Progress Energy

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MAY 2 9 2003

SERIAL: BSEP 03-0062
TSC-2003-01

10 CFR 50.90

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
Request For License Amendments Regarding Implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program to Address the Requirements of Appendix H to 10 CFR Part 50

REFERENCES:

1. Letter from U.S. NRC to Carl Terry (Boiling Water Reactor Vessel and Internals Project), "Safety Evaluation Regarding EPRI Proprietary Reports, 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," dated February 1, 2002.

2. NRC Regulatory Issue Summary 2002-05, NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program, dated April 8, 2002. (ADAMS Accession Number ML020660522)

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Progress Energy Carolinas, Inc. (PEC) (i.e., formerly known as Carolina Power & Light (CP&L) Company) is requesting changes to the Updated Final Safety Analysis Report (UFSAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The requested changes will remove the current BSEP reactor material specimen surveillance schedule from the UFSAR and specify that BSEP, Units 1 and 2 will participate in an integrated surveillance program (ISP) developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

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In Reference 1, the NRC determined that the ISP proposed by the BWRVIP is an acceptable alternative to all existing BWR plant-specific reactor pressure vessel surveillance programs, for the purpose of maintaining compliance with the requirements of 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." In Reference 2, the NRC stated that licensees who elect to participate in the ISP should submit a license amendment request to incorporate the ISP into the licensing basis for their BWR facility. In accordance with the NRC position in Reference 2, PEC is submitting this license amendment request in order to implement the ISP for BSEP, Units 1 and 2. Enclosure 1 to this letter provides the description and evaluation of the proposed change. PEC has evaluated the proposed change in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and determined that this change does not involve a significant hazards consideration. Enclosure 2 provides a copy of the applicable pages of the BSEP UFSAR marked to show the proposed change. Enclosure 3 provides, for information only, a copy of marked-up BSEP, Unit 1 Technical Specification Bases pages. In accordance with TS 5.5.10, "Technical Specification (TS) Bases Control Program," these TS Bases changes will be implemented following approval of the license amendment to adopt the BWRVIP ISP. These TS Bases pages are being submitted for information only and do not require issuance by the NRC.

During the BSEP, Unit 2 refueling outage conducted in March 2003, reactor pressure vessel surveillance specimen number 1 was withdrawn. 10 CFR Part 50, Appendix H, paragraph IV.A specifies that a test results report for a withdrawn surveillance capsule must be submitted to the NRC within one year, unless an extension has been granted by the Director, Office of Nuclear Reactor Regulation. Therefore, PEC requests either NRC approval of this license amendment application by December 1, 2003, or approval of an extension until December 1, 2004, for submittal of a test results report.

PEC requests that the amendments, once approved, be issued effective immediately, to be implemented within 60 days following issuance.

In accordance with 10 CFR 50.91(b), PEC is providing the State of North Carolina a copy of the proposed license amendments.

Please refer any questions regarding this submittal to Mr. Edward T. O'Neil, Manager – Support Services, at (910) 457-3512.

Sincerely,

John S. Keenan

John S. Keenan

WRM/wrm

Enclosures:

1. Evaluation of Proposed License Amendment Request
2. Marked-up UFSAR Pages
3. Marked-up Unit 1 Technical Specification Bases Pages
4. List of Regulatory Commitments

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, and agents of Carolina Power & Light Company.

Dean S. Mason
Notary (Seal)

My commission expires: Aug. 29, 2004

cc (with enclosures):

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Evaluation of Proposed License Amendment Request

Subject: Implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program to Address the Requirements of Appendix H to 10 CFR Part 50

1.0 Introduction

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Progress Energy Carolinas, Inc. (PEC) (i.e., formerly known as Carolina Power & Light (CP&L) Company) requests a change to the Updated Final Safety Analysis Report (UFSAR) for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The requested change will remove the current facility reactor material surveillance capsule removal schedules from the UFSAR and specify that the facilities will participate in an integrated surveillance program (ISP) developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP).

