CP&L

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MAY 1 1 2000

SERIAL: BSEP 00-0070

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

Gentlemen:

On June 5, 1998, the NRC issued license Amendments 203 and 233, which approved the conversion of the Technical Specifications (TS) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2 from the current Technical Specifications to the Improved Technical Specifications, as contained in Revision 1 of NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4." TS 5.5.10, "Technical Specifications (TS) Bases Control Program," Item d requires changes made to the TS bases, implemented without prior NRC approval, to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). In accordance with TS 5.5.10.d, Carolina Power & Light (CP&L) Company submits the revisions made to the BSEP TS Bases between May 7, 1999, and May 8, 2000.

This submittal does not contain any regulatory commitments. Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing, at (910) 457-2073.

Sincerely,

Warren J. Dorman
Manager - Regulatory Affairs
Brunswick Steam Electric Plant

MAT/mat

Enclosures:

1. Summary List of Revisions
2. Bases Page Replacement Instructions
3. Unit 1 and Unit 2 - Replacement Bases Pages

A001

cc (with enclosures):

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ENCLOSURE 1

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

Summary List of Revisions

Technical Specification 5.5.10, "Technical Specifications (TS) Bases Control Program," Item d requires changes made to the TS bases, implemented without prior NRC approval, to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). The following table summarizes the changes made to the TS bases for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2 between May 7, 1999, and May 8, 2000.

BSEP TS Bases Revisions
May 7, 1999, to May 8, 2000

Rev. No.	Date Incorp.	Title/Description
Unit 1 - Rev. 8 Unit 2 - Rev. 8	6/22/99	Cross-Tie Breaker Requirements Revision 8 for Units 1 and 2 modified the Limiting Condition for Operation (LCO) section of B 3.8.7 to clarify the requirement to maintain the cross-tie breakers between the redundant 4.16 kV Emergency Buses open when one or both units are in Modes 1, 2, or 3.
Unit 2 - Rev. 9	2/16/00	Main Turbine Bypass System Operability Revision 9 for Unit 2 modified B3.7.6, Action A.1 to state that three or more inoperable bypass valves render the Main Turbine Bypass system inoperable versus one or more inoperable bypass valves.
Unit 1 - Rev. 9 Unit 2 - Rev. 10	2/25/00	Nuclear Service Water Header Requirements During Shutdown Revision 9 for Unit 1 and Revision 10 for Unit 2 modified B 3.7.2, Surveillance Requirement (SR) 3.7.2.4 to clarify that a nuclear service water header, on a shutdown unit, may be out of service without declaring diesel generators inoperable. Safety Relief Valve (SRV) Test Pressure Revision 9 for Unit 1 and Revision 10 for Unit 2 modified B 3.4.3, SR 3.4.3.2 and B 3.5.1, SR 3.5.1.11 to identify that the adequate pressure for testing SRVs is 945 psig.

BSEP TS Bases Revisions May 7, 1999, to May 8, 2000		
Rev. No.	**Date Incorp.**	**Title/Description**
Unit 1 - Rev. 10	3/17/00	Elimination of Safety Limit MCPR Cycle Specific Restriction Revision 10 for Unit 1 modified the Safety Limits section of B 2.1.1 to eliminate a statement that the Unit 1 Safety Limit Minimum Critical Power Ratio values were cycle specific. This change was based on Amendment 207 for the Unit 1 Technical Specifications, which removed this cycle specific limitation.
Unit 1 - Rev. 11 Unit 2 - Rev. 11	3/31/00	Flowrate Measurement Clarification Revision 11 for Units 1 and 2 modified the bases for SR 3.6.4.3.2 and SR 3.7.3.2 to indicate that the flowrates for the these surveillances are determined using installed, calibrated flow orifice plates.

ENCLOSURE 2

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

Bases Page Replacement Instructions

Bases Unit 1 - B 2.0 - B 3.3 (Book 1)	
Remove	**Insert**
Title Page, Revision 7	Title Page, Revision 11
LOEP 1, Revision 7	LOEP 1, Revision 11
B 2.0-4, Revision 0	B 2.0-4, Revision 10

Bases Unit 1 - B 3.4 - B 3.10 (Book 2)	
Remove	**Insert**
LOEP 1, Revision 7	LOEP 1, Revision 11
LOEP 3, Revision 6	LOEP 3, Revision 11
LOEP 4, Revision 7	LOEP 4, Revision 8
B 3.4-14, Revision 0	B 3.4-14, Revision 9
B 3.5-16, Revision 0	B 3.5-16, Revision 9
B 3.6-84, Revision 0	B 3.6-84, Revision 11
B 3.7-19 through B 3.7-25, Revision 0	B 3.7-19 through B 3.7-25, Revision 9
B 3.7-26 through B 3.7-27, Revision 0	B 3.7-26 through B 3.7-27, Revision 11
B 3.7-28 through B 3.7-41, Revision 0	B 3.7-28 through B 3.7-42, Revision 9
B 3.8-75 through B 3.8-78, Revision 6	B 3.8-75 through B 3.8-78, Revision 8
B 3.8-79 through B 3.8-85, Revision 7	B 3.8-79 through B 3.8-85, Revision 8

Bases Unit 2 - B 2.0 - B 3.3 (Book 1)	
Remove	**Insert**
Title Page, Revision 7	Title Page, Revision 11
LOEP 1, Revision 7	LOEP 1, Revision 11

Bases Unit 2 - B 3.4 - B 3.10 (Book 2)	
Remove	**Insert**
LOEP 1, Revision 7	LOEP 1, Revision 11
LOEP 3, Revision 6	LOEP 3, Revision 11
LOEP 4, Revision 7	LOEP 4, Revision 8
B 3.4-14, Revision 0	B 3.4-14, Revision 10
B 3.5-16, Revision 0	B 3.5-16, Revision 10
B 3.6-84, Revision 0	B 3.6-84, Revision 11
B 3.7-19 through B 3.7-25, Revision 0	B 3.7-19 through B 3.7-25, Revision 10
B 3.7-26 through B 3.7-27, Revision 0	B 3.7-26 through B 3.7-27, Revision 11
B 3.7-28 through B 3.7-41, Revision 0	B 3.7-28 through B 3.7-42, Revision 10
B 3.8-75 through B 3.8-78, Revision 6	B 3.8-75 through B 3.8-78, Revision 8
B 3.8-79 through B 3.8-85, Revision 7	B 3.8-79 through B 3.8-85, Revision 8

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

Unit 1 and Unit 2
Replacement Bases Pages

Bases Unit 1 - B 2.0 - B 3.3 (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 11

