difference between the CTS limits and the ITS limits are insignificant and the reactivity difference allowed by this change is negligible and consistent with current analyses. This change is consistent with the STS.

L3 CTS 5.6.2 specifies the water level to which spent fuel storage pool is designed and maintained to prevent inadvertent draining. The water level in CTS 5.6.2 is lowered from 115 ft to 11 inches to 94 ft 7 inches in ITS 4.3.2. The CTS level is the design level that the spent fuel storage pool can be drained with the fuel pool gates installed. The ITS level is the minimum design level the spent fuel storage pool is drained with the fuel pool gates removed. The gates are removed during refueling outages to transfer fuel between the spent fuel storage pool and the reactor vessel. This minimum design level prevents exposing the fuel in the event of a complete loss of spent fuel pool level. This change is implemented because lowering the specified design level reflecting the lowest plant design level associated with the spent fuel pool does not impact safety since the spent fuel pool level is maintained controlled by ITS 3.7. The lower of the two design limits (94 ft 7 inches) is incorporated in ITS 4.3.2.

Specification 5.5 - Less Restrictive

L1 CTS Surveillance 4.6.6.1.d.1 requires that the pressure drop across the combined HEPA filters and charcoal absorber banks be less than 8.5 inches water gauge while operating the SGT filter train at a flow rate of 3000 cfm ± 10%. ITS 5.5.7.d, for the SGT subsystems, requires that the pressure drop across the combined HEPA filters, the prefilter, and the charcoal absorbers be less than or equal to 8.5 inches water gauge when the filter train is tested at a flowrate of 2700 cfm to 3300 cfm (3000 cfm ± 10%). The limitation is slightly relaxed to allow the pressure drop across the SGT subsystem filter train (including the prefilter) to be equal to 8.5 inches water gauge and still be Operable. BSEP calculations and evaluations have demonstrated the SGT subsystems Operability and assumed the pressure drop across the filter train (i.e., the HEPA filters, the prefilter, and charcoal absorbers) is 8.5 inches water gauge, not less than 8.5 inches water gauge. The SGT subsystem prefilter in the pressure drop acceptance criterion ensures consistency with the BSEP calculations and evaluations regarding SGT filter train pressure drop.

Specification 5.6 - Less Restrictive

L1 This change proposes to relax the requirement (CTS 6.9.1.4 and 6.9.1.6) for submitting the Personnel Exposure and Monitoring Report and the Annual Radiological Environmental Operating Report. The CTS require the reports to be submitted by March 1 and May 1 of each year, respectively. This proposed change will allow the reports to be submitted by April 30 and May 15 of each year, respectively. Given that the reports are still required to be provided to the NRC on or before April 30 or May 15, as applicable, and cover the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner for the interval between March 1 and April 30 and May 1 and May 15. Additionally, there is no requirement for the NRC to approve the reports. Therefore, this change has no impact on the safe operation of the plant.

L2 CTS 6.9.1.8, 6.9.1.9, and 6.9.1.10 requires the Radioactive Effluent Release Report to be submitted on a semiannual basis. ITS 5.6.3 specifies that this report shall be submitted in accordance with 10 CFR 50.36a. This change allows the report to only be submitted annually since 10 CFR 50.36a(a)(2) requires the report to be submitted on an annual basis. Given that the report is still required to be provided to the NRC and covers the previous calendar year, report completion and submittal is clearly not necessary to assure operation in a safe manner. Additionally, there is no requirement for the NRC to approve the report. Therefore, this change has no impact on the safe operation of the plant.

Specification 5.7 - Less Restrictive

L1 CTS 6.12, which provides high radiation area access control alternatives pursuant to 10 CFR 20.203(c)(2) (revised 10 CFR 20.1601(c)), is revised as a result of the changes to 10 CFR 20, the guidance provided in Regulatory Guide 8.38 (Control of Access to High and Very High Radiation Areas in Nuclear Power Plants), and current industry technology in controlling access to high radiation areas. ITS 5.7 includes updated high radiation area access control alternatives in accordance with 10 CFR 20.1601(c). The ITS 5.7 changes include a capping dose rate to differentiate a high radiation area from a very high radiation area, additional requirements for groups entering high radiation areas, and clarification of the need for communication and control of workers in high radiation areas. This change provides acceptable alternate methods for controlling access to high radiation areas. As a result, this change will not decrease control of exposures from external sources in restricted areas.

Table L lists all CTS requirements that have been relaxed and which pertain to Category I though XII and to the specific listing of changes discussed above. Table L is organized by ITS section and includes: the section designation, followed by the discussion of change identifier, e.g., 1.1 L1 (ITS Section 1.1, DOC L1); a summary description of the change; CTS and ITS LCO references; and a reference to the applicable change categories as discussed above (if applicable) and a "Characterization" of the discussion change.

For the reasons presented above, these less restrictive requirements are acceptable because they will not affect the safe operation of the plant. The TS requirements that remain are consistent with current licensing practices, operating experience, and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected.

D. Relocated Less Restrictive Requirements

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The BSEP design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in STS, and thus provide a basis for ITS. A significant number of changes to the CTS involved the removal of specific requirements and detailed information from individual specifications evaluated to be Types 1 through 4 that follow:

Type 1 – Details of System Design and System Description Including Design Limits

Type 2 – Descriptions of Systems or Plant Operation

Type 3 - Procedural Details for Meeting TS Requirements and Related Reporting Requirements

Type 4 - Performance Requirements for Indication-only Instrumentation and Alarms

The following discussions address why each of the four types of information or specific requirements are not required to be included in ITS.

Details of System Design and System Description Including Design Limits *(Type 1)*

The design of the facility is required to be described in the UFSAR by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings, and maintained in accordance with an NRC-approved QA plan (reference in the UFSAR). In 10 CFR 50.59 controls are specified for changing the facility as described in the UFSAR, and in 10 CFR 50.54(a) criteria are specified for changing the QA plan. In ITS, the Bases also contain descriptions of system design. ITS 5.5.10 specifies controls for changing the Bases. Removing details of system design from the CTS is acceptable because this information will be adequately controlled in the UFSAR, controlled design documents and drawings or the TS Bases, as appropriate.

Descriptions of Systems or Plant Operation *(Type 2)*

The plans for the normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures including procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR. In ITS, the Bases also contain descriptions of system or plant operation. It is acceptable to remove details of systems or plant operation from the TS because this type of information will be adequately controlled in the UFSAR, and the TS Bases, as appropriate.

Procedural Details for Meeting TS and Related Reporting Requirements *(Type 3)*

Details for performing action and surveillance requirements are more appropriately specified in the plant procedures required by ITS 5.4.1, the UFSAR, and ITS Bases. For example, control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and has previously been determined to be unnecessary as a TS restriction. As indicated in Generic Letter 91-04, allowing this procedural control is consistent with the vast majority of other SRs that do not dictate plant conditions for surveillances. Prescriptive procedural information in an action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is therefore required in any event. Other changes to procedural details include those associated with limits retained in the ITS.

The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled in the UFSAR and Bases, as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Similarly, removal of reporting requirements from LCOs is appropriate because ITS 5.6, 10 CFR 50.36 and 10 CFR 50.73 adequately cover the reports deemed to be necessary.

Performance Requirements for Indication-Only Instrumentation and Alarms *(Type 4)*

Indication-only instrumentation, test equipment, and alarms are usually not required to be operable to support TS operability of a system or component unless these items are included in TS as source range monitoring instrumentation, remote shutdown monitoring instrumentation, post-accident monitoring instrumentation, and reactor coolant system leakage detection instrumentation. Thus, with the exception of the source range monitoring instrumentation, remote shutdown monitoring instrumentation, post accident monitoring instrumentation, and reactor coolant system leakage detection instrumentation, STS do not include operability requirements for indication-only equipment. The availability of such indication instruments, monitoring instruments, and alarms, and necessary compensatory activities if they are not available, are more appropriately specified in plant operational, maintenance, and annunciator response procedures required by ITS 5.4.1. Removal of requirements for indication-only instrumentation and alarms from the CTS is acceptable because they will be adequately controlled in plant procedures.

Table RL lists CTS specifications and detailed information removed from individual specifications that are relocated to licensee-controlled documents in ITS. Table RL is organized by ITS section and includes: the section designation followed by the discussion of change identifier, e.g., 3.1.1 LA1 (ITS Section 3.1.1, DOC LA 1); CTS reference; a summary description of the change; the name of the document that retains the CTS requirements; the method for controlling future changes to relocated requirements; a characterization of the change; and a reference to the specific change type, as discussed above, for not including the information or specific requirements in ITS.

The NRC staff has concluded that these types of detailed information and specific requirements are not necessary to ensure the effectiveness of ITS to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement: (1) TS Bases controlled by ITS 5.5.10, "Technical Specifications Bases Control Program;" (2) UFSAR (includes the Technical Requirements Manual (TRM by reference) controlled by 10 CFR 50.59; (3) the Offsite Dose Calculation Manual (ODCM) controlled by ITS 5.5.1; and (4) the QA plans as approved by the NRC and referenced in the UFSAR and controlled by 10 CFR Part 50, Appendix B and 10 CFR 50.54(a). For each of these changes, Table RL also lists the licensee-controlled documents and the TS or regulatory requirements governing changes to those documents.

To the extent that requirements and information have been relocated to licensee-controlled documents, such information and requirements are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, where such information and requirements are contained in LCOs and associated requirements in the CTS, the NRC staff has concluded that they do not fall within any of the four criteria in the Final Policy Statement (discussed in Part II of this safety evaluation). Accordingly, existing detailed information and specific requirements, such as generally described above, may be deleted from the CTS.

E. Relocated Specifications

The Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria may be relocated from existing TS (an NRC-controlled document) to appropriate licensee-controlled documents. These requirements include the LCOs, Action Statements (Actions), and associated SRs. In its application, the licensee proposed relocating such specifications to the UFSAR (includes the TRM by reference), and the ODCM, as appropriate. The staff has reviewed the licensee's submittals, and finds that relocation of these requirements to the UFSAR (and TRM) and ODCM is acceptable, in that changes to the UFSAR will be adequately controlled by 10 CFR 50.59 and changes to the ODCM will be controlled by ITS 5.5.1. These provisions will continue to be implemented by appropriate plant procedures; i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures.

The licensee, in electing to implement the specifications of STS, also proposed, in accordance with the criteria in the Final Policy Statement, to entirely remove certain TS from the CTS and place them in licensee-controlled documents noted in Table R. Table R lists all specifications and specific CTS details that are relocated, based on the Final Policy Statement, to licensee-controlled documents in ITS. Table R provides: a CTS reference; a summary description of the requirement; the name of the document that retains the CTS requirements; the method for controlling future changes to relocated requirements; and a characterization of the discussion of change. The NRC staff evaluation of each relocated specification and specific CTS detail presented in Table R is provided below.

<u>CTS 3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION</u>

The control rod withdrawal block instrumentation is relocated to the TRM. This instrumentation consists of the APRM, the IRM, the SRM and the scram discharge volume (SDV), and as shown in Table 3.3.4-1, shall be Operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2 in the CTS.

3/4.3.4.1 APRM

The APRM control rod block functions to prevent a control rod withdrawal error at power transient utilizing LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power, however, the rod block function is not used to mitigate a DBA or transient.

3/4.3.4.3 Source Range Monitors

The SRM control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown, and startup conditions. No DBA or transient analysis takes credit for rod block signals initiated by the SRMs.

3/4.3.4.4 Intermediate Range Monitors

The IRM control rod block functions to prevent a control rod withdrawal error during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No DBA or transient analysis takes credit for rod block signals initiated by IRMs.

3/4.3.4.5 Scram Discharge Volume

The SDV control rod block functions to prevent control rod withdrawals during power range operation, utilizing SDV signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume remaining to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No DBA or transient takes credit for rod block signals initiated by the SDV instrumentation.

The control rod block LCO and SRs applicable to the APRM, SRM, IRM, and the SDV instrumentation do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the control rod withdrawal block instrumentation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.3.5.3 ACCIDENT MONITORING INSTRUMENTATION

The accident monitoring instrumentation is relocated to the TRM. The accident monitoring instrumentation channels shown in Table 3.3.5.3-1 of the CTS are required to be Operable. Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding according to prediction, i.e., automatic safety systems are performing properly, and deviations from expected accident course are minimal. The NRC position on application of the deterministic screening criteria to post-accident monitoring instrumentation is that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all

Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the BSEP Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have remained in TS. The instruments not meeting these criteria have been relocated from the TS to the TRM. The following summarizes the BSEP position for those instruments currently in TS.

From an NRC SER dated 5/14/85, Subject: Emergency Response Capability: Conformance to R.G. 1.97, Revision 2.

Type A Variables

1. Reactor Pressure Vessel (RPV) Pressure
2. RPV Water Level
3. Suppression Pool Water Temperature
4. Suppression Pool Water Level
5. Drywell Pressure
6. Drywell Temperature
7. Suppression Pool Pressure
8. Drywell and Suppression Pool Hydrogen and Oxygen Concentration

From R.G. 1.97 and CP&L submittal to the NRC dated 2/01/84, "Emergency Response Capability, Regulatory Guide 1.97, Revision 1."

Other Type, Category 1 Variables

Primary Containment Area Radiation - High Range

For other post-accident monitoring instrumentation currently in TS, their loss is not risk-significant since the variable they monitored did not qualify as a Type A or Category 1 variable (one that is important to safety and needed by the operator, so that the operator can perform necessary normal actions).

Since the screening criteria have not been satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO and SRs may be relocated to the TRM. The instruments to be relocated are as follows:

1. Suppression Chamber Atmosphere Temperature
2. Safety/Relief Valve Position Indication
3. Turbine Building Ventilation Monitor
4. Offgas Stack Ventilation Monitor
5. Drywell Radiation (Airborne Radiation Monitors)

The accident monitoring instrumentation and the associated LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the accident monitoring instrumentation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS TABLE 3/4.3.3 ECCS ACTUATION INSTRUMENTATION

The ECCS actuation instrumentation consisting of the core spray bus power monitor, the LPCI bus power monitor, the HPCI bus power monitor, and the automatic depressurization system (ADS) bus power monitor is relocated to the TRM. The ECCS actuation instrumentation channels shown in Table 3.3.3-1 are required to be Operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

3/4.3.3.1.e Core Spray Bus Power Monitor

3/4.3.3.2.f LPCI Bus Power Monitor

3/4.3.3.3.e HPCI Bus Power Monitor

3/4.3.3.4.g ADS Bus Power Monitor

The bus power monitors for the RHR (LPCI), Core Spray, HPCI, and ADS trip systems alarm if a fault is detected in the power system to the appropriate system's logic. No DBA or transient analyses take credit for the bus power monitors. This instrumentation provides a monitoring/alarm function only.

The ECCS actuation instrumentation LCO and SRs associated with the RHR (LPCI), Core Spray, HPCI, and ADS bus power monitors do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the ECCS Actuation Instrumentation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of Public health and safety.

CTS 3/4.3.3 ECCS ACTUATION INSTRUMENTATION

The ECCS actuation instrumentation is relocated to the TRM. The ECCS actuation instrumentation channels shown in Table 3.3.3-1 are required to be Operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2.

3/4.3.3.4.a' ADS Inhibit Switch

The ADS inhibit switch allows the operator to defeat ADS actuation as directed by the emergency operating procedures under conditions for which ADS would not be desirable. For example, during an ATWS event, low pressure ECCS system activation would dilute sodium pentaborate injected by the SLC system thereby reducing the effectiveness of the SLC system shutdown.

instruments are not assumed to mitigate a DBA or transient since an accidental chlorine release is not a DBA or transient.

3/4.3.5.5.3 Control Room Envelope Smoke Protection

The control room envelope smoke protection instrumentation functions to permit continuous occupancy of the control room emergency zone during an external smoke event. During a smoke event, the CBHVAC system automatically isolates and enters the radiation/smoke protection mode and the emergency air filtration units begin operation to minimize smoke intrusion and provide a positive pressure in the control room envelope. However, the instruments are not assumed to mitigate a DBA or transient since an external smoke event is not a DBA or transient.

The control room emergency ventilation chlorine isolation and smoke protection LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the control room emergency ventilation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.3.5.1 SEISMIC MONITORING INSTRUMENTATION

Specifications for the seismic monitoring instrumentation are relocated to the TRM. This instrumentation shown in Table 3.5.1-1 is required to be Operable. In the event of an earthquake, seismic monitoring instrumentation is required to determine the magnitude of the seismic event. These instruments do not perform any automatic action. They are used to measure the magnitude of the seismic event for comparison to the design basis of the plant to ensure the design margins for plant equipment and structures have not been violated. Since the determination of the magnitude of the seismic event is performed after the event has occurred, this instrumentation has no bearing on the mitigation of any DBA or transient.

