Britt T. McKinney
Sr. Vice President & Chief Nuclear Officer

PPL Susquehanna, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3149 Fax 570.542.1504
btmckinney@pplweb.com

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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 287 TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 255 TO LICENSE NPF-22: REVISION TO TECHNICAL SPECIFICATIONS 5.5.6 AND 5.5.12
PLA-6020

Docket Nos. 50-387 and 50-388

In accordance with the provisions of 10 CFR 50.90, PPL Susquehanna, LLC is submitting a request for an amendment to the Technical Specifications for Susquehanna Units 1 and 2.

The proposed amendment implements TSTF-343, Rev. 1, and TSTF-479, Rev. 0. TSTF-343 allows the performance of visual examinations on the primary containment pursuant to ASME Section XI Code, Subsections IWL and IWE in lieu of the visual examinations required by Regulatory Guide 1.163 and NEI 94-01. TSTF-479 extends the provisions of SR 3.0.2 to other IST frequencies that are not specified in the Technical Specifications.

Note: Only the extension of SR 3.0.2 applicability portion of TSTF-479 is being implemented with this amendment. A previous amendment implemented the OM Code portions of the TSTF.

These proposed changes have been reviewed by the Plant Operations Review Committee and by the Susquehanna Review Committee.

The Enclosure to this letter provides a description of the proposed changes. Attachment 1 provides the existing Technical Specification pages marked-up to show the proposed change. Attachment 2 provides the corresponding TS Bases "markup" pages. No new regulatory commitments are made herein.

A047

We request approval of the proposed License Amendment by January 1, 2007, with the amendment being implemented within 30 days following approval in order to use this exemption in the spring 2007 Susquehanna SES Unit 2 Refueling Outage.

In accordance with 10 CFR 50.91(b), PPL Susquehanna, LLC is providing the Commonwealth of Pennsylvania with a copy of this proposed License Amendment request.

If you have any questions regarding this submittal, please contact Mr. C. T. Coddington at (610) 774-4019.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 9-7-06

B. T. McKinney

B. T. McKinney

Enclosure: PPL Susquehanna Evaluation of the Proposed Changes

Attachments:
Attachment 1 - Proposed Technical Specification Changes Units 1 & 2, (Mark-ups)
Attachment 2 – Proposed Technical Specification Bases Changes Units 1 & 2, (Mark-ups provided for information.)

cc: NRC Region I
Mr. A. J. Blamey, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

Enclosure to PLA-6020

PPL Susquehanna, LLC Evaluation of Proposed Changes

Technical Specifications 5.5.6 and 5.5.12

1. DESCRIPTION
2. PROPOSED CHANGES
3. BACKGROUND
4. TECHNICAL ANALYSIS
5. REGULATORY SAFETY ANALYSIS
   - 5.1 No Significant Hazards Consideration
   - 5.2 Applicable Regulatory Requirements/Criteria
6. ENVIRONMENTAL CONSIDERATION
7. REFERENCES

PPL EVALUATION

Subject: TECHNICAL SPECIFICATION 5.5.6 and 5.5.12

1.0 DESCRIPTION

This is a request to amend Operating Licenses NPF-14 and NPF-22 for PPL Susquehanna, LLC (PPL), Susquehanna Steam Electric Station (SSES) Units 1 and 2 respectively.

Change to Technical Specification 5.5.6

The proposed change revises the Administrative Controls, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

Change to Technical Specification 5.5.12

The proposed change revises Primary Containment Leakage Rate Testing Program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

As a result of this proposed change, SSES will be required to perform one less visual inspection of the containment during the ten year interval. However, the requirements for inspection in Subsection IWE and IWL of Section XI are more rigorous than those currently required to be performed by the Technical Specifications.

The proposed changes have been approved by the NRC for plant-specific Technical Specifications several times. See References 5, 6, 7, 8, 9, and 10.

2.0 PROPOSED CHANGES

Change to Technical Specification 5.5.6

The proposed change will revise:

Inservice Testing Program

This specification is revised to indicate that the Inservice Testing Program shall include testing frequencies that are non-standard Frequencies utilized in the Inservice Testing Program in which the provisions of SR 3.0.2 are applicable. Specifically, it is revised to state:

> "The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice test activities."

Change to Technical Specification 5.5.12

This specification is revised to add the following exceptions to Regulatory Guide 1.163, "Performance- Based Containment Leak-Testing Program,"

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The SSES TS SR 3.6.1.1 Bases does not contain a description of the basis for the Surveillance Frequency. Such a description is required by the ISTS Writer's Guide. Therefore, the Bases change to describe the basis for the Surveillance Frequency is added for consistency.

