



Progress Energy

DEC 10 2004

SERIAL: BSEP 04-0160

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

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

Ladies and Gentlemen:

In accordance with Technical Specification (TS) 5.5.10 for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., is submitting Revision 38 to the BSEP, Unit 1 TS Bases and Revision 35 to the BSEP, Unit 2 TS Bases.

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

Sincerely,

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

WRM/wrm

Enclosures:

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

ADDI

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Summary of Revisions to Technical Specification Bases			
Revision	Affected Unit	Date Implemented	Title/Description
38 35	1 2	December 1, 2004	<p>Title: Implementation of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Reactor Pressure Vessel Integrated Surveillance Program to Address the Requirements of Appendix H to 10 CFR Part 50 (TSC-2003-01)</p> <p>Description: Revision 38 and 35 revise Technical Specification (TS) Bases 3.4.9 to reflect implementation of the reactor pressure vessel integrated surveillance program (ISP) developed by the BWRVIP. NRC approval for the Brunswick Plant to adopt the ISP was issued on January 14, 2004, as Amendments 229 and 257 to the facility operating licenses for Unit 1 and Unit 2, respectively.</p>
			<p>Title: Clarification of Surveillance Requirement 3.8.1.10 Purpose (TSB-2004-07)</p> <p>Description: Revisions 38 and 35 incorporate a clarification to the description of the purpose for Surveillance Requirement 3.8.1.10.</p>
			<p>Title: Administrative Correction to the List of Effective Pages (LOEP)</p> <p>Description: Revision 35 for Unit 2 also includes a correction to the LOEP to remove pages B 3.3.1.3-1 through B 3.3.1.3-9. The actual Bases pages were removed as part of Bases Revision 27, which was submitted by letter dated March 28, 2003 (i.e., Serial: BSEP 03-0054); however, the LOEP was not accurately updated at that time.</p>

Page Replacement Instructions	
Remove	Insert
Unit 1 -Bases Book 1	
Cover Page, Revision 37	Cover Page, Revision 38
LOEP-1, Revision 37	LOEP-1, Revision 38
Unit 1 - Bases Book 2	
LOEP-1, Revision 36	LOEP-1, Revision 38
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B 3.4.9-3, Revision 31	B 3.4.9-3, Revision 38
B 3.4.9-5, Revision 31	B 3.4.9-5, Revision 38
B 3.4.9-6, Revision 31	B 3.4.9-6, Revision 38
B 3.4.9-9, Revision 36	B 3.4.9-9, Revision 38
B 3.8.1-28, Revision 31	B 3.8.1-28, Revision 38

Page Replacement Instructions	
Remove	Insert
Unit 2 - Bases Book 1	
Cover Page, Revision 34	Cover Page, Revision 35
LOEP-1, Revision 34	LOEP-1, Revision 35
LOEP-2 Revision 33	LOEP-2 Revision 35
LOEP-3 Revision 33	LOEP-3 Revision 35
LOEP-4 Revision 33	LOEP-4 Revision 35
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B 3.4.9-3, Revision 30	B 3.4.9-3, Revision 35
B 3.4.9-5, Revision 30	B 3.4.9-5, Revision 35
B 3.4.9-6, Revision 30	B 3.4.9-6, Revision 35
B 3.4.9-9, Revision 33	B 3.4.9-9, Revision 35
B 3.8.1-28, Revision 30	B 3.8.1-28, Revision 35

BSEP 04-0160
Enclosure 3

**Unit 1 Technical Specification Bases
Replacement Pages**

Unit 1 - Bases Book 1
Replacement Pages

BASES

TO

THE FACILITY OPERATING LICENSE DPR-71

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 1

CAROLINA POWER & LIGHT COMPANY

REVISION 38

LIST OF EFFECTIVE PAGES - BASES

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(continued)

BASES

BACKGROUND
(continued)

K_{IC} for development of P/T limit curves has been approved by the ASME through Code Case N-640 (Ref. 4).

The actual shift in the RT_{NDT} of the vessel plate and weld materials will be established periodically by removing and evaluating representative irradiated reactor specimens from selected reactors in accordance with BWRVIP-86A (Reference 5) and Appendix H of 10 CFR 50 (Reference 6). For BNP, the limiting reactor vessel material with respect to P/T curves is the N16 nozzle material. The shift in the RT_{NDT} of this material has been established in accordance with Regulatory Guide 1.99, Revision 2 (RG 1.99) (Reference 7).

