Pacific Gas and
Electric Company

Gregory M. Rueger
Senior Vice President–
Generation and
Chief Nuclear Officer

US Mail:
Mail Code B32
Pacific Gas and Electric Company
PO Box 770000
San Francisco, CA 94177-0001

Overnight Mail:
Mail Code B32
Pacific Gas and Electric Company
77 Beale Street, 32nd Floor
San Francisco, CA 94105-1814

415.973.4684
Fax: 415.973.2313

November 16, 2001

PG&E Letter DCL-01-116

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 01-06
Revision to Technical Specification 5.5.16 for a Change in the 10 CFR 50, Appendix J, Integrated Leak Rate Test Interval

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP) respectively. This License Amendment Request (LAR) revises Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," to allow a one-time extension of the ten-year interval for the performance-based leakage rate testing program for Type A tests as prescribed by Nuclear Energy Institute (NEI) Report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and applied by Title 10, Code of Federal Regulations, Section 50 (10 CFR 50), Appendix J, Option B. The ten-year interval between integrated leakage rate tests (ILRTs) will be extended to a fifteen-year interval from the previous ILRTs. The last ILRTs were completed on May 4, 1994, for DCPP Unit 1 and April 30, 1993, for DCPP Unit 2.

This application represents a risk-informed licensing change. The results from the previous ILRTs and containment inspections support deferral of the test. The proposed change meets the criteria of Regulatory Guide 1.174 for risk-informed changes.

The existing schedules for the reactor containment buildings inspections under the requirements of the American Society of Mechanical Engineers Section XI Code Subsections IWE and IWL and the existing 10 CFR 50 Appendix J Type B and C testing programs are not being modified by this request.

A017

Enclosure 1 contains a description of the proposed change, the supporting technical analyses, and the significant hazards determination. Enclosures 2 and 3 contain marked-up and revised TS pages, respectively. Enclosure 4 provides the probabilistic risk assessment calculation that supports the change.

The change in this LAR is not required to address an immediate safety concern. However, in order to facilitate scheduling and avoid preparatory costs associated with the eleventh refueling outage for Unit 2 currently scheduled for February 2003, PG&E requests that this amendment be approved no later than April 28, 2002. PG&E requests the LAR be made effective upon NRC issuance, to be implemented within 30 days from the date of issuance.

Sincerely,

Gregory M. Rueger
Senior Vice President - Generation and Chief Nuclear Officer

cc: Edgar Bailey, DHS
Ellis W. Merschoff
David L. Proulx (w/o Enclosure 4)
Girija S. Shukla
Diablo Distribution (w/o Enclosure 4)

Enclosures
KJS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	)	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	)	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

Gregory M. Rueger, of lawful age, first being duly sworn upon oath says that he is Senior Vice President - Generation and Chief Nuclear Officer of Pacific Gas and Electric Company; that he has executed LAR 01-06 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Gregory M. Rueger
Senior Vice President - Generation and Chief Nuclear Officer

Subscribed and sworn to before me this 16th day of November, 2001.

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1204640
Notary Public - California
San Luis Obispo County
My Comm. Expires Dec 12, 2002

PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION 5.5.16 FOR A CHANGE IN THE 10 CFR 50, APPENDIX J, INTEGRATED LEAK RATE TEST INTERVAL

1.0 DESCRIPTION

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

The proposed change revises Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," to allow a one-time extension of the ten-year interval for the performance-based leakage rate testing program for Type A tests as prescribed by Nuclear Energy Institute (NEI) Report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and applied by Title 10, Code of Federal Regulations, Section 50 (10 CFR 50), Appendix J, Option B.

This letter does not request any changes to the intervals for the 10 CFR 50 Appendix J Type B tests or Type C tests, the schedules for the containment liner or concrete inservice inspections, or the schedules for the 10 CFR 50.65 maintenance rule inspections.

2.0 PROPOSED CHANGE

TS 5.5.16.a requires the following:

> "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 Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995."

The approved one-time deferral of the integrated leakage rate tests (ILRT) would add the following to TS 5.5.16.a:

> "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."

The proposed TS change is noted on the markup TS page provided in Enclosure 2. The revised TS is provided in Enclosure 3. The probabilistic risk assessment (PRA) calculation that supports the change is contained in Enclosure 4.

3.0 BACKGROUND

3.1 Containment Building Description

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 reactor coolant system (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 DCPP Hosgri Earthquake is a 7.5 magnitude seismic event that produces a peak horizontal ground acceleration of 0.75 times the acceleration of gravity.

The concrete outline and equipment locations are shown in Chapter 1 of the DCPP Final Safety Analysis Report (FSAR) Update. The exterior shell consists of a 142 feet high cylinder, topped with a hemispherical dome. The minimum thickness of the concrete walls is 3.6 feet, and the minimum thickness of the concrete roof is 2.5 feet. Both Unit containments have a nominal inside diameter of 140 feet and a nominal inside height of 212 feet. The concrete floor pad is 153 feet in diameter with a minimum thickness of 14.5 feet, 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 provides the pressure boundary and which prevents the release of radioactive materials to the environment under any postulated accident condition. The wall liner is 3/8 inches thick, except for the bottom section (approximately 4 feet high) next to the base mat where the thickness is 3/4 inches. The dome liner is 3/8 inches thick. The thickness of the base mat liner is 1/4 inches and this liner is covered with a 24 inches thick concrete floor slab for protection of the liner. The top of the concrete floor slab is at an elevation of 91.0 feet.

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

Most of the reinforcement bars in the concrete shell are placed near the outside face of the shell to minimize temperature stresses; however 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 vertical shear and membrane 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 base mat.

The equipment hatch is an 18 feet 6 inches nominal diameter opening for transportation of equipment through the containment wall. The opening is bound by a 3 inch thick, 24 inch wide, 18 feet 6 inches inside diameter steel band, welded into the liner plate. An approximately 22 feet squared area of the liner around the steel band is thickened to 1-1/2 inches so that the liner material displaced by the opening is replaced.

The personnel hatch provides access to the inside of containment through a 9 feet diameter, 17 feet 6 inches long penetration sleeve with bulkheads and sealed access doors at both ends. The penetration sleeve is made of 3/4 inch and 3/8 inch plate steel plate, welded to a 13 feet diameter liner insert plate. The bulkheads are made of 1-1/8 inch steel plate stiffeners.

The emergency hatch is similar to the personnel hatch except it is smaller. The access door is 30 inches in diameter. The penetration sleeve is 5 feet in diameter and is constructed of a 1/2 inch plate which is welded to a 110 inch diameter, 1 inch 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 inches 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 Information Notice 92-20 are not applicable to DCPP and no inservice inspections of stainless steel bellows are required.

3.2 Containment Leakage Rate Test Requirements

10 CFR 50 Appendix J specifies the leakage rate test requirements for primary reactor containments. The test requirements ensure that:
(a) leakage through containment or systems and components penetrating containment does not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during its service

life. DCPP has adopted Option B of Appendix J, which requires that the ILRT be performed at periodic intervals based on performance of the containment system.

Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” endorses NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995. NEI 94-01 provides methods acceptable to the NRC staff for complying with the provisions of Option B of 10 CFR 50 Appendix J. NEI 94-01 includes the criterion that Option B Type A testing be performed at an interval of at least once per ten years.

This request does not modify the existing containment building inspections under the requirements of the American Society of Mechanical Engineers (ASME) Section XI Subsections IWE and IWL or the existing leakage rate testing programs under Appendix J Type B and Type C.

3.3 Purpose for Proposed Amendment

The proposed one-time relaxation in ILRT interval will eliminate the burden of performing a test for which data exists indicating it is unnecessary. This one-time relaxation will make resources available for other work that provides a greater benefit for maintaining protection of the health and safety of the public and will reduce worker dose.

4.0 TECHNICAL ANALYSES

4.1 Leakage Rate Testing Interval

Adoption of the 10 CFR 50 Appendix J Option B performance-based containment leakage rate testing program allowed a change in the frequency of measuring primary containment leakage in Type A, B, and C tests.

The surveillance interval for Type A testing specified in NEI 94-01 is at least 1-in-10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3 (i.e., past component performance, service, design, safety impact, and cause determination).

The containment leakage rate testing interval is based upon an evaluation which looks at the "as found" leakage history to determine the interval for leakage rate testing which provides assurance that leakage rate limits will be maintained. The Type A test interval was changed from 3-in-10 years to

1-in-10 years in Amendment number 110 for DCCP Unit 1 and Amendment 109 for DCPP Unit 2.

The currently allowed interval of 1-in-10 years for containment leakage rate testing was based upon a generic evaluation documented in NUREG-1493. NUREG-1493 made the following observations with regard to decreasing the test frequency:

- Reducing the Type A (ILRT) testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because an ILRT will identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, the interval between integrated leakage rate testing can be increased with minimal effect on public risk.
- While Type B and C tests identify the vast majority (approximately 97 percent) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Because leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

4.2 Continuation of Type B and C Tests

The existing 10 CFR 50 Appendix J Type B and Type C testing programs will not be modified by this change. PG&E will continue to follow option B of Appendix J and NEI 94-01 with a frequency not to exceed 60 months.

4.3 Inservice Inspections

Several types of inservice inspection including, ASME Section XI inspections, containment liner inspections, and 10 CFR 50.65 Maintenance Rule inspections, are performed on the containment buildings. The results of all inspections indicate that the DCPP containment is structurally sound and is in good condition.

Inservice inspection of the DCPP containment buildings is conducted in accordance with the requirements of the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of the ASME Section XI Code. Subsection IWE provides the rules and requirements for inservice inspection of the metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Subsection IWL provides the rules and requirements for in-service inspection of the containment concrete shell.

Containment Liner Inspections

As required by IWE and the implementing rulemaking in 10 CFR 50.55a(g)(6)(ii)(B), all accessible areas of the containment liners in each unit received a VT general examination by trained and qualified visual examiners between September 9, 1996, and September 9, 2001. The results for the in-service examination indicate that there are some minor areas of coating degradation occurring, but nothing significant that would adversely impact either the containment structural integrity or leak tightness. All identified areas of minor coating degradation were evaluated and found to be limited in scope, with no significant liner material loss, and no potential for acting as precursors to significant containment liner failures.

IWE-1240 requires areas that are likely to experience accelerated degradation and aging be classified as Examination Category E-C and meet the examination requirements identified in Table IWE-2500-1. Based on the results of documented historical issues, the sump liner for each unit has been classified as Examination Category E-C (augmented examination areas). These augmented examination areas were carefully overlaid with a grid system as required in IWE-2500 and ultrasonically examined for wall thinning. This allowed an assessment of the condition of the backside of the liner in this area, which was determined to be sound and free from detectable wastage. A visual examination of the augmented inspection area noted no significant attack of the liner. The thickness information will be utilized to monitor for any adverse material wastage trends as required in Table IWE-2500-1 footnote (2). Note that in the majority of the augmented inspection area of the liner, the liner is 3/4 inches thick, rather than the 3/8 inches general thickness of the liner at higher elevations.

Several plants have identified through-holes in their containment liner due to corrosion of the backside of the liner that is against concrete. These defects in the liner were identified in areas where the protective coating had blistered. Behind the blistered areas of paint, a rust area was discovered. Corrective actions were developed to remove the defective portion of the liner, inspect for further damage, and perform a containment liner weld repair. Upon removal of the containment liner, wood and cloth material were discovered embedded in the concrete. The cause of the through wall defects were the results of the foreign material being in contact with the containment liner plate which interfered with the normal tendency for the concrete's alkalinity to inhibit corrosion of the embedded steel. The occlusion of the surface at the point of contact between the steel and the foreign material created a point of active corrosion, influenced by the residual moisture in the foreign material. The cause of the foreign material

being embedded in the concrete was not determined by the responsible utilities.

