Progress Energy

JUN 0 4 2004

SERIAL: BSEP 04-0065

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62
Submittal of Technical Specification Bases Changes

Ladies and Gentlemen:

In accordance with Technical Specification (TS) 5.5.10 for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., is submitting Revision 36 to the BSEP, Unit 1 TS Bases and Revision 33 to the BSEP, Unit 2 TS Bases.

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

Sincerely,

ETO'N

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

MAT/mat

Enclosures:

1. Summary of Revisions to Technical Specification Bases
2. Page Replacement Instructions
3. Unit 1 Technical Specification Bases Replacement Pages
4. Unit 2 Technical Specification Bases Replacement Pages

A001

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Summary of Revisions to Technical Specification Bases			
Revision	**Affected Unit**	**Date Implemented**	**Title/Description**
36[1] 33[1]	1 2	June 3, 2004	**Title:** Changes of Test Conditions for CAC-V16/17 Testing (TSB-2004-01) **Description:** This change corrects the bases for Surveillance Requirement 3.6.1.5.4, associated with the Reactor Building-to-Suppression Chamber Vacuum Breakers, to state that the differential pressure test is with respect to the drywell and reactor building, not the suppression chamber and reactor building.
			Title: Updating of UFSAR Citations in the Technical Specification Bases (TSB-2004-03) **Description:** These changes are administrative in nature and update various references in the Technical Specification (TS) bases to reflect Updated Final Safety Analysis Report numbering changes.
			Title: HPCI Function Suction and Transfer Update (TSB-2004-04) **Description:** This change updates B 3.3.5.1, "ECCS Instrumentation," and B 3.5.1, "ECCS - Operating," to better address High Pressure Coolant Injection (HPCI)

[1] Note: Revision 36 for Unit 1 and Revision 33 for Unit 2 incorporated four Bases change packages (i.e., TSB-2004-01, TSB-2004-03, TSB-2004-04, and TSB-2004-05).

Summary of Revisions to Technical Specification Bases			
Revision	**Affected Unit**	**Date Implemented**	**Title/Description**
			design basis for auto suction transfer from the Condensate Storage Tank to the suppression pool. In addition, the change clarifies/updates HPCI system design and licensing basis information.
			Title: Revising the CREV System Bases for Unfiltered Inleakage (TSB-2004-05) **Description:** This change updates TS Bases Section 3.7.3, "CREV System," to reflect the NRC approved control room unfiltered inleakage assumption of 10,000 cfm for the loss-of-coolant accident analysis.

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Unit 1 Technical Specification Bases
Replacement Pages

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.19 (continued)

APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.3) and core flow (SR 3.3.1.1.18), respectively.

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

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

REFERENCES

1. UFSAR, Section 7.2.
2. UFSAR, Chapter 15.0.
3. UFSAR, Section 7.2.2.
4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
5. 10 CFR 50.36(c)(2)(ii).
6. NEDO-23842, Continuous Control Rod Withdrawal in the Startup Range, April 18, 1978.
7. UFSAR, Section 5.2.2.
8. UFSAR, Appendix 5A.
9. UFSAR, Section 6.3.1.

(continued)

BASES

REFERENCES (continued)

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

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

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

13. UFSAR, Table 7-4.

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

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

16. NEDC-32410P-A, Supplement 1, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, November 1997.

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

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

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

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

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

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. For automatic swaps from CST suction to suppression pools suction, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. The sequence is intended to ensure that a pump suction flow path is continuously maintained during the transfer. Two level switches are used to detect high water level in the suppression pool and two level switches are used to detect low CST level. Actuation of any one switch will cause the automatic swap from CST suction to suppression pools suction.

The HPCI System provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level—High trip, at which time the HPCI turbine trips, which causes the turbine's stop valve and the injection valve to close. This variable is monitored by two transmitters, which are, in turn, connected to two trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a two-out-of-two logic to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level—Low Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level—Low Level 3; and confirmed Reactor Vessel Water Level—Low Level 1; and CS or RHR (LPCI Mode) Pump Discharge Pressure—High are all present and the ADS Timer has timed out. There are two transmitters for Reactor Vessel Water Level—Low Level 3 and one transmitter for confirmed Reactor Vessel Water Level—Low Level 1 in

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.e. Reactor Vessel Shroud Level (continued)

flow diversion occurs when reactor water level is below the Reactor Vessel Shroud Level.