3.0 BACKGROUND

Change to Technical Specification 5.5.6

In 1990, the ASME published the initial edition of the ASME OM Code that provides rules for inservice testing of pumps and valves. The ASME OM Code replaced Section XI of the Boiler and Pressure Vessel Code for inservice testing of pumps and valves.

Change to Technical Specification 5.5.12

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code (the Code). The final rule, Subpart 50.55a (g)(6)(ii)(B) of Title 10 of the Code of Federal Regulations (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

4.0 TECHNICAL ANALYSIS

Change to Technical Specification 5.5.6

The Technical Specification Inservice Testing Program is revised to indicate that the provisions of SR 3.0.2 are applicable to other IST Frequencies that are not specified in the Program. The Inservice Test Program may have Frequencies for testing that are based on risk and do not conform to the standard testing Frequencies specified in the Technical Specifications. For example, an Inservice Testing Program may use ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Plants," in lieu of stroke time testing. The Frequency of the Surveillance may be determined through a mix of risk informed and performance based means in accordance with the Inservice Testing Program. This is consistent with the guidance in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," which indicates that the 25% extension of the interval specified in the Frequency would apply to increased frequencies of 2 years or less the same way that it applies to regular frequencies. If a test interval is specified in 10 CFR 50.55a, the TS SR 3.0.2 Bases indicates that the requirements of the regulation take precedence over the TS.

Change to Technical Specification 5.5.12

The Technical Specification requirements for the Primary Containment Leakage Rate Testing Program specify that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Position C.3 of the regulatory guide states that "Section 9.2.1, 'Pretest Inspection and Test Methodology,' of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of Regulatory Guide 1.163 and NEI 94-01, the concrete surfaces of the containment must be visually examined in accordance with the ASME Section XI Code, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every five years, and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The visual examinations performed pursuant to Subsection IWL may be performed at any time during power operation or during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during maintenance or refueling outages since this in the only time that the liner plate is fully accessible.

The visual examinations performed pursuant to Subsections IWL and IWE are more rigorous than those performed pursuant to Regulatory Guide 1.163 and NEI 94-01. For example, Subarticle IWE-2320 requires the general visual examination to be the responsibility of an individual who is knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. Subsection IWE, Subarticle-2330 requires the examination to be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation.

Similarly, Subarticle IWL-2320 states that:

> "The Responsible Engineer shall be a Registered Professional Engineer experienced in evaluating the inservice condition of structural concrete. The Responsible Engineer shall have knowledge of the Design and

Construction Codes and other criteria used in design and construction of concrete containments in nuclear power plants."

The Responsible Engineer shall be responsible for the following:

(a) development of plans and procedures for examination of concrete surfaces;

(b) approval, instruction, and training of concrete examination personnel;

(c) evaluation of examination results;

(d) preparation or review of Repair/Replacement Plans and procedures;

(e) review of procedures for pressure tests following repair/replacement procedures;

(f) submittal of report to the Owner documenting results of examinations and repairs.

Based on the above, the Responsible Engineer will ensure that a comprehensive visual examination of the concrete is performed in accordance with Code requirements except where relief has been granted by the NRC. Furthermore, with respect to examinations performed pursuant to both Subsections IWL and IWE, visual examinations of both the concrete surfaces and the liner plate are reviewed by the ANII, in accordance with IWA-2110 and IWA-2120. The combination of the Code requirements for the rigor of the visual examinations plus the third party review will more than offset the fact that one fewer visual examination of the concrete will be performed during a 10-year interval. The fact that the concrete visual examination pursuant to Subsection IWL may be performed during power operation as opposed to during a refueling outage will have no effect on the quality of the examination and will provide flexibility in scheduling of the visual examinations.

5.0 REGULATORY SAFETY ANALYSIS

Change to Technical Specification 5.5.6

The proposed change revises the Administrative Controls, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

Change to Technical Specification 5.5.12

The proposed change revises Primary Containment Leakage Rate Testing Program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

5.1 No Significant Hazards Consideration

PPL Susquehanna, LLC (PPL) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 of occurrence or consequences of an accident previously evaluated?**

   Response: No.

Change to Technical Specification 5.5.6

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

Change to Technical Specification 5.5.12

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a (g)(4) for components classified as Code Class CC.

The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the Primary Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility.

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.

<u>**Change to Technical Specification 5.5.6**</u>

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3.