In one of the events described above, insitu leak rate tests were performed prior to the removal of the liner plate. In all cases the leak rate tests demonstrated the leakage to be well within acceptable limits. As such, there were no significant safety consequences or implications as a result of the liner holes that were identified. It is believed that the concrete provided a significant barrier to maintain the local leak rate within acceptable limits. At one plant, a "localized" leak rate test was performed, while the plant with the larger repair area had been scheduled for an ILRT that outage. Both tests had satisfactory results.

At DCPP the liner coatings are inspected each outage by a qualified inspector. This inspection is designed to identify any similar defects as those identified above. Additional Maintenance Rule inspections are performed on a regular basis, every 5 to 10 years, and are intended to identify similar defects in the coatings. The Section XI, Subsection IWE inspections also identify liner defects through visual inspections of the coatings.

To date no coating defects have been identified at DCPP which have led to identification of through holes in either containment liner.

Containment Concrete Inspections

The IWL in-service examination applies to the VT-3C visual examination of the containment concrete shell. Portions of the concrete surface that are covered by the liner (including penetration sleeves), foundation material or backfill, or are otherwise obstructed by adjacent structures, components or parts are exempt from examination. VT-3C examination of the containment concrete employs a three-tier acceptance process similar to that described in American Concrete Institute ACI 349.3R-96. This procedure 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.

The IWL examination consisted of a visual examination by trained and qualified examiners of 100 percent of the accessible exterior concrete surface of the Unit 1 and 2 containment structures. The Unit 1 examination was performed from October 2000 to April 2001. The Unit 2 examination was performed from April 2001 to June 2001. For Unit 1

and Unit 2, over 90 percent of the entire concrete surface area was examined.

The results of the IWL examination concluded the Unit 1 and 2 containment buildings are structurally sound. It was concluded that all indications on the containment concrete were acceptable and that they will not have any adverse effect on the structural integrity of the containment and that there is no loss of structural capacity. Based on the results of the IWL examination, no repairs of the Unit 1 or 2 containment concrete were required and no areas of the concrete shells have been classified as augmented examination areas.

Containment Seal, Gasket, and Bolt Inspections

Inservice inspection (ISI) relief requests for seals and gaskets and for examination and testing of bolting were submitted to the NRC in letter DCL-00-075 dated May 11, 2000. The NRC approved these relief requests in a letter dated September 28, 2000. As stated in the relief requests, the alternate examinations of Appendix J Type B (hatches) testing will be performed at least once during each containment ten-year inspection interval. Thus the extension requested for Type A testing does not affect the interval of these alternate examinations in that they will continue to be performed once in the ten-year inspection interval. When seals are disassembled, they are inspected by maintenance personnel. Results of inspections of seals, gaskets, and bolts have shown that 10 CFR 50 Appendix J Type B (hatches) component seals have had an excellent performance record. Seals typically last many years when properly protected during refueling outages. There have been only a few bolting replacements resulting from inspection requirement changes. Thus the schedule for examination of the seals, gaskets, and bolts will continue to provide assurance that the integrity of the containment pressure boundary is maintained.

Therefore the one-time extension of the Type A containment testing from 1-in-10 years to 1-in-15 years is acceptable since the inspections of the containment concrete, containment liner, and containment seals, gaskets, and bolts have shown that they are in good condition and that they are structurally sound.

4.4 <u>Maintenance Rule Inspections</u>

An appropriate program has been developed and implemented to meet the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [the Maintenance Rule]. Periodic assessment, (a)(3) of the Maintenance Rule, shows that

the program for monitoring the condition and effectiveness of structures meets the requirements of 10 CFR 50.65.

The Maintenance Rule Civil implementation program as defined in MA1.NE1 and CMR.WI1 takes credit for the ASME Section XI, subsections IWE and IWL programs. No separate condition monitoring inspections were performed of the containment liner and concrete shell. An assessment of the results of these inspections shows that the structures meet their design basis. Both containment structures are in (a)(2) status.

4.5 Previous Integrated Leak Rate Test (ILRT) Results

Previous Type A test results confirmed that the reactor containment structure is low leakage and represents minimal risk of increased leakage. The risk is minimized by continued Type B and Type C testing for containment penetrations with direct communication with the containment atmosphere. Also, the ISI Program (IWE/IWL) and Maintenance Rule inspections provide confidence in containment integrity through identification of degradation of the containment liner and concrete which may impact containment structural integrity and leak tightness.

To date, six Type A tests, including preoperational and operational testing, have been performed on Unit 1, and two Type A tests, including preoperational and operational testing, have been performed on Unit 2. As indicated in Table 1 below, there is considerable margin between the Type A test results and the TS 5.5.16 limit of 0.75 L_a, where L_a is equal to 0.1 percent of the containment air weight per day at the peak calculated containment internal pressure for the design basis loss of coolant accident (47 psig). These test results demonstrate that both units have low leakage containments.

Table 1
DCPP Units 1 and 2 Type A Test Results

Unit	Date	Mass Point Leakage (%)	Acceptance Limit (%)	Test Pressure (psig)
1	Dec, 1975	0.0466	0.075	47.2
1	Nov, 1978	0.0219	0.075	26.7
1	Feb, 1982	0.0156	0.075	26.7
2	Aug, 1984	0.043	0.075	47.4
1	Apr, 1985	0.053	0.075	48.1
1	May, 1988	0.023	0.075	47.8
2	Nov, 1993	0.0251	0.075	48.3
1	Apr, 1994	0.0429	0.075	49.3

These results are below the acceptance limit of 0.075 weight percent per day (0.75 L_a). The current as-left minimum path leak rate, including Type B and C leakage, is 0.069 L_a for Unit 1 and 0.060 L_a for Unit 2. Therefore, DCPP has maintained a significant margin between actual containment leakage and the TS 5.5.16 limit of 0.75 L_a. The testing history and structural capability of the containment have established that DCPP has had acceptable containment leakage rates with considerable margin and that the structural integrity of containment is assured. Due to the large margin available, and no identified mechanism that would cause significant degradation of containment, a 5 year extension of the ILRT interval would not be expected to result in containment leakage above the acceptable limit.

4.6 <u>Accident Analysis Impact</u>

The one-time increase from a ten-year interval to a fifteen-year interval for the Type A leakage rate test will not affect any accident parameters discussed in the DCPP FSAR Update. The Type A test interval does not affect the operation of any safety related equipment credited for accident mitigation. A change in the Type A test interval does not affect the containment initial conditions nor the leak rates that are assumed in the accident analyses. Therefore, the one-time increase from a ten-year interval to a fifteen-year interval for the Type A leakage rate test will not affect the accident analyses results in the DCPP FSAR Update.

4.7 <u>Defense-in-Depth</u>

Defense-in-depth is maintained by the robust containment design (which is not affected by the proposed change), on-going performance monitoring, and inspection activities.

The proposed change to the inspection interval does not have any effect on containment design margins or isolation system capability. The proposed change does not impact the initial conditions assumed in the safety analyses or the mitigation capability of the containment systems.

The results of the IWL examination have concluded the containment concrete was acceptable with no loss of structural capacity and no areas of the concrete shell have been classified as augmented examination areas. The results of the IWE containment liner inspections have not identified any significant degradations that could adversely impact the containment structural integrity or leak tightness, such as through-holes in the containment liner. The structural capability of the containment has been shown by the previous Type A testing results which have established that DCPP has had acceptable containment leakage rates with considerable margin.

On going performance monitoring and inspection of the containment liner and containment concrete provides assurance the containment structural and leak-tight integrity will be maintained in the absence of an ILRT within the fifteen year interval.

The containment structure is passive. Under normal operating conditions, there is no significant environmental or operational stress present that would contribute to its degradation. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests. No passive failures of containment have occurred at DCPP.

RG 1.174 recommends the use of risk analysis techniques to assist in demonstrating that the proposed change in the ILRT test interval is consistent with the defense-in-depth philosophy. To satisfy this recommendation, the change in the conditional containment failure probability (CCFP) was estimated. The estimated CCFPs for ILRT test intervals of 3-in-10 years, 1-in-10 years, and 1-in-15 years are 8.26E-1, 8.28E-1 and 8.29E-1, respectively. The change in the CCFP was estimated to be 0.001 for the change from the current ILRT interval of 1-in-10 years to the proposed one-time interval of 1-in-15 years. The change in the CCFP was estimated to be 0.003 for the cumulative change of going from the initial DCPP ILRT interval of 3-in-10 years to the proposed one-time interval of 1-in-15 years. It is concluded that the defense-in-depth is maintained based on the very small change in the CCFP for the proposed change.

Therefore, based on the DCPP containment design and past performance, on-going performance monitoring and inspection activities, and the very small impact on the RG 1.174 CCFP, it is concluded there is no reduction in defense-in-depth as a result of this change.

4.8 Risk Insights

Overall plant risk due to containment leakage is relatively small, given the small probability of containment leakage occurring. The predominant contributors to the risk (population dose) are containment bypass accident scenarios (i.e., steam generator tube rupture and interfacing system loss-of-coolant accident scenarios). These contributors would not be impacted by the proposed change in the ILRT interval.

The risk impact of containment structural life is measured by a pathway created for radionuclides if the containment is challenged such as in a loss-of-coolant accident or severe accident. Such leakage does not create any new accident scenarios, nor does it contribute to initiation of an accident.

The purpose of Appendix J leakage rate testing is to detect containment leakage resulting from failures in the containment isolation boundary before an accident occurs. Such leakage could be the result of leakage through containment penetrations, through airlocks, or through containment structural faults. The Appendix J Type B and C tests, which are unaffected by this proposed change, will continue to detect leakage through containment valves, penetrations, and airlocks as required. The only potential failures that would not be detected by Type B and C testing are mechanical failures of the containment shell (i.e., resulting from degradations or modifications to the containment shell). Inspections performed under ASME Section XI Subsections IWE and IWL are designed to detect such flaws.

The containment structure is passive. Under normal operating conditions, there is no significant environmental or operational stress present that would contribute to its degradation. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests. No passive failures of containment have occurred at DCPP.

The containment capacity was also analyzed as part of the Individual Plant Examination (IPE) study to determine the pressure retaining capability of the containment structure. Failure of containment is defined as leakage of containment atmosphere through the liner. The important postulated failure modes and leak areas with associated variability were estimated. The containment capacity analysis was then utilized in the development of the source terms. The risk analysis supporting this license amendment request uses the source term information to assess the public health (i.e., risk).

NUREG-1493 provides the generic technical bases for defining new containment leakage-testing requirements. These generic bases are developed partially based on a quantitative assessment of the risk impact of extending ILRT (and local leak rate test) test intervals for sample Light Water Reactor Power Plants. The NUREG-1493 results are generic in nature and are applicable to all power plants. Based on information provided in NUREG-1493, the potential increase in population dose attributable to the extension of the containment leakage rate Type A test interval would be extremely small. NUREG-1493 includes the results of a sensitivity study performed to explore the risk impact of several alternate leak rate test schedules. Alternative 6 from this study examines relaxing the ILRT interval from 3-in-10 years to 1-in-20 years. Using best estimate data, NUREG-1493 concludes that the increase in population exposure risk to

those in the vicinity of the five representative plants ranges from 0.02 to 0.16 percent. This low impact on risk is attributable to:

- ILRTs can potentially identify very small additional potential leakage paths that are not detectable by Type B and C tests;
- the insensitivity of risk to containment leak rate; and
- a low likelihood of ILRT identified leakage exceeding twice the allowable.

NUREG-1493 concludes that even increasing the ILRT interval to once per 20 years would "lead to an imperceptible increase in risk." The proposed extension of the test interval is bounded by the analyses of NUREG-1493. The associated increase in risk is not significant. An additional benefit in increasing the ILRT interval is to reduce occupational dose experienced by the test performers.

These risk insights are augmented by the plant specific quantitative analyses below.

