Progress Energy

AUG 2 0 2003

SERIAL: BSEP 03-0126

U.S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
SUBMITTAL OF TECHNICAL SPECIFICATION BASES CHANGES FOR REVISION 30 (UNIT 1) AND 29 (UNTI 2)

Ladies and Gentlemen:

In accordance with Technical Specification (TS) 5.5.10 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, Progress Energy Carolinas, Inc. is submitting Revisions 30 and 29 to the BSEP, Unit 1 and 2 TS Bases, respectively. Revisions 30 and 29 were implemented on August 6, 2003.

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing/Regulatory Programs, at (910) 457-2073.

Sincerely,

Edward T. O'Neil
Manager - Support Services
Brunswick Steam Electric Plant

WRM/wrm

Enclosures:
1. Summary of Revisions to Technical Specification Bases
2. Page Replacement Instructions
3. Replacement Technical Specification Bases Pages

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Summary of Revisions to Technical Specification Bases			
Revision	**Affected Units**	**Date Implemented**	**Title/Description**
30 29	Unit 1 Unit 2	August 6, 2003	**Title:** Issuance of Amendment Regarding Pressure/Temperature Limit Curves **Description:** Revisions 30 and 29 incorporate the bases changes associated with Facility Operating License Amendments 228 (Unit 1) and 256 (Unit 2), issued June 18, 2003 (ADAMS Accession Number ML031690683).

Page Change Instructions

Unit 1 - Bases Book 1	
Remove	**Insert**
Title Page - Revision 29	Title Page - Revision 30
LOEP-1, Revision 29	LOEP-1, Revision 30

Unit 1 - Bases Book 2	
Remove	**Insert**
LOEP-1, Revision 29	LOEP-1, Revision 30
B 3.4-39, Revision 17	B 3.4-39, Revision 30
B 3.4-40, Revision 17	B 3.4-40, Revision 30
B 3.4-41, Revision 17	B 3.4-41, Revision 30
B 3.4-44, Revision 17	B 3.4-44, Revision 30
B 3.4-47, Revision 17	B 3.4-47, Revision 30

Unit 2 - Bases Book 1	
Remove	**Insert**
Title Page - Revision 28	Title Page - Revision 29
LOEP-1, Revision 28	LOEP-1, Revision 29

Unit 2 - Bases Book 2	
Remove	**Insert**
LOEP-1, Revision 27	LOEP-1, Revision 29
LOEP-2, Revision 27	LOEP-2, Revision 29
B 3.4-39, Revision 17	B 3.4-39, Revision 29
B 3.4-40, Revision 17	B 3.4-40, Revision 29
B 3.4-41, Revision 17	B 3.4-41, Revision 29
B 3.4-44, Revision 17	B 3.4-44, Revision 29
B 3.4-47, Revision 17	B 3.4-47, Revision 29
B 3.6-66, Revision 27	B 3.6-66, Revision 29
B 3.6-67, Revision 18	B 3.6-67, Revision 29
B 3.6-71, Revision 21	B 3.6-71, Revision 29

Replacement Technical Specification Bases Pages Revisions 30 (Unit 1) and 29 (Unit 2)

Unit 1
Bases Book 1
Replacement Pages

BASES

TO

THE FACILITY OPERATING LICENSE DPR-71

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 1

CAROLINA POWER & LIGHT COMPANY

REVISION 30

LIST OF EFFECTIVE PAGES - BASES

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List of Effective Pages - Book I	
LOEP-1	30
LOEP-2	25
LOEP-3	24
LOEP-4	27
i	21
ii	21
B 2.0-1	0
B 2.0-2	0
B 2.0-3	25
B 2.0-4	24
B 2.0-5	24
B 2.0-6	24
B 2.0-7	24
B 2.0-8	0
B 3.0-1	0
B 3.0-2	0
B 3.0-3	0
B 3.0-4	0
B 3.0-5	0
B 3.0-6	0
B 3.0-7	0
B 3.0-8	0
B 3.0-9	0
B 3.0-10	0
B 3.0-11	0
B 3.0-12	0
B 3.0-13	28
B 3.0-14	28
B 3.0-15	28
B 3.0-16	28
B 3.1-1	0
B 3.1-2	0
B 3.1-3	0
B 3.1-4	0
B 3.1-5	0
B 3.1-6	0
B 3.1-7	0
B 3.1-8	0
B 3.1-9	0
B 3.1-10	0
B 3.1-11	0
B 3.1-12	0
B 3.1-13	0
B 3.1-14	0
B 3.1-15	0
B 3.1-16	0
B 3.1-17	25
B 3.1-18	25
B 3.1-19	0
B 3.1-20	0
B 3.1-21	0
B 3.1-22	0
B 3.1-23	0
B 3.1-24	23
B 3.1-25	0
B 3.1-26	23
B 3.1-27	23
B 3.1-28	0
B 3.1-29	0
B 3.1-30	0
B 3.1-31	0
B 3.1-32	0
B 3.1-33	25
B 3.1-34	25
B 3.1-35	25
B 3.1-36	25
B 3.1-37	24
B 3.1-38	24
B 3.1-39	24
B 3.1-40	24
B 3.1-41	24
B 3.1-42	24
B 3.1-43	24
B 3.1-44	24
B 3.1-45	0
B 3.1-46	0
B 3.1-47	0
B 3.1-48	24

