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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 REVISIONS 16 THROUGH 19

Ladies and Gentlemen:

In accordance with Technical Specification (TS) 5.5.10.d for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, Carolina Power & Light (CP&L) Company is submitting Revisions 16 through 19 to the BSEP Unit 1 and Unit 2 TS Bases. Enclosure 1 provides a list of the revision numbers, a description of each revision, the date of implementation, and the BSEP units affected. Instructions for replacing the pages contained in the TS Bases books are provided in Enclosure 2. Enclosure 3 provides replacement TS Bases pages for both BSEP units.

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

Sincerely,

David C DiCello

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WRM/wrm

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

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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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FOR REVISIONS 16 THROUGH 19

Summary of Revisions to Technical Specification Bases

Summary of Revisions to Technical Specification Bases			
Revision	**Affected Units**	**Date Implemented**	**Title/Description**
16	1 and 2	September 25, 2001	**Title**: Primary Containment Isolation Valve Operability **Description**: This Bases revision clarified the conditions for which normally closed primary containment isolation valves are considered operable.
17	1 and 2	October 5, 2001	**Title**: Revised Pressure-Temperature Curves **Description**: This Bases revision reflects Amendments 214 and 241 which revised the pressure-temperature curves for operation up to 19 effective full power years.
18	1 and 2	November 29, 2001	**Title**: Revised Excess Flow Check Valve Testing Frequency **Description**: This Bases revision reflects Amendments 215 and 242 which revised Surveillance Requirement 3.6.1.3.7 to allow a representative sample, approximately 20 percent, of excess flow check valves (EFCVs) to be tested every 24 months, such that each EFCV is tested once every 10 years.

Summary of Revisions to Technical Specification Bases			
Revision	**Affected Units**	**Date Implemented**	**Title/Description**
19	1 and 2	November 29, 2001	**Title**: Clarification of Drywell Floor Drain Sump Flow Monitoring Instrumentation **Description:** This Bases revision clarifies discussion of the Drywell Floor Drain Sump Flow Monitoring Instrumentation regarding alarms associated with this instrumentation and the requirements to perform a channel functional test.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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FOR REVISIONS 16 THROUGH 19

Technical Specification Bases Pages Replacement Instructions

Unit 1	
Remove	**Insert**
Bases Book 1	
Title Page, Revision 15	Title Page, Revision 19
LOEP-1, Revision 15	LOEP-1, Revision 19
Bases Book 2	
LOEP-1, Revision 15	LOEP-1, Revision 19
LOEP-2, Revision 3	LOEP-2, Revision 18
LOEP-3, Revision 15	LOEP-3, Revision 18
Table of Contents Page i, Revision 0	Table of Contents Page i, Revision 18
B 3.4-20 - B 3.4-24, Revision 0	B 3.4-20 - B 3.4-24, Revision 19
B 3.4-39 - B 3.4-47, Revision 0	B 3.4-39 - B 3.4-47, Revision 17
B 3.6-18, Revision 0	B 3.6-18, Revision 16
B 3.6-27 - B 3.6-83, Revision 0	B 3.6-27 - B 3.6-83, Revision 18
B 3.6-84, Revision 11	B 3.6-84, Revision 18
---	B 3.6-85, Revision 18

Unit 2	
Remove	**Insert**
Bases Book 1	
Title Page, Revision 15	Title Page, Revision 19
LOEP-1, Revision 15	LOEP-1, Revision 19
Bases Book 2	
LOEP-1, Revision 15	LOEP-1, Revision 19
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Table of Contents Page i, Revision 0	Table of Contents Page i, Revision 18
B 3.4-20 - B 3.4-24, Revision 0	B 3.4-20 - B 3.4-24, Revision 19
B 3.4-39 - B 3.4-47, Revision 0	B 3.4-39 - B 3.4-47, Revision 17
B 3.6-18, Revision 0	B 3.6-18, Revision 16
B 3.6-27 - B 3.6-83, Revision 0	B 3.6-27 - B 3.6-83, Revision 18
B 3.6-84, Revision 11	B 3.6-84, Revision 18
---	B 3.6-85, Revision 18

ENCLOSURE 3

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 REVISIONS 16 THROUGH 19

Replacement Bases Pages – Units 1 and 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 19

LIST OF EFFECTIVE PAGES - BASES

Page No.	Revision No.
Title Page	19
List of Effective Pages - Book 1	
LOEP-1	19
LOEP-2	3
LOEP-3	12
LOEP-4	12
i	0
ii	0
B 2.0-1	0
B 2.0-2	0
B 2.0-3	0
B 2.0-4	10
B 2.0-5	0
B 2.0-6	0
B 2.0-7	0
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	0
B 3.0-14	0
B 3.0-15	0
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	0
B 3.1-18	0
B 3.1-19	0
B 3.1-20	0
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B 3.1-22	0
B 3.1-23	0
B 3.1-24	0
B 3.1-25	0
B 3.1-26	0
B 3.1-27	0
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	0
B 3.1-34	0
B 3.1-35	0
B 3.1-36	0
B 3.1-37	0
B 3.1-38	0
B 3.1-39	0
B 3.1-40	0
B 3.1-41	0
B 3.1-42	0
B 3.1-43	0
B 3.1-44	0
B 3.1-45	0
B 3.1-46	0
B 3.1-47	0
B 3.1-48	0

(continued)

Unit 1
Bases Book 2 Replacement Pages

LIST OF EFFECTIVE PAGES - BASES

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

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LIST OF EFFECTIVE PAGES - BASES (continued)

Page No.	Revision No.	Page No.	Revision No.
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	0	B 3.6-46	18
B 3.6-3	0	B 3.6-47	18
B 3.6-4	0	B 3.6-48	18
B 3.6-5	0	B 3.6-49	18
B 3.6-6	0	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	18
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	18
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	18
B 3.6-23	0	B 3.6-67	18
B 3.6-24	0	B 3.6-68	18
B 3.6-25	0	B 3.6-69	18
B 3.6-26	0	B 3.6-70	18
B 3.6-27	18	B 3.6-71	18
B 3.6-28	18	B 3.6-72	18
B 3.6-29	18	B 3.6-73	18
B 3.6-30	18	B 3.6-74	18
B 3.6-31	18	B 3.6-75	18
B 3.6-32	18	B 3.6-76	18
B 3.6-33	18	B 3.6-77	18
B 3.6-34	18	B 3.6-78	18
B 3.6-35	18	B 3.6-79	18
B 3.6-36	18	B 3.6-80	18
B 3.6-37	18	B 3.6-81	18
B 3.6-38	18	B 3.6-82	18

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B 3.6-83	18	B 3.7-40	9
B 3.6-84	18	B 3.7-41	9
B 3.6-85	18	B 3.7-42	9
B 3.7-1	0	B 3.8-1	0
B 3.7-2	0	B 3.8-2	0
B 3.7-3	0	B 3.8-3	0
B 3.7-4	0	B 3.8-4	0
B 3.7-5	0	B 3.8-5	14
B 3.7-6	0	B 3.8-6	6
B 3.7-7	0	B 3.8-7	6
B 3.7-8	0	B 3.8-8	6
B 3.7-9	15	B 3.8-9	6
B 3.7-10	15	B 3.8-10	6
B 3.7-11	15	B 3.8-11	6
B 3.7-12	15	B 3.8-12	6
B 3.7-13	15	B 3.8-13	6
B 3.7-14	15	B 3.8-14	6
B 3.7-15	15	B 3.8-15	6
B 3.7-16	15	B 3.8-16	6
B 3.7-17	15	B 3.8-17	6
B 3.7-18	15	B 3.8-18	6
B 3.7-19	15	B 3.8-19	6
B 3.7-20	15	B 3.8-20	6
B 3.7-21	9	B 3.8-21	6
B 3.7-22	9	B 3.8-22	6
B 3.7-23	9	B 3.8-23	6
B 3.7-24	9	B 3.8-24	6
B 3.7-25	9	B 3.8-25	6
B 3.7-26	11	B 3.8-26	6
B 3.7-27	11	B 3.8-27	6
B 3.7-28	9	B 3.8-28	6
B 3.7-29	9	B 3.8-29	6
B 3.7-30	9	B 3.8-30	6
B 3.7-31	9	B 3.8-31	6
B 3.7-32	9	B 3.8-32	6
B 3.7-33	9	B 3.8-33	6
B 3.7-34	9	B 3.8-34	6
B 3.7-35	9	B 3.8-35	6
B 3.7-36	9	B 3.8-36	6
B 3.7-37	9	B 3.8-37	6
B 3.7-38	9	B 3.8-38	6
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

The UFSAR (Ref. 1), requires means for detecting RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when LEAKAGE rates above normal background levels are detected and also to supply quantitative measurement of LEAKAGE rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two independently monitored variables, such as drywell floor drain sump flow changes and drywell gaseous or particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump flow monitoring system.

The drywell floor drain sump flow monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges, floor drains, the Reactor Building Closed Cooling Water System, and drywell cooler drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump is provided with two sump pumps. A flow transmitter in the common discharge line of the drywell floor drain sump pumps inputs to a flow integrator. In addition to the required instrumentation, the starting frequency and run duration of a sump pump motor are monitored by timer circuitry to provide a signal (alarm) in the control room indicating that LEAKAGE has reached a specified limit.

(continued)

BASES

BACKGROUND (continued)	The primary containment atmosphere radioactivity monitoring systems (particulate and gaseous) continuously monitor the primary containment atmosphere for airborne particulate and gaseous radioactivity. The primary containment atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying LEAKAGE rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times. A significant increase of radioactivity, which may be attributed to a sudden increase in RCPB steam or reactor water LEAKAGE, is annunciated in the control room.
APPLICABLE SAFETY ANALYSES	A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 3 and 4). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm and/or indication of excess LEAKAGE in the control room. A control room alarm/indication allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 5). Therefore, these actions provide adequate response before a significant break in the RCPB can occur. RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).
LCO	The one channel of drywell floor drain sump flow monitoring system is required to quantify the unidentified LEAKAGE from the RCS. The required drywell floor drain sump flow monitoring system instrumentation includes the flow transmitter and integrator, as well as a flow totalizer. One channel of the other monitoring systems (particulate or gaseous) provides early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump flow monitoring system inoperable, no other required instrumentation can provide the equivalent information to quantify LEAKAGE. However, the primary containment atmosphere radioactivity monitor will provide indication of changes in LEAKAGE.