4.9 Probabilistic Risk Assessment

PG&E has performed a risk assessment of the proposed one-time extension of the containment Type A leakage rate test interval from a ten-year interval to a fifteen-year interval. The risk assessment is performed in accordance with the guidelines set forth in NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Report TR-104285, and the NRC regulatory guidance on the use of PRA findings and risk insights in support of a licensee request for changes to a plant's licensing basis, discussed in RG 1.174.

The updated Level 1 internal events PRA model is used for this risk assessment calculation in combination with the IPE Level 2 model.

PG&E has developed and maintained a living PRA maintenance program to maintain a state-of-the-art PRA model that reflects the as-built and operated configuration. The DCPP PRA was originally reviewed by the NRC in 1988 and the model was found to be "beyond the state of the art" by the NRC. Additionally, the NRC has reviewed both the DCPP IPE and individual plant examination of external events submittals. A background and summary of the PRA model, including a description of the model updates, the model peer review, and the model quality was submitted to the NRC in PG&E letter DCL-01-015, dated February 16, 2001.

The change in plant risk associated with extending the Type A ILRT test interval was evaluated based on the change in the predicted man-rem/year frequency and the Large Early Release Frequency (LERF).

The Type A testing has no impact on the protection of the reactor core and therefore there is no impact on the core damage frequency (CDF) due to a change in the Type A testing interval.

The approach used for evaluating the change in risk associated with increasing the current 1-in-10 years intervals to 1-in-15 years intervals for Type A test is similar to that presented in EPRI TR-104285 and NUREG-1493. Namely, the analysis performed examined DCPP specific accident sequences in which the containment integrity remains intact or the containment is impaired.

The overall risk assessment approach was to:

- Develop a risk model for the accident classes defined in EPRI TR-104285 in terms of DCPP specific accident class frequency and bounding man-rem dose (population dose) estimates. The man-rem dose estimates from Indian Point Nuclear Generating Unit No. 3 (IP3) were used for DCPP. The IP3 man-rem estimates are contained in the Entergy Nuclear Operations, Inc. submittal for the IP3 ILRT extension request dated January 18, 2001 that was approved by the NRC on April 17, 2001. The IP3 man-rem estimates were evaluated by PG&E to be bounding and appropriate for the DCPP ILRT analysis (see Enclosure 4 for details).
- Quantify the risk for the original (3-in-10 years), the current (1-in-10 years), and the proposed (one-time extension to 1-in-15 years) Type A test interval. Although the Type A test interval change is from the current 1-in-10 years to a proposed 1-in-15 years, the risk evaluation considers the original Type A test interval of 3-in-10 years as a reference point (i.e. as a base line). The 3-in-10 years test interval is used as the base line point because the change from a 3-in-10 years interval to a 1-in-10 years interval was justified qualitatively based on the 10 CFR 50 Appendix J Option B rule change and was not supported by a quantitative risk evaluation. The information for the base line is summarized in Table 2.

Table 2
Baseline Mean Consequences - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Man-rem (50-Miles)	Man-rem/yr (50-Miles)
1	No Containment Failure	1.15E-06	1.41E+06	1.63E+00
2	Large IsoLation Failures (Failure-to-Close)	2.00E-07	4.94E+07	9.88E+00
3a	Small Pre-existing Liner Breach Failure	6.53E-07	1.41E+07	9.20E+00
3b	Large Pre-existing Liner Breach Failure	2.14E-07	4.94E+07	1.06E+01
4	Small isoLation Failure-to-Seal (Type B test)	N/A	N/A	N/A
5	Small isoLation Failure-to-Seal (Type C test)	N/A	N/A	N/A
6	Other isoLation Failures, e.g., Dependent Failures	2.07E-09	4.94E+07	1.02E-01
7	Failure Induced by Phenomena (Early and Late Failures)	7.82E-06	1.41E+08	1.10E+03
8	Containment Bypassed (SGTR & V-Sequence)	3.50E-07	5.33E+09	1.86E+03
CDF	All CET End States	1.04E-05	N/A	3.00E+03

N/A = Not Analyzed

- Evaluate the risk impact of extending the Type A test interval from the current 1-in-10 years interval to a 1-in-15 years interval. The risk impact was calculated in terms of the total risk (i.e. considering contribution from the large, small, late and early containment failures) as well as the change in the LERF figure of merit in accordance with RG 1.174. The details of the approach used, assumptions, and steps taken to perform the PRA analysis are discussed in Enclosure 4. The results of the DCPP risk assessment are presented in Table 3.

Table 3- Summary Risk Assessment Results

Case	Man-Rem Risk (per year)			LERF (per year)		Risk Impact (percent)		
	Total	Increase relative to base line (3-in-10)	Increase relative to current interval (1-in-10)	Increase relative to base line (3-in-10)	Increase relative to current interval (1-in-10)	% Cont. from Class 1 & 3 to Total Risk	Increase relative to base line (3-in-10)	Increase relative to current interval (1-in-10)
3-in-10	2.997E+3	-	-	-	-	0.71%	-	-
1-in-10	3.001E+3	4.10	-	N/A	-	0.84%	0.14%	-
1-in-15	3.002E+3	5.09	9.99E-1	2.57E-8	1.14E-8	0.87%	0.17%	0.03%

N/A = Not Analyzed

Based on the results it is concluded that the change in type A test interval from 1-in-10 years to 1-in-15 years increases the total integrated risk of the associated specific accident sequences by 0.03 percent. That is, the risk impact when compared to other severe accident induced risks is negligible. This is consistent with NUREG-1493 conclusions that even increasing the ILRT interval to 1-in-20 years would not have a significant impact on risk. Using the RG 1.174 LERF figure-of-merit criteria, the increase in LERF resulting from a change in the Type A test interval to 1-in-15 years is risk insignificant (low risk).

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed extension to the Type A testing interval from 1-in-10 years to 1-in-15 years will not increase the probability of an accident previously evaluated. The determination of containment integrity is not an accident

initiator. The containment Type A testing interval extension does not involve a plant modification and the testing interval extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. Research documented in NUREG-1493 has determined that Type B and C tests can identify the vast majority (approximately 97 percent) of all potential leakage paths. Experience at Diablo Canyon Power Plant (DCPP) demonstrates that excessive containment leakage paths are detected by Type B and C local leakage rate tests. Type B and C testing will identify any containment opening, such as a valve, that would otherwise be detected by the Type A tests.

NUREG-1493 concluded that increasing the Type A test interval to 1-in-20 years leads to an imperceptible increase in risk. A DCPP plant specific probabilistic risk assessment of the change in the Type A testing interval from 1-in-10 years to 1-in-15 years determined the total integrated risk of the associated specific accident sequences increases by 0.03 percent. This risk impact when compared to other severe accident induced risks is negligible. The increase in the Regulatory Guide (RG) 1.174 large early release fraction (LERF) figure-of-merit criteria resulting from a change in the Type A test interval from 1-in-10 years to 1-in-15 years is risk insignificant.

Testing and inspection provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The structural capability of the containment has been shown by the Type A testing results that have established that DCPP has had acceptable containment leakage rates with considerable margin. Inspections required by 10 CFR 50.65 and American Society of Mechanical Engineers code are performed in order to identify indications of containment degradation that could affect leak tightness. The results of containment concrete examination have concluded the containment concrete has had no loss of structural capacity and no areas of the concrete shell have experienced accelerated degradation or aging. The results of containment liner inspections have not identified any significant degradations that could adversely impact the containment structural integrity or leak tightness, such as through-holes in the containment liner. Due to the large containment leakage rate margin available, and no identified mechanism that would cause significant degradation of containment, a 5 year extension of the ILRT interval would not be expected to result in containment leakage above the acceptable limit.

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 extension of the Type A testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no physical changes being made to the plant and there are no changes in operation of the plant that could introduce a new failure mode, creating an accident.

The containment structure is passive. Under normal operating conditions, there is no significant environmental or operational stress present that would contribute to its degradation. Passive failures resulting in significant containment structural leakage are therefore extremely unlikely to develop between Type A tests.

Therefore, the proposed change 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 extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing results in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the increase in the Type A testing interval would have a minimal effect on this risk because 97 percent of the potential leakage paths are detected by Type B and C testing.

A DCPP plant specific probabilistic risk assessment of the change in the Type A testing interval from 1-in-10 years to 1-in-15 years determined the total integrated risk of the associated specific accident sequences increases by 0.03 percent. This risk impact when compared to other severe accident induced risks is negligible. The increase in RG 1.174 LERF figure-of-merit criteria resulting from a change in the Type A test interval from 1-in-10 years to 1-in-15 years is risk insignificant.

Deferral of Type A testing for DCPP does not increase the level of risk to the public due to loss of capability to detect and measure containment leakage or loss of containment structural capability. Other containment testing methods and inspections will assure all limiting conditions of operation will continue to be met. The margin of safety inherent in existing accident analyses is 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 amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TSs. Limiting containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident. 10 CFR 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." DCPP previously selected Option B. RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Exceptions to the requirements of RG 1.163 are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states:

"The Regulatory Guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide."

Therefore, the change proposed by this application does not require an exemption from Option B of 10 CFR 50, Appendix J. The incorporation of the 15-year interval for the Type A containment leakage rate test, which is

an exception to the 10-year Type A containment leakage rate test required by NEI 94-01, into the TS satisfies the requirements of 10 CFR 50, Appendix J, Option B, Section V.B.

The change in the Type A test interval from a 1-in-10 years to a 1-in-15 years interval has no impact on the CDF. The change in LERF is not considered a significant change as defined by the RG 1.174. Therefore the change meets the RG 1.174 risk guidelines.

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

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 REFERENCES

1. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.
2. Nuclear Energy Institute (NEI) Report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995.
3. American National Standard ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements."
4. NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995.
5. American Society of Mechanical Engineers (ASME) Section XI code, Sections IWE and IWL, Reactor Building Containment Inspections.
6. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998.

7. PG&E Letter DCL-00-075, "Inservice Inspection Relief Requests for Containment," dated May 11, 2000.
8. NRC Letter "Diablo Canyon Power Plant Units 1 and 2 - Safety Evaluation For Inservice Inspection Relief Requests - #CNT-E1, #CNT-E2, #CNT-E3, #CNT-E4, #CNT-E5, #CNT-L1 Regarding Containment Examination Requirements JAC Nos. MA8630 and MA8631," dated September 28, 2000.
9. Electric Power Research Institute (EPRI) Report EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.
10. PG&E Letter DCL-01-015, "Relief Request for Application of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds," dated February 16, 2001.
11. New York Power Authority Letter IPN-00-062, "Proposed Change to Section 6.14 of the Administrative Section of the Technical Specifications," dated September 6, 2000.
12. Entergy Nuclear Operations, Inc., Letter IPN-01-007, "Indian Point Unit 3 Nuclear Power Plant, Docket No. 50-286 License No. DPR-64, Supplemental Information Regarding Proposed Change to Section 6.1.4 of the Administrative Section of the Technical Specifications," dated January 18, 2001, including Attachment II to IPN-01-007, Calculation IP3-CALC-VC-03357, and "Risk Impact Assessment of Extending Containment Type A Test Interval."
13. NRC Letter "Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment RE: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MB0178)," dated April 17, 2001.
14. NUREG-1335, "Individual Plant Examination: Submittal Guidance," dated August 1989.
15. NUREG/CR-4551, "Evaluation of Severe Accidents Risks: Zion Unit 1," Brookhaven National Laboratory, Volume 7, Revision 7, dated October 1990.
16. PG&E Letter for Response to Generic Letter 88-20, Individual Plant Examination, "Individual Plant Examination Report for Diablo Canyon Power Plant Units 1 and 2," dated April 1992.

MARKED-UP TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
5.0-24	5.0-24

5.5 Programs and Manuals

5.5.15 <u>Safety Function Determination Program (SFDP)</u> (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 <u>Containment Leakage Rate Testing Program</u>

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

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

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

d. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

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

   b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

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

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

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.