2.0 Description of the Proposed License Amendment

The BSEP reactor pressure vessel material surveillance program is discussed in Section 5.3.1.6 of the UFSAR. The withdrawal schedule for the reactor pressure vessel material specimens is currently contained in UFSAR Table 5.3.1-2. As a participant in the ISP, BSEP is not scheduled to remove a material specimen. Instead, the current NRC-approved revision of the BWRVIP ISP states that BSEP Units 1 and 2 will use material specimen results from other units in accordance with the ISP. Therefore, the proposed change revises the UFSAR to incorporate a description of BSEP's participation in the ISP, and revises the UFSAR table for reactor pressure vessel surveillance capsules to specify that the remaining specimens will be standby specimens (i.e., no removal schedule specified). The UFSAR will state that BSEP will participate in the NRC-approved version of the ISP, which is currently described in Reference 1 and approved by Reference 2. PEC will evaluate any subsequent NRC-approved changes to the ISP, and will either incorporate these changes into the UFSAR or seek NRC approval of a proposed alternative material specimen surveillance program in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

PEC will make supporting changes to the Technical Specification (TS) Bases in accordance with TS 5.5.10, "Technical Specification (TS) Bases Control Program," following approval of the license amendment to implement the BWRVIP ISP. Enclosure 3 provides marked-up TS Bases pages for Unit 1; similar changes are planned for the affected Unit 2 TS Bases pages. These pages are being submitted for information only and do not require issuance by the NRC.

3.0 Background

In References 3 and 4, as supplemented by References 5 and 6, the BWRVIP described the technical basis for the development and implementation of an ISP intended to support operation of all United States BWR reactor pressure vessels through the completion of each facility's current 40-year operating license.

The BWRVIP ISP was developed in response to an issue raised by the NRC regarding the potential lack of adequate unirradiated baseline Charpy V-notch (CVN) data for one or more materials in plant-specific reactor pressure vessel surveillance programs at several BWRs. The lack of baseline properties would inhibit the ability to effectively monitor changes in the fracture toughness properties of reactor pressure vessel materials in accordance with 10 CFR 50, Appendix H. The BWRVIP ISP, as approved by the NRC, resolves this issue.

Implementation of the ISP will provide additional benefits. When the original surveillance materials were selected for plant-specific surveillance programs, the state of knowledge concerning reactor pressure vessel material response to irradiation and post-irradiation fracture toughness was not the same as it is today. As a result, many facilities did not include what would be identified today as the plant's limiting reactor pressure vessel materials in their surveillance programs. The effort to identify and evaluate materials from other BWRs, which may better represent a facility's limiting materials, should improve the overall evaluation of BWR reactor pressure vessel embrittlement. Second, the inclusion of data from the testing of BWR Owners' Group (BWROG) Supplemental Surveillance Program (SSP) capsules will improve overall quality of the data used to evaluate BWR reactor pressure vessel embrittlement. Furthermore, occupational exposure will be reduced due to elimination of the need for some facilities to remove material specimens. Overall, the combined benefits of the ISP are substantial. Implementation of the ISP is also expected to reduce the cost of surveillance testing and analysis since surveillance materials that are of little or no value will no longer be tested, either because they lack adequate unirradiated baseline CVN data or because they are not the best representative materials.

In Reference 7, the NRC determined that the ISP proposed by the BWRVIP is an acceptable alternative to all existing BWR plant-specific reactor pressure vessel surveillance programs for the purpose of maintaining compliance with the requirements of 10 CFR 50, Appendix H. Reference 7 stated that licensees electing to participate in the ISP should provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP and to address the compatibility of comparison of neutron fluences calculated for each reactor pressure vessel with neutron fluences calculated for the surveillance capsules in the ISP.

In Reference 8, the NRC stated that licensees who elect to participate in the ISP should submit a license amendment request to incorporate this program into their licensing basis.