LIST OF EFFECTIVE PAGES - BASES

Page No.	Revision No.	Page No.	Revision No.
Title Page	11	B 3.1-7	0
		B 3.1-8	0
List of Effective Pages - Book 1		B 3.1-9	0
		B 3.1-10	0
LOEP-1	11	B 3.1-11	0
LOEP-2	3	B 3.1-12	0
LOEP-3	5	B 3.1-13	0
LOEP-4	3	B 3.1-14	0
		B 3.1-15	0
i	0	B 3.1-16	0
ii	0	B 3.1-17	0
		B 3.1-18	0
B 2.0-1	0	B 3.1-19	0
B 2.0-2	0	B 3.1-20	0
B 2.0-3	0	B 3.1-21	0
B 2.0-4	10	B 3.1-22	0
B 2.0-5	0	B 3.1-23	0
B 2.0-6	0	B 3.1-24	0
B 2.0-7	0	B 3.1-25	0
B 2.0-8	0	B 3.1-26	0
		B 3.1-27	0
B 3.0-1	0	B 3.1-28	0
B 3.0-2	0	B 3.1-29	0
B 3.0-3	0	B 3.1-30	0
B 3.0-4	0	B 3.1-31	0
B 3.0-5	0	B 3.1-32	0
B 3.0-6	0	B 3.1-33	0
B 3.0-7	0	B 3.1-34	0
B 3.0-8	0	B 3.1-35	0
B 3.0-9	0	B 3.1-36	0
B 3.0-10	0	B 3.1-37	0
B 3.0-11	0	B 3.1-38	0
B 3.0-12	0	B 3.1-39	0
B 3.0-13	0	B 3.1-40	0
B 3.0-14	0	B 3.1-41	0
B 3.0-15	0	B 3.1-42	0
		B 3.1-43	0
B 3.1-1	0	B 3.1-44	0
B 3.1-2	0	B 3.1-45	0
B 3.1-3	0	B 3.1-46	0
B 3.1-4	0	B 3.1-47	0
B 3.1-5	0	B 3.1-48	0
B 3.1-6	0		

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. In conjunction with LCOs, the limiting safety system settings, defined in LCO 3.3.1.1 as the Allowable Values, establish the threshold for protective system action to prevent exceeding acceptable limits, including this reactor vessel water level SL, during Design Basis Accidents. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 2). Therefore, it is required

(continued)

Bases Unit 1 - B 3.4 - B 3.10 (Book 2)
Replacement Pages

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

(continued)

LIST OF EFFECTIVE PAGES - BASES (continued)

Page No.	Revision No.	Page No.	Revision No.
B 3.6-83	0	B 3.7-40	9
B 3.6-84	11	B 3.7-41	9
		B 3.7-42	9
B 3.7-1	0		
B 3.7-2	0	B 3.8-1	0
B 3.7-3	0	B 3.8-2	0
B 3.7-4	0	B 3.8-3	0
B 3.7-5	0	B 3.8-4	0
B 3.7-6	0	B 3.8-5	0
B 3.7-7	0	B 3.8-6	6
B 3.7-8	0	B 3.8-7	6
B 3.7-9	0	B 3.8-8	6
B 3.7-10	0	B 3.8-9	6
B 3.7-11	0	B 3.8-10	6
B 3.7-12	0	B 3.8-11	6
B 3.7-13	0	B 3.8-12	6
B 3.7-14	0	B 3.8-13	6
B 3.7-15	0	B 3.8-14	6
B 3.7-16	0	B 3.8-15	6
B 3.7-17	0	B 3.8-16	6
B 3.7-18	0	B 3.8-17	6
B 3.7-19	9	B 3.8-18	6
B 3.7-20	9	B 3.8-19	6
B 3.7-21	9	B 3.8-20	6
B 3.7-22	9	B 3.8-21	6
B 3.7-23	9	B 3.8-22	6
B 3.7-24	9	B 3.8-23	6
B 3.7-25	9	B 3.8-24	6
B 3.7-26	11	B 3.8-25	6
B 3.7-27	11	B 3.8-26	6
B 3.7-28	9	B 3.8-27	6
B 3.7-29	9	B 3.8-28	6
B 3.7-30	9	B 3.8-29	6
B 3.7-31	9	B 3.8-30	6
B 3.7-32	9	B 3.8-31	6
B 3.7-33	9	B 3.8-32	6
B 3.7-34	9	B 3.8-33	6
B 3.7-35	9	B 3.8-34	6
B 3.7-36	9	B 3.8-35	6
B 3.7-37	9	B 3.8-36	6
B 3.7-38	9	B 3.8-37	6
B 3.7-39	9	B 3.8-38	6
		B 3.8-39	6

(continued)

LIST OF EFFECTIVE PAGES - BASES (continued)

Page No.	Revision No.	Page No.	Revision No.
B 3.8-40	6	B 3.8-81	8
B 3.8-41	6	B 3.8-82	8
B 3.8-42	6	B 3.8-83	8
B 3.8-43	6	B 3.8-84	8
B 3.8-44	6	B 3.8-85	8
B 3.8-45	6	B 3.8-86	7
B 3.8-46	6	B 3.8-87	7
B 3.8-47	6	B 3.8-88	7
B 3.8-48	6	B 3.8-89	7
B 3.8-49	6	B 3.8-90	7
B 3.8-50	6	B 3.8-91	7
B 3.8-51	6		
B 3.8-52	6	B 3.9-1	0
B 3.8-53	6	B 3.9-2	0
B 3.8-54	6	B 3.9-3	0
B 3.8-55	6	B 3.9-4	0
B 3.8-56	6	B 3.9-5	0
B 3.8-57	6	B 3.9-6	0
B 3.8-58	6	B 3.9-7	0
B 3.8-59	6	B 3.9-8	0
B 3.8-60	6	B 3.9-9	0
B 3.8-61	6	B 3.9-10	0
B 3.8-62	6	B 3.9-11	0
B 3.8-63	6	B 3.9-12	0
B 3.8-64	6	B 3.9-13	0
B 3.8-65	6	B 3.9-14	0
B 3.8-66	6	B 3.9-15	0
B 3.8-67	6	B 3.9-16	0
B 3.8-68	6	B 3.9-17	0
B 3.8-69	6	B 3.9-18	0
B 3.8-70	6	B 3.9-19	0
B 3.8-71	6	B 3.9-20	0
B 3.8-72	6	B 3.9-21	0
B 3.8-73	6	B 3.9-22	0
B 3.8-74	6	B 3.9-23	0
B 3.8-75	8	B 3.9-24	0
B 3.8-76	8	B 3.9-25	0
B 3.8-77	8	B 3.9-26	0
B 3.8-78	8	B 3.9-27	0
B 3.8-79	8	B 3.9-28	0
B 3.8-80	8	B 3.9-29	0

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test. Adequate pressure at which this test is to be performed, to avoid damaging the valve, is 945 psig. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is considered OPERABLE.

The 24 month Frequency was developed based on the SRV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.2.2.2.

2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, Supplement 1, March 1996.

3. UFSAR, Chapter 15.

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

5. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.11

A manual actuation of each required ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the SRV discharge lines. This is demonstrated by the response of the turbine control or bypass valve; by a change in the measured flow; or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this SR. Adequate pressure at which this SR is to be performed, to avoid damaging the valve, is 945 psig. Reactor startup is allowed prior to performing this SR because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions and provides adequate time to complete the Surveillance. SR 3.5.1.10 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has demonstrated that these components will usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency is considered to be acceptable from a reliability standpoint.

SR 3.5.1.12

This SR ensures that the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is less than or equal to the maximum value assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 13. This SR is modified by a Note that allows the instrumentation portion of the response time to be assumed to be the design instrumentation response time. Therefore, the instrumentation response time is excluded from the ECCS

(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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.4 (continued)

DG renders the DG inoperable but does not affect the OPERABILITY of the SW System. As such, if the automatic transfer of the cooling water supply valves associated with a DG fails this Surveillance, the DG should be considered inoperable. However, the SW System is still OPERABLE.

It is not necessary to declare the DG inoperable if the service water supply valves to the affected DG are administratively controlled to ensure cooling water is supplied to the DG and two NSW pumps are operable on the corresponding NSW header that the DG is aligned to. This ensures that a single active failure will not result in more than one DG not receiving cooling water (Ref. 5).