Reactor Vessel Shroud Level signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Shroud Level Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of the Reactor Vessel Shroud Level Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the systems that the valves are opened for are not required to be OPERABLE in MODES 4 and 5 and are normally not used).

HPCI System

3.a. Reactor Vessel Water Level—Low Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. The Reactor Vessel Water Level—Low Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in Reference 2.

Reactor Vessel Water Level—Low Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a. Reactor Vessel Water Level—Low Level 2 (continued)

The Reactor Vessel Water Level—Low Level 2 Allowable Value is low enough to avoid a HPCI System start from normal reactor level transients (e.g., a reactor scram without the loss of feedwater flow) and is high enough to avoid ADS timer start and initiation of low pressure ECCS at Reactor Vessel Water Level—Low Level 3 during a small LOCA (up to 1" nominal) where HPCI provides the preferred source of makeup. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Four channels of Reactor Vessel Water Level—Low Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of ADS actuation. The Drywell Pressure—High Function is not assumed in accident or transient analyses. It is retained since it is a potentially significant contributor to risk.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure—High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level—High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level—High signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs) which precludes an unanalyzed event.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. Reactor Vessel Water Level—High (continued)

Reactor Vessel Water Level—High signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. Both Reactor Vessel Water Level—High signals are required in order to close the HPCI turbine stop valve. This ensures that no single instrument failure can preclude HPCI initiation.

The Reactor Vessel Water Level—High Allowable Value is high enough to avoid interfering with HPCI System operation during reactor water level recovery resulting from low reactor water level events and low enough to prevent flow from the HPCI System from overflowing into the MSLs. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of Reactor Vessel Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.d Condensate Storage Tank Level—Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is assumed to provide the protection described in References 2 and 6.

The Condensate Storage Tank Level—Low signal is initiated from two level switches. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level—Low Function

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

3.d Condensate Storage Tank Level—Low (continued)

Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST. With HPCI flow manually reduced to no more than 2000 gpm as the CST is being depleted, the allowable value is also sufficient to prevent any air entrainment that could cause pump damage during the time it takes for the suction transfer to be completed.

Two channels of the Condensate Storage Tank Level—Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.e. Suppression Chamber Water Level—High

Excessively high suppression pool water could impact operation of the HPCI and Reactor Core Isolation Cooling (RCIC) exhaust vacuum breakers resulting in an inoperable HPCI or RCIC System. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

The Function is assumed to actuate for the small line break events (up to 1" nominal) where HPCI is the preferred event response system.

The Suppression Chamber Water Level—High signal is initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Chamber Water Level—High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which the HPCI and RCIC exhaust vacuum breakers become inoperable. The Allowable Value is referenced from the suppression chamber water level zero. Suppression chamber water level zero is one inch below the torus centerline.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4.b, 5.b. ADS Timer

The purpose of the ADS Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The ADS Timer Function is assumed to be OPERABLE for the accident analyses of References 2 and 5 that require ECCS initiation and assume unavailability of the HPCI System.

There are two ADS Timer relays, one in each of the two ADS trip systems. The Allowable Value for the ADS Timer is chosen to be long enough to allow HPCI to start and avoid an inadvertent blowdown yet short enough so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the ADS Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Reactor Vessel Water Level—Low Level 1

The Reactor Vessel Water Level—Low Level 1 Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level—Low Level 3 signals. In order to prevent spurious initiation of the ADS due to spurious Level 3 signals, a Level 1 signal must also be received before ADS initiation commences.