The seismic monitoring instrumentation LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the seismic monitoring instrumentation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.3.5.6 CHLORIDE INTRUSION MONITORS

Specifications for the chloride intrusion monitors are relocated to the TRM. These monitors shown in Table 3.3.5.6-1 are required to be Operable with alarm setpoints set consistent with the values shown in the Trip Setpoint columns of Table 3.3.5.6-2.

This specification ensures the Operability of the chloride intrusion monitors thereby, providing adequate warning of leakage in the main condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. Chloride concentration is maintained to reduce the possibility of failure in the reactor coolant system pressure boundary caused by corrosion. Poor reactor coolant water chemistry contributes to the long-term degradation of system materials and, thus is not of immediate importance to the plant operator. In summary, the chloride intrusion monitor requirements serve a long-term preventative purpose rather than mitigative purpose.

The chloride intrusion monitors LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the chloride intrusion monitors, as relocated to the TRM, will require a safety evaluation pursuant to 10CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.3.5.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Specifications for the radioactive liquid effluent monitoring instrumentation are relocated to the ODCM. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.5.8-1 are required to be Operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints are determined in accordance with the ODCM. The radioactive liquid effluent monitoring instrumentation is neither a safety system nor is connected to the reactor coolant system. This instrumentation is used for the purpose of showing conformance to the discharge limits of 10 CFR Part 20. It is not installed to detect excessive reactor coolant leakage. The radioactive liquid effluent monitors are used to provide continuous check on the release of radioactive liquid effluent from the normal plant liquid effluent flow paths. These TS require the Licensee to maintain Operability of various liquid effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive effluent monitors.

The radioactive liquid effluent monitoring instrumentation LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the radioactive liquid effluent monitoring instrumentation, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Any changes must also conform to 10 CFR Part 20 and ITS 5.5.1. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.3.5.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Specifications for gaseous effluents are relocated to the ODCM. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.5.9-1 are required to be Operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints are determined in accordance with the methodology as described in the ODCM.

The radioactive gaseous effluent monitoring instrumentation is neither a safety system nor is it connected to the reactor coolant system. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The radioactive gaseous effluent monitors are used routinely to provide continuous check on the releases of radioactive gaseous effluents from the normal plant gaseous effluent flow paths. These TS require the Licensee to maintain Operability of various effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive effluent monitors. In addition, the explosive gas monitor instrumentation is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gaseous radwaste treatment system is adequately monitored, which will help ensure that the concentration is maintained below the flammability limit of hydrogen. However, the offgas system is designed to contain detonations and will not affect the function of any safety related equipment. The concentration of hydrogen in the offgas stream is not an initial assumption of any DBA or transient analysis.

The radioactive gaseous effluent monitoring instrumentation LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the radioactive gaseous effluent monitoring instrumentation, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Any changes must also conform to 10 CFR Part 20 and ITS 5.5.1. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.4.4 CHEMISTRY

The chemistry specification is relocated to the TRM. The chemistry of the reactor coolant system is required to be maintained within the limits specified in Table 3.4.4-1. Poor reactor coolant water chemistry may contribute to the long term degradation of system materials and thus is not of immediate importance to the plant operator. Reactor coolant water chemistry is monitored for a variety of reasons. One reason is to reduce the possibility of failures in the reactor coolant system pressure boundary caused by corrosion. The chemistry monitoring activity serves a long term preventative rather than mitigative purpose.

The reactor coolant system chemistry LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the reactor coolant system chemistry specification, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.4.8 STRUCTURAL INTEGRITY

Specifications for ASME Code Class 1, 2, and 3 components are relocated to the TRM. The structural integrity of ASME Code Class 1, 2, and 3 components is required to be maintained in accordance with Specification 4.4.8. The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components' life. Other TS require important systems to be Operable (for example, HPCI 3/4.5.1) and in a ready state for mitigative action. This TS is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this specification to ensure immediate Operability of safety systems.

Further, this TS prescribes inspection requirements which are typically performed during plant shutdown. It is, therefore, not directly important for responding to DBAs.

The structural integrity LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the structural integrity LCO and SRs, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.7.2 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

Specifications for the control room emergency ventilation system are relocated to the TRM. The control room emergency ventilation system is required to be Operable with:

3/4.7.2.a An Operable Smoke Protection Mode consisting of two Operable Control Room Emergency Filtration Subsystems.

3/4.7.2.b An Operable Chlorine Protection Mode

The CREV smoke protection mode functions permit continuous occupancy of the control room emergency zone during an external smoke event. During a smoke event, the CBHVAC system automatically isolates and enters the radiation/smoke protection mode and the emergency air filtration units begin operation to minimize smoke intrusion and provide a positive pressure in the control room envelope. However, the smoke protection mode of CBHVAC is not assumed to mitigate a DBA or transient since smoke intrusion is not a DBA or transient.

The CREV chlorine protection mode functions to permit continuous occupancy of the control room emergency zone during a toxic gas event. In the event of a chlorine release, the CBHVAC system enters a full recirculation mode, with no outdoor air intake. However, the chlorine protection mode of CBHVAC is not assumed to mitigate a DBA or transient since an accidental chlorine release is not a DBA or transient.

The CREV chlorine and smoke protection mode LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the CREV, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.7.3 FLOOD PROTECTION

Specifications for flood protection are relocated to the TRM. Flood protection is required to be provided for all safety-related systems, components, and structures when the water level of the intake canal exceeds 17'6" Mean Sea Level U.S. Geological Survey (USGS) datum.

This TS has provisions for high intake canal level and level instrumentation. A high intake canal water level is a preliminary indication of flood conditions. Flooding is not a design basis accident or transient; thus, intake canal water level is not credited in any safety analysis. The flood protection TS requirements were put in place to ensure that facility protective actions will be taken and operation will be terminated in the event of flood conditions. This requirement is adequately controlled in plant emergency procedures.

The flood protection LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding flood protection, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.7.6 SEALED SOURCE CONTAMINATION

Specifications for sealed source contamination are relocated to the TRM. Each sealed source containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material is required to be free of greater than or equal to 0.005 microcuries of removable contamination.

The limitations on sealed source contamination are intended to ensure that the total body dose or individual organ irradiation dose does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated SRs bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation.

The sealed source contamination LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding sealed source contamination, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.9.5 COMMUNICATIONS

The communications specifications are relocated to the TRM. Direct communication is required to be maintained between the control room and refueling platform personnel. Communication between the control room and refueling platform personnel is maintained to ensure that refueling personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling platform personnel (such as the insertion of a control rod prior to loading fuel). However, the refueling system design accident or transient response does not take credit for communications.

The communications LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the communications specifications, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.9.6 CRANE AND HOIST OPERABILITY

The specifications for crane and hoist Operability are relocated to the TRM. All cranes and hoists used for handling fuel assemblies or control rods within the reactor pressure vessel are required to be Operable.

Operability of the refueling platform equipment (e.g., cranes and hoists) ensures that cranes and hoists have sufficient load capacity for handling fuel assemblies and/or control rods and the core internals and pressure vessel are protected from excessive lifting force if they are inadvertently engaged during lifting operations. Although the interlocks designed to provide the above capabilities can prevent damage to the refueling platform equipment and core internals, they are not assumed to function to mitigate the consequences of a design basis accident. Additionally, in analyzing the control rod withdrawal error during refueling, if any one of the operations involved in initial failure or error is followed by any other single equipment failure or single operator error, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to violation of any limits. Hence, the refueling platform cranes and hoists are not part of the primary success path in mitigating the control rod withdrawal error during refueling.

The crane and hoist Operability LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the crane and hoist Operability specifications, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

Specifications for crane travel over the spent fuel storage pool are relocated to the TRM. Loads in excess of 1600 pounds are required to be prohibited from travel over fuel assemblies in the spent fuel storage pool racks. The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event the load is dropped, the activity release will be limited to that contained in a single fuel assembly and any possible distortion of the fuel in the storage racks will not result in a critical array. Administrative monitoring of loads moving over the fuel storage racks serves as a backup to the crane interlocks.

Although this TS supports the maximum refueling accident assumption in the DBA, the crane travel limits are not monitored and controlled during reactor operation; they are checked on a periodic basis to ensure Operability. The deterministic criteria for TS retention are, therefore, not satisfied.

The crane travel - spent fuel storage pool LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding crane travel over the spent fuel storage pool, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.1.1 LIQUID EFFLUENTS - CONCENTRATION

Specifications for liquid effluents are relocated to the ODCM. The concentration of radioactive material released in liquid effluents to Unrestricted Areas after dilution in the discharge canal is required to be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml.

10 CFR Part 20, BII(2) refers to releases to an unrestricted area of radioactive material in concentrations that exceed the specified limits. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Neither does the system comprise a part of the safety sequence analysis or a part of the primary coolant pressure boundary. Effluent control is for protection against radiation hazards from licensed activities, not accidents.

The liquid effluents concentration LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding liquid effluents concentration, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.1.2 DOSE - LIQUID EFFLUENTS

Specifications for the dose of liquid effluents are relocated to the ODCM. The dose or dose commitment to a Member of the PUBLIC from radioactive materials in liquid effluents released to Unrestricted Areas is required to be limited:

a. During any calendar quarter to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ, and

b. During any calendar year to less than or equal to 6 mrem to the total body and to less than or equal to 20 mrem to any organ.

Limitations of the quarterly and annual projected doses to members of the public which results from cumulative liquid effluent discharges during normal operation over extended periods is intended to ensure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from any design bases accident or transient.

The dose of liquid effluents LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the dose of liquid effluents, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

Specifications for liquid radwaste are relocated to the ODCM. The liquid radwaste treatment system is required to be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site to UNRESTRICTED AREAS would exceed 0.12 mrem to the total body or 0.4 mrem to any organ in a 31-day period.

The requirement for a liquid waste treatment system in 10 CFR Part 50, Appendix A, GDC 60, pertains to controlling the release of site liquid effluents during normal operational occurrences. No loss of primary coolant is involved; neither is an accident condition assumed or implied. The limits for release in 10 CFR Part 50, Appendix I, Sec. II.A, for liquids are design objectives for operation.

The liquid radwaste treatment system LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the liquid radwaste system, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.1 GASEOUS EFFLUENTS - DOSE RATE

Specifications for the dose rate of gaseous effluents are relocated to the ODCM. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the Site Boundary is required to be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

This LCO limits the dose rate due to gaseous effluents in unrestricted areas at any time to a value less than the yearly dose limit of 10 CFR Part 20. This provides reasonable assurance that no member of the public is exposed to annual average concentrations which exceed the limits of 10 CFR Part 20 Appendix B, Table II. This is a limit which applies to normal operation of the plant. It is not assumed as an initial condition of any design basis accident or transient analysis and is not relied upon to limit the consequences of such events.

The gaseous effluent dose rate LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the gaseous effluent dose rate, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.2 GASEOUS EFFLUENTS - DOSE NOBLE GASES

Specifications for the dose of noble gases are relocated to the ODCM. The air dose due to noble gases released in gaseous effluents from the site to areas at and beyond the Site Boundary is required to be limited to the following:

a. During any calendar quarter: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation;

b. During any calendar year: Less than or equal to 20 mrad for gamma radiation and less than or equal to 40 mrad for beta radiation.

Limitation of the quarterly and annual air doses from noble gases in plant gaseous effluents during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any design basis accident or transient.

The gaseous effluents - dose noble gases LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the noble gases dose, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.3 GASEOUS EFFLUENTS - DOSE IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

Specifications for iodine, tritium, and radionuclide gaseous effluents are relocated to the ODCM. The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the Site Boundary are required to be limited to the following:

a. During any calendar quarter: Less than or equal to 15 mrem to any organ;

b. During any calendar year: Less than or equal to 30 mrem to any organ; and

c. Less than 0.1% of the limits of 3.11.2.3(a) and (b) as a result of burning contaminated oil.

Limitation of the quarterly and annual projected doses to members of the public from radionuclides other than noble gases during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any design basis accident or transient.

The gaseous effluents - dose iodine-131, iodine-133, tritium, and radionuclides in particulate form LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding doses from iodine-131, iodine-133, tritium, and radionuclides, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

Specifications for gaseous radwaste treatment are relocated to the ODCM. The gaseous radwaste treatment system is required to be in operation. The gaseous radwaste treatment system (Offgas) system reduces the activity level of the non-condensible fission product gases from fuel defects removed from the main condenser prior to their release to the environs.

The Operability of the offgas system as well as the ventilation exhaust treatment system is required to meet the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50 (i.e.; releases of radioactive materials in gaseous effluents will be kept ALARA). The Operability of the offgas system is not assumed in the analysis of any design bases accident or transient. However, offgas activity is an initial condition of a design basis accident and is being retained in BSEP ITS LCO 3.7.5. Therefore, there is no need to retain this requirement.
The gaseous radwaste treatment system LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the gaseous radwaste treatment system, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.5 VENTILATION EXHAUST TREATMENT SYSTEM

Specifications for the ventilation exhaust treatment system are relocated to the ODCM. The ventilation exhaust treatment system is required to be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from the site to areas at and beyond the Site Boundary, would exceed 0.6 mrem to any organ over 31 days.

This LCO is intended to provide reasonable assurance that releases of radioactive materials during normal operation of the plant are ALARA and to help assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. Additionally, the only ventilation exhaust treatment systems covered by this specification are those installed for turbine buildings' ventilation. These objectives are not related to protection of the public from any design basis accident or transient.

The ventilation exhaust treatment system LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the ventilation exhaust treatment system, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.2.8 DRYWELL VENTING OR PURGING

Specifications for drywell venting or purging are relocated to the ODCM. The drywell is required to be purged to the environment at a rate in conformance with Specification 3.11.2.1. The drywell vent and purge system is used primarily to control primary containment pressure during reactor operation and also used to reduce drywell airborne radioactivity and nitrogen levels before personnel entry. This LCO is intended to provide reasonable assurance that releases from normal drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas. These limits are not related to protection of the public from the consequences of any DBA or transient.

The drywell venting or purging LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the drywell venting or purging, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.3 SOLID RADIOACTIVE WASTE

Specifications for solid radioactive waste are relocated to the UFSAR. The solid radwaste system is required to be used in accordance with a process control program to process wet radioactive wastes to meet shipping and burial ground requirements. The solid radwaste system is a logical continuation of the liquid radwaste system. It operates by the same requirement for effluent control, identified as 10 CFR Part 50, Appendix A, GDC 60. The system serves to control operational release of solid waste, not accidental release.

The solid radioactive waste LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding solid radioactive waste, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.11.4 TOTAL DOSE (40 CFR PART 190)

Specifications for total dose to the public are relocated to the ODCM. The annual (calendar year) dose or dose commitment to any Member of the PUBLIC, due to releases of radioactivity and radiation, from uranium fuel cycle sources are required to be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which are required to be limited to less than or equal to 75 mrem).

This LCO limits the annual doses to individual members of the public from all plant sources. The LCO is intended to assure that normal operation of the plant is in compliance with the provisions of 40 CFR Part 190. These limits are not related to protection of the public from any design basis accident or transient.

The total dose (40 CFR Part 190) LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the total dose, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.12.1 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Specifications for the radiological environmental monitoring program are relocated to the ODCM. The radiological environmental monitoring program are required to be conducted as specified in Table 3.12.1-1. The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operations. This program monitors the long term impact of normal plant operations and is not related to protection of the public from the consequences of any DBA or transient.

The radiological environmental monitoring program LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the radiological environmental monitoring program, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.12.2 LAND USE CENSUS

Specifications for the land use census are relocated to the ODCM. A land use census are required to be conducted and are required to identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest resident, and the nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the location in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broadleaf vegetation.)

Broadleaf vegetable sampling of at least 3 different kinds of vegetation may be performed at the Site Boundary in each of 2 different direction sectors with the highest D/Qs in lieu of the garden census. Specifications for broadleaf vegetation sampling in Table 3.12.1-1(4c) are required to be followed, including analysis of control samples.

The land use census required by this specification supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operation. This program ensures that changes in the use of areas at or beyond the Site Boundary are identified and changes made to the radiological environmental monitoring program, if required.

The land use census LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the land use census, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CTS 3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

Specifications for the interlaboratory comparison program are relocated to the ODCM. Analyses are required to be performed on radioactive materials supplied as part of an interlaboratory comparison program that has been approved by the Commission.

The interlaboratory comparison program required by this specification confirms the accuracy of the measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operation. This program ensures independent checks on the precision and accuracy of the instrumentation used in the measurements of radioactive material for the radiological environmental monitoring program are performed.

The interlaboratory comparison program LCO and SRs do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding interlaboratory comparisons, as relocated to the ODCM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of public health and safety.