SR 3.7.2.5

This SR verifies that the automatic isolation valves of the SW System will automatically align to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR also verifies the automatic start capability of the required NSW pumps.

Operating experience has demonstrated that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

A Note indicates that the isolation of the SW System to individual components (e.g., an RBCCW heat exchanger) does not affect the OPERABILITY of the SW System. Isolation of SW System flow to an individual component must be performed such that an active component failure will not result in diverting SW System flow from the safety related components.

REFERENCES

1. BNP Calculation PCN G0050A-10, BSEP Unit No. 1 Service Water System Hydraulic Analysis, Revision 6, 7/29/93.

2. BNP Calculation PCN G0050A-12, BSEP Unit No. 2 Service Water System Hydraulic Analysis, Revision 5, 8/11/92.

(continued)

BASES

REFERENCES (continued)

3. UFSAR, Chapter 6.2.

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

5. TRM 3.16.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of CREV System is the radiation protection portion of the radiation/smoke protection mode and includes two redundant high efficiency air filtration subsystems for emergency treatment of recirculated air or outside supply air. Each subsystem consists of a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber bank, an emergency recirculation fan, and the associated ductwork and dampers. HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorber banks provide a holdup period for gaseous iodine, allowing time for decay. Two control room supply fans provide the required flow to maintain the pressure of the control room positive with respect to the outside atmosphere.

The CREV System is a standby system that is common to both Unit 1 and Unit 2, parts of which also operate during normal unit operations to maintain the control room environment. The two CREV subsystems must be OPERABLE if conditions requiring CREV System OPERABILITY exist in either Unit 1 or Unit 2. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREV System automatically switches to the radiation/smoke protection mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems. Outside air is taken in at the normal ventilation intake and is mixed with the recirculated air before being passed through one of the CREV subsystems for removal of airborne radioactive particles.

(continued)

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 > 3000 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

LCO (continued)

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

Also, three control room supply fans must be OPERABLE to ensure positive pressure can be maintained in the control room with respect to the outside atmosphere and to meet single failure criteria.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that SR 3.7.3.3 can be met. However, it is acceptable for access doors to be opened for normal control room entry and exit and not consider it to be a failure to meet the LCO.

APPLICABILITY

In MODES 1, 2, and 3, the CREV System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one CREV subsystem inoperable, the inoperable CREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREV subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREV System capability. The 7 day

(continued)

BASES

ACTIONS

A.1 (continued)

Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if any Required Action and associated Completion Time of Condition A cannot be met or two CREV subsystems are inoperable, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREV subsystem may be placed in the radiation/smoke protection mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product released. Actions must continue until the OPDRVs are suspended.

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

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. These actions place the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that each CREV subsystem in a standby mode starts from the control room and continues to operate. This SR includes initiating flow through the associated HEPA filter and charcoal adsorber bank. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every 31 days provides an adequate check on this system. Since the CREV subsystems do not have installed heaters, each subsystem need only be operated for ≥ 15 minutes to demonstrate the function of the subsystem. The 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, 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. 5), system flow rate is determined using installed calibrated flow orifice plates. Specific test frequencies and additional information are discussed in Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)."

SR 3.7.3.3

This SR verifies the integrity of the control room envelope and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to the outside atmosphere, is periodically tested to verify proper function of the CREV System. During the radiation/smoke protection mode of operation, the CREV System is designed to slightly pressurize the control room with respect to any potentially contaminated external atmosphere, including the outside atmosphere and adjacent building atmospheres, to minimize unfiltered inleakage. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.3 (continued)

CREV System is designed to maintain this positive 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 two control room supply fans 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 15.6.4.5.5.

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

5. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC portion of the Control Building Heating, Ventilation, and Air Conditioning System (hereinafter referred to as the Control Room AC System) provides temperature and humidity control for the control room during normal and accident conditions.

The Control Room AC System consists of three 50% capacity subsystems that provide cooling of recirculated control room air and outside air. Each manually controlled subsystem consists of a heating coil, a cooling coil, a supply fan, a compressor-condenser unit, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

The Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. Two of the three subsystems provide the required temperature control to maintain a suitable control room environment for a sustained occupancy of 30 persons. The normal design conditions for the control room environment are 75°F and 50% relative humidity. The Control Room AC System operation in maintaining the control room temperature is discussed in the UFSAR, Section 6.4.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy following a design basis event.

The Control Room AC System components are arranged in redundant subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of an active component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

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

LCO

Three redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least two subsystems are available, assuming a single failure disables one subsystem. A failure of two or more control room AC subsystems could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in all three subsystems. These components include the cooling coils, supply fans, compressor-condenser units, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with a potential for draining the reactor vessel (OPDRVs).

(continued)

BASES (continued)

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystems are adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystems could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystems can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1

With two control room AC subsystems inoperable, the Control Room AC System may not be capable of performing the intended function. However, since the BNP control room is common to both Units 1 and 2, the risk associated with continued operation for a relatively short time could be less than that associated with an immediate controlled shutdown of both units. Therefore, additional time is allowed to restore one of the inoperable control room AC subsystems to OPERABLE status. The 72 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining the Control Room AC System OPERABLE. This time period also ensures that the probability of an accident (requiring Control Room AC System OPERABILITY) occurring during periods when two control room AC subsystems are inoperable is minimal.

C.1 and C.2

In MODE 1, 2, or 3, if Required Action A.1 or B.1 cannot be completed within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

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

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, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 or B.1 cannot be completed within the associated Completion Time, the OPERABLE control room AC subsystem or subsystems may be placed immediately in operation. This action ensures that the remaining subsystem(s) is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

E.1

If three control room AC subsystems are inoperable in MODE 1, 2, or 3, the Control Room AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

F.1, F.2, and F.3

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, the Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with three control room AC subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. UFSAR, Section 6.4.2.

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

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System for the purposes of this specification consists of the components in the following flow path from the main condenser SJAEs to the plant stack. Offgas is discharged from the main condenser via the SJAEs and diluted with steam to keep hydrogen levels below explosive concentrations. The offgas is then passed through an Offgas Recombiner System where hydrogen and oxygen are catalytically recombined into water. After recombination, the offgas is routed to an offgas condenser to remove moisture. The offgas then passes through a 30 minute delay pipe before entering the Augmented Offgas Charcoal Adsorber System. The radioactivity of the offgas recombiner effluent is monitored downstream of the offgas condenser prior to entering the 30 minute delay pipe. The Augmented Offgas Charcoal Adsorber System provides a long delay period for radioisotope decay as the offgas passes through the system. Offgas exiting the Augmented Offgas Charcoal Adsorber System is routed to the plant stack for release to the environment.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 11.3 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

BASES (continued)

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 243,600 μCi/second) and is based on the original licensed RATED THERMAL POWER.

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System failure.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from significant sources of radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample (taken at the discharge of the main condenser air ejector prior to dilution or discharge) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by $\geq$ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is indicated (by the condenser air ejector noble gas activity monitor), to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.3.

2. 10 CFR 100.