With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 8 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump flow monitoring system is inoperable. This allowance is provided because other instrumentation (listed in Reference 1) is available to monitor RCS LEAKAGE.

B.1 and B.2

With both gaseous and particulate primary containment atmosphere radioactivity monitoring channels inoperable (i.e., the required primary containment atmosphere monitoring system), grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic LEAKAGE information. Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmosphere radioactivity monitoring channels are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be met, 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 MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmosphere radioactivity monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies, for the radioactivity monitoring channels only, the required alarm

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.2 (continued)

setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is consistent with the Brunswick refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 5.2.5.
2. Regulatory Guide 1.45, May 1973.
3. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through—Wall Flaws, April 1968.
4. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.
5. UFSAR, Section 5.2.5.2.2.
6. 10 CFR 50.36(c)(2)(ii).

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.

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. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 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 III, Appendix G (Ref. 2).

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

(continued)

BASES

BACKGROUND (continued)

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 P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

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 1 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 Figures 3.4.9-3 or 3.4.9-4, 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

LCO (continued)

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODES 1, 2, and 3 must be corrected so that the RCPB is returned to a condition that has been verified as safe by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation within acceptable limits, an engineering evaluation is required to determine if RCS operation can continue. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, 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 vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 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 A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

(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 Figure 3.4.9-3 or 3.4.9-4 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 (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 and SR 3.4.9.2

Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

SR 3.4.9.1 is modified by a Note that requires the Surveillance to be performed only during system heatup and cooldown operations. SR 3.4.9.2 is modified by a Note that requires the Surveillance to be performed only during inservice leakage and hydrostatic testing.

SR 3.4.9.3

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.4 and SR 3.4.9.5

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.4 and SR 3.4.9.5 (continued)

Performing the Surveillance within 30 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the differential temperature requirement of SR 3.4.9.4 is to compare the temperature of the reactor coolant in the dome to the bottom head drain temperature.

As specified in procedures, an acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.5 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.4 and SR 3.4.9.5 are modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Note also states the SR is only required to be met during recirculation pump startup, since this is when the stresses occur.

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

(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. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, 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," dated November 8, 2000.
10. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)	Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. PCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).
LCO	PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA. The power operated, automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. Primary containment purge and vent valves > 8 inches must be blocked to prevent opening > 50° (approximately 55%). While the reactor building-to-suppression chamber vacuum breakers isolate primary containment penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO 3.6.1.5, "Reactor Building-to-Suppression Chamber Vacuum Breakers." The valves covered by this LCO are listed with their associated stroke times in Reference 4. The normally closed PCIVs are considered OPERABLE when any one of the following condtions is met: (1) manual valves are closed or opened in accordance with appropriate administrative controls, (2) automatic valves are de-activated and secured in their closed position, (3) blind flanges are in place, or (4) closed systems are intact. MSIVs are exempt from Type C testing limits and must meet specific leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing. This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.7

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) is OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break signal. This may be accomplished by cycling the EFCVs through one complete cycle of full travel. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 12). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

The analyses in References 2 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be $\leq$ 11.5 scfh when tested at $\geq P_t$ (25 psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of the Primary Containment Leakage Rate Testing Program. The Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Ref. 6), and conforms with Regulatory Guide 1.163 (Ref. 7) and Nuclear Energy Institute (NEI) 94-01 (Ref. 8) except for the following:

a. BNP may use standard glass tube and ball type flowmeters with an accuracy of 5% of full scale. This is an exception to the flowmeter accuracy requirements of ANSI/ANS 56.8-1994 (Ref. 9) referenced in NEI 94-01 (Ref. 8), Section 8.0. The basis for this exception is described in Reference 10.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

b. Local leak rate testing of the MSIVs may be performed at a pressure less than P_a. This is an exemption from the requirements of 10 CFR 50 Appendix J (Ref. 6). The basis for this exemption is described in Reference 11.

The Frequency is required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. UFSAR, Chapter 15.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. 10 CFR 50.36(c)(2)(ii).
4. Technical Requirements Manual.
5. UFSAR, Section 15.2.3.
6. 10 CFR 50, Appendix J, Option B.
7. NRC Regulatory Guide 1.163, Performance-Based Containment Leak-Rate Testing Program, September 1995.
8. Nuclear Energy Institute (NEI) 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J, July 26, 1995.
9. ANSI/ANS 56.8-1994.
10. NRC SER; Issuance of Amendment No. 181 to Facility Operating License No. DPR-71 and Amendment No. 213 to Facility Operating License No. DPR-62 Regarding 10 CFR 50 Appendix J, Option B - Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 95-0316) (TAC Nos. M93679 and M93680); dated February 1, 1996.

(continued)

BASES

REFERENCES (continued)

11. NRC SER, Brunswick 1 & 2 - Amendments No. 10 and 36 to Operating Licenses Revising Technical Specifications to Grant Exemptions from Specific Requirements of 10 CFR 50 Appendix J, dated November 8, 1977.

12. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 and 2 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Refs. 1 and 2). Among the inputs to the design basis analysis is the initial drywell average air temperature (Refs. 1 and 2). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 300°F (Ref. 3). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 8 hour Completion Time is acceptable, considering the sensitivity of the analysis to allow significant variations in this parameter, and provides sufficient time to correct minor problems.

B.1 and B.2

If the drywell average air temperature cannot be restored to within the limit 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. Drywell air temperature is monitored in all quadrants and at various elevations (referenced to mean sea level). Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1 (continued)

The following locations are monitored to obtain the drywell average temperature:

a. Below 5 ft elevation;

b. Between 10 ft and 23 ft elevation;

c. Between 28 ft and 45 ft elevation;

d. Between 70 ft and 80 ft elevation; and

e. Above 90 ft elevation.

The 24 hour Frequency of the SR is based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal drywell air temperature condition.

REFERENCES

1. UFSAR, Section 6.2.

2. GE-NE-T23-00735-01, Brunswick Steam Electric Plant Units 1 and 2 High Drywell Bulk Average Temperature Analysis, October 1996.

3. UFSAR, Section 6.2.1.1.1.

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Reactor Building-to-Suppression Chamber Vacuum Breakers

BASES

BACKGROUND

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression chamber-to-drywell vacuum breakers. The design of the external (reactor building-to-suppression chamber) vacuum relief provisions consists of two vacuum breakers (a mechanical vacuum breaker and a pneumatically operated butterfly valve), located in series in each of two 20 inch lines. The two lines from the reactor building merge to a common 20 inch line which connects to the suppression chamber airspace. Each path is capable of relieving 100% of design flow. The butterfly valve is actuated by a differential pressure switch. The normal pneumatic supply for the butterfly valve is the Non-interruptible Instrument Air System. A Nitrogen Backup System is provided to each butterfly valve and is automatically aligned to the valves following a loss of coolant accident (LOCA) signal and a subsequent primary containment isolation or following a loss of offsite power. Additionally, the Nitrogen Backup System automatically aligns to the valves to maintain system pressure when the Non-interruptible Instrument Air System pressure drops to approximately 95 psig. The mechanical vacuum breaker is self actuating similar to a check valve. Both mechanical vacuum breakers can be locally operated for testing purposes. The two vacuum breakers in series must be closed to maintain a leak tight primary containment boundary.

The Nitrogen Backup System is of safety grade quality and complies with the intent of Generic Letter 84-09 (Ref. 1). This system consists of two independent and redundant subsystems. Each subsystem supplies safety grade nitrogen from a nitrogen bottle rack to one pneumatic butterfly valve via a pressure control valve and has sufficient capacity to provide 22 hours of valve operation including design system leakage. The pressure control valve reduces the nitrogen bottle supply pressure of ≥ 1130 psig to a normal system pressure of approximately 95 psig.

(continued)

BASES

BACKGROUND (continued)

A negative differential pressure across the drywell wall is caused by depressurization of the drywell. Events that cause this depressurization are cooling cycles, inadvertent primary containment spray actuation, and steam condensation in the event of a primary system rupture. Reactor building-to-suppression chamber vacuum breakers prevent an excessive negative differential pressure across the primary containment boundary. Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation following a small break LOCA results in a significant pressure transient and becomes important in sizing the external (reactor building-to-suppression chamber) vacuum breakers.

The external vacuum breakers are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere. Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 2 as part of the accident response of the containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 0.5 psid (Ref. 2). Additionally, of the two series reactor building-to-suppression chamber vacuum breakers, one is assumed to fail in a closed position to satisfy the single active failure criterion. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight with positive primary containment pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers:

a. A small break loss of coolant accident followed by actuation of both primary containment spray loops and

b. A large break loss of coolant accident followed by actuation of both primary containment spray loops.

The results of these two cases show that the external vacuum breakers, with a full open setpoint of 0.5 psid, are capable of maintaining the differential pressure within design limits.

The reactor building-to-suppression chamber vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two vacuum breakers (mechanical vacuum breaker and pneumatic butterfly valve) in each of the two lines from the reactor building to the common line connected to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber. For a pneumatic butterfly valve to be OPERABLE for opening, both the Non-interruptible Instrument Air System and the Nitrogen Backup System shall be capable of supplying the pneumatic operator.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell atmosphere and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which, after the suppression chamber-to-drywell vacuum breakers open (due to differential pressure between the suppression chamber and

(continued)

APPLICABILITY (continued)

drywell), would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor building-to-suppression chamber vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

A Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each of the two lines from the reactor building to the common line connected to the suppression chamber air space.

With one or more lines with one vacuum breaker not closed, the leak tight primary containment boundary may be threatened. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status or the open vacuum breaker closed within 72 hours. The 72 hour Completion Time is consistent with requirements for inoperable suppression chamber-to-drywell vacuum breakers in LCO 3.6.1.6, "Suppression Chamber-to-Drywell Vacuum Breakers." The 72 hour Completion Time takes into account the redundant capability afforded by the remaining breakers, the fact that the OPERABLE breaker in each of the lines is closed, and the low probability of an event occurring that would require the vacuum breaker to be OPERABLE during this period.