CONCLUSION

The relocated CTS discussed above are not required to be in the TS under 10 CFR 50.36 and do not meet any criteria in 10 CFR 50.36(c)(2)(ii). They are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to public health and safety. In addition, the NRC staff finds that sufficient regulatory controls exist under the regulations cited above to maintain the effect of the provisions in these specifications. The NRC staff has concluded that appropriate controls have been established for all of the current specifications, information, and requirements that are being moved to licensee-controlled documents. This is the subject of a license condition established herewith. Until incorporated in the UFSAR and ODCM, changes to these specifications, information, and requirements will be controlled in accordance with the current applicable procedures that control these documents. Following implementation, the NRC will audit the removed provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59. Accordingly, these specifications, information, and requirements, as described in detail in this Safety Evaluation, may be relocated from CTS and placed in the UFSAR or other licensee-controlled documents as specified in the licensee's letter dated November 1, 1996 as modified by the licensee's October 13, 1997 letter (Revision A) and the licensee's April 24, 1998 letter (Revision D).

F. Control of Specifications, Requirements, and Information Removed from the CTS

The facility and procedures described in the UFSAR and TRM, incorporated into the UFSAR by reference, can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures records are maintained and establishes appropriate control over requirements removed from CTS and over future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with other applicable regulatory requirements; for example, the ODCM can be changed in accordance with ITS 5.5.1, the emergency plan implementing procedures (EPIPs) can be changed in accordance with 10 CFR 50.54(q); and the administrative instructions that implement the Quality Assurance Manual (QAM) can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. Temporary procedure changes are also controlled by 10 CFR 50.54(a). The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the licensee's QA Program Description for BSEP and such applicable regulations as 10 CFR 50.59.

The licensee committed in a letter dated November 1, 1996, to confirm that CTS requirements designated for placement in the UFSAR or the TRM are appropriately reflected in these documents, or that they will be included in the next required update of these documents. This is the subject of a license condition established herewith. The licensee has also committed to maintain an auditable record of, and an implementation schedule for, the procedure changes associated with the implementation of ITS. The licensee will maintain the documentation of these changes in accordance with the record retention requirements in the QA Program Description, Volume 1 of the November 1, 1996, letter, as modified by the licensee's October 13, 1997, letter

(Revision A), the April 24, 1998 letter (Revision D), and the May 22, 1998 letter (Revision E), includes a list of the changes involving specific requirements that have been removed from the CTS. For each of these changes, Volume 1 also includes the licensee-controlled documents and the TS or regulatory requirements governing changes to these documents.

G. EVALUATION OF OTHER TS CHANGES INCLUDED IN THE APPLICATION FOR CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS

ITS 3.1.2 CHANGE IN REACTIVITY ANOMALY SURVEILLANCE FREQUENCY

The licensee proposed modifying the reactivity anomaly surveillance frequency from once per full power month to 1100 MWD (megawatt-days)/metric ton (approximately 41 days). Plant specific operating history has shown that reactivity anomaly surveillance acceptance criteria have been continually met over the 1100 MWD/metric ton frequency. The STS suggest a value of 1000 MWD/short ton. The change being requested for BSEP is from 1000 MWD/T (i.e., megawatt days/short ton) to 1100 MWD/T (i.e., megawatt days/metric ton). This change is requested because the process computer provides burn-up in terms of MWD/metric ton. Because 1000 MWD/short ton is approximately equivalent to 1100 MWD/metric ton ((1000 MWD/short ton) x (1.102 short ton/metric ton) = 1102 MWD/metric ton), the change from 1000 MWD/short ton to 1100 MWD/metric ton is administrative in nature and consistent with the intent of STS and is acceptable.

ITS 3.1.5 MODIFIED REQUIRED ACTION FOR INOPERABLE CONTROL ROD SCRAM ACCUMULATOR(S)

The licensee proposed modifying the required actions for inoperable control rod scram accumulators. Currently, the STS state that if the required actions for inoperable scram accumulators are not met, then place the reactor mode switch in shutdown immediately. The licensee has proposed changing the required action to an immediate manual reactor scram rather than placing the mode switch in shutdown. The licensee has stated a manual scram will avoid undesirable challenges to the safety systems on BSEP Unit 2 resulting from Group 1 isolation (which includes mainsteam isolation valve closure on Unit 2 when steam line flow is greater than 40%) which is initiated when moving the mode switch to shutdown. The same change in required actions is incorporated in the Unit 1 ITS for consistency with Unit 2. The STS provide an exception to the mode switch movement by adding a note stating that moving the mode switch to shutdown is not required if all inoperable control rod scram accumulators are associated with fully inserted control rods. Therefore, a manual scram meets the intent of the STS, and is acceptable.

ITS 3.3.2.1 ROD BLOCK MONITOR OPERABILITY REQUIREMENTS

The licensee proposed modifying the Rod Block Monitor (RBM) operability requirements. Currently, the RBM is required to be operable when (1) thermal power is greater than or equal to 30% of Rated Thermal Power (RTP) and less than 90% of RTP with the Minimum Critical Power Ratio (MCPR) less than 1.70. The value of 30% will be replaced by 29% in the new Brunswick

ITS. The change is required to ensure that ITS are consistent with setpoint values identified in NEDC-31654P, "Maximum Extended Operating Domain Analysis for Brunswick Steam Electric Plant," dated February 1989. This change represents an operational condition more restrictive to plant operations, and is acceptable to the staff.

ITS 3.10.1 CONDUCT OF INSERVICE HYDROSTATIC AND LEAK TESTING

The licensee has proposed conducting hydrostatic and leak testing while the reactor is considered to be in mode 4 with the coolant temperature above 212°F, provided a number of mode 3 LCOs are met. The mode 3 LCO's include secondary containment isolation instrumentation, secondary containment, secondary containment isolation valves, and standby gas treatment. The proposed changes are consistent with STS, with the plant design bases, and are consistent with previously approved changes for other BWRs. Therefore, these changes are acceptable.

EMERGENCY CORE COOLING SYSTEM (ECCS) ALLOWED OUTAGE TIMES

The licensee has proposed to add AOTs for the following ECCS combinations:

1. HPCI and one low pressure injection or spray subsystem inoperable (72 hours)
2. One required ADS valve and one low pressure injection or spray subsystem inoperable (72 hours)
3. One LPCI pump and one core spray subsystem inoperable (72 hours)
4. One required ADS valve and HPCI inoperable (72 hours)
5. One LPCI pump inoperable in each LPCI subsystem (7 days)

Currently, STS allow for a 72 hour outage time for conditions 1 and 2. This allowed outage time is based on a reliability study (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) and has been found to be acceptable through operating experience, and has been approved for other BWR/4s in the conversion to STS.

In addition, BSEP has proposed conditions 3 and 4 which are similar to the BWR/4 STS with the allowed outage time of 72 hours. Conditions 1 through 4 were analyzed in NEDC-31624, "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, July 1990. The LOCA analysis for each of the four ECCS conditions states that adequate core cooling is provided. However, in the above conditions, the redundancy is reduced such that an additional single active component failure may not maintain the ability to provide adequate core cooling. Therefore, the proposed BSEP TS have a restrictive Completion Time of 72 hours. Therefore, the changes are consistent with changes previously approved for other BWRs, and are consistent with the plant LOCA analysis and are acceptable.

Brunswick has also proposed to add condition 5, augmenting the existing LCO of one LPCI subsystem inoperable. For condition 5, this change has been previously approved for other BWR/4's; this change allows one LPCI pump in each LPCI subsystem to be inoperable for 7 days.

The BSEP LPCI system is designed with two pumps per subsystem. Each of the four pumps is powered from a separate diesel generator, such that a single failure of a diesel generator will only affect one pump. With one LPCI pump in each subsystem inoperable, the assumptions of the accident analysis can still be met, assuming no additional single failure. This condition is analogous to one LPCI subsystem being inoperable. In fact, under worst case break location conditions, it results in more ECCS subsystems remaining capable of injecting than when only one LPCI subsystem is completely inoperable. Therefore, the change is acceptable.

ITS 3.5.1 REDUCTION IN NUMBER OF REQUIRED AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) VALVES

The licensee has proposed to reduce the number of ADS valves required to be operable from seven to six. This change is based on the analysis summarized in NECD-31624P, "Brunswick Steam Electric Plant Units 1 & 2 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," Revision 2, July 1990, which was approved by the staff in safety evaluation reports (SERs) dated June 1, 1989 and January 10, 1991. This analysis shows that adequate core cooling is provided during a small break LOCA and a simultaneous HPCI System failure with two of seven ADS valves out of service. Therefore, the change is acceptable.

ITS 3.5.1 INCREASE IN ADS MINIMUM PRESSURE OPERABILITY REQUIREMENT

The licensee has proposed an increase in the minimum pressure above which ADS is required to be operable from 113 psig to 150 psig to provide consistency in ECCS and RCIC operability requirements. The ADS is required to operate to lower system pressure sufficiently so that the LPCI and Core Spray Systems can provide makeup to mitigate such accidents. Since LPCI and CS are analyzed to begin injection into the RPV at pressures well above 150 psig, there is no safety significance in the ADS not being operable between 113 and 150 psig. Additionally, the ADS is a backup to the HPCI system which is required to be operable at greater than 150 psig. Therefore, the change is acceptable.

ITS RELAXATION IN LPCI AND CS FLOW ACCEPTANCE CRITERIA

By letter dated November 1, 1996, as supplemented by letters dated October 13, 1998, February 26, 1998, March 13, 1998, and April 24, 1998, the licensee proposed to revise the LPCI and CS flow acceptance criteria under operating conditions from 17000 to 14000 gpm per loop (2 pumps) for LPCI and from 4625 to 4100 gpm for each core spray pump. The Brunswick LOCA analysis, "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDE-31624P, Revision 2, July 1990, assumes a minimum LPCI loop flow of 14000 gpm and a minimum core spray loop flow of 4100 gpm. Therefore, this change is acceptable.

By letter dated November 1, 1996, as supplemented by letters dated October 13, 1998, February 26, 1998, March 13, 1998, and April 24, 1998, Carolina Power & Light Company (CP&L) proposed changes to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2 for the conversion to the Improved Technical Specifications (ITS). The proposed changes include, as a "beyond scope" change to ITS Surveillance Requirement (SR) 3.5.2.5, a reduction in the minimum core spray pump flow acceptance criterion (from 4650 gpm to 4100 gpm) under shutdown conditions.

In support of its request, the licensee stated that the lower CS flow as well as reduced low pressure coolant injection (LPCI) loop flow (2 LPCI pumps per loop) (from 17000 gpm to 14000 gpm) were used in the "Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDE-31624P, Revision 2, July 1990. That short-term cooling reanalysis demonstrated that adequate core cooling is provided by 3 operable LPCI pumps and one operable CS pump during a recirculation loop suction line break event (the case that results in worst case consequences) and by one operable CS pump and one operable LPCI pump during a recirculation loop discharge line break event. This analysis which assumed that the plant was operating at full power has been reviewed and accepted by the NRC staff. The licensee stated that the LOCA reanalysis (NEDE-31624P) at the reduced flowrate for the low pressure ECCS systems indicates that the peak cladding temperature does not exceed 1600°F. CP&L did not perform a long-term cooling reanalysis using the reduced LPCI and CS flowrate and stated that the earlier LOCA analysis, "General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10 CFR 50 Appendix K," NEDO-20655A, which included both short-term and long-term analyses, remains valid. The NEDO-20655A analysis was based on higher values for minimum LPCI and CS loop flows than the proposed flowrate.

The NEDO-20655A long-term cooling analysis demonstrated that only one low pressure ECCS subsystem is required, post-LOCA, to maintain adequate reactor vessel water level. Since the current TS minimum CS loop flow (4625 gpm) is adequate, it is reasonable to assume, based upon engineering judgement, that the proposed minimum CS loop flow of 4100 gpm will be sufficient to maintain adequate reactor vessel water level in the event of an inadvertent draindown. Furthermore, two low pressure ECCS subsystems are required to be operable, and this redundancy ensures adequate reactor water level is maintained in the event of an inadvertent reactor draindown. In addition, periodic trending of CS pump performance for indications of degradation is required by ITS 5.5.6 as part of the Inservice Testing Program and is addressed by plant procedures. These procedures will ensure that any adverse trends in equipment performance are identified and appropriate corrective actions taken. Finally, the CS pumps are components subject to the monitoring requirements of the maintenance rule, 10 CFR 50.65, thus providing further assurance that degraded conditions will be identified and corrected.

The staff has reviewed the information provided in the licensee's submittals in support of the reduced CS pump flow criterion under shutdown conditions. Based upon the reasoning presented above, the staff finds that the proposed reduction in the CS pump flow criterion during shutdown conditions is acceptable.

CTS 3/4.6.1.5 DELETION OF CONTAINMENT INTERNAL PRESSURE TS

The licensee proposes to delete CTS 3/4.6.1.5, Primary Containment Internal Pressure. The CTS is based on the initial assumption of 1.75 psig in the safety analysis, and is required in Modes 1, 2, and 3. A recent GE evaluation (NEDC-32466P, Supplement 1) showed that an initial drywell pressure of 2.5 psig is acceptable for ensuring containment pressure design limits are not exceeded. This initial pressure was used in determining a new P_a, and was submitted to the staff to support the Brunswick power uprate amendment (Brunswick letter BSEP-96-0123, dated April 2, 1996). The Brunswick power uprate was approved by license amendments 183 and 214, dated November 1, 1996. This CTS is not needed since the reactor protection system (RPS) high drywell pressure scram will trip the unit prior to exceeding 2.5 psig (the allowable value is 1.8 psig, with a trip setpoint of 1.7 psig) effectively placing the unit in Mode 3. While the RPS trip is not required in Mode 3, the Emergency Operating Procedures (EOPs) will govern actions if the drywell pressure exceeds 1.8 psig, effectively bounding the 2.5 psig limit. The EOPs will require entry into the RPV control and primary containment control actions. These actions require steps to be taken to reduce primary containment pressure to less than 1.8 psig. The negative pressure limit (-0.5 psig) is essentially controlled by the proper operation of the reactor building-to-suppression chamber vacuum breakers and the suppression chamber-to-drywell vacuum breakers. These vacuum breakers are designed to ensure the negative pressure design limit of the primary containment is not exceeded, and are designed to open at -0.5 psid. Thus, the internal pressure cannot exceed the current -0.5 psig setpoint (which is also in CTS to preclude the negative pressure design limit of the primary containment from being exceeded) under normal circumstances (i.e., non-accident conditions). Since the vacuum breakers and their setpoints are required by the ITS during Modes 1, 2, and 3 (ITS 3.6.1.5 and ITS 3.6.1.6), the negative pressure limit part of the CTS is also not needed. Therefore, the staff finds that the proposed deletion of CTS 3/4.6.1.5 is acceptable.

<u>ITS 5.5.12 ADDITIONAL EXCEPTIONS IN PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM</u>

CTS 6.8.3.4, "Primary Containment Leakage Rate Testing Program" (renumbered 5.5.12 in ITS), requires compliance with the guidelines contained in Regulatory Guide 1.163, dated September 1995, as modified by exceptions which are specifically described in the TS. Currently, there are two exceptions described in the TS; the licensee has proposed to add three more exceptions.

The new exceptions, designated as 5.5.12.c., 5.5.12.d., and 5.5.12.e., are reviewed below.

5.5.12.c. Reduced duration Type A tests may be performed using the criteria and Total Time method specified in Bechtel Topical Report BN-TOP-1, Revision 1.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the NRC staff for implementing Option B of Appendix J to 10 CFR Part 50. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

NEI 94-01, Section 8.0, "Testing Methodologies for Type A, B and C Tests," states that these tests should be performed using the technical methods and techniques specified in ANSI/ANS 56.8-1994, "or other alternative testing methods that have been approved by the NRC." Some licensees wish to use the alternative testing methodology contained in Bechtel Topical Report BN-TOP-1, Revision 1, "Testing Criteria For Integrated Leakage Rate Testing of Primary Containment Structures For Nuclear Power Plants," dated November 1, 1972. The staff approved use of BN-TOP-1 in 1972 and it has been used ever since, primarily because it allows Type A tests (containment integrated leakage rate tests) to be completed in as little as 6 hours instead of the typical 24 hours. Although Option B and ANSI/ANS 56.8-1994 allow tests as short as 8 hours and may be preferable to the dated methodology of BN-TOP-1, the licensee proposes to retain BN-TOP-1 as an option for performing Type A tests. BN-TOP-1 still provides acceptable results and, therefore, continues to be acceptable for plants under either Option A or Option B of Appendix J.