4.0 Regulatory Requirements and Guidance

10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," invokes 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." 10 CFR Part 50, Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary, including reactor pressure vessels. In order to support evaluations to demonstrate that compliance with these requirements will be maintained, information regarding irradiated reactor pressure vessel material properties and the neutron fluence level of a licensee's reactor pressure vessel is necessary. Therefore, 10 CFR 50.60 also invokes 10 CFR 50, Appendix H, which requires implementation of a reactor pressure vessel material surveillance program.

10 CFR 50, Appendix H, paragraph III.C addresses an alternative to individual plant-specific reactor pressure vessel programs. In accordance with paragraph III.C of Appendix H, a reactor pressure vessel ISP may be implemented, with approval of the Director of the Office of Nuclear Reactor regulation, by two or more facilities with similar design and operating features. Paragraph III.C.1 also sets forth specific criteria upon which approval of an ISP should be based. The criteria include:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

b. Each reactor must have an adequate dosimetry program.

c. There must be adequate arrangement for data sharing between plants.

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

In addition, paragraph III.C.2 specifies that no reduction is permitted in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor. Finally, paragraph III.C.3 states that no reduction in the amount of testing is permitted unless previously authorized by the Director of the Office of Nuclear Reactor Regulation.

In Reference 7, the NRC documented the BWRVIP ISP met the above criteria.

Based on the above, PEC has determined that the proposed changes do not alter compliance with any applicable regulatory requirement or criteria. Therefore, the proposed changes do not require any exemptions or relief from regulatory requirements, other than modifying the basis for

BSEP's compliance with the requirements of 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

5.0 Technical Analysis

BWRVIP-78 (Reference 3) described the technical basis related to material selection and testing on which the BWRVIP ISP was developed. The report primarily addressed the methodology established to identify existing plant-specific surveillance capsules and surveillance capsules from the SSP initiated by the BWROG in the late 1980s, which contain important surveillance materials for inclusion in the ISP. In this case, "important" surveillance materials are those which best represent the actual limiting (i.e., in terms of predicted fracture behavior) plate and weld materials from which BWR reactor pressure vessels were constructed. The report also established the connection between the identified surveillance materials and the specific reactor pressure vessel plate or weld materials which they represent and provided a proposed test matrix for the ISP.

BWRVIP-86 (Reference 4) addressed determination of ISP surveillance capsule withdrawal and testing dates, information on ISP project administration, additional information on neutron fluence determination issues, additional information on data utilization and sharing, and information on licensing aspects of ISP implementation.

The NRC approval of the technical basis for the ISP (Reference 7) stated that licensees electing to participate in the ISP should provide information regarding what specific neutron fluence methodology will be implemented as part of participation in the ISP.

In Reference 9, as supplemented in Reference 10, PEC submitted proposed changes to the BSEP, Units 1 and 2 pressure-temperature limit curves for operation to 32 effective full power years (EFPY); the existing pressure-temperature limit curves support operation to 19 EFPY. The neutron fluence evaluation supporting the pressure-temperature limit curves changes was developed using the methodologies described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 1, 2001. Evaluation of irradiation effects on vessel beltline materials was performed using Regulatory Guide 1.99, Revision 2, "Radiation Enbrittlement of Reactor Vessel Materials." NRC approval of the Reference 9 submittal is pending.

Reference 7 also stated that licensees should address the compatibility of comparison of neutron fluences calculated for each reactor pressure vessel with neutron fluences calculated for the surveillance capsules in the ISP. The BWRVIP will evaluate the neutron fluence of the surveillance capsules withdrawn as part of the ISP using a method consistent with Regulatory Guide 1.190. This will ensure compatibility of the methods used to calculate reactor pressure vessel and capsule neutron fluence.

6.0 Regulatory Requirements and Guidance

The regulatory requirements for an ISP are discussed in Section 4.0 above. In Reference 7, the NRC concluded that the BWRVIP ISP met the regulatory criteria for approval of an ISP.

