Pacific Gas and Electric Company®

James M. Welsch
Vice President, Nuclear Generation

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

805.545.3242
E-Mail: JMW1@pge.com

10 CFR 50.90

May 12, 2016

PG&E Letter DCL-16-057

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

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 16-03
Application to Adopt NEI 94-01 Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"

References: 1. NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008 (ADAMS Accession No. ML100620847)

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed 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 implement the performance based option of 10 CFR 50, Appendix J.

The proposed change revises Technical Specification (TS) 5.5.16 to replace the reference to Regulatory Guide 1.163, and 10 CFR 50, Appendix J, "Option B – Performance-Based Requirements," with a reference to NEI 94-01, Revision 2-A (Reference 1), dated October 2008. PG&E will adopt Reference 1 to implement a performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. In addition, this LAR proposes to remove the third exception under TS 5.5.16 for the one-time 15-year Type A test interval beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2.

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than May 12, 2017. PG&E requests the license amendment(s) be made effective upon NRC issuance, to be implemented within 180 days from the date of issuance.

A001
A006
A053
NRR

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter includes no revisions to existing regulatory commitments.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

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

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

Executed on May 12, 2016.

Sincerely,

James M. Welsch
Vice President, Nuclear Generation

E1d7/4418/50559745
Enclosure
cc: Diablo Distribution
cc/enc: John P. Reynoso, NRC Acting Senior Resident Inspector
Marc L. Dapas, NRC Region IV
Gonzalo L. Perez, Branch Chief, California Department of Public Health
Balwant K. Singal, NRR Senior Project Manager

Evaluation of the Proposed Change

License Amendment Request 16-03
Application to Adopt NEI 94-01 Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"

ATTACHMENTS:

1. Proposed Technical Specification Page Markup
2. Proposed Retyped Technical Specification Page
3. Evaluation of Risk Significance of Permanent ILRT Extension
4. Proposed Updated Final Safety Analysis Report Markup (for Information Only)
5. Glossary of Acronyms

EVALUATION

1. SUMMARY DESCRIPTION

For convenience, Enclosure Attachment 5 includes a glossary of acronyms used within this enclosure.

Pursuant to 10 CFR 50.90, PG&E requests an amendment to the DCPP Operating License to revise TS 5.5.16, "Containment Leakage Rate Testing Program," to allow the following:

- Increase in the existing Type A test interval from 10 years to 15 years in accordance with NEI 94-01, Revision 2-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," (Reference 31).
- Adopt the use of ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements," (Reference 3), as referenced in NEI 94-01, Revision 2-A.
- Adopt an allowable test interval extension of 9 months, which is shorter than the currently allowed 25 percent grace, for the Type A, Type B, and Type C leakage tests in accordance with NEI 94-01, Revision 2-A.

The proposed change revises TS 5.5.16 to replace the reference to RG 1.163, and 10 CFR 50, Appendix J, "Option B – Performance-Based Requirements," with a reference to NEI 94-01, Revision 2-A, dated October 2008. PG&E will then adopt NEI 94-01, Revision 2-A to implement a performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J, for DCPP. In addition, this LAR proposes to remove the third exception under TS 5.5.16 for the one-time 15 year Type A test interval beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2.

2. DETAILED DESCRIPTION

2.1 <u>Current Containment Leakage Rate Testing Program</u>

DCPP TS 5.5.16, "Containment Leakage Rate Testing Program," currently states, in part:

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 RG 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.

2.2 Proposed TS Changes Description

The proposed TS change would allow an increase in the Type A test interval from its current 10 year frequency to a maximum of 15 years, in accordance with NEI 94-01, Revision 2-A. In addition, this LAR proposes to adopt an allowable test interval extension of 9 months, for Type A, Type B, and Type C leakage tests in accordance with NEI 94-01, Revision 2-A, for non-routine, emergent conditions. The changes are described as follows:

- RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 43) is being replaced with NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October 2008 in TS 5.5.16.

- This LAR also proposes the following administrative change to TS 5.5.16:

  The information regarding the one-time 15-year Type A test interval beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2 is being deleted since the tests associated with these exceptions were successfully completed as scheduled.

The proposed change will revise TS 5.5.16 to state, in part:

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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based

Option of 10 CFR Part 50, Appendix J," dated October 2008, 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.

A markup of TS 5.5.16 is provided in Attachment 1 and a retype of TS 5.5.16 is provided in Attachment 2.

Attachment 3 contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of RG 1.174, Revision 2 (Reference 44) and RG 1.200, Revision 2 (Reference 45). The risk assessment concluded that increasing the Type A testing required performance interval from 10 to 15 years represents a small change per RG 1.174 in the DCPP risk profile.

3. TECHNICAL EVALUATION

3.1 Description of Containment System

The reactor containment building for each unit is a steel-lined, reinforced concrete cylindrical building with a dome roof that completely encloses the reactor and RCS. It ensures that leakage of radioactive materials to the environment is minimized even if gross failure of the RCS were to occur immediately following a DCPP Hosgri Earthquake.

The exterior shell consists of a 142-ft high cylinder, topped with a hemispherical dome. The thickness of the concrete cylindrical walls is 3-2/3-ft and the thickness of the concrete roof is 2-1/2-ft. Both Units' containments have a nominal inside diameter of 140-ft. The concrete floor pad is 153-ft in diameter with a thickness of 14-1/2-ft, with the reactor cavity near the center of the floor pad.

A continuous welded steel liner plate is provided on the entire inside face of the containment which prevents the release of radioactive materials to the

environment under any postulated accident condition. The wall liner is 3/8 in. thick, except for the bottom section (approximately 4-ft high) next to the basemat where the thickness is 3/4 in. The dome liner is 3/8 in. thick. The thickness of the basemat liner is 1/4 in. and this liner is covered with a 24-in. thick concrete floor slab for protection of the liner. The top of the concrete floor slab is at the 91-ft elevation.

An anchorage system is provided to prevent instability of the liner during an earthquake. The bottom of the wall liner is attached to the basemat by an anchorage system that consists of reinforcing bars attached to the wall liner. The wall liner and dome liner are anchored to the concrete with L-shaped welded studs placed in approximately an equilateral triangle pattern. The basemat liner is anchored to the basemat concrete through steel T-shaped sections anchored in the basemat.

Most of the reinforcement bars in the concrete shell are placed near the outside face of the shell to minimize temperature stresses; diagonal bars are also provided for seismic loads in the bottom portion of the shell and inside layer reinforcing is provided elsewhere to assure liner anchorage. There are no vertical reinforcing or special bars inclined at 45 degrees as is commonly used to resist tangential shears in cylindrical walls. Instead, diagonal bars inclined at 60 degrees are used to resist both membrane shear and vertical tension in the cylindrical walls. The concrete dome reinforcing bars are placed in a geodesic pattern matching the wall reinforcing. Diagonal bars from the cylindrical wall become a part of the geodesic pattern of the dome, forming continuous loops with both ends anchored in the basemat.

The equipment hatch is an 18-1/2-ft normal diameter opening for transportation of equipment through the containment wall. The opening is bound by a 3 in. thick, 2 ft wide, 18-1/2-ft inside-diameter steel band, welded to the liner plate. An approximately 21-5/6 ft square area of the liner around the steel band is thickened to 1-1/2 in. so that the liner material displaced by the opening is replaced.

The personnel hatch provides access to the inside of containment through a 9-ft diameter 17-1/2-ft long penetration sleeve with bulkheads and sealed access doors at both ends. The penetration sleeve is made of 3/4-in. and 3/8-in. steel plate, welded to a 13-ft diameter liner insert plate. The bulkheads are made of 1-1/8-in. steel plate stiffeners.

The emergency hatch is similar to the personnel hatch except that it is smaller. The access door is 30 in. in diameter. The penetration sleeve is 5-ft in diameter and is constructed of a 1/2-in. plate which is welded to a 110-in. diameter, 1-in. thick liner insert plate.

Typically, penetrations consist of a sleeve embedded in concrete and welded to the liner. A portion of the liner adjacent to the sleeve is made of a thicker plate thickness of 1-1/8 in. to replace the material displaced by the penetration and to reduce local stress concentrations.

The DCPP containment isolation system does not contain any stainless steel bellows that are credited as a containment pressure boundary. Therefore, the concerns of IN 92-20 are not applicable to DCPP and no inservice inspections of stainless steel bellows are required.

3.1.1 Containment Overpressure on ECCS Performance

The ECCS is designed so that adequate NPSH is provided to system pumps. Atmospheric pressure is equal to vapor pressure for NPSH concerns and no credit is taken for sub-cooling. Containment overpressure is not credited in the calculation of available NPSH.

Adequate NPSH is shown to be available for all ECCS pumps during the recirculation phase of a postulated LOCA. A calculation was performed to show the maximum expected flow rates (i.e., run out conditions) for the RHR Pumps, SI Pumps, and the CCPs during both the injection and recirculation phases of post LOCA recovery. The calculation concluded that in all cases the NPSH available is higher than the NPSH required for each of the ECCS pumps.

RHR Pumps:

The NPSH of the RHR Pumps was evaluated for normal plant shutdown operation, and for both the injection and recirculation modes of operation for the design basis accident. The limiting NPSH requirement is evaluated based on recirculation operation. The NPSH evaluation was based on all pumps (i.e., both RHR pumps, CCP1 and CCP2, both SI pumps, and both containment spray pumps operating at the maximum design (run out) flow rates.

Safety Injection and Centrifugal Charging Pumps 1 and 2:

The NPSH for the SI pumps and CCP1 and CCP2 was evaluated for both the injection and recirculation modes of operation for the design basis accident. The limiting NPSH available requirement is evaluated based on the end of the injection mode of operation. The limiting NPSH was determined from the elevation head and vapor pressure of the water in the RWST, which is at atmospheric pressure, and the pressure drop in the suction piping from the tank to the pumps. The NPSH evaluation is based on all pumps operating at the maximum design flow rates. Following switchover to the recirculation mode, adequate NPSH is supplied from the containment recirculation sump by the booster action of the RHR pumps.

3.2 Justification for the Proposed Technical Specification Change

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. 10 CFR 50, Appendix J also ensures that periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests:

1) Type A tests, intended to measure the containment overall integrated leakage rate;

2) Type B tests, intended to detect leakage paths and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for containment penetrations, and;

3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths.

Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not challenged by Type B and C testing.

In 1995, 10 CFR 50, Appendix J, was amended to provide a performance-based Option B for the containment leakage testing requirements. Option B allows test intervals for Type A, Type B, and Type C testing to be determined by using a performance-based approach. Performance-based test intervals are determined by consideration of the operating history of a component and the resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50, Appendix J refers to both the performance history necessary to extend the test intervals as well as to the acceptance criteria necessary to meet the requirements of Option B.

Also in 1995, RG 1.163 was issued. RG 1.163 endorsed NEI 94-01, Revision 0, (Reference 30) with certain modifications and additions. Option B, in conjunction with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory Type A test performance history (i.e., two consecutively successful Type A tests) to reduce the test frequency for the containment Type A test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 41) and EPRI TR-104285 (Reference 16), both of which showed that the risk increase associated with

extending the Type A test surveillance interval was very small. In addition to the 10-year Type A test interval, provisions for extending the test interval an additional 15 months was considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, Revision 0, but stated that such extension "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Rev. 2-A, was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, subject to the limitations and conditions noted in Section 4.0 of the NRC SER for NEI 94-01. The SER is provided in the front portion of the NEI 94-01, Revision 2-A report. NEI 94-01, Revision 2-A, includes provisions for extending Type A test intervals up to a maximum of 15 years and incorporates the regulatory positions stated in RG 1.163. It details a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. Justification for extending test intervals is based on the performance history and risk considerations.

The NRC has provided the following guidance concerning the use of test interval extensions in the deferral of Type A tests beyond the 15-year interval specified in NEI 94-01, Revision 2-A, NRC SER, Section 3.1.1.2:

> "As noted above, Section 9.2.3, NEI TR 94-01 [sic], Rev. 2, states, 'Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history.' However, Section 9.1 states that the 'required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes.' The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01 [sic], Rev. 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists."

In 2012, NEI 94-01, Revision 3-A (Reference 32), was issued. DCPP has evaluated the additional extension of Type C intervals permitted by NEI 94-01, Revision 3-A, and has chosen not to adopt NEI 94-01, Revision 3-A, at this time. However, this LAR employs the methodology currently endorsed by NEI 94-01, Revision 3-A, for the required confirmatory risk impact assessment since this is the most up to date guidance available.

3.2.1 Current DCPP Type A Requirements

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." On March 1, 1996, the NRC approved License Amendments 109 and 110 for DCPP (Reference 27) authorizing the implementation of Option B for Type A tests. Currently, TS 5.5.16 requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program is required to meet the guidelines contained in RG 1.163. RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0, as an acceptable method for complying with the provisions of Appendix J, Option B.

RG 1.163, Section C.1 states that licensees intending to comply with 10 CFR 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01, Revision 0, rather than using test intervals specified in ANSI/ANS 56.8-1994 (Reference 2). NEI 94-01, Revision 0, Section 11.0 refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a (where L_a is the maximum allowable leakage rate at design pressure). The elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Adoption of an Appendix J, Option B performance based containment leakage rate testing program altered the frequency of measuring containment leakage in the Type A, B, and C tests but did not alter the testing methodology by which Appendix J leakage testing is performed. The test frequency is based on an evaluation of the AF leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The allowed frequency for Type A testing as documented in NEI 94-01, Revision 0, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak tightness for differing types of containment designs, including a concrete containment similar to the DCPP containment structure. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests from the original three tests per 10 years to one test per 20 years was found to lead to an imperceptible increase in risk because Type A tests identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to overall containment leakage rate and the small number of leakage paths detected solely by Type A testing, NUREG-1493 concluded that increasing

the interval between performances of Type A tests is possible with a minimal impact on public risk.

3.2.2 DCPP 10 CFR 50, Appendix J, Option B Licensing History

March 1, 1996 - License Amendments No. 109 and 110 (Reference 27)

This amendment revised the DCPP Unit 1 (Amendment 110) and Unit 2 (Amendment 109) TS to allow use of 10 CFR 50, Appendix J, Option B, for Type A, B, and C containment leak rate testing implementing the performance based leakage rate testing program as permitted by 10 CFR 50, Appendix J.

April 22, 2002 – License Amendments No. 150 and 150 (Reference 21)

This amendment revised the DCPP Unit 1 and Unit 2 TS 5.5.16.a.3, "Containment Leakage Rate Testing Program," to allow a one-time, 5-year extension to the 10-year interval for performing the next Type A test at DCPP. The change allowed Type A testing within 15 years from the last Type A test, which was performed in May, 1994, at Unit 1 and April, 1993, at Unit 2.