The proposed TS describes the use of BN-TOP-1 as an exception to Regulatory Guide 1.163. Strictly speaking, the use of BN-TOP-1 does not constitute an exception to Regulatory Guide 1.163; it conforms to the provision, quoted above, that allows the use of other alternative testing methods that have been approved by the NRC. Nevertheless, the staff has no objection to specifically citing BN-TOP-1 in the TS so as to avoid any confusion as to its acceptability. Therefore, the staff finds the proposed TS concerning BN-TOP-1 to be acceptable.

5.5.12.d. Performance of Type C leak rate testing of the hydrogen and oxygen monitor isolation valves is not required.

By letter dated May 12, 1987, the staff granted to the licensee an exemption from the Type C testing requirements of Appendix J for the hydrogen and oxygen monitor isolation valves. After Appendix J was revised in 1995, the old requirements of Appendix J were retained and redesignated as "Option A - Prescriptive Requirements." A new option was added, "Option B - Performance-Based Requirements." Further, Option B, section V.B.1., states: "Specific exemptions to Option A of this appendix that have been formally approved by the AEC or NRC, according to 10 CFR 50.12, are still applicable to Option B of this appendix if necessary, unless specifically revoked by the NRC." Similarly, section 1.1 of NEI 94-01 states that the performance-based testing requirements contained in Option B do not invalidate exemptions granted before the issuance of Option B. Therefore, the exemption granted for the subject valves is still in effect. The licensee has stated that they wish to list this exemption in the TS for completeness. Although it is not an exception per se from the Regulatory Guide, the staff has no objection to specifically citing it in the TS so as to avoid any confusion as to its acceptability. Therefore, the staff finds the proposed TS to be acceptable.

5.5.12.e. Performance of Type C leak rate testing of the main steam isolation valves at a pressure less than P_a instead of leak rate testing at P_a as specified in ANSI/ANS 56.8-1994.

By letter dated November 8, 1977, the staff granted to the licensee an exemption from Appendix J to allow the main steam isolation valves (MSIVs) to be Type C tested at a test pressure of 25 psig rather than P_a, the calculated peak containment internal pressure related to the design basis loss-of-coolant accident. P_a is specified in the ITS as being 49 psig. As explained above, this exemption is still in effect. The proposed TS does not specify 25 psig, but the surveillance requirement for MSIV leakage rate testing, ITS 3.6.1.3.9, does, and the original exemption itself also specifies the allowable test pressure. Again, although this exemption is not an exception per se from the Regulatory Guide, the staff has no objection to specifically citing it in the TS so as to avoid any confusion as to its acceptability. Therefore, the staff finds the proposed TS to be acceptable.

EXTENSION OF SURVEILLANCE FREQUENCIES FROM A 18-MONTH TO A 24-MONTH REFUELING INTERVAL (Part 1 of 7) - ELECTRICAL AREA

NRC Generic Letter (GL) 91-04, "Changes In Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," has provided guidance to all holders of operating licenses of nuclear power reactors requesting changes to surveillance intervals to accommodate a 24-month fuel cycle. The basis for the longer cycle is that licensees are planning to use improved reactor fuels because of the significant economic benefits. A longer fuel cycle increases the interval between refueling outages and can impact the performance of the associated technical specification (TS) surveillance requirements.

The staff has required that licensees extending their fuel cycle conduct an evaluation to determine the safety significance of the change. The evaluation is required to confirm that historical plant maintenance and surveillance data support all conclusions reached. Licensees are also required to confirm that assumptions in the plant licensing basis would not be invalidated by performing surveillance requirements at intervals needed to accommodate a 24-month fuel cycle.

By letter dated November 1, 1996, Carolina Power & Light Company (CP&L) submitted an amendment request proposing to revise the Brunswick Steam Electric Plant (BSEP) TS for Units 1 and 2. The proposed revisions would extend the frequency of the selected surveillance requirements (SR) needed to support the adoption of the 24-month fuel cycle at BSEP. The proposed changes are based on guidance in NRC GL 91-04.

The licensee is proposing to extend the following surveillance intervals from 18 to 24 months in the BSEP TS.

Surveillance Requirement 3.8.1.8

This surveillance requirement encompasses the transfer of the unit power supply from the normal offsite circuit to the preferred offsite circuit; and manual transfer of the unit power supply from the preferred offsite circuit to the alternate circuit.

Surveillance Requirement 3.8.1.9

This surveillance demonstrates the diesel generators' (DGs) capability to reject the largest single load without exceeding predetermined voltage and frequency while maintaining a specified margin to the overspeed trip setpoint.

Surveillance Requirement 3.8.1.10

This surveillance demonstrates that DG non-critical protective functions are bypassed on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal and that critical protective functions trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide alarms during abnormal engine conditions.

Surveillance Requirements 3.8.1.11

This surveillance is not part of the Improved TS and is added as new by the licensee. This surveillance demonstrates that the DGs can start and run at a power factor ≤ 0.9 for 60 minutes loaded to ≥3500 kW and ≤3850 kW. The licensee has added this surveillance to ensure that the DG is tested under load conditions that are as close to design conditions as possible.

Surveillance Requirement 3.8.1.12

This surveillance verifies that an actual or simulated ECCS initiation signal is capable of overriding the test mode feature to return each DG to ready-to- load operation to ensure that the DG availability under accident conditions is not comprised as a result of testing.

Surveillance Requirement 3.8.1.13

This surveillances verifies interval between each sequenced load block is within ± 10 percent of the design interval for each load sequence relay. The above assures that under accident conditions loads are sequentially connected to the bus by individual load timers which control the permissive and starting signals to motor breakers.

Surveillance Requirement 3.8.1.14

This surveillance demonstrates DG operation during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal.

Surveillance Requirement 3.8.4.5

This surveillance demonstrates the battery charger's capability to meet design requirements.

Surveillance Requirements 3.8.4.6

This surveillance is related to a battery service test. The battery service test is a test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the dc electrical power system.

The licensee has determined that the extension of the surveillance intervals for all the above surveillance requirements are consistent with NRC GL 91-04. Additionally, the licensee has cited the following considerations to support acceptability of the proposed changes. The design, in conjunction with technical specification requirements which limit the extent and duration of inoperable ac and dc power sources, provides substantial redundancy in ac and dc power sources The DG starting capability to operate under load is demonstrated every 31 days and its ability to reach rated speed and frequency within required time limits is demonstrated every 184 days which will provide prompt identification of any substantial DG degradation or failure. The battery parameters such as float voltage, electrolyte level, and specific gravity are periodically monitored during the operating cycle to verify battery operability and will provide prompt identification of any substantial battery degradation or failure.

Additionally, the licensee has conducted a review of the surveillance test history for each of these surveillances to validate the above conclusion. This historical review of the surveillances test history demonstrates that there are no failures that would invalidate the conclusion and that the impact of the extension of the surveillance intervals from 18 to 24 months on system availability is minimal. The licensee has also determined that the proposed extension of the above surveillance intervals will not invalidate assumptions made in the original Safety Analysis Report for BSEP. Based on the above, the staff finds the proposed changes to be acceptable.

EXTENSION OF SURVEILLANCE FREQUENCIES FROM A 18-MONTH TO A 24-MONTH REFUELING INTERVAL (Part 2 of 7) - INSTRUMENT AND CONTROL AREA

By letter dated November 1, 1996, Carolina Power & Light Company proposed changes to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed changes involve an extension of instrumentation and miscellaneous surveillance test intervals to support 24-month operating cycles. The proposed TS changes were evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24 Month Fuel Cycle," dated April 2, 1991. The proposed TS changes have been divided into two categories. The categories are: (1) changes identified as "Non-Instrumentation Changes", and (2) other changes involving the Channel Calibration Frequency identified as "Instrumentation Changes." This evaluation addresses the Instrumentation Changes.

GL 91-04 identifies seven steps for the evaluation of Instrumentation changes. The licensee evaluated the effect of longer calibration intervals on the TS instrumentation by performing a review of the surveillance test history for all instrumentation including, where necessary, instrument drift. Historical calibration data for components currently calibrated once per 18 months were evaluated to assess the acceptability of extending the calibration interval to 24 months. The failure history evaluation and drift study demonstrates that instrument drift has not exceeded the current allowable limits. In performing the drift study, an effort was made to retrieve all recorded Channel Calibration data for associated instruments, when available, for the past operating cycles. By obtaining all recorded calibration data for the past several cycles of operation, a true representation of instrument drift can be determined.

The licensee has performed a drift evaluation, where necessary, using calibration data obtained from surveillance tests of affected instruments by make, model number, and range. The drift evaluation was performed using a computer model for drift determination developed by General Electric (GE) and based on NEDC-31336, "GE Instrument Setpoint Methodology," which was previously approved by the NRC.

The Boiling Water Reactor Owners' Group (BWROG) committee for Calibration Interval Extension determined that the drift module of the GE Instrument Setpoint Methodology could be used to determine instrument drift for periods up to 30 months (the maximum calibration interval permitted by TS for a 24 month cycle plus 25%) based on actual instrument performance in plant environments. In order to evaluate the impact of extended calibration intervals on drift, GE developed the "General Electric Instrument Trending Analysis System" (GEITAS). This program has been successfully used to determine the drift for other plants including Limerick Generating Station and Peach Bottom Atomic Power Station, both of which received approval for the extension of Channel Calibration SRs to 24 months (based on the use of plant specific data).

The GEITAS program analysis produced drift values at intervals from one to thirty months. The drift values were compared with the drift uncertainty associated with specific instrument setpoint analysis results or in the case of an instrument with no specific setpoint analysis, such as monitoring instruments, a comparison was made to appropriate instrument design criteria. The results of the GEITAS evaluation showed acceptable 30 month drift values within the setpoint analysis drift allowances in those cases where a sufficient amount of historical data to satisfy the computer algorithms and the majority of the as-found and as-left values were within acceptable limits. The calculated drift value was compared to the allowance for the associated instruments as calculated in the associated setpoint analysis and it was verified that technical specification limits provide sufficient margin over the analytical limit to allow for instrument inaccuracies. If an instrument was not in service long enough to establish a calculated drift number, the surveillance interval was extended to a 24 month interval based on justification obtained from the instrument manufacturer. The licensee stated that in no case was a setpoint of an instrument changed to accommodate a drift error larger than previously evaluated and that in all cases the calculated drift fell within the assumptions of the safety analysis. Therefore, in no case was it necessary to change the existing safe shutdown analysis to accommodate a larger drift error.

The licensee developed a program such that instruments with Technical Specification calibration surveillance frequencies extended to 24 months will be monitored to identify all occurrences of instruments found outside the licensee's setpoint analysis assumptions. As-found and as-left calibration data will be recorded for each calibration activity. When as-found conditions are outside allowable limits, an evaluation will be performed to determine if the assumptions made to extend the calibration frequency are still valid and to evaluate the effect on plant safety.

Based on the above evaluation, the staff concludes that the proposed TS change in instrumentation surveillance frequency to a 24 month interval is consistent with the guidance of GL 91-04 for 24 month fuel cycles and the staff approval of NEDC-31336 for setpoint methodology and is, therefore, acceptable. The evaluation of the impact of the proposed changes to the extension of the instrument channel surveillance interval from 18 to 24 months and the proposed changes to plant procedures preserves the system function within the limits of the existing safety analyses. The channel uncertainties projected over the 30 month period can be accommodated within the current Technical Specification setpoint and safety analysis limit. The licensee will implement a monitoring program for assessing the effects of the increased instrument calibration surveillance intervals on future instrument drift which satisfies the GL 91-04 guidance for instrument monitoring and is, therefore, acceptable.

EXTENSIONS OF SURVEILLANCE INTERVALS FROM 18 MONTHS TO 24 MONTHS (Part 3 of 7)

INTRODUCTION

By letter dated November 1, 1996, as supplemented by letters dated October 13, 1998, February 26, 1998, March 13, 1998, and April 24, 1998, Carolina Power & Light Company proposed changes to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2 for the conversion to the Improved Technical Specifications (ITS). The proposed changes include extensions of instrumentation and miscellaneous surveillance test intervals to support 24-month operating cycles.

Most of these surveillance interval extensions have previously been reviewed by the technical staff and are addressed in earlier safety evaluations (SEs) provided in support of the ITS conversion. This SE addresses the remaining interval extensions recently identified by the licensee as not having been addressed in our consolidated draft SE for the ITS conversion.

In Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24 Month Fuel Cycle," dated April 2, 1991, the NRC provides guidelines to evaluate the impact of extended surveillance interval from 18 months to 24 months. GL 91-04 requires the licensees to: (a) evaluate the effect of the increased interval on safety; (b) confirm that the historical plant maintenance and surveillance data support the conclusion; and (c) ensure that the assumptions in the licensing basis would not be invalidated. The NRC advised all licensees that for extending surveillance intervals successful equipment performance history is required. A successful equipment performance history is exhibited by evaluating the surveillance test results, corrective and preventive maintenance history, and operating history for the affected equipment and systems. If the maintenance, testing, and performance history of the affected systems and equipment are determined to be satisfactory, and the assumptions in the plant licensing basis are not invalidated, then performing surveillance at the bounding surveillance interval limit of 30 months (1.25 times 24 months) is permitted by ITS Surveillance Requirement (SR) 3.0.2.

Attachment 4 of the amendment request discusses CP&L's adherence with the GL 91-04 guidelines in evaluating the impact of the 24 months extended surveillance interval. In the Attachment, the licensee confirmed that the licensing basis or the applicable analyzed events were reviewed for each affected surveillance and any impact on the key assumptions was evaluated. The licensee also stated that the individual SR justifications discussed any case in which the extended frequency affected the BSEP UFSAR.

The staff reviewed the licensee's stated approach and methodology and the staff found the approach presented in Attachment 4 acceptable.

EVALUATION

The following is an evaluation of each of the 24-month surveillance interval extensions previously unreviewed by the technical staff.

STANDBY LIQUID CONTROL (SLC) SYSTEM FUNCTIONAL TEST INTERVAL

Current Technical Specification (CTS) 4.1.5.c.1 specifies an 18 month functional testing requirement for the SLC System. Improved Technical Specification (ITS) SR 3.1.7.7 specifies the Frequency for the same Surveillance Requirement. The Surveillance Frequency of this SR is being increased from effectively once every 36 months on a Staggered Test Basis to once every 48 months on a Staggered Test Basis. This SR ensures that the SLC System is capable of injecting into the reactor pressure vessel by verifying a flow path and also by firing one of the explosive valves. The SLC System is a backup safety system to the Control Rod Drive (CRD) System. In the event of a low probability failure of the CRD System, the SLC System is designed to bring the reactor subcritical during the most reactive point in core life. The SLC System is designed so that all active components are single failure proof. In addition, each of the SLC System pumps is tested in accordance with the Inservice Testing Program per ITS SR 3.1.7.6 which verifies system capacity. ITS SR 3.1.7.2 and ITS SR 3.1.7.3 ensure the temperature in the SLC System tank and SLC pump suction piping is maintained to prevent the precipitation of sodium pentaborate. ITS SR 3.1.7.4 verifies the continuity of the charge in the explosive valves. These tests ensure that the SLC System is Operable during the operating cycle. Finally, the explosive valves are designed to be highly reliable.

The licensee cites the inherent system and component reliability and the testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined that the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification for extending the surveillance interval for ITS SR 3.1.7.7 and the staff found the extension acceptable. The licensee addressed all the requirements and guidelines provided in the Generic letter. According to NUREG 1433, operating experience shows these components pass the SR when performed on the 18 month Frequency. This experience indicates that these components will similarly pass the SR when performed on a 24 month Frequency. Part of the basis in NUREG 1433 for conducting the SR on a 18 month Frequency is the need to perform this surveillance under the conditions that apply during a plant outage and the potential for unplanned transient if the surveillance were performed with the reactor at power. Extending the SR to the new 24 month refueling interval will ensure that the SR will continue to be performed under plant outage conditions. The proposed change is therefore acceptable.

SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVE FUNCTIONAL TEST INTERVAL

CTS 4.1.3.1.3 specifies the frequency for functional testing the SDV vent and drain valves as "when the reactor protection system logic is tested per Specification 4.3.1.2." (i.e., once per 18 months). ITS SR 3.1.8.3 specifies the Frequency for the same Surveillance Requirement. The Frequency of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that the SDV vent and drain valves close in ≤ 30 seconds after receipt of an actual or simulated scram signal, and open when the actual or simulated scram signal is reset.