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

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND	The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of four valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.1.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the Speed Control System or load limit restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows through connecting piping and bypass valve pressure reducers to the condenser.
APPLICABLE SAFETY ANALYSES	The Main Turbine Bypass System is assumed to function during the generator load rejection transient, the turbine trip transient, and the feedwater controller failure maximum demand transient, as described in the UFSAR, Section 15.2.1 (Ref. 2), Section 15.2.2 (Ref. 3), and Section 15.1.2 (Ref. 4). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties. The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the minimum number of bypass valves, specified in the COLR, to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the turbine generator load rejection transient. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves as specified in the COLR inoperable), and the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 4 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on manufacturer's recommendations (Ref. 6), is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the COLR. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. UFSAR, Section 7.7.1.4.
2. UFSAR, Section 15.2.1.
3. UFSAR, Section 15.2.2.
4. UFSAR, Section 15.1.2.
5. 10 CFR 50.36(c)(2)(ii).
6. GE Service Information Letter No. 413, Main Steam Bypass Valve Testing, October 4, 1984.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are well below the 10 CFR 100 (Ref. 3) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 4).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

BASES (continued)

LCO

The specified water level (19 feet 11 inches above the top of irradiated fuel assemblies seated in the High Density Fuel Storage System racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, the water level is the minimum required for fuel movement within the spent fuel storage pool. This water level corresponds to 36 feet 11 3/4 inches spent fuel storage pool level.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

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 A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable considering that the water volume in the pool is normally stable, and the availability of spent fuel storage pool level indicators and associated alarms.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 15.7.1.
3. 10 CFR 100.
4. Regulatory Guide 1.25, March 23, 1972.
5. 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

Based on the number of safety significant electrical loads associated with each bus listed in Table B 3.8.7-1, if one or more of the buses becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other buses, such as MCCs and distribution panels, which help comprise the AC and DC distribution systems are not listed in Table B 3.8.7-1. The loss of electrical loads associated with these buses may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these buses become inoperable due to a failure not affecting the OPERABILITY of a bus listed in Table B 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the bus must be declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in Table B 3.8.7-1 (e.g., loss of a 4.16 kV emergency bus, which results in de-energization of all buses powered from the 4.16 kV emergency bus), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV emergency bus).

In addition, tie breakers and transfer switches between redundant safety related AC and DC power distribution subsystems, if they exist, must be open. This includes control power transfer switches associated with the 4.16 kV and 480 V emergency buses and transfer switches associated with the ESS and DG panels. The requirement for tie breakers to be open between redundant buses and divisions is necessary to ensure independence of the redundant buses and divisions. Independence of the redundant buses and divisions is required to ensure that single-failure criteria are satisfied. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). Except as noted below, if any tie breakers are closed or transfer switches aligned to the alternate supply,

(continued)

BASES

LCO (continued)

the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related, redundant electrical power distribution subsystems. It does not, however, preclude redundant Class IE 4.16 kV emergency buses from being powered from the same offsite circuit.

An exception to the requirement to maintain the cross-tie breakers open between the 4.16 kV emergency buses applies during breaker setup and/or Appendix R and Station Blackout testing of cross-tie breakers and their associated control circuitry. During these evolutions it is permissible to close one of the two series cross-tie breakers between the 4.16 kV buses if the remaining series cross-tie breaker has been racked-out and removed from its cubicle and control power for the closed breaker is maintained at all times while the breaker is closed; under these conditions independence of the redundant buses and divisions is maintained such that single-failure criteria are satisfied.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 4 and 5 and other conditions in which AC and DC electrical power distribution subsystems are required are covered in the Bases for LCO 3.8.8, "Distribution Systems—Shutdown."

ACTIONS

A.1

With one AC electrical power distribution subsystem inoperable due to either inoperable load group E3 bus(es) or inoperable load group E4 bus(es), the remaining AC electrical power distribution load groups are capable of

(continued)

BASES

ACTIONS

A.1 (continued)

supporting the minimum safety functions necessary to shut down the operating reactor and maintain both reactors in a safe condition, assuming no single failure in the remaining AC electrical power distribution load groups, when Unit 2 is in MODE 4 or 5. (If Unit 2 is in MODE 1, 2, or 3, then the Unit 2 ACTIONS of Specification 3.8.7, "Distribution Systems—Operating," require restoration of the associated AC electrical power distribution subsystem within 8 hours of the inoperability.) The overall reliability is reduced in Condition A, because a single failure in a remaining load group could result in the minimum required ESF functions not being supported. As a result, Required Action A.1 limits the time period to perform planned maintenance on a Unit 2 load group to 7 days. This is acceptable based on the following:

a. The other unit's load group buses are not as critical to the operating unit (fewer operating unit loads) as the operating unit's load group buses.

b. Performing maintenance on these components will increase the reliability of the Class 1E AC Electrical Power Distribution System.

c. The 7 day Completion Time provides a reasonable time frame for performance of planned maintenance.

During the planned maintenance of the load group buses, if a condition is discovered on these buses requiring corrective maintenance, this maintenance may be performed within the 7 day Completion Time of Required Action A.1.

The Class 1E AC Electrical Power Distribution System is divided into four load groups. Each load group consists of a primary emergency bus, its downstream secondary emergency bus, 120 VAC vital bus, and transformers and interconnecting cables. The buses associated with each of the four load groups are defined as follows:

(continued)

BASES

ACTIONS

A.1 (continued)

Load group E1 consists of 4.16 kV bus E1, 480 V bus E5, and 120 VAC vital bus 1E5.

Load group E2 consists of 4.16 kV bus E2, 480 V bus E6, and 120 VAC vital bus 1E6.

Load group E3 consists of 4.16 kV bus E3, 480 V bus E7, and 120 VAC vital bus 2E7.

Load group E4 consists of 4.16 kV bus E4, 480 V bus E8, and 120 VAC vital bus 2E8.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an AC bus in a load group in a different division is inoperable and subsequently returned OPERABLE, this LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours (since initial failure to meet the LCO) to restore the AC Electrical Power Distribution System. At this time an AC bus in a load group in a different division could again become inoperable, and the load group removed under Condition A could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". This results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition A was entered. The 176 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

If while in Condition A, emergency buses associated with another load group become inoperable (e.g., buses in load groups E3 and E4 are concurrently inoperable), Condition B and F must be entered, as appropriate.

(continued)

BASES

ACTIONS (continued)

B.1

With one or more required AC buses or distribution panels in one division inoperable for reasons other than Condition A, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining AC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses and distribution panels must be restored to OPERABLE status within 8 hours.

The Condition B worst scenario is one division without AC power (i.e., no offsite power to the division and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining division by stabilizing the unit and restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.

b. The low potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP).")

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, a DC bus is inoperable and subsequently

(continued)

BASES

ACTIONS

B.1 (continued)

returned OPERABLE, this LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 176 hours, since initial failure to meet the LCO, to restore the AC electrical power distribution system. At this time a DC bus could again become inoperable, and the AC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition B was entered. The 176 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1, C.2, C.3 and C.4

Condition C applies to the 125 VDC buses listed in Table B 3.8.7-1 which can be supplied from either a normal or an alternate DC source. These buses are listed below:

a. 125 VDC Control Power Buses for 4.16 kV Switchgear E1, E2, E3, and E4;

b. 125 VDC Control Power Buses for 480 V Switchgear E5, E6, E7, and E8;

c. 125 VDC ESS Logic Cabinets H58, H59, H60, and H61; and

d. 125 VDC DG Panels DG-1, DG-2, DG-3, and DG-4.