June 26, 2007 – License Amendments No. 197 and 198 (Reference 18)

This amendment revised the DCPP Unit 1 (Amendment 197) and Unit 2 (Amendment 198) TS 5.5.16, "Containment Leakage Rate Testing Program," to comply with the requirements of 10 CFR 50.55a(g)(4) for components classified as ASME Section III, Code Class CC consistent with TSTF-343, "Containment Structural Integrity," (Reference 46). The revision allows the performance of visual examinations of the containment pursuant to ASME Section XI, Subsections IWE and IWL, (Reference 7) in lieu of the visual examinations performed pursuant to RG 1.163.

January 15, 2009 – License Amendments No. 203 and 204 (Reference 19)

This amendment revised the DCPP Unit 1 (Amendment 203) and Unit 2 (Amendment 204) TS 5.5.16, "Containment Leakage Rate Testing Program," to specify a lower peak calculated containment internal pressure following a large-break LOCA and the containment design pressure at DCPP. This revision specifies a new lower peak calculated containment internal pressure following steam generator replacement, which contains adequate margin to the UFSAR containment internal pressure analysis values.

3.2.3 Continued Acceptability of TS Amendments 197 (Unit 1) and 198 (Unit 2)

By application dated December 29, 2006 (Reference 11), PG&E requested changes to the TSs for DCPP, Units 1 and 2.

The proposed amendments revised TS 5.5.16, "Containment Leakage Rate Testing Program," to comply with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC consistent with TSTF-343, "Containment Structural Integrity." This regulation requires licensees to update their containment inservice inspection requirements in accordance with subsections IWE and IWL of ASME 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).

Amendments 197 and 198 revised TS 5.5.16.a by adding 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.

NEI 94-01, Revision 2-A, Section 1.1 states (in part), "generally, a FSAR describes plant testing requirements, including containment testing. In some cases, FSAR testing requirements differ from those of Appendix J. In many cases, Technical Specifications were approved that incorporated exemptions to provisions of Appendix J. Additionally, some licensees have requested and received exemptions after their Technical Specifications were issued. The alternate performance-based testing requirements contained in Option B of Appendix J will not invalidate such exemptions. However, any exemptions to the provisions of 10 CFR 50, Appendix J to be maintained in force as part of the Containment Leakage Testing Program should be clearly identified as part of the plant's program documentation."

By letter dated June 25, 2008, the NRC issued the final SER for NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," (Reference 14). The SER states, "if the exemptions were issued after the Technical Specifications were approved, when the licensee amends the TS requirements to the new test interval (for Type A, Type B or Type C tests), it should explicitly describe which exemptions the licensee wants to continue with and which exemptions it will not use during the implementation of the new test intervals. This information should be part of the TS amendment request. The

NRC staff requests that this section be clarified to state that this approach is acceptable provided the NRC has a chance to review the licensee's choice, as part of the TS amendment."

Conclusion:

DCPP will continue to implement the provisions of TS 5.5.16.a.1 and TS 5.5.16.a.2.

3.2.4 Type A Test History

As noted previously, TS 5.5.16 currently requires Type A, B, and C testing to be performed in accordance with RG 1.163, which endorses the methodology for complying with Option B. Since the adoption of Option B, the performance leakage rates are calculated in accordance with NEI 94-01, Revision 0, Section 9.1.1 for Type A testing. Tables 3.2.5-1 and 3.2.5-2 list the past periodic Type A test results for DCPP Units 1 and 2.

Table 3.2.5-1, DCPP Unit 1 Type A Test History

Test Date	Leakage Rate (Percent Weight per Day)
December, 1975 (Pre-Operational)	0.0466
November 1978	0.0219
February, 1982(1)	0.0156
April, 1985	0.0530
May, 1988	0.0230
April, 1994	0.0429
March, 2009 (SGRP)(2)	0.0355

Notes: (1) The February 1982 test was performed at reduced pressure
(2) The 2009 Type A testing was performed immediately following the SGRP. The SGRP did not modify or cut the containment structure.

Table 3.2.5-2, DCPP Unit 2 Type A Type A Test History

Test Date	Leakage Rate (Percent Weight per Day)
November, 1977 (Pre-Operational)	0.0165
August, 1984	0.0430
April, 1990	0.0340
November, 1993	0.0251
April, 2008 (SGRP)(1)	0.0146

Note: (1) The 2008 Type A test was performed immediately following the SGRP. The SGRP did not modify or cut the containment structure.

The results of the last two Type A tests performed in April 1994 and March 2009 (Unit 1), and November 1993 and April 2008 (Unit 2), are less than half of the maximum allowable containment leakage rate of 0.1 weight-percent per day. As a result, DCPP remains on an extended Type A test frequency since both Type A tests met the acceptance criteria defined by NEI 94-01, Revision 0. The current Type A test interval frequency for both Units 1 and 2 is 10 years. This LAR is proposing to extend the Type A test interval frequency for both Units 1 and 2 from one test every 10 years to one test every 15 years.

3.3 Plant Specific Confirmatory Analysis

3.3.1 Methodology

An evaluation has been performed to provide a risk assessment of permanently extending the Type A test interval from ten to fifteen years. The risk assessment follows the guidelines from the following:

- NEI 94-01 Revision 3-A;
- the methodology used in EPRI TR-104285;
- the NEI "Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," November 2001 (Reference 17);
- NRC regulatory guidance on the use of PRA as stated in RG 1.200, Revision 2, as applied to Type A tests;
- risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174, Revision 2;
- the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 20); and
- The methodology used in EPRI 1018243, dated October 2008 (Reference 15), inclusive of EPRI Report No. 1009325, Revision 2-A.

NEI 94-01, Revision 2-A, contains an SER that supports using EPRI Report No. 1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," for performing risk impact assessments in support of Type A test extensions. The guidance provided in Appendix H of EPRI Report No. 1009325, Revision 2-A, builds on the EPRI Risk Assessment methodology found in EPRI TR-104285. This methodology is employed to determine the appropriate risk information for use in evaluating the impact of the proposed Type A test frequency changes.

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant, that containment isolation

failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, it is desirable to show that extending the Type A test interval will not lead to a substantial increase in risk from containment isolation failures for DCPP.

In the SER issued by the NRC (Reference 24), the NRC concluded that the methodology in EPRI Report No. 1009325, Revision 2, was acceptable for referencing by licensees proposing to amend their TS to extend the Type A test interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SER. Table 3.3.1-1 addresses each of the four limitations and conditions for the use of EPRI Report No. 1009325, Revision 2.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions

Limitation/Condition (From Section 4.2 of SER)	DCPP Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the Type A test interval extension.	DCPP PRA technical adequacy is addressed in Section 3.3.2 of this LAR. The PRA is either consistent with the requirements of RG 1.200 or where gaps exist, the gaps have been addressed, as is relevant to this Type A test interval extension, as detailed in Attachment 3.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions		
	Limitation/Condition (From Section 4.2 of SER)	**DCPP Response**
2.a	The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the Type A test interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SER.	RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of CDF greater than 1.0E-6/year and less than 1.0E-5/year and increases in LERF greater than 1.0E-7/year and less than 1.0E-6/year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 3 in 10 years to 1 in 15 years is estimated as 1.53E-7/year for Unit 1 and 1.35E-7/year for Unit 2 using the EPRI Report No. 1018243 guidance (this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included), and baseline LERF is 2.26E-6/year for Unit 1 and 2.18E-6/year for Unit 2. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174. When external event risk is included, the increase in LERF resulting from a change in the Type A test interval from 3 in 10 years to 1 in 15 years is estimated as 7.87E-7/year for Unit 1 and 8.09E-7/year for Unit 2 using the EPRI Report No. 1018243 guidance, and baseline LERF is 8.78E-6/year for Unit 1 and 8.45E-6/year for Unit 2. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174.

Table 3.3.1-1, EPRI Report No. 1009325 Revision 2 Limitations and Conditions

	Limitation/Condition (From Section 4.2 of SER)	DCPP Response
2.b	Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive.	The effect resulting from changing the Type A test frequency to 1 per 15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.076 person-rem/year for Unit 1 and 0.067 person-rem/year for Unit 2. The results of this calculation meet these criteria. Moreover, the risk impact for the Type A test interval extension is small per RG 1.174.
2.c	In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in a previous one-time 15-year Type A test interval extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage point.	The increase in the CCFP from the 3 in 10-year Type A test interval to 1 in 15-year interval is 0.81 percent for Unit 1 and 0.80 percent for Unit 2. Therefore, this increase is judged to be small.
3.	The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.	The representative containment leakage for Class 3b sequences is 100 L_a based on the guidance provided in EPRI 1018243, as discussed in Attachment 3, Section 4.0 of this submittal.
4.	A LAR is required in instances where containment over-pressure is relied upon for ECCS performance.	Containment overpressure is not relied upon for ECCS Performance and is further discussed in Section 3.1.6 of this enclosure.

3.3.2 Technical Adequacy of the DCPP PRA

Internal Events PRA Quality Statement for Permanent 15-Year Type A Test Interval Extension

PG&E conducted an Internal Events Peer Review in December 2012. The full-scope Peer Review that included internal events and internal floods portions of the DCPP PRA was performed in accordance with RG 1.200, Revision 2, and ASME/ANS RA-Sa-2009 (Reference 8). All findings have been either resolved by additional analysis or evaluated in terms of their impact on the Type A test interval extension, and dispositioned as presented in Enclosure Attachment 3, Table A-1 Internal Events PRA Peer Review – Facts and Observations.

No changes have been made to the Internal Events or Internal Floods PRA models since the Peer Review, that would constitute an upgrade.

Internal floods findings and their disposition are presented in Attachment 3, Table A-2. Findings associated with internal floods are addressed but have no impact on the Type A test interval extension application.

FPRA Quality Statement for Permanent 15-Year Type A Test Interval Extension

The DCPP FPRA is adequate to support the Type A test interval extension analysis. The DCPP FPRA was reviewed in January 2008, as part of the pilot application of the NEI 07-12 (Reference 33) Peer Review process. The 2008 Peer Review was conducted against the requirements of the ANS Standard "FPRA Methodology," ANSI/ANS-58.23-2007 (Reference 4). At the time of this first Peer Review, certain technical elements of the FPRA had not been completed, and it was agreed that the second phase of the Peer Review would be performed when all the technical elements of the FPRA were completed. The second phase of the Peer Review was completed in December 2010. The 2010 Peer Review was conducted against the requirements of Section 4 of the ASME/ANS RA-Sa-2009.

The F&Os noted by the Peer Review and their dispositions are provided in Attachment 3 of this submittal, Table A-3.

All FPRA related F&Os, except SF-A5-01 against SR SF-A5, have been addressed and dispositioned as closed. SF-A5-01 tracks the implementation of a recommendation related to fire brigade training requirement dealing with seismically induced fires. This item has no impact on the Type A test interval extension analysis.

Per the 2010 Peer Review, the DCPP FPRA met CAT II or better in all SRs except two, SRs CF-A1 and FSS-D7. These two SRs are listed in Enclosure Attachment 3, Table A-3. These SRs have since been addressed and are now

considered as met at CAT-II. Table A-3 also lists the SRs from the 2008 Peer Review that did not meet CAT-II, or better, quality requirements. However, as documented in Enclosure Attachment 3, Tables V-1 and V-2, these 2008 SRs have been re-reviewed during the 2010 Peer Review and all of the SRs were found met at CAT-II or better. No changes have been made to the FPRA model since the Peer Reviews that would constitute an upgrade. Based on the Peer Reviews, independent third-party reviews, and the resolution of F&Os, the DCPP FPRA model includes no deviations from NUREG/CR-6850 (Reference 40) approaches, and contains no UAMs.

DCPP has received License Amendments No. 225 and 227, dated April 14, 2016, (Reference 28), for implementation of NFPA 805 on Units 1 and 2, respectively. DCPP has implemented required changes to the Operating Licenses and TSs and is in the process of implementing the program and installing modifications as committed to in the LAR, SE, and License Amendments. Thus, it is anticipated that all of the FPRA related modifications will be completed prior to the next scheduled Type A tests for Units 1 and 2 in the first quarter of 2019 and 2018, respectively. Therefore, the NFPA 805 post modification FPRA model is deemed applicable and was used as described in Enclosure Attachment 3 for determining the PRA impacts of the Type A test frequency extension.

<u>SPRA Quality Statement for Permanent 15-Year Type A Test Extension</u>

PG&E conducted a SPRA Peer Review in January 2013. The full-scope Peer Review also included a review of seismic hazard and fragility analyses and was performed in accordance with RG 1.200, Revision 2, and ASME/ANS RA-Sa-2009. The seismic hazard update incorporates the most recent site-specific seismic data. The Peer Review team reviewed the methodologies used in the hazard and fragility of existing analyses and found them to be acceptable. The current SPRA model provides a reasonable estimate of the seismic CDF and LERF for the purposes of the Type A test interval extension analysis.

Section 5 of the ASME/ANS RA-Sa-2009 contains a total of 77 SRs under three technical elements. As a result of this review, a total of 60 F&Os were generated. These included 5 “Best Practices,” 18 “Suggestions,” and 37 “Findings.” Enclosure Attachment 3, Table A-4 presents the SPRA Peer Review F&Os and their effect on the Type A test interval extension analysis.

The SPRA is deemed applicable and was used as described in Enclosure Attachment 3 for determining the PRA impacts of the Type A test frequency extension.

3.3.3 Summary of Plant Specific Risk Assessment Results

The findings of the DCPP Risk Assessment contained in Attachment 3 confirm the general findings of previous studies that the risk impact associated with extending the Type A test interval from 3 in 10 years to 1 in 15 years is small per RG 1.174.

The conclusions that are based on the results presented in Attachment 3, Section 5.2, and the sensitivity calculations presented in Attachment 3, Section 5.3, are summarized below:

- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines small changes in risk as resulting in increases of CDF less than 1.0E-5/year and increases in LERF less than 1.0E-6/year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 3 in 10 years to 1 in 15 years is estimated as 1.53E-7/year for Unit 1 and 1.35E-7/year for Unit 2 using the EPRI Report No. 1018243 guidance (this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included), and baseline LERF is 2.26E-6/year for Unit 1 and 2.18E-6/year for Unit 2. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174. When external event risk is included, the increase in LERF resulting from a change in the Type A test interval from 3 in 10 years to 1 in 15 years is estimated as 7.87E-7/year for Unit 1 and 8.09E-7/year for Unit 2 using the EPRI Report No. 1018243 guidance, and baseline LERF is 8.78E-6/year for Unit 1 and 8.45E-6/year for Unit 2. As such, the estimated change in LERF is determined to be small using the acceptance guidelines of RG 1.174.

- The effect resulting from changing the Type A test frequency to 1 in 15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.076 person-rem/year for Unit 1 and 0.067 person-rem/year for Unit 2. EPRI Report No. 1009325 states that a very small population dose is defined as an increase of less than or equal to 1.0 person-rem per year, or less than or equal to 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended Type A test intervals. The results of this calculation meet these criteria.