CP&L stated that ITS SR 3.1.8.2 ensures that the mechanical components and portion of the valve logic remain OPERABLE. ITS SR 3.1.8.2 requires that the SDV vent and drain valves be cycled fully closed and fully open on a more frequent basis during the operating cycle. CP&L stated that this test does not ensure that the logic of the SDV vent and drain valves is operable, but logic systems are inherently more reliable as acknowledged in the NRC Safety Evaluation Report, dated August 2, 1993 relating to the extension of Peach Bottom Atomic Power Station, Units number 2 and 3, surveillance interval extension from 18 to 24 months. The SER stated, "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC 30936P) show that the overall reliability of safety systems is not dominated by the reliability of the logic systems, but by the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of a mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

The licensee cites the inherent equipment reliability (as demonstrated by years of operating experience in the nuclear and non-nuclear industry) and more frequent stroke testing of the subject valves in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification for extending the surveillance interval for SR 3.1.8.3 and the staff found the extension acceptable, because:

(1) the licensee addressed all the requirements and guidelines provided in the Generic letter. According to NUREG 1433, operating experience show these components pass the SR when performed during the 18 month frequency;

(2) BWR Standard Technical Specification NUREG -1433 bases for SR 3.1.8.3 (page B3.1-50) specifies the required SDV vent and drains closure time as 60 seconds after the receipt of the scram signal which is longer than the BSEP ITS SDV vent and drain valve closure time of 30 seconds. The closure time is based on the bounding leakage case evaluated in the accident analysis. The LOGIC SYSTEM FUNCTION in LCO 3.3.1.1 and the scram time testing of the control rods in LCO 3.1.3 overlap with this SR and provide testing of the assumed safety function; and

(3) the 24 month testing is based on the need to perform this SR during outage and avoid the occurrence of unplanned transient if performed during reactor power operation.

The proposed change is therefore acceptable.

REACTOR PROTECTION SYSTEM (RPS) LOGIC SYSTEM FUNCTIONAL TEST AND CHANNEL FUNCTIONAL TEST INTERVALS

CTS 4.3.1.2 specifies a frequency for the RPS Logic System Functional Test (LSFT) as once every 18 months and CTS Table 4.3.1-1 specifies a frequency for the Channel Functional Test of Function 11 (Reactor Mode Switch in Shutdown Position) as 18 months. In ITS SR 3.3.1.1.12, the frequency of the Channel Functional Test is specified as once every 24 months for ITS Table 3.3.1.1-1 Function 10 (Reactor Mode Switch in Shutdown Position). In ITS SR 3.3.1.1.15, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. These SRs ensure that RPS logic and the RPS Reactor Mode Switch in Shutdown Position Function will function as designed in response to an analyzed event. Extending the surveillance test interval for the RPS LSFT is acceptable because the RPS is verified to be operating properly throughout the operating cycle by the performance of Channel Checks and, in some cases, Channel Functional Tests. This testing ensures that a significant portion of the RPS circuitry is operating properly and will detect significant failures of this circuitry. Additional justification for extending the surveillance test interval is that the RPS network, including the actuating logic, is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the RPS LSFT and Reactor Mode Switch in Shutdown Position Channel Functional Test surveillance intervals to 24 months. The proposed change is therefore acceptable.

RPS RESPONSE TIME TESTING INTERVAL

CTS 4.3.1.3 specifies the Frequency for RPS Response Time Testing as once every 18 months. In ITS SR 3.3.1.1.17, the frequency for RPS Response Time Testing is every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that RPS trip logic functions within the response time assumed in the analyses of the applicable analyzed event. Extending the interval between response time tests for this function is acceptable because the RPS is verified to be operating properly throughout the operating cycle by the performance of Channel Checks and, in some cases, Channel Functional Tests. This testing ensures that a significant portion of the RPS circuitry is operating properly and will detect significant failures of this circuitry. Additional justification for extending the surveillance test interval is that the RPS network, including the actuating logic, is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and the testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the RPS Response Time Testing surveillance interval to 24 months. The proposed change is therefore acceptable.

ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.3.6.1.2 specifies the Frequency for the ATWS-RPT LSFT as once every 18 months. In ITS SR 3.3.4.1.5, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that ATWS-RPT trip logic will function as designed to ensure proper response during an analyzed event. Extending the SR interval for this function is acceptable because the ATWS-RPT logic is tested every 92 days by the Channel Functional Test in ITS SR 3.3.4.1.2. This testing of the ATWS-RPT logic system ensures that a significant portion of the circuitry is operating properly and will detect significant failures of this circuitry. The ATWS-RPT logic including the actuating logic is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the ATWS-RPT Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.3.3.2 specifies the frequency for the ECCS LSFT as once every 18 months. In ITS SR 3.3.5.1.5, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that ECCS logic will function as designed to ensure proper response during an analyzed event. ECCS systems are tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The testing of the ECCS system ensures that a significant portion of the ECCS circuitry is operating properly and will detect significant failures of this circuitry. The ECCS network including the actuating logic is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the ECCS Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.3.7.2 specifies the frequency for the RCIC System LSFT as once every 18 months. In ITS SR 3.3.5.2.5, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that RCIC logic will function as designed to ensure proper response during an analyzed event. The RCIC System is tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The testing of the RCIC System ensures that a significant portion of the RCIC circuitry is operating properly and will detect significant failures of this circuitry. The

RCIC (with HPCI as backup) including the actuating logic is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the RCIC System Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.3.2.2 specifies the frequency for the Isolation Actuation Instrumentation LSFT as once every 18 months. In ITS SR 3.3.6.1.7, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that Isolation Actuation Instrumentation logic will function as designed to ensure proper response during an analyzed event. Most PCIVs are tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The testing of the PCIVs ensures that a significant portion of the Isolation Actuation Instrumentation circuitry is operating properly and will detect significant failures of this circuitry. The Primary Containment Isolation System including the actuating logic, with the exception of the Main Stack Radiation-High Function, is designed to be single failure proof and therefore, is highly reliable. The Main Stack Radiation-High Function is supplied from a single stack monitor which provides a signal to two one-out-of-one logic trip systems. The Main Stack Radiation-High Function is redundant to other isolation signals for the same primary containment isolation valves. These redundant signals are single failure proof and highly reliable. Additionally, the Allowable Value of the Main Stack Radiation-High Function is a small fraction of 10 CFR 100 limits. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-

30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the Primary Containment Isolation Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

ISOLATION INSTRUMENTATION RESPONSE TIME TESTING INTERVAL

CTS 4.3.2.3 specifies the frequency for the Isolation System Response Time testing as once every 18 months. In ITS SR 3.3.6.1.8, the frequency for the Isolation Instrumentation Response Time testing is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that Isolation Actuation Instrumentation and components will function as designed to ensure proper response during an analyzed event. Most PCIVs are tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The testing of the PCIVs ensures the components are operating properly and will detect significant failures. The Primary Containment Isolation System including the actuating logic is designed to be single failure proof and therefore, is highly reliable. The licensee cites the inherent system and component reliability and the testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the Isolation Instrumentation Response Time Testing surveillance interval to 24 months. The proposed change is therefore acceptable.

SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.3.2.2 specifies the frequency for the Isolation Actuation Instrumentation LSFT as once every 18 months. In ITS SR 3.3.6.2.5, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that Secondary Containment Isolation Instrumentation logic will function as designed to ensure proper response during an analyzed event. The secondary containment isolation dampers and the Standby Gas Treatment System including the actuating logic are designed to be single failure proof and therefore, are highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the Secondary Containment Isolation Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM INSTRUMENTATION LOGIC SYSTEM FUNCTIONAL TEST (LSFT) INTERVAL

CTS 4.7.2.d.2 specifies the frequency for the CREV System functional test (including initiation logic) as once every 18 months. In ITS SR 3.3.7.1.4, the frequency for the LSFT is specified as once every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that CREV System Instrumentation logic will function as designed to ensure proper response during an analyzed event. The CREV System is tested on a more frequent basis during the operating cycle in accordance with ITS 3.7.3. The testing of the CREV System ensures that a significant portion of the CREV System circuitry is operating properly and will detect significant failures of this circuitry. The CREV System including the actuating logic is designed to be single failure proof and therefore, is highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the CREV System Instrumentation LSFT surveillance interval to 24 months. The proposed change is therefore acceptable.

REACTOR COOLANT SYSTEM LEAK DETECTION INSTRUMENTS CHANNEL CALIBRATION INTERVAL

CTS 4.4.3.1.a and CTS 4.4.3.1.b establish 18 months as the required Frequencies for performance of Channel Calibration of the Reactor Coolant System leak detection instruments. ITS SR 3.4.5.3 will extend the required Frequency for these SRs to 24 months. Therefore, the surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR ensures that the primary containment atmosphere particulate and gaseous monitoring system and the drywell floor drain sump flow monitoring system are Operable and within the established calibration requirements. The primary containment atmosphere particulate and gaseous monitoring systems provide backup and diversity to the drywell floor drain sump flow monitoring system, in that they both will alert the operators to unanticipated leakage from the reactor coolant pressure boundary. The Reactor Coolant System leakage detection instrumentation does not provide for the actuation of any safety devices. The equipment provides a monitoring function only which alerts the operator to a potential plant problem. The setpoint of these devices is not an assumption in any event analysis. ITS SR 3.4.5.1 and SR 3.4.5.2 require that a Channel Check and Channel Functional Test be performed more frequently. These more frequent tests provide assurance during the operating cycle that each instrument is Operable. Therefore, a drift calculation was not performed for these functions. The licensee cites the design of the instrumentation and more frequent testing in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the Reactor Coolant System leak detection instruments Channel Calibration intervals to 24 months. The proposed change is therefore acceptable.

ECCS SYSTEM FUNCTIONAL TEST INTERVAL AND HPCI LOW PRESSURE FLOW TEST INTERVAL

CTS 4.5.3.1.d, CTS 4.5.3.2.c, CTS 4.5.1.c.1, CTS 4.5.1.c.2 and CTS 4.5.1.c.3 specify "once per 18 months" as the frequency for the system functional test of Core Spray (CS), LPCI and HPCI and the low pressure HPCI flow test. ITS SR 3.5.1.8 and ITS SR 3.5.1.9 specify a 24 month frequency for these tests. Therefore, the surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

The ECCS system functional test (ITS SR 3.5.1.8) verifies that the HPCI pump can develop specified flow rate against a system head corresponding to the reactor pressure. ITS SR 3.5.1.9 verifies that ECCS injection/spray subsystem actuates on an actual or simulated auto initiation.

The licensee stated that:

(1) the ECCS network has built-in redundancy so that no single failure prevents the starting of the ECCS;

(2) the logic system is reliable and increased actuation testing interval will have no safety impact. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability in concluding that the impact from this change on system availability is small;

(3) the HPCI low pressure test (ITS SR 3.5.1.7) requires testing every three months to ensure required flow at normal operating pressures. This test could detect significant failures of the HPCI pump and turbine;

(4) HPCI also has a back up system of ADS combined with the low pressure ECCS; and

(5) this review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification and found it acceptable for extending the surveillance interval for SR 3.5.1.8 and 3.5.1.9. In addition, the staff noted that:

(1) the licensee addressed all the requirements and guidelines provided in the Generic letter. According to NUREG 1433, operating experience show these components pass the SR when performed during the 18 month frequency; and

(2) the 24 month testing for SR 3.5.1.8 is based on the need to perform this SR under conditions that apply just prior to or during a startup from plant outage.

The proposed change is therefore acceptable.

AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) SYSTEM FUNCTIONAL TEST INTERVAL

CTS 4.5.2.a specifies "once per 18 months" as the frequency for the ADS system functional test. CTS 4.5.2.b specifies "once per 18 months" as the frequency for the manual operation of each ADS valve. ITS SR 3.5.1.10 and ITS SR 3.5.1.11 specify a 24 month Frequency for these tests. Therefore, the surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

ITS SR 3.5.1.10, the ADS system functional test, and ITS SR 3.5.1.11, manual operation of each ADS valve, are performed to demonstrate that the ADS function operates as designed when initiated either by an actual or simulated initiation signal and that the valve and solenoid are functioning properly. The manual operation of the ADS valves also ensures that no blockage exists in the SRV discharge lines.

CP&L stated that extending the interval between SR performances will not have a significant impact on reliability because the ADS uses two independent and redundant trip systems. The SRVs associated with the ADS are equipped with remote manual switches so that the entire system can be operated manually as well as automatically. The ADS primary function is to serve as a backup to the HPCI System. Furthermore for the ADS system functional test, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability in concluding that the impact from this change on system availability is small.

The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification and found it acceptable for extending the surveillance interval for SR 3.5.1.10 and 3.5.1.11. In addition, the staff note that:

(1) CP&L addressed all the requirements and guidance provided in the Generic Letter (GL 91-04), which is discussed in detail in Attachment 4 of the amendment request;

(2) the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this surveillance to provide complete testing of the assumed safety function;

(3) the 24 month testing for SR 3.5.1.11 (per NUREG SR 3.5.1.12) is based on the need to perform this SR under conditions that apply just prior to or during a startup from plant outage. The SR is performed prior to or during an outage in order to avoid unplanned transient which could occur if the SR is performed during power; and

(4) operating experience shows these components pass the SR when performed during the 18 month frequency.

The proposed change is therefore acceptable.

ECCS RESPONSE TIME TESTING INTERVAL

CTS 4.5.1.c.4, 4.5.3.1.e, and 4.5.3.2.d specify the frequency for ECCS Response Time Testing as once every 18 months. In ITS SR 3.5.1.12, the Frequency for ECCS Response Time Testing is every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

ITS SR 3.5.1.12 verifies the ECCS response time for each ECCS injection/spray is within the limit. This SR ensures that the ECCS function within the response time assumed in the analyses of the applicable analyzed event. Extending the interval between response time tests for this function is acceptable because the ECCS is verified to be operating properly throughout the operating cycle by the performance of Channel Checks, Channel Functional Tests and pump, valve and flow tests. This testing ensures that a significant portion of the associated ECCS is operating properly and will detect significant failures.

Additional justification for extending the surveillance test interval is that the ECCS network, including the actuating logic, is designed to be single failure proof and therefore, is highly reliable. The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact of this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined that the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the ECCS Response Time Testing surveillance interval to 24 months. The proposed change is therefore acceptable.

LOW PRESSURE ECCS SYSTEM FUNCTIONAL TEST INTERVAL

CTS 4.5.3.1.d and CTS 4.5.3.2.c specify "once per 18 months" as the frequency for the system functional test of Core Spray (CS) and LPCI. ITS SR 3.5.2.6 specifies a 24 month Frequency for these tests. Therefore, the surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. The low pressure ECCS system functional test (ITS SR 3.5.2.6) ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of CS and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions.

CP&L stated that the ECCS network has built-in redundancy so that no single failure prevents the starting of the ECCS. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability in concluding that the impact from this change on system availability is small.

The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined that the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification and found it acceptable for extending the surveillance interval for SR 3.5.2.6.

CP&L addressed all the requirements and guidance provided in the Generic Letter (GL 91-04), which is discussed in detail in Attachment 4 of the amendment request. Operating experience also shows that these components pass the SR when performed during the 18 month frequency.

In addition, the 24 month testing for SR 3.5.2.6 is based on the need to perform this SR under conditions that apply just prior to or during a startup from plant outage. The SR is performed prior to or during an outage in order to avoid unplanned transient which could occur if the SR is performed during power.

The proposed change is therefore acceptable.

LOW PRESSURE ECCS RESPONSE TIME TESTING INTERVAL

CTS 4.5.3.1.e and 4.5.3.2.d specify the frequency for ECCS Response Time Testing as once every 18 months. In ITS SR 3.5.2.7, the Frequency for ECCS Response Time Testing is every 24 months. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

SR ensures that the ECCS function within the response time assumed in the analyses of the applicable analyzed event. Extending the interval between response time tests for this function is acceptable because the ECCS is verified to be operating properly on a more frequent basis by the performance of Channel Checks, Channel Functional Tests and pump, valve and flow tests. This testing ensures that a significant portion of the associated ECCS is operating properly and will detect significant failures. Additional justification for extending the surveillance test interval is that the ECCS network, including the actuating logic, is designed to be single failure proof and therefore, is highly reliable. The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The staff reviewed the licensee's justification and found it acceptable for extending the surveillance interval for SR 3.5.2.7. CP&L addressed all the requirements and guidance provided in the Generic Letter (GL 91-04), which is discussed in detail in Attachment 4 of the amendment request. Operating experience also show these components pass the SR when performed during the 18 month frequency.

The proposed change is therefore acceptable.