7.0 No Significant Hazards Consideration

PEC has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change adopts an integrated surveillance program (ISP) for reactor vessel material specimen surveillances. The ISP ensures that the reactor pressure vessel will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change. The initial conditions and methodologies used in accident analyses remain unchanged. The proposed change does not revise the design assumptions for systems or components used to mitigate the consequences of accidents. The accident analyses results are not affected by this proposed change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change adopts an integrated surveillance program (ISP) for reactor vessel material specimen surveillances. The ISP ensures that the reactor pressure vessel will continue to meet all applicable fracture toughness requirements. No physical changes to the facilities will result from the proposed change. The proposed change does not affect the design or operation of any system, structure, or component in the facilities. The safety functions of the related systems, structures, or components are not changed in any manner, nor is the reliability of any system, structure, or component reduced. The change does not affect the manner by which the facilities are operated and does not change any facility, structure, or component. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change has no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The proposed change does not affect any plant safety parameters or setpoints. No physical or operational changes to the facilities will result from the proposed change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PEC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

8.0 Environmental Considerations

A review has determined that the proposed amendments change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20 and impose a new surveillance requirement. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

9.0 References

1. Letter from Carl Terry (BWRVIP) to U.S. NRC, "Project No. 704 – BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," dated November 12, 2002. (ADAMS Accession Number ML023190487)

2. Letter from William H. Bateman (USNRC) to Carl Terry (BWRVIP), "NRC Staff Review of BWRVIP-86-A, ' BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,'" dated December 16, 2002. (ADAMS Accession Number ML023500309)

3. Letter from Carl Terry (BWRVIP) to U.S. NRC, "Project No. 704 –BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan (BWRVIP-86)," dated December 22, 1999.

4. Letter from Carl Terry (BWRVIP) to U.S. NRC, "Project No. 704 – BWRVIP-86: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program Implementation Plan," EPRI Technical Report 1000888, dated December 22, 2000.

5. Letter from Carl Terry (BWRVIP) to U.S. NRC, "Project No. 704 – BWRVIP Response to NRC Request for Additional Information Regarding BWRVIP-78," dated December 15, 2000. (ADAMS Accession Number ML003778471)

6. Letter from Carl Terry (BWRVIP) to U.S. NRC, "Project No. 704 – BWRVIP Response to Second NRC Request for Additional Information Regarding BWRVIP-78," dated May 30, 2001. (ADAMS Accession Number ML011560296)

7. Letter from William H. Bateman (USNRC) to Carl Terry (BWRVIP), "Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" dated February 1, 2002. (ADAMS Accession Number ML020380691)

8. NRC Regulatory Issue Summary 2002-05, "NRC Approval of Boiling Water Reactor Pressure Vessel Integrated Surveillance Program," dated April 8, 2002. (ADAMS Accession Number ML020660522)

9. Letter from John S. Keenan (PEC) to USNRC, "Request for License Amendments to Revise Technical Specification Pressure-Temperature Limit Curves," dated June 26, 2002. (ADAMS Accession Number ML021890061)

10. Letter from John S. Keenan (PEC) to USNRC, "Response to Request for Additional Information, Proposed License Amendment to Revise Pressure-Temperature Curve Limits (NRC TAC Nos. MB5579 and MB5580)," dated November 22, 2002. (ADAMS Accession Number ML023370591)

10.0 Precedents

The NRC approved the BWRVIP ISP in February 2002. The NRC has approved participation of the following licensees in the ISP:

1. Letter from Kahtan N. Jabbour (USNRC) to Mr. J. A. Scalice (Tennessee Valley Authority), "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Re: Implementation of the Boiling-Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program to Address the Requirements of Appendix H to 10 CFR Part 50 (TAC Nos. MB6677 and MB6678)," dated January 28, 2003. (ADAMS Accession Number ML030290418)

2. Letter from John F. Stang (USNRC) to Mr. William T. O'Connor, Jr. (Detroit Edison Company), "Fermi 2 – Issuance of Amendment Re: Implementation of the BWRVIP Reactor Pressure Vessel ISP To Address The Requirements of Appendix H to 10 CFR Part 50 TAC No. MB5840) (ADAMS Accession Number ML023300129)