The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is

no increase in individual or cumulative occupational exposure. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

<u>Change to Technical Specification 5.5.12</u>

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50, paragraph 55a (g)(4) for components classified as Code Class CC.

The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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.

<u>Change to Technical Specification 5.5.6</u>

The proposed change revises the Inservice Testing Program for consistency with the requirements of 10 CFR 50.55a(f)(4) for pumps and valves which are classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3. The safety function of the affected pumps and valves will be maintained. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

Change to Technical Specification 5.5.12

The proposed change revises the Improved Standard Technical Specification Administrative Controls program requirements for consistency with the requirements of 10 CFR 50, paragraph 55a (g)(4) for components classified as Code Class CC.

The change affects the frequency of visual examinations that will be performed for the concrete surfaces of containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The safety function of the containment as a fission product barrier will be maintained.

5.2 Applicable Regulatory Requirements/Criteria

Change to Technical Specification 5.5.6

NRC regulation, 10 CFR 50.55a, defines the requirements for applying industry codes to each licensed nuclear powered facility. Licensees are required by 10 CFR 50.55a(f)(4)(i) to initially prepare programs to perform inservice testing of certain ASME Section III, Code Class 1, 2, and 3 pumps and valves during the initial 120-month interval. The regulations require that programs be developed utilizing the latest edition and addenda incorporated into paragraph (b) of 10 CFR 50.55a on the date 12 months prior to the date of issuance of the operating license subject to the limitations and modification identified in paragraph (b).

Change to Technical Specification 5.5.12

The regulatory basis for TS 3.6.1.1, "Primary Containment," is to ensure that the containment is capable of remaining leaktight following a loss of coolant accident. This ensures that offsite radiation exposures are maintained within the limits of 10 CFR 100.

10 CFR 50, Appendix A, General Design Criterion 16, "Design," requires that reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

6.0 ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna, LLC has evaluated the proposed changes and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed changes do not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

7.0 REFERENCES

1. 10 CFR 50.55a

2. SECY-99-017, "Proposed Amendment to 10 CFR 50.55a"

3. NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants"

4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program."

5. Letter dated January 18, 2000, to W. R. McCollum, Jr., Duke Energy Corporation, "Oconee Nuclear Station Units 1, 2, and 3 RE: Issuance of Amendments (TAC Nos. MA6568, MA6569, and MA6570)." Amendment Nos. 310.

6. Letter dated June 6, 2001, to J. B. Beasley, Jr., Southern Nuclear Operating Company, Inc, "Vogtle Electric Generating Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB1097 and MB1098)." Amendment Nos. 122 and 100.

7. Letter dated January 30, 2001, to C. H. Cruse, Constellation Nuclear, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 RE: Containment Tendon Surveillance Program – Amendment (TAC Nos. MB0011 and MB0012)." Amendment Nos. 240 and 214.

8. Letter dated January 31, 2001, to T. F. Plunkett, Florida Power and Light Company, "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Changes to Containment Structural Integrity Technical Specifications (TAC Nos. MA9047 and MA9048)." Amendment Nos. 210 and 204.

9. Letter dated March 19, 2004, to R. R. Overbeck, Arizona Public Service Company, "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Issuance of Amendment on Containment Tendon Surveillance Program and Containment Leakage Rate Testing Program (TAC Nos. MC1069, MC1070, and MC1071)." Amendment Nos. 151.

10. Letter dated March 17, 2004, to R. A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Issuance of Amendment Re: Containment Tendon Surveillance Program and Containment Leakage Rate Testing Program." Amendment No. 152.

Attachment 1 to PLA-6020

Proposed Technical Specification Changes

Units 1 & 2

(Mark-ups)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in the ASME Operation and Maintenance Code and applicable Addenda are as follows:

ASME Operation and Maintenance Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Operation and Maintenance Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

a. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

~~a~~c. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 4, 1992 Type A test shall be performed no later than May 3, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests:

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in the ASME Operation and Maintenance Code and applicable Addenda are as follows:

ASME Operation and Maintenance Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Operation and Maintenance Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

a. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

b. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

~~a~~c. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the October 31, 1992 Type A test shall be performed no later than October 30, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (Continued)

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Attachment 2 to PLA-6020

Technical Specification Bases Changes

(Mark-ups Provided for Information)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The primary containment concrete visual examinations may be performed during either power

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1 (continued)

operation, e.g., performed concurrently with other primary containment inspection-related activities, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside primary containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock leakage testing (SR 3.6.1.2.1) or resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.6) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to each startup after performing a required leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR Frequencies are as required by the Primary Containment Leakage Rate Testing Program. These periodic testing requirements verify that the primary containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