RCIC SYSTEM FUNCTIONAL TEST AND RCIC LOW PRESSURE FLOW TEST INTERVAL

CTS 4.7.4.c specifies "once per 18 months" as the frequency for the system functional test and low pressure flow test of RCIC. ITS SR 3.5.3.4 and ITS SR 3.5.3.5 specify a 24 month Frequency for these tests. Therefore, the surveillance test interval of these SRs is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. The RCIC system functional test ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of RCIC will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This includes verifying that the RCIC suction is automatically transferred when CST level is low (CST 4.7.3.c.3). The increased interval between SR performances is acceptable because RCIC is not a system that is assumed in the safety analysis. The functions performed by RCIC can be performed by HPCI and Technical Specifications do not permit HPCI and RCIC to be inoperable concurrently. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." The licensee cites the inherent system and component reliability in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The RCIC low pressure flow test (ITS SR 3.5.3.4) ensures the RCIC system is capable of performing the its design function at low reactor pressure. In addition to the low pressure test for which the frequency is being extended, ITS SR 3.5.3.3 requires that RCIC is tested every 3 months to ensure required flow at normal operating pressure. Although conducted at normal operating pressure, this test would detect significant failures of the RCIC turbine or pump that could lead to the failure of the RCIC System to perform its design function at low reactor pressures. The licensee cites the inherent system and component reliability and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend RCIC System functional test and RCIC low pressure flow test surveillance intervals to 24 months. The proposed change is therefore acceptable.

CONTAINMENT ATMOSPHERE DILUTION (CAD) SYSTEM VALVE CYCLING TEST INTERVAL

CTS 4.6.6.2.b.1 specifies "once per 18 months" as the frequency for the cycling of each CAD System power operated, excluding automatic, valve in the flow path through at least one complete cycle of travel. ITS SR 3.6.3.2.3 specifies a 24 month Frequency for this test. Therefore, the surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. ITS SR 3.6.3.2.3, cycling of each CAD System power operated (excluding automatic) valve in the flow path through at least one complete cycle, is performed to demonstrate that the valves are mechanically OPERABLE and functioning properly. Extending the interval between SR performances will not have a significant impact on reliability because the CAD System is a manually initiated system and includes two subsystems. These two subsystems include redundant flow paths such that no single failure of an active component (e.g., a power operated valve in the flow path to primary containment) will render the system inoperable. In addition, most of these power operated valves are cycled on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The licensee cites the system design and more frequent testing performed during the operating cycle in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review of the surveillance test history determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend the CAD System valve cycling test surveillance interval to 24 months. The proposed change is therefore acceptable.

INTEGRATED LEAK TEST INTERVAL

CTS 6.8.3.1.2 specifies the frequency for system integrated leak testing as each refueling interval. The surveillance test interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. ITS 5.5.2 requires an "Integrated leak test requirement for each system at 24 months or less." The licensee concluded that the impact of this change to the testing requirement, if any, on safety is small. The licensee stated that most portions of the subject systems included in this program are visually walked down, while the plant is operating, during plant testing and/or operator/system engineer walkdowns and housekeeping/safety walkdowns. If leakage is observed from these systems, corrective actions will be taken to repair the leakage. Plant radiological surveys will also identify potential sources of leakage. Based on assurance provided by the visual observations during plant operation and the surveys provide monitoring of the systems at a greater frequency then once per 24 months, the licensee concluded that the impact on safety as a result of the proposed changes is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend Integrated Leak Test surveillance interval to 24-months. Therefore, the proposed change is acceptable.

STANDBY GAS TREATMENT (SGT) SYSTEM FILTER TEST INTERVAL

CTS 4.6.6.1.b and 4.6.6.1.d specify the frequencies for SGT System Filter Testing. These frequencies are once per 18-months and after 720 hours of charcoal adsorber operation. ITS 5.5.7 specifies the frequencies for SGT System Filter Testing. One of the surveillance test intervals for this testing is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. ITS 5.5.7 requires HEPA filter in-place penetration and bypass leakage testing after each complete or partial replacement of a HEPA filter bank. ITS 5.5.7 also requires charcoal adsorber in-place penetration and bypass leakage testing after each complete or partial replacement of a charcoal adsorber bank. Additionally, ITS 5.5.7 requires in-place filter testing after any structural maintenance on the HEPA filter or charcoal adsorber housings or following significant painting, fire, or chemical release in any ventilation zone communicating with the SGT System. The licensee states that testing after filter maintenance, fire, chemical release, painting, HEPA replacement, or charcoal replacement, will ensure that potential changes in HEPA filter efficiency and carbon adsorber bypass leakage are detected. The licensee cites the inherent system and component reliability, in addition to the testing performed during the operating cycle, in concluding that the impact from this change on system availability is small. The licensee confirmed this conclusion by performing a review of the surveillance test history. This review determined that there are no failures that would invalidate the conclusion that the impact on system availability is small. The licensee has also determined the proposed surveillance interval extension does not invalidate assumptions made in the BSEP UFSAR.

The licensee has performed the required evaluations and the required confirmations for compliance with GL 91-04 criteria to extend SGT System charcoal adsorber test surveillance interval to 24-months. Therefore, the proposed change is acceptable.

CONCLUSION

The staff reviewed this information provided in the licensee's submittals and determined the results to be satisfactory. Thus, the surveillance interval extensions from 18 to 24-months do not degrade equipment reliability or the safe operation of the plant. Therefore, the staff finds that extending the

18-month surveillance intervals to 24 months is acceptable. Each surveillance to which this change applies is described in the appropriate section of this safety evaluation.

EXTENSION OF SURVEILLANCE FREQUENCIES FROM A 18-MONTH TO A 24-MONTH REFUELING INTERVAL (Part 4 of 7) - PLANT SYSTEMS AREA

INTRODUCTION

By letter dated November 1, 1996, the Carolina Power & Electric Company (licensee) requested an amendment to the Technical Specifications (TSs) for the Brunswick Steam Electric Plant, Units 1 and 2. The proposed TS amendment would convert the Current Technical Specifications (CTS) into the Improved Technical Specifications (ITS) which is in a format consistent with NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 4." Included in the submittal, the licensee also requested an extension to the test intervals of selected surveillance requirements (SR) in the CTS from 18 months to 24 months (with an additional 25-percent grace period) to accommodate its planned change to a 24-month fuel cycle. Additional information justifying the 24-month SR frequencies was provided by letter dated June 17, 1997. The following evaluation resulted from a review of the licensee's requested changes to the test frequencies for certain SRs in Section 3.7 of the CTS.

Generic Letter 91-04 provides generic guidance to support the development of TS revisions to allow a 24-month fuel cycle and includes requirements to evaluate the effects on safety for an increase in surveillance intervals to accommodate a 24-month fuel cycle. The licensee's evaluation should conclude that the net effect on safety is small, that historical plant maintenance and surveillance data support the proposed surveillance interval extension, and that the assumptions in the plant design and safety analyses are still bounding with the incorporation of a 24-month surveillance interval.

The licensee divided the changes in the surveillance intervals into two categories, i.e., (1) changes involving the channel calibration frequency identified as "instrumentation changes," and (2) other changes identified as "non-instrumentation changes." The licensee has evaluated the proposed surveillance interval extensions in accordance with the guidance provided in GL 91-04 and provided justification for each non-instrumentation SR in CTS Section 3.7 that is being retained in the ITS. The licensee's conclusion on the impact of the proposed changes on system availability, and safety, and the basis for these conclusions are addressed below.

ITS SR 3.7.2.5, SERVICE WATER SYSTEM FUNCTIONAL TEST

CTS 4.7.1.2.b requires that a functional test be performed on the service water system (SWS) once every 18 months. In ITS SR 3.7.2.5, the frequency for the SWS functional test is changed to once every 24 months. The licensee therefore requested to increase the surveillance test interval of this SR from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR is required to verify the operability of the SWS actuation circuitry to ensure its capability of automatic initiation.

The licensee evaluated the change in test frequency and concluded that the impact of this change, if any, on system availability is minimal. This conclusion is based on the fact that the SWS circuitry is designed to meet the requirements of IEEE 279 which ensures that no single circuit fault can prevent the safety system function. Furthermore, the service water pump and valves which are required to function are tested on a more frequent basis in accordance with the Inservice Testing Program. This more frequent testing, although it does not test the actual initiation signal, verifies the operability of the majority of the SWS circuitry. The licensee also reviewed the surveillance test history of the SWS and confirmed that there are no failures that would invalidate this conclusion.

The staff reviewed the information presented by the licensee and concluded that the proposed change does not have a significant effect on safety and follows the guidance of GL 91-04. Therefore, the proposed change to ITS 3.7.2.5 is acceptable.

ITS 3.7.3.4 CONTROL ROOM EMERGENCY VENTILATION SYSTEM FUNCTIONAL TESTS

CTS 4.7.2.d.2 and CTS 4.7.2.d.4 require that functional tests be performed on the control room emergency ventilation (CREV) system every 18 months. In ITS SR 3.7.3.4, the frequency for the CREV system functional tests is changed to once every 24 months. The licensee therefore requested to increase the surveillance test intervals of this SR from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period. This SR is required to verify the operability of the CREV system actuation circuitry to ensure the integrity of the control room envelope.

The licensee evaluated the change in test frequency and concluded that the impact of this change, if any, on system availability is minimal. This conclusion is based on the fact that the CREV system circuitry is designed to meet the requirements of IEEE 279 and to be single failure proof such that no single circuit fault can prevent the safety system actuation. Furthermore, the components and dampers associated with the CREV system are tested on a more frequent basis. This more frequent testing, although it does not test the actual initiation signal, verifies the operability of the majority of the CREV system circuitry. The licensee also reviewed the plant historical test data for this SR and confirmed that there are no failures that would invalidate this conclusion.

The staff reviewed the information presented by the licensee and concluded that the proposed changes to the surveillance tests interval do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes to ITS SR 3.7.3.4 are acceptable.

ITS SR 3.7.3.2 CREV FILTERS TESTS

ITS SR 3.7.3.2 requires performing CREV filters testing in accordance with ITS 5.5.7, "ventilation filter testing program" to determine the CREV system operability. CTS 4.7.2.b requires that the CREV system filters testing be performed at least once every 18 months. ITS 5.5.7 requires that the CREV filters testing be performed once every 24 months; and after each completion or partial replacement of the HEPA filter bank or charcoal absorber filter bank; after any structural maintenance on the HEPA filter or charcoal absorber housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the system. The licensee therefore requested to increase the surveillance test intervals for this SR from 18 months to 24 months for a maximum interval of 30 months including the 25% grace period. This SR is required to ensure the capability of the CREV system charcoal absorbers and HEPA filters to perform their safety function.

The licensee evaluated the change in test frequency and concluded that the impact on system availability, if any, as a result of this change is minimal. This conclusion is based on the fact that the CREV system is normally in standby and its active components and power supplies are designed with redundancy to meet the single active failure criteria, which ensure system availability in the event of a failure of one of the system components. ITS 5.5.7.c requires testing the CREV system every 24 months; and after any structural maintenance on the HEPA filter or charcoal absorber housings or following painting, fire, or chemical release in any ventilation zone communicating with the CREV system. By testing after filters maintenance, fire, chemical release, painting, HEPA replacement; potential changes in HEPA filter efficiency and carbon absorber bypass leakage will be detected prior to the detection by conducting the 18-month surveillance tests. Furthermore, the CREV system active components and power supplies are designed with redundancy, which will ensure system availability in the event of a component failure. The licensee also reviewed the surveillance test history for the system and confirmed that there are no failures that would invalidate this conclusion.

The staff reviewed the information presented by the licensee and concluded that the proposed changes to the test intervals for the CREV system filters do not have a significant effect on safety and follow the guidance of GL 91-04. Therefore, the proposed changes in ITS 5.5.7 regarding CREV system filters test frequency are acceptable.

CONCLUSION

Based on the above discussion, the staff concludes that the proposed technical specification changes regarding the surveillance interval increase from 18 to 24 months (30 months with grace period) for testing the service water system and control room emergency ventilation system and its filters, as proposed in the licensee's submittal, are acceptable and the licensee's evaluations for these changes are developed within the guidelines of Generic Letter 91-04. Therefore, the staff finds acceptable the licensee's proposed increase in surveillance intervals for these ITS to accommodate a 24-month fuel cycle.

EXTENSION OF SURVEILLANCE FREQUENCIES FROM A 18-MONTH TO A 24-MONTH REFUELING INTERVAL (Part 5 of 7) - CONTAINMENT SYSTEMS AREA

By application dated November 1, 1996, CP&L requested changes to the BSEP Units 1 and 2 Technical Specifications (TS). Application included proposed changes necessary to permit 24-month operating cycles. TS Section 3/4.6 specifies containment systems operability and surveillance requirements. This evaluation addresses each of the proposed changes to TS Section 3/4.6 associated with 24-month cycles.

VACUUM BREAKER BYPASS LEAKAGE TEST INTERVAL

Proposed change: Current Technical Specification (CTS) 4.6.2.1.e.2 requires that drywell-to-suppression chamber vacuum breaker leakage be verified to be within limits every 18 months. In Improved Technical Specification (ITS) SR 3.6.1.1.2, the frequency is changed to once every 24 months. The surveillance interval of the vacuum breaker SR would be increased from once every 18 months to once every 24 months for a maximum interval of 30 months, including the 25% grace period.

Licensee's Justification: The licensee's justification is provided in Volume 9 of an enclosure to the application. The drywell vacuum breakers are the sole active components which constitute a potential suppression pool bypass pathway. The test provides interim assurance that the pressure suppression function would not be compromised by bypass steam leakage which would not be directed through the suppression pool water. It is not practical to perform this test during power operation. The vacuum breakers are also functionally (stroke) tested on a more frequent basis by ITS SR 3.6.1.6.2 to ensure their operability. In addition, ITS SR 3.6.1.6.1 verifies the vacuum breakers are closed every 14 days. Although the more operability frequent tests do not directly ensure the leak tightness of the drywell to suppression chamber vacuum breakers, they do ensure the valves are functional and closed. Based on the passive design of the suppression chamber-to-drywell vacuum breakers and the more frequent functional testing of the drywell-to-suppression chamber vacuum breakers, the impact, if any, from this change on component and system availability is small. A review of the surveillance test history was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small.

Staff Evaluation: In Generic Letter 91-04, the staff issued guidance to licensees for preparation of amendment requests for 24-month operating cycles. The generic letter stated:
Technical Specifications (TS) that specify an 18-month surveillance interval could be changed to state that these surveillances are to be performed once per refueling interval. The notation for surveillance intervals would then be changed to include the definition of a "Refueling Interval" with the existing "R" notation for surveillances that are generally performed during a refueling outage. The frequency for the interval indicated by this notation would also be changed from 18 months to "At least once every 24 months." The provision to extend surveillances by 25 percent of the specified interval would extend the time limit for completing these surveillances from the existing limit of 22.5 months to a maximum of 30 months.

Enclosure 1 of GL 91-04 stated:

> The NRC staff has reviewed a number of requests to extend 18-month surveillances to the end of a fuel cycle and a few requests for changes in surveillance intervals to accommodate a 24-month fuel cycle. The staff has found that the effect on safety is

> small because safety systems use redundant electrical and mechanical components and because licensees perform other surveillances during plant operation that confirm that these systems and components can perform their safety functions. Nevertheless, licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small. Licensees should confirm that historical plant maintenance and surveillance data support this conclusion. Also, licensees should confirm that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle. In consideration of these confirmations, the licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components.

As indicated by the generic letter, the staff has made a generic determination that the effect of the 18-to-24-month extension on safety is small and that it is therefore technically acceptable to extend 18-month test intervals to 24 months subject to a licensee confirmation that no licensing basis assumptions would be invalidated. The licensee has indicated that the proposed TS changes for 24-month cycles have been evaluated for conformance to the facility design basis and that appropriate FSAR update changes will be made where necessary and are identified in the "discussion of changes" sections of the application. No such changes were identified for the vacuum breakers.

The licensee has performed the required evaluations and provided the required confirmations for compliance with the GL 91-04 criteria to extend the vacuum breaker surveillance interval to 24 months. The proposed change is therefore acceptable.

CONTAINMENT ISOLATION VALVE FUNCTIONAL TEST INTERVAL

Proposed change: Current Technical Specifications 4.6.3.2 and 4.6.6.2.b.2 specify the frequency for primary containment isolation valve functional testing as once every 18 months. The interval of this SR is being increased in ITS SR 3.6.1.3.6 from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

License's Justification: This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. Some PCIVs are stroke tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The stroke testing of these PCIVs tests a significant portion of the PCIV's circuitry and will detect failures of this circuitry. The PCIVs, including the actuating logic, are designed to be single failure proof and therefore, are highly reliable. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station surveillance intervals from 18 to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." Based on the above discussion the impact, if any, of this change on system availability is small.

A review of the surveillance test history was performed to validate the above conclusion. This review of the surveillance test history, documented in "24- Month Surveillance Test History Study," demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small from a change to Current Technical Specification 4.6.3.2 as implemented in the proposed SR 3.6.1.3.6.