B.1

With one or more lines with two vacuum breakers not closed, primary containment integrity is not maintained. Therefore, one open vacuum breaker in each line must be closed within 2 hours. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 2 hours.

(continued)

BASES

ACTIONS (continued)

C.1

With one vacuum breaker inoperable solely due to its associated nitrogen backup subsystem being inoperable, the leak tight primary containment boundary is intact. In this Condition, the vacuum breakers in the redundant line are adequate to mitigate the primary containment depressurization. However, overall reliability is reduced because a single failure could result in a loss of the capability to mitigate an event that causes a containment depressurization following a LOCA and a subsequent primary containment isolation. The 31 day Completion Time is acceptable because of the OPERABLE vacuum breakers in the redundant line, the normal pneumatic supply is available to the vacuum breaker, and the low probability of a LOCA and a subsequent primary containment isolation occurring during the period the nitrogen backup subsystem is inoperable.

D.1

With two vacuum breakers inoperable solely due to their associated nitrogen backup subsystems being inoperable, the leak tight primary containment boundary is intact. Since the normal pneumatic supply is available to each vacuum breaker, the vacuum breakers are still capable of mitigating any event that causes a containment depressurization except following a LOCA and a subsequent primary containment isolation. The 7 day Completion Time is acceptable because the normal pneumatic supply is available to each vacuum breaker and because of the low probability of a LOCA and a subsequent primary containment isolation occurring during the period the nitrogen backup subsystems are inoperable.

E.1

With one line with one or more vacuum breakers inoperable for opening, the leak tight primary containment boundary is intact. However, with one line with one or more vacuum breakers inoperable for opening for reasons other than its associated nitrogen backup subsystem being inoperable (Condition C), overall system reliability is reduced because a single failure in one of the vacuum breakers in the

(continued)

BASES

ACTIONS

E.1 (continued)

redundant line could threaten the ability to mitigate an event that causes a containment depressurization. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status within 72 hours. This is consistent with the Completion Time for Condition A and the fact that the leak tight primary containment boundary is being maintained.

F.1

With two lines with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the function of the vacuum breakers is lost if the vacuum breakers are inoperable for reasons other than the Nitrogen Backup System being inoperable (Condition D). Therefore, all vacuum breakers in one line must be restored to OPERABLE status within 2 hours. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 2 hours.

G.1 and G.2

If any Required Action and associated Completion Time cannot be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

The bank of nitrogen bottles supplying each nitrogen backup subsystem header is required to be verified to be pressurized to ≥ 1130 psig to ensure sufficient motive force is available to the pneumatic butterfly valve actuators following a LOCA and subsequent primary containment isolation. A nitrogen bottle pressure of ≥ 1130 psig assures sufficient capacity to actuate and cycle the pneumatic butterfly valve for 22 hours including design system leakage. This Surveillance may be satisfied by

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1 (continued)

verifying the absence of the Nitrogen Backup System low pressure alarms. The 24 hour Frequency is based on engineering judgment in view of the fact that adequate indication of pressure is available to the operator and the Frequency has also been shown to be acceptable through operating experience.

SR 3.6.1.5.2

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

SR 3.6.1.5.3

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This SR ensures that the safety analysis assumptions are valid. This is accomplished by manually verifying that each mechanical vacuum breaker is free to open and verifying each pneumatic butterfly valve operates through at least one complete cycle of full travel. The 92 day Frequency of this SR was developed based upon Inservice Testing Program requirements to perform valve testing at least once every 92 days.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.5.4

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of $\leq$ 0.5 psid is valid. This is accomplished by demonstrating that the force required to open each mechanical vacuum breaker is $\leq$ 0.5 psid and demonstrating that each pneumatic butterfly valve opens at $\geq$ 0.4 psid and $\leq$ 0.5 psid with suppression chamber pressure negative with respect to reactor building pressure. The 24 month Frequency has been demonstrated to be acceptable, based on operating experience, and is further justified because of other Surveillances performed more frequently that convey the proper functioning status of each vacuum breaker.

SR 3.6.1.5.5

To ensure the pneumatic butterfly valves have sufficient capacity to actuate and cycle following a LOCA and subsequent primary containment isolation, Nitrogen Backup System leakage must be within the design limit. This SR ensures that overall system leakage is within a design limit of 0.65 scfm. This is accomplished by measuring the nitrogen bottle supply pressure decrease while maintaining approximately 95 psig to the nitrogen backup subsystem during the test with an initial nitrogen bottle supply pressure of $\geq$ 1130 psig. The system leakage test is performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is concluded to be acceptable from a reliability standpoint.

SR 3.6.1.5.6

This SR ensures that in the event a LOCA and subsequent primary containment isolation occurs, the Nitrogen Backup System will actuate to perform its design function and supply nitrogen gas at the required pressure to the pneumatic operators of the butterfly valves. The 24 month

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.6 (continued)

Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NRC Generic Letter GL 84-09, Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii).

2. UFSAR, Section 6.2.

3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

BACKGROUND

The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 10 internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber, which allow flow from the suppression chamber atmosphere to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the suppression chamber-drywell boundary. Each vacuum breaker is a self actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

A negative differential pressure across the drywell wall is caused by depressurization of the drywell. Events that cause this depressurization are cooling cycles, inadvertent drywell spray actuation, and steam condensation from sprays or subcooled water reflood of a break in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or spill of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, the drywell atmosphere is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow from a recirculation line break, or drywell spray actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is less than the suppression chamber pressure, there will be an

(continued)

BASES

BACKGROUND (continued)

increase in the height of the downcomer waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary.

The safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid (Ref. 1). Additionally, 3 of the 10 internal vacuum breakers are assumed to fail in a closed position (Ref. 1). The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that 8 of 10 vacuum breakers be OPERABLE (the additional vacuum breaker is required to meet the single failure criterion) are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The total cross sectional area of the main vent system between the drywell and suppression chamber needed to fulfill this requirement has been established as a minimum of 51.5 times the total break area. In turn, the vacuum relief capacity between the drywell and suppression chamber should be 1/16 of the total main vent cross sectional area, with the valves set to operate at ≤ 0.5 psid differential pressure. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight until the suppression pool is at a positive pressure relative to the drywell.

The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

Only 8 of the 10 vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers, however, are required to be closed (except when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell atmosphere and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one of the required vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining seven OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one of the eight required vacuum breakers inoperable, 72 hours is allowed to restore at least one of the inoperable vacuum breakers to OPERABLE status so that

(continued)

BASES

ACTIONS

A.1 (continued)

plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

B.1

With one vacuum breaker not closed, communication between the drywell and suppression chamber airspace could occur, and, as a result, there is the potential for primary containment overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not available, an alternate method of verifying that the vacuum breakers are closed is to verify that the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 8 hour Completion Time is considered adequate to perform this test.

C.1 and C.2

If any Required Action and associated Completion Time can not be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified closed (except when the vacuum breaker is performing its intended design function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel and procedural controls to ensure the drywell is normally maintained at a higher pressure than the suppression chamber, and has been shown to be acceptable through operating experience. This verification is also required within 2 hours after any discharge of steam to the suppression chamber from any source.

A Note is added to this SR which allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

SR 3.6.1.6.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This is accomplished by verifying each required vacuum breaker operates through at least one complete cycle of full travel. This SR ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR was developed, based on Inservice Testing Program requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after a discharge of steam to the suppression chamber from any source.

SR 3.6.1.6.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.5 psid is valid. The 24 month Frequency is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.3 (continued)

need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency has been demonstrated to be acceptable, based on operating experience, and is further justified because of other surveillances performed more frequently that convey the proper functioning status of each vacuum breaker.

REFERENCES

1. UFSAR, Section 6.2.

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven systems (i.e., the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System). Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation - the original limit for the end of a LOCA blowdown was 170°F, based on the Bodega Bay and Humboldt Bay Tests;

b. Primary containment peak pressure and temperature - design pressure is 62 psig and design temperature is 340°F (Ref. 1);

c. Condensation oscillation loads - maximum allowable initial temperature is 110°F; and

d. Chugging loads - these only occur at < 135°F; therefore, there is no initial temperature limit because of chugging.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is a spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (Reference 1 for LOCAs and for the pool temperature analyses required by Reference 2). An initial pool temperature of 95°F is assumed for the Reference 1 containment analyses. Reactor shutdown at a pool temperature of 110°F and vessel depressurization at a pool temperature of 120°F are assumed for the Reference 1 analyses. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during unit testing.

Suppression pool average temperature satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are:

a. Average temperature ≤ 95°F with THERMAL POWER > 1% RATED THERMAL POWER (RTP) and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.

b. Average temperature ≤ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in unit risk.

(continued)

BASES

LCO (continued)

c. Average temperature ≤ 110°F with THERMAL POWER ≤ 1% RTP. This requirement ensures that the unit will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

At 1% RTP, heat input is approximately equal to normal system heat losses.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

A.1 and A.2

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power, the initial conditions exceed the conditions assumed for the Reference 1 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above those assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to allow the suppression pool average temperature to be restored below the limit. Additionally, when suppression pool temperature is > 95°F, increased monitoring of the suppression pool temperature is required to ensure that it remains ≤ 110°F. The once per hour Completion Time is adequate based on past experience, which has shown that pool temperature increases relatively slowly except when testing that adds heat to the suppression pool is being performed. Furthermore, the once per hour Completion Time is considered adequate in view of other indications in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

(continued)

BASES

ACTIONS (continued)

B.1

If the suppression pool average temperature cannot be restored to within limits 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 power must be reduced to ≤ 1% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce power from full power conditions in an orderly manner and without challenging plant systems.

C.1

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP, and when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105°F, all testing must be immediately suspended to preserve the heat absorption capability of the suppression pool. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

D.1 and D.2

Suppression pool average temperature > 110°F requires that the reactor be shut down immediately. This is accomplished by manually scramming the reactor. Further cooldown to Mode 4 within 36 hours is required at normal cooldown rates (provided pool temperature remains ≤ 120°F). Additionally, when suppression pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours, and the plant must be brought to at least 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.