Condition A permits the de-energization of the E3 load group bus(es) or the E4 load group bus(es) for planned maintenance when Unit 2 is in MODE 4 or 5. During a 4.16 kV or 480 V bus outage it is desirable to clear both the normal and alternate sources of DC control power to the bus for personnel safety. The de-energized AC bus is inoperable and not capable of supplying its loads regardless of the availability of DC control power. Hence, entry into Condition C as a result of performing maintenance under Condition A is not necessary; Condition D would apply.

(continued)

BASES

ACTIONS

C.1, C.2, C.3 and C.4 (continued)

With one or more DC electrical power distribution subsystems inoperable due to loss of normal DC source, the remaining DC electrical power distribution subsystem(s) are capable of supporting the minimum safety functions necessary to shutdown the reactor and maintain it in a safe shutdown condition, provided safety function is not lost and assuming no single failure. However, the overall reliability is reduced because a single failure in the DC electrical power distribution system could result in a loss of two of four AC electrical load groups and the minimum required ESF functions not being supported. Therefore, action must be immediately initiated to transfer the DC electrical power distribution system to its alternate source and the affected supported equipment immediately declared inoperable. Upon completion of the transfer of the affected supported equipment's DC electrical power distribution subsystem to its OPERABLE alternate DC source, the affected supported equipment may be declared OPERABLE again. The ESS logic cabinets transfer automatically upon loss of the normal source. For an ESS logic cabinet, verification that the automatic transfer has occurred and alternate power is available to the ESS logic cabinet will satisfy Required Action C.2. By allowance of the option to declare affected supported equipment inoperable with associated DC electrical power distribution subsystems inoperable due to loss of normal DC source, more conservative restrictions are implemented in accordance with the affected system LCOs' ACTIONS. When any control power transfer switch associated with the 4.16 kV and 480 V emergency buses or any transfer switch associated with the ESS and DG panels is transferred to the alternate source, a single failure in the DC system could render two of four AC electrical load groups inoperable. Therefore, to prevent indefinite operation in this degraded condition, power from the normal DC source must be restored in 7 days.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. Required Actions C.1 and C.2 should be completed as quickly as possible. The 7 day Completion Time of Required Action C.4 is considered to be acceptable due to the low potential for an event in conjunction with a single failure

(continued)

BASES

ACTIONS

C.1, C.2, C.3 and C.4 (continued)

of a redundant component and is consistent with the allowed Completion Time for an inoperable DC electrical power subsystem specified in Specification 3.8.4, "DC Sources—Operating."

The second Completion Time for Required Action C.4 establishes a limit on the maximum time allowed for any combination of required electrical power distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours, since initial failure to meet the LCO, to restore the DC electrical power distribution system. At this time, an AC bus could again become inoperable, and the DC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time Condition C was entered. The 176 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

D.1

With one DC electrical power distribution subsystem inoperable for reasons other than Condition C, the remaining DC electrical power distribution subsystem is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC electrical power distribution subsystem must be restored to OPERABLE status within 7 days by powering the bus from the associated batteries or chargers.

(continued)

BASES

ACTIONS

D.1 (continued)

Condition D represents one division without adequate DC power, potentially with both the battery(s) significantly degraded and the associated charger(s) nonfunctioning. In this situation the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining divisions, and restoring power to the affected division.

The 7 day Completion Time is consistent with the allowed Completion Time for an inoperable DC electrical power subsystem specified in Specification 3.8.4, "DC Sources—Operating". Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 7 days, is acceptable because of:

a. The potential for decreased safety when requiring a change in plant conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without DC power, while not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division;

c. The low potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of required electrical power distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition D is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours, since initial failure to

(continued)

BASES

ACTIONS

D.1 (continued)

meet the LCO, to restore the DC electrical power distribution system. At this time, an AC bus could again become inoperable, and the DC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time Condition D was entered. The 176 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

E.1 and E.2

If the inoperable electrical power distribution subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit 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.

F.1

Condition F corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one AC or DC electrical power distribution subsystem is lost, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the AC and DC electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. This includes verifying that distribution bus tie breakers are open and control power transfer switches associated with the 4.16 kV and 480 V emergency buses and transfer switches associated with the ESS and DG panels are aligned to their normal DC sources. The correct breaker alignment ensures the appropriate separation and independence of the electrical buses are maintained, and power is available to each required bus. The verification of energization of the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. This may be performed by verification of absence of low voltage alarms or by verifying a load powered from the bus is operating. The 7 day Frequency takes into account the redundant capability of the AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

SR 3.8.7.2

This Surveillance verifies that no combination of more than two power conversion modules (consisting of either two lighting inverters or one lighting inverter and one plant uninterruptible power supply unit) are aligned to Division II (bus B). Two power conversion modules aligned to Division II (bus B) was an initial assumption in the DC battery load study. Limiting two power conversion modules to be aligned to Division II ensures the associated batteries will supply DC power to safety related equipment during a design basis event. The 7 day Frequency takes into account the redundant capability of the DC electrical power distribution subsystems and indications available in the control room to alert the operator of power conversion module misalignment.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

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

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BASES

TO

THE FACILITY OPERATING LICENSE DPR-62

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 2

CAROLINA POWER & LIGHT COMPANY

REVISION 11

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B 3.8-41	6	B 3.8-82	8
B 3.8-42	6	B 3.8-83	8
B 3.8-43	6	B 3.8-84	8
B 3.8-44	6	B 3.8-85	8
B 3.8-45	6	B 3.8-86	7
B 3.8-46	6	B 3.8-87	7
B 3.8-47	6	B 3.8-88	7
B 3.8-48	6	B 3.8-89	7
B 3.8-49	6	B 3.8-90	7
B 3.8-50	6	B 3.8-91	7
B 3.8-51	6		
B 3.8-52	6	B 3.9-1	0
B 3.8-53	6	B 3.9-2	0
B 3.8-54	6	B 3.9-3	0
B 3.8-55	6	B 3.9-4	0
B 3.8-56	6	B 3.9-5	0
B 3.8-57	6	B 3.9-6	0
B 3.8-58	6	B 3.9-7	0
B 3.8-59	6	B 3.9-8	0
B 3.8-60	6	B 3.9-9	0
B 3.8-61	6	B 3.9-10	0
B 3.8-62	6	B 3.9-11	0
B 3.8-63	6	B 3.9-12	0
B 3.8-64	6	B 3.9-13	0
B 3.8-65	6	B 3.9-14	0
B 3.8-66	6	B 3.9-15	0
B 3.8-67	6	B 3.9-16	0
B 3.8-68	6	B 3.9-17	0
B 3.8-69	6	B 3.9-18	0
B 3.8-70	6	B 3.9-19	0
B 3.8-71	6	B 3.9-20	0
B 3.8-72	6	B 3.9-21	0
B 3.8-73	6	B 3.9-22	0
B 3.8-74	6	B 3.9-23	0
B 3.8-75	8	B 3.9-24	0
B 3.8-76	8	B 3.9-25	0
B 3.8-77	8	B 3.9-26	0
B 3.8-78	8	B 3.9-27	0
B 3.8-79	8	B 3.9-28	0
B 3.8-80	8	B 3.9-29	0

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test. Adequate pressure at which this test is to be performed, to avoid damaging the valve, is 945 psig. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is considered OPERABLE.

The 24 month Frequency was developed based on the SRV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has demonstrated that these components will usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.2.2.2.

2. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, Supplement 1, March 1996.