Staff Evaluation: As noted above, the licensee performed an evaluation of the surveillance test history and concluded that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

EXCESS FLOW CHECK VALVE TEST FREQUENCY

Proposed Change: Current Technical Specification 4.6.3.4 specifies the frequency for excess flow check valve (EFCV) testing as once every 18 months. The Surveillance Test Interval of this SR would be increased in ITS SR 3.6.1.3.7 from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that each EFCV will check excess flow to provide assurance that predicted radiological consequences will not be exceeded during a postulated instrument line break. EFCVs are designed with a restricting orifice, that, in the event of the failure of the check valve to function, will significantly limit line flow during a postulated instrument line break. Furthermore, instrument lines are seismically mounted and evaluated to withstand an design basis seismic event. Based on the design of the EFCV and the design of the instrumentation line tubing, the impact, if any, of this change on system availability is small.

A review of the surveillance test history was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history and found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

MSIV LEAKAGE TEST FREQUENCY

Proposed Change: Current Technical Specification 4.6.1.2.2 requires that the Main Steam Isolation Valves (MSIVs) be leak tested every 18 months. In ITS 3.6.1.3.9, the frequency is now defined as in accordance with the Primary Containment Leakage Rate Testing Program. With the change to a 24-month operating cycle, the surveillance interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that the MSIVs are capable of maintaining an essentially leak tight barrier. The MSIVs were designed and tested for closure in the event of a main steam line break and to provide an isolation barrier to maintain primary containment following a design basis accident. As such, the valves were designed to close during emergency steam flow conditions following rupture of the main steam line downstream of the valve. Furthermore, the valves were designed for the limiting system pressure and temperature. Finally, BNP is designed with two MSIVs per main steam line to ensure no active single failure will result in a loss of main steam line isolation capability. Based on the redundant design of the MSIVs, the impact, if any, from this change on component and system availability is small.

A review of the surveillance test history was performed to validate the above conclusion. This review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small.

Staff Evaluation: As described in the licensee's justification above, the licensee performed an evaluation of the surveillance test history and found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

DRYWELL VACUUM BREAKER SET POINT

Proposed Change: Current Technical Specification 4.6.4.1.d.1 requires that the suppression chamber-to-drywell vacuum breaker opening set point be verified every 18 months. In the proposed ITS SR 3.6.1.6.3, the frequency is specified as once every 24 months. The Surveillance Test Interval of this SR would thus be increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that each suppression chamber-to-drywell vacuum breaker is capable of performing its safety function assumed in the safety analysis assumptions. Proposed SR 3.6.1.6.2 requires that each vacuum breaker must be functionally tested by cycling to ensure that it opens adequately to perform its design function and returns to the fully closed position. This more frequent test performed during the operating cycle, although not ensuring the specified set point, does ensure that the vacuum breaker is capable of being cycled open and shut. Based on the more frequent testing the impact, if any, of this change on system availability is small.

A review of the surveillance test history was performed to validate the above conclusion. This review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

REACTOR BUILDING TO SUPPRESSION CHAMBER VACUUM BREAKERS SET POINT

Proposed Change: Current Technical Specification 4.6.4.2.1.b.1 requires that the reactor building-to-suppression chamber vacuum breaker check valve opening set point be verified every 18 months. CTS 4.6.4.2.1.b.2 requires that the reactor building-to-suppression chamber vacuum breaker butterfly valve opening set point be verified every 18 months. In the proposed ITS SR 3.6.1.5.4, the frequency of these surveillances is specified as once every 24 months. The surveillance interval of this SR would thus be increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that each reactor building-to-suppression chamber vacuum breaker check valve and vacuum breaker butterfly valve is capable of performing its safety function as assumed in the safety analysis assumptions. ITS SR 3.6.1.5.3 requires that each vacuum breaker must be functionally tested once per 92 days by cycling each reactor building-to-suppression chamber vacuum breaker check valve and vacuum breaker butterfly valve to ensure that it opens adequately to perform its design function and returns to the fully closed position. This more frequent test performed during the operating cycle, although not ensuring the specified set point, does ensure that the vacuum breaker check valve and vacuum breaker butterfly valves are capable of being cycled open and shut. Furthermore, the vacuum relief system design for the active components provides two 100% redundant relief paths. Therefore, based on the more frequent testing and the design of the vacuum relief system, the impact, if any, of this change on system availability is small.

A review of the surveillance test history was performed to validate the above conclusion. This review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact on system availability, if any, is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

NITROGEN BACKUP SYSTEM LEAKAGE RATE TEST INTERVAL

Proposed Change: Current Technical Specification 4.6.4.2.2.b requires that each nitrogen backup system be demonstrated OPERABLE by verifying that each subsystem leakage rate is less than or equal to the specified limit. In ITS SR 3.6.1.5.5, the frequency of this SR is specified as once every 24 months. The surveillance interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that reactor building-to-suppression chamber vacuum breaker butterfly valves are capable of performing their safety function as assumed in the safety analysis. ITS 3.6.1.5.1 requires verification that the nitrogen bottle supply of each subsystem is maintained at the required pressure once per 24 hours. Although this SR does not fully verify that the nitrogen supply system will perform its safety function, it does ensure minimum system pressure is maintained and that any significant degradation of the system will be detected. Furthermore, the vacuum relief system design for the active components provides two 100% redundant relief paths. Therefore, based on the more frequent testing and the design of the vacuum relief system, the impact, if any, of this change on system availability will be small as a result of the change in Surveillance Frequency.

A review of the surveillance test history was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact of this change, if any, on system availability is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

NITROGEN BACKUP SYSTEM FUNCTIONAL TEST INTERVAL

Proposed Change: Current Technical Specification 4.6.4.2.2.c specifies the Frequency for the system functional test of the Nitrogen Backup System as once every 18 months. In ITS SR 3.6.1.5.6, the Frequency of this SR is specified as once every 24 months. The surveillance interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: This SR ensures that the Nitrogen Backup System will perform its safety function (supplying motive force to the reactor building-to-suppression chamber vacuum breaker butterfly valves). The Nitrogen Backup System and its actuating logic are designed to be single failure proof and therefore, are highly reliable. Furthermore, Industry reliability studies for BWRs prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability. Based on the above, the impact, if any, of this change on system availability will be small.

A review of the surveillance test history was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact of this change, if any, on system availability is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

SECONDARY CONTAINMENT LEAKAGE TEST INTERVAL

Proposed Change: Current Technical Specification 4.6.5.1.b establishes 18 months as the required Frequency for performance of SRs that verify that the secondary containment internal pressure can be drawn down and maintained at the required vacuum. In ITS SR 3.6.4.1.3 the interval for these would be increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: The purpose of these tests is to ensure secondary containment boundary integrity by demonstrating that secondary containment vacuum assumed in the safety analysis can be maintained under design basis conditions. Extending the surveillance interval for this verification of secondary containment integrity is acceptable because secondary containment is maintained at a negative pressure during normal operation, and secondary containment structural integrity is maintained through administrative controls which ensure that no significant changes will be made to the secondary containment structure without proper evaluation. Any event which would cause significant structural degradation, such as seismic event, would require a plant evaluation. Therefore, based on the above, the impact, if any, on system availability will be small as a result of the change in Surveillance Frequency.

A review of the surveillance test history for each of these Surveillance requirements was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact of this change, if any, on system availability is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

SECONDARY CONTAINMENT ISOLATION DAMPER FUNCTIONAL TEST FREQUENCY

Proposed Change: Current Technical Specifications 4.6.5.2.c.1 and 4.6.5.2.c.2 establishes 18 months as the frequency for SRs that verify that on a containment isolation test signal each isolation damper actuates to its isolation position within the required time. In ITS SR 3.6.4.2.1 and SR 3.6.4.2.2 the interval of these SRs would be increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: Secondary containment isolation dampers (SCIDs) are stroke tested on a more frequent basis during the operating cycle in accordance with the Inservice Testing Program. The stroke testing of these SCIDs test a significant portion of the SCIDs circuitry and will detect failures of this circuitry and degradation of the SCIDs. The SCIDs, including the actuating logic, are designed to be single failure proof and therefore, are highly reliable. Furthermore, it is stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Units 2 and 3 surveillance intervals from 18 months to 24 months: "Industry reliability studies for boiling water reactors (BWRs), prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability." Based on the above discussion the impact, if any of this change on system availability is small.

A review of the surveillance test history for each of these Surveillance requirements was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact of this change, if any, on system availability is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable

STANDBY GAS TREATMENT (SGT) SYSTEM FUNCTIONAL TEST FREQUENCY

Proposed Change: Current Technical Specifications 4.6.6.1.d.2 establishes 18 months as the required Frequency for performance of the SR that verifies that each SGT filter train starts and that the associated dampers open on an initiation signal. ITS SR 3.6.4.3.3 performs the same test but the required Frequency is extended to 24 months. Therefore, the surveillance interval of this SR is being increased from once every 18 months to once every 24 months for a maximum interval of 30 months including the 25% grace period.

Licensee's Justification: Extending the surveillance interval for this verification is acceptable because the system is operated every 31 days to satisfy the requirements of ITS 3.6.4.3.1. This test will detect significant failures affecting system operation that would be detected by conducting the 18 month surveillance test. In addition, the SGT System active components and power supplies are designed with redundancy to meet the single active failure criteria, which will ensure system availability in the event of a failure of one of the system components. Based on the above discussion, the impact, if any of this change on system availability will be small.

A review of the surveillance test history for each of these Surveillance requirements was performed to validate the above conclusion. This historical review of the surveillance test history demonstrates that there are no failures that would invalidate the conclusion that the impact of this change, if any, on system availability is small.

Staff Evaluation: As noted in the licensee's justification above, the licensee performed an evaluation of the surveillance test history that found that there were no failures that would invalidate the conclusion that the impact on system availability, if any, is small. Also, the design basis review did not indicate any inconsistencies with the proposed amendment. Thus, there are no plant-specific circumstances that preclude 24-month intervals. Therefore, based on compliance with the GL 91-04 criteria, the proposed change is acceptable.

EXTENSION OF SURVEILLANCE INTERVAL FOR CONDENSER VACUUM PUMP ISOLATION (Part 6 of 7)

By letter dated October 13, 1997 (Revision A), the Carolina Power and Light Company proposed a revision to the Brunswick 1 and 2 Technical Specifications (TS) to extend the test surveillance interval from 18 to 24 months for the functional test for condenser vacuum pump isolation on high main steam line radiation.

The condenser vacuum pump isolation instrumentation initiates a trip of the two condenser vacuum pumps and closure of the associated line isolation valve following events in which the main steam line radiation monitors exceed a predetermined value.

The licensee evaluated the proposed TS changes in accordance with the guidance in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24 Month Fuel Cycle, dated April 2, 1991, and the licensee's historical maintenance and surveillance data support the extended intervals of 24 months. The licensee also stated that if the as-found conditions are outside allowable limits, an evaluation will be performed to determine whether the assumptions made to extend the calibration frequency are still valid and will evaluate the effects on plant safety.

Based on the discussion above, the staff concludes that the proposed changes to extend the functional test for the condenser vacuum pump isolation on high main steam line radiation are consistent with the guidance of GL 91-04 for 24 month fuel cycles, and are, therefore, acceptable.

REVISED SETPOINTS AND EXTENSION OF SURVEILLANCE FREQUENCIES FROM A 18-MONTH TO A 24-MONTH REFUELING INTERVAL (Part 7 of 7) - ELECTRICAL AREA

NRC GL 91-04, "Change in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," has provided guidance to all holders of operating licenses of nuclear power reactors requesting changes to surveillance intervals to accommodate a 24-month fuel cycle. The basis for the longer cycle is that licensees are planning to use improved reactor fuels because of the significant economic benefits. A longer fuel cycle increases the interval between refueling outages and can impact the performance of technical specification (TS) surveillance requirements (SRs).

CP&L has requested that the TS at Brunswick be converted following the guidance in GL 94-01 and the generic ITS for a BWR/4 (NUREG-1433). The EELB was requested to evaluate several specific changes that were considered outside the scope of the review for the conversion to the ITS.

The EELB has reviewed the proposed changes and finds them acceptable as discussed in the following evaluation.

The licensee is proposing the following changes based on guidance in the ITS.
Change 1: This change revises BNP's Current Technical Specification (CTS) Table 3.3.3-2, "Emergency Core Cooling System Actuation Instrumentation Setpoints," Trip Functions 3a and 3b. This change will eliminate the trip setpoints (TSPs) for the 4.16kV Emergency Undervoltage-Loss of Voltage and Degraded Voltage Functions (Loss of Power (LOP) Instrumentation) and leave only the Allowable Values, which is consistent with the ITS. Also a revised Allowable Value has been proposed for the Loss of Voltage Function.

Change 2: This change will revise BNP's CTS 4.8.2.5, "Reactor Protection System Electric Power Monitoring," trip setpoints. The change will use the current trip setpoints in the TS as the Allowable Values and thereby the TS will contain only Allowable Values.

The staff discussed the current TSP methodology with the licensee and has concluded that the TSP methodology provides adequate margin between the TSPs, Allowable Values and safety analysis limits at BNP. Additionally, the staff concludes that plant calibration procedures will ensure adequate control of the TSPs and the assumptions regarding accuracy, measurement and test equipment accuracy, and setting tolerance will be maintained within acceptable limits. Finally, the changes are considered consistent with the ITS. Based on the above, the staff finds the proposed changes acceptable.

The licensee is proposing the following changes based on the guidance in NRC GL 94-01:

Change 3: CTS 4.3.3-1 and CTS 4.3.3.2 established once every refueling outage (18 months) as the required frequency for channel calibration for the LOP instrumentation. The licensee is proposing to extend the SR for the Degraded Voltage Function to 24 months. The channel calibration interval for the Loss of Voltage Function will remain at 18 months due to drift considerations.

Change 4: CTS 4.3.3.2 requires the LOP Instrumentation Logic System Functional Test (LSFT) once every 18 months. The licensee is proposing to extend this SR to 24 months.

Change 5: CTS 4.8.2.5.b requires the Reactor Protection System (RPS) Electric Power Monitoring instrumentation simulated automatic actuation test every 18 months. The licensee is proposing to extend this SR to 24 months.

Change 6: CTS 4.8.2.5.b establishes 18 months as the channel calibration frequency for the RPS Electric Power Monitoring instrumentation. The licensee is proposing to extend this SR to 24 months.

NRC GL 91-04 has provided guidance to licensees for requesting changes to surveillance intervals related to 24-month fuel cycle extensions. A longer fuel cycle increases the interval between refueling outages and can impact the performance of TS surveillance requirements. The staff has required that licensees evaluate the effect on safety when requesting surveillance intervals that correspond to a 24-month fuel cycle. Those evaluations should support a conclusion that the effect on safety is small with confirmation that historical maintenance and surveillance data does not invalidate that conclusion. Also, licensees must address instrument drift when proposing an increase in the surveillance interval for calibration of instrumentation that performs safety functions, including those instruments that provide the capability for a safe plant shutdown.

CP&L has reviewed surveillance test history of the LOP and RPS Electric Monitoring instrumentation. The historical review of the surveillance test data demonstrated that there were no failures that invalidated the conclusion that the impact on system availability was minimal as a result of a 24-month refueling interval. Additionally, the licensee has evaluated the impact of the extended instrumentation calibration interval on associated setpoint calculations and concluded that the drift assumptions remained valid.

Additionally, the extension of the surveillance intervals is further supported by industry reliability studies which show that overall unreliability of safety systems is not dominated by the failure of relays and their contacts but by components such as pumps and valves which are consequently tested more often. Based on the above, the staff finds the proposed changes acceptable.

CHANGES IN CERTAIN ALLOWABLE VALUES, ALLOWED OUTAGE TIMES, AND SURVEILLANCE FREQUENCIES - INSTRUMENT AND CONTROLS AREA

By letters dated November 1, 1996, October 13, 1997, October 28, 1997, February 26, 1998, and March 13, 1998, Carolina Power and Light Company requested an amendment to revise certain instrumentation allowable values, allowed outage times, and surveillance frequencies in the Technical Specifications (TS) for the Brunswick Steam Electric Plant Unit Nos. 1 and 2.

The licensee is in the process of converting the current Technical Specifications (CTS) to the improved Standard Technical Specifications format. In support of this effort, the licensee has proposed the revision of certain instrumentation allowable values, allowed outage times, and surveillance frequencies contained in the CTS.

The proposed allowable values are based on uncertainties associated with the entire loop of the instrumentation circuitry (trip unit and sensor) and were calculated in accordance with the licensee's setpoint methodology described in Design Guide VIII.0050, "Instrument Setpoints." The staff has previously reviewed and accepted this setpoint methodology for other Brunswick allowable values.

The proposed allowable values were calculated by applying calibration based errors to the trip setpoints, thereby establishing an operability limit associated with the entire loop of each instrumentation function. The proposed allowable value changes are within the analytical limit for each function and do not affect the existing margins between operating conditions and reactor trip setpoints. Therefore, the proposed allowable value changes do not affect the existing licensing basis, and are, therefore, acceptable.