- The increase in the CCFP from the 3 in 10-year interval to 1 in 15-year interval is 0.808 percent for Unit 1 and 0.799 percent for Unit 2. EPRI Report No. 1009325, states that increases in CCFP of less than or equal to 1.5 percent is small.

Therefore, increasing the Type A testing interval to 15 years is considered to be not significant since it represents a small change, per RG 1.174, to the DCPP risk profile.

3.3.4 Previous Assessments

The NRC, in NUREG-1493, has previously concluded that:

- Reducing the frequency of Type A tests from 3 per 10 years to 1 per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because Type A tests identify only a few potential containment leakage paths that cannot be identified by Type B and Type C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between Type A tests is possible with minimal impact on public risk. The impact of relaxing the Type A test frequency beyond 1 in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, Type A tests also test the integrity of the containment structure.

The findings for DCPP confirm these general findings on a plant-specific basis considering the severe accidents evaluated for DCPP, the DCPP containment failure modes, and the local population surrounding DCPP.

3.4 Non-Risk Based Assessment

Consistent with the defense-in-depth philosophy discussed in RG 1.174, DCPP has assessed other non-risk based considerations relevant to the proposed amendment. DCPP has multiple inspections and testing programs that are designed to identify any degrading conditions that might affect the containment structure. These inspections and testing programs also ensure that the containment structure remains capable of meeting its design functions. These programs are discussed below.

3.4.1 Coating Quality Monitoring Program

The Coating Quality Monitoring Program is designed to provide added assurance of continued acceptable performance of coatings inside the containments.

In September 1984, following the Unit 2 Type A test, PG&E undertook a comprehensive review of the paint systems in containment. As a result of the investigation and analysis associated with this review, PG&E implemented a

formal Coating Quality Monitoring Program to provide added assurance of continued acceptable paint system performance to address all containment coatings (previous NRC commitment contained in References 9 and 10). The program requires a walkdown of containments during each scheduled refueling outages (RFOs) to verify the general paint condition by thorough visual inspection. This walkdown is to be conducted by individuals qualified in coatings and coating systems. In addition, paint adhesion testing was conducted; special attention was given to the Unit 2 areas where blistering was previously discovered to monitor the general paint system condition, the adhesion of the primer, and its hardness. This special inspection was conducted until adequate data was collected to ensure that unqualified coatings inside containment were in good condition and did not show degradation. The results of this special inspection of Unit 2 after three RFOs provided sufficient information to conclude that these coatings performed adequately. Therefore, based on the good performance of the coatings, the special inspection portion of the Coating Quality Monitoring Program was discontinued after completion of the third RFO for Unit 2.

However, in accordance with the Coating Quality Monitoring Program, PG&E continues the general inspections of the containment coatings scheduled each RFO, which include inspection of all quality and non-quality coatings. The general inspection is part of surveillance activities to assure continued acceptable performance of quality-related coatings inside containment.

Specifically, the general inspection provides a survey of the condition of coatings inside containment. Deficient coatings identified are recorded describing locations, types, quantities, and modes of failure. These are repaired within a reasonable time, and are evaluated to assure that there are no safety concerns. If significant failures are observed during the general inspection, the special monitoring and testing can be reinstated and expanded, as deemed necessary by the coatings inspector. An overall report is issued after each RFO to document the condition and assessment of the coatings.

<u>Results of Recent Coatings Inspections</u>

Spring 2014, 1R18

The 1R18 coatings inspection concluded that the majority of the coatings inside the Unit 1 containment are in good condition. Containment liner plates were inspected visually from the 91-ft, 117-ft, and 140-ft elevations. Isolated spots of coating defects, all in sizes less than 1/2 square ft, were identified and repaired in accordance with the Containment Field Coatings Procedure during 1R18. General walkdown and specific visual inspections of coated structures, systems, and components inside the containment were conducted by qualified coatings inspectors. All coated surfaces of steel and concrete were examined from accessible elevations at 91-ft, 117-ft, and 140-ft for visible defects such as

blistering, cracking, rusting, peeling, or delamination. Any identified visual defect in the coating was documented for each coated item. Notifications were initiated for further evaluation of these defective areas. Any defective coating with potential to fail and generate debris was either removed or reported as unqualified and added to the Unqualified Coatings Log. Coating repair was recommended where necessary and prioritized.

The total area of deficient coatings identified during 1R18 that could not be repaired or determined to be Category 2 coatings with potential to delaminate and form debris is 137-1/2 square ft. The Unit 1 coating margin for unqualified coatings after 1R18 is 3677 square ft.

All delaminating coatings with potential to become a source of debris have either been removed or included in the Unqualified Coatings Log which tracks the margin of coating debris allowable for safe operation of the ECCS. The balance of coated items pending repair following 1R18 pose no safety or operability concern.

Fall 2014, 2R18

The 2R18 coatings inspection concluded that the majority of the coatings inside the Unit 2 containment are in good condition. Containment liner plates were visually inspected from 91-ft, 117-ft, and 140-ft elevations. Isolated spots of coating defects, all in sizes less than 1/10 square ft were identified. All defects that were accessible within the outage scope were repaired in accordance with the Containment Field Coatings Procedure before the end of 2R18.

General walkdown and specific visual inspections of coated structures, systems, and components inside the containment were conducted by qualified coatings inspectors. All coated surfaces of steel and concrete were examined from accessible elevations at 91-ft, 117-ft, and 140-ft for visible defects such as blistering, cracking, rusting, peeling, or delamination. Any identified visual defect in the coating was documented for each coated item. Notifications were initiated for further evaluation of these defective areas. Any defective coating with potential to fail and generate debris was either removed or reported as unqualified and added to the Unqualified Coatings Log. Coating repair was recommended where necessary and prioritized.

The total area of deficient coatings identified during 2R18 that could not be repaired or determined to be Category 2 coatings with potential to delaminate and form debris is 98 square ft. The Unit 2 coating margin for unqualified coatings after 2R18 is 2,143 square ft.

All delaminating coatings with potential to become a source of debris have either been removed or included in the Unqualified Coatings Log which tracks the margin of coating debris allowable for safe operation of the ECCS. The balance

of coated items pending repair following 2R18 pose no safety or operability concern.

3.4.2 Containment Inservice Inspection Plan

The incorporation into 10 CFR 50.55a of ASME Section XI Subsections IWL and IWE, 1992 Edition with 1992 Addenda, for the inservice inspection of containment concrete and metal liner, became effective September 9, 1996. The September 9, 1996, effective date established the beginning of the DCPP Containment ISI Program. The guidance contained in NRC IN 97-29, "Containment Inspection Rule," requires that all repair and replacement activities are in accordance with IWE and IWL requirements effective September 9, 1996, with the completion of first inspection period exams no later than September 9, 2001. As required, expedited containment concrete and liner exams were completed successfully by September 9, 2001.

The Containment ISI Program Plan for DCPP, Units 1 and 2, includes the containment concrete shell (ASME Section III, Reference 6, Code Class CC) and containment metallic liner (ASME Section III, Code Class MC). The Containment ISI Program Plan supplements the DCPP Units 1 and 2 ISI Program Plan for Class 1, 2, and 3 pressure retaining components and their supports, and together these two volumes comprise the ISI Program Plan.

This second interval Containment ISI Program Plan implements ASME Code Section XI, Subsections IWE and IWL, 2001 Edition with 2003 Addenda, within the limits and modifications of 10 CFR 50.55a. IWE exams of the metallic containment liner are performed on 40-month periods within the 10-year interval starting May 9, 2008. Concrete shell exams occur on a 5-year frequency as specified by IWL-2410(a), starting November 2000, and August 2001, for Unit 1 and Unit 2, respectively.

Examiners performing General Visual and Detailed Visual examinations of concrete using ASME Section XI guidance are qualified and certified to Level II in the VT-3C method by examination every 3 years. Examiners performing General Visual and VT-3 examinations of the containment liner are qualified and certified to Level II in the VT-3 method by examination every 3 years. Examiners performing VT-1 examinations are qualified and certified to Level II by examination every 3 years. Alternatively, Level III examiners in the visual method are qualified and certified by examination every 5 years and may perform all referenced examinations.
Repairs and replacement to the containment shell are conducted in accordance with IWL-4000. Repairs and replacement of the metallic liner are in accordance with IWE-3124. The exemptions are as follows:

1. The proper performance of concrete shell and metallic liner examinations in accordance with the requirements and frequency of Section XI,

Subsections IWE and IWL are required to satisfy DCPP TS 5.5.16.a.1 and TS 5.5.16.a.2. These examinations supersede those required by 10 CFR 50, Appendix J as required by the exemptions in TS 5.5.16.a.

2. 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 and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

Due to misapplication of IWL-2421, the second examination of the Unit 1 containment concrete shell in the first interval that was due by November 2005, (5 years after the original examination in 2000) was not performed. This issue was entered into the plant corrective action program for resolution (Reference 13). The subsequent Unit 1 examination was performed and completed successfully in October 2010.

Remaining first interval examinations of the Unit 2 containment concrete shell were completed successfully by August 2006, as required.

ASME Code Exemptions

Concrete areas inaccessible due to adjacent structures or components, or due to coverage of foundation material, backfill, or embedment are not required to be examined per IWL-1220(b). Embedded or inaccessible areas of the containment liner are not required to be examined per IWE-1220.

Code Cases

Use of Code Cases which may apply to the Containment ISI Program are governed by relief requests subject to NRC review and approval until such time as they are incorporated into RG 1.147, or otherwise approved for use by the NRC.

Pressure Testing Requirements

Pressure tests of containment, which includes Types A, B, and C tests, are scheduled in accordance with NEI 94-01, Revision 0, and RG 1.163, per DCPP Administrative Procedure AD13.DC5, "Containment Leakage Rate Testing Program."

<u>10 CFR 50.55a(b)(2)(i) Limitations and Modifications</u>

General Limitations and Modifications

1. In accordance with 10 CFR 50.55a(b)(2), references to Section XI of the ASME Boiler and Pressure Vessel Code, and includes the 1977 Edition (Division 1) through the 2003 Addenda (Division 1).

2. In accordance with 10 CFR 50.55a(b)(2)(vi), successive 120-month interval updates must be implemented in accordance with paragraph (g)(4)(ii).

Examination of Concrete Containments

1. In accordance with 10 CFR 50.55a(b)(2)(viii), licensees applying Subsection IWL, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section shall apply paragraphs (b)(2)(viii)(E) through (b)(2)(viii)(G).

2. In accordance with 10 CFR 50.55a(b)(2)(viii)(E), licensees shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary report required by IWA-6000:

   - a description of the type and estimated extent of degradation, and the conditions that led to the degradation;

   - an evaluation of each area and the result of the evaluation; and

   - a description of the necessary corrective action.

3. In accordance with 10 CFR 50.55a(b)(2)(viii)(F), personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA-2300. The "owner-defined" personnel qualification provisions in IWL-2310(d) are not approved for use.

4. In accordance with 10 CFR 50.55a(b)(2)(viii)(G), corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA-1400. This criterion does not apply to DCPP as the containments are not post-tensioned.

Examination of Metallic Liners

1. In accordance with 10 CFR 50.55a(b)(2)(ix), licensees applying Subsection IWE, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I).

2. In accordance with 10 CFR 50.55a(b)(2)(ix)(A), licensees shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report required by IWA-6000:

   - a description of the type and estimated extent of degradation, and the conditions that led to the degradation;

   - an evaluation of each area and the result of the evaluation; and

   - a description of necessary corrective action.

3. In accordance with 10 CFR 50.55a(b)(2)(ix)(B), when performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in IWA-2210-1 may be decreased, provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

4. In accordance with 10 CFR 50.55a(b)(2)(ix)(F), VT-1 and VT-3 examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 examination method shall be qualified in accordance with IWA-2300. The owner-defined personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 examinations are not approved for use.

5. In accordance with 10 CFR 50.55a(b)(2)(ix)(G), the VT-3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of ASME Section XI Table IWE-2500-1.

   The VT-1 examination method must be used to conduct the examination of Item E4.11 of ASME Section XI Table IWE-2500-1.

Examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval.

The owner-defined visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 examinations.

6. In accordance with 10 CFR 50.55a(b)(2)(ix)(H), containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of ASME Section XI Table IWE-2500-1 must be examined using the VT-3 examination method.

   Flaws or degradation identified during the performance of VT-3 examination must be examined in accordance with the VT-1 examination method.

   The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation.

   As an alternative to performing a VT-3 examination of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

7. In accordance with 10 CFR 50.55a(b)(2)(ix)(H), the ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

Table 3.4.2-1 DCPP Unit 1 IWE Containment Examination Schedule	
ISI 2nd Interval Period 1	5/9/2008 – 9/8/2011
	Planned Outages 1R15 - 2/25/2009 1R16 - 10/2/2010
ISI 2nd Interval Period 2	9/9/2011 – 1/8/2015
	Planned Outages 1R17 - 4/22/2012 1R18 - 2/2/2014
ISI 2nd Interval Period 3	1/9/2015 – 5/8/2018
	Planned Outages 1R19 - 09/27/2015 1R20 - 04/30/2017
ISI 3rd Interval Period 1	5/9/2018 – 9/8/2021
	Planned Outages 1R21 1R22
ISI 3rd Interval Period 2	9/9/2021 – 1/8/2025
	Planned Outages 1R23 1R24
ISI 3rd Interval Period 3	1/9/2025 – 4/9/2028
	Planned Outages 1R25 1R26

Table 3.4.2-2 DCPP Unit 2 IWE Containment Examination Schedule	
ISI 2nd Interval Period 1	5/9/2008 – 9/8/2011
	Planned Outages 2R15 - 10/3/2009 2R16 - 5/1/2011
ISI 2nd Interval Period 2	9/9/2011 – 1/8/2015
	Planned Outages 2R17 - 2/3/2013 2R18 - 9/28/2014
ISI 2nd Interval Period 3	1/9/2015 – 5/8/2018
	Planned Outages 2R19 - 5/1/2016 2R20 - 2/4/2018
ISI 3rd Interval Period 1	5/9/2018 – 9/8/2021
	Planned Outages 2R21 2R22
ISI 3rd Interval Period 2	9/9/2021 – 1/8/2025
	Planned Outages 2R23 2R24
ISI 3rd Interval Period 3	1/9/2025 – 4/9/2028
	Planned Outages 2R25 2R26

Containment Surfaces Subject to Examination

3.4.2-3 Examination Category E-A, Containment Liner Surfaces				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-1 Containment Liner (Containment Vessel Pressure Retaining Boundary)				
E1.11	Accessible Surface Areas	General Visual	1	Each Inspection Period
E1.12	Wetted Surfaces of Submerged Areas	None*	*Note: DCPP sump design precludes wetted/submerged surfaces of the metallic liner. E1.12 does not apply.	
E1.30	Moisture Barriers	None*	*Note: DCPP liner design does not use moisture barriers. E1.30 does not apply.	

