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Ref: 10CFR50.90
[Proprietary Information Enclosed]

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December 26, 2001

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 01-14
REVISION TO TECHNICAL SPECIFICATION (TS) 5.5.16
CONTAINMENT LEAKAGE RATE TESTING PROGRAM

Gentlemen:

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

The proposed change will revise TS 5.5.16 entitled Containment Leakage Rate Testing Program. This request proposes a one-time extension of the ten-year period of the performance-based leakage rate testing program for Type A tests as prescribed by NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J," and applied by 10CFR50, Appendix J, Option B. The ten-year interval between integrated leakage rate tests is to be extended to 15 years from the previous integrated leakage rate tests, which were completed on December 7, 1993 (Unit 1) and December 1, 1997 (Unit 2). The change reflects a one-time deferral of the next Type A Containment Integrated Leak Rate Test (ILRT) to no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2). This proposed change is based on and has been evaluated using the "risk informed" guidance in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

A001

Attachment 1 provides a detailed description of the proposed changes, a safety analysis of the proposed changes, TXU Electric's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected Technical Specification pages marked-up to reflect the proposed changes. Attachment 3 provides retyped Technical Specification pages which incorporate the requested changes.

Enclosure 1 provides the "Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension" that supports this license amendment request. Ricky Summit Consulting, Inc., considers information contained in Appendix B "Surrogate Person-Rem Methodology (RSC 01-44)" to Enclosure 1 "Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension" to be proprietary. In accordance with the requirements of 10CFR2.790(b) for withholding of proprietary information from public disclosure, an Affidavit is enclosed (Enclosure 2). Correspondence with respect to the proprietary aspects of the supporting Rick Summit Consulting, Inc., Affidavit should be addressed to Ricky Summit Consulting, Inc., 342 Ebenezer Road, Knoxville, TN 37923. Enclosure 3 provides a non- proprietary version of Enclosure 1.

TXU Electric requests approval of the proposed License Amendment by May 31, 2002, to be implemented within 60 days of the issuance of the license amendment. This would allow deferral of the next ILRT Type A Test, currently scheduled for the ninth refueling outage in the fall of 2002 (Unit 1) and the ninth refueling outage in the fall of 2006 (Unit 2).

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

This communication contains the following revised commitment to be implemented upon NRC approval of the License Amendment Request:

Commitment \Number	Commitment Description
09030	Type A test should be conducted in accordance with provisions of NEI 94-01, Rev. 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, as modified by the following exception: 1. NEI 94-01 - 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)." Upon implementation of the one time deferral, the visual examinations (as required by NEI 94-01) will be performed approximately every other refueling outage until the Type A test outage and prior to the next Type A test.

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

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

Executed on December 26, 2001.

Sincerely,

C. L. Terry

By: MC Blevins

M. R. Blevins
Vice President & Deputy to
Senior Vice President & Principal Nuclear Officer

CBC/cbc

Attachments
1. Description and Assessment
2. Markup of Technical Specifications pages
3. Retyped Technical Specification Pages

Enclosures
1. Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension [Proprietary]
2. Affidavit (request to withhold proprietary information in Enclosure 1
3. Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension [Non-Proprietary version of Enclosure 1]

c - E. W. Merschoff, Region IV
C. E. Johnson, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Authur C. Tate
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Texas Department of Public Health
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ATTACHMENT 1 to TXX-01187

DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

By this letter, TXU Electric requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. Proposed change LAR-01-14 is a request to revise Technical Specifications (TS) 5.5.16, "Containment Leakage Rate Testing Program" for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2. TXU Electric requests approval of the proposed License Amendment by May 31, 2002, to be implemented within 60 days of the issuance of the license amendment. The proposed amendment will allow for a one-time extension of the current interval between the Type A tests from 10 to 15 years. The change reflects a one-time deferral of the next Type A Containment Integrated Leak Rate Test (ILRT) to no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2). This would allow deferral of the next ILRT Type A Test, currently scheduled for the ninth refueling outage in the fall of 2002 (Unit 1) and the ninth refueling outage in the fall of 2006 (Unit 2).

The CPSES Final Safety Analysis Report (Section 6.2.6) (Reference 1) will be updated as required after the License Amendment Request is approved and implemented.

2.0 PROPOSED CHANGE

Technical Specification Section 5.5.16.a currently 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-Test Program, dated September 1995"

Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," (Reference 2) endorses NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995 (Reference 3) and prepared by the Nuclear Energy Institute (NEI). NEI 94-01 provides methods acceptable to the NRC staff for complying with the provisions of Option B as described in Regulatory Guide 1.163. NEI 94-01 includes the criterion that Option B Type A testing be performed at a frequency of at least once per 10 years.

This proposed change in the current licensing basis is a one-time extension of the test interval from 10 years to 15 years. The approved one-time deferral of the integrated leakage rate test would be incorporated into Technical Specification 5.5.16.a by adding:

> "...as modified by the following exception:

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

In summary the proposed change will revise TS 5.5.16 entitled Containment Leakage Rate Testing Program to allow a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) to no later than December 15, 2008, for Unit 1 and December 9, 2012, for Unit 2. This proposed change is based on and has been evaluated using the risk informed guidance in Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 4).

3.0 BACKGROUND

Containment leakage tests are performed to verify that Containment leakage is maintained below the acceptable limits stated in Technical Specification 5.5.16. The leakage tests ensure the public health and safety in the case of a design basis accident that would release radioactivity to the containment.

The leakage testing program consists of the following types of periodic tests: (1) Type A Test - measures the overall integrity of the containment system, (2) Type B Test - measures leakage rates across pressure retaining or leakage limiting boundaries other than valves, and (3) Type C Test - measures containment isolation valve leakage rates. This request does not modify the existing Appendix J Type B and Type C testing programs nor does it change the Appendix J Type A, Type B, or Type C Test methods. The change is a one-time exception to the Type A Test frequency.

