CP&L
A Progress Energy Company

John S. Keenan
Vice President
Brunswick Nuclear Plant

JAN 1 7 2001

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TSC 00TSC10

10 CFR 50.90
10 CFR 50.55a(a)(3)(i)

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

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 AND REQUEST FOR APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW CHECK VALVE TESTING FREQUENCY

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Parts 50.90 and 2.101, Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. The proposed license amendments will relax the Surveillance Requirement (SR) 3.6.1.3.7 frequency by allowing a representative sample, approximately 20 percent, of excess flow check valves (EFCVs) to be tested every 24 months, such that each EFCV is tested once every 10 years. This change incorporates Technical Specification Task Force (TSTF) Item Number 334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," approved by the NRC on September 18, 2000.

EFCVs are included in the Inservice Testing (IST) Program for BSEP, Units 1 and 2. For the third 10-year interval, Refueling Justification RFJ-01 was included in the IST Program to justify testing these valves at each refueling outage instead of the quarterly test requirements for check valves stipulated by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Therefore, in accordance with 10 CFR 50.55a(a)(3)(i), CP&L is requesting relief from the requirements of the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, for BSEP, Units 1 and 2. A relief request, designated as VRR-14, is provided in Enclosure 10 which will match the IST Program requirements to those of the Surveillance Requirement in the proposed license amendments.

Markups of the Unit 1 Bases pages associated with the proposed license amendments are included in Enclosure 9. These pages are provided for information only and do not require issuance by the NRC.

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CP&L is providing, in accordance with 10 CFR 50.91(b), Mr. Mel Fry of the State of North Carolina a copy of this request.

Approval of these proposed license amendments and relief request is requested by July 31, 2001, to support planning activities associated with the upcoming BSEP, Unit 1 refueling outage (i.e., B114R1).

Please refer any questions regarding this submittal to Mr. David C. DiCello, Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,

John S Keenan

John S. Keenan

WRM/wrm

Enclosures:

1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Environmental Considerations
4. Page Change Instructions
5. Typed Technical Specification Page - Unit No. 1
6. Typed Technical Specification Page - Unit No. 2
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9. Marked-up Technical Specification Bases Pages - Unit No. 1
10. Relief Request VRR-14, Revision 0

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: August 29, 2004

cc (with enclosures):

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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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REQUEST FOR LICENSE AMENDMENTS AND REQUEST FOR
APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW
CHECK VALVE TESTING FREQUENCY

Basis For Change Request

The proposed change is to relax Brunswick Steam Electric Plant (BSEP), Units 1 and 2 Surveillance Requirement (SR) 3.6.1.3.7 by allowing a "representative sample" of excess flow check valves (EFCVs) (i.e., a sample of approximately 20 percent of the EFCV population) to be tested every 24 months, such that each EFCV will be tested at least once every 10 years. This change incorporates Technical Specification Task Force (TSTF) Item Number 334, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," approved by the NRC on September 18, 2000. The proposed change is being requested to minimize personnel radiation exposure during refueling outages, reduce outage critical path time without significantly impacting the risk to the general public, and increase the availability of instrumentation during outages.

Current Requirement

Surveillance Requirement 3.6.1.3.7 states:

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	Verify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months

Proposed Change

Surveillance Requirement 3.6.1.3.7 is being revised to state:

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months

Basis For Proposed Change

EFCVs are used in the BSEP, Units 1 and 2 containments to limit the release of process fluid in the event of a reactor instrumentation line rupture. EFCVs are not required to close in response to a containment isolation signal, nor are they required to operate under post loss-of-coolant accident (LOCA) conditions.

BSEP Technical Specifications (TS) SR 3.6.1.3.7 currently requires verification of the actuation capability of each reactor instrumentation line EFCV every 24 months. This SR demonstrates that each reactor instrumentation line EFCV is operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break downstream of the valve. The 24 month frequency is based on the typical performance of this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power.