PROPOSED TECHNICAL SPECIFICATIONS PAGE

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995." The ten-year interval between performance of the integrated leakage rate (Type A) test, beginning May 4, 1994, for Unit 1 and April 30, 1993, for Unit 2, has been extended to 15 years.

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

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

d. Leakage rate acceptance criteria are:

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

   2. Air lock testing acceptance criteria are:

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

      b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

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

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

PROBABALISTIC RISK ASSESSMENT CALCULATION TO SUPPORT CHANGE IN 10 CFR 50 APPENDIX J INTEGRATED LEAK RATE TEST INTERVAL

PACIFIC GAS & ELECTRIC COMPANY

NUCLEAR SAFETY ASSESSMENT AND LICENSING

PROBABILISTIC RISK ASSESSMENT

CALCULATION FILE NO. **PRA01-07 Revision 1**

SUBJECT: Risk Impact Assessment of Extending Containment Type A Test Interval

PREPARED BY: ______________________ DATE: 10/11/01
A. Afzali

VERIFIED BY: KAHubbard DATE: 10/11/01
K.A. Hubbard

VERIFIED IN ACCORDANCE WITH: CF3.ID15

APPROVED BY: ______________________ DATE: 10/12/01

This file contains: 32 pages total
26 pages text
6-page attachment (including cover page)

0.0 RECORD OF REVISION

REV. 0 Original calculation.

REV. 1 Several editorial changes were made to this calculation file.

1.0 PURPOSE

Provide a risk impact assessment on extending the plant's integrated leak rate test (ILRT) interval from 10 to 15 years.

2.0 INTRODUCTION

In October 26, 1995, the NRC revised 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The revision allowed individual plants to select containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Diablo Canyon Power Plant (DCPP) Units 1 and 2 selected the requirements under Option B as its testing program.

The surveillance testing requirements as proposed in NEI 94-01 [Ref. 1] for Type A testing is at least once per 10 years, based on an acceptable performance history (defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than 1.01-L_a).

The Unit 1 current 10-year Type A test is due to be performed during refueling outage 1R12, in 2004. The Unit 2 Type A test is scheduled for 2R11, in 2003.

This calculation provides a risk impact assessment on extending the plant's ILRT interval from 10 to 15 years. The risk assessment is performed in accordance with the guidelines set forth in NEI 94-01 [Ref. 1], the methodology used in EPRI TR-104285 [Ref. 2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, RG 1.174 [Ref. 3].

The updated PRA model [Ref. 4] is used for this risk assessment calculation in combination with the IPE Level 2 Model [Ref. 5]. This combination of the models ensures the fidelity of the core damage model.

It should be noted that attempts have been made to perform this calculation and present the results in a similar manner to the Indian Point 3 (IP3) ILRT submittal [Ref. 6], to ensure consistency with an approved change request.

3.0 ASSUMPTIONS AND CALCULATIONS

3.1 Assumptions

Consistent with the IP3 ILRT submittal [Ref. 6], the following maximum leakage rates are assumed:

- The maximum containment leakage for Class-1 sequences is 2 L_a.
- The maximum containment leakage for Class 2 sequences is 35 L_a.
- The maximum containment leakage for Class-3a sequences is 10 L_a.
- The maximum containment leakage for Class-3b sequences is 35 L_a.

- The maximum containment leakage for Class 6 sequences is 35 L_a.
- The maximum containment leakage for Class 7 sequences is 100 L_a.

Class 8 sequences are containment bypass sequences. Therefore, the potential releases are directly to the environment and the containment structure will not impact the release magnitude.

All RISKMAN calculations were performed with RISKMAN Release 3.00 on PC S/N B4T6H (PC0000090625), which has been certified for use under procedure IDAP CF2.ID2 as documented "RISKMAN for Windows and DOS SQA Plan."

3.2 Calculation

A simplified bounding approach for evaluating the change in risk associated with increasing the interval from 10 to 15 years for the Type A test was used. This approach is similar to that presented in EPRI TR-104285 [Ref. 2] and NUREG-1493 [Ref. 8]. Namely, the analysis performed examined DCPP-specific accident sequences in which containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class-1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or steam generator manway leakage. (EPRI TR-104285 Class-3 sequences)
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left 'opened' following a plant post-maintenance test. For example, a valve failing to close following a valve stroke test. (EPRI TR-104285 Class 6 sequences)
- Accident sequences involving containment failure induced by severe accident phenomena (EPRI TR-104285 Class 7 sequences), containment bypassed (EPRI TR-104285 Class 8 sequences), or large containment isolation failures (EPRI TR-104285 Class 2 sequences). Small containment isolation 'failure-to-seal' events (EPRI TR-104285 Class 4 and 5 sequences) were not evaluated in this analysis. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test intervals.

The steps taken to perform this risk assessment are as follows:

Step 1 -- Quantify the baseline risk in terms of frequency per reactor year for six of the eight accident classes presented in Table 2 (accident classes 4 and 5 are not evaluated).

Step 2 -- Develop man-rem dose (population dose) per reactor year for the six accident classes evaluated.

Step 3 -- Evaluate the risk impact of extending the Type A test interval from 10 to 15 years.

Step 4 -- Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [Ref. 3].

Step 1 - Quantify the baseline risk in terms of frequency per reactor year

This step involves the review of the DCPP IPE [Ref. 5] containment event tree (CET). The CET characterizes the response of the containment to important severe accident sequences. The CET used in this evaluation is based on important phenomena and systems-related events identified in NUREG-1335 [Ref. 9], NUREG/CR-4550 [Ref. 10], and on plant features that influence the phenomena.

As previously described, the extension of the Type A interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena. As a result, the CET containment isolation model was reviewed for applicable isolation failures and their impact on the overall plant risk. Specifically, the results of the IPE Level 2 analysis and the current Level 1 model are used to predict the likelihood of having a small/large breach in the containment liner that is undetected by the Type A ILRT.

The containment isolation model found in Reference 4 examined the five issues associated with containment isolation in NUREG-1335 [9]: (1) the identity of pathways that could significantly contribute to containment isolation failure, (2) the signals required to automatically isolate the containment penetration, (3) the potential generating signals for all initiating events, (4) the examination of testing and maintenance procedures, and (5) the quantification of each containment isolation mode. These issues are addressed as follows:

(1) <u>Pathways that could significantly contribute to containment isolation failure</u>
Significant fission product release to the environment may occur through containment penetrations from drain lines from sumps inside containment that are ultimately routed into the primary auxiliary building and piping that communicates directly with the containment atmosphere. The following piping elements are not considered in this analysis:

- Piping that communicates directly with the Reactor Coolant System (RCS) is not considered in the containment isolation failure analysis. Such failures are considered to be failures of the pressure boundary between the RCS and low-pressure systems (i.e., an interfacing system LOCA).

- For the LERF calculation, non-isolated penetrations that have a potential equivalent flow area smaller than a 3-inch diameter hole were excluded. The rationale for this exclusion is that containment leakage through smaller diameter piping will not preclude further containment pressurization, and, in any case, any release of fission products from such sources will be small. Per DCPP PRA Calculation File N.1 [Ref. 11], the minimum size for a hole in DCPP containment, which allows one containment volume change in one hour at a constant design pressure, is 3 inches.

- Containment penetrations that are not used and are closed during power operation are not included. Exceptions include certain large containment lines (e.g., containment purge lines) and steam generator blowdown and blowdown sample lines. These lines are periodically opened during plant operation. In addition, preexisting containment leaks are also considered in this analysis.

- Penetration lines that have no direct contact with the RCS or the containment atmosphere are not included.

- Penetration lines that are required for safety functions, and are therefore not isolated, are not included.

- Penetration lines that have a negligible failure frequency; e.g., multiple failures of three or more valves are required to fail the isolation of a penetration.

In the DCPP PRA, the containment isolation function is represented by Top Events WL (automatic isolation of the common drain line of the reactor cavity sump and containment sumps); CP (automatic isolation of the containment pressure and vacuum relief line and the containment purge lines); CI (automatic isolation of the containment penetrations that connect to the RCS or containment atmosphere, that are permitted to be open during power operation, that have a flow area less than an equivalent 3-inch diameter hole, and that are not already included in Top Event WL.); and IV (isolation of a ruptured steam generator [SGTR initiating event]). Top Event IV is only questioned for the SGTR initiating event and its contribution is covered by the non-isolated SGTR core damage scenarios. Top event WL is mainly addressed from the core damage point of view. Its impact on the containment isolation probability is addressed via sequences where top event OI (manual isolation of containment) has failed. A full description of the pertinent containment isolation top events is presented below.

- Top Event WL -- Automatic isolation of the common drain line of the reactor cavity sump and containment sumps. Success of this top event requires at least one isolation valve in the containment structure sump pump discharge line to close, if both are open prior to the initiating event, and to remain closed for 24 hours after the initiating event. This would ensure an adequate recirculation sump water level following a LOCA. If isolation fails, water transfer to the floor drain receiver tanks may result from operation of the containment sump pumps and as a result of high containment pressure. The PRA model in such circumstances assumes that the water level in the containment is not sufficient to allow recirculation from the sump. The sump drain line would eventually clear of water and provide a release path from the containment.

- Top Event CP -- Automatic isolation of the containment pressure and vacuum relief line and the containment purge lines. These are the only containment penetrations that connect to the containment atmosphere, are permitted to be open during power operation, and have an equivalent flow area more than a 3-inch diameter hole. Also included in this top event is the large opening (more than 3 inches) in the containment (basic event CPLEK), due to valve misalignment, that will not be isolated by the containment isolation signal. Top Event CP is successful if no misalignment is present, and if the lines are closed initially or if one or more lines are open and at least one isolation valve in each open line closes automatically on a Phase A or containment ventilation isolation signal.

- Top Event CI -- Automatic isolation of the containment penetrations that connect to the RCS or containment atmosphere, that are permitted to be open during power operation, that have a flow area less than an equivalent 3-inch diameter hole, and that are not already included in Top Event WL. The five penetrations considered in this top event are:

  - RCP Seal Water Return and Letdown Line
  - Reactor Coolant Drain Tank (RCDT) Discharge Line
  - RCDT Vent Header

- o RCDT Nitrogen Supply Line
- o Pressurizer Relief Tank (PRT) Nitrogen Supply Line

These penetration lines are isolated by a Phase A containment isolation signal if open initially. Success of Top Event CI requires that at least one isolation valve in each of the five lines is closed. Also included in this top event are the small containment leaks (less than 3 inches) (basic event CILEAK) due to valve misalignment that do not isolate on containment isolation signal.

If automatic isolation (Top Event CI, CP, or WL) fails, the operator action to isolate the containment is modeled in Top Event OI. In the DCPP PRA model, however, credit was not given to operator action to isolate the containment when top event CP failed.

Figures 1 and 2 present the fault trees for top events CP and CI, respectively. The support system dependencies are not included in the fault trees because the DCPP PRA model uses the large event tree/small fault tree approach. The support system dependencies are addressed in the pertinent event trees by using the appropriate split fraction for a given support system configuration.

The following changes have been made to the CP and CI split fraction constructions:

- The basic event impact CPLEK = S is added to the construction equation for split fractions CP1 through CP6. This condition is added to exclude the contribution of the misalignment mechanism from the total failure probability of the above split fractions. The misalignment, which is used to calculate Class 6 frequency, is addressed outside the PRA model (to be consistent with IP3 submittal).

- The basic event impact CILEAK = S is added to the construction equation for split fractions CI1 through CI6. This condition is added to exclude the contribution of the misalignment mechanism from the total failure probability of the above split fractions. The misalignment, which is used to calculate Class 6 frequency, is addressed outside the PRA model.

(2&3) <u>Signals required to automatically isolate the containment penetration and potential generating signals for all initiating events</u>
Containment isolation signals, including those generated by unique plant initiators, required to automatically isolate the containment penetration, are included in the PRA model.

