Pacific Gas and
Electric Company®

James R. Becker
Vice President
Diablo Canyon Operations and
Station Director

Diablo Canyon Power Plant
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Fax: 805.545.4234

December 29, 2006

PG&E Letter DCL-06-142

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

Docket Nos. 50-275 and 50-323
Diablo Canyon Power Plant (DCPP) Units 1 and 2
License Amendment Request 06-09,
Revision to Technical Specification 5.5.16, "Containment Leakage Rate Testing Program" (TSTF-343)

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to update Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program."

The proposed change revises TS 5.5.16 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 CFR 50.55a(b)(2)(ix).

This LAR is consistent with NRC-approved Industry/Technical Specification Task Force (TSTF) Traveler number TSTF-343, "Containment Structural Integrity."

Enclosure 1 contains a description of the proposed changes, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and retyped (clean) TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes for information only. TS Bases changes are provided for information only and will be implemented pursuant to TS 5.5.14, "Technical Specifications Bases Control Program," at the time the license amendments are implemented.

Pacific Gas and Electric Company (PG&E) has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or



A017
A047

environmental assessment needs to be prepared in connection with the issuance of this amendment.

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than December 20, 2007. PG&E requests the license amendment(s) be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

This communication contains no new or revised commitments.

If you have any questions or require additional information, please contact Stan Ketelsen at 805-545-4720.

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

Executed on December 29, 2006.

James R. Becker
*Vice President - Diablo Canyon Operations and Station Director*

mjrm/4557
Enclosures

| | |
|---|---|
| cc: | Edgar Bailey, DHS |
| | Bruce S. Mallett |
| | Terry W. Jackson |
| | Diablo Distribution |
| cc/enc: | Alan B. Wang |

# EVALUATION

## 1.0 SUMMARY DESCRIPTION

This letter is a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Operating Licenses by incorporating the attached change into the DCPP Unit 1 and 2 Technical Specifications (TS).

The proposed changes would revise TS 5.5.16, "Containment Leakage Rate Testing Program," for DCPP Units 1 and 2. The proposed changes are based on the NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler 343, Revision 1, "Containment Structural Integrity," (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 CFR 50.55a(b)(2)(ix). The proposed change also revises the TS Bases for Surveillance Requirement (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.

As a result of this change, DCPP will be required to perform fewer visual inspections 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.

# 2.0 DETAILED DESCRIPTION

## 2.1 Proposed Changes

The proposed change would revise TS 5.5.16 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 TS Bases for SR 3.6.1.1 are 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 included for information only.

The proposed TS changes are noted on the marked-up TS page provided in Enclosure 2. The proposed retyped TS pages are provided in Enclosure 3. The revised TS Bases pages are provided for information only in Enclosure 4.

## 2.2 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 required licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

The containment structure is a steel lined, reinforced concrete structure. It consists of a vertical cylindrical structure with a hemispherical dome.

## 3.0 TECHNICAL EVALUATION

### 3.1 Containment Design Basis

The TS 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 alternating between units for a 2-unit plant, 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.

### 3.2 Containment Safety Analysis Basis

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 fewer visual examinations 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.

# 4.0 REGULATORY ANALYSIS

## 4.1 No Significant Hazards Consideration

Pacific Gas and Electric Company (PG&E) 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 or consequences of an accident previously evaluated?

Response: No.

The proposed change revises the Technical Specification (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. Does the proposed change create the possibility of a new or different 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 of the 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 a 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 a 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.

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 of the 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 evaluation, PG&E concludes that the proposed change presents a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

### 4.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for Pressurized Water Reactor Improved Standard Technical Specification (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 change will not reduce the leak-tightness of the containment.

### 4.3 Conclusions

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.

## 5.0 ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that it does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6.0 REFERENCES

### 6.1 Precedent

1. 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 No. 310

2. 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

3. 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

4. Letter 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 No. 151

5. 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

Proposed Technical Specification Changes (marked-up)

INSERT 1

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. The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

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.

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," D dated September 1995, ~~The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.~~ INSERT 1

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 47 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 overall leakage rate acceptance criterion 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 $\geq 10$ psig.

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

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

(continued)

Proposed Technical Specification Changes (retyped)

| <u>Remove Page</u> | <u>Insert Page</u> |
|---|---|
| 5.0-24 | 5.0-24 |
| 5.0-24a | 5.0-24a |

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

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.

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 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 ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

   3. The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 47 psig.

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

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment overall leakage rate acceptance criterion 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 $\geq 10$ psig.

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

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

5.5.17 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, of the following:

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

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

(continued)

Changes to Technical Specification Bases Pages
(For information only)

INSERT 1

The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during 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.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, 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 containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

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 1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements as specified in the Containment Leakage Rate Testing Program (Ref. 1). Failure to meet air lock and purge valve with resilient seal leakage limits specified in the 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 $\leq 0.75\ L_a$ 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. SR Frequencies are as required by Containment Leakage Rate Testing Program. These periodic testing

(continued)

BASES

| | |
|---|---|
| SURVEILLANCE REQUIREMENTS | <u>SR 3.6.1.1</u> (continued)<br>requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.<br><br><u>SR 3.6.1.2</u><br>Not Used |
| REFERENCES | 1. 10 CFR 50, Appendix J, Option B, and 10 CFR 50.55a.<br>2. FSAR, Chapter 15.<br>3. FSAR, Section 6.2.<br>4. ASME Code, Section XI, Subsection IWL and Subsection IWE |