The Boiling Water Reactor Owners' Group (BWROG) has developed a basis, described in Topical Report B21-00658-01, "Excess Flow Check Valve Testing Frequency Relaxation," dated November 1988, for reducing the EFCV testing frequency. This report was initially submitted to the NRC as part of a Duane Arnold Energy Center proposed license amendment on April 12, 1999. The BWROG report was supplemented by BWROG letter dated January 6, 2000, "Generic Response to NRC Request For Additional information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements." The report was approved for use by an NRC Safety Evaluation dated March 14, 2000. Additionally, issues raised by the NRC in the March 14, 2000, Safety Evaluation were addressed in the issuance of General Electric Topical Report NEDO-32977-A (i.e., BWROG Topical Report B21-00658-01), "Excess Flow Check Valve Testing Relaxation," dated June 2000. TSTF-334 was previously submitted to the NRC and was approved on September 18, 2000.

The BWROG topical report concluded that the change in EFCV test frequency has an insignificant impact on EFCV reliability. The topical report evaluated the reliability of EFCVs at various boiling water reactor plants, including BSEP, based on information covering a 10-year period. Industry experience with EFCVs indicate that they have very low failure rates. A large portion of the reported test failures was related to test methodologies and not actual valve failures.

The BWROG, in the topical report, stated that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with the design-basis LOCA would be of a sufficiently low probability to be outside of the design basis. The high reliability of these valves and the low risk significance associated with an EFCV failure to isolate an instrument line break are the primary bases for this change as documented in the BWROG report referenced above.

The approach used to evaluate this issue for BSEP, Units 1 and 2, is consistent with the BWROG topical report. The primary consideration associated with the proposed change is the verification of valve isolation function, which limits the reactor coolant release following the rupture of an instrument line. Based on a review of available testing records since 1995, CP&L has determined that there has not been a failure of a BSEP EFCV to isolate.

At BSEP, Units 1 and 2, each instrument line connected to the reactor coolant pressure boundary is equipped with an EFCV (Reference: Updated Final Safety Analysis Report Subsection 6.2.4.1.2). For each line, a 0.25-inch flow-restricting orifice is provided (Reference: UFSAR Subsection 6.2.4.2). Leakage from an instrument line rupture upstream of the EFCV would be minimized by the line size or the flow-restricting orifice in the line. The rate and quantity of process fluid loss from an instrument line rupture is well within the capability of the reactor coolant make-up systems (i.e., the High Pressure Coolant Injection, Reactor Core Isolation Cooling, or Residual Heat Removal systems). The proposed change does not alter the plants' instrument line design in any manner. The integrity and functional performance of the secondary containment and Standby Gas Treatment system are not impaired by this proposed change. The potential offsite radiological exposure from an unisolated instrument line break is bounded by the main steam line break analysis and is substantially below the guidelines of 10 CFR 100. Therefore, a failure of an EFCV, although not expected as a result of this proposed change, is bounded by the previous evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by this proposed change.

The reduced testing associated with this proposed change will result in an increase in the availability of the instrumentation during the outages, a significant saving in outage critical-path time and cost without significantly impacting the health and safety of the general public, and significant dose savings.

The enclosed TS mark-ups have been revised based on the recommended changes contained in TSTF-334, Revision 2. The Reviewer's Note, associated with TSTF-334, addresses the requirements for adopting the relaxation, including the selection of performance criteria and basis to ensure that the licensee's corrective action program can provide meaningful feedback for appropriate corrective actions. Any EFCV failures that may occur will be documented in the BSEP Corrective Action Program as a surveillance test failure. The Corrective Action Program will ensure that the failure will be evaluated to identify common failure mode, industry experience, and to review for similar component failure history.

The TS Bases are being revised to state that a representative sample of EFCVs consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). The Bases will also state that EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments.

Precedents

Precedents exist for the proposed revision. Duane Arnold, Fermi 2, and Browns Ferry 2 and 3 have previously submitted requests for relaxing the surveillance frequency for EFCV testing using BWROG Topical Report B21-00658-01 as a basis for the change. The Duane Arnold and Fermi 2 applications have been approved by the NRC.