(4) <u>The examination of testing and maintenance procedures</u>
Failures attributed to valve test and maintenance procedures are represented in the DCPP PRA model by the CPLEK and CILEAK basic events. These basic events are excluded from the PRA model and their contributions to Class 6 sequences were evaluated outside the PRA model as discussed below.

(5) <u>The quantification of each containment isolation mode</u>
The containment isolation fault tree models all pertinent containment isolation modes.

For this analysis, and consistent with the IP3 submittal [Ref. 6], the probability of a liner breach (due to excessive leakage) at the time of core damage is considered. As stated earlier, these are Event Class-3A (small liner breach) and Event Class-3B (large liner breach). These events model the "Class-3" sequence depicted in EPRI TR-104285 [Ref. 2]. Again, based on the IP3

submittal, the probabilities for Class-3A and Class-3B are estimated to be 0.021 and 0.064, respectively, and are described in greater details below under Class-3 sequences.

Calculation of Release Frequencies

As stated earlier, the PRA model is used to calculate release frequencies. The current Level 1 PRA model, in combination with the IPE Level 2 model, is used in this calculation. The Level 1 model is used to calculate plant damage state frequencies. The Level 2 model is used to address the physical progression of plant damage states from the onset of core damage through potential release of radionuclides from containment. Figure 3 presents the Level 2 event tree. Since the DCPP PRA model is a large event tree/small fault tree model, the Level 2 event tree model is an extensive tree and includes all potential consequences. The Level 2 results are summarized in Table 1 (See Attachment 1 for the Riskman printouts).

Table 1 - Summary Results for Release Frequency Update

Description of Release Category Group	Updated Frequency (per year)
Small, Early Containment Failures	3.22E-07
Large, Early Containment Failures	2.71E-07
Late Containment Failures	7.43E-06
Containment Bypass	3.50E-07
Long Term Containment Intact	2.02E-06
Total	1.04E-05

For this analysis, the LERF figure of merit is calculated as follows:

LERF = Freq. of "Large, Early Containment Failures" + Frequency of "Containment Bypass"
LERF = 2.71E-07 + 3.50E-07 = 6.20E-07 per year

Class-1 Sequences

This group consists of all core damage accident progression bins for which containment remains intact. From Table 1, Class-1 frequency is 2.02E-06 per year. However, since the IPE model did not include containment isolation failure modes covered by Classes 3A and 3B, and contributor to Class 6 (basic event CPLEK) is taken out for this calculation and a more accurate Class-1 frequency is calculated by:

CLASS-1-FREQ = "IPE Long Term Containment Intact" frequency - ($FREQ_{Class3A}$ + $FREQ_{Class3B}$ + $FREQ_{Class6}$)

CLASS-1-FREQ = 2.02E-06 - (1.02E-05*0.064 + 1.02E-05*0.021 + 2.031E-04*1.02E-5) = 1.15E-06 per year

CLASS-1-FREQ = 1.15E-06/year

Class 2 Sequences

This group consists of all core damage accident progression bins for which a pre-existing leakage occurs due to failure to isolate the containment in response to an accident. These sequences are dominated by failure-to-close of large (>3-inch diameter) containment isolation valves. In the IP3 submittal, where the PRA model is based on the large fault tree/small event tree approach, this frequency is calculated by:

CLASS-2-FREQUENCY = $PROB_{Large\ CI}$ * CDF

Where: $PROB_{Large\ CI}$ = random large containment isolation failure probability (i.e., large valves)

For DCPP, the frequency can be directly calculated by running the model. Riskman code allows for creation of groups that sum up the contribution from all sequences involving a particular failure event or set of failure events. For the calculation of frequency of the Class 2 sequences, the failure sequences of interest are those involving the containment isolation valve (>3-inch in diameter) failures (not including pre-existing holes or misalignments.) Therefore, group CPS is created where top event CP = S. This gives the core damage frequency (CDF) estimate for sequences where containment isolation has been successful. Subtracting the total CDF from the CPS CDF gives the CDF estimate where large containment isolation valves have failed to isolate. Therefore:

Internal Events CDF = 1.04E-05 per year (C.9, Rev. 9 calculation file)
Internal Events CDF with CP = S = 1.02E-05 per year (ILRT Model)
CLASS - 2 - FREQUENCY = CPF Group Frequency = 1.04E-05 – 1.02E-05 = 2.0E-07
CLASS-2-FREQUENCY = 2.0E-07/year

The associated maximum containment leakage for this Class is 35 L_a.

Class-3 Sequences

This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (i.e., containment liner) exists. The containment leakage for these sequences can be either small (2 L_a to 35 L_a) or Large (>35 L_a).

Using the IP3 approach (which is reported here for the documentation sake), to calculate the probability that a liner leak will be large (Event Class-3B), use was made of the data presented in NUREG-1493 [Ref. 8]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Since 21 L_a does not constitute a large release (refer to the writeup in Step 4), no large releases have occurred based on the 144 ILRTs reported in NUREG-1493.

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the ?^{2} distribution. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [Ref. 12]. The χ^2 distribution is really a family of distributions, which range in shape from that of the exponential to that of the normal distribution. Each distribution is identified by the degrees of freedom, v. For time-truncated tests (versus failure-truncated tests), an estimate of the probability of a large leak using the χ^2 distribution can be calculated as χ^2_{95} (v = 2n+2)/2N, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and χ^2_{95} (2) = 5.99, the 95th percentile estimate of the probability of a large leak is calculated as 5.99/(2*144) = 0.021.

To calculate the probability that a liner leak will be small (Event Class-3A), use was made of the data presented in NUREG-1493. The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of 1.0 L_a.

However, of these 23 'failures', only 4 were found by an ILRT; the others were found by Types B and C testing or errors in test alignments. Therefore, the number of failures considered for "small releases" are 4 out of 144. Similar to the event Class-3B probability, the estimated failure probability for small release is found by using the χ^2 distribution. The χ^2 distribution is calculated by n=4 (number of small leaks) and N=144 (number of events) which yields a $\chi^2(10) = 18.3070$. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070/(2*144) = 0.064.

The respective frequencies per year are determined as follows:

CLASS-3A-FREQUENCY = $PROB_{Class\text{-}3A}$ * CDF
CLASS-3B-FREQUENCY = $PROB_{Class\text{-}3B}$ * CDF

Where:
$PROB_{Class\text{-}3A}$ = probability of small pre-existing containment liner leakage
= 0.064

$PROB_{Class\text{-}3B}$ = probability of large pre-existing containment liner leakage
= 0.021

Note that the CDF in this case should only include sequences where top event CP has been successful because core damage sequences that include failure of top event CP are included in Class-2 frequency. Therefore,

CLASS-3A-Frequency = (Group CPS CDF) * 0.064
CLASS-3B-Frequency = (Group CPS CDF) * 0.021

CLASS-3A-FREQUENCY = 0.064 * 1.02E-05/year = 6.53E-07/year
CLASS-3B-FREQUENCY = 0.021 * 1.02E-05/year = 2.14E-07/year

For this analysis the associated maximum containment leakage for Class-3A is 10 L_a and for Class-3B is 35 L_a.

Class 4 Sequences
This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type B test components occurs. Because these failures are detected by Type B tests, this group is not evaluated any further, consistent with approved methodology.

Class 5 Sequences
This group consists of all core damage accident progression bins for which a failure-to-seal containment isolation failure of Type C test components occurs. Because these failures are detected by Type C tests, this group is not evaluated any further, consistent with approved methodology.

Class 6 Sequences
This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. The IP3 valve misalignment failure probability [2.031E-04] was used here for Class 6 frequency estimation.

The frequency per year for these sequences is determined as follows:

CLASS-6-FREQUENCY = PROB $_{LargeT\&M}$ * (Group CPS CDF)

Where:
PROB $_{LargeT\&M}$ = random Large containment isolation failure probability due to valve misalignment
= 2.031E-04 [Ref. 6]

CLASS-6-FREQUENCY = 2.031E-04 * 1.02E-05/year = 2.07E-09/year

For this analysis the associated maximum containment leakage for this group is 35 L_a.

Note that in the DCPP IPE model, basic event CPLEK, with the probability of 2.35E-07, represents the valve misalignment failure mechanism. However, conservatively, the contribution of basic event CPLEK is excluded and the IP3 estimate of valve misalignment probability is used in this analysis.

Class 7 Sequences
This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena (Early and Late Failures). For this analysis the associated maximum containment leakage for this group is 35 L_a.

CLASS-7-FREQUENCY = TOT_CFL + CFE

Where:
TOT_CFL = total late containment failure frequency = 7.43E-06/year [Table 1]

CFE = early containment failure frequency (excluding bypass failures)
= TOT-CFE - CI

Where:
TOT-CFE = total early contnmt failure freq = 3.22E-07 + 2.71E-07 = 5.92E-07/yr [Table 1]

CI = Freq. Of Class 2 Frequency

(Note that Class-3A, Class-3B, and Class-6 are also early containment failures, but since their contribution is calculated outside the model, they do not need to be subtracted from the Class-7 model).

CI = 2.0E-07 per year

Therefore:

CFE = 5.92E-07 – 2.00E-07 = 3.92E-07/year

CLASS-7-FREQUENCY = 7.43E-06 + 3.92E-07 = 7.82E-06/year

CLASS-7-FREQUENCY = 7.82E-06/year

Note that in the IP3 analysis, early failures are not broken down to small or large failures. The DCPP model considers both types of failures modes. The inclusion of the "Small, Early Containment Failure" in TOT-CFE, hence Class-7 sequences, is conservative since the maximum containment leakage for Class-7 sequences is 100 L_a, whereas the leakage for "Small Containment Failures" is 10 L_a (same as Class-3A). One approach to address this issue is to create a sub category of Class 7 sequences (for small early failures). However, the results

are insensitive to the inclusion of the "Small, Early Containment Failure" in Class 7. Therefore, to stay consistent with the IP3 analysis, Class 7 sequences also include the small containment failure group of sequences.

Class 8 Sequences
This group consists of all core damage accident progression bins in which containment bypass occurs.

CLASS-8-FREQUENCY = CONT-BYPASS Frequency = 3.50E-07/year [Table 1]

Note: For this class, the maximum release is not based on normal containment leakage, because the releases are released directly to the environment. Therefore, the containment structure will not impact the release magnitude.

Table 2
Mean Containment Frequencies Measures - Given Accident Class

Class	Description	Frequency (per Rx-year)
1	No Containment Failure	1.15E-06
2	Large Containment Isolation Failures (Failure-to-close)	2.00E-07
3A	Small Isolation Failures (Liner breach)	6.53E-07
3B	Large Isolation Failures (Liner Breach)	2.14E-07
4	Small Isolation Failure - Failure-to-Seal (Type B test)	Not Analyzed
5	Small Isolation Failure - Failure-to-Seal (Type C test)	Not Analyzed
6	Containment Isolation Failures (dependent failures, personnel errors)	2.07E-09
7	Severe Accident Phenomena Induced Failure (Early and Late Failures)	7.82E-06
8	Containment Bypassed (SGTR & V-Sequence, induced SGTR)	3.50E-07
Core Damage	Core Damage -- All CET End states	1.04E-05

Step 2 - Develop plant specific man-rem dose (population dose) per reactor year

To simplify the overall approach, the total dose can be calculated by:

Total Dose = $\Sigma F_i * D_i$

Where F_i is the frequency of event class i and D_i is the population dose for the event class.

The F_i term is calculated above and is summarized in Table 2. The D_i term requires a Level III analysis, which uses the Level II results and other parameters to calculate the population dose. The D_i term is a combination of the source term category definitions (defined in the Level II analysis) and other parameters such as power rating, demographic and physical characteristics of the site and the surrounding area. Since DCPP does not have a Level III analysis, the D_i is calculated by using the dose estimates and approach that are provided in IP3 ILRT one-time exemption request. This approach is expected to yield bounding results since:

- Total population in the vicinity of DCPP is considerably smaller than at IP3. The overall population of the area around the plant (within 50 miles) is estimated to be approximately 500,000 people. The plant area around IP3 supports a population in the range of 15 million. This, in turn, translates into a substantially higher population dose to the general population.