In development of the P/T curves (Reference 8), it is assumed that the 1/4t (ID) flaw with a cooldown is controlling based on the following:

1. Due to attenuation effects, the fluence is significantly higher at the 1/4t location compared to the 3/4t location. Therefore, the ART_{NDT} is significantly higher at the 1/4t location.
2. The thermal tensile stress due to a 100°F/hr heatup (for a 3/4t flaw) is about the same as the thermal tensile stress due to a 100°F/hr cooldown (for a 1/4t flaw).
3. The allowable material property (i.e., K_{IA} or K_{IC}) is lower at the end of a cooldown transient where thermal stresses are a maximum compared to the end of the heatup transient.
4. For the reactor pressure vessel (RPV) (i.e., a thin cylinder), the pressure stress is essentially constant through the wall, so the 1/4t and 3/4t pressure stresses are not significantly different.

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

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 9), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 8 provides the curves and limits in this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition. RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO The elements of this LCO are:

- a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are $\leq 100^{\circ}\text{F}$ in any 1 hour period, during RCS heatup and cooldown;
- b. RCS pressure and temperature are within the applicable limits in Figures 3.4.9-3, 3.4.9-4, or 3.4.9-5 and heatup or cooldown rates are $\leq 30^{\circ}\text{F}$ in any 1 hour period, during RCS inservice leak and hydrostatic testing;
- c. The temperature difference between the reactor vessel bottom head coolant and the RPV coolant is $\leq 145^{\circ}\text{F}$ during recirculation pump startup;
- d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is $\leq 50^{\circ}\text{F}$ during recirculation pump startup;
- e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and
- f. The reactor vessel flange and the head flange temperatures are $\geq 70^{\circ}\text{F}$ when tensioning the reactor vessel head bolting studs.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

ASME Code, Section XI, Appendix E (Ref. 9), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel bellline.

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

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

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1 and C.2

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

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 9), may be used to support the evaluation; however, its use is restricted to evaluation of the bellline.

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

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**

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

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

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

REFERENCES

1. Calculation 0B21-1029, "Instrument Uncertainty for RCS Pressure/Temperature Limits Curve," Revision 0.
2. 10 CFR 50, Appendix G.
3. 1989 Edition of the ASME Code, Section XI, Appendix G.
4. ASME Code Case N-640. "Alternate References Fracture Toughness for Development of P-T Limit Curves Section XI. Division 1."
5. EPRI Report TR-1003346, BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, October 2002.
6. 10 CFR 50, Appendix H.
7. Regulatory Guide 1.99, Revision 2, May 1988.
8. Calculation 0B11-0005, "Development of RPV Pressure-Temperature Curves For BNP Units 1 and 2 For Up To 32 EFPY of Plant Operation," Revision 1.
9. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
10. 10 CFR 50.36(c)(2)(ii).

BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.8.1.10

Consistent with Regulatory Guide 1.9 (Ref. 11), paragraph C.2.2.12, this Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature) are bypassed on an ECCS initiation test signal. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 24 month Frequency is based on engineering judgment, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has demonstrated that these components will pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. To minimize testing of the DGs, the Note allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.11

Brunswick Nuclear Plant performs a 60 minute run greater than or equal to the continuous rating (3500 kW) which bounds the maximum expected post-accident DG loading. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for

(continued)

BSEP 04-0160
Enclosure 4

**Unit 2 Technical Specification Bases
Replacement Pages**

Unit 2 - Bases Book 1
Replacement Pages

BASES

TO

THE FACILITY OPERATING LICENSE DPR-62

TECHNICAL SPECIFICATIONS

FOR

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 2

CAROLINA POWER & LIGHT COMPANY

REVISION 35

LIST OF EFFECTIVE PAGES - BASES

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LOEP-4	35	B 3.1.3-3	30
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B 3.0-2	30	B 3.1.5-2	30
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(continued)

BASES

BACKGROUND
(continued)

K_{IC} for development of P/T limit curves has been approved by the ASME through Code Case N-640 (Ref. 4).

The actual shift in the RT_{NDT} of the vessel plate and weld materials will be established periodically by removing and evaluating representative irradiated reactor specimens from selected reactors in accordance with BWRVIP-86A (Reference 5) and Appendix H of 10 CFR 50 (Reference 6).