Marked-up UFSAR Pages

d) The maximum NDT temperature of all material opposite the center of the active fuel of the core was determined. For all other base material in the vessel, dropweight tests were performed per ASTM E208 to assure that the NDT temperature was at or below + 40°F; however, the actual NDT temperature was not determined. The NDT temperature of Category A and B welds was at or below + 10°F. Impact properties of all other "as-fabricated" carbon and low alloy steel pressure containing material and the vessel support skirt material met the requirements of the ASME Code, Section III, N-330 at a temperature no higher than 40°F.

e) "Weak" direction Charpy V-Notch specimens were not obtained because they were not required by the ASME Code at the time of vessel purchase. The reactor vessel was designed and fabricated to the requirements of Section III of the 1965 ASME Code with Addenda through Winter 1967. Section III of the 1971 ASME Code and Code Case 1514 do not apply to the Brunswick Steam Electric Plant (BSEP) vessels since they were not in existence at the time of purchase.

f) Longitudinal and circumferential weld joints in the reactor vessel are oriented so as not to intersect openings or penetrations, wherever practical. The region of highest neutron flux occurs at approximately mid-plane of the core.

g) Sufficient longitudinal Charpy V-Notch specimens are available as part of the Materials Surveillance Program (see Section 5.3.1.6) so that the upper shelf fracture energy levels could be determined for those materials included in the surveillance program.

h) Refer to Appendix 5.3A, Part 2, paragraph 1.9.30, for the RPV specification requirements that were imposed on beltline material to reduce sensitivity to irradiation embrittlement.

5.3.1.6 Material Surveillance. The surveillance test program provides for the preparation of a series of Charpy V-Notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a welded joint of reactor steels, which simulates a welded joint in the core area of the reactor vessel (Reference 5.3.1-2). The specimens and neutron monitor wires were placed near core mid-height adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall. The specimens were installed prior to initial criticality. Selected groups of specimens may be removed at intervals over the lifetime of the reactor and tested to compare with the documented unirradiated mechanical properties for the material. The reactor material irradiation surveillance specimens are removed and examined to determine changes in material properties at the intervals shown in Table 5.3.1-2. The results of these examinations are used to update the Pressure-Temperature Limit curves in Technical Specifications, as applicable. The cumulative effective full power years is determined at least once per 24 months.

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The number of capsules, number and type of specimens, and an estimated withdrawal schedule is presented in Table 5.3.1-2. The neutron flux wire was located in the capsule located at 30° azimuth, and was removed at approximately one year to verify flux calculations.

Neither ASTM-E-185-72 nor 10CFR50.55a, Appendix H existed at the time the BSEP vessels were purchased. Reactor vessel material surveillance program for BSEP, however, does meet the intent of ASTM-E-185-66, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The BSEP program is essentially identical to that stated for the LaSalle County Station Preliminary Safety Analysis Report (PSAR), Amendment Number 4, in answer to question 4.8. Also, the BSEP program of reactor vessel surveillance is shown in Reference 5.3.1-3 to be completely responsive to 10CFR50, Appendix H.

The greater than 1 Mev fast neutron flux density and fluence (integrated neutron flux) at the reactor vessel wall capsule holder position of BSEP Unit 1 have been determined to be 1.4×10^9 n/cm^2 sec (at full power) and 4.7×10^{16} n/cm^2, respectively, following the analysis of irradiated iron and copper flux dosimeters (Reference 5.3.1-4).

More detailed information on the material surveillance program is provided in Appendix 5.3B. Part A is applicable to Unit 1 (NEDO-24161) and Part B is applicable to Unit 2 (NEDO-24157).

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel top head is secured to the reactor vessel by studs, nuts, and bushings which are designed to be tightened with a stud tensioner.

INSERT FOR UFSAR SECTION 5.3.1.6

The reactor pressure vessel (RPV) materials surveillance program was designed to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment. Neither ASTM E-185-1972 nor 10 CFR 50, Appendix H existed at the time the BSEP vessels were purchased. The reactor vessel material surveillance program for BSEP; however, does meet the intent of ASTM E185-1966, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels."