- The meteorological data for the two sites are different. However, the total effects of the release for DCPP will be similar to IP3. This is based on the consideration that the dispersion and deposition of the releases within the area of concern (out to a 50 mile radius) is sufficiently large to assure that the overall transport and deposition of radionuclides would be comparable.

- Physically, the presence of relatively large populations in the immediate vicinity of the plant (within approximately 7 miles) yields a proportionally higher dose to the public than the population residing in areas further removed from the site. Since DCPP has essentially no population residing within this zone, the net effect of adapting the IP3 results will yield even more conservative results for DCPP.

- DCPP environs are primarily fairly low population zones, with the principal uses being agricultural and recreational. Extrapolation based on the IP3 analysis will result in extremely conservative estimates. IP3 has a lot of inhabited areas. It is expected that agricultural impacts will be somewhat different – and it is difficult to assess effects on the food chain such as ingestion through milk and produce. However, because of the higher population around IP3, it is judged that the aggregate doses of the IP3 will bound the aggregate doses of Diablo Canyon.

Plant and Source Terms for both plants are expected to be consistent and comparable on the basis that:

- Both plants are Westinghouse 4-loop PWRs.

- Both units operate at similar power levels; IP3 runs a little lower (3025 versus 3411 for the DCPP units) – the difference will result in slightly smaller source terms for IP3, but the source terms are sufficiently close for assessment purposes. An overall difference of 13% does not affect the validity of the source term selected. The nature and composition of the source terms are expected to be very similar since the design and physical capabilities of the plants are very similar.

- Containment structures are very similar in terms of volume and overall dimensions.

- Containment structural limits are very similar (design pressures).

Therefore, it is judged that the IP3 results are limiting for DCPP. The differences between the overall populations surrounding the plants will assure that the IP3 results are limiting for DCPP in terms of aggregate doses.

Note that this judgment is consistent with the EPRI TR-104285 assertion that "in as much as the comparison is made relative to a baseline, the differences [in constituents of the population dose calculation] not considered in the analysis, would not impact the conclusions drawn."

IP3 reports 1.41E+06 man-rem and 5.33E+09 man-rem for the population dose taken out to 50 miles for the design-basis normal containment leak rate of 0.1% and containment bypass, respectively. Therefore, using the same methodology as presented in the IP3 submittal, the man-rem for accident classes 1 to 8 are as follows:

Class-1 = 1.41E+06 x 1.0 L_a = 1.41E+06 man-rem
Class-2 = 1.41E+06 x 35 L_a = 4.94E+07 man-rem
Class-3A = 1.41E+06 x 10 L_a = 1.41E+07 man-rem
Class-3B = 1.41E+06 x 35 L_a = 4.94E+07 man-rem
Class-4 = Not analyzed
Class-5 = Not analyzed
Class-6 = 1.41E+06 x 35 L_a = 4.94E+07 man-rem
Class-7 = 1.41E+06 x 100 L_a = 1.41E+08 man-rem
Class-8 = 5.33E+09

The above values are summarized in Table 3 below.

Table 3
Man-Rem Measures - Given Accident Class

Class	Description	Man-Rem (50-Miles)
1	No Containment Failure	1.41E+06
2	Large Containment Isolation Failures (Failure-to-close)	4.94E+07
3A	Small Isolation Failures (Liner breach)	1.41E+07
3B	Small Isolation Failures (Liner breach)	4.94E+07
4	Small Isolation Failure - failure-to-seal (Type B test)	N/A
5	Small Isolation Failure - failure-to-seal (Type C test)	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	4.94E+07
7	Failure Induced by Phenomena (Early and Late Failures)	1.41E+08
8	Containment Bypassed (SGTR & V-Sequence)	5.33E+09

The above results, when combined with the results presented in Table 2, yield the DCPP baseline mean consequence measures for each accident class. These results are presented in Table 4 below.

Table 4
Baseline Mean Consequences Measures - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Man-rem (50-Miles)	Man-rem/yr (50-Miles)
1	No Containment Failure	1.15E-06	1.41E+06	1.63E+00
2	Large Isolation Failures (Failure-to-close)	2.00E-07	4.94E+07	9.88E+00
3A	Small Pre-existing Liner Breach Failure	6.53E-07	1.41E+07	9.20E+00
3B	Large Pre-existing Liner Breach Failure	2.14E-07	4.94E+07	1.06E+01
4	Small Isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small Isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	2.07E-09	4.94E+07	1.02E-01
7	Failure Induced by Phenomena (Early and Late Failures)	7.82E-06	1.41E+08	1.10E+03
8	Containment Bypassed (SGTR & V-Sequence)	3.50E-07	5.33E+09	1.86E+03
CDF	All CET End States	1.04E-05	N/A	2.9973E+03

Based on the above values, the percent risk contribution ($\%Risk_{BASE}$) for Class-1 and Class-3 is as follows:

$$\%Risk_{BASE} = [(CLASS1_{BASE} + CLASS3A_{BASE} + CLASS3B_{BASE}) / Total_{BASE}] \times 100$$

Where:
$CLASS1_{BASE}$ = Class-1 man-rem/year = 1.63 man-rem/year [Table 4]

$CLASS3A_{BASE}$ = Class-3A man-rem/year = 9.20 man-rem/year [Table 4]

$CLASS3B_{BASE}$ = Class-3B man-rem/year = 10.6 man-rem/year [Table 4]

$Total_{BASE}$ = total man-rem year for baseline interval = 2.9973E+03 man-rem/year [Table 4]

$$\%Risk_{BASE} = [21.4 / 2.9973E+03] \times 100\%$$
$$\%Risk_{BASE} = 0.71\%$$

Therefore, the total baseline risk contribution of leakage, represented by Class-1 and Class-3 accident scenarios is 0.71%.

<u>Step 3 - Evaluate risk impact of extending Type A test interval from 10 to 15 years</u>

According to NUREG-1493 [Ref. 8], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10 years will increase the average time that a leak detectable only by an ILRT goes undetected from 18 to 60 months. (The average time for not detecting is calculated by multiplying the test interval by ½ and multiplying by 12 to convert from "years" to "months".) If the test interval is extended to 1-in-15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 90 months (1/2 * 15 * 12). Since ILRTs only detect

about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% increase in the overall probability of leakage. This value is determined by multiplying 3% and the ratio of the average time for not detecting for the increased ILRT test interval (60 months) to the baseline average time for not detecting of 18 months. For a 15-yr test interval, the result is a 15% increase in the overall probability of leakage (i.e., 3 * 90/18). Thus, increasing the ILRT test interval from 10 to 15 years results in a 5% increase in the overall probability of leakage.

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class-1 and Class-3 sequences. In addition, the increased probability of not detecting excessive leakage has no impact on the frequency of occurrence for Class-1 sequences. Therefore, for Class-1 sequences, to determine the risk contribution of leakage for a 10-year test interval, the man-rem/year results for Class-1 sequences are multiplied by the increase in overall probability of leakage (10% or 1.1) times 2 L_a. For Class-3 sequences, the release magnitude is not impacted by the change in test interval, (a small or large liner opening remains the same, even though the probability of not detecting the liner opening increases). Thus, only the frequency of Class-3 sequences is impacted. Therefore, for Class-3 sequences, the risk contribution is determined by multiplying the Class-3 accident frequency by the increase in probability of leakage of 1.1. (Recall that for a 10-year interval, there is a 10% increase on the overall probability of leakage.) The results of this calculation are presented in Table 5.

Table 5
Mean Consequence Measures for 10-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Man-rem (50-Miles)	Man-rem/yr (50-Miles)
1	No Containment Failure	1.07E-06	3.10E+06	3.31E+00
2	Large Isolation Failures (Failure-to-close)	2.00E-07	4.94E+07	9.88E+00
3A	Small Pre-existing Liner Breach Failure	7.18E-07	1.41E+07	1.01E+01
3B	Large Pre-existing Liner Breach Failure	2.36E-07	4.94E+07	1.16E+01
4	Small Isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small Isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	2.07E-09	4.94E+07	1.02E-01
7	Failure Induced by Phenomena (Early and Late Failures)	7.82E-06	1.41E+08	1.10E+03
8	Bypass (SGTR)	3.50E-07	5.33E+09	1.86E+03
CDF	All CET End States	1.04E-05	N/A	3.0014E+3

Based on the above values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class-1 and Class-3 is as follows:

$\%Risk_{10} = [(CLASS1_{10} + CLASS3A_{10} + CLASS3B_{10}) / Total_{10}] \times 100$

Where:
$CLASS1_{10}$ = Class-1 man-rem/year = 3.31E+00 man-rem/year [Table 5]

$CLASS3A_{10}$ = Class-3A man-rem/year = 1.01E+01 man-rem/year [Table 5]

$CLASS3B_{10}$ = Class-3B man-rem/year = 1.16E+01 man-rem/year [Table 5]

$Total_{10}$ = total man-rem year for 10-year interval = 3.0014E+03 man-rem/year [Table 5]

$$\%Risk_{10} = [23.30/\ 3.0014E{+}3] \times 100$$
$$\%Risk_{10} = 0.84\%$$

Therefore, the total Type A 10-year ILRT interval risk contribution of leakage, represented by Class-1 and Class-3 accident scenarios is 0.84%.

The percent risk increase ($\Delta\%Risk_{10}$) due to a 10-year ILRT over the baseline case is as follows:

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE} \times 100.0$$

Where:
$Total_{BASE}$ =total man-rem/year for baseline interval = 2.9973E+03 man-rem/year [Table 4]

$Total_{10}$ = total man-rem/year for 10-year interval = 3.0014E+03 man-rem/year [Table 5]

$$\Delta\%Risk_{10} = [(3.0014E{+}03 - 2.9973E{+}03) / 2.9973E{+}01] \times 100.0$$
$$\Delta\%Risk_{10} = 0.14\%$$

Therefore, the increase in risk contribution because of relaxed 10-year ILRT test frequency from 3-in-10 to 1-in-10 years is 0.14%.

<u>Risk Impact due to 15-year Test Interval</u>

The risk contribution for a 15-year interval is similar to the 10-year interval. The difference is in the increase in probability of leakage value. For this case, the value is 15 percent or 1.15. (Recall that for a 10-year interval there is a 10% increase on the overall probability of leakage.) In addition, the containment leakage used for the 10-year test interval for both Class-1 and Class-3 are used in the 15-year interval evaluation. The results for this calculation are presented in Table 6.

Table 6

Mean Consequence Measures for 15-Year Test Interval - Given Accident Class

Class	Description	Frequency (per Rx-yr)	Man-rem (50-Miles)	Man-rem/yr (50-Miles)
1	No Containment Failure	1.02E-06	3.24E+06	3.32E+00
2	Large Isolation Failures (Failure-to-close)	2.00E-07	4.94E+07	9.88E+00
3A	Small Pre-existing Liner Breach Failure	7.51E-07	1.41E+07	1.06E+01
3B	Large Pre-existing Liner Breach Failure	2.46E-07	4.94E+07	1.22E+01
4	Small isolation Failure-to-Seal (Type B test)	Not Analyzed	N/A	N/A
5	Small isolation Failure-to-Seal (Type C test)	Not Analyzed	N/A	N/A
6	Other Isolation Failures (e.g., Dependent Failures)	2.07E-09	4.94E+07	1.02E-01
7	Failure Induced by Phenomena (Early and Late Failures)	7.82E-06	1.41E+08	1.10E+03
8	Bypass (SGTR)	3.50E-07	5.33E+09	1.86E+03
CDF	All CET End States	1.04E-05	N/A	3.0024E+03

Based on the above values, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class-1 and Class-3 is as follows:

$\%Risk_{15} = [(CLASS1_{15} + CLASS3A_{15} + CLASS3B_{15}) / Total_{15}] \times 100$

Where:

$CLASS1_{15}$ = Class-1 man-rem/year = 3.32E+00 man-rem/year [Table 6]

$CLASS3A_{15}$ = Class-3a man-remlyear = 1.06E+1 man-rem/year [Table 6]

$CLASS3B_{15}$, = Class-3b man-rem/year = 1.22E+1 man-rem/year [Table 6]

$Total_{15}$ = total man-rem year for 10-year interval = 3.0024E+03 man-rem/year [Table 6]

$\%Risk_{15} = [(3.32E+00+ 1.06E+01 + 1.22 E+01) / 3.0024E+03] \times 100\%$
$\%Risk_{15} = 0.87\%$

Therefore, the total Type A 15-year ILRT interval risk contribution of leakage, represented by Class-1 and Class-3 accident scenarios is 0.87%.