Reactor Vessel Water Level—Low Level 1 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Allowable Value for

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Reactor Vessel Water Level—Low Level 1 (continued)

Reactor Vessel Water Level—Low Level 1 is selected at the RPS Level 1 scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases discussion of this Function. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of Reactor Vessel Water Level—Low Level 1 Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and RHR (LPCI Mode) Pump Discharge Pressure—High

The Pump Discharge Pressure—High signals from the CS and RHR pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure—High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in References 2 and 5 with an assumed HPCI unavailability. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one CS pump (both channels for the pump) indicate the high discharge pressure condition or two RHR pumps in one LPCI loop (one channel for each pump) indicate a high discharge pressure condition. The Pump Discharge Pressure—High Allowable Value is less than the pump discharge pressure when the pump is operating at all flow ranges and high enough to avoid any condition that results in a discharge

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic and simulated automatic operation for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed 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 that these components will usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDC-31624P, Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2), July 1990.
6. UFSAR, Section 9.2.6.2.
7. NEDC-30936-P-A, BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Parts 1 and 2, December 1988.

BASES

SURVIELLANCE REQUIREMENTS

SR 3.3.7.1.4 (continued)

While this surveillance can be 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 found to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.4.4.1.
2. UFSAR, Section 15.7.1
3. 10 CFR 50.36(c)(2)(ii).
4. GENE-770-06-1-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

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BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8</u> (continued)

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

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

REFERENCES

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

BASES

BACKGROUND (continued)

The ADS (Ref. 4) consists of 7 of the 11 SRVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems so that these subsystems can provide coolant inventory makeup. Each of the SRVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5 and 6. The required analyses and assumptions are defined in Reference 7. The results of these analyses are also described in Reference 8.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 10. For a large recirculation loop suction pipe break LOCA, failure of one 250 VDC power supply is considered the most severe failure. For a small break LOCA, one 250 VDC power supply failure was combined with a conservative

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APPLICABLE SAFETY ANALYSES (continued)	out of service assumption to bound the single failure combinations in a single analysis. In addition to failing HPCI, one CS pump and one LPCI pump (due to the power supply failure); two ADS valves were assumed out of service (Ref. 10). This combination results in an allowance for a single ADS valve failure with no additional analysis and it results in no accident mitigation credit being assumed for HPCI in any LOCA analysis. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage. The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 11).
LCO	Each ECCS injection/spray subsystem and six of seven ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems. With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 9 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 9. LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR shutdown cooling isolation pressure in MODE 3, if they are capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes the period when the required RHR pump is not operating and the period when the system is being realigned to or from the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.
APPLICABILITY	All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

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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 drywell 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

(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 6-11.

BASES

BACKGROUND (continued)	The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will slightly pressurize the control room to prevent infiltration of air from surrounding buildings. CREV System operation in maintaining control room habitability is discussed in the UFSAR, Sections 6.4 and 9.4, (Refs. 1 and 2, respectively).
APPLICABLE SAFETY ANALYSES	The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the design basis accident presented in the UFSAR (Ref. 3). The radiation/smoke protection mode of the CREV System is assumed (explicitly or implicitly) to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident. The radiological doses to control room personnel as a result of a DBA are summarized in Reference 3. Postulated single active failures that may cause the loss of outside or recirculated air from the control room are bounded by BNP radiological dose calculations for control room personnel. The CREV System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA if unfiltered leakage into the control room is > 10,000 cfm. The CREV System is considered OPERABLE when the individual components necessary to support the radiation protection mode are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated: a. Emergency recirculation fan is OPERABLE; b. HEPA filter and charcoal adsorber bank are not excessively restricting flow and are capable of performing their filtration and adsorption functions; and

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.3.3 (continued) pressure at a flow rate of ≤ 2200 cfm to the control room in the radiation/smoke protection mode. To adequately demonstrate the capability of a CREV subsystem to maintain positive pressure, no more than one control room supply fan may be in operation during performance of this test. The Frequency of 18 months on a STAGGERED TEST BASIS is based on the low probability of significant degradation of the control room boundary occurring between surveillances. SR 3.7.3.4 This SR verifies that on an actual or simulated initiation signal, each CREV subsystem starts and operates. This SR includes ensuring outside air flow is diverted to the HEPA filter and charcoal adsorber bank of each CREV subsystem. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.7.1 overlaps this SR to provide complete testing of the safety function. Operating experience has demonstrated that the components will usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was found to be acceptable from a reliability standpoint.
REFERENCES	1. UFSAR, Section 6.4. 2. UFSAR, Section 9.4. 3. UFSAR, Section 6.4.4.1. 4. 10 CFR 50.36(c)(2)(ii). 5. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.