The original surveillance test program provided for the preparation of a series of Charpy V-Notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a welded joint of reactor steels, which simulated a welded joint in the core area of the reactor vessel (Reference 5.3.1-2). The specimen holders were placed near core mid-height adjacent to the reactor vessel wall at azimuth locations 30°, 120°, and 300°. The specimens were installed prior to initial criticality. The neutron flux wire was installed in the capsule located at 30° azimuth, and was removed after approximately one year of operation to verify flux calculations.

The number of surveillance specimen capsules and types of specimens are presented in Table 5.3.1-2. The original material surveillance program was designed for removal and testing of selected groups of these specimens at intervals over the lifetime of the reactor. Following removal, these specimens would be tested and compared to the documented unirradiated mechanical properties for the material. The results of these examinations would then be used to update the pressure-temperature limit curves in Technical Specifications, as applicable.

More detailed information on the original Brunswick Units 1 and 2 material surveillance program is provided in Appendix 5.3B. Part A is applicable to Unit 1 (NEDO-24161) and Part B is applicable to Unit 2 (NEDO-24157).

In May 2002, the NRC approved revisions to the BSEP Operating Licenses and Technical Specifications to allow an increase in the maximum power level at each BSEP unit from 2558 megawatts thermal (MWt) to 2923 MWt. This represents a power increase of approximately 15 percent. For operation at 2923 MWt, fluence assessments were performed for the Brunswick Units 1 and 2 pressure vessel beltline regions based on the guidance specified in Regulatory Guide 1.190 (Reference 5.3.1-3). Exposures of the materials were determined on a plant and fuel cycle specific basis. Based on this assessment, the 32 effective full power year (EFPY) peak fluence for Unit 1 is 2.5E18 n/cm^2 and for Unit 2 is 2.4E18 n/cm^2. (Reference 5.3.1-4).

In 2003, the NRC approved BSEP Units 1 and 2 participation in the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as described in BWRVIP-78 (Reference 5.3.1-5) and BWRVIP-86-A (Reference 5.3.1-6). The ISP meets the requirements of 10 CFR 50, Appendix H and ASTM E185-1972. The current ISP withdrawal schedule is based on the latest NRC-approved revision of BWRVIP-86-A (Reference 5.3.1-7). Based on this schedule, no capsules are scheduled to be withdrawn from either Brunswick Units 1 or Unit 2.

PAGE NOT CHANGED
PROVIDED FOR INFORMATION
ONLY

TABLE 5.3.1-1
REACTOR PRESSURE VESSEL MATERIALS

COMPONENT	FORM	MATERIAL	SPEC. (ASTM/ASME) AND CODE CASE (cc)
Heads, shell	rolled plate	low alloy steel	SA533 Gr B Class 1 cc 1338 - 4 alt. 1
Closure flange	forged rings	low alloy steel	A508 Cl 2 cc 1332 Para. 5 & cc 1359 - 1
Nozzles	forged shapes	low alloy steel	A508 Cl 2 cc 1332 Para. 5 & cc 1359 - 1
Control rod drive stub tubes	tube	Ni-Cr-Fe	SB167 cc 1420
Control rod drive housing	pipe	austenitic stainless steel	SA312 grade TP 304 SA182 grade F 304
In-core housing	pipe	austenitic stainless steel	SA213 grade TP 304 SA182 grade F 304
Vessel support skirt cylinder	rolled plate	low alloy steel	SA533 Gr B Class 1 cc 1338 - 4 alt. 1
Shroud support	rolled plate	Ni-Cr-Fe	SB168 cc 1338 - 4 alt. 1
Nozzle thermal sleeves	pipe	austenitic stainless steel & Ni-Cr-Fe	SA312 Type 304 L, SA336 F8 cc 1359-1 SB168 cc 1338 - 4 alt. 1 & SB166
Seal leak detector nozzles	forging pipe		SA1B2 Type 316 NG SA376 Type 316 NF (Unit 2 Only)
	forging	Ni-Cr-Fe	SB166
Vessel support skirt transition (welding to bottom head)	rolled plate	low alloy steel	SA533 Gr B Class 1 cc 1338 - 4 alt. 1
Cladding	weld overlay	austenitic stainless steel	*