The percent increase in risk (in terms of man-rem/yr) of these associated specific sequences is computed as follows.

$\%Risk_{10\text{-}15} = [(MAN\text{-}REM_{15} - MAN\text{-}REM_{10}) / MAN\text{-}REM_{10}] \times 100$

Where:

$MAN\text{-}REM_{10}$ man-rem/year of 15 years interval (for Classes 1, 3A and 3B) [Table 5]
2.51E+01 man-rem/yr

$MAN\text{-}REM_{15}$ man-rem/year of 15 years interval (for Classes 1, 3A and 3B) [Table 6]

2.61E+01 man-rem/yr

$\%Risk_{10\text{-}15}$ = [(2.61E+01 – 2.51E+01)/2.51E+01] x 100
$\%Risk_{10\text{-}15}$ = 4.0%

Therefore, the change in Type A test frequency from 1-in-10 years to 1-in-15 years increases the risk of those associated specific accident sequences by 4.0 %.

The percent increase on the total integrated plant risk for these accident sequences is computed as follows:

$\%TOTAL_{10\text{-}15}$= [($TOTAL_{15}$ - $TOTAL_{10}$) / $TOTAL_{10}$] x 100

Where:
$TOTAL_{10}$ = total man-rem/year for 10-year interval = 3.0014E+03 man-rem/year [Table 5]
$TOTAL_{15}$ = total man-rem/year for 15-year interval = 3.0024E+03 man-rem/year [Table 6]

$\%TOTAL_{10\text{-}15}$ = 3.32E-02%

Therefore, the risk impact on the total integrated plant risk for these accident sequences influenced by Type A testing is only 0.03%.

The percent risk increase ($\Delta\%Risk_{15}$) due to a 15-year ILRT over the baseline case is as follows:

$\Delta\%Risk_{15}$ = [($TOTAL_{15}$ - $Total_{BASE}$) / $Total_{BASE}$ x 100

Where:
$Total_{BASE}$ = total man-rem/year for baseline interval = 2.9973E+03 man-rem/year [Table 4]
$Total_{15}$ = total man-rem/year for 15-year interval = 3.0024+03 man-rem/year [Table 6]

$\Delta\%Risk_{15}$ = [(2.9973E+03 - 3.0024E+03)/2.997E+03] x 100
$\Delta\%Risk_{15}$ = 0.17%

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from 3-in-10 to 1-in-15 years is 0.17%.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF)

The risk impact associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from containment could, in fact, result in a large release due to failure to detect a pre-existing leak during the relaxation period. For this evaluation, only Class-3 sequences have the potential to result in large releases if a pre-existing leak were present. Class-1 sequences are not considered as potential large release pathways because, for these sequences, the containment remains intact. Therefore, the containment leak rate is expected to be small (less than 2 L_a). A large leak rate would imply an impaired containment, such as Classes 2, 3, 6, and 7.

Late releases are excluded regardless of the size of the leak because late releases are, by definition, not a LERF event. At the same time, sequences in the DCPP IPE that result in the large releases (e.g., large isolation valve failures) are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, for this evaluation, the frequency

of Class-3B sequences is used as the LERF. This frequency, based on a 10-year test interval, is 2.35E-07/year [From Table 5].

Regulatory Guide (RG) 1.174 [Ref. 3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. The RG defines very small changes in risk as resulting in increases of CDF below 1.0E-06/year and increases in LERF below 1.0E-07/year. Since the ILRT does not impact CDF, the relevant figure of merit is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

As described in Step 3, extending the ILRT interval from 1-in-10 to 1-in-15 years will increase the average time that a leak detectable only by an ILRT goes undetected from 60 to 90 months. Since ILRT only detects about 3% of leaks (the rest are identified during LLRTs), the result for a 15-yr ILRT interval is a 15% increase in the overall probability of leakage versus 10% for a 10-year interval. Thus, increasing the ILRT test interval from 10 years to 15 years results in a 5% increase in the overall probability of leakage. Multiplying the LERF frequency (2.35E-05/year) by the increase in the overall probability of leakage (0.05) gives an increase in LERF of 1.14E-08/year, which is well below the RG 1.174 risk significance screening criteria.

Note that the risk increase, as measured by the LERF figure of merit, from the 3-in-10 year to the 1-in-15 year interval is (2.14E-07 * (15%-3%)) = 2.57E-08/year, which is still risk insignificant.

4.0 SUMMARY AND CONCLUSIONS

4.1 Summary

- The baseline risk contribution of leakage, represented by Class-1 and Class-3 accident scenarios, is 0.71%.
- Type A 10-year ILRT interval risk contribution of leakage, represented by Class-1 and Class-3 accidents scenarios, is 0.84%.
- Type A 15-year ILRT interval risk contribution of leakage, represented by Class-1 and Class-3 accidents scenarios, is 0.87%.
- The man-rem/year increase in risk contribution from the extending the ILRT test frequency from the current 1-in-10 year interval to 1-in-15 years is 4.0%.
- The total integrated increase in risk contribution from extending the ILRT test frequency from the current 1-in-10 year interval to 1-in-15 years is 3.32E-02%.
- The change in the LERF figure of merit for extending the ILRT test frequency from the current 1-in-10 year interval to 1-in-15 years is 1.14E-08/year.
- The change in the LERF figure of merit for extending the ILRT test frequency from the original 3-in-10 year interval to 1-in-15 years is 2.57E-08/year.

4.2 Conclusions

Based on the above results, it is concluded that:

- The change in Type A test frequency from 1-in-10 to 1-in-15 years increases the risk of the associated specific accident sequences by 4.0%. The risk impact of those accident sequences impacted by Type A testing on the total integrated plant risk is 0.033%. That is, the risk impact when compared to other severe accident induced risks is negligible. This is consistent with NUREG-1493 conclusions that even increasing the ILRT interval to 1-in-20 years would not have a significant impact on risk.
- Using RG 1.174 LERF figure of merit criteria, the increase in LERF resulting from a change in the Type A test interval to 1-in-15 years is risk insignificance (low risk).
- The defense-in-depth, or safety margins, are not significantly affected by allowing a one-time extension of the ILRT to 15 years on the basis that, consistent with NUREG-1493, this analysis finds that:
    - Type B and C tests are very effective in identifying potential leak paths (about 97%),
    - ILRT-identified leakage unlikelihood to exceed twice the allowable leakage; and
    - The insensitivity of risk to containment leak rate.

5.0 REFERENCES

1. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 26, 1995, Revision 0
2. EPRI TR-104285, "Risk Assessment of Revised Containment Leak Rate Testing Intervals," August 1994
3. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998
4. PRA Calculation File C.9, "Quantification of CDF and LERF for the DCC0 Combined Internal, Seismic, and Fire Model," Revision 9, March 15, 2001
5. PRA Calculation File PRA01-08, "Regeneration of Level 2 Model," Revision 0, September 28, 2001
6. IP3-CALC-VC-03357, Revision 0, "Risk Impact Assessment of Extending Containment Type A Test Interval," January 4, 2001, submitted to the NRC via an Indian Point 3 Nuclear Power Plant letter, Docket Number 50-286
7. RISKMAN SQA Plan, CPAR 01-01, "RISKMAN for Windows, Release 3.00," February 28, 2001
8. NUREG-1493, "Performance-Based Containment Leak-Test Program," Final Report, July 1995
9. NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989
10. NUREG/CR-4551, "Evaluation of Severe Accidents Risks: Zion Unit 1," Brookhaven National Laboratory, Volume 7, Revision 7, October 1990