BASES

BACKGROUND (continued)	Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. The starting sequence of all automatically connected loads needed to recover the unit or maintain it in a safe condition is provided in UFSAR, Table 8-7 (Ref. 4). Ratings for the DGs satisfy the requirements of Safety Guide 9 (Ref. 5). Each DG has the following ratings: a. 3500 kW—continuous; and b. 3850 kW—2000 hours.
APPLICABLE SAFETY ANALYSES	The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 6) and Chapter 15 (Ref. 7), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits"; Section 3.5, "Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System"; and Section 3.6, "Containment Systems." The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the onsite or offsite AC sources OPERABLE during accident conditions in the event of: a. An assumed loss of all offsite power; and b. A worst case single failure. AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).
LCO	Two Unit 1 and two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E Distribution System and four separate and independent DGs (1, 2, 3, and 4) ensure availability of the required power to shut down the reactor and maintain it in a safe

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

systems. Due to the shared configuration of certain systems (required to mitigate DBAs and transients) between BNP Units 1 and 2, all four DGs are required to be OPERABLE to supply power to these systems when either one or both units are in MODE 1, 2, or 3. In order to reduce the potential consequences associated with removing a required offsite circuit from service during the performance of this Surveillance, reduce consequences of a potential perturbation to the electrical distribution systems during the performance of this Surveillance, and reduce challenges to safety systems, while at the same time avoiding the need to shutdown both units to perform this Surveillance, Note 2 only precludes satisfying this Surveillance Requirement for DG 1 and DG 2 when Unit 1 is in MODE 1, 2, or 3. During the performance of this Surveillance with Unit 1 not in MODE 1, 2, or 3 and with Unit 2 in MODE 1, 2, or 3; the applicable ACTIONS of the Unit 1 and Unit 2 Technical Specifications must be entered if a required offsite circuit, DG 1, DG 2, or other supported Technical Specification equipment is rendered inoperable by the performance of this Surveillance. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. UFSAR, Section 8.3.1.2.
2. UFSAR, Sections 8.2 and 8.3.
3. NRC Diagnostic Evaluation Team Report for Brunswick Steam Electric Plant dated August 2, 1989, from J.M. Taylor (NRC) to S.H. Smith, Jr. (CP&L).
4. UFSAR, Table 8-7.
5. Safety Guide 9.
6. UFSAR, Chapter 6.
7. UFSAR, Chapter 15.
8. 10 CFR 50.36(c)(2)(ii).
9. Regulatory Guide 1.93, December 1974.
10. Generic Letter 84-15.

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Unit 2 Technical Specification Bases Replacement Pages

Unit 2 - Bases Book 1
Replacement Pages

BASES

TO

THE FACILITY OPERATING LICENSE DPR-62

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 2

CAROLINA POWER & LIGHT COMPANY

REVISION 33

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.19 (continued)

APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation drive flow properly correlate with THERMAL POWER (SR 3.3.1.1.3) and core flow (SR 3.3.1.1.18), respectively.

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

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

REFERENCES

1. UFSAR, Section 7.2.
2. UFSAR, Chapter 15.0.
3. UFSAR, Section 7.2.2.
4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
5. 10 CFR 50.36(c)(2)(ii).
6. NEDO-23842, Continuous Control Rod Withdrawal in the Startup Range, April 18, 1978.
7. UFSAR, Section 5.2.2.
8. UFSAR, Appendix 5A.
9. UFSAR, Section 6.3.1.

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REFERENCES (continued)

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

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

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

13. UFSAR, Table 7-4.

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

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

16. NEDC-32410P-A, Supplement 1, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, November 1997.

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

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

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

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

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

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. For automatic swaps from CST suction to suppression pools suction, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. The sequence is intended to ensure that a pump suction flow path is continuously maintained during the transfer. Two level switches are used to detect high water level in the suppression pool and two level switches are used to detect low CST level. Actuation of any one switch will cause the automatic swap from CST suction to suppression pools suction.