3.4.2-4 Examination Category E-A, Containment Liner Surfaces				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-2 Containment Liner (Containment Vessel Pressure Retaining Boundary)				
E1.11	Accessible Surface Areas	General Visual	1	Each Inspection Period
E1.12	Wetted Surfaces of Submerged Areas	None*	*Note: DCPP sump design precludes wetted/submerged surfaces of the metallic liner. E1.12 does not apply.	
E1.30	Moisture Barriers	None*	*Note: DCPP liner design does not use moisture barriers. E1.30 does not apply.	

3.4.2-5 Examination Category E-A, Containment Liner Surfaces				
Item No.	Item Description	Exam Method	Number of Components	Exam Frequency
Unit-1 Containment Liner Pressure Retaining Bolting				
E1.11	Bolting When Installed - Accessible Surface Areas	General Visual or VT-3 (Note 1)	All Bolting Installed During Exam	100 Percent Each Period Unless Disassembled (See Below)
E1.11	Bolting When Disassembled (Any Reason)	VT-3 / VT-1 (Note 1)	All Bolting Disassembled During Exam	100 Percent Each Inspection Period
E1.11	Penetration #58 (Mini Equipment)	VT-3 / VT-1 (Note 1)	12 Studs, 24 Nuts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #60 (Mini Equipment)	VT-3 / VT-1 (Note 1)	12 Studs, 24 Nuts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #63 (Vacuum Pressure Relief)	VT-3 / VT-1 (Note 1)	12 Bolts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #64 (Fuel Transfer Tube)	VT-3 / VT-1 (Note 1)	Bolts not normally removed due to QOTTC	100 Percent Each Period If Disassembled During Exam
E1.11	Equipment Hatch (Always Disassembled)	VT-3 / VT-1 (Note 1)	48 Eye Bolts and Nuts	100 Percent - Always Disassembled During Exam
Note 1: All bolting is examined as part of Item E1.11. 10 CFR 50.55a(b)(2)(ix)(G) requires all bolting to receive a VT-3 exam instead of the General Visual exam at least once during the interval if not disassembled. 10 CFR 50.55a(b)(2)(ix)(H) requires bolting disassembled at the time of the exam to always receive a VT-3 exam instead of the General Visual exam, and that flaws or degradation identified during the performance of the VT-3 exam must be examined using the VT-1 method.				

3.4.2-6 Examination Category E-A, Containment Liner Surfaces				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-2 Containment Liner Pressure Retaining Bolting				
E1.11	Bolting When Installed - Accessible Surface Areas	General Visual or VT-3 (Note 1)	All Bolting if Installed During Exam	100 Percent General Visual x 2, plus VT-3 100 Percent x 1 Each Period Unless Disassembled (See Below)
E1.11	Bolting When Disassembled (Any Reason)	VT-3 / VT-1 (Note 1)	All Bolting if Disassembled During Exam	Each Inspection Period
E1.11	Penetration #58 (Mini Equipment)	VT-3 / VT-1 (Note 1)	12 Studs, 24 Nuts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #60 (Mini Equipment)	VT-3 / VT-1 (Note 1)	12 Studs, 24 Nuts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #63 (Vacuum Pressure Relief)	VT-3 / VT-1 (Note 1)	12 Bolts	100 Percent Each Period When Disassembled During Exam
E1.11	Penetration #64 (Fuel Transfer Tube)	VT-3 / VT-1 (Note 1)	Bolts Not Normally Removed Due to QOTTC	100 Percent Each Period If Disassembled During Exam
E1.11	Equipment Hatch (Always Disassembled)	VT-3 (See E-C for VT-1 Augment)	48 Eye Bolts and Nuts	100 Percent Each Period (Always Disassembled During Exam)
Note-1: All bolting is examined as part of Item E1.11. 10 CFR 50.55a(b)(2)(ix)(G) requires all bolting to receive a VT-3 exam instead of the General Visual exam at least once during the interval if not disassembled. 10 CFR 50.55a(b)(2)(ix)(H) requires bolting disassembled at the time of the exam to always receive a VT-3 exam instead of the General Visual exam, and that flaws or degradation identified during the performance of the VT-3 exam must be examined using the VT-1 method.				

3.4.2-7 Examination Category E-C, Containment Liner Surfaces Requiring Augmented Examination				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-1 Containment Liner including Pressure Retaining Bolting (Containment Vessel Pressure Retaining Bolting)				
E4.11	Accessible Surface Areas	VT-1	Note-1	Note-1 and Note-2
E4.12	Surface Area Grid - Minimum Wall Thickness Location	Ultrasonic Thickness	Note-1	Note-1 and Note-2
Note-1: Containment surfaces requiring augmented examination are those identified in IWE-1240. Currently, no surface areas meeting the requirements of IWE-1240 are identified. The recirculation sump wall adjacent to the self-contained sump structure is no longer a thickness grid area. Note-2: The extent of examination shall be 100 percent for each inspection period until the areas examined remain essentially unchanged for the next inspection period. Such areas then no longer require augmented examination in accordance with IWE-2420(c).				

3.4.2-8 Examination Category E-C, Containment Liner Surfaces Requiring Augmented Examination				
Item No.	Item Description	Examination Method	Number of Components	Examination Requirement
Unit-2 Containment Liner including Pressure Retaining Bolting (Containment Vessel Pressure Retaining Bolting)				
E4.11	Accessible Surface Areas	VT-1	Note-1	Note-1 and Note-2
E4.12	Surface Area Grid - Minimum Wall Thickness Location	Ultrasonic Thickness	Note-1	Note-1 and Note-2
Note-1: Containment surfaces requiring augmented examination are those identified in IWE-1240. Currently, no surface areas meeting the requirements of IWE-1240 are identified. The recirculation sump wall adjacent to the self-contained sump structure is no longer a thickness grid area. Note-2: The extent of examination shall be 100 percent for each inspection period until the areas examined remain essentially unchanged for the next inspection period. Such areas then no longer require augmented examination in accordance with IWE-2420(c).				

3.4.2-9 Examination Category L-A, Containment Concrete Shell				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-1 Containment Concrete Shell				
L1.11	All Accessible Surface Areas	General Visual	1	100 Percent x 2 IWL-2510 Accessible Areas
L1.12	Suspect Areas	Detailed Visual	Any Areas Identified During General Visual	IWL-2510 Suspect Areas (Note-1)
Note-1: Any suspect areas identified during the General Visual exam must receive a Detailed Visual exam. The number of items and frequency of examination for the Detailed Visual exam depend on results of the General Visual exam. All examination results are reviewed and evaluated by the responsible registered professional civil engineer.				

3.4.2-10 Examination Category L-A, Containment Concrete Shell				
Item No.	Item Description	Examination Method	Number of Components	Examination Frequency
Unit-2 Containment Concrete Shell				
L1.11	All Accessible Surface Areas	General Visual	1	100 Percent x 2 IWL-2510 Accessible Areas
L1.12	Suspect Areas	Detailed Visual	Any Areas Identified During General Visual	IWL-2510 Suspect Areas (Note-1)
Note-1: Any suspect areas identified during the General Visual exam must receive a Detailed Visual exam. The number of items and frequency of examination for the Detailed Visual exam depend on results of the General Visual exam. All examination results are reviewed and evaluated by the responsible registered professional civil engineer.				

<u>Results of Recent IWE Inspections – Outage</u>

Spring 2012, 1R17,

Typical conditions of small bare metal spots, flaking, paint cracking, dents, surface oxidation, and gouges were found throughout the Unit 1 containment. Many of these conditions are legacy locations as indicated from previous examination reports. Bulges were observed to be present around the vertical section of the containment liner and appear to have originated during

construction. No evidence of paint distress or metal fatigue around any of these bulges was observed. Previous examination reports identified bulges in the same general areas. Surface oxidation was observed in a few locations. The identified bare metal spots were the results of dents and rub marks from equipment impacts. The indications identified were evaluated by the Engineering Department and determined to be acceptable.

Spring 2013, 2R17

Typical conditions of small bare metal spots, paint flaking, paint cracking, dents, surface oxidation, and gouges were found throughout the Unit 2 containment. Many of these conditions were identified in previous exams. Bulges were present around the vertical section of the liner and appear to be from the original construction. No evidence of paint distress or metal fatigue around any bulges was noted. Bare metal spots resulting from impacts and rubs from moving equipment impacts, as well as grinding and sanding marks from various stages of work were present throughout the containment liner. Surface oxidation exists in a few of the bare metal spots. Some grinding and sanding marks are under the original paint and most likely from the initial construction and others are from recent repairs. No cracks were found during the exam. The indications identified were evaluated by the Engineering Department and determined to be acceptable.

Results of Recent IWL Examinations

Unit 1 Examinations

A concrete examination was performed in June, July, and October 2010, to meet ISI requirements and evaluate the engineering properties of the concrete for the Unit 1 containment structure at DCPP. The next examination of the Unit 1 containment began in 2015 to satisfy the five year examination frequency requirement and is ongoing.

Examination of the containment concrete employs a three-tier acceptance process similar to that described in ACI 349.3R-96 (Reference 1). DCPP Procedure, NDE VT 3C-1, "VT-3C Visual Examination of the Containment Concrete Shell," incorporates the first tier criteria that permits direct acceptance by the examiner, and provides data recording requirements for implementation of the second tier (responsible engineer acceptance) and third tier (responsible engineer evaluation) in the event that first tier criteria are not met. Certain conditions were considered acceptable if their dimensions or observed effects were not severe and were within first-tier limits. The procedure permits direct acceptance by the examiner for first-tier criteria. For observations exceeding the initial first-tier limits, further evaluation by the responsible engineer was required to determine acceptability. Observations exceeding a second-tier set of

quantitative limits required further technical evaluation and analysis to validate the existing condition or repair to preserve structural function.

In the 2010 examination, a total of 990 reportable indications were recorded, of which 12 are greater than second-tier indications. The indications found were primarily leaching, passive cracks, and form tie repairs (Plugs). Of the 990 reportable indications, 241 were passive cracks. An additional 620 indications of leaching were recorded. Another 85 indications of deteriorated form tie repairs (Plugs) were recorded, all above the 140-ft elevation.

Reported concrete indications that are greater than the second tier limits and require further engineering evaluation are all limited to the concrete cover depth (minimum 2 in.) only and do not affect the structural reinforcing steel bars. Further engineering evaluation included additional investigations (rebar scan to measure actual concrete cover) for all indications in order to evaluate and accept as-is without repair. In all cases, the concrete cover was greater than the required minimum 2 in. and the reinforcing bars at these locations always had a net minimum 2 in. of undisturbed concrete cover.

No repair is required on any of the indications listed in the report. These indications are reported as part of the 2010 IWL inspection results for Unit 1 containment exterior concrete and they will continue to be monitored per the IWL inspections. The next IWL inspection of the Unit 1 containment began in 2015 to satisfy the five year examination frequency requirement and is ongoing. Any changes in indications will be evaluated during the ongoing Unit 1 IWL inspection

Based on the engineering evaluations, the subject indications do not affect the ability of the containment structure to perform its intended design function. The containment exterior structure (concrete shell) continues to remain capable of performing its design functions.

Unit 2 Examinations

A concrete examination was performed to meet ISI requirements and evaluate the engineering properties of the concrete for the Unit 2 containment structure at DCPP from May 2011, to June 2011. The examination of all areas was completed by August 2011.

Acceptance criteria for the examination employed a three-tiered hierarchy. Certain conditions were considered acceptable if their dimensions or observed effects were not severe and were within first-tier limits. The procedure permitted direct acceptance by the examiner for first-tier criteria. For observations exceeding the initial first-tier limits, further evaluation by the responsible engineer was needed to determine acceptability. Observations exceeding a second-tier set of quantitative limits required further technical evaluation and analysis to validate the existing condition or repair to preserve structural function.

A total of 2096 reportable indications were documented during the examination, of which 2076 are greater than first-tier and 20 are greater than second-tier indications. Of the reportable indications, 1034 were passive cracks, 89 percent of these indications were located on the dome.

Reported concrete indications that were greater than the second tier limits and required engineering evaluation were examined in detail. The engineering evaluation process included additional investigations (rebar scan to measure actual concrete cover) for all indications to evaluate and accept them as-is without repair. In all cases, the concrete cover was greater than the required minimum 2 in. and the reinforcing bars at these locations always had a net minimum 2 in. of undisturbed concrete cover. Since the additional examinations showed that the indications are all limited to the concrete cover, these indications do not affect the structural reinforcing steel bars and the structural integrity of the containment exterior is not adversely impacted.

No repair was required for any of the indications noted during the 2011 IWL inspection for the Unit 2 containment exterior concrete. The indications will continue to be monitored during future IWL inspections.

Based on the above evaluations, the subject indications do not affect the ability of the containment structure to perform its intended design function. The containment exterior structure (concrete shell) continues to remain capable of performing its design functions.

3.4.3 Supplemental Inspection Requirements

NEI 94-01, Revision 2-A requires that a general visual examination of accessible interior and exterior surfaces of the containment, for structural deterioration that may affect the containment leak-tight integrity, be conducted. This inspection must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years in accordance with the following sections of NEI 94-01, Revision 2-A:

- Section 9.2.1, “Pretest Inspection and Test Methodology”
- Section 9.2.3.2, “Supplemental Inspection Requirements”

However, DCPP License Amendments 197 and 198 were approved on June 26, 2007, to allow the performance of the visual examinations of the containment pursuant to ASME Code Section XI, Subsections IWE and IWL, in lieu of the visual examinations performed pursuant to RG 1.163. The containment visual examination is implemented by the “Containment Inservice Inspection Program

Plan (ASME XI, Subsections IWE and IWL)." This plan fulfills the surveillance requirements as all areas of the shell and liner which are accessible for direct or qualified remote examination are subject to these requirements. As a result, no supplemental inspections that are required by NEI 94-01, Revision 2-A, will be implemented.

3.4.4 Containment Leakage Rate Testing Program – Type B and Type C Testing Program

The reactor containment leakage test program includes performance of Type B and Type C tests, in accordance with 10 CFR 50, Appendix J, Option B and RG 1.163. Type B tests are intended to detect leakage paths and measure leakage for certain reactor containment penetrations such as airlocks, hatches, flanges and electrical penetrations. Type C tests are intended to measure containment isolation valve leakage rates. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below acceptable limits. In accordance with TS 5.5.16, the acceptance criteria for allowable unit startup maximum pathway total of Types B and C leakage is 0.6 L_a which is equal to 60 percent of L_a.

Tables 3.4.4-1 and 3.4.4-2 provide Type B and C data trend summaries for DCPP Units 1 and 2. All Type B and Type C test results for the AF and AL outage leakage summations from 2008 through 2015 met acceptance criteria.