This request represents a cost beneficial licensing change. The integrated leak rate test imposes significant expense on the station while the safety benefit of performing it within 10 years, versus 15 years, is minimal. Cost savings have been estimated for the Unit 1 ninth refueling outage at $410,000 for actually performing the test and eliminating $25,000 per hour for each hour of critical path outage time (the number of critical path hours is variable).

4.0 TECHNICAL ANALYSIS

The proposed changes have been evaluated to determine that current regulations and applicable requirements continue to be met, that adequate defense-in-depth and sufficient safety margins are maintained, and that any increases in core damage frequency (CDF) and large early release frequency (LERF) are small and consistent with the NRC Safety Goal Policy Statement (Reference 5), and the acceptance criteria in 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, (Reference 4) and Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," August 1999 (Reference 6).

4.1 Traditional Engineering Considerations

In License Amendment 51/37 (Reference 17), TXU Electric committed to testing as required by 10 CFR 50, Appendix J, Option B, and in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995."

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B, and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493 (Reference 7). NUREG-1493 made the following observations with regard to decreasing the test frequency:

"Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the same fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing had minimal impact on public risk."

While Type B and C tests identify the vast majority of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

The surveillance frequency for Type A testing in NEI 94-01 is at least once per 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.O L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Based on the last ILRT Type A tests (Unit 1 - December 7, 1993, and Unit 2 - December 7, 1997), the current interval for CPSES is once every 10 years (Unit 1 and Unit 2). As allowed by Section 9.2.3 of NEI 94-01 (Reference 3) which states in part, "A pre-operational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a pre-operational Type A test can be treated as a periodic test," CPSES did perform an evaluation (for Unit 1 and Unit 2) to document the acceptability of using the pre-operational Type A test as one of the two Type A tests.

A Type A test can detect containment leakage due to a loss of structural capability. All other sources of containment leakage detected in a Type A test analysis can be detected by the Type B and C tests.

4.1.1 Inspections

4.1.1.1 IWE/ IWL Inservice Inspection (ISI) activities to support ILRT

For Inservice Inspection (ISI) the applicable ASME Section XI Code for both units is the 1986 Edition, no Addenda.

As required by 10CFR50.55a, Inservice Inspection (ISI) of the CPSES Containment building is conducted in accordance with the requirements of Subsections IWE and IWL of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI. Subsection IWE provides the rules and requirements for inservice inspection of penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. CPSES requested and received approval for Relief Requests E-1, "Metallic containment shell and penetration liners and their integral attachments" and L-1, "Concrete Containment Components" (Reference 8). The relief requests allow use of the 1998 Edition of the Subsections IWE and IWL of the ASME Code, supplemented by licensee commitments. CPSES completed the first interval inspections for Unit 1 and Unit 2 in September 2001, in accordance with Subsections IWE and IWL of ASME Code Section XI, with acceptable results.

NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," discussed the inadequate local leak rate testing of two-ply stainless steel bellows. CPSES has no such bellows that act as part of the containment boundary.

The ASME Code Section XI IWE and IWL containment inspections provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

4.1.1.2 Maintenance Rule Monitoring to support ILRT

Containment Building structure and containment isolation functions are monitored under the maintenance rule to ensure functions are maintained and that maintenance is effective.

Maintenance Rule baseline inspections were performed in May 1998. The inspection results indicated that an appropriate program had been developed and implemented to meet the requirements of 10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 9) [the Maintenance Rule]. The inspection determined that the program for monitoring the condition and effectiveness of the containment structure and isolation functions were appropriate and met the intent of the Maintenance Rule. The results were documented in an NRC Inspection Report (Reference 10).

4.1.1.3 Containment Visual Inspection

As required by NEI 94-01 (Reference 3) and R.G. 1.163, part C.3 (Reference 2), visual examinations are performed of accessible interior and exterior surfaces of the containment system for structural deterioration. These examinations are currently performed prior to the Type A test and during two other refueling outages when the ILRT is on a 10 year interval. Upon implementation of the one-time deferral, the visual examination will be performed approximately every other refueling outage until the Type A test outage and prior to the next Type A test.

4.1.2 Previous Integrated Leakage Rate Test Results Inspections

Previous Type A tests confirmed that the CPSES reactor containment structure has leakage well under acceptance limits and represents minimal risk to increased leakage. The risk is minimized by continued Type B and Type C testing for direct communication with containment atmosphere. Also, the Inservice Inspection (ISI) program and maintenance rule monitoring provide confidence in containment integrity.

The results for the last Type A test for CPSES are listed below.

Date	As Found Leakage	Acceptance Limit	Test Pressure (psia)
12/07/1993 (Unit 1)*	0.05638% wt/day	0.10% wt/day	63.36
12/01/1997 (Unit 2)*	0.0317% wt/day	0.10% wt/day	63.47

* The commercial operation dates for Unit 1 and Unit 2 are August 13, 1990, and August 3, 1993, respectively. As noted in Section 4.1 above, CPSES did perform an evaluation (Unit 1 and Unit 2) to document the acceptability of using the pre-operational tests as one of the successful tests.

The testing history and structural capability of the containment have established that Comanche Peak Steam Electric Station has had acceptable containment leakage rates with considerable margin, that the structural integrity of containment is assured, and that there is negligible impact in extending the Type A test interval on a one-time basis.

4.1.3 Plant Operational Performance

During power operation, instrument air leaks from air-operated valves inside containment and pressurizes the containment building. Containment pressure is monitored and conditions approaching the limits allowed by the Technical Specifications are annunciated. Because it is routinely necessary