*The finished surface shall have a composition meeting the following requirements:

Cr	18.0 - 22.0	percent
Ni	8.0 - 12.0	percent
Si max	0.90	percent
Mn max	2.50	percent
C max	0.08	percent
Ferrite	3	percent, minimum

TABLE 5.3.1-2

BRUNSWICK SURVEILLANCE SPECIMENS

UNIT 1

CAPSULE NO. & LOCATION	NUMBER OF SPECIMENS	TYPE OF SPECIMENS	MATERIAL	WITHDRAWAL INTERVAL **
1* 30° Azimuth	3 3 2 12 12 12	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	~~**~~ Standby ~~**~~
2 120° Azimuth	3 3 2 8 8 8	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	
3 300° Azimuth	2 2 2 8 8 8	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	Capsule shall be withdrawn during the refueling outage immediately preceding or following the accumulation of 8 effective full power years. Capsule was withdrawn at 8.7 EFPY

UNIT 2

CAPSULE NO. & LOCATION	NUMBER OF SPECIMENS	TYPE OF SPECIMENS	MATERIAL	WITHDRAWAL INTERVAL **
1* 30° Azimuth	3 3 2 8 8 8	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	** Capsule was withdrawn at 17.35 EFPY
2 120° Azimuth	3 3 2 8 8 8	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	~~**~~ Standby
3 300° Azimuth	2 2 2 8 8 8	Tensile Tensile Tensile Charpy Charpy Charpy	Base Weld HAZ Base Weld HAZ	Capsule shall be withdrawn during the refueling outage immediately preceding or following the accumulation of 10 effective full power years. Capsule was withdrawn at 10.9 EFPY

* Flux wire - 1 year.

** The schedule for removal of the second and third capsules (CAPSULE No.2 and 1) shall be proposed after the results of the first capsule (Capsule No. 3) have been evaluated.

Based on Reference 5.3.1-6.

INSERT REFERENCES

REFERENCES: SECTION 5.3

5.3.1-1 GE Report - NEDO 10115.

5.3.1-2 Higgins, J. P., "Modified Surveillance Program for General Electric BWR Pressure Vessel Steels," APED-5490, GE Atomic Power Equipment Department, May 1967.

5.3.1-3 ~~"Reactor Vessel Material Surveillance Program," (Attach 1 to Letter, NG-77-968, dated August 24, 1977, from CP&L to NRC).~~

5.3.1-4 ~~G. C. Martin, D. E. Farmer, W. W. Sabol, "Determination of Fast Neutron Flux Density and Fluence: Brunswick Unit 1 Power Station," March 10, 1980 (attach to Letter G-KBI-0-53, dated March 26, 1980, from GE to CP&L).~~

5.3.3-1 "Feedwater Sparger Cold Flow Vibration Tests," GE Report - NEDO-20554.

5.3.3-2 "Design and Analysis of Control Rod Drive Reactor Vessel Penetrations," APED-5703, November 1968.

5.3.3-3 Kobsa, I. R. and Wetzel, V. R., "Design and Analysis of Control Rod Drive Reactor Vessel Penetrations," APED-5703, GE Atomic Power Equipment Department, November 1968.

5.3.3-4 "An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident," NEDO-10029, July 1969.

5.3.3-5 Letter, GE-79-158, dated June 14, 1979, from CP&L to NRC.

5.3.3-6 CP&L Report, "Review of Recirculation System Safe End NDE Program," December 22, 1978, Revision 1 (forwarded by Letter, GD-79-134, dated January 15, 1979 from CP&L to NRC).

5.3.3-7 CP&L Report, - "BSEP Unit 2, Recirculation System Safe End Inspection," (Encl 1 to Letter, GD-79-1017, dated April 17, 1979, from CP&L to NRC).

5.3.3-8 Stress Report -218, BWR-CP&L, Brunswick Steam Electric Plant, Chicago Bridge and Iron (CBI).