The HPCI System provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level—High trip, at which time the HPCI turbine trips, which causes the turbine's stop valve and the injection valve to close. This variable is monitored by two transmitters, which are, in turn, connected to two trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a two-out-of-two logic to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level—Low Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level—Low Level 3; and confirmed Reactor Vessel Water Level—Low Level 1; and CS or RHR (LPCI Mode) Pump Discharge Pressure—High are all present and the ADS Timer has timed out. There are two transmitters for Reactor Vessel Water Level—Low Level 3 and one transmitter for confirmed Reactor Vessel Water Level—Low Level 1 in

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.e. Reactor Vessel Shroud Level (continued)

flow diversion occurs when reactor water level is below the Reactor Vessel Shroud Level.

Reactor Vessel Shroud Level signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Shroud Level Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of the Reactor Vessel Shroud Level Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves that this Function isolates (since the systems that the valves are opened for are not required to be OPERABLE in MODES 4 and 5 and are normally not used).

HPCI System

3.a. Reactor Vessel Water Level—Low Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. The Reactor Vessel Water Level—Low Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in Reference 2.

Reactor Vessel Water Level—Low Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a. Reactor Vessel Water Level—Low Level 2 (continued)

The Reactor Vessel Water Level—Low Level 2 Allowable Value is low enough to avoid a HPCI System start from normal reactor level transients (e.g., a reactor scram without the loss of feedwater flow) and is high enough to avoid ADS timer start and initiation of low pressure ECCS at Reactor Vessel Water Level—Low Level 3 during a small LOCA (up to 1" nominal) where HPCI provides the preferred source of makeup. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Four channels of Reactor Vessel Water Level—Low Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of ADS actuation. The Drywell Pressure—High Function is not assumed in accident or transient analyses. It is retained since it is a potentially significant contributor to risk.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure—High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level—High

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Reactor Vessel Water Level—High signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs) which precludes an unanalyzed event.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. Reactor Vessel Water Level—High (continued)

Reactor Vessel Water Level—High signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. Both Reactor Vessel Water Level—High signals are required in order to close the HPCI turbine stop valve. This ensures that no single instrument failure can preclude HPCI initiation.

The Reactor Vessel Water Level—High Allowable Value is high enough to avoid interfering with HPCI System operation during reactor water level recovery resulting from low reactor water level events and low enough to prevent flow from the HPCI System from overflowing into the MSLs. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of Reactor Vessel Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.d Condensate Storage Tank Level—Low

Low level in the CST indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valves between HPCI and the CST are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CST. However, if the water level in the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes. The Function is assumed to provide the protection described in References 2 and 6.

The Condensate Storage Tank Level—Low signal is initiated from two level switches. The logic is arranged such that either level switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level—Low Function

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

3.d Condensate Storage Tank Level—Low (continued)

Allowable Value is high enough to ensure adequate pump suction head while water is being taken from the CST. With HPCI flow manually reduced to no more than 2000 gpm as the CST is being depleted, the allowable value is also sufficient to prevent any air entrainment that could cause pump damage during the time it takes for the suction transfer to be completed.

Two channels of the Condensate Storage Tank Level—Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.e. Suppression Chamber Water Level—High

Excessively high suppression pool water could impact operation of the HPCI and Reactor Core Isolation Cooling (RCIC) exhaust vacuum breakers resulting in an inoperable HPCI or RCIC System. Therefore, signals indicating high suppression pool water level are used to transfer the suction source of HPCI from the CST to the suppression pool to eliminate the possibility of HPCI continuing to provide additional water from a source outside containment. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CST suction valve automatically closes.

The Function is assumed to actuate for the small line break events (up to 1" nominal) where HPCI is the preferred event response system.

The Suppression Chamber Water Level—High signal is initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Chamber Water Level—High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which the HPCI and RCIC exhaust vacuum breakers become inoperable. The Allowable Value is referenced from the suppression chamber water level zero. Suppression chamber water level zero is one inch below the torus centerline.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

4.b, 5.b. ADS Timer

The purpose of the ADS Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The ADS Timer Function is assumed to be OPERABLE for the accident analyses of References 2 and 5 that require ECCS initiation and assume unavailability of the HPCI System.