3. UFSAR, Chapter 15.

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

5. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.11

A manual actuation of each required ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the SRV discharge lines. This is demonstrated by the response of the turbine control or bypass valve; by a change in the measured flow; or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this SR. Adequate pressure at which this SR is to be performed, to avoid damaging the valve, is 945 psig. Reactor startup is allowed prior to performing this SR because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure is adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions and provides adequate time to complete the Surveillance. SR 3.5.1.10 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has demonstrated that these components will usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency is considered to be acceptable from a reliability standpoint.

SR 3.5.1.12

This SR ensures that the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is less than or equal to the maximum value assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 13. This SR is modified by a Note that allows the instrumentation portion of the response time to be assumed to be the design instrumentation response time. Therefore, the instrumentation response time is excluded from the ECCS

(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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.4 (continued)

DG renders the DG inoperable but does not affect the OPERABILITY of the SW System. As such, if the automatic transfer of the cooling water supply valves associated with a DG fails this Surveillance, the DG should be considered inoperable. However, the SW System is still OPERABLE.

It is not necessary to declare the DG inoperable if the service water supply valves to the affected DG are administratively controlled to ensure cooling water is supplied to the DG and two NSW pumps are operable on the corresponding NSW header that the DG is aligned to. This ensures that a single active failure will not result in more than one DG not receiving cooling water (Ref. 5).

SR 3.7.2.5

This SR verifies that the automatic isolation valves of the SW System will automatically align to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. This SR also verifies the automatic start capability of the required NSW pumps.

Operating experience has demonstrated that these components will usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

A Note indicates that the isolation of the SW System to individual components (e.g., an RBCCW heat exchanger) does not affect the OPERABILITY of the SW System. Isolation of SW System flow to an individual component must be performed such that an active component failure will not result in diverting SW System flow from the safety related components.

REFERENCES

1. BNP Calculation PCN G0050A-10, BSEP Unit No. 1 Service Water System Hydraulic Analysis, Revision 6, 7/29/93.

2. BNP Calculation PCN G0050A-12, BSEP Unit No. 2 Service Water System Hydraulic Analysis, Revision 5, 8/11/92.

(continued)

BASES

REFERENCES (continued)

3. UFSAR, Chapter 6.2.

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

5. TRM 3.16.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Ventilation (CREV) System

BASES

BACKGROUND

The CREV System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of CREV System is the radiation protection portion of the radiation/smoke protection mode and includes two redundant high efficiency air filtration subsystems for emergency treatment of recirculated air or outside supply air. Each subsystem consists of a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber bank, an emergency recirculation fan, and the associated ductwork and dampers. HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorber banks provide a holdup period for gaseous iodine, allowing time for decay. Two control room supply fans provide the required flow to maintain the pressure of the control room positive with respect to the outside atmosphere.

The CREV System is a standby system that is common to both Unit 1 and Unit 2, parts of which also operate during normal unit operations to maintain the control room environment. The two CREV subsystems must be OPERABLE if conditions requiring CREV System OPERABILITY exist in either Unit 1 or Unit 2. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREV System automatically switches to the radiation/smoke protection mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems. Outside air is taken in at the normal ventilation intake and is mixed with the recirculated air before being passed through one of the CREV subsystems for removal of airborne radioactive particles.

(continued)

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 > 3000 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

LCO (continued)

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

Also, three control room supply fans must be OPERABLE to ensure positive pressure can be maintained in the control room with respect to the outside atmosphere and to meet single failure criteria.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that SR 3.7.3.3 can be met. However, it is acceptable for access doors to be opened for normal control room entry and exit and not consider it to be a failure to meet the LCO.

APPLICABILITY

In MODES 1, 2, and 3, the CREV System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREV System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one CREV subsystem inoperable, the inoperable CREV subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREV subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced CREV System capability. The 7 day

(continued)

BASES

ACTIONS

A.1 (continued)

Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if any Required Action and associated Completion Time of Condition A cannot be met or two CREV subsystems are inoperable, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREV subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREV subsystem may be placed in the radiation/smoke protection mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product released. Actions must continue until the OPDRVs are suspended.

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

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREV subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. These actions place the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that each CREV subsystem in a standby mode starts from the control room and continues to operate. This SR includes initiating flow through the associated HEPA filter and charcoal adsorber bank. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every 31 days provides an adequate check on this system. Since the CREV subsystems do not have installed heaters, each subsystem need only be operated for ≥ 15 minutes to demonstrate the function of the subsystem. The 31 day Frequency is based on the known reliability of the equipment and the two subsystem redundancy available.

SR 3.7.3.2

This SR verifies that the required CREV testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, 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. 5), system flow rate is determined using installed calibrated flow orifice plates. Specific test frequencies and additional information are discussed in Specification 5.5.7, "Ventilation Filter Testing Program (VFTP)."

SR 3.7.3.3

This SR verifies the integrity of the control room envelope and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to the outside atmosphere, is periodically tested to verify proper function of the CREV System. During the radiation/smoke protection mode of operation, the CREV System is designed to slightly pressurize the control room with respect to any potentially contaminated external atmosphere, including the outside atmosphere and adjacent building atmospheres, to minimize unfiltered inleakage. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.3 (continued)

CREV System is designed to maintain this positive 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 two control room supply fans 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 15.6.4.5.5.

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

5. ESR 99-00055, SBGT and CBEAF Technical Specification Surveillance Flow Measurement.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Air Conditioning (AC) System

BASES

BACKGROUND

The Control Room AC portion of the Control Building Heating, Ventilation, and Air Conditioning System (hereinafter referred to as the Control Room AC System) provides temperature and humidity control for the control room during normal and accident conditions.

The Control Room AC System consists of three 50% capacity subsystems that provide cooling of recirculated control room air and outside air. Each manually controlled subsystem consists of a heating coil, a cooling coil, a supply fan, a compressor-condenser unit, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

The Control Room AC System is designed to provide a controlled environment under both normal and accident conditions. Two of the three subsystems provide the required temperature control to maintain a suitable control room environment for a sustained occupancy of 30 persons. The normal design conditions for the control room environment are 75°F and 50% relative humidity. The Control Room AC System operation in maintaining the control room temperature is discussed in the UFSAR, Section 6.4.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room AC System is to maintain the control room temperature for a 30 day continuous occupancy following a design basis event.

The Control Room AC System components are arranged in redundant subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single failure of an active component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Redundant detectors and controls are provided for control room temperature control. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

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

LCO

Three redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least two subsystems are available, assuming a single failure disables one subsystem. A failure of two or more control room AC subsystems could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in all three subsystems. These components include the cooling coils, supply fans, compressor-condenser units, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with a potential for draining the reactor vessel (OPDRVs).

(continued)

BASES (continued)

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystems are adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystems could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystems can provide the required protection, and the availability of alternate safety and nonsafety cooling methods.

B.1

With two control room AC subsystems inoperable, the Control Room AC System may not be capable of performing the intended function. However, since the BNP control room is common to both Units 1 and 2, the risk associated with continued operation for a relatively short time could be less than that associated with an immediate controlled shutdown of both units. Therefore, additional time is allowed to restore one of the inoperable control room AC subsystems to OPERABLE status. The 72 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining the Control Room AC System OPERABLE. This time period also ensures that the probability of an accident (requiring Control Room AC System OPERABILITY) occurring during periods when two control room AC subsystems are inoperable is minimal.