Table 3.4.4-1 DCPP Unit 1 Type B and C Testing Combined AF/AL Trend Summary					
RFO	2009	2010	2012	2014	2015
	1R15	1R16	1R17	1R18	1R19
AF Minimum Path (lbm/day)	59.199	40.74	36.49	44.246	62.085
Fraction of L_a (percent)	8.17	5.37	4.81	5.83	8.18
AL Maximum Path (lbm/day)	75.875	63.67	63.04	72.591	78.22
Fraction of L_a (percent)	10.00	8.39	8.31	9.56	10.31
AL Minimum Path (lbm/day)	45.90	37.50	39.06	23.48	25.90
Fraction of L_a (percent)	6.05	4.94	5.15	3.09	3.41

Table 3.4.4-2 DCPP Unit 2 Types B and C Testing Combined AF/AL Trend Summary					
RFO	2008	2009	2011	2013	2014
	2R14	2R15	2R16	2R17	2R18
AF Min Path (lbm/day)	51.202	32.781	35.45	36.953	35.025
Fraction of La (percent)	6.99	4.32	4.67	4.87	4.61
AL Max Path (lbm/day)	71.322	58.615	77.074	82.56	71.289
Fraction of L_a (percent)	9.74	7.72	10.15	10.88	9.39
AL Min Path (lbm/day)	45.56	28.52	36.24	22.86	18.74
Fraction of L_a (percent)	6.22	3.76	4.77	3.01	2.47

As discussed in NUREG-1493, Type B and Type C tests can identify the vast majority of all potential containment leakage paths. Type B and Type C testing will continue to provide a high degree of assurance that containment integrity is maintained.

A review of the Type B and Type C test results of AF and AL outage leakage summations from 2008 through 2015 met acceptance criteria and the regulatory requirements as described below:

Unit 1:

- The AF minimum pathway leak rate average for DCPP Unit 1 shows an average of 6.47 percent of L_a with a high of 8.18 percent L_a, since 2009.
- The AL maximum pathway leak rate average for DCPP Unit 1 shows an average of 9.31 percent of L_a with a high of 10.31 percent L_a, since 2009.

Unit 2:

- The AF minimum pathway leak rate average for DCPP Unit 2 shows an average of 5.09 percent of L_a with a high of 6.99 percent L_a, since 2008.
- The AL maximum pathway leak rate average for DCPP Unit 2 shows an average of 9.58 percent of L_a with a high of 10.88 percent L_a, since 2008.

<u>Statistics on Number of Components on Extended Intervals</u>

Number of Components in Type B and Type C Test Program:

157 total components per Unit are tested, which includes the personnel airlock; personnel airlock seals (2); emergency airlock; emergency airlock seals (2); equipment hatch seals (1); mini equipment hatch (Pen 58) seals (1); mini equipment hatch (Pen 60) seals (1); fuel transfer tube seals (1); electrical penetrations (42); and containment isolation valves (105).

29 components per Unit will be tested at nominal surveillance intervals, since these components are not eligible for extended intervals due to TS limitations and refueling outage use.

1 component on Unit 1 and 5 components on DCPP Unit 2 will be tested at nominal surveillance intervals, since these components are not eligible for extended intervals due to performance history.

3.4.5 Type B and Type C Test Program Implementation Review

Tables 3.4.5-1 and 3.4.5-2 identify the components that were on extended Type B and C test intervals. The components have not demonstrated acceptable performance during the previous two outages for DCPP Units 1 and 2 and the current outage 2R19:

Table 3.4.5-1, DCPP Unit 1 Type B and C Test Program Implementation Review						
1R18 - 2014						
Component	AF SCCM	Admin Limit SCCM	AL SCCM	Cause of Failure	Corrective Action	Scheduled Interval
Electrical Penetration 9E	119	100	427	Penetration Canister Leakage	Under Evaluation(1)	30 Months
1R19 - 2015						
Component	AF SCCM	Admin Limit SCCM	AL SCCM	Cause of Failure	Corrective Action	Scheduled Interval
None						

Note: (1) Temporary repair with sealant was applied and penetration replacement is being evaluated.

Table 3.4.5-2, DCPP Unit 2 Type B and C Test Program Implementation Review						
2R17 - 2013						
Component	AF SCCM	Admin Limit SCCM	AL SCCM	Cause of Failure	Corrective Action	Scheduled Interval
NSS-2-9355B	1454	300	40	Foreign Material (Metal Shavings) Found Inside Valve	Skim Cut Stem Plug/Lapped Valve Seat	30 Months
2R18 - 2014						
Component	AF SCCM	Admin Limit SCCM	AL SCCM	Cause of Failure	Corrective Action	Scheduled Interval
CVCS-2-8109	6180	2000	33	Debris and Rust Found Between Disc and Body	Valve Cleaned, Foreign Material Removed	30 Months
2R19 – 2016 (in progress)[1]						
Component	AF SCCM	Admin Limit SCCM	AL SCCM	Cause of Failure	Corrective Action	Scheduled Interval
CS-2-9011B	11853	5100	Not completed at this time.	Under Evaluation	Under Evaluation	30 Months (currently 60 months)
NSS-2-9355A	554	300	Not completed at this time.	Under Evaluation	Under Evaluation	30 Months

Note (1) LAR is being submitted during 2R19.

3.5 Operating Experience

During the various examinations and tests conducted in support of the containment related programs previously mentioned, issues that did not meet established criteria or that provide indication of degradation, are placed into DCPP's corrective action program and corrective actions are planned and executed.

The following site specific and industry events have been evaluated for impact on containment:

- NRC IN 1989-79, "Degraded Coatings and Corrosion of Steel Containment Vessels" (Reference 34)
- NRC IN 1992-20, "Inadequate Local Leak Rate Testing" (Reference 35)

- NRC IN 1997-10, "Liner Plate Corrosion in Concrete Containments" (Reference 36)
- NRC IN 2004-09, "Corrosion of Steel Containment and Containment Liner" (Reference 37)
- NRC IN 2010-12, "Containment Liner Corrosion" (Reference 38)
- NRC IN 2014-07, "Degradation of Leak Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner" (Reference 39)

Each of these areas is discussed in detail in Sections 3.5.1 through 3.5.6, respectively.

3.5.1 NRC IN 1989-79, "Degraded Coatings and Corrosion of Steel Containment Vessels"

The NRC issued IN 1989-79, dated December 1, 1989, to alert addressees of the discovery of severely degraded coatings and the corrosion of steel ice condenser containment vessels that are caused by boric acid and collected condensation in the annular space between the steel shell and the surrounding concrete shield building.

> "On August 24, 1989, Duke Power Company reported significant coating damage and base metal corrosion on the outer surface of the steel shell of the McGuire Unit 2 containment which was discovered during a pre-integrated leak rate test inspection (as required by Appendix J to 10 CFR Part 50). Subsequently, Duke Power identified similar degradation of the McGuire Unit 1 containment, which is essentially identical to the Unit 2 structure.
>
> "Both units have ice condenser-type containments consisting of a freestanding steel shell surrounded by a concrete shield building. Between the shell and the shield building is a 6-foot-wide annular space. The steel shells have a nominal thickness near the annulus floor of 1 inch. The degraded area on the shells of both units is limited to 30-foot circumferential sections no higher than 1+ inches [sic] above the annulus floors. The average depth of the corrosion is 0.1 inch with pits of up to 0.125 inch. Corrosion that is up to 0.03-inch-deep was also found in areas below the level of the annulus floor on the Unit 2 shell, where concrete was removed to expose the shell surface. This below-floor corrosion is due to a lack of sealant at the interface between the shell and the annulus floor."

Discussion:

In July 2010, PG&E responded to a NRC RAI regarding the DCPP LRA (Reference 13). One of the requests asked DCPP to "Describe the potential effects of steel liner plate corrosion issues discussed in NRC INs 89-79, 97-10 and 2004-09 on the containment liners for DCPP Units 1 and 2," since their ASME Section XI IWE AMP did not discuss those specific INs.

In the response to the RAI, DCPP noted that INs 1989-79, 1997-10, and 2004-09 discuss containment liner corrosion events of differing severities that have occurred in boiling water reactor drywells and suppression pools, PWR ice condenser liners, and PWR reinforced concrete structural liners such as DCPP's. The DCPP ASME Section XI, Subsection IWE AMP containment liner inspection procedure, NDE VT 3-L, Units 1 and 2, "VT-3 Visual Examination of the Containment Liner," specifically addresses inspection of the containment liner for corrosion and degraded liner surfaces. DCPP specific examinations have routinely detected minor surface irregularities. Additional inspection methods have been performed to determine the extent and origin, if possible, of the irregularities. This level of detection demonstrates that conditions and surface indications of liner degradation have a high probability of being detected and addressed, thus ensuring the containment liner license renewal intended function is maintained. The periodic (40-month) inspection frequency has been specified by ASME Code as being sufficient to detect incipient indications of damage before it becomes widespread.

3.5.2 NRC IN 1992-20, "Inadequate Local Leak Rate Testing"

The issue discussed in IN 1992-20, "Inadequate Local Leak Rate Testing," was based on events at four different plants, Quad Cities, Dresden Nuclear Station, Perry Nuclear Plant, and the Clinton Station. The common issue in the four events was the failure to adequately perform Type B and C testing on different penetration configurations leading to problems that were discovered during Type A tests in the first three cases.
In the event at Quad Cities, the two-ply bellows design was not properly subjected to Type B test pressure and the conclusion of the utility was that the two-ply bellows design could not be Type B tested as configured.

In the events at both Dresden and Perry, flanges were not considered a leakage path when the Type C test was designed. This omission led to a leakage path that was not discovered until the plant performed a Type A test.

In the event at Clinton, relief valve discharge lines that were assumed to terminate below the suppression pool minimum drawdown level were discovered to terminate at a level above that datum. These lines needed to be reconfigured and the valves should have been Type C tested.

Discussion:

DCPP Units 1 and 2 have stainless steel bellows attached to their fuel transfer tubes; however, they do not provide a containment isolation function. The fuel transfer tubes at DCPP are welded directly to the containment liner via a split ring as a transitional piece. The expansion bellows are also welded to the fuel transfer tube on either side of the containment barrier in order to connect the fuel transfer tube to the reactor cavity, inside containment, and the fuel transfer canal, in the fuel handling building. There are no other steel expansion bellows that form part of the containment boundary at DCPP.

3.5.3 NRC IN 1997-10, "Liner Plate Corrosion in Concrete Containments"

The NRC issued IN 1997-10 on March 13, 1997, to alert addressees to occurrences of corrosion in the liner plates of reinforced and pre-stressed concrete containments, and to detrimental effects such as corrosion could have on containment reliability and availability under design-basis and beyond-design-basis events.

Inspections of several containment liners showed various degrees of corrosion. In January 1993, an NRC inspector pointed out corrosion of the drywell liner at Brunswick Unit 2. During NRC structural assessment reviews at Trojan and Beaver Valley Unit 1, the staff noted peeled coating and spots of liner corrosion. The structural assessment at Robinson Unit 2 also revealed discoloration of the vertical portion of the containment liner at an insulation joint. Finally, before the Type A test of the containment at Salem Unit 2 in 1993, the staff noted minor corrosion of the containment liner.

Of the five occurrences cited above, four (Trojan, Beaver Valley Unit 1, Salem Unit 2, and Robinson Unit 2) were found to be benign from the standpoint of safety. Corrosion of the liner plate at Brunswick Units 1 and 2 was significant from the standpoint of safety. Prior to the restart of Brunswick Units 1 and 2, the affected areas were cleaned and repaired.
Discussion:

In the response to the RAI (Reference 13), DCPP noted that INs 1989-79, 1997-10, and 2004-09 discuss containment liner corrosion events of differing severities that have occurred in boiling water reactor drywells and suppression pools, PWR ice condenser liners, and PWR reinforced concrete structural liners such as DCPP's. The DCPP ASME Section XI, Subsection IWE AMP containment liner inspection procedures, NDE VT 3-L, Units 1 and 2, "VT-3 Visual Examination of the Containment Liner," specifically addresses inspection of the containment liner for corrosion and degraded liner surfaces. DCPP specific examinations have routinely detected minor surface irregularities. Additional inspection methods have been performed to determine the extent and origin, if possible, of the irregularities. This level of detection demonstrates that conditions and/or surface

indications of liner degradation have a high probability of being detected and addressed, thus ensuring the containment liner license renewal intended function is maintained. The periodic (40-month) inspection frequency has been specified by ASME Code as being sufficient to detect incipient indications of damage before it becomes widespread.

3.5.4 NRC IN 2004-09, "Corrosion of Steel Containment and Containment Liner"

The NRC issued this IN to alert addressees to recent occurrences of corrosion in freestanding metallic containments and in liner plates of reinforced and pre-stressed concrete containments. Any corrosion (metal thinning) of the liner plate or freestanding metallic containment could change the failure threshold of the containment under a challenging environmental or accident condition. Thinning changes the geometry of the containment shell or liner plate and may reduce the design margin of safety against postulated accident and environmental loads. Recent experience has shown that the integrity of the moisture barrier seal at the floor-to-liner or floor-to-containment junction is important in avoiding conditions favorable to corrosion and thinning of the containment liner plate material. Inspections of containment at the floor level, as well as at higher elevations, have identified various degrees of corrosion and containment plate thinning.

Discussion:

Containment liner plate corrosion has been an industry issue for many years. DCPP is aware of these issues and has taken appropriate action to identify potential problems. Previous NRC INs have been evaluated, including: IN 1989-79, IN 1991-10, and IN 1997-29. Thorough inspection of the liner plate under both the Coatings Monitoring Program and the Containment Inspection Program have been performed and will continue to be performed to preclude problems similar to those noted in the INs.

Five specific issues were posed by this IN: corrosion and thinning of the liner plate at the floor-to-liner junction, corrosion due to freestanding water, corrosion due to inadequate coatings, liner plate degradation due to foreign objects embedded in the exterior concrete, and degraded moisture barrier seal at the floor-to-liner junction.

The floor-to-liner junction at DCPP is thoroughly inspected through both the ISI Containment Inspection Program and the Coatings Quality Monitoring Program, with no adverse indications identified to date. The coatings applied to the liner and floor slab are safety-related immersion coatings. An area of greater concern than the floor-to-liner junction is the RHR recirculation sump area. The RHR recirculation sump area is a potentially corrosive environment for which an augmented inspection is performed. The floor-to-liner junction is a dry (non-corrosive) area and does not require augmented inspections. The augmented inspections are performed in accordance with ASME Section XI, IWE-2500. The

augmented examination includes an ultrasonic thickness inspection of the containment liner in the RHR sump area. The liner within the sump area is gridded into 12-inch by 12-inch squares and ultrasonic readings are obtained at the grid intersection points.

Corrosion due to freestanding water (from a clogged drain, etc.) would be detected under the coatings monitoring inspections, performed every outage. Any issues would be identified and resolved according to the Containment Coatings Monitoring Program. The DCPP Coatings Monitoring Program provides assurance that identified problems are evaluated by the Engineering Department such that adequate corrective actions can be implemented.