There are two ADS Timer relays, one in each of the two ADS trip systems. The Allowable Value for the ADS Timer is chosen to be long enough to allow HPCI to start and avoid an inadvertent blowdown yet short enough so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the ADS Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Reactor Vessel Water Level—Low Level 1

The Reactor Vessel Water Level—Low Level 1 Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level—Low Level 3 signals. In order to prevent spurious initiation of the ADS due to spurious Level 3 signals, a Level 1 signal must also be received before ADS initiation commences.

Reactor Vessel Water Level—Low Level 1 signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Allowable Value for

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.c, 5.c. Reactor Vessel Water Level—Low Level 1 (continued)

Reactor Vessel Water Level—Low Level 1 is selected at the RPS Level 1 scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases discussion of this Function. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

Two channels of Reactor Vessel Water Level—Low Level 1 Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and RHR (LPCI Mode) Pump Discharge Pressure—High

The Pump Discharge Pressure—High signals from the CS and RHR pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure—High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in References 2 and 5 with an assumed HPCI unavailability. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling functions. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one CS pump (both channels for the pump) indicate the high discharge pressure condition or two RHR pumps in one LPCI loop (one channel for each pump) indicate a high discharge pressure condition. The Pump Discharge Pressure—High Allowable Value is less than the pump discharge pressure when the pump is operating at all flow ranges and high enough to avoid any condition that results in a discharge

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.3.5.1.5 The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic and simulated automatic operation for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed 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 that these components will usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 5.2.
2. UFSAR, Section 6.3.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDC-31624P, Brunswick Steam Electric Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (Revision 2), July 1990.
6. UFSAR, Section 9.2.6.2.
7. NEDC-30936-P-A, BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Parts 1 and 2, December 1988.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.3.7.1.4</u> (continued) While this surveillance can be 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 found to be acceptable from a reliability standpoint.
REFERENCES	1. UFSAR, Section 6.4.4.1. 2. UFSAR, Section 15.7.1 3. 10 CFR 50.36(c)(2)(ii). 4. GENE-770-06-1-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

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BASES

SURVEILLANCE REQUIREMENTS

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

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

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

REFERENCES

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

BASES

BACKGROUND (continued)

The ADS (Ref. 4) consists of 7 of the 11 SRVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems so that these subsystems can provide coolant inventory makeup. Each of the SRVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5 and 6. The required analyses and assumptions are defined in Reference 7. The results of these analyses are also described in Reference 8.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 10. For a large recirculation loop suction pipe break LOCA, failure of one 250 VDC power supply is considered the most severe failure. For a small break LOCA, one 250 VDC power supply failure was combined with a conservative

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BASES

APPLICABLE SAFETY ANALYSES (continued)

out of service assumption to bound the single failure combinations in a single analysis. In addition to failing HPCI, one CS pump and one LPCI pump (due to the power supply failure); two ADS valves were assumed out of service (Ref. 10). This combination results in an allowance for a single ADS valve failure with no additional analysis and it results in no accident mitigation credit being assumed for HPCI in any LOCA analysis. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 11).

LCO

Each ECCS injection/spray subsystem and six of seven ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 9 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 9.

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR shutdown cooling isolation pressure in MODE 3, if they are capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes the period when the required RHR pump is not operating and the period when the system is being realigned to or from the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is $\leq$ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

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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 drywell 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.6.3.2.2</u> (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. <u>SR 3.6.3.2.3</u> 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 6-11.