For BNP, the limiting reactor vessel material with respect to P/T curves is the N16 nozzle material. The shift in the RT_{NDT} of this material has been established in accordance with Regulatory Guide 1.99, Revision 2 (RG 1.99) (Reference 7).

In development of the P/T curves (Reference 8), it is assumed that the 1/4t (ID) flaw with a cooldown is controlling based on the following:

1. Due to attenuation effects, the fluence is significantly higher at the 1/4t location compared to the 3/4t location. Therefore, the ARTNDT is significantly higher at the 1/4t location.
2. The thermal tensile stress due to a 100°F/hr heatup (for a 3/4t flaw) is about the same as the thermal tensile stress due to a 100°F/hr cooldown (for a 1/4t flaw).
3. The allowable material property (i.e., KIA or KIC) is lower at the end of a cooldown transient where thermal stresses are a maximum compared to the end of the heatup transient.
4. For the reactor pressure vessel (reactor pressure vessel) (i.e., a thin cylinder), the pressure stress is essentially constant through the wall, so the 1/4t and 3/4t pressure stresses are not significantly different.

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

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 9), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

<p>APPLICABLE SAFETY ANALYSES</p>	<p>The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 8 provides the curves and limits in this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition. RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).</p>
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<p>LCO</p>	<p>The elements of this LCO are:</p> <ol style="list-style-type: none"> a. RCS pressure and temperature are within the applicable limits specified in Figures 3.4.9-1 and 3.4.9-2, and heatup or cooldown rates are $\leq 100^{\circ}\text{F}$ in any 1 hour period, during RCS heatup and cooldown; b. RCS pressure and temperature are within the applicable limits in Figure 3.4.9-3, 3.4.9-4, or 3.4.9-5 and heatup or cooldown rates are $\leq 30^{\circ}\text{F}$ in any 1 hour period, during RCS inservice leak and hydrostatic testing; c. The temperature difference between the reactor vessel bottom head coolant and the RPV coolant is $\leq 145^{\circ}\text{F}$ during recirculation pump startup; d. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is $\leq 50^{\circ}\text{F}$ during recirculation pump startup; e. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-2, prior to achieving criticality; and f. The reactor vessel flange and the head flange temperatures are $\geq 70^{\circ}\text{F}$ when tensioning the reactor vessel head bolting studs.
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(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

ASME Code, Section XI, Appendix E (Ref. 9), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel bellline.

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

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

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1 and C.2

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

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 9), may be used to support the evaluation; however, its use is restricted to evaluation of the bellline.

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

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**

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

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

SR 3.4.9.6 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs.

SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature is $\leq 80^{\circ}\text{F}$ in MODE 4.

SR 3.4.9.8 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature is $\leq 100^{\circ}\text{F}$ in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. Calculation 0B21-1029, "Instrument Uncertainty for RCS Pressure/Temperature Limits Curve," Revision 0
2. 10 CFR 50, Appendix G.
3. 1989 Edition of the ASME Code, Section XI, Appendix G.
4. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."
5. EPRI Report TR-1003346, BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, October 2002.
6. 10 CFR 50, Appendix H.
7. Regulatory Guide 1.99, Revision 2, May 1988.
8. Calculation 0B11-0005, "Development of RPV Pressure-Temperature Curves For BNP Units 1 and 2 For Up To 32 EFPY of Plant Operation," Revision 1.
9. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
10. 10 CFR 50.36(c)(2)(ii).

BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.8.1.10

Consistent with Regulatory Guide 1.9 (Ref. 11), paragraph C.2.2.12, this Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature) are bypassed on an ECCS initiation test signal. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 24 month Frequency is based on engineering judgment, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has demonstrated that these components will pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. To minimize testing of the DGs, the Note allows a single test (instead of two tests, one for each unit) to satisfy the requirements for both units. This is allowed since the main purpose of the Surveillance can be met by performing the test on either unit. If the DG fails one of these Surveillances, the DG should be considered inoperable on both units, unless the cause of the failure can be directly related to only one unit.

SR 3.8.1.11

Brunswick Nuclear Plant performs a 60 minute run greater than or equal to the continuous rating (3500 kW) which bounds the maximum expected post-accident DG loading. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for

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