Corrosion due to inadequate coatings would be detected under the coatings monitoring inspections, which are performed every outage. Any deficiencies would be identified and resolved according to Containment Coatings Monitoring Program.

As industry operating experience has shown, the back of the liner plate can be subject to corrosion as a result of foreign objects embedded in the concrete; however, it is very difficult to identify this type of corrosion until evidence of degradation appears on the interior accessible surface of the liner plate. The Coatings Monitoring Program, performed every outage, would identify this type of degradation as a deficiency and then route the issue to the Engineering Department for further evaluation. In addition, the integrity of the exterior concrete shell is important to the protection of the liner plate. To meet the requirements of ASME Section XI, Subsection IWL, DCPP procedures provide for inspection of the accessible surface of the containment concrete. The purpose of this inspection is to determine the general structure condition by identifying areas of concrete deterioration or distress. The acceptance criteria for the concrete shell are such that any conditions that could affect the liner plate (concrete degradation, discoloration, moisture seepage, etc.) are identified. The results of these inspections at DCPP to date have not revealed adverse conditions of any magnitude that would warrant concern about corrosion on the back of the liner plate.

In response to IN 1989-79, DCPP evaluated the installation of a moisture barrier seal at the floor-to-liner junction. It was subsequently determined that the installation of a seal was not necessary. The RHR recirculation sump area and the liner-to-concrete interface area were included in the Containment Coatings Monitoring Program with special attention to ensure that boric acid corrosion of the containment liner near the basemat interface is averted. This area is also included as a line item in the coatings report generated after each outage to ensure that potential problems are identified. Both the Containment Coatings Monitoring Program and the ISI Containment Inspection Program monitor this area.

3.5.5 NRC IN 2010-12, "Containment Liner Corrosion"

This IN was issued to alert plant operators to three events that occurred where the steel liner of the containment building was corroded and degraded. At Beaver Valley and Brunswick plants, material had been found in the concrete which trapped moisture against the liner plant and corroded the steel. In one case, it was material intentionally placed in the building, and in the other case it was foreign material which had inadvertently been left in the form when the wall was poured. But the result in both cases was that the material trapped moisture against the steel liner plate leading to corrosion. In the third case, Salem, an insulating material placed between the concrete floor and the steel liner plate adsorbed moisture and led to corrosion of the liner plate.

Discussion:

The DCPP containment structure is reinforced concrete with a mild carbon steel liner plate. The wall liner is made of 3/8-in. plate, except for the bottom section (approximately 4-ft high) where 3/4-in. plate is used with local reinforcement of the liner around penetration openings. During construction, the liner plate was blast cleaned to Joint Surface Preparation Standard SP 10 and coated with 3/1000-in. inorganic zinc primer both inside and out. The inside of the liner plate is coated with an epoxy coating.

A detailed review was conducted and found that the exterior concrete structure, liner plate, penetrations, and penetration boundaries do not contain embedded items that could potentially cause corrosion on the concrete side of the containment liner plate.

DCPP Procedure ISI VT GEN-1, "General Visual Examination of the Containment Liner," provides the performance requirements for General Visual examination of accessible surfaces of the containment liner. This procedure meets the requirements of ASME Code Section XI, 2001 Edition with 2003 Addenda, as modified by 10 CFR 50.55a. This procedure applies to the General Visual examination of the containment liner, associated structure, and structural attachments, as required by Subsection IWE. In addition to ISI VT GEN-1, the Coating Quality Monitoring Program (DCPP Procedure MIP CT-2.0, "Coating Quality Monitoring Program (DCP-210)") was developed to provide assurance of continued acceptable performance of coatings inside the containment structures. The Coating Quality Monitoring Program is performed during every refueling outage, which is a DCPP commitment to the NRC. This inspection is conducted on all accessible coated surfaces of the containment, which includes the liner plate. IN 2010-12 notes that Beaver Valley has committed to performing supplemental volumetric inspections of 1 square ft samples in at least 75 random locations of each unit's containment liner. Beaver Valley also committed to performing supplemental volumetric examinations of a minimum of eight 1 square ft areas in locations that OE shows are susceptible to localized pitting

corrosion. Additionally, Salem committed in its license renewal submittals that they would perform supplemental and augmented examinations of the liner plates at random and non-random locations. The DCPP IWE program consists of the code required visual inspections of the liner plate (augmented ultrasonic inspection in the RHR sump was performed prior to sump replacement with a closed system that made this augmented exam unnecessary). During the inspection process, DCPP identified a small number of suspect areas and then investigated those areas with surface or volumetric exams. The results of these exams have shown no significant degradation or loss of plate section.

3.5.6 NRC IN 2014-07, "Degradation of Leak Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Metallic Liner"

The containment basemat metallic shell and liner plate seam welds of PWRs are embedded in a 3-ft to 4-ft thick concrete floor during construction and are typically covered by a leak chase channel system that incorporates pressurizing test connections. This system allows for pressure testing of the seam welds for leak-tightness during construction and also while in service, as required. A typical basemat shell or liner weld leak-chase channel system consists of steel channel sections that are fillet welded continuously over the entire bottom shell or liner seam welds and subdivided into zones, each zone with a test connection.

Each test connection consists of a small carbon or stainless steel tube (less than 1-in. diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak chase channel. After the initial tests, steel threaded plugs or caps are installed in the test tap to seal the leak-chase volume. Gasketed cover plates or countersunk plugs are attached to the top of the access box flush with the containment floor. In some cases, the leak chase channels with plugged test connections may extend vertically along the circumference of the cylindrical containment shell or liner to a certain height above the floor.

Discussion:

The interface of the DCPP containment liner to the concrete base was not constructed with a moisture barrier. The majority of visible leak chase channels are above the concrete basemat (91-ft elevation). There are leak chase channels on the floor liner plate embedded below the 91-ft elevation in 2 ft of concrete. However, there are no leak chase test fittings that extend up through the containment floor at elevation 91-ft. The leak chase channels on the containment wall liner penetrate into the concrete floor/basemat and communicate with the basemat leak chase channels. Therefore, there is no need for test fittings to extend up through the floor.

Based on the review, the leak chase configuration is not similar to the examples in the IN. Therefore, DCPP is not susceptible to the condition.

3.6 License Renewal Aging Management

PG&E requested license renewal for DCPP from the NRC in letter DCL-09-079 (Reference 12) submitted November 23, 2009, but has subsequently put the application on hold. The following programs, which are part of the supporting basis for this submittal, are also Aging Management Programs for DCPP:

3.6.1 <u>Containment Inservice Inspection Program</u>

The Containment ISI Program is divided into two subsections, ASME Section XI, IWE and ASME Section XI, IWL. The DCPP ASME Section XI IWE Program manages loss of material and loss of sealing through the containment. It provides aging management of the concrete containment steel liner. IWE inspections are carried out in order to find and manage containment liner aging effects that could result in the loss of intended function. Included in the inspection program are the containment liner plate and its integral attachments, containment hatches and airlocks, and pressure-retaining bolting. For the second containment inspection interval commencing in May 2008, DCPP performed IWE containment ISIs in accordance with the 2001 Edition of ASME Section XI, Subsection IWE (with the 2002 and 2003 addenda) supplemented with the applicable requirements of 10 CFR 50.55a(b)(2)(ix). In addition, the ASME Section XI, Subsection IWE Program is consistent with provisions in 10 CFR 50.55a that specify the use of the ASME Code edition in effect 12 months before the start of the inspection interval.

The DCPP ASME Section XI IWL Program manages cracking due to expansion, loss of bond, and loss of material (spalling, scaling) and provides an approach for aging management of the conventionally-reinforced concrete containment buildings for Units 1 and 2. The design of the DCPP Units 1 and 2 containment buildings does not include post-tensioned tendons. The ASME Section XI, Subsection IWL inspections are performed in order to find and manage containment concrete aging effects that could result in loss of intended function. Accessible surfaces of the containment exterior dome are included within this inspection program.

<u>Containment Inservice Inspection Program - License Renewal Aging Management Commitments</u>

During the LRA process, several RAIs were issued to DCPP regarding the Containment ISI Program.

During review of plant-specific OE in the application, it was noted that gaps were identified in isolated spots along the liner plate and floor interface on the 91-ft

elevation during the 2R15. The issue was documented showing that no corrosion was found at the liner/concrete interface and that the concrete was in good condition (no cracks or delamination). Recommendations were made to seal the gaps to prevent any liquid intrusion into the gaps and minimize the potential for corrosion of the carbon steel liner. The staff questioned how the program would effectively manage aging of the carbon steel containment liner during the period of extended operation if permanent remediation by sealing the gap between the liner plate and concrete is not completed. In the response to the RAI, PG&E committed to completing the Unit 2 gap repair work prior to the period of extended operation by installation of sealant (caulking). (Reference 13)

In the LRA, PG&E stated that the DCPP ASME Section XI, Subsection IWL Program is in accordance with IWL-2400. Visual examinations of 100 percent of the accessible surfaces on the concrete shells will be completed on 10-year cycles, with each unit being examined every 5 years. However, the 2001 edition of ASME Section XI, Subsection IWL-2410 states that concrete shall be examined in accordance with IWL-2510 at 1, 3, and 5 years following the completion of the containment Structural Integrity Test CC-6000 and every 5 years thereafter. In addition, the requirements in ASME Section XI, Subsection IWL-2421 only apply to sites with multiple units that have containments with unbounded post-tensioning systems. The staff questioned how the program is consistent with the recommendations in the GALL Report (Reference 42) Section XI.S2, "ASME Section XI, Subsection IWL." In addition, the staff questioned the basis for selecting the 10-year inspection frequency for each unit and what impact it would have on the detection of aging effects.

In the response to the RAI, PG&E stated that the basis behind the 10-year frequency was an incorrect interpretation of ASME Section XI paragraphs IWL-2410 and IWL-2421. This error caused the Unit 1 containment concrete inspection per Subsection IWL to not be performed in the outage closest to 2005, as required. The issue was entered into DCPP's Corrective Action Program for resolution. The issue was determined to not apply to Unit 2, as the examinations for Unit 2 were completed as required. In October 2010, the Unit 1 examination was scheduled during 1R16 and was completed successfully. Significant testing of the containment structure was performed in the surveillance interval, including the Type A test and the containment structural integrity test. No adverse indications were found during the performed tests. Based on the testing performed, it is reasonable to conclude that the containment structural function was adequately maintained. PG&E committed to revise the plant procedures which perform the concrete inspections per ASME Section XI, Subsection IWL within a 5-year interval after receiving approval of the license renewal from the NRC.

3.6.2 Protective Coating Monitoring and Maintenance Program

Compliant with GALL Report section XI.S8, "Protective Coating Monitoring and Maintenance Program," the DCPP Protective Coating Monitoring and Maintenance Program is an existing program that manages cracking, blistering, flaking, peeling, and delamination of Service Level I coatings subjected to indoor air in the containment structure.

3.6.3 <u>10 CFR 50, Appendix J</u>

The DCPP 10 CFR 50, Appendix J Program manages loss of sealing, leakage through containment, loss of leak tightness, and loss of material. The program detects pressure boundary degradation in the reactor containment and all systems and components penetrating containment that are covered under the 10 CFR 50, Appendix J program. The program includes the steel liner of the concrete containment and its integral attachments as well as welds, gaskets, seals, and bolted connections for the containment pressure boundary access points. The 10 CFR 50, Appendix J program consists of tests performed in accordance with the regulations and guidance provided in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," (Option B); RG 1.163, "Performance-Based Containment Leak-Testing Program;" NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J;" and ANSI/ANS 56.8 - 1994, "Containment System Leakage Testing Requirements." Containment leak rate tests are performed in accordance with 10 CFR Part 50, Appendix J, Option B to assure that leakage through the reactor containment and systems and components penetrating containment does not exceed allowable leakage limits specified in the DCPP TS. Periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment and the systems and components penetrating containment. A Type A test is performed during a period of reactor shutdown at the frequency specified in 10 CFR Part 50, Appendix J, Option B. Type B and C tests are performed on isolation valves and containment access penetrations at frequencies that comply with the requirements of 10 CFR 50 Appendix J, Option B.

All containment leak rate tests are performed and documented in accordance with plant procedures.

3.7 NRC Safety Evaluation Report Limitations and Conditions

3.7.1 <u>Limitations and Conditions Applicable to NEI, 94-01 Revision 2-A</u>

The NRC staff found that the use of NEI 94-01, Revision 2-A, was acceptable for referencing by licensees proposing to amend their TSs to permanently extend the Type A test surveillance interval to 15 years, provided the following conditions as listed in Table 3.7.1-1 were satisfied:

Table 3.7.1-1: NEI 94-01, Rev. 2-A, Limitations and Conditions

Limitation/Condition (From Section 4.0 of SER)	DCPP Response
For calculating the Type A leakage rate, the licensee should use the definition in the NEI 94-01, Revision 2-A, in lieu of that in ANSI/ANS-56.8-2002. (Refer to SER Section 3.1.1.1.)	DCPP will utilize the definition in NEI 94-01, Revision 2-A, Section 5.0.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SER Section 3.1.1.3.)	Reference Tables 3.4.2-1 and 3.4.2-2 of this submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SER Section 3.1.3.)	Reference Section 3.4.2 of this submittal.
The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SER Section 3.1.4.)	There are no major modifications planned.
The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI 94-01, Revision 2, related to extending the Type A test interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SER Section 3.1.1.2.)	DCPP will follow the requirements of NEI 94-01, Revision 2-A, Section 9.1. In accordance with the requirements of NEI 94-01, Revision 2-A, SER Section 3.1.1.2, DCPP will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.
For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the Type A test surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment Type A test data.	Not applicable. DCPP was not licensed under 10 CFR Part 52.

3.8 Conclusion

NEI 94-01, Revision 2-A, dated October 2008, describes an NRC endorsed approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporated the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01,

Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. DCPP is adopting the guidance of NEI 94-01, Revision 2-A, for the DCPP, 10 CFR 50, Appendix J testing program plan.

Based on the previous Type A tests conducted at DCPP, it may be concluded that the permanent extension of the containment Type A test interval from 10 to 15 years represents minimal risk to increased containment leakage. The risk is minimized by continued Type B and Type C testing performed in accordance with Option B of 10 CFR 50, Appendix J and the overlapping inspection activities performed as part of the following DCPP inspection programs:

- Containment Inservice Inspection Plan (IWE and IWL)
- 10 CFR 50, Appendix J Program Containment Inspections per TS 5.5.16
- Protective Coating Monitoring and Maintenance Program

This conclusion is supplemented by risk analysis studies, including the DCPP risk analysis provided in Attachment 3 of this submittal. The risk analysis concluded that increasing the Type A test interval to 15 years is considered to be insignificant since it represents a very small change to the DCPP risk profile.

4. REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. 10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of 10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." 10 CFR 50, Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, 10 CFR 50, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B, and Type C testing did not alter the basic method by which 10 CFR 50, Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that reviewed AF leakage history to determine the frequency for leakage testing which provides assurance that leakage

limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage.

EPRI Report No. 1009325, Revision 2 provided a risk impact assessment for optimized Type A test intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94-01, Revision 2-A, Section 9.2.3.1 states that Type A test intervals of up to 15 years are allowed by this guideline. EPRI 1018243, "The Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," indicates that, in general, the risk impact associated with Type A test interval extensions for intervals up to 15 years is small. However, plant-specific confirmatory analyses are required.

Per Section 5.0, Conclusion, of the SER for NEI 94-01, Revision 2, "The NRC staff reviewed NEI TR 94-01 [sic], Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01 [sic], Revision 2, the NRC staff determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J. This guidance includes provisions for extending Type A ILRT test intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC staff finds that the Type A testing methodology as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by NEI 94-01, Revision 2, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

> "For EPRI Report No. 1009325, Revision 2, a risk-informed methodology using plant-specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC staff finds that the proposed methodology satisfies the key principles of risk-informed decision making applied to changes to TSs as delineated in RG 1.177 and RG 1.174.
>
> "The NRC staff, therefore, found that this guidance is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.0 of this SE."

Thus, PG&E proposes to adopt NEI 94-01, Revision 2-A, to implement a performance-based leakage testing program in accordance with Option B to 10 CFR 50, Appendix J, for DCPP.

4.2 Precedent

This request is similar in nature to the following license amendments to extend the Type A test frequency to 15 years in accordance with NEI 94-01, Revision 2-A, as previously authorized by the NRC:

- Nine Mile Point Nuclear Station, Unit 2 (Reference 25)
- Arkansas Nuclear One, Unit 2 (Reference 22)
- Palisades Nuclear Plant (Reference 23)
- Virgil C. Summer Nuclear Station, Unit 1 (Reference 29)

4.3 Significant Hazards Consideration

Pacific Gas and Electric Company (PG&E) has evaluated whether or not a significant hazards consideration is involved with the proposed license amendment by focusing on the three standards set forth in Title 10 of the Code of Federal Regulation (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 license amendment adopts the Nuclear Regulatory Commission (NRC)-accepted guidelines of Nuclear Energy Institute (NEI) Report 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," for development of the Diablo Canyon Power Plant (DCPP) Units 1 and 2 performance-based Technical Specification 5.5.16, "Containment Leakage Rate Testing Program." NEI 94-01 allows, based on risk and performance, an extension of Type A containment leak test intervals. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses.

The findings of the DCPP risk assessment confirm the general findings of previous studies that the risk impact with extending the containment leak rate is small, per the guidance provided in Regulatory Guide (RG) 1.174, Revision 2.

Since the license amendment is implementing a performance-based containment testing program, the proposed license amendment does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. The requirements for leakage rate tests and acceptance criteria will not be changed by this license amendment. Therefore, the containment will continue to perform its design function as a barrier to fission product releases.

The proposed license amendment also deletes an exception previously granted to allow one time extensions of the Type A test frequency for DCPP. This exception was for an activity that has already taken place; therefore, the deletion is solely an administrative action that has no effect on any component and no physical impact on how the units are operated.

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 license amendment to implement a performance-based Type A testing program does not change the design or operation of structures, systems, or components of the plant. In addition, the proposed changes would not impact any other plant system or component.

The proposed license amendment would continue to ensure containment integrity and would ensure operation within the bounds of existing accident analyses. There are no accident initiators created or affected by the proposed changes.

The proposed license amendment also deletes an exception previously granted to allow one time extensions of the Type A test frequency for DCPP. This exception was for an activity that has already taken place; therefore, the deletion is solely an administrative action and does not change how the units are operated or maintained.

Therefore, the proposed license amendment does not create the possibility of a new or different accident from any accident previously evaluated.

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

Response: No.

The proposed license amendment to implement the performance-based Type A testing program does not affect plant operations, design functions, or any analysis that verifies the capability of a structure, system, or component of the plant to perform a design function. In addition, this change does not affect safety limits, limiting safety system setpoints, or limiting conditions for operation.

The specific requirements and conditions of Technical Specification 5.5.16, "Containment Leakage Rate Testing Program," exist to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leak rate limit specified by the Technical Specifications is maintained. This ensures that the margin of safety in the plant safety analysis is maintained. The proposed amendment will ensure that the design, operation, testing methods and acceptance criteria for Type A tests specified in applicable codes and standards would continue to be met since these are not affected by implementation of a performance-based Type A testing interval.

The proposed amendment also deletes an exception previously granted to allow one time extensions of the Type A test frequency for DCPP. This exception was for an activity that has taken place; therefore, the deletion is solely an administrative action and does not change how the unit is operated and 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 does not involve a 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.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be impacted 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. ENVIRONMENTAL CONSIDERATION

The proposed amendment 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 license 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 needs to be prepared in connection with the proposed amendment.

6. REFERENCES

1. American Concrete Institute, ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," dated January 1, 1996.

2. American National Standards Institute and American Nuclear Society, ANSI/ANS 56.8-1994, "American National Standard for Containment System Leakage Testing Requirements," dated August 4, 1994.

3. American National Standards Institute and American Nuclear Society, ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements," dated November 2002.

4. American National Standards Institute and American Nuclear Society, ANSI/ANS-58.23-2007, "Fire PRA Methodology," withdrawn February 2, 2009, Superseded by ASME/ANS RA-Sa-2009 (Reference 8).

5. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1977 Edition through 2003 Addenda.

6. American Society of Mechanical Engineers, ASME Section III, "Rules for Construction of Nuclear Facility Components."

7. American Society of Mechanical Engineers, ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition through 2003 Addenda.

8. American Society of Mechanical Engineers and American Nuclear Society, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," dated February 2009.

9. PG&E Letter DCL-85-058, "Additional Information on Paint," dated February 09, 1985.

10. PG&E Letter DCL-85-108, "Additional Information on Paint," dated March 13, 1985.

11. PG&E Letter DCL-06-142, "License Amendment Request 06-09, Revision to Technical Specification 5.5.16, 'Containment Leakage Rate Testing Program' (TSTF-343)," dated December 29, 2006 (ADAMS Accession No. ML070160261).

12. PG&E Letter DCL-09-079, "License Renewal Application," dated November 23, 2009 (ADAMS Accession No. ML093340086).

13. PG&E Letter DCL-10-077, "Response to NRC Letter dated June 21, 2010, Request for Additional Information (Set 5) for the Diablo Canyon License Renewal Application," dated July 19, 2010 (ADAMS Accession No. ML102530195).

14. Electric Power Research Institute, EPRI Report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007.

15. Electric Power Research Institute, EPRI Report No. 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated October 2008.

16. Electric Power Research Institute, EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.

17. John M Gisclon (EPRI), William Parkinson, and Ken Canavan (Data Systems and Solutions), "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals," developed for NEI, dated October 2001 (ADAMS Accession No. ML012990239).

18. NRC Letter from A. Wang to J. Keenan (PG&E), "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Technical Specification 5.5.16, 'Containment Leakage Rate Testing Program,' For Consistency with 10 CFR 50.55a(g)(40) (TAC Nos. MD3977 and MD3978)," dated June 26, 2007 (ADAMS Accession No. ML071370731).

19. NRC Letter from A. Wang to J. Conway (PG&E), "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Revision to Technical Specification 5.5.16, 'Containment Leakage Rate Testing Program,' (TAC Nos. MD8042 and MD8043)," dated January 15, 2009 (ADAMS Accession No. ML090080387).

20. Constellation Nuclear Letter from C. Cruse to the NRC, "Response to Request for Additional Information Concerning the License Amendment

Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002 (ADAMS Accession No. ML020920100).

21. NRC Letter from G. Shukla to G. Rueger (PG&E), "Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Revision of Technical Specifications Section 5.5.16, for a One-Time Extension of the 10 CFR Part 50, Appendix J, Integrated Leak Rate Test Interval (TAC Nos. MB3515 and MB3517)," dated April 22, 2002 (ADAMS Accession No. ML020100111).

22. NRC Letter from K. Kalyanam to Vice President, Operations (ANO) "Arkansas Nuclear One, Unit No. 2 – Issuance of Amendment Re: Technical Specification Change to Extend the Type A Test Frequency to 15 Years (TAC No. ME4090)," dated April 7, 2011 (ADAMS Accession No. ML110800034).

23. NRC Letter from M. Chawla to Vice President, Operations (PNGS) "Palisades Nuclear Plant – Issuance of Amendment to Extend the Containment Type A Leak Rate Test Frequency to 15 Years (TAC No. ME5997)," dated April 23, 2012 (ADAMS Accession No. ML120740081).

24. NRC Letter from M. Maxin to J. Butler (NEI), "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J' and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals' (TAC No. MC9663)," dated June 25, 2008 (ADAMS Accession No. ML081140105).

25. NRC Letter from R. V. Guzman to S. L. Belcher (NMP), "Nine Mile Point Nuclear Station Unit No. 2 – Issuance of Amendment Re: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650)" dated March 30, 2010 (ADAMS Accession No. ML100730032).

26. NRC Letter from S. Bahadur to B. Bradley (NEI), "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J' (TAC No. ME2164)," dated June 8 2012 (ADAMS Accession No. ML121030286).

27. NRC Letter from S. Bloom to G. Rueger (PG&E), "Issuance of Amendments for Diablo Canyon Nuclear Power Plant Unit No. 1 (TAC No. M94379) and Unit No. 2 (TAC No. M94380)," dated March 1, 1996 (ADAMS Accession No. ML022390548).

28. NRC Letter from S. Lingam to E. Halpin (PG&E), "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Transition to a Risk-Informed Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c) (CAC Nos. MF2333 and MF2334)," dated April 14, 2016.

29. NRC Letter from S. Williams to T.D. Gatlin (VCSNS), "Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385)," dated February 5, 2014 (ADAMS Accession No. ML13326A204).

30. Nuclear Energy Institute, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 21, 1995 (ADAMS Accession No. ML11327A025).

31. Nuclear Energy Institute, NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated October 2008 (ADAMS Accession No. ML100620847).

32. Nuclear Energy Institute, NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012 (ADAMS Accession No. ML12221A202).

33. Nuclear Energy Institute, NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," dated June 2010 (ADAMS Accession No. ML102230070).

34. Nuclear Regulatory Commission, NRC IN 1989-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," dated December 1, 1989.

35. Nuclear Regulatory Commission, NRC IN 1992-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992.

36. Nuclear Regulatory Commission, NRC IN 1997-10, "Liner Plate Corrosion in Concrete Containments," dated March 13, 1997.

37. Nuclear Regulatory Commission, NRC IN 2004-09, "Corrosion of Steel Containment and Containment Liner," dated April 27, 2004.

38. Nuclear Regulatory Commission, NRC IN 2010-12, "Containment Liner Corrosion," dated June 18, 2010.

39. Nuclear Regulatory Commission, NRC IN 2014-07, "Degradation of Leak Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," dated May 5, 2014.

40. Nuclear Regulatory Commission, NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Detailed Methodology, Final Report," dated September 2005.

41. Nuclear Regulatory Commission, NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995.

42. Nuclear Regulatory Commission, NUREG-1801, Revision 2 "Generic Aging Lessons Learned (GALL) Report," dated December 2010 (ADAMS Accession No. ML103490041).

43. Nuclear Regulatory Commission, Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058).

44. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (ADAMS Accession No. 100910006).

45. Nuclear Regulatory Commission, Regulatory Guide 1.200, Revision 2, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities," dated March 2009 (ADAMS Accession No. ML090410014).

46. Technical Specification Task Force, TSTF-343, Revision 1, "Containment Structural Integrity," dated July 7, 1999.

Proposed Technical Specification Page Markup

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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008~~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 43.5 psig. The containment design pressure 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)

Proposed Retyped Technical Specification Page

Remove Page	Insert Page
5.0-16	**5.0-16**

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 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, 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.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.5 psig. The containment design pressure 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)

Proposed Updated Final Safety Analysis Report Markup

TABLE 6.1-1

CRITERIA	TITLE	APPLICABILITY								
Engineered Safety Features		Containment Functional Design	Containment Heat Removal Systems	Containment Air Purification and Cleanup Systems	Containment Isolation System	Combustible Gas Control in Containment	Emergency Core Cooling System	Control Room Habitability System	Technical Support Center Habitability System	Auxiliary Feedwater System
Section		6.2.1	6.2.2	6.2.3	6.2.4	6.2.5	6.3	6.4.1	6.4.2	6.5
5. <u>Regulatory Guides (contd.)</u>										
~~Regulatory Guide 1.163, September 1995~~	~~Performance-Based Containment Leak-Test Program~~	~~X~~			~~X~~					
Regulatory Guide 1.197, Revision 0, May 2003	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors							X		
6. <u>NRC NUREG</u>										
NUREG-0737, November 1980	Clarification of TMI Action Plan Requirements		X		X	X	X	X	X	X
7. <u>NRC Generic Letters</u>										
Generic Letter 89-10, June 1989	Safety-Related Motor-Operated Valve Testing and Surveillance		X	X	X	X	X			X
Generic Letter 95-07, August 1995	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves						X			

6.2.1.1.5 General Design Criterion 55, 1967 – Containment Periodic Leakage Rate Testing

The containment is designed so that integrated leakage rate testing can be done at the design pressure periodically during the plant's lifetime.

6.2.1.1.6 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The containment is designed as a barrier to maintain control over plant radioactive effluents, whether gaseous, liquid, or solid meeting the radiological limits of 10 CFR Part 100. Appropriate holdup capacity is provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment.

6.2.1.1.7 10 CFR Part 50, Appendix J, Option B – Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

The containment is designed to allow for conductance of a performance-based containment leakage rate testing program for Type A containment integrated leak rate tests (ILRT) and Type B testing for the air lock door seals.

10 CFR 50, Appendix J, Option B testing, as modified by approved exemptions, is performed in accordance with Technical Specification 5.5.16, and the guidelines contained in NEI 94-01, Revision 2-A. (References 61 and 62)

6.2.1.1.8 10 CFR Part 50, Appendix K, Part I.A – ECCS Evaluation Models, Sources of Heat during the LOCA

The containment is designed to accommodate the largest credible energy release following a postulated pipe break taking into account the heat sources listed in Paragraph I.A of 10 CFR Part 50, Appendix K.

~~6.2.1.1.9 Regulatory Guide 1.163, September 1995 – Performance-Based Containment Leak-Test Program~~

~~The containment is designed to allow the use of a performance-based leak test program, including the leakage rate test methods, procedures, and analyses as required by Regulatory Guide 1.163, September 1995.~~

6.2.1.2 Description of Short-Term Mass and Energy Releases and Containment Subcompartment Analysis

6.2.1.2.1 Short-Term Mass and Energy Release Analysis

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses. These analyses are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within

6.2.1.3.4 General Design Criterion 54, 1967 – Containment Leakage Rate Testing

Containment leakage rate testing was conducted after completion of construction and installation of all penetrations to verify its conformance with required performance (refer to Section 6.2.1.4.1).