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. The average temperature is determined using an algorithm with inputs from OPERABLE suppression pool water temperature channels. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

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

2. NUREG-0783.

3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the steam exhaust lines in the turbine driven systems (i.e., High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System) and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 86,450 ft^3 at the low water level limit of -31 inches and 89,750 ft^3 at the high water level limit of -27 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the SRV quenchers, main vents, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from SRV discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to SRV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of References 1 and 2 remains valid. Suppression pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).
LCO	A limit that suppression pool water level be ≥ -31 inches and ≤ -27 inches is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.
APPLICABILITY	In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS-Shutdown."
ACTIONS	A.1 With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as main vents are covered, HPCI and RCIC turbine exhausts are covered, and SRV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the Drywell Spray System. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If suppression pool water level cannot be restored to within limits 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. The 24 hour Frequency of this SR has been shown to be acceptable based on operating experience. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.

REFERENCES

1. UFSAR, Section 6.2.1.1.3.2.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each RHR subsystem contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. Service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink.

The heat removal capability of two RHR pumps in one subsystem is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (SRV). With only one RHR pump in one subsystem OPERABLE, the available heat removal capability results in suppression pool temperatures, after a DBA LOCA, for which inadequate NPSH would be available for required RHR and Core Spray pumps. Therefore, to ensure adequate NPSH is available for required RHR and Core Spray pumps, two RHR pumps are required to be OPERABLE in a subsystem. SRV leakage and High Pressure Coolant Injection System and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

References 1 and 2 contain the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The intent of the analyses is to demonstrate that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Refs. 1 and 2). To ensure that these requirements are met, two independent RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when two pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both a release of radioactive material to the primary containment and a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool

(continued)

BASES

ACTIONS

A.1 (continued)

cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

Required Action A.1 is modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one RHR suppression pool cooling subsystem is inoperable. This allowance is provided because of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem.

B.1

With two RHR suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to remove heat from primary containment are available.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

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 since the RHR suppression pool cooling mode is manually initiated. 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. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each RHR pump develops a flow rate ≥ 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment pressure and temperature can be maintained below the design limits during a DBA (Ref. 2). The normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 4) is covered by the requirements of LCO 3.5.1, "ECCS—Operating." This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is 92 days.

REFERENCES

1. UFSAR, Section 6.2.1.1.3.2.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. 10 CFR 50.36(c)(2)(ii).
4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Oxygen Concentration

BASES

BACKGROUND	The primary containment is designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inert, that is, oxygen concentration < 4.0 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The capability to inert the primary containment and maintain oxygen < 4.0 v/o works together with the Containment Atmosphere Dilution System (LCO 3.6.3.2, "Containment Atmosphere Dilution (CAD) System") to provide redundant and diverse methods to mitigate events that produce hydrogen and oxygen. For example, an event that rapidly generates hydrogen from zirconium metal water reaction could result in excessive hydrogen in primary containment, but oxygen concentration will remain < 5.0 v/o and no combustion can occur. Long term generation of both hydrogen and oxygen from radiolytic decomposition of water may eventually result in a combustible mixture in primary containment if the initial primary containment oxygen concentration exceeded 4.0 v/o during operation in the applicable conditions. This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.
APPLICABLE SAFETY ANALYSES	The Reference 1 calculations assume that the primary containment is inerted when a Design Basis Accident (DBA) loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce combustible gas mixtures in the primary containment. Oxygen, which is subsequently generated by radiolytic decomposition of water, is diluted by the CAD System more rapidly than it is produced. Primary containment oxygen concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

The primary containment oxygen concentration is maintained < 4.0 v/o to ensure that an event that produces any amount of hydrogen and oxygen does not result in a combustible mixture inside primary containment.

APPLICABILITY

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen and oxygen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible during a scheduled power reduction to ≤ 15% RTP. As long as reactor power is ≤ 15% RTP, the potential for an event that generates significant hydrogen and oxygen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a scheduled power reduction ≤ 15% RTP, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

If oxygen concentration is ≥ 4.0 v/o at any time while operating in MODE 1, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration must be restored to < 4.0 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 4.0 v/o because of the availability of other hydrogen and oxygen mitigating systems (e.g., Containment Atmosphere Dilution System) and the low probability and long duration of an event that would generate significant amounts of hydrogen and oxygen occurring during this period.

(continued)

BASES

ACTIONS (continued)	B.1 If oxygen concentration cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, power must be reduced to ≤ 15% RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.6.3.1.1 The primary containment must be determined to be inerted by verifying that oxygen concentration is < 4.0 v/o. The 7 day Frequency is based on the slow rate at which oxygen concentration can change and on other indications of abnormal conditions (which would lead to more frequent checking by operators in accordance with plant procedures). Also, this Frequency has been shown to be acceptable through operating experience.
REFERENCES	1. UFSAR, Section 6.2.5. 2. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Containment Atmosphere Dilution (CAD) System

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < 5.0 volume percent (v/o).

The CAD System is manually initiated and consists of two 100% capacity subsystems. Each subsystem consists of a common liquid nitrogen supply tank, an electric vaporizer, and connected piping to supply the drywell and suppression chamber volumes. The liquid nitrogen supply tank and electric vaporizers are common components which are shared between the CAD subsystems of the two units. Piping from the liquid nitrogen supply tank downstream of the vaporizers is split and routed to each unit. Each pipe to a particular unit is divided to provide the capability to supply nitrogen to both the drywell and the suppression chamber. The nitrogen storage tank contains ≥ 4350 gal, which is adequate for 30 days of CAD subsystem operation.

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

APPLICABLE SAFETY ANALYSES

To evaluate the potential for hydrogen and oxygen accumulation in primary containment following a LOCA, hydrogen and oxygen generation is calculated (as a function of time following the initiation of the accident). The assumptions stated in Reference 1 are used to maximize the amount of hydrogen and oxygen generated. The calculation confirms that when the mitigating systems are actuated in accordance with emergency operating procedures, the peak oxygen concentration in primary containment is < 5.0 v/o (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Hydrogen and oxygen may accumulate within primary containment following a LOCA as a result of:

a. A metal water reaction between the zirconium fuel rod cladding and the reactor coolant; or

b. Radiolytic decomposition of water in the Reactor Coolant System.

The CAD System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref.3).

LCO

The CAD System (two CAD subsystems) must be OPERABLE with an OPERABLE flow path capable of supplying nitrogen to the drywell. This ensures operation of at least one CAD subsystem in the event of a worst case single active failure. Operation of at least one CAD subsystem is designed to maintain primary containment post-LOCA oxygen concentration < 5.0 v/o for 30 days.

APPLICABILITY

In MODE 1 when primary containment oxygen concentration is required to be < 4.0 v/o (i.e., primary containment inerted) in accordance with LCO 3.6.3.1, "Primary Containment Oxygen Concentration," the CAD System is required to maintain the oxygen concentration within primary containment below the flammability limit of 5.0 v/o following a LOCA. This ensures that the relative leak tightness of primary containment is adequate and prevents damage to safety related equipment and instruments located within primary containment.

In MODE 1, when primary containment oxygen concentration is not required to be < 4.0 v/o in accordance with LCO 3.6.3.1, "Primary Containment Oxygen Concentration," and in MODE 2, the potential for an event that generates significant hydrogen and oxygen is low, the primary containment need not be inert, and the CAD System is not required to be OPERABLE. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a scheduled power reduction < 15% RTP (i.e., when primary containment oxygen concentration is not required to be < 4.0 v/o in accordance with LCO 3.6.3.1), is low enough that these "windows," when the primary containment is not inerted and the CAD System is not required to be OPERABLE, are also justified.

(continued)

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 30 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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.2 (continued)

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

SR 3.6.3.2.3

Cycling each power operated valve, excluding automatic valves, in the CAD System flow path through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. While this Surveillance may be performed with the reactor at power, the 24 month Frequency of the Surveillance is intended to be consistent with expected fuel cycle lengths. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Safety Guide 7, March 1971.
2. UFSAR, Section 6.2.5.3.2.1, Amendment No. 9.
3. 10 CFR 50.36(c)(2)(ii).
4. UFSAR, Table 6.2.4-1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND	The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment. The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."
APPLICABLE SAFETY ANALYSES	There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Refs. 1 and 2) and a fuel handling accident inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment. Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES (continued)

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components or irradiated fuel assemblies located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, at least one door in each access to the Reactor Building must be closed, and the sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary 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.

(continued)

BASES

ACTIONS (continued)

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, C.2, and C.3

Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause 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. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable. Suspension of these activities 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 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 irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one secondary containment access door in each access opening are closed ensures that the infiltration of outside air of such magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in each access opening is closed. The 24 month Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will establish and maintain a negative pressure in the secondary containment. This is confirmed by demonstrating that one SGT subsystem can maintain $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate $\leq$ 3000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, this test is used to ensure secondary containment boundary integrity. Since this SR is a secondary containment test, it need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

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

2. UFSAR, Section 15.6.4.

3. UFSAR, Section 15.7.1.

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

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

6. Regulatory Guide 1.52, Revision 1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

BASES

BACKGROUND

The function of the SCIDs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1, 2, and 3). Secondary containment isolation within the time limits specified for those isolation dampers designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIDs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of active (automatic) devices.