11. PRA Calculation File N.1, "DCPP PRA Plant Damage States," Revision 4, May 1, 2000

12. Patrick D. T. O'Connor, "Practical Reliability Engineering," John Wiley & Sons, 2nd Ed, 1985

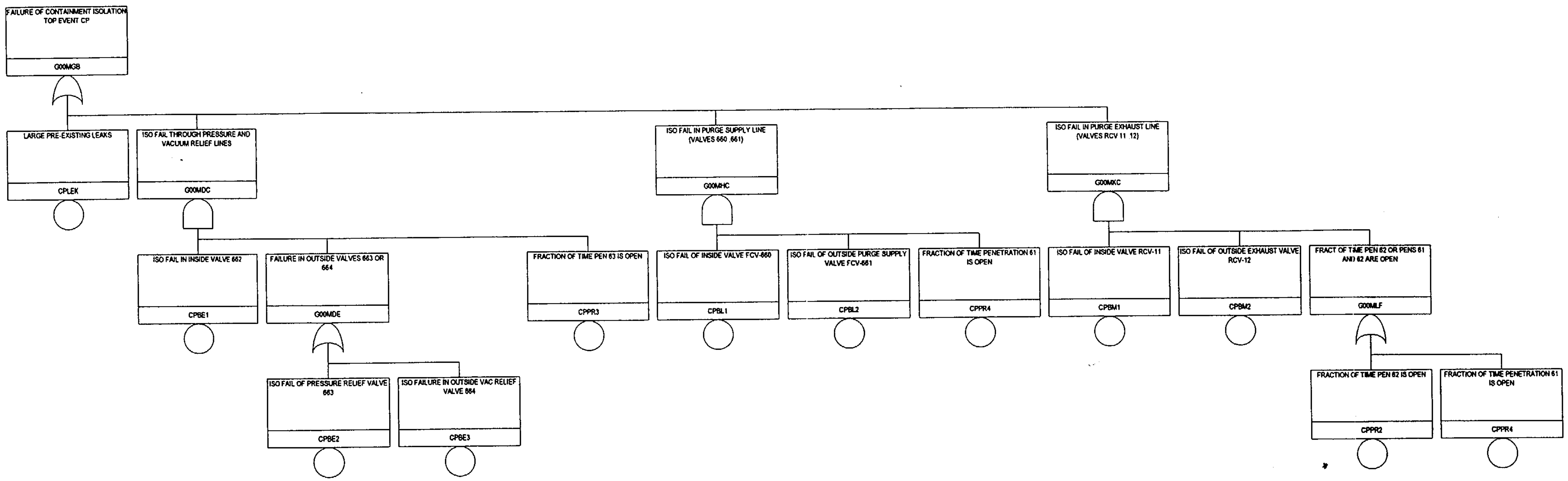


Figure 1- Large Containment Isolation Fault Tree

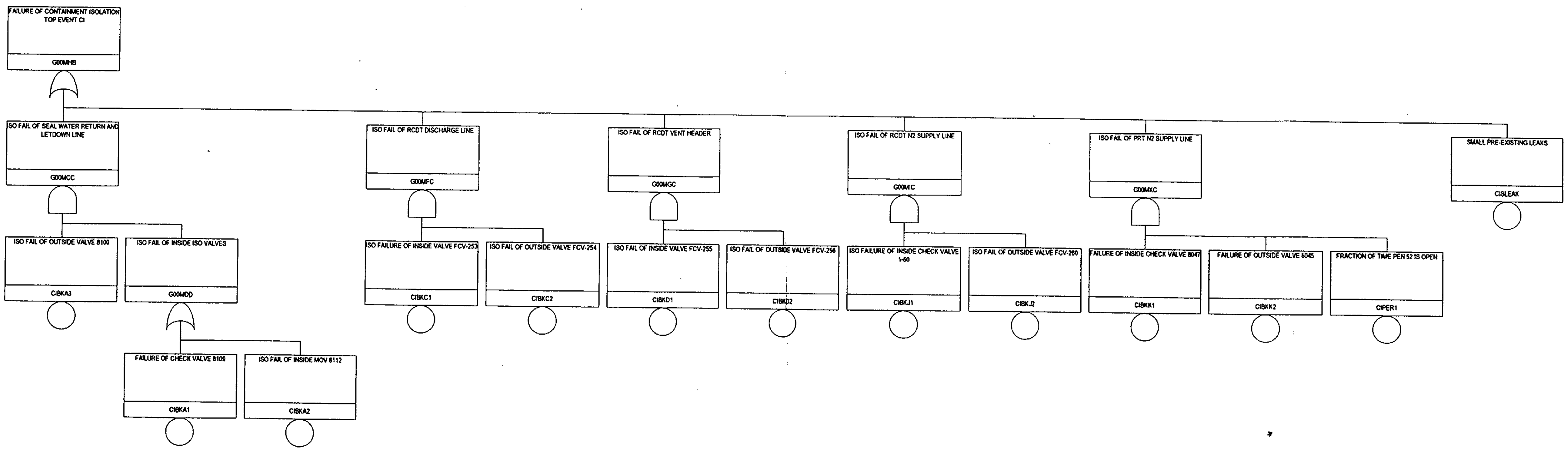


Figure 2- Small Containment Isolation Fault Tree

CALCULATION FILE PRA91-07 Rev. 1

Sheet 24

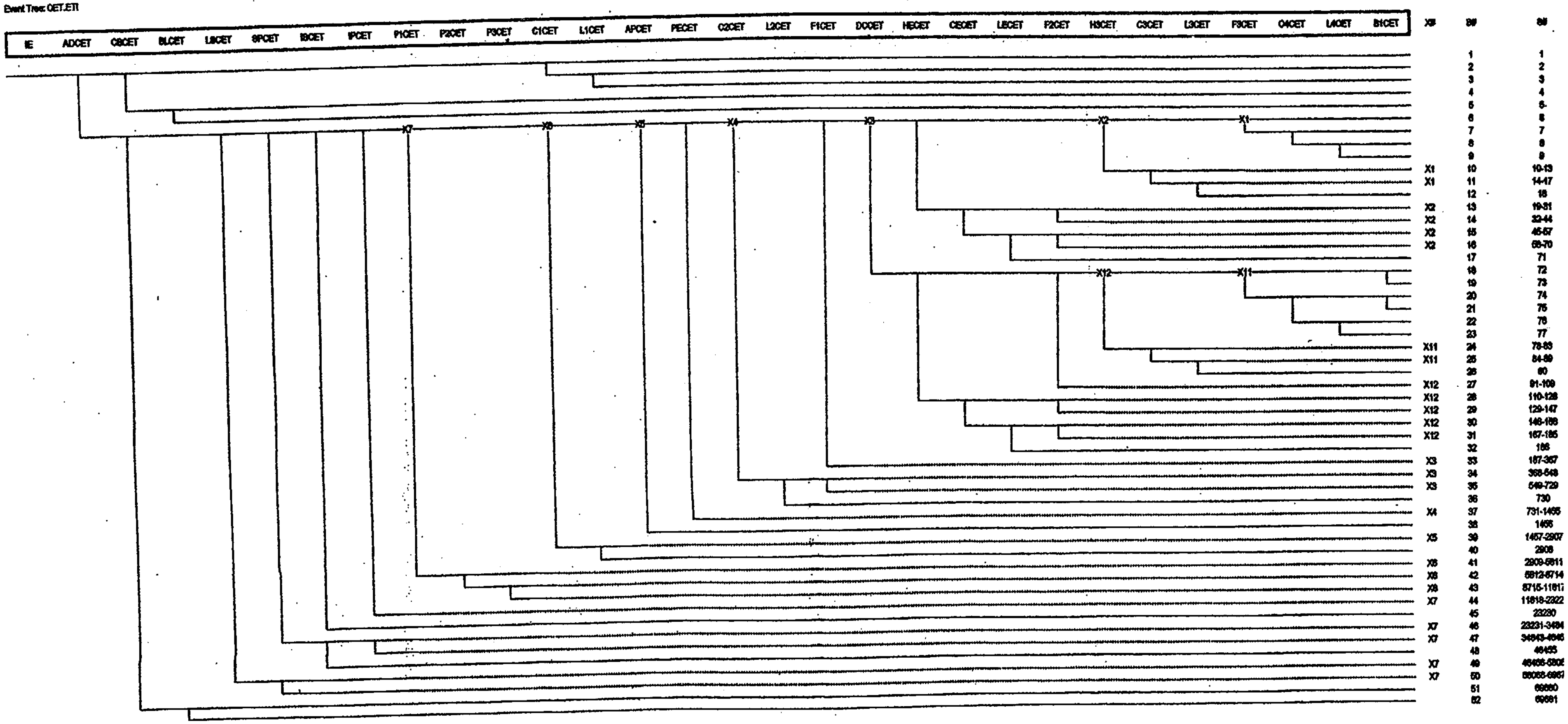
Figure 3- Containment Event Tree

Page I

16:12:30 September

MOBEL Name:

Event Tree: CET.ETI



Top Event IE - Entry State. This top event simply is an entry point to the event tree. The plant damage state bins are normally used as the entry states (similar to initiating events for the plant model event trees) to the containment event tree and are characterized by the thermodynamic conditions in the reactor coolant system and in the containment at the time of severe core damage, and the availability or unavailability of both passive and active plant features that can terminate the accident or mitigate the release of radioactive materials into the environment. In order to simplify CET quantification, the PDSs are grouped into key plant damage states (KPDS).

A representative accident sequence is selected for each KPDS. These representative sequences were analyzed in detail with MAAP to characterize the timing of important events (such as the onset of severe core damage and reactor vessel melt-through), as well as the nature of the core damage and fission product release.

Top Event ADCET - Failure to Arrest Core Damage and Prevent Vessel Breach

Top Event CBCET - Containment Bypass Prior to Core Damage

Top Event BLCET - Large Bypass Prior to Core Damage

Top Event LSCET — Induced PORV Failure

Top Event SPCET — RCP Seal Cooling Unavailable

Top Event ISCET — Induced Steam Generator Tube Rupture

Top Event IPCET — Induced RCS Hot Leg or Surge Line Failure

Top Event P1CET — RCS Pressure at Vessel Breach Exceeds 200 psia

Top Event P2CET — RCS Pressure at Vessel Breach Exceeds 650 psia

Top Event P3CET — RCS Pressure at Vessel Breach Exceeds 2000 psia

Top Event C1CET — Containment Failure Prior to Vessel Breach

Top Event L1CET — Large Containment Failure Prior to Vessel Breach

Top Event APCET — Containment Is Failed by an In-Vessel Steam Explosion

Top Event PECET — High Pressure Melt Ejection

Top Event C2CET — Containment Failure at Vessel Breach

Top Event L2CET — Large Containment Failure at Vessel Breach

Top Event F1CET — Containment Fan Coolers Inoperable After Vessel Breach

Top Event DCCET — Debris Is Not Cooled

Top Event HECET — Hydrogen Burn within 4 Hours of Vessel Breach

Top Event CECET — Containment Failure due to Early Hydrogen Burn

Top Event LECET — Large Containment Failure from Early Burn

Top Event F2CET — Containment Fan Coolers Fail After Early Burn or Debris Uncoolable

Top Event H3CET — Late Burn of Combustible Gases

Top Event C3CET — Late Containment Failure due to Burn

Top Event L3CET — Large, Late Containment Failure due to Hydrogen Burn

Top Event F3CET — Long Term Operation Failure of Containment Fan Coolers

Top Event C4CET — Long-Term Overpressurization

Top Event L4CET — Large Long Term Containment Failure

Top Event BICET — Basemat Penetration

Attachment 1- Riskman Reports

Model: ILRT
Master Frequency File: ILRTRAN
Bin Totals for Group: SEARLY
(Quantified Results only for Groups w/o TE and SF Constraints)

7:54 AM 9/18/2001
Page 1

Bin	Frequency (Quantified)	Frequency (Saved Sequences)	Bin Description
RC14	1.4849E-007	1.4849E-007	
RC16U	1.3057E-007	1.3057E-007	
RC14U	2.4142E-008	2.4141E-008	
RC13	1.7668E-008	1.7668E-008	
RC16	6.7008E-010	6.7008E-010	
RC15U	1.5057E-011	1.5056E-011	
RC13U	1.3186E-012	1.3186E-012	
RC15	7.8779E-013	7.8779E-013	
Total	**3.2155E-007**	**3.2155E-007**	

Model: ILRT
Master Frequency File: ILRTRAN
Bin Totals for Group: LEARLY
(Quantified Results only for Groups w/o TE and SF Constraints)

7:49 AM 9/18/2001
Page 1

Bin	Frequency (Quantified)	Frequency (Saved Sequences)	Bin Description
RC04	1.2022E-007	1.2022E-007	
RC02	7.5459E-008	7.5459E-008	
RC04U	5.9317E-008	5.9317E-008	
RC01	1.0815E-008	1.0815E-008	
RC03U	3.0539E-009	3.0539E-009	
RC03	1.4648E-009	1.4648E-009	
RC01U	2.6745E-010	2.6745E-010	
RC02U	4.1995E-011	4.1995E-011	
Total	**2.7063E-007**	**2.7063E-007**	

Model: ILRT
Master Frequency File: ILRTRAN
Bin Totals for Group: LATE
(Quantified Results only for Groups w/o TE and SF Constraints)

7:47 AM 9/18/2001
Page 1

Bin	Frequency (Quantified)	Frequency (Saved Sequences)	Bin Description
RC10	2.7528E-006	2.7528E-006	
RC12U	2.0422E-006	2.0422E-006	
RC06	8.9781E-007	8.9781E-007	
RC08U	6.1812E-007	6.1812E-007	
RC21	6.0515E-007	6.0515E-007	BASEMENT MELT THROUGH AND NO CONTAINMENT FAILURE
RC10U	2.9018E-007	2.9018E-007	
RC06U	1.0209E-007	1.0209E-007	
RC12	9.5302E-008	9.5302E-008	
RC08	2.6934E-008	2.6934E-008	
RC05	3.1024E-010	3.1024E-010	
RC07	4.6608E-011	4.6608E-011	
RC05U	0.0000E+000		
RC07U	0.0000E+000		
RC09U	0.0000E+000		
RC11	0.0000E+000		
RC11U	0.0000E+000		
RC09	0.0000E+000		
Total	**7.4309E-006**	**7.4309E-006**	

Model: ILRT
Master Frequency File: ILRTRAN
Bin Totals for Group: CNTBYP
(Quantified Results only for Groups w/o TE and SF Constraints)

8:37 AM 9/18/2001
Page 1

Bin	Frequency (Quantified)	Frequency (Saved Sequences)	Bin Description
RC17	3.3502E-007	3.3502E-007	SGTR
RC18	1.4570E-008	1.4570E-008	ISLOCA
Total	**3.4959E-007**	**3.4959E-007**	

Model: ILRT
Master Frequency File: ILRTRAN
Bin Totals for Group: LGTINT
(Quantified Results only for Groups w/o TE and SF Constraints)

7:51 AM 9/18/2001
Page 1

Bin	Frequency (Quantified)	Frequency (Saved Sequences)	Bin Description
RC19	1.4112E-006	1.4112E-006	SIGNIFICANT CORE DAMAGE PREVENTED
RC20	6.1127E-007	6.1127E-007	LONG TERM CONTAINMENT INTACT
Total	**2.0224E-006**	**2.0224E-006**	