As noted in table B 3.6.1.3-1, an exemption to Appendix J is provided that isolation barriers which remain water filled or a water seal remains in the line post-LOCA are tested with water and the leakage is not included in the Type B and C 0.60 L_a total.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

measures drywell to suppression chamber leakage to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. The allowable limit is 10% of the acceptable SSES $A/\sqrt{k}$ design valve. For SSES, the $A/\sqrt{k}$ design value is .0535 ft^2.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and determining the leakage. The leakage test is performed when the 10 CFR 50, Appendix J, Type A test is performed in accordance with the Primary Containment Leakage Rate Testing Program. This testing Frequency was developed considering this test is performed in conjunction with the Integrated Leak rate test and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 24 months is required until the situation is remedied as evidenced by passing two consecutive tests.

SR 3.6.1.1.3

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through downcomers into the suppression pool. This SR measures suppression chamber-to-drywell vacuum breaker leakage to ensure the leakage paths that would bypass the suppression pool are within allowable limits. The total allowable leakage limit is 30% of the SR 3.6.1.1.2 limit. The allowable leakage per set is 12% of the SR 3.6.1.1.2 limit.

The leakage is determined by establishing a 4.3 psi differential pressure across the drywell-to-suppression chamber vacuum breakers and verifying the leakage. The

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.3 (continued)

leakage test is performed every 24 months. The 24 month Frequency was developed considering the surveillance must be performed during a unit outage. A Note is provided which allows this Surveillance not to be performed when SR 3.6.1.1.2 is performed. This is acceptable because SR 3.6.1.1.2 ensures the OPERABILITY of the pressure suppression function including the suppression chamber-to-drywell vacuum breakers.

REFERENCES

1. FSAR, Section 6.2.
2. FSAR, Section 15.
3. 10 CFR 50, Appendix J, Option B.
4. Nuclear Energy Institute, 94-01
5. ANSI/ANS 56.8-1994
6. Final Policy Statement on Technical Specifications Improvements July 22, 1993 (58 FR 39132)
7. Standard Review Plan 6.2.4, Rev. 1, September 1975

BASES

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The primary containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other primary containment inspection-related activities, or during a

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SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1 (continued)

maintenance or refuel outage. The visual examinations of the steel liner plate inside primary containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock leakage testing (SR 3.6.1.2.1) or resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.6) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to each startup after performing a required leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

As noted in Table B 3.6.1.3-1, an exemption to Appendix J is provided that isolation barriers which remain filled or a water seal remains in the line post-LOCA are tested with water and the leakage is not included in the Type B and C 0.60 L_a test.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell to suppression chamber leakage to ensure that the leakage paths that would bypass the suppression pool are within allowable limits. The allowable limit is 10% of the acceptable SSES $A/\sqrt{k}$ design value. For SSES, the $A/\sqrt{k}$ design value is .0535 ft^2.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and determining the leakage. The leakage test is performed when the 10 CFR 50, Appendix J, Type A test is performed in accordance with the Primary Containment Leakage Rate Testing Program. This testing Frequency was developed considering this test is performed in conjunction with the Integrated Leak rate test and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 24 months is required until the situation is remedied as evidenced by passing two consecutive tests.

SR 3.6.1.1.3

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through downcomers into the suppression pool. This SR measures suppression chamber-to-drywell vacuum breaker leakage to ensure the leakage paths that would bypass the suppression pool are within allowable limits. The total allowable leakage limit is 30% of the SR 3.6.1.1.2 limit. The allowable leakage per set is 12% of the SR 3.6.1.1.2 limit.

The leakage is determined by establishing a 4.3 psi differential pressure across the drywell-to-suppression chamber vacuum breakers and verifying the leakage. The leakage test is performed every 24 months. The 24 month Frequency was developed considering the surveillance must be performed during a unit outage. A Note is provided which allows this Surveillance not to be performed when SR 3.6.1.1.2 is performed. This is acceptable because SR 3.6.1.1.2 ensures the OPERABILITY of the pressure suppression function including the suppression chamber-to-drywell vacuum breakers.

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REFERENCES

1. FSAR, Section 6.2.
2. FSAR, Section 15.
3. 10 CFR 50, Appendix J, Option B.
4. Nuclear Energy Institute, 94-01
5. ANSI/ANS 56.8-1994
6. Final Policy Statement on Technical Specifications Improvements July 22, 1993 (58 FR 39132)
7. Standard Review Plan 6.2.4, Rev. 1, September 1975