BASES

BACKGROUND (continued)	The CREV System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body. A single CREV subsystem will slightly pressurize the control room to prevent infiltration of air from surrounding buildings. CREV System operation in maintaining control room habitability is discussed in the UFSAR, Sections 6.4 and 9.4, (Refs. 1 and 2, respectively).
APPLICABLE SAFETY ANALYSES	The ability of the CREV System to maintain the habitability of the control room is an explicit assumption for the design basis accident presented in the UFSAR (Ref. 3). The radiation/smoke protection mode of the CREV System is assumed (explicitly or implicitly) to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident. The radiological doses to control room personnel as a result of a DBA are summarized in Reference 3. Postulated single active failures that may cause the loss of outside or recirculated air from the control room are bounded by BNP radiological dose calculations for control room personnel. The CREV System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	Two redundant subsystems of the CREV System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA if unfiltered leakage into the control room is > 10,000 cfm. The CREV System is considered OPERABLE when the individual components necessary to support the radiation protection mode are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated: a. Emergency recirculation fan is OPERABLE; b. HEPA filter and charcoal adsorber bank are not excessively restricting flow and are capable of performing their filtration and adsorption functions; and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.3 (continued)

pressure at a flow rate of ≤ 2200 cfm to the control room in the radiation/smoke protection mode. To adequately demonstrate the capability of a CREV subsystem to maintain positive pressure, no more than one control room supply fan may be in operation during performance of this test. The Frequency of 18 months on a STAGGERED TEST BASIS is based on the low probability of significant degradation of the control room boundary occurring between surveillances.

SR 3.7.3.4

This SR verifies that on an actual or simulated initiation signal, each CREV subsystem starts and operates. This SR includes ensuring outside air flow is diverted to the HEPA filter and charcoal adsorber bank of each CREV subsystem. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.7.1 overlaps this SR to provide complete testing of the safety function. Operating experience has demonstrated that the components will usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.4.
2. UFSAR, Section 9.4.
3. UFSAR, Section 6.4.4.1.
4. 10 CFR 50.36(c)(2)(ii).
5. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.

BASES

BACKGROUND (continued)	Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading of the DGs in the process. The starting sequence of all automatically connected loads needed to recover the unit or maintain it in a safe condition is provided in UFSAR, Table 8-7 (Ref. 4). Ratings for the DGs satisfy the requirements of Safety Guide 9 (Ref. 5). Each DG has the following ratings: a. 3500 kW—continuous; and b. 3850 kW—2000 hours.
APPLICABLE SAFETY ANALYSES	The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 6) and Chapter 15 (Ref. 7), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits"; Section 3.5, "Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System"; and Section 3.6, "Containment Systems." The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the onsite or offsite AC sources OPERABLE during accident conditions in the event of: a. An assumed loss of all offsite power; and b. A worst case single failure. AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).
LCO	Two Unit 1 and two Unit 2 qualified circuits between the offsite transmission network and the onsite Class 1E Distribution System and four separate and independent DGs (1, 2, 3, and 4) ensure availability of the required power to shut down the reactor and maintain it in a safe

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

systems. Due to the shared configuration of certain systems (required to mitigate DBAs and transients) between BNP Units 1 and 2, all four DGs are required to be OPERABLE to supply power to these systems when either one or both units are in MODE 1, 2, or 3. In order to reduce the potential consequences associated with removing a required offsite circuit from service during the performance of this Surveillance, reduce consequences of a potential perturbation to the electrical distribution systems during the performance of this Surveillance, and reduce challenges to safety systems, while at the same time avoiding the need to shutdown both units to perform this Surveillance, Note 2 only precludes satisfying this Surveillance Requirement for DG 3 and DG 4 when Unit 2 is in MODE 1, 2, or 3. During the performance of this Surveillance with Unit 2 not in MODE 1, 2, or 3 and with Unit 1 in MODE 1, 2, or 3; the applicable ACTIONS of the Unit 1 and Unit 2 Technical Specifications must be entered if a required offsite circuit, DG 3, DG 4, or other supported Technical Specification equipment is rendered inoperable by the performance of this Surveillance. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. UFSAR, Section 8.3.1.2.
2. UFSAR, Sections 8.2 and 8.3.
3. NRC Diagnostic Evaluation Team Report for Brunswick Steam Electric Plant dated August 2, 1989, from J. M. Taylor (NRC) to S. H. Smith, Jr. (CP&L).
4. UFSAR, Table 8-7.
5. Safety Guide 9.
6. UFSAR, Chapter 6.
7. UFSAR, Chapter 15.
8. 10 CFR 50.36(c)(2)(ii).
9. Regulatory Guide 1.93, December 1974.
10. Generic Letter 84-15.

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