6.2.1.3.5 General Design Criterion 55, 1967 – Containment Periodic Leakage Rate Testing

Periodic ILRT of the containment is performed as part of the DCPP Containment Periodic Leakage Rate Testing Program (refer to Sections 6.2.1.3.7 and 6.2.1.3.9).

6.2.1.3.6 General Design Criterion, 70, 1967 – Control of Releases of Radioactivity to the Environment

The containment, in conjunction with the containment isolation system (CIS) (refer to Section 6.2.4.4.12), is designed to be a barrier to maintain control over plant radioactive effluents, whether gaseous, liquid, or solid. The containment is designed to withstand the effects of a LOCA (refer to Section 6.2.1.3.3), ensuring that the offsite radiological exposures resulting from a LOCA are within the limits of 10 CFR Part 100 (refer to Section 15.5).

6.2.1.3.7 10 CFR Part 50, Appendix J, Option B – Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

10 CFR 50, Appendix J, Option B testing, as modified by approved exemptions, is performed in accordance with Technical Specification 5.5.16, and the guidelines contained in NEI 94-01, Revision 2-A. (References 61 and 62)

~~10 CFR Part 50, Appendix J, Option B, Type A and B testing, as modified by approved exemptions, are performed in accordance with Technical Specification 5.5.16, Containment Leakage Rate Testing Program~~. The ILRT is performed at a P_a, calculated peak containment internal pressure, of 43.5 psig. The maximum allowable leakage rate is not greater than 0.10% of the containment air weight per day. For the discussion of Type C testing (testing of containment penetrations) refer to Section 6.2.4.4.12.

6.2.1.3.8 10 CFR Part 50, Appendix K, Part I.A – ECCS Evaluation Models, Sources of Heat during the LOCA

The initial conditions for the LOCA and MSLB events of the containment integrity analyses are established at the maximum calculated power for the reactor with additional margins for instrument error for the sources of heat listed in accordance with 10 CFR Part 50, Appendix K, Part I.A (refer to Section 6.2D.3.1.4).

~~6.2.1.3.9 Regulatory Guide 1.163, September 1995 – Performance-Based Containment Leak-Test Program~~

~~The DCPP Containment Leakage Rate Testing Program utilizes a performance-based approach, consistent with Regulatory Guide 1.163, September 1995, to comply with the requirements of 10 CFR Part 50, Appendix J, Option B (refer to Section 6.2.1.3.7).~~

Refer to Table 6.2-39 for exceptions to GDC 57, 1971.

10 CFR 50, Appendix J, Option B testing, as modified by approved exemptions, is performed in accordance with Technical Specification 5.5.16, and the guidelines contained in NEI 94-01, Revision 2-A. (References 61 and 62)

6.2.4.1.12 General Design Criterion 70, 1967 – Control of Releases of Radioactivity to the Environment

The containment is designed as a barrier to maintain control over plant radioactive effluents, whether gaseous, liquid, or solid to meeting the radiological limits of 10 CFR Part 100. Appropriate holdup capacity is provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment.

6.2.4.1.13 10 CFR Part 50, Appendix J, Option B - Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

The CIS is designed to enable implementation of a performance-based containment leakage rate testing program for Type C local leak rate tests with approved exemptions.

~~6.2.4.1.14 Regulatory Guide 1.163, September 1995 - Performance-Based Containment Leak-Test Program~~

~~The CIS is designed to allow the use of a performance-based leak-test program, including the leakage-rate test methods, procedures, and analyses as required by Regulatory Guide 1.163, September 1995.~~

6.2.4.1.15 NUREG-0737 (Item II.E.4.2), November 1980 - Clarification of TMI Action Plan Requirements

Item II.E.4.2 - Dependability of Containment Isolation:

Position (1) - The CIS is designed with diverse parameters sensed for the initiation of containment isolation.

Position (2) - The CIS process penetrations are classified as nonessential, essential, and safety system process lines for the determination of those penetrations isolated by a containment isolation signal.

Position (3) - The CIS nonessential systems use either manually sealed closed valves or are automatically isolated on a Phase A containment isolation signal. Additionally, essential systems are automatically isolated on a Phase B isolation signal.

Position (4) - The CIS is designed so that re-setting of a containment isolation signal will not result in the automatic re-opening of any containment isolation valves. Ganged re-opening cannot result from a single operator action after the containment isolation signal has been reset.

6.2.4.4.9 General Design Criterion 55, 1971 – Reactor Coolant Pressure Boundary Penetrating Containment

The CIS is designed such that for each line that is part of the reactor coolant pressure boundary that penetrates containment is provided with containment isolation valves in compliance with GDC 55, 1971. Refer to Section 6.2.4.2.1 and Table 6.2-39 for penetration and configuration details with regards to GDC 55, 1971.

6.2.4.4.10 General Design Criterion 56, 1971 – Primary Containment Isolation

The CIS is designed such that each line that connects directly to the containment atmosphere and penetrates containment is provided with containment isolation valves in compliance with GDC 56, 1971. Refer to Section 6.2.4.2.1 and Table 6.2-39 for penetration and configuration details with regards to GDC 56, 1971.

6.2.4.4.11 General Design Criterion 57, 1971 – Closed System Isolation Valves

The CIS is designed such that each line that penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere is provided with at least one containment isolation valve in compliance with GDC 57, 1971. Refer to section 6.2.4.2.1 and Table 6.2-39 for penetration and configuration details with regards to GDC 57, 1971.

6.2.4.4.12 General Design Criterion, 70, 1967 – Control of Releases of Radioactivity to the Environment

The CIS, in conjunction with the containment (refer to Section 6.2.1.3.6), is designed to be a barrier to maintain control over plant radioactive effluents, whether gaseous, liquid, or solid. The CIS is designed to withstand the effects of a LOCA (refer to Section 6.2.4.4.2), ensuring that the offsite radiological exposures resulting from a LOCA are within the limits of 10 CFR Part 100 (refer to Section 15.5).

6.2.4.4.13 10 CFR Part 50, Appendix J, Option B – Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

, and the guidelines contained in NEI 94-01, Revision 2-A. (References 61 and 62)

Testing of containment penetrations (Type C testing) is performed in accordance with the Technical Specifications 5.5.16, Containment Leakage Rate Testing Program, as required by 10 CFR 50.54(o), and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. As a general requirement, all containment isolation valves will be tested periodically with a gas to determine leaktightness. Refer to Section 6.2.1 for information regarding Type A (ILRT) and Type B (air locks, electrical penetrations, hatches, etc.) testing.

Exceptions to this requirement are those valves not required to be testable by Appendix J to 10 CFR Part 50, and certain valves that cannot be isolated for air testing. These include the first of double check valves to the RCS and valves for which such testing

would require draining significant portions of the RHR system or the SIS. Even if these systems were drained, the presence of other valves associated with the systems would make it impractical to determine the source of any measured leakage. Where a quantitative leakage test is necessary, provisions are made for each valve to measure the inflow of the pressurizing medium, collect and measure leakage, or calculate the leakage from the rate of pressure drop. The test pressure on the valve will be at a differential pressure of not less than the peak calculated containment internal pressure related to the design basis LOCA (P_a). The P_a value specified in Technical Specification 5.5.16 bounds the calculated LOCA containment integrity results in Section 6.2D.3.2.6.

Check valves and single-disk gate valves will have the test pressure applied to the inboard side of the valve. Exceptions are the three RHR injection lines. The valves in these lines will be tested from the outboard side, as there is no practical method to test from the inboard side. Diaphragm valves may be tested on either side since their leakage characteristics are the same in either direction. Double-disk gate valves may be tested by applying the test pressure between the disks. Globe valves may be tested by pressurizing either the inboard side or under the seat.

Piping systems are provided with test vents (TV) and test connections (TC) or have other provisions to allow periodic leakage testing of the containment isolation valves, as required. Locations of TC and TV are shown on the penetration diagram (refer to Figure 6.2-19). In most cases, equipment vents or drains can be used as TC or TV.

~~6.2.4.4.14 Regulatory Guide 1.163, September 1995 – Performance-Based Containment Leak-Test Program~~

~~The DCPP Containment Leakage Rate Testing Program is performed in accordance with the Technical Specifications. The DCPP Containment Leakage Rate Testing Program utilizes a performance based approach, consistent with Regulatory Guide 1.163, September 1995, to comply with the requirements of 10 CFR Part 50, Appendix J, Option B (refer to Section 6.2.4.4.13).~~

6.2.4.4.15 NUREG-0737 (Item II.E.4.2), November 1980 – Clarification of TMI Action Plan Requirements

Item II.E.4.2 - Containment Isolation Dependability:

Position (1) – The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals.

Immediate isolation of the containment is accomplished automatically. There are two automatic phases of containment isolation at DCPP. Phase A isolates all nonessential process lines but does not affect safety injection, containment spray, component cooling water supplied to the reactor coolant pumps and containment fan coolers, and steam and auxiliary feedwater lines. Phase B isolates all process lines except safety injection, containment spray, auxiliary feedwater, and the containment fan coolers component

50. Westinghouse Letter PGE-91-533, Safety Evaluation for Containment Spray Flow Reduction, February 7, 1991.

51. Westinghouse Letter PGE-89-673, RWST Setpoint Evaluation, July 24, 1989.

52. Deleted in Revision 18.

53. Deleted in Revision 18.

54. Deleted in Revision 18.

55. J. C. Griess, A. L. Bacarella, Design Considerations of Reactor Containment Spray Systems - Part III. The Corrosion of Materials in Spray Solutions, ORNL-TM-2412, Part III, December 1969.

56. Deleted in Revision 18.

57. Deleted in Revision 18.

58. Deleted in Revision 18.

59. Westinghouse Mass and Energy Release Data for Containment Design, WCAP-8264-P-A, Rev. 1, (Proprietary), WCAP-8312-A, August 1975.

60. Ice Condenser Containment Pressure Transient Analysis Methods, WCAP-8077 (Proprietary), WCAP-8078, March 1973.

6.2.7 REFERENCE DRAWINGS

Figures representing controlled engineering drawings are incorporated by reference and are identified in Table 1.6-1. The contents of the drawings are controlled by DCPP procedures.

61. License Amendments XXX/XXX, "Revision to Implement Performance Based Option of 10 CFR 50 Appendix J," dated XX/XX/XXXX

62. NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008

Glossary of Acronyms

Glossary of Acronyms

1R15	Unit 1 Fifteenth Refueling Outage
1R16	Unit 1 Sixteenth Refueling Outage
1R17	Unit 1 Seventeenth Refueling Outage
1R18	Unit 1 Eighteenth Refueling Outage
1R19	Unit 1 Nineteenth Refueling Outage
1R20	Unit 1 Twentieth Refueling Outage
1R21	Unit 1Twenty-first Refueling Outage
1R22	Unit 1 Twenty-second Refueling Outage
1R23	Unit 1 Twenty-third Refueling Outage
1R24	Unit 1 Twenty-fourth Refueling Outage
1R25	Unit 1 Twenty-fifth Refueling Outage
1R26	Unit 1 Twenty-sixth Refueling Outage
2R14	Unit 2 Fourteenth Refueling Outage
2R15	Unit 2 Fifteenth Refueling Outage
2R16	Unit 2 Sixteenth Refueling Outage
2R17	Unit 2 Seventeenth Refueling Outage
2R18	Unit 2 Eighteenth Refueling Outage
2R19	Unit 2 Nineteenth Refueling Outage
2R20	Unit 2 Twentieth Refueling Outage
2R21	Unit 2 Twenty-first Refueling Outage
2R22	Unit 2 Twenty-second Refueling Outage
2R23	Unit 2 Twenty-third Refueling Outage
2R24	Unit 2 Twenty-fourth Refueling Outage
2R25	Unit 2 Twenty-fifth Refueling Outage
2R26	Unit 2 Twenty-sixth Refueling Outage
ACI	American Concrete Institute
AF	As Found
AL	As Left
AMP	Aging Management Plan
ANO	Arkansas Nuclear One
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CAT	Capability Category as defined by ASME/ANS RA-Sa-2009
CCFP	Conditional Containment Failure Probability
CCP	Centrifugal Charging Pump
CCP1	Centrifugal Charging Pump 1
CCP2	Centrifugal Charging Pump 2
CDF	Core Damage Frequency

CF-A1	Supporting Requirement defined by ASME/ANS RA-Sa-2009 PRA Standard
CFR	Code of Federal Regulations
CVCS	Chemistry and Volume Control System
DCPP	Diablo Canyon Power Plant
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
F&Os	Findings and Observations
FPRA	Fire Probabilistic Risk Assessment
FSAR	Final Safety Analysis Report
FSS-D7	Supporting Requirement defined by ASME/ANS RA-Sa-2009 PRA Standard
ft	Feet
GALL	Generic Aging Lessons Learned
GE	Area defined as East of G-Line in DCPP Area Location Plan Drawing
GW	Area defined as West of G-Line in DCPP Area Location Plan Drawing
ILRT	Integrated Leak Rate Testing - referred to as Type A testing
in.	Inches
IN	Information Notice
ISI	Inservice Inspection
IWE	Table title in ASME Section XI
IWL	Table title in ASME Section XI
L_a	0.1 percent of containment air weight per day
LAR	License Amendment Request
lbm/day	Pound mass per day
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Testing - referred to as Types B and C testing
LOCA	Loss of Cooling Accident
LRA	License Renewal Application
Max	Maximum
MIC	Microbial induced corrosion
Min	Minimum
MIP	Modification Installation Procedure
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NMP	Nine Mile Point (Nuclear Generating Station)
No.	Number
NPSH	Net Positive Suction Head

NRC	Nuclear Regulatory Commission
NSS	Nuclear Sampling System
OE	Operating Experience
PG&E	Pacific Gas and Electric Company
PNGS	Palisades Nuclear Generating Station
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QOTTC	Quick Opening Transfer Tube Cover
RAI	Request for Additional Information
RCS	Reactor Coolant System
RFO	Refueling Outage
RG	Regulatory Guide
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SCCM	Standard Cubic Centimeters per Minute
SE	Safety Evaluation
SER	Safety Evaluation Report
SGRP	Steam Generator Replacement Project
SI	Safety Injection
SPRA	Seismic Probabilistic Risk Assessment
SR	Supporting Requirement
TR	Topical Report
TS	Technical Specification
TSTF	Technical Specifications Task Force
UAM	Unreviewed Analysis Methods
UFSAR	Updated Final Safety Analysis Report
VCSNS	Virgil C. Summer Nuclear Station
VT	Visual Testing (as defined by ASME Section XI)