Automatic SCIDs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

APPLICABLE SAFETY ANALYSES

The SCIDs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Refs. 1 and 2) and a fuel handling accident inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIDs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Maintaining SCIDs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment. SCIDs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	SCIDs form a part of the secondary containment boundary. The SCID safety function is related to control of offsite radiation releases resulting from DBAs. The isolation dampers are considered OPERABLE when their associated accumulators are pressurized, their isolation times are within limits, and the dampers are capable of actuating on an automatic isolation signal. The dampers covered by this LCO, along with their associated stroke times, are listed in Reference 5.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIDs is required. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIDs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment. Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.
ACTIONS	The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

(continued)

BASES

ACTIONS (continued)

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCID. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIDs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCID.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCID inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic SCID, a closed manual damper, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration, and the probability of a DBA, which requires the SCIDs to close, occurring during this short time is very low.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that secondary containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time of once per 92 days is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to devices located in high radiation areas and allows them to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With two SCIDs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic damper, a closed manual damper, and a blind flange. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIDs to close, occurring during this short time, is very low.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation dampers. This clarifies that only Condition A is entered if one SCID is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, 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.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the 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 irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

Verifying that the isolation time of each automatic SCID is within limits, by cycling each SCID through one complete cycle of full travel and measuring the isolation time, is required to demonstrate OPERABILITY. The isolation time test ensures that the SCID will isolate in the required time period. The Frequency of this SR is once per 24 months. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.4.2.2

Verifying that each automatic SCID closes on a secondary containment isolation signal is required to minimize leakage of radioactive material from secondary containment following

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.2 (continued)

a DBA or other accidents. This SR ensures that each automatic SCID will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
2. UFSAR, Section 15.6.4.
3. UFSAR, Section 15.7.1.
4. 10 CFR 50.36(c)(2)(ii).
5. Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The function of the SGT System is to ensure that the release of radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) is minimized by filtration and adsorption prior to exhausting to the environment.

The SGT System consists of a suction duct, two parallel and independent filter trains with associated blowers, valves and controls, and a discharge vent.

Each filter train consists of (components listed in order of the direction of the air flow):

a. A moisture separator;

b. An electric heater;

c. A prefilter;

d. A high efficiency particulate air (HEPA) filter;

e. Two in-line charcoal adsorber beds;

f. A second HEPA filter; and

g. A centrifugal fan.

The SGT System is designed to restore and maintain secondary containment at a negative pressure of at least 0.25 inches water gauge relative to the atmosphere following a secondary containment isolation signal. Maintaining this negative pressure is based on a SGT System flow rate of at least 3000 cfm. A secondary containment negative pressure of 0.25 inches water gauge minimizes the release of radioactivity from secondary containment by ensuring primary containment leakage is treated prior to release.

The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 1). The prefilter removes large particulate matter, while the HEPA

(continued)

BASES

BACKGROUND (continued)	filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber beds remove gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber. The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following an initiation signal, both SGT charcoal filter train fans start.
APPLICABLE SAFETY ANALYSES	The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Refs. 2, 3, and 4). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment. The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).
LCO	Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

(continued)

BASES (continued)

ACTIONS

A.1

With one SGT subsystem inoperable in MODE 1, 2, or 3, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

In MODE 1, 2, or 3, if one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time or both SGT subsystems are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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

With one SGT subsystem inoperable during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs, the inoperable subsystem must be restored to OPERABLE status in 31 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 31 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the probability and consequences of an event requiring the radioactivity release control function during this period.

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, D.2.2, and D.2.3

During movement of irradiated fuel assemblies, in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action C.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order 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 in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

E.1, E.2, and E.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

applicable, actions must immediately be initiated to suspend OPDRVs in order 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 irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, by initiating (from the control room) flow through the HEPA filters and charcoal adsorbers, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on automatic control for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6), except as specified in Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)". The VFTP includes testing HEPA

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.2 (continued)

filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). It is noted that, per the basis provided by ESR 99-00055 (Ref. 7), system flow rate is determined using installed calibrated flow orifice plates. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has demonstrated that these components will usually pass the Surveillance when performed at the 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.5.1.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. UFSAR Section 15.6.4.
4. UFSAR Section 15.7.1.
5. 10 CFR 50.36(c)(2)(ii).
6. Regulatory Guide 1.52, Revision 1.
7. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.

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

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

The UFSAR (Ref. 1), requires means for detecting RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when LEAKAGE rates above normal background levels are detected and also to supply quantitative measurement of LEAKAGE rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two independently monitored variables, such as drywell floor drain sump flow changes and drywell gaseous or particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump flow monitoring system.

The drywell floor drain sump flow monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges, floor drains, the Reactor Building Closed Cooling Water System, and drywell cooler drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump is provided with two sump pumps. A flow transmitter in the common discharge line of the drywell floor drain sump pumps inputs to a flow integrator. In addition to the required instrumentation, the starting frequency and run duration of a sump pump motor are monitored by timer circuitry to provide a signal (alarm) in the control room indicating that LEAKAGE has reached a specified limit.

(continued)

BASES

BACKGROUND (continued)

The primary containment atmosphere radioactivity monitoring systems (particulate and gaseous) continuously monitor the primary containment atmosphere for airborne particulate and gaseous radioactivity. The primary containment atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying LEAKAGE rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times. A significant increase of radioactivity, which may be attributed to a sudden increase in RCPB steam or reactor water LEAKAGE, is annunciated in the control room.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 3 and 4). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm and/or indication of excess LEAKAGE in the control room.

A control room alarm/indication allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 5). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

The one channel of drywell floor drain sump flow monitoring system is required to quantify the unidentified LEAKAGE from the RCS. The required drywell floor drain sump flow monitoring system instrumentation includes the flow transmitter and integrator, as well as a flow totalizer. One channel of the other monitoring systems (particulate or gaseous) provides early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump flow monitoring system inoperable, no other required instrumentation can provide the equivalent information to quantify LEAKAGE. However, the primary containment atmosphere radioactivity monitor will provide indication of changes in LEAKAGE.

With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 8 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump flow monitoring system is inoperable. This allowance is provided because other instrumentation (listed in Reference 1) is available to monitor RCS LEAKAGE.

B.1 and B.2

With both gaseous and particulate primary containment atmosphere radioactivity monitoring channels inoperable (i.e., the required primary containment atmosphere monitoring system), grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic LEAKAGE information. Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmosphere radioactivity monitoring channels are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be met, 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 MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmosphere radioactivity monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies, for the radioactivity monitoring channels only, the required alarm

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.2 (continued)

setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is consistent with the Brunswick refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 5.2.5.
2. Regulatory Guide 1.45, May 1973.
3. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through—Wall Flaws, April 1968.
4. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.
5. UFSAR, Section 5.2.5.2.2.
6. 10 CFR 50.36(c)(2)(ii).

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.

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. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 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 III, Appendix G (Ref. 2).

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

(continued)

BASES

BACKGROUND (continued)

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 P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

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 1 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 Figures 3.4.9-3 or 3.4.9-4, 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

LCO (continued)

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODES 1, 2, and 3 must be corrected so that the RCPB is returned to a condition that has been verified as safe by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation within acceptable limits, an engineering evaluation is required to determine if RCS operation can continue. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, 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 vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 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 A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

(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 Figure 3.4.9-3 or 3.4.9-4 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 (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 and SR 3.4.9.2

Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

SR 3.4.9.1 is modified by a Note that requires the Surveillance to be performed only during system heatup and cooldown operations. SR 3.4.9.2 is modified by a Note that requires the Surveillance to be performed only during inservice leakage and hydrostatic testing.

SR 3.4.9.3

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.4 and SR 3.4.9.5

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.4 and SR 3.4.9.5 (continued)

Performing the Surveillance within 30 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the differential temperature requirement of SR 3.4.9.4 is to compare the temperature of the reactor coolant in the dome to the bottom head drain temperature.

As specified in procedures, an acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.5 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.4 and SR 3.4.9.5 are modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Note also states the SR is only required to be met during recirculation pump startup, since this is when the stresses occur.

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

(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. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, 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," dated November 8, 2000.
10. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)	Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. PCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).
LCO	PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA. The power operated, automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. Primary containment purge and vent valves > 8 inches must be blocked to prevent opening > 50° (approximately 55%). While the reactor building-to-suppression chamber vacuum breakers isolate primary containment penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO 3.6.1.5, "Reactor Building-to-Suppression Chamber Vacuum Breakers." The valves covered by this LCO are listed with their associated stroke times in Reference 4. The normally closed PCIVs are considered OPERABLE when any one of the following conditions is met: (1) manual valves are closed or opened in accordance with appropriate administrative controls; (2) automatic valves are de-activated and secured in their closed position; (3) blind flanges are in place, or (4) closed systems are intact. MSIVs are exempt from Type C testing limits and must meet specific leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing. This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.7

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) is OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break signal. This may be accomplished by cycling the EFCVs through one complete cycle of full travel. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 12). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.9

The analyses in References 2 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be $\leq$ 11.5 scfh when tested at $\geq P_t$ (25 psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of the Primary Containment Leakage Rate Testing Program. The Primary Containment Leakage Rate Testing Program has been established in accordance with 10 CFR 50.54(o) to implement the requirements of 10 CFR Part 50, Appendix J, Option B (Ref. 6), and conforms with Regulatory Guide 1.163 (Ref. 7) and Nuclear Energy Institute (NEI) 94-01 (Ref. 8) except for the following:

a. BNP may use standard glass tube and ball type flowmeters with an accuracy of 5% of full scale. This is an exception to the flowmeter accuracy requirements of ANSI/ANS 56.8-1994 (Ref. 9) referenced in NEI 94-01 (Ref. 8), Section 8.0. The basis for this exception is described in Reference 10.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

b. Local leak rate testing of the MSIVs may be performed at a pressure less than P_a. This is an exemption from the requirements of 10 CFR 50 Appendix J (Ref. 6). The basis for this exemption is described in Reference 11.

The Frequency is required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. UFSAR, Chapter 15.

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

3. 10 CFR 50.36(c)(2)(ii).

4. Technical Requirements Manual.

5. UFSAR, Section 15.2.3.

6. 10 CFR 50, Appendix J, Option B.

7. NRC Regulatory Guide 1.163, Performance-Based Containment Leak-Rate Testing Program, September 1995.

8. Nuclear Energy Institute (NEI) 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J, July 26, 1995.

9. ANSI/ANS 56.8-1994.

10. NRC SER; Issuance of Amendment No. 181 to Facility Operating License No. DPR-71 and Amendment No. 213 to Facility Operating License No. DPR-62 Regarding 10 CFR 50 Appendix J, Option B - Brunswick Steam Electric Plant, Units 1 and 2 (BSEP 95-0316) (TAC Nos. M93679 and M93680); dated February 1, 1996.

(continued)

BASES

REFERENCES (continued)

11. NRC SER, Brunswick 1 & 2 - Amendments No. 10 and 36 to Operating Licenses Revising Technical Specifications to Grant Exemptions from Specific Requirements of 10 CFR 50 Appendix J, dated November 8, 1977.

12. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 and 2 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Refs. 1 and 2). Among the inputs to the design basis analysis is the initial drywell average air temperature (Refs. 1 and 2). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 300°F (Ref. 3). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 8 hour Completion Time is acceptable, considering the sensitivity of the analysis to allow significant variations in this parameter, and provides sufficient time to correct minor problems.

B.1 and B.2

If the drywell average air temperature cannot be restored to within the limit 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. Drywell air temperature is monitored in all quadrants and at various elevations (referenced to mean sea level). Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1 (continued)

The following locations are monitored to obtain the drywell average temperature:

a. Below 5 ft elevation;

b. Between 10 ft and 23 ft elevation;

c. Between 28 ft and 45 ft elevation;

d. Between 70 ft and 80 ft elevation; and

e. Above 90 ft elevation.

The 24 hour Frequency of the SR is based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal drywell air temperature condition.

REFERENCES

1. UFSAR, Section 6.2.

2. GE-NE-T23-00735-01, Brunswick Steam Electric Plant Units 1 and 2 High Drywell Bulk Average Temperature Analysis, October 1996.

3. UFSAR, Section 6.2.1.1.1.

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Reactor Building-to-Suppression Chamber Vacuum Breakers

BASES

BACKGROUND

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression chamber-to-drywell vacuum breakers. The design of the external (reactor building-to-suppression chamber) vacuum relief provisions consists of two vacuum breakers (a mechanical vacuum breaker and a pneumatically operated butterfly valve), located in series in each of two 20 inch lines. The two lines from the reactor building merge to a common 20 inch line which connects to the suppression chamber airspace. Each path is capable of relieving 100% of design flow. The butterfly valve is actuated by a differential pressure switch. The normal pneumatic supply for the butterfly valve is the Non-interruptible Instrument Air System. A Nitrogen Backup System is provided to each butterfly valve and is automatically aligned to the valves following a loss of coolant accident (LOCA) signal and a subsequent primary containment isolation or following a loss of offsite power. Additionally, the Nitrogen Backup System automatically aligns to the valves to maintain system pressure when the Non-interruptible Instrument Air System pressure drops to approximately 95 psig. The mechanical vacuum breaker is self actuating similar to a check valve. Both mechanical vacuum breakers can be locally operated for testing purposes. The two vacuum breakers in series must be closed to maintain a leak tight primary containment boundary.

The Nitrogen Backup System is of safety grade quality and complies with the intent of Generic Letter 84-09 (Ref. 1). This system consists of two independent and redundant subsystems. Each subsystem supplies safety grade nitrogen from a nitrogen bottle rack to one pneumatic butterfly valve via a pressure control valve and has sufficient capacity to provide 22 hours of valve operation including design system leakage. The pressure control valve reduces the nitrogen bottle supply pressure of $\geq$ 1130 psig to a normal system pressure of approximately 95 psig.

(continued)

BASES

BACKGROUND (continued)

A negative differential pressure across the drywell wall is caused by depressurization of the drywell. Events that cause this depressurization are cooling cycles, inadvertent primary containment spray actuation, and steam condensation in the event of a primary system rupture. Reactor building-to-suppression chamber vacuum breakers prevent an excessive negative differential pressure across the primary containment boundary. Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation following a small break LOCA results in a significant pressure transient and becomes important in sizing the external (reactor building-to-suppression chamber) vacuum breakers.

The external vacuum breakers are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere. Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 2 as part of the accident response of the containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 0.5 psid (Ref. 2). Additionally, of the two series reactor building-to-suppression chamber vacuum breakers, one is assumed to fail in a closed position to satisfy the single active failure criterion. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight with positive primary containment pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers:

a. A small break loss of coolant accident followed by actuation of both primary containment spray loops and

b. A large break loss of coolant accident followed by actuation of both primary containment spray loops.

The results of these two cases show that the external vacuum breakers, with a full open setpoint of 0.5 psid, are capable of maintaining the differential pressure within design limits.

The reactor building-to-suppression chamber vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two vacuum breakers (mechanical vacuum breaker and pneumatic butterfly valve) in each of the two lines from the reactor building to the common line connected to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber. For a pneumatic butterfly valve to be OPERABLE for opening, both the Non-interruptible Instrument Air System and the Nitrogen Backup System shall be capable of supplying the pneumatic operator.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell atmosphere and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which, after the suppression chamber-to-drywell vacuum breakers open (due to differential pressure between the suppression chamber and

(continued)

BASES

APPLICABILITY (continued)

drywell), would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor building-to-suppression chamber vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

A Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each of the two lines from the reactor building to the common line connected to the suppression chamber air space.

With one or more lines with one vacuum breaker not closed, the leak tight primary containment boundary may be threatened. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status or the open vacuum breaker closed within 72 hours. The 72 hour Completion Time is consistent with requirements for inoperable suppression chamber-to-drywell vacuum breakers in LCO 3.6.1.6, "Suppression Chamber-to-Drywell Vacuum Breakers." The 72 hour Completion Time takes into account the redundant capability afforded by the remaining breakers, the fact that the OPERABLE breaker in each of the lines is closed, and the low probability of an event occurring that would require the vacuum breaker to be OPERABLE during this period.

B.1

With one or more lines with two vacuum breakers not closed, primary containment integrity is not maintained. Therefore, one open vacuum breaker in each line must be closed within 2 hours. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 2 hours.

(continued)

BASES

ACTIONS (continued)

C.1

With one vacuum breaker inoperable solely due to its associated nitrogen backup subsystem being inoperable, the leak tight primary containment boundary is intact. In this Condition, the vacuum breakers in the redundant line are adequate to mitigate the primary containment depressurization. However, overall reliability is reduced because a single failure could result in a loss of the capability to mitigate an event that causes a containment depressurization following a LOCA and a subsequent primary containment isolation. The 31 day Completion Time is acceptable because of the OPERABLE vacuum breakers in the redundant line, the normal pneumatic supply is available to the vacuum breaker, and the low probability of a LOCA and a subsequent primary containment isolation occurring during the period the nitrogen backup subsystem is inoperable.

D.1

With two vacuum breakers inoperable solely due to their associated nitrogen backup subsystems being inoperable, the leak tight primary containment boundary is intact. Since the normal pneumatic supply is available to each vacuum breaker, the vacuum breakers are still capable of mitigating any event that causes a containment depressurization except following a LOCA and a subsequent primary containment isolation. The 7 day Completion Time is acceptable because the normal pneumatic supply is available to each vacuum breaker and because of the low probability of a LOCA and a subsequent primary containment isolation occurring during the period the nitrogen backup subsystems are inoperable.

E.1

With one line with one or more vacuum breakers inoperable for opening, the leak tight primary containment boundary is intact. However, with one line with one or more vacuum breakers inoperable for opening for reasons other than its associated nitrogen backup subsystem being inoperable (Condition C), overall system reliability is reduced because a single failure in one of the vacuum breakers in the

(continued)

BASES

ACTIONS

E.1 (continued)

redundant line could threaten the ability to mitigate an event that causes a containment depressurization. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status within 72 hours. This is consistent with the Completion Time for Condition A and the fact that the leak tight primary containment boundary is being maintained.

F.1

With two lines with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the function of the vacuum breakers is lost if the vacuum breakers are inoperable for reasons other than the Nitrogen Backup System being inoperable (Condition D). Therefore, all vacuum breakers in one line must be restored to OPERABLE status within 2 hours. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 2 hours.

G.1 and G.2

If any Required Action and associated Completion Time cannot be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

The bank of nitrogen bottles supplying each nitrogen backup subsystem header is required to be verified to be pressurized to ≥ 1130 psig to ensure sufficient motive force is available to the pneumatic butterfly valve actuators following a LOCA and subsequent primary containment isolation. A nitrogen bottle pressure of ≥ 1130 psig assures sufficient capacity to actuate and cycle the pneumatic butterfly valve for 22 hours including design system leakage. This Surveillance may be satisfied by

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1 (continued)

verifying the absence of the Nitrogen Backup System low pressure alarms. The 24 hour Frequency is based on engineering judgment in view of the fact that adequate indication of pressure is available to the operator and the Frequency has also been shown to be acceptable through operating experience.

SR 3.6.1.5.2

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor building-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

SR 3.6.1.5.3

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This SR ensures that the safety analysis assumptions are valid. This is accomplished by manually verifying that each mechanical vacuum breaker is free to open and verifying each pneumatic butterfly valve operates through at least one complete cycle of full travel. The 92 day Frequency of this SR was developed based upon Inservice Testing Program requirements to perform valve testing at least once every 92 days.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.5.4

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of ≤ 0.5 psid is valid. This is accomplished by demonstrating that the force required to open each mechanical vacuum breaker is ≤ 0.5 psid and demonstrating that each pneumatic butterfly valve opens at ≥ 0.4 psid and ≤ 0.5 psid with suppression chamber pressure negative with respect to reactor building pressure. The 24 month Frequency has been demonstrated to be acceptable, based on operating experience, and is further justified because of other Surveillances performed more frequently that convey the proper functioning status of each vacuum breaker.

SR 3.6.1.5.5

To ensure the pneumatic butterfly valves have sufficient capacity to actuate and cycle following a LOCA and subsequent primary containment isolation, Nitrogen Backup System leakage must be within the design limit. This SR ensures that overall system leakage is within a design limit of 0.65 scfm. This is accomplished by measuring the nitrogen bottle supply pressure decrease while maintaining approximately 95 psig to the nitrogen backup subsystem during the test with an initial nitrogen bottle supply pressure of ≥ 1130 psig. The system leakage test is performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is concluded to be acceptable from a reliability standpoint.