C.1 and C.2

In MODE 1, 2, or 3, if Required Action A.1 or B.1 cannot be completed within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

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

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, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 or B.1 cannot be completed within the associated Completion Time, the OPERABLE control room AC subsystem or subsystems may be placed immediately in operation. This action ensures that the remaining subsystem(s) is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

E.1

If three control room AC subsystems are inoperable in MODE 1, 2, or 3, the Control Room AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

F.1, F.2, and F.3

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, the Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with three control room AC subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. UFSAR, Section 6.4.2.

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

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System for the purposes of this specification consists of the components in the following flow path from the main condenser SJAEs to the plant stack. Offgas is discharged from the main condenser via the SJAEs and diluted with steam to keep hydrogen levels below explosive concentrations. The offgas is then passed through an Offgas Recombiner System where hydrogen and oxygen are catalytically recombined into water. After recombination, the offgas is routed to an offgas condenser to remove moisture. The offgas then passes through a 30 minute delay pipe before entering the Augmented Offgas Charcoal Adsorber System. The radioactivity of the offgas recombiner effluent is monitored downstream of the offgas condenser prior to entering the 30 minute delay pipe. The Augmented Offgas Charcoal Adsorber System provides a long delay period for radioisotope decay as the offgas passes through the system. Offgas exiting the Augmented Offgas Charcoal Adsorber System is routed to the plant stack for release to the environment.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the UFSAR, Section 11.3 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

BASES (continued)

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 243,600 μCi/second) and is based on the original licensed RATED THERMAL POWER.

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, main steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the gross gamma activity rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment, the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System failure.

B.1, B.2, B.3.1, and B.3.2

If the gross gamma activity rate is not restored to within the limits in the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from significant sources of radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample (taken at the discharge of the main condenser air ejector prior to dilution or discharge) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by $\geq$ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is indicated (by the condenser air ejector noble gas activity monitor), to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.3.

2. 10 CFR 100.

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

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND	The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 85% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of ten valves connected to the main steam lines between the main steam isolation valves and the turbine stop valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the UFSAR, Section 7.7.1.4 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the Speed Control System or load limit restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows through connecting piping and bypass valve pressure reducers to the condenser.
APPLICABLE SAFETY ANALYSES	The Main Turbine Bypass System is assumed to function during the generator load rejection transient, the turbine trip transient, and the feedwater controller failure maximum demand transient, as described in the UFSAR, Section 15.2.1 (Ref. 2), Section 15.2.2 (Ref. 3), and Section 15.1.2 (Ref. 4). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR and MCPR penalties. The Main Turbine Bypass System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the minimum number of bypass valves, specified in the COLR, to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the turbine generator load rejection transient. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (three or more bypass valves as specified in the COLR inoperable), and the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR and MCPR limits accordingly. The 4 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the APLHGR and MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on manufacturer's recommendations (Ref. 6), is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the COLR. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. UFSAR, Section 7.7.1.4.
2. UFSAR, Section 15.2.1.
3. UFSAR, Section 15.2.2.
4. UFSAR, Section 15.1.2.
5. 10 CFR 50.36(c)(2)(ii).
6. GE Service Information Letter No. 413, Main Steam Bypass Valve Testing, October 4, 1984.

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are well below the 10 CFR 100 (Ref. 3) exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.25 (Ref. 4).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core. The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

(continued)

BASES (continued)

LCO

The specified water level (19 feet 11 inches above the top of irradiated fuel assemblies seated in the High Density Fuel Storage System racks) preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, the water level is the minimum required for fuel movement within the spent fuel storage pool. This water level corresponds to 36 feet 11 3/4 inches spent fuel storage pool level.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

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 A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable considering that the water volume in the pool is normally stable, and the availability of spent fuel storage pool level indicators and associated alarms.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 15.7.1.
3. 10 CFR 100.
4. Regulatory Guide 1.25, March 23, 1972.
5. 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

Based on the number of safety significant electrical loads associated with each bus listed in Table B 3.8.7-1, if one or more of the buses becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other buses, such as MCCs and distribution panels, which help comprise the AC and DC distribution systems are not listed in Table B 3.8.7-1. The loss of electrical loads associated with these buses may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should one or more of these buses become inoperable due to a failure not affecting the OPERABILITY of a bus listed in Table B 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the bus must be declared inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in Table B 3.8.7-1 (e.g., loss of a 4.16 kV emergency bus, which results in de-energization of all buses powered from the 4.16 kV emergency bus), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV emergency bus).

In addition, tie breakers and transfer switches between redundant safety related AC and DC power distribution subsystems, if they exist, must be open. This includes control power transfer switches associated with the 4.16 kV and 480 V emergency buses and transfer switches associated with the ESS and DG panels. The requirement for tie breakers to be open between redundant buses and divisions is necessary to ensure independence of the redundant buses and divisions. Independence of the redundant buses and divisions is required to ensure that single-failure criteria are satisfied. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s). Except as noted below, if any tie breakers are closed or transfer switches aligned to the alternate supply,

(continued)

BASES (continued)

LCO (continued)

the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related, redundant electrical power distribution subsystems. It does not, however, preclude redundant Class IE 4.16 kV emergency buses from being powered from the same offsite circuit.

An exception to the requirement to maintain the cross-tie breakers open between the 4.16 kV emergency buses applies during breaker setup and/or Appendix R and Station Blackout testing of cross-tie breakers and their associated control circuitry. During these evolutions it is permissible to close one of the two series cross-tie breakers between the 4.16 kV buses if the remaining series cross-tie breaker has been racked-out and removed from its cubicle and control power for the closed breaker is maintained at all times while the breaker is closed; under these conditions independence of the redundant buses and divisions is maintained such that single-failure criteria are satisfied.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 4 and 5 and other conditions in which AC and DC electrical power distribution subsystems are required are covered in the Bases for LCO 3.8.8, "Distribution Systems—Shutdown."

ACTIONS

A.1

With one AC electrical power distribution subsystem inoperable due to either inoperable load group E1 bus(es), or inoperable load group E2 bus(es), the remaining AC electrical power distribution load groups are capable of

(continued)

BASES

ACTIONS

A.1 (continued)

supporting the minimum safety functions necessary to shut down the operating reactor and maintain both reactors in a safe condition, assuming no single failure in the remaining AC electrical power distribution load groups, when Unit 1 is in MODE 4 or 5. (If Unit 1 is in MODE 1, 2, or 3, then the Unit 1 ACTIONS of Specification 3.8.7, "Distribution Systems—Operating," require restoration of the associated AC electrical power distribution subsystem within 8 hours of the inoperability.) The overall reliability is reduced in Condition A, because a single failure in a remaining load group could result in the minimum required ESF functions not being supported. As a result, Required Action A.1 limits the time period to perform planned maintenance on a Unit 1 load group to 7 days. This is acceptable based on the following:

a. The other unit's load group buses are not as critical to the operating unit (fewer operating unit loads) as the operating unit's load group buses.

b. Performing maintenance on these components will increase the reliability of the Class 1E AC Electrical Power Distribution System.

c. The 7 day Completion Time provides a reasonable time frame for performance of planned maintenance.

During the planned maintenance of the load group buses, if a condition is discovered on these buses requiring corrective maintenance, this maintenance may be performed within the 7 day Completion Time of Required Action A.1.

The Class 1E AC Electrical Power Distribution System is divided into four load groups. Each load group consists of a primary emergency bus, its downstream secondary emergency bus, 120 VAC vital bus, and transformers and interconnecting cables. The buses associated with each of the four load groups are defined as follows:

(continued)

BASES

ACTIONS

A.1 (continued)

Load group E1 consists of 4.16 kV bus E1, 480 V bus E5, and 120 VAC vital bus 1E5.