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Unit 1
Bases Book 2
Replacement Pages

LIST OF EFFECTIVE PAGES - BASES

Page No.	Revision No.	Page No.	Revision No.
List of Effective Pages - Book 2		B 3.4-33	0
		B 3.4-34	0
LOEP-1	30	B 3.4-35	0
LOEP-2	26	B 3.4-36	0
LOEP-3	25	B 3.4-37	0
LOEP-4	29	B 3.4-38	0
LOEP-5	25	B 3.4-39	30
		B 3.4-40	30
i	18	B 3.4-41	30
ii	7	B 3.4-42	17
		B 3.4-43	17
B 3.4-1	0	B 3.4-44	30
B 3.4-2	0	B 3.4-45	17
B 3.4-3	25	B 3.4-46	17
B 3.4-4	21	B 3.4-47	30
B 3.4-5	1	B 3.4-48	0
B 3.4-6	21	B 3.4-49	0
B 3.4-7	0		
B 3.4-8	0	B 3.5-1	0
B 3.4-9	0	B 3.5-2	0
B 3.4-10	0	B 3.5-3	0
B 3.4-11	0	B 3.5-4	0
B 3.4-12	0	B 3.5-5	0
B 3.4-13	0	B 3.5-6	0
B 3.4-14	9	B 3.5-7	0
B 3.4-15	0	B 3.5-8	0
B 3.4-16	0	B 3.5-9	0
B 3.4-17	0	B 3.5-10	0
B 3.4-18	0	B 3.5-11	0
B 3.4-19	0	B 3.5-12	0
B 3.4-20	19	B 3.5-13	0
B 3.4-21	19	B 3.5-14	0
B 3.4-22	19	B 3.5-15	0
B 3.4-23	19	B 3.5-16	9
B 3.4-24	19	B 3.5-17	0
B 3.4-25	24	B 3.5-18	0
B 3.4-26	24	B 3.5-19	0
B 3.4-27	24	B 3.5-20	0
B 3.4-28	24	B 3.5-21	0
B 3.4-29	0	B 3.5-22	0
B 3.4-30	0	B 3.5-23	0
B 3.4-31	0	B 3.5-24	0
B 3.4-32	0	B 3.5-25	0

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The criticality curve provides limits for both heatup and cooldown during criticality. Development of the curves considered instrument uncertainty values of 10°F for temperature and 15 psig for pressure plus an additional 15 psig for pressure instrument location (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel (including its appurtenances) is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel (including its appurtenances).

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section XI, Appendix G (Ref. 3).

(continued)

BASES

BACKGROUND (continued)

The P/T limit curves in this Specification were developed in accordance with the 1989 Edition of the ASME Code, Section XI, Appendix G (Ref. 3). These P/T limit curves were developed using the initiation fracture toughness, K_{IC}, for the allowable material fracture toughness. The use of K_{IC} for development of P/T limit curves has been approved by the ASME through Code Case N-640 (Ref. 4).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with the UFSAR (Ref. 5) and Appendix H of 10 CFR 50 (Ref. 6). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 7.

The bounding P/T limit curves are based on the N16 instrumentation nozzles. These nozzles are located at the top of the beltline region and were determined to be the limiting material with respect to the P/T curves.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 2 requirement that they be at least 40°F above the noncritical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 8), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 9 provides the curves and limits in this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are ≤ 100°F in any 1 hour period, during RCS heatup and cooldown;

b. RCS pressure and temperature are within the applicable limits in Figure 3.4.9-3, 3.4.9-4, or 3.4.9-5 and heatup or cooldown rates are ≤ 30°F in any 1 hour period, during RCS inservice leak and hydrostatic testing;

c. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup;

d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and

f. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified as safe by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored. With the applicable limits of Figures 3.4.9-3, 3.4.9-4, or 3.4.9-5 exceeded during inservice hydrostatic and leak testing operations, the maximum temperature change shall be limited to 10°F in any 1 hour period during restoration of the P/T limit parameters to within limits.