The proposed surveillance frequency changes involve an extension of instrumentation test intervals to support 24 month operating cycles. The surveillance frequency and allowed outage time changes are based on NEDC-30936P-A, "BWR Owners Group Technical Specification Improvement Methodology With Demonstration for BWR ECCS Actuation Instrumentation," NEDC-31336, "GE Instrument Setpoint Methodology," and "General Electric Instrument Trending Analysis System" (GEITAS). NEDC-30936P-A and NEDC-31336 have been previously approved by the staff and accepted as a basis for other Brunswick surveillance frequency and allowed outage time revisions. GEITAS has been successfully used to determine drift for other plants that have received approval for extension of channel calibration surveillance requirements to 24 months and other Brunswick surveillance frequency revisions. Based on discussions with the staff, the licensee clarified in a letter dated May 22, 1998 (Revision E), the applicability of NEDC-31336P. Therefore, the proposed allowed outage time and surveillance frequency changes are acceptable.

Based on the above evaluation, the staff concludes that the proposed changes in instrumentation allowable values, allowed outage times, and surveillance frequencies incorporated in the TSs are consistent with approved topical reports, the licensee's setpoint methodology, and licensing basis, and are, therefore, acceptable.

EXTENSION OF ROD WORTH MINIMIZER SURVEILLANCE FREQUENCY

Brunswick current technical specifications (CTS) require a channel functional test to be performed prior to withdrawal of control rods for the purpose of making the reactor critical and when the RWM is initiated during a plant shutdown. Improved Technical Specifications (ITS) surveillance requirements (SRs) 3.3.2.1.2 and 3.3.2.1.3 are similar to CTS 4.1.4.1.1 except a test frequency is specified as 92 days. This change effectively extends the channel functional test to 92 days, i.e the channel test is not required to be performed if a start up or shutdown occurs within 92 days of a previous start up or shutdown. If maintenance is performed on the RWM during this 92 day period, it is required that testing be performed before the system is returned to service. Testing prior to each start up and shutdown increases the wear on the instruments, thereby reducing the reliability of the instruments. The effect on safety due to the extended surveillance is small. This change was previously approved for Peach Bottom. We agree with the licensee that surveillance beyond the quarterly surveillance is not required to assure the instruments safety function. The proposed extended surveillance frequency of 92 days is acceptable.

EXTENSION OF SERVICE WATER ALLOWED OUTAGE TIMES

By letter dated November 1, 1996, as supplemented by letters dated October 13, 1997, February 26, 1998, March 13, 1998, April 24, 1998, and May 22, 1998, Carolina Power and Light Company (the licensee) proposed changes to the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, for the conversion to the Improved Technical Specifications (ITS). The proposed changes include extension of certain allowed outage times (AOTs) for certain combinations of operable service water pumps. This proposed extension of AOTs is beyond the scope of the ITS conversion process and must be considered separately. This evaluation documents the staff's review in this regard.

The BSEP service water system (SWS) provides water for cooling selected systems and components. The SWS serving each unit is subdivided into two major headers, the nuclear service water (NSW) header and the conventional service water (CSW) header, which are normally operated independently. Two NSW pumps for each unit provide cooling water to the nuclear header, while three CSW pumps for each unit provide cooling water to the conventional header. While the CSW pumps are normally aligned to supply cooling water to the conventional header, each of the CSW pumps can be individually aligned to provide cooling water to either the conventional header or to the nuclear header; the NSW pumps can only be aligned to provide cooling water to the nuclear header. The emergency diesel generators (EDGs) can only receive cooling water from the nuclear header, but other safety-related heat loads can be aligned to either the nuclear header or the conventional header. One NSW pump and CSW pump for each unit receive electrical power from one of the unit emergency buses, while the second NSW pump and CSW pump receive electrical power from the other unit emergency bus. The third CSW pump for each unit receives electrical power from one of the emergency buses of the other unit. The motor operated valve that supplies service water from the NSW header to the vital service header and the motor operated valve that supplies service water to the residual heat removal service water (RHRSW) suction header are each powered by the same emergency bus, while the two motor operated valves that supply service water for these functions from the CSW header are powered from the other emergency bus. Consequently, the CSW header and NSW header can become inoperable (depending on the status of the required power supplies) irrespective of whether the required number of CSW and NSW pumps are operable.

During a loss-of-coolant accident (LOCA) and/or a loss of offsite power (LOOP) event, the operable EDGs and NSW pumps associated with both units automatically start to provide water via the nuclear header for cooling the EDGs and the emergency core cooling system (ECCS) heat loads. Each EDG has a pressure switch for detecting low service water supply pressure and if low service water pressure is sensed, service water supply is automatically transferred to the nuclear header of the other unit. After 10 minutes into the event, remote operator action (from the control room) is credited for starting the CSW pumps and for making valve alignments to provide cooling water for long-term decay heat removal. Single failure scenarios and flow diversion possibilities can be quite complex due to various pump and system alignment possibilities and power supply considerations.

Current requirements for maintaining operability of the service water system for the Brunswick units are contained in TS Section 3.7.1.2. Three NSW pumps serving the site (any combination of Unit 1 and Unit 2 NSW pumps) and two CSW pumps serving each unit must be operable in operational conditions 1, 2, and 3. For a site NSW pump to be operable, it must be capable of supplying its associated unit NSW header and for a CSW pump to be considered operable, it must be capable of supplying both the CSW and NSW headers. From this initial configuration, assuming a single failure, at least two NSW pumps will remain operable to provide cooling water for the four EDGs, and at least one CSW pump for each unit will remain operable to support long-term decay heat removal requirements. The TS requirements include AOTs to allow for situations when the total number of required NSW pumps and CSW pumps are not operable. The licensee has requested an extension of the AOTs for certain combinations of operable service water pumps to allow 72 hour AOTs in place of the shutdown actions that are currently required, and a 72 hour AOT is proposed for a new pump combination that was not previously addressed by the existing service water TS requirements. The licensee's request is for specific NSW pump and CSW pump combinations where the SWS (including required flow paths, pumps, valves, and emergency power supplies) is still able to satisfy its design-basis cooling functions for all postulated scenarios, as long as single failure considerations are not imposed. The specific TS requirements that are proposed by the licensee include the following:

a. TS requirement 3.7.1.2, Action a.1.a, currently requires a plant shutdown in the event one required CSW pump is inoperable and the subject unit NSW pump is not powered from a separate emergency bus from the operable CSW pump. For this situation, ITS requirement 3.7.2, Action D.1, would allow continued operation for 72 hours.

b. TS requirement 3.7.1.2, Action a.2.a, currently requires a plant shutdown in the event the required CSW pumps for a unit are inoperable and two of the three operable NSW pumps are not associated with the affected unit. For this situation, ITS requirement 3.7.2, Action E.1, would allow continued operation for 72 hours.

c. TS requirement 3.7.1.2, Action a.4.a, currently requires a plant shutdown in the event that one required CSW pump and one required NSW pump are inoperable, if none of the operable NSW pumps are associated with the affected unit. For this situation, ITS requirement 3.7.2, Actions F.1 and F.2, would allow continued operation for 72 hours.

d. TS requirement 3.7.1.2, Action a.4.b, currently requires a plant shutdown in the event that one required CSW pump and one required NSW pump are not operable, if the remaining operable CSW pump and the remaining operable affected unit NSW pump are powered by the same emergency bus. For this situation, ITS requirement 3.7.2, Actions F.1 and F.2, would allow continued operation for 72 hours.

e. ITS requirement 3.7.2, Actions G.1, G.2.1, and G.2.2, are proposed to allow continued operation for 72 hours in the event that none of the required CSW pumps for an affected unit are operable at the same time that one of the required NSW pumps is inoperable, if the two remaining operable NSW pumps are associated with the affected unit. Since this condition is not specifically addressed by the existing service water TS requirements, TS requirement 3.0.3 would require a plant shutdown for this situation.

Through analysis and computer modeling, the licensee has determined that adequate cooling water is provided to ECCS loads and to two of the four EDGs by one NSW pump during a design-basis accident, and that one NSW pump can provide the required cooling capacity to support all four EDGs and the subject unit vital header loads (including allowances for leakage across the isolation valves between the nuclear and conventional header piping when one of the two headers is depressurized). The licensee's analysis was included in the staff's review associated with TS Amendments 164 (Unit 1) and 195 (Unit 2), which was completed in August 1993, and additional review of the licensee's analysis in conjunction with this TS amendment request was not performed. However, the licensee's analysis may be subject to further NRC review and audit in the future as deemed necessary.

While a total of three NSW pumps for the site and two CSW pumps for each operating unit must be operable to account for potential component failures and flow diversions whenever a unit is operating in Modes 1, 2, or 3, the number of NSW and CSW pump combinations that are required to be operable can be relaxed for 72 hours if the SWS can continue to perform its design-basis functions in the absence of single failure considerations. This is consistent with the staff's policy and past practices relative to AOTs. The staff reviewed the specific NSW and CSW pump combinations for which the licensee proposes to allow 72 hour AOTs (as discussed in the background section, above), and the staff has determined that the proposed pump combinations will continue to satisfy the SWS design-basis functions as long as single failure considerations are not imposed. Therefore, the licensee's proposal to allow 72 hour AOTs for the specific NSW and CSW pump combinations cited above is acceptable to the staff.

The staff has reviewed the licensee's request to allow 72 hour AOTs for certain combinations of operable NSW and CSW pumps. The NSW and CSW pump combinations that have been proposed will satisfy SWS design-basis flow requirements for heat removal as long as single failure considerations are not imposed. Consistent with staff policy and past practice, 72 hour AOTs are appropriate for this situation. Therefore, the licensee's request is acceptable to the staff.

RELOCATION OF MAIN STEAM TUNNEL HIGH TEMPERATURE ISOLATION INSTRUMENTATION

The proposed change would relocate the isolation actuation instrumentation for the main steam line tunnel and turbine building area temperatures to the Technical Requirements Manual (TRM).

The turbine building area temperature function and instruments of the main steam line tunnel temperature function that sense main steam line temperature outside the main steam isolation valve (MSIV) pit are provided to detect and initiate an MSIV isolation. However, these primary containment isolation instruments constitute only one method of determining steam leakage in their respective areas. In addition to temperature monitoring, excess coolant inventory loss is detected by the reactor vessel low water level functions and main steam line high flow functions which continue to be required by TS. The turbine building area temperature instrumentation is not assumed to mitigate any accident described in the BSEP UFSAR and the only main steam line tunnel temperature instruments assumed in the mitigation of analyzed events are the MSIV instruments that sense temperature in the MSIV pit.

Comparison to Screening Criteria:

1. The Main Steam Line Tunnel Temperature and Turbine Building Area Temperature Functions are not used for, nor capable of detecting, a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Main Steam Line Tunnel Temperature and Turbine Building Area Temperature Instrumentation are not process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission protect barrier.

3. The Turbine Building Area Temperature Instrumentation is not part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The main steam line tunnel temperature instruments that sense main steam line temperature outside the MSIV pit are not assumed as part of the primary success path that functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Credit is not taken in pressure-temperature analyses, radiation dose calculations, or equipment qualification for the operation of the Turbine Building Area Temperature Function or instruments of the Main Steam Line Tunnel Temperature Function that sense main steam line temperature outside the MSIV pit. In addition, adequate redundancy is available by other TS-required instruments to detect a steam leak in the main steam tunnel or the turbine building and perform the associated primary containment isolation function.

4. As discussed in Appendix B of the Application of Selection Criteria to the BNP Technical Specifications, CP&L found the requirements of the Turbine Building Area Temperature Function and instruments of the Main Steam Line Tunnel Temperature Function that sense main steam line temperature outside the MSIV pit not being met to be a non-significant risk contributor to core damage frequency and offsite releases.

In addition, adequate redundancy is available by other TS-required instruments to detect a steam leak in the main steam tunnel or the turbine building and perform the associated primary containment isolation function.

The isolation actuation instrumentation LCO and SRs associated with the turbine building area temperature function and instruments of the main steam line tunnel temperature function that sense line temperature outside the MSIV pit do not meet the criteria in 10 CFR 50.36. Therefore, in accordance with the NRC Final Policy Statement, these specifications are relocated out of the ITS. Any changes to these former requirements regarding the isolation actuation instrumentation, as relocated to the TRM, will require a safety evaluation pursuant to 10 CFR 50.59. Thus, sufficient regulatory controls exist to ensure continued protection of the public health and safety.

H. IMPLEMENTATION OF REACTOR STABILITY LONG-TERM SOLUTION

On November 1, 1996, the Carolina Power and Light Company (CP&L) requested amendments to modify the technical specifications for the Brunswick Steam Electric Plant, Units 1 and 2, to allow for transition to the long-term stability solution known as Enhanced Option I-A (I-A) (reference 1). Supplemental information was provided on December 18, 1997 (reference 2). This modification will be completed as part of the conversion to the standard technical specifications (TS) format. I-A consists of several modifications to the plant hardware and operating procedures that, taken as a whole, provide a means for reliably detecting and avoiding reactor instabilities which could challenge plant safety limits if left alone. The modification of the current plant TS consists of eliminating the procedures which are required to conform to the stability Interim Corrective Actions and inserting the procedures (including hardware surveillance requirements and operator actions) to implement I-A. Generic I-A TS were reviewed by the staff and approved as NEDO-32339, Supplement 4 (reference 3). Errata to this document (reference 4) were submitted October 20, 1997, and approved by the staff in reference 5. This Safety Evaluation will evaluate the adequacy of the implementation of the guidance in NEDO-32339, Supplement 4. Approval for use of the new TS pages will be granted when CP&L changes to the standard technical specifications.

The staff compared the proposed I-A TS pages presented in reference 1 for BSEP Units 1 and 2 with those approved in NEDO-32339, Supplement 4. The following sections were either modified or added:

a. Section 3.2.3, Fraction of Core Boiling Boundary, which includes LCO 3.2.3 and SR 3.2.3.1

b. Specification 3.3.1.1, Reactor Protection System Instrumentation, including Table 3.3.1.1-1

c. Specification 3.3.1.3, Period Based Detection System, which includes LCO 3.3.1.3 and SRs 3.3.1.3.1, 3.3.1.3.2, and 3.3.1.3.3

d. Section 5.6.5 relating to the Core Operating Limits Report (COLR) has been updated to add the following Enhanced Option I-A specific references.

1. NEDO-32339-A, "Reactor Stability Long Term Stability Solution: Enhanced Option I-A," July 1995.

2. NEDO-32339-P Supplement 1, "Reactor Stability Long Term Stability Solution: Enhanced Option I-A ODYSY Computer Program," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

3. NEDO-32339 Supplement 3, "Reactor Stability Long Term Stability Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

A discussion with justification of all of the above TS is included in reference 2. CP&L has appropriately modified their TS in accordance with the outline presented in reference 3.

The staff has concluded that CP&L has adequately implemented the proposed Technical Specifications necessary to implement Long Term Stability Solution Enhanced Option I-A.

IV. STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendment. The State official for the State of North Carolina had no comments.

V. ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on May 27, 1998 (63 FR 29039) for the ITS conversion.

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this ITS conversion amendment will not have a significant effect on the quality of the human environment.

With respect to other TS changes included in the application for conversion to improved Technical Specifications, the items change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 25103, 63 FR 6970, 63 FR 6971, 63 FR 6972, 63 FR 6973, 63 FR 6974, 63 FR 6975, 63 FR 6976, 63 FR 6979, 63 FR 6980, 63 FR 6981, 62 FR 8793). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

VI. CONCLUSION

The improved BSEP TS provide clearer, more readily understandable requirements to ensure safe operation of the plant. The NRC staff concludes that they satisfy the guidance in the Commission's policy statement with regard to the content of technical specifications, and conform to the model provided in NUREG-1433 with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the improved BSEP TS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36 and other applicable standards. On this basis, the NRC staff concludes that the proposed improved BSEP TS are acceptable.

The NRC staff has also reviewed the plant-specific changes to CTS as described in this evaluation. On the basis of the evaluations described herein for each of the changes, the NRC staff concludes that these changes are acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and, (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

VII. REFERENCES

1. Letter from William R. Campbell (CP&L) to USNRC, "Enhanced Option I-A Technical Specifications," November 1, 1996.

2. Letter from C. S. Hinnant (CP&L) to USNRC, "Supplemental Request for License Amendments: Enhanced Option I-A Stability Technical Specifications," December 18, 1997.

3. NEDO-32339-A, Supplement 4, "Reactor Stability Long Term Stability Solution: Enhanced Option I-A Generic Technical Specifications," December 1996.

4. Errata to NEDO-32339-A, Supplement 4, October 20, 1997.

5. SER for NEDO-32339, Revision 1, February 25, 1998.

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