TXU Power
Power

TXU Power
Comanche Peak Steam
Electric Station
P. O. Box 1002 (E01)
Glen Rose, TX 76043
Tel: 254 897 5209
Fax: 254 897 6652
mike.blevins@txu.com

Mike Blevins
Senior Vice President &
Chief Nuclear Officer

Ref: 10CFR50.90

CPSES-200602339
Log # TXX-06191
File # 00236

December 19, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 06-010
REVISION TO TECHNICAL SPECIFICATION 5.5.16,
"CONTAINMENT LEAKAGE RATE TESTING PROGRAM"

Dear Sir or Madam:

Pursuant to 10CFR50.90, TXU Generation Company LP (TXU Power) hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. This change request applies to both units.

The proposed change will revise TS 5.5.16 entitled "Containment Leakage Rate Testing Program". The proposed change revises TS 5.5.16, "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).

A017
A047

Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, TXU Power's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected Technical Specifications (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides proposed changes to the Technical Specification Bases for information only. These changes will be processed per CPSES site procedures. Attachment 4 provides retyped Technical Specifications pages which incorporate the requested changes. Attachment 5 provides retyped Technical Specifications Bases pages which incorporate the proposed changes (for information only).

TXU Power requests approval of the proposed License Amendment by December 31, 2007, to be implemented within 120 days of the issuance of the license amendment. The approval date was administratively selected to allow for NRC review but the plant does not require this amendment to allow continued safe full power operations.

In accordance with 10CFR50.91(b), TXU Power is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new or revised commitments.

Should you have any questions, please contact Mr. Carl Corbin at (254) 897-0121.

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

Executed on December 19, 2006.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Director, Oversight and Regulatory Affairs

CBC

Attachments
1. Description and Assessment
2. Proposed Technical Specifications Changes (Mark-up)
3. Proposed Technical Specifications Bases Changes (Markup For Information Only)
4. Retyped Technical Specifications Pages
5. Retyped Technical Specifications Bases Pages (for information)

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

Ms. Alice Rogers
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78756-3189

ATTACHMENT 1 to TXX-06191

DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

By this letter, TXU Generation Company LP (TXU Power) requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. Proposed change LAR 06-010 is a request to revise Technical Specifications (TS) 5.5.16, "Containment Leakage Rate Testing Program," for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2.

The proposed changes are based on the NRC-approved Technical Specifications Task Force Improved Standard Technical Specifications Change Traveler 343, Rev. 1, "Revision to TS 5.5.16 and associated TS Bases for Containment Leakage Rate Testing Program" (TSTF-343). The proposed changes are consistent with the wording in section 5.5.16 of NUREG-1431, Revision 3.1, "Standard Technical Specifications, Westinghouse Plants" (STS), since STS has already incorporated TSTF-343. The proposed change revises TS 5.5.16, "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). The proposed change also revises TS Bases Surveillance Requirements SR 3.6.1.1. The TS Bases for SR 3.6.1.1 is revised for consistency with the requirements of the ASME Code Section XI, Subsection IWL, and applicable addenda as required by 10 CFR 50.55a.

CPSES Final Safety Analysis Report Chapter 6 will be updated as a result of this License Amendment Request.

2.0 PROPOSED CHANGE

The proposed change would revise TS 5.5.16 by adding 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 TS Bases for SR 3.6.1.1 is revised for consistency with the requirements of the ASME Code Section XI, Subsection IWL, and applicable addenda as required by 10 CFR 50.55a. The TS Bases changes are provided for information only.

3.0 BACKGROUND

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.

The containment structure is a fully continuous, steel-lined, reinforced concrete structure. It consists of a vertical cylinder and a hemispherical dome.

4.0 TECHNICAL ANALYSIS

The Technical Specification requirements for the 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 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 use 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 must be reviewed by an Inspector employed by a State or municipality of the United States or an Inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance, 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 ANALYSIS

5.1 No Significant Hazards Consideration

TXU Power has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   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 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. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   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. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   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 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.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluations, TXU Power concludes that the proposed amendment(s) present no significant hazards under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for PWR ISTS 3.6.1, "Containment," is to ensure that the containment is capable of remaining leak-tight 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 leak-tight 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.

This Technical Specification change will not reduce the leak-tightness of the containment. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

TXU Power has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. TXU Power has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

7.0. PRECEDENTS

7.1 10 CFR 50.55a, "Codes and Standards."

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

7.3 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

7.4 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.

7.5 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.

7.6 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.

7.7 Letter dated March 19, 2004, to G. 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.

7.8 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 2 to TXX-06191

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UP)

Pages 5.0-27

5.5 Programs and Manuals (continued)

5.5.16. Containment Leakage Rate Testing Program

a. A program shall be established to implement 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 ~~exception~~: 98

INSERT 1

~~1.~~ 3 NEI 94-01 – 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)."

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

e. The provision of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves. 66

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

(continued)

INSERT 1 FOR TECHINCAL SPECIFICATIONS PAGE 5.0-27

exceptions:

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 the 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 CFR50, 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.

ATTACHMENT 3 to TXX-06191

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES
(Markup For Information Only)

Pages B 3.6-4

BASES

ACTIONS (continued)

B.1 and B.2

If 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 6 hours and to MODE 5 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

INSERT 2

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program 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. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 15.
3. FSAR, Section 6.2.

INSERT 2 FOR TECHINCAL SPECIFICATIONS BASES PAGE 3.6-4

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

ATTACHMENT 4 to TXX-06191

RETYPED TECHNICAL SPECIFICATIONS PAGES

Pages 5.0-27
5.0-28
5.0-28a

5.5.16. Containment Leakage Rate Testing Program

a. A program shall be established to implement 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:

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 the 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, 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.

3. NEI 94-01 – 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)."

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

(continued)

5.5.16. Containment Leakage Rate Testing Program (continued)

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ P_a.

e. The provision of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves.

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

5.5.17 Technical Requirements Manual (TRM)

The TRM contains selected requirements which do not meet the criteria for inclusion in the Technical Specification but are important to the operation of CPSES. Much of the information in the TRM was relocated from the TS.

Changes to the TRM shall be made under appropriate administrative controls and reviews. Changes may be made to the TRM without prior NRC approval provided the changes do not require either a change to the TS or NRC approval pursuant to 10 CFR 50.59. TRM changes require approval of the Plant Manager*.

5.5.18 Configuration Risk Management Program (CRMP)

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed Completion Time has been granted. The program shall include the following elements:

a. Provisions for the control and implementation of a Level 1, at-power, internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

(continued)

* Duties may be performed by the Vice President of Nuclear Operations if that organizational position is assigned.

5.5 Programs and Manuals

5.5.18 Configuration Risk Management Program (CRMP) (continued)

b. Provisions for performing an assessment prior to entering the LCO Action for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Action for unplanned entry into the LCO Action.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues, and external events, qualitatively or quantitatively.

5.5.19 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer for the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.

ATTACHMENT 5 to TXX-06191

RETYPED TECHNICAL SPECIFICATIONS BASES PAGES

Pages B 3.6-4

ACTIONS (continued)

B.1 and B.2

If 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 6 hours and to MODE 5 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

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities, or during a maintenance or refueling outage. The visual examinations of the steel liner plate inside 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 and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program 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. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 15.
3. FSAR, Section 6.2.