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 8), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8 (continued)

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.6 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature is ≤ 80°F in MODE 4. SR 3.4.9.8 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature is ≤ 100°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. Calculation 0B21-1029, "Instrument Uncertainty for RCS Pressure/Temperature Limits Curve," Revision 0.
2. 10 CFR 50, Appendix G.
3. 1989 Edition of the ASME Code, Section XI, Appendix G.
4. ASME Code Case N-640. "Alternate References Fracture Toughness for Development of P-T Limit Curves Section XI. Division 1."
5. UFSAR, Section 5.3.1.6 and Appendix 5.3B.
6. 10 CFR 50, Appendix H.
7. Regulatory Guide 1.99, Revision 2, May 1988.
8. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
9. Calculation 0B11-0005, "Development of RPV Pressure-Temperature Curves For BNP Units 1 and 2 For Up To 32 EFPY of Plant Operation," Revision 1.
10. 10 CFR 50.36(c)(2)(ii).

Unit 2
Bases Book 1
Replacement Pages

BASES

TO

THE FACILITY OPERATING LICENSE DPR-62

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 2

CAROLINA POWER & LIGHT COMPANY

REVISION 29

LIST OF EFFECTIVE PAGES - BASES

Page No.	Revision No.
Title Page	29
List of Effective Pages - Book 1	
LOEP-1	29
LOEP-2	27
LOEP-3	27
LOEP-4	27
i	27
ii	27
B 2.0-1	0
B 2.0-2	0
B 2.0-3	27
B 2.0-4	27
B 2.0-5	27
B 2.0-6	27
B 2.0-7	27
B 2.0-8	0
B 3.0-1	0
B 3.0-2	0
B 3.0-3	0
B 3.0-4	0
B 3.0-5	0
B 3.0-6	0
B 3.0-7	0
B 3.0-8	0
B 3.0-9	0
B 3.0-10	0
B 3.0-11	0
B 3.0-12	0
B 3.0-13	25
B 3.0-14	25
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B 3.0-16	25
B 3.1-1	0
B 3.1-2	0
B 3.1-3	0
B 3.1-4	0
B 3.1-5	0
B 3.1-6	0

Page No.	Revision No.
B 3.1-7	0
B 3.1-8	0
B 3.1-9	0
B 3.1-10	0
B 3.1-11	0
B 3.1-12	0
B 3.1-13	0
B 3.1-14	0
B 3.1-15	0
B 3.1-16	0
B 3.1-17	27
B 3.1-18	27
B 3.1-19	0
B 3.1-20	0
B 3.1-21	0
B 3.1-22	0
B 3.1-23	0
B 3.1-24	22
B 3.1-25	0
B 3.1-26	22
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B 3.1-28	0
B 3.1-29	0
B 3.1-30	0
B 3.1-31	0
B 3.1-32	0
B 3.1-33	27
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B 3.1-41	27
B 3.1-42	27
B 3.1-43	28
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B 3.1-45	0
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B 3.1-47	0
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Unit 2
Bases Book 2
Replacement Pages

(continued)

LIST OF EFFECTIVE PAGES - BASES (continued)

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B 3.5-26	0	B 3.6-39	18
B 3.5-27	0	B 3.6-40	18
B 3.5-28	0	B 3.6-41	18
B 3.5-29	0	B 3.6-42	18
B 3.5-30	0	B 3.6-43	18
		B 3.6-44	18
B 3.6-1	0	B 3.6-45	18
B 3.6-2	27	B 3.6-46	23
B 3.6-3	0	B 3.6-47	23
B 3.6-4	0	B 3.6-48	18
B 3.6-5	0	B 3.6-49	18
B 3.6-6	27	B 3.6-50	18
B 3.6-7	0	B 3.6-51	18
B 3.6-8	0	B 3.6-52	18
B 3.6-9	0	B 3.6-53	27
B 3.6-10	0	B 3.6-54	18
B 3.6-11	0	B 3.6-55	18
B 3.6-12	0	B 3.6-56	27
B 3.6-13	0	B 3.6-57	18
B 3.6-14	0	B 3.6-58	18
B 3.6-15	0	B 3.6-59	18
B 3.6-16	0	B 3.6-60	18
B 3.6-17	0	B 3.6-61	18
B 3.6-18	16	B 3.6-62	18
B 3.6-19	0	B 3.6-63	18
B 3.6-20	0	B 3.6-64	18
B 3.6-21	0	B 3.6-65	18
B 3.6-22	0	B 3.6-66	29
B 3.6-23	0	B 3.6-67	29
B 3.6-24	0	B 3.6-68	18
B 3.6-25	0	B 3.6-69	21
B 3.6-26	27	B 3.6-70	21
B 3.6-27	18	B 3.6-71	29
B 3.6-28	18	B 3.6-72	21
B 3.6-29	18	B 3.6-73	21
B 3.6-30	18	B 3.6-74	21
B 3.6-31	18	B 3.6-75	21
B 3.6-32	18	B 3.6-76	21
B 3.6-33	27	B 3.6-77	21
B 3.6-34	18	B 3.6-78	21
B 3.6-35	18	B 3.6-79	21
B 3.6-36	18	B 3.6-80	18
B 3.6-37	18	B 3.6-81	21
B 3.6-38	18	B 3.6-82	21