SR 3.6.1.5.6

This SR ensures that in the event a LOCA and subsequent primary containment isolation occurs, the Nitrogen Backup System will actuate to perform its design function and supply nitrogen gas at the required pressure to the pneumatic operators of the butterfly valves. The 24 month

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.6 (continued)

Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NRC Generic Letter GL 84-09, Recombiner Capability requirements of 10 CFR 50.44(c)(3)(ii).
2. UFSAR, Section 6.2.
3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

BACKGROUND

The function of the suppression-chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 10 internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber, which allow flow from the suppression chamber atmosphere to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the suppression chamber-drywell boundary. Each vacuum breaker is a self actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

A negative differential pressure across the drywell wall is caused by depressurization of the drywell. Events that cause this depressurization are cooling cycles, inadvertent drywell spray actuation, and steam condensation from sprays or subcooled water reflood of a break in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or spill of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, the drywell atmosphere is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow from a recirculation line break, or drywell spray actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is less than the suppression chamber pressure, there will be an

(continued)

BASES

BACKGROUND (continued)	increase in the height of the downcomer waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.
APPLICABLE SAFETY ANALYSES	Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary. The safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid (Ref. 1). Additionally, 3 of the 10 internal vacuum breakers are assumed to fail in a closed position (Ref. 1). The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and the requirement that 8 of 10 vacuum breakers be OPERABLE (the additional vacuum breaker is required to meet the single failure criterion) are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. The total cross sectional area of the main vent system between the drywell and suppression chamber needed to fulfill this requirement has been established as a minimum of 51.5 times the total break area. In turn, the vacuum relief capacity between the drywell and suppression chamber should be 1/16 of the total main vent cross sectional area, with the valves set to operate at $\leq$ 0.5 psid differential pressure. Design Basis Accident (DBA) analyses assume the vacuum breakers to be closed initially and to remain closed and leak tight, until the suppression pool is at a positive pressure relative to the drywell. The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

Only 8 of the 10 vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers, however, are required to be closed (except when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell atmosphere and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one of the required vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining seven OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one of the eight required vacuum breakers inoperable, 72 hours is allowed to restore at least one of the inoperable vacuum breakers to OPERABLE status so that

(continued)

BASES

ACTIONS

A.1 (continued)

plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

B.1

With one vacuum breaker not closed, communication between the drywell and suppression chamber airspace could occur, and, as a result, there is the potential for primary containment overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not available, an alternate method of verifying that the vacuum breakers are closed is to verify that the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 8 hour Completion Time is considered adequate to perform this test.

C.1 and C.2

If any Required Action and associated Completion Time can not be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified closed (except when the vacuum breaker is performing its intended design function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

the differential pressure between the suppression chamber and drywell is maintained > 0.5 times the initial differential pressure for 1 hour without nitrogen makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel and procedural controls to ensure the drywell is normally maintained at a higher pressure than the suppression chamber, and has been shown to be acceptable through operating experience. This verification is also required within 2 hours after any discharge of steam to the suppression chamber from any source.

A Note is added to this SR which allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers.

SR 3.6.1.6.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This is accomplished by verifying each required vacuum breaker operates through at least one complete cycle of full travel. This SR ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR was developed, based on Inservice Testing Program requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after a discharge of steam to the suppression chamber from any source.

SR 3.6.1.6.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of 0.5 psid is valid. The 24 month Frequency is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.6.1.6.3</u> (continued) need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency has been demonstrated to be acceptable, based on operating experience, and is further justified because of other surveillances performed more frequently that convey the proper functioning status of each vacuum breaker.
REFERENCES	1. UFSAR, Section 6.2. 2. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven systems (i.e., the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System). Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation - the original limit for the end of a LOCA blowdown was 170°F, based on the Bodega Bay and Humboldt Bay Tests;

b. Primary containment peak pressure and temperature - design pressure is 62 psig and design temperature is 340°F (Ref. 1);

c. Condensation oscillation loads - maximum allowable initial temperature is 110°F; and

d. Chugging loads - these only occur at < 135°F; therefore, there is no initial temperature limit because of chugging.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is a spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (Reference 1 for LOCAs and for the pool temperature analyses required by Reference 2). An initial pool temperature of 95°F is assumed for the Reference 1 containment analyses. Reactor shutdown at a pool temperature of 110°F and vessel depressurization at a pool temperature of 120°F are assumed for the Reference 1 analyses. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during unit testing.

Suppression pool average temperature satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are:

a. Average temperature ≤ 95°F with THERMAL POWER > 1% RATED THERMAL POWER (RTP) and no testing that adds heat to the suppression pool is being performed. This requirement ensures that licensing bases initial conditions are met.

b. Average temperature ≤ 105°F with THERMAL POWER > 1% RTP and testing that adds heat to the suppression pool is being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the 110°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ 95°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > 95°F is short enough not to cause a significant increase in unit risk.

(continued)

BASES

LCO (continued)

c. Average temperature ≤ 110°F with THERMAL POWER ≤ 1% RTP. This requirement ensures that the unit will be shut down at > 110°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down.

At 1% RTP, heat input is approximately equal to normal system heat losses.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

A.1 and A.2

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power, the initial conditions exceed the conditions assumed for the Reference 1 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above those assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to allow the suppression pool average temperature to be restored below the limit. Additionally, when suppression pool temperature is > 95°F, increased monitoring of the suppression pool temperature is required to ensure that it remains ≤ 110°F. The once per hour Completion Time is adequate based on past experience, which has shown that pool temperature increases relatively slowly except when testing that adds heat to the suppression pool is being performed. Furthermore, the once per hour Completion Time is considered adequate in view of other indications in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

(continued)

BASES

ACTIONS (continued)

B.1

If the suppression pool average temperature cannot be restored to within limits 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 power must be reduced to ≤ 1% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce power from full power conditions in an orderly manner and without challenging plant systems.

C.1

Suppression pool average temperature is allowed to be > 95°F with THERMAL POWER > 1% RTP, and when testing that adds heat to the suppression pool is being performed. However, if temperature is > 105°F, all testing must be immediately suspended to preserve the heat absorption capability of the suppression pool. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

D.1 and D.2

Suppression pool average temperature > 110°F requires that the reactor be shut down immediately. This is accomplished by manually scramming the reactor. Further cooldown to Mode 4 within 36 hours is required at normal cooldown rates (provided pool temperature remains ≤ 120°F). Additionally, when suppression pool temperature is > 110°F, increased monitoring of pool temperature is required to ensure that it remains ≤ 120°F. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ 120°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 200 psig within 12 hours, and the plant must be brought to at least 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.

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. The average temperature is determined using an algorithm with inputs from OPERABLE suppression pool water temperature channels. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
2. NUREG-0783.
3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the steam exhaust lines in the turbine driven systems (i.e., High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System) and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 86,450 ft^3 at the low water level limit of -31 inches and 89,750 ft^3 at the high water level limit of -27 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the SRV quenchers, main vents, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from SRV discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to SRV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of References 1 and 2 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

A limit that suppression pool water level be ≥ -31 inches and ≤ -27 inches is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS-Shutdown."

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as main vents are covered, HPCI and RCIC turbine exhausts are covered, and SRV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the Drywell Spray System. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)

BASES

ACTIONS (continued)	B.1 and B.2 If suppression pool water level cannot be restored to within limits 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.
SURVEILLANCE REQUIREMENTS	SR 3.6.2.2.1 Verification of the suppression pool water level is to ensure that the required limits are satisfied. The 24 hour Frequency of this SR has been shown to be acceptable based on operating experience. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.
REFERENCES	1. UFSAR, Section 6.2.1.1.3.2. 2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995. 3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each RHR subsystem contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. Service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink.

The heat removal capability of two RHR pumps in one subsystem is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (SRV). With only one RHR pump in one subsystem OPERABLE, the available heat removal capability results in suppression pool temperatures, after a DBA LOCA, for which inadequate NPSH would be available for required RHR and Core Spray pumps. Therefore, to ensure adequate NPSH is available for required RHR and Core Spray pumps, two RHR pumps are required to be OPERABLE in a subsystem. SRV leakage and High Pressure Coolant Injection System and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

References 1 and 2 contain the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The intent of the analyses is to demonstrate that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Refs. 1 and 2). To ensure that these requirements are met, two independent RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when two pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both a release of radioactive material to the primary containment and a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool

(continued)

BASES

ACTIONS

A.1 (continued)

cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

Required Action A.1 is modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one RHR suppression pool cooling subsystem is inoperable. This allowance is provided because of the redundant RHR suppression pool cooling capabilities afforded by the OPERABLE subsystem.

B.1

With two RHR suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to remove heat from primary containment are available.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

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 since the RHR suppression pool cooling mode is manually initiated. 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. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each RHR pump develops a flow rate ≥ 7700 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment pressure and temperature can be maintained below the design limits during a DBA (Ref. 2). The normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 4) is covered by the requirements of LCO 3.5.1, "ECCS—Operating." This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is 92 days.

REFERENCES

1. UFSAR, Section 6.2.1.1.3.2.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. 10 CFR 50.36(c)(2)(ii).
4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Oxygen Concentration

BASES

BACKGROUND

The primary containment is designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inert, that is, oxygen concentration < 4.0 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The capability to inert the primary containment and maintain oxygen < 4.0 v/o works together with the Containment Atmosphere Dilution System (LCO 3.6.3.2, "Containment Atmosphere Dilution (CAD) System") to provide redundant and diverse methods to mitigate events that produce hydrogen and oxygen. For example, an event that rapidly generates hydrogen from zirconium metal water reaction could result in excessive hydrogen in primary containment, but oxygen concentration will remain < 5.0 v/o and no combustion can occur. Long term generation of both hydrogen and oxygen from radiolytic decomposition of water may eventually result in a combustible mixture in primary containment if the initial primary containment oxygen concentration exceeded 4.0 v/o during operation in the applicable conditions. This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.

APPLICABLE SAFETY ANALYSES

The Reference 1 calculations assume that the primary containment is inerted when a Design Basis Accident (DBA) loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce combustible gas mixtures in the primary containment. Oxygen, which is subsequently generated by radiolytic decomposition of water, is diluted by the CAD System more rapidly than it is produced.

Primary containment oxygen concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

(continued)

BASES (continued)

LCO

The primary containment oxygen concentration is maintained < 4.0 v/o to ensure that an event that produces any amount of hydrogen and oxygen does not result in a combustible mixture inside primary containment.