Inservice Testing Program

EFCVs are included in the Inservice Testing (IST) Program for BSEP, Units 1 and 2. For the third 10-year interval, which started on May 10, 1998, Refueling Justification RFJ-01 was included in the IST Program to justify testing these valves at each refueling outage instead of the quarterly test requirements for check valves stipulated by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

In accordance with 10 CFR 50.55a(a)(3)(i), CP&L is requesting relief from the requirements of the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, for BSEP, Unit Nos. 1 and 2. A new relief request, designated as VRR-14, is included in Enclosure 10 for NRC approval. This proposed relief request will match the IST Program requirements to those of the SR in the proposed license amendments. As part of implementing the proposed license amendments, Refueling Justification RFJ-01 will be superseded and replaced with Relief Request VRR-14 for the remainder of the third 10-year interval.

References:

1. Boiling Water Reactor Owners' Group Topical Report B21-00658-01, "Excess Flow Check Valve Testing Frequency Relaxation," dated November 1988.
2. General Electric Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.
3. BSEP, Unit Nos. 1 and 2, Updated Final Safety Analysis Report, Subsection 6.2.4.1.2.
4. BSEP, Unit Nos. 1 and 2, Updated Final Safety Analysis Report, Subsection 6.2.4.2.

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
REQUEST FOR LICENSE AMENDMENTS AND REQUEST FOR APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW CHECK VALVE TESTING FREQUENCY

10 CFR 50.92 Evaluation

Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, to relax Surveillance Requirement 3.6.1.3.7 by allowing a "representative sample" of excess flow check valves to be tested every 24 months, such that each excess flow check valve will be tested at least once every 10 years. The proposed change is being requested to minimize personnel radiation exposure during refueling outages, reduce outage critical path time without significantly impacting the risk to the general public, and increase the availability of instrumentation during outages. CP&L has concluded that the proposed change to the BSEP, Unit 1 and 2 Technical Specifications does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The current surveillance requirement frequency requires each reactor instrumentation line excess flow check valve to be tested every 24 months. The excess flow check valves at BSEP are designed to close automatically in the event of a line break downstream of the valve. The proposed change allows a reduction in the number of excess flow check valves to be tested every 24 months to approximately 20 percent of the valves each operating cycle. Industry operating experience demonstrates a high level of reliability for these excess flow check valves. A failure of an excess flow check valve to isolate cannot initiate previously evaluated accidents. Therefore, there is no increase in the probability of occurrence of an accident as a result of this proposed change.

   The postulated failure of an excess flow check valve to isolate is bounded by the limiting analysis in the Updated Final Safety Analysis Report (UFSAR). For a postulated break of an instrument line upstream of an excess flow check valve, leakage from the line rupture would be minimized by the line size or the flow-restricting orifice in the line. The rate and quantity of process fluid loss from an instrument line rupture is well within the capability of the reactor coolant make-up systems. The proposed change does not alter the design of the plants' instrument lines in any manner, and the integrity and functional performance of the secondary containment and Standby Gas Treatment system are not affected by this proposed change. The potential offsite radiological exposure associated with a postulated instrument line rupture upstream of an excess flow check valve is bounded by the main steam line break analysis and is substantially below the guidelines of 10 CFR 100. Therefore, the proposed license amendments do not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change allows a reduced number of excess flow check valves to be tested each operating cycle. No other change in requirements are being proposed. Industry operating experience demonstrates the high reliability of the excess flow check valves. The potential failure of an excess flow check valve to isolate is bounded by the main steam line break analysis. The proposed license amendments do not physically alter the plants and will not alter the operation of the structures, systems, and components described in the UFSAR. Therefore, a new or different kind of accident will not be created.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. Industry experience with excess flow check valves indicates that they have very low failure rates. The postulated failure of an excess flow check valve to isolate as a result of reduced testing is bounded by the limiting analysis in the UFSAR, which is the main steam line break analysis. For a postulated break of an instrument line upstream of an excess flow check valve, leakage from the line rupture would be minimized by the line size or the flow-restricting orifice in the line. The rate and quantity of process fluid loss from an instrument line rupture is well within the capability of the reactor coolant make-up systems. The proposed change does not alter the design of the plants' instrument line design in any manner, and the integrity and functional performance of the secondary containment and standby gas treatment system are not affected by this proposed change. Therefore, the proposed license amendments do not involve a significant reduction in the margin of safety.