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The criticality curve provides limits for both heatup and cooldown during criticality. Development of the curves considered instrument uncertainty values of 10°F for temperature and 15 psig for pressure plus an additional 15 psig for pressure instrument location (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel (including its appurtenances) is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel (including its appurtenances).

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section XI, Appendix G (Ref. 3).

(continued)

BASES

BACKGROUND

The P/T limit curves in this Specification were developed in accordance with the 1989 Edition of the ASME Code, Section XI, Appendix G (Ref. 3). These P/T limit curves were developed using the initiation fracture toughness, K_{IC}, for the allowable material fracture toughness. The use of K_{IC} for development of P/T limit curves has been approved by the ASME through Code Case N-640 (Ref. 4).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with the UFSAR (Ref. 5) and Appendix H of 10 CFR 50 (Ref. 6). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 7.

The bounding P/T limit curves are based on the N16 instrumentation nozzles. These nozzles are located at the top of the beltline region and were determined to be the limiting material with respect to the P/T curves.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 2 requirement that they be at least 40°F above the noncritical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 8), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

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BASES (continued

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 9 provides the curves and limits in this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are ≤ 100°F in any 1 hour period, during RCS heatup and cooldown;

b. RCS pressure and temperature are within the applicable limits in Figure 3.4.9-3, 3.4.9-4, or 3.4.9-5, and heatup or cooldown rates are ≤ 30°F in any 1 hour period, during RCS inservice leak and hydrostatic testing;

c. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup;

d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and

f. The reactor vessel flange and the head flange temperatures are ≥ 70°F when tensioning the reactor vessel head bolting studs.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified as safe by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored. With the applicable limits of Figures 3.4.9-3, 3.4.9-4, or 3.4.9-5 exceeded during inservice hydrostatic and leak testing operations, the maximum temperature change shall be limited to 10°F in any 1 hour period during restoration of the P/T limit parameters to within limits.

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 8), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8 (continued)

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.6 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature is ≤ 80°F in MODE 4. SR 3.4.9.8 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature is ≤ 100°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. Calculation 0B21-1029, "Instrument Uncertainty for RCS Pressure/Temperature Limits Curve," Revision 0.
2. 10 CFR 50, Appendix G.
3. 1989 Edition of the ASME Code, Section XI, Appendix G.
4. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."
5. UFSAR, Section 5.3.1.6 and Appendix 5.3B.
6. 10 CFR 50, Appendix H.
7. Regulatory Guide 1.99, Revision 2, May 1988.
8. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
9. Calculation 0B11-0005, "Development of RPV Pressure-Temperature Curves For BNP Units 1 and 2 For Up To 32 EFPY of Plant Operation," Revision 1.
10. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABILITY (continued)

In MODE 3, both the hydrogen and oxygen production rates and the total amounts produced after a LOCA would be less than those calculated for the Design Basis Accident LOCA. Thus, if the analysis were to be performed starting with a LOCA in MODE 3, the time to reach a flammable concentration would be extended beyond the time conservatively calculated for MODE 1. The extended time would allow hydrogen removal from the primary containment atmosphere by other means and also allow repair of an inoperable CAD subsystem, if CAD were not available. Therefore, the CAD System is not required to be OPERABLE in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the CAD System is not required to be OPERABLE in MODES 4 and 5.

ACTIONS

A.1

If the CAD System (one or both subsystems) is inoperable, it must be restored to OPERABLE status within 31 days. In this Condition, the oxygen control function of the CAD System is lost. However, alternate oxygen control capabilities may be provided by the Containment Inerting System. The 31 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action to prevent exceeding this limit, and the availability of other hydrogen mitigating systems.

Required Action A.1 has been modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the CAD System (one or both subsystems) is inoperable. This allowance is provided because of the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit, and the availability of other hydrogen mitigating systems.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Action A.1 cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 8 hours. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.1

Verifying that there is ≥ 4350 gal of liquid nitrogen supply in the CAD System will ensure at least 29 days of post-LOCA CAD operation. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required. The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems.

SR 3.6.3.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

(continued)

BASES

ACTIONS

A.1 (continued)

containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Movement of recently irradiated fuel assemblies in the secondary containment and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable. Suspension of this activity shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement

(continued)