APPLICABILITY

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen and oxygen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible during a scheduled power reduction to ≤ 15% RTP. As long as reactor power is ≤ 15% RTP, the potential for an event that generates significant hydrogen and oxygen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a scheduled power reduction ≤ 15% RTP, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

If oxygen concentration is ≥ 4.0 v/o at any time while operating in MODE 1, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration must be restored to < 4.0 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 4.0 v/o because of the availability of other hydrogen and oxygen mitigating systems (e.g., Containment Atmosphere Dilution System) and the low probability and long duration of an event that would generate significant amounts of hydrogen and oxygen occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1

If oxygen concentration cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, power must be reduced to ≤ 15% RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1

The primary containment must be determined to be inerted by verifying that oxygen concentration is < 4.0 v/o. The 7 day Frequency is based on the slow rate at which oxygen concentration can change and on other indications of abnormal conditions (which would lead to more frequent checking by operators in accordance with plant procedures). Also, this Frequency has been shown to be acceptable through operating experience.

REFERENCES

1. UFSAR, Section 6.2.5.

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Containment Atmosphere Dilution (CAD) System

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < 5.0 volume percent (v/o).

The CAD System is manually initiated and consists of two 100% capacity subsystems. Each subsystem consists of a common liquid nitrogen supply tank, an electric vaporizer, and connected piping to supply the drywell and suppression chamber volumes. The liquid nitrogen supply tank and electric vaporizers are common components which are shared between the CAD subsystems of the two units. Piping from the liquid nitrogen supply tank downstream of the vaporizers is split and routed to each unit. Each pipe to a particular unit is divided to provide the capability to supply nitrogen to both the drywell and the suppression chamber. The nitrogen storage tank contains ≥ 4350 gal, which is adequate for 30 days of CAD subsystem operation.

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

APPLICABLE SAFETY ANALYSES

To evaluate the potential for hydrogen and oxygen accumulation in primary containment following a LOCA, hydrogen and oxygen generation is calculated (as a function of time following the initiation of the accident). The assumptions stated in Reference 1 are used to maximize the amount of hydrogen and oxygen generated. The calculation confirms that when the mitigating systems are actuated in accordance with emergency operating procedures, the peak oxygen concentration in primary containment is < 5.0 v/o (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Hydrogen and oxygen may accumulate within primary containment following a LOCA as a result of:

a. A metal water reaction between the zirconium fuel rod cladding and the reactor coolant; or

b. Radiolytic decomposition of water in the Reactor Coolant System.

The CAD System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref.3).

LCO

The CAD System (two CAD subsystems) must be OPERABLE with an OPERABLE flow path capable of supplying nitrogen to the drywell. This ensures operation of at least one CAD subsystem in the event of a worst case single active failure. Operation of at least one CAD subsystem is designed to maintain primary containment post-LOCA oxygen concentration < 5.0 v/o for 30 days.

APPLICABILITY

In MODE 1 when primary containment oxygen concentration is required to be < 4.0 v/o (i.e., primary containment inerted) in accordance with LCO 3.6.3.1, "Primary Containment Oxygen Concentration," the CAD System is required to maintain the oxygen concentration within primary containment below the flammability limit of 5.0 v/o following a LOCA. This ensures that the relative leak tightness of primary containment is adequate and prevents damage to safety related equipment and instruments located within primary containment.

In MODE 1, when primary containment oxygen concentration is not required to be < 4.0 v/o in accordance with LCO 3.6.3.1, "Primary Containment Oxygen Concentration," and in MODE 2, the potential for an event that generates significant hydrogen and oxygen is low, the primary containment need not be inert, and the CAD System is not required to be OPERABLE. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a scheduled power reduction < 15% RTP (i.e., when primary containment oxygen concentration is not required to be < 4.0 v/o in accordance with LCO 3.6.3.1), is low enough that these "windows," when the primary containment is not inerted and the CAD System is not required to be OPERABLE, are also justified.

(continued)

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 30 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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.2 (continued)

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

SR 3.6.3.2.3

Cycling each power operated valve, excluding automatic valves, in the CAD System flow path through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. While this Surveillance may be performed with the reactor at power, the 24 month Frequency of the Surveillance is intended to be consistent with expected fuel cycle lengths. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Safety Guide 7, March 1971.
2. UFSAR, Section 6.2.5.3.2.1, Amendment No. 9.
3. 10 CFR 50.36(c)(2)(ii).
4. UFSAR, Table 6.2.4-1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Dampers (SCIDs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Refs. 1 and 2) and a fuel handling accident inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

(continued)

BASES (continued)

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components or irradiated fuel assemblies located in secondary containment, can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, at least one door in each access to the Reactor Building must be closed, and the sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) must be OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary 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.

(continued)

BASES

ACTIONS (continued)

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, C.2, and C.3

Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause 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. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is operable. Suspension of these activities 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 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 irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1 and SR 3.6.4.1.2

Verifying that secondary containment equipment hatches and one secondary containment access door in each access opening are closed ensures that the infiltration of outside air of such magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in each access opening is closed. The 24 month Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will establish and maintain a negative pressure in the secondary containment. This is confirmed by demonstrating that one SGT subsystem can maintain $\geq$ 0.25 inches of vacuum water gauge for 1 hour at a flow rate $\leq$ 3000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, this test is used to ensure secondary containment boundary integrity. Since this SR is a secondary containment test, it need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
2. UFSAR, Section 15.6.4.
3. UFSAR, Section 15.7.1.
4. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Dampers (SCIDs)

BASES

BACKGROUND	The function of the SCIDs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1, 2, and 3). Secondary containment isolation within the time limits specified for those isolation dampers designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary. The OPERABILITY requirements for SCIDs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of active (automatic) devices. Automatic SCIDs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.
APPLICABLE SAFETY ANALYSES	The SCIDs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Refs. 1 and 2) and a fuel handling accident inside secondary containment (Refs. 1 and 3). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIDs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Maintaining SCIDs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment. SCIDs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	SCIDs form a part of the secondary containment boundary. The SCID safety function is related to control of offsite radiation releases resulting from DBAs. The isolation dampers are considered OPERABLE when their associated accumulators are pressurized, their isolation times are within limits, and the dampers are capable of actuating on an automatic isolation signal. The dampers covered by this LCO, along with their associated stroke times, are listed in Reference 5.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIDs is required. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIDs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment. Moving irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3.
ACTIONS	The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

(continued)

BASES

ACTIONS (continued)

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCID. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIDs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCID.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCID inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic SCID, a closed manual damper, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration, and the probability of a DBA, which requires the SCIDs to close, occurring during this short time is very low.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that secondary containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time of once per 92 days is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to devices located in high radiation areas and allows them to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With two SCIDs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic damper, a closed manual damper, and a blind flange. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIDs to close, occurring during this short time, is very low.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation dampers. This clarifies that only Condition A is entered if one SCID is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, 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.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the 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 irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

Verifying that the isolation time of each automatic SCID is within limits, by cycling each SCID through one complete cycle of full travel and measuring the isolation time, is required to demonstrate OPERABILITY. The isolation time test ensures that the SCID will isolate in the required time period. The Frequency of this SR is once per 24 months. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.4.2.2

Verifying that each automatic SCID closes on a secondary containment isolation signal is required to minimize leakage of radioactive material from secondary containment following

(continued)

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.6.4.2.2</u> (continued)

a DBA or other accidents. This SR ensures that each automatic SCID will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
2. UFSAR, Section 15.6.4.
3. UFSAR, Section 15.7.1.
4. 10 CFR 50.36(c)(2)(ii).
5. Technical Requirements Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The function of the SGT System is to ensure that the release of radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) is minimized by filtration and adsorption prior to exhausting to the environment.

The SGT System consists of a suction duct, two parallel and independent filter trains with associated blowers, valves and controls, and a discharge vent.

Each filter train consists of (components listed in order of the direction of the air flow):

a. A moisture separator;

b. An electric heater;

c. A prefilter;

d. A high efficiency particulate air (HEPA) filter;

e. Two in-line charcoal adsorber beds;

f. A second HEPA filter; and

g. A centrifugal fan.

The SGT System is designed to restore and maintain secondary containment at a negative pressure of at least 0.25 inches water gauge relative to the atmosphere following a secondary containment isolation signal. Maintaining this negative pressure is based on a SGT System flow rate of at least 3000 cfm. A secondary containment negative pressure of 0.25 inches water gauge minimizes the release of radioactivity from secondary containment by ensuring primary containment leakage is treated prior to release.

The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 1). The prefilter removes large particulate matter, while the HEPA

(continued)

BASES

BACKGROUND (continued)	filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber beds remove gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber. The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following an initiation signal, both SGT charcoal filter train fans start.
APPLICABLE SAFETY ANALYSES	The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Refs. 2, 3, and 4). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment. The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).
LCO	Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

(continued)

BASES (continued)

ACTIONS

A.1

With one SGT subsystem inoperable in MODE 1, 2, or 3, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

In MODE 1, 2, or 3, if one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time or both SGT subsystems are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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

With one SGT subsystem inoperable during movement of irradiated fuel assemblies in secondary containment, during CORE ALTERATIONS, or during OPDRVs, the inoperable subsystem must be restored to OPERABLE status in 31 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 31 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the probability and consequences of an event requiring the radioactivity release control function during this period.

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BASES

ACTIONS (continued)

D.1, D.2.1, D.2.2, and D.2.3

During movement of irradiated fuel assemblies, in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action C.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order 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 in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

E.1, E.2, and E.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

applicable, actions must immediately be initiated to suspend OPDRVs in order 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 irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving 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 of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, by initiating (from the control room) flow through the HEPA filters and charcoal adsorbers, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on automatic control for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6), except as specified in Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)". The VFTP includes testing HEPA

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.2 (continued)

filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). It is noted that, per the basis provided by ESR 99-00055 (Ref. 7), system flow rate is determined using installed calibrated flow orifice plates. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has demonstrated that these components will usually pass the Surveillance when performed at the 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.5.1.
2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
3. UFSAR Section 15.6.4.
4. UFSAR Section 15.7.1.
5. 10 CFR 50.36(c)(2)(ii).
6. Regulatory Guide 1.52, Revision 1.
7. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.