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
REQUEST FOR LICENSE AMENDMENTS AND REQUEST FOR
APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW
CHECK VALVE TESTING FREQUENCY

Environmental Considerations

Carolina Power & Light (CP&L) Company is requesting a revision to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, to relax Surveillance Requirement 3.6.1.3.7 by allowing a "representative sample" of excess flow check valves to be tested every 24 months, such that each excess flow check valve will be tested at least once every 10 years. The proposed change is being requested to minimize personnel radiation exposure during refueling outages, reduce outage critical path time without significantly impacting the risk to the general public, and increase the availability of instrumentation during outages. CP&L has concluded that the proposed change to the BSEP, Unit 1 and 2 Technical Specifications is eligible for categorical exclusion from performing an environmental assessment. In support of this determination, an evaluation of each of the three (3) criteria set forth in 10 CFR 51.22(c)(9) is provided below.

1. The proposed license amendments do not involve a significant hazards consideration, as shown in Enclosure 2.

2. The proposed license amendments do not result in a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite. The potential failure of an excess flow check valve to isolate as a result of the proposed reduction in test frequency is bounded by the current main steam line break analysis. The proposed license amendments do not introduce any new equipment nor require the excess flow check valves or current mitigating systems to perform a different type of function than they are presently designed to perform. The proposed license amendments do not alter the function of the excess flow check valves or other existing equipment and will ensure that the consequences of any previously evaluated accident do not increase. Therefore, CP&L has concluded that there will not be a significant increase in the types or amounts of any effluent that may be released offsite and, as such, the proposed relaxation in the testing frequency for the excess flow check valves does not involve irreversible environmental consequences beyond those already associated with normal operation.

3. The proposed license amendments do not result in an increase in individual or cumulative occupational radiation exposure. The proposed change is being requested to minimize personnel radiation exposure during refueling outages by reducing the number of excess flow check valves that are tested during each operating cycle. Therefore, the proposed

license amendments are expected to result in a decrease in individual or cumulative occupational radiation exposure.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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Page Change Instructions

UNIT 1	
Removed page	Inserted page
3.6-13	3.6-13

UNIT 2	
Removed page	Inserted page
3.6-13	3.6-13

ENCLOSURE 5

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 AND REQUEST FOR
APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW
CHECK VALVE TESTING FREQUENCY

Typed Technical Specification Page - Unit No. 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 6

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 AND REQUEST FOR
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CHECK VALVE TESTING FREQUENCY

Typed Technical Specification Page - Unit No. 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 7

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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REQUEST FOR LICENSE AMENDMENTS AND REQUEST FOR
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CHECK VALVE TESTING FREQUENCY

Marked-up Technical Specification Page - Unit No. 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7 EFCVs	Verify a representative sample of ~~each~~ reactor instrumentation line ~~EFCV~~ actuates to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 8

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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CHECK VALVE TESTING FREQUENCY

Marked-up Technical Specification Page - Unit No. 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.7	Verify ~~each~~ a representative sample of reactor instrumentation line ~~EFCV~~ EFCVs actuate~~s~~ to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.8	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Inservice Testing Program
SR 3.6.1.3.9	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 9

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 AND REQUEST FOR
APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW
CHECK VALVE TESTING FREQUENCY

Revised Technical Specification Bases Pages - Unit No. 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

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

a representative sample of

SR 3.6.1.3.7

(s)

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve s actuates to the isolation position on an actual or simulated instrument line break signal. This may be accomplished by cycling the EFCV through one complete cycle of full travel. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has demonstrated that these components will pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

INSERT 1

INSERT 2

SR 3.6.1.3.8

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

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

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

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

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

REFERENCES

1. UFSAR, Chapter 15.

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

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

4. Technical Requirements Manual.

5. UFSAR, Section 15.2.3.

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

(continued)

BASES

REFERENCES (continued)

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

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

9. ANSI/ANS 56.8-1994.

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

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

INSERT 3

INSERT 1:

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the samples are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time.