Load group E2 consists of 4.16 kV bus E2, 480 V bus E6, and 120 VAC vital bus 1E6.

Load group E3 consists of 4.16 kV bus E3, 480 V bus E7, and 120 VAC vital bus 2E7.

Load group E4 consists of 4.16 kV bus E4, 480 V bus E8, and 120 VAC vital bus 2E8.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an AC bus in a load group in a different division is inoperable and subsequently returned OPERABLE, this LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours (since initial failure to meet the LCO) to restore the AC Electrical Power Distribution System. At this time an AC bus in a load group in a different division could again become inoperable, and the load group removed under Condition A could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock". This results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition A was entered. The 176 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

If while in Condition A, emergency buses associated with another load group become inoperable (e.g., buses in load groups E1 and E2 are concurrently inoperable), Condition B and F must be entered, as appropriate.

(continued)

BASES

ACTIONS (continued)

B.1

With one or more required AC buses or distribution panels in one division inoperable for reasons other than Condition A, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining AC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, and distribution panels must be restored to OPERABLE status within 8 hours.

The Condition B worst scenario is one division without AC power (i.e., no offsite power to the division and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining division by stabilizing the unit and restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.

b. The low potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP).")

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, a DC bus is inoperable and subsequently

(continued)

BASES

ACTIONS

B.1 (continued)

returned OPERABLE, this LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 176 hours, since initial failure to meet the LCO, to restore the AC electrical power distribution system. At this time a DC bus could again become inoperable, and the AC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This results in establishing the "time zero" at the time this LCO was initially not met, instead of at the time Condition B was entered. The 176 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1, C.2, C.3 and C.4

Condition C applies to the 125 VDC buses listed in Table B 3.8.7-1 which can be supplied from either a normal or an alternate DC source. These buses are listed below:

a. 125 VDC Control Power Buses for 4.16 kV Switchgear E1, E2, E3, and E4;

b. 125 VDC Control Power Buses for 480 V Switchgear E5, E6, E7, and E8;

c. 125 VDC ESS Logic Cabinets H58, H59, H60, and H61; and

d. 125 VDC DG Panels DG-1, DG-2, DG-3, and DG-4.

Condition A permits the de-energization of the E1 load group bus(es) or the E2 load group bus(es) for planned maintenance when Unit 1 is in MODE 4 or 5. During a 4.16 kV or 480 V bus outage it is desirable to clear both the normal and alternate sources of DC control power to the bus for personnel safety. The de-energized AC bus is inoperable and not capable of supplying its loads regardless of the availability of DC control power. Hence, entry into Condition C as a result of performing maintenance under Condition A is not necessary; Condition D would apply.

(continued)

BASES

ACTIONS

C.1, C.2, C.3 and C.4 (continued)

With one or more DC electrical power distribution subsystems inoperable due to loss of normal DC source, the remaining DC electrical power distribution subsystem(s) are capable of supporting the minimum safety functions necessary to shutdown the reactor and maintain it in a safe shutdown condition, provided safety function is not lost and assuming no single failure. However, the overall reliability is reduced because a single failure in the DC electrical power distribution system could result in a loss of two of four AC electrical load groups and the minimum required ESF functions not being supported. Therefore, action must be immediately initiated to transfer the DC electrical power distribution system to its alternate source and the affected supported equipment immediately declared inoperable. Upon completion of the transfer of the affected supported equipment's DC electrical power distribution subsystem to its OPERABLE alternate DC source, the affected supported equipment may be declared OPERABLE again. The ESS logic cabinets transfer automatically upon loss of the normal source. For an ESS logic cabinet, verification that the automatic transfer has occurred and alternate power is available to the ESS logic cabinet will satisfy Required Action C.2. By allowance of the option to declare affected supported equipment inoperable with associated DC electrical power distribution subsystems inoperable due to loss of normal DC source, more conservative restrictions are implemented in accordance with the affected system LCOs' ACTIONS. When any control power transfer switch associated with the 4.16 kV and 480 V emergency buses or any transfer switch associated with the ESS and DG panels is transferred to the alternate source, a single failure in the DC system could render two of four AC electrical load groups inoperable. Therefore, to prevent indefinite operation in this degraded condition, power from the normal DC source must be restored in 7 days.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. Required Actions C.1 and C.2 should be completed as quickly as possible. The 7 day Completion Time of Required Action C.4 is considered to be acceptable due to the low potential for an event in conjunction with a single failure

(continued)

BASES

ACTIONS

C.1, C.2, C.3 and C.4 (continued)

of a redundant component and is consistent with the allowed Completion Time for an inoperable DC electrical power subsystem specified in Specification 3.8.4, "DC Sources—Operating."

The second Completion Time for Required Action C.4 establishes a limit on the maximum time allowed for any combination of required electrical power distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours, since initial failure to meet the LCO, to restore the DC electrical power distribution system. At this time, an AC bus could again become inoperable, and the DC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time Condition C was entered. The 176 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

D.1

With one DC electrical power distribution subsystem inoperable for reasons other than Condition C, the remaining DC electrical power distribution subsystem is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC electrical power distribution subsystem must be restored to OPERABLE status within 7 days by powering the bus from the associated batteries or chargers.

(continued)

BASES

ACTIONS

D.1 (continued)

Condition D represents one division without adequate DC power, potentially with both the battery(s) significantly degraded and the associated charger(s) nonfunctioning. In this situation the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining divisions, and restoring power to the affected division.

The 7 day Completion Time is consistent with the allowed Completion Time for an inoperable DC electrical power subsystem specified in Specification 3.8.4, "DC Sources—Operating". Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 7 days, is acceptable because of:

a. The potential for decreased safety when requiring a change in plant conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without DC power, while not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division;

c. The low potential for an event in conjunction with a single failure of a redundant component.

The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of required electrical power distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition D is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 176 hours, since initial failure to

(continued)

BASES

ACTIONS

D.1 (continued)

meet the LCO, to restore the DC electrical power distribution system. At this time, an AC bus could again become inoperable, and the DC electrical power distribution system could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time Condition D was entered. The 176 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

E.1 and E.2

If the inoperable electrical power distribution subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, the unit 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.

F.1

Condition F corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one AC or DC electrical power distribution subsystem is lost, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the AC and DC electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. This includes verifying that distribution bus tie breakers are open and control power transfer switches associated with the 4.16 kV and 480 V emergency buses and transfer switches associated with the ESS and DG panels are aligned to their normal DC sources. The correct breaker alignment ensures the appropriate separation and independence of the electrical buses are maintained, and power is available to each required bus. The verification of energization of the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. This may be performed by verification of absence of low voltage alarms or by verifying a load powered from the bus is operating. The 7 day Frequency takes into account the redundant capability of the AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

SR 3.8.7.2

This Surveillance verifies that no combination of more than two power conversion modules (consisting of either two lighting inverters or one lighting inverter and one plant uninterruptible power supply unit) are aligned to Division II (bus B). Two power conversion modules aligned to Division II (bus B) was an initial assumption in the DC battery load study. Limiting two power conversion modules to be aligned to Division II ensures the associated batteries will supply DC power to safety related equipment during a design basis event. The 7 day Frequency takes into account the redundant capability of the DC electrical power distribution subsystems and indications available in the control room to alert the operator of power conversion module misalignment.

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

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