5.3.3-9 GE Report NEDO-22196, DRF B11-00227, March 1983, Reactor Pressure Vessel Thermal Cycle Fatigue Evaluation for BSEP Units 1 & 2.

5.3.3-10 Structural Integrity Report SIR-93-037, "Fatigue Usage to Date for Reactor Pressure Vessel Components, BSEP Unit 1 and Unit 2", April 16, 1993.

5.3.3-11 SECY-95-245, NRC Policy Issue, Completion of Fatigue Action Plan.

INSERT FOR REFERENCES

5.3.1-3 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

5.3.1-4 General Electric Report GE-NE-A22-00113-07-01, Revision 0, Task T0301, "RPV Fracture Toughness Evaluation," June 2001.

5.3.1-5 Letter from Carl Terry (BWRVIP) to USNRC, "Project No. 704 - BWR Vessel and Internals Project – BWR Integrated Surveillance Program Plan (BWRVIP-78)," dated December 22, 1999.

5.3.1-6 Letter from Carl Terry (BWRVIP) to USNRC, "Project No. 704 – BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," dated November 12, 2002. (ADAMS Accession Number ML023190487)

5.3.1-7 Letter from William H. Bateman (USNRC) to Carl Terry (BWRVIP), "NRC Staff Review of BWRVIP-86-A, 'BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan,'" dated December 16, 2002. (ADAMS Accession Number ML023500309)

Marked-up Unit 1 Technical Specification Bases Pages

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

This Specification contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The criticality curve provides limits for both heatup and cooldown during criticality.

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

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

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The P/T limit curves in this Specification were developed in accordance with the 1989 Edition of the ASME Code, Section XI, Appendix G (Ref. 3). These P/T limit curves were developed using the initiation fracture toughness, K_{Ic}, for the allowable material fracture toughness. The use of

(continued)

BASES

BACKGROUND (continued)

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

REPLACE WITH INSERT

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

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

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

The criticality limits include the Reference 1 requirement that they be at least 40°F above the noncritical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. ~~8~~ 9), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

INSERT FOR BASES BACKGROUND SECTION

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 (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.

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. 8

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

LCO

The elements of this LCO are:

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

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

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

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

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

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

(continued)

BASES

LCO (continued)

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

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

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

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

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

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

APPLICABILITY

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

ACTIONS

A.1 and A.2

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

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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

9

ASME Code, Section XI, Appendix E (Ref. 8), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

C.1 and C.2

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

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This engineering evaluation will determine the effect of the P/T limit violation on the fracture toughness properties of the RCS. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 8 9), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 and SR 3.4.9.2

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

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

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

SR 3.4.9.3

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

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

SR 3.4.9.4 and SR 3.4.9.5

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.4 and SR 3.4.9.5 (continued)

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

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

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

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

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

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

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

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

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. 1989 Edition of the ASME Code, Section XI, Appendix G.

4. ASME Code Case N-640. "Alternate References Fracture Toughness for Development of P-T Limit Curves Section XI. Division 1."

Replace with Insert

5. UFSAR, Section 5.3.1.6 and Appendix 5.3B.

6. 10 CFR 50, Appendix H.

7. Regulatory Guide 1.99, Revision 2, May 1988.

8. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

9. Calculation 0B11-0005. "Development of RPV Pressure-Temperature Curves For BNP Units 1 and 2 For Up To 32 EFPY of Plant Operation," dated November 8, 2000.

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

INSERT FOR BASES REFERENCES

5. BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan.
6. 10 CFR 50, Appendix H.
7. Regulatory Guide 1.99, Revision 2, May 1988.
8. Calculation 0B11-0012, "Neutron Exposure Evaluations for the Core Shroud and Pressure Vessel Brunswick Units 1 and 2," (latest approved version).
9. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

List of Regulatory Commitments

The following table identifies those actions committed to by Progress Energy Carolinas, Inc. (i.e., formerly known as Carolina Power & Light (CP&L) Company) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to the Manager – Support Services at the Brunswick Steam Electric Plant.

Commitment	Schedule
1. No commitments were made in this request.	N/A