INSERT 2:

The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A (Ref. 12). Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

INSERT 3:

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

ENCLOSURE 10

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 AND REQUEST FOR
APPROVAL OF RELIEF REQUEST – REVISED EXCESS FLOW
CHECK VALVE TESTING FREQUENCY

Relief Request VRR-14, Revision 0

RELIEF REQUEST: VRR-14 (Rev. 0)

SUBJECT: Relaxed Surveillance Frequency For Excess Flow Check Valves

SYSTEMS:

Nuclear Steam Supply
Reactor Recirculation
Core Spray
High Pressure Cooling Injection
Reactor Core Isolation Cooling

COMPONENTS FOR WHICH RELIEF IS REQUESTED:

This request for relief is applicable to the following Brunswick Steam Electric Plant (BSEP), Unit 1 and 2 components:

1/2-B21-F008	1/2-B21-F049C	1/2-B21-F058G	1/2-E51-F043A
1/2-B21-F014A	1/2-B21-F042B	1/2-B21-F058H	1/2-E51-F043B
1/2-B21-F014B	1/2-B21-F044B	1/2-B21-F058L	1/2-E51-F043C
1/2-B21-F014C	1/2-B21-F046B	1/2-B21-F058M	1/2-E51-F043D
1/2-B21-F014D	1/2-B21-F047D	1/2-B21-F058N	
1/2-B21-F014E	1/2-B21-F048B	1/2-B21-F058P	1/2-B32-F042A
1/2-B21-F014F	1/2-B21-F049D	1/2-B21-F058R	1/2-B32-F041A
1/2-B21-F014G	1/2-B21-F050A	1/2-B21-F058S	1/2-B32-F042B
1/2-B21-F014H	1/2-B21-F050B	1/2-B21-F058T	1/2-B32-F041B
1/2-B21-F014K	1/2-B21-F050D	1/2-B21-F058U	1/2-B32-F058A
1/2-B21-F014J	1/2-B21-F052A	1/2-B21-F060	1/2-B32-F039C
1/2-B21-F014L	1/2-B21-F052B		1/2-B32-F039A
1/2-B21-F014M	1/2-B21-F052C	1/2-B21-IV-2149	1/2-B32-F006A
1/2-B21-F014N	1/2-B21-F050C	1/2-B21-IV-2455	1/2-B32-F005A
1/2-B21-F014P	1/2-B21-F052D	1/2-B21-IV-2456	1/2-B32-F042D
1/2-B21-F014R	1/2-B21-F054	1/2-B21-IV-2196	1/2-B32-F041D
1/2-B21-F014S	1/2-B21-F056		1/2-B32-F042C
1/2-B21-F040	1/2-B21-F058A	1/2-E21-F017A	1/2-B32-F041C
1/2-B21-F042A	1/2-B21-F058B	1/2-E21-F017B	1/2-B32-F058B
1/2-B21-F044A	1/2-B21-F058C		1/2-B32-F039B
1/2-B21-F046A	1/2-B21-F058D	1/2-E41-F023A	1/2-B32-F039D
1/2-B21-F047C	1/2-B21-F058E	1/2-E41-F023B	1/2-B32-F005B
1/2-B21-F048A	1/2-B21-F058F	1/2-E41-F023C	1/2-B32-F006B
		1/2-E41-F023D	

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CATEGORY:

AC

CLASS:

1 and 2

ASME SECTION XI CODE REQUIREMENT:

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition, is the applicable Code of record for the third 10-year interval Inservice Testing (IST) Program at BSEP, Units 1 and 2. Subarticle IWV-1100, Valve Testing, requires that valve testing be performed in accordance with the requirements in American National Standards Institute (ANSI)/ASME Operations and Maintenance Standards (OM-10). The ASME Code, Section XI, Table IWA-1600-1, Reference Standards and Specification, references the 1987 Edition with 1988 Addenda of the ASME/ANSI OM (Part 10).

OM-10 Paragraph 4.3.2.1, Exercising Test Frequency, requires that check valves, Category C valves, be exercised nominally every three months.

OM-10 Paragraph 4.3.3.2, Exercising Requirements, requires that check valves be exercised to the positions in which they perform their safety functions or examined at least once every reactor refueling outage.

OM-10, Paragraph 4.1, Valve Position Verification, requires verification of valve position for valves with remote position indicators at least once every two years to verify accurate position indication.

REQUESTED RELIEF:

In accordance with 10 CFR 50.55a(a)(3)(i), Carolina Power & Light (CP&L) Company requests relief from the requirements of Paragraph 4.3.2.2 of OM-10 which requires that check valves be exercised to the positions in which they perform their safety functions or examined every reactor refueling outage. CP&L also requests relief from the requirements of Paragraph 4.1 of OM-10 for verification of valve position.

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PROPOSED ALTERNATIVE:

CP&L proposes to test a representative sample of excess flow check valves consisting of an approximately equal number of excess flow check valves every 24 months, such that each excess flow check valve will be tested at least once every 10 years.

In addition, CP&L proposes to verify the open position indication at a frequency more often than what the ASME Code requires, but verify the close position indication in conjunction with excess flow check valve exercise tests.

BASIS FOR REQUESTING RELIEF:

An excess flow check valve is provided in each instrument process line that penetrates the primary containment and is part of the reactor coolant pressure boundary. The excess flow check valve is designed so that: (1) it will not close accidentally during normal operation, (2) will close if a rupture of the instrument line is indicated downstream of the valve, (3) can be re-opened when appropriate, and (4) has its status indicated in the control room.

Because of the design of excess flow check valves, verifying their closure indication requires a simulated instrument line break. Based on the burden and costs associated with testing these excess flow check valves, CP&L is proposing to perform the exercise tests and valve position verification tests on a sampling basis (i.e., approximately an equal number of excess flow check valves every 24 months such that each excess flow check valve is tested at least once every 10 years).

CP&L has determined that alternative excess flow check valve testing will provide an acceptable level of quality and safety for the following reasons:

1. Excess flow check valves are a simple and reliable device. The major components are a poppet and spring. The spring holds the poppet open only under static conditions, such that the valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve.

2. The Boiling Water Reactor Owners' Group (BWROG) has developed a basis, documented in Topical Report B21-00658-01, "Excess Flow Check Valve Testing Frequency Relaxation," dated November 1988, for reducing the EFCV testing frequency. This report was initially submitted to the NRC as part of a Duane Arnold Energy Center proposed license amendment on April 12, 1999. The BWROG report was supplemented by BWROG letter dated January 6, 2000, "Generic Response to NRC Request For Additional information on Lead Plant

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Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements." The report was approved for use by an NRC Safety Evaluation dated March 14, 2000. Additionally, issues raised by the NRC in the March 14, 2000, Safety Evaluation were addressed in the issuance of General Electric Topical Report NEDO-32977-A (i.e., BWROG Topical Report B21-00658-01), "Excess Flow Check Valve Testing Relaxation," dated June 2000. Technical Specification Task Force (TSTF) Item Number 334 (i.e., TSTF-334) was previously submitted to the NRC and was approved on September 18, 2000.

The BWROG topical report concluded that the change in excess flow check valve test frequency has an insignificant impact on excess flow check valve reliability. The topical report evaluated the reliability of excess flow check valves at various boiling water reactor plants, including BSEP, based on information covering a 10-year period. Industry experience with excess flow check valves indicate that they have very low failure rates. A large portion of the reported test failures at other plants was related to test methodologies and not actual valve failures.

Excess flow check valves have been extremely reliable throughout the industry. At BSEP, since 1995, no excess flow check valves have failed to close due to actual valve failure.

3. An orifice is installed on each of the affected instrument lines. The orifice limits leakage to a quantity where the integrity and functional performance of secondary containment and the associated safety systems are maintained. The process fluid loss for a postulated rupture of an instrument line is within the capability of the reactor coolant makeup systems.

4. The reduced testing associated with the alternative will result in an increase in the availability of the associated instrumentation during plant refueling outages. The reduced testing associated with the alternative will also reduce occupational radiological exposure.

5. The relief request will match the IST Program requirements to those of the Surveillance Requirement in the proposed license amendments accompanying this relief request.

REFERENCES:

ASME Code, Section XI, *Rules For Inservice Inspection of Nuclear Power Plants Components*, 1989 Edition.

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Part 10 of the American National Standards Institute/ASME Operations and Maintenance Standards (OM-10), *Inservice Testing of Valves in Light Water Reactor Power Plants*, 1987 Edition with 1988 Addenda.

Updated Final Safety Analysis Report, Subsection 6.2.4.2.

NRC Safety Guide 11 (Regulatory Guide 1.11), *Instrument Lines Penetrating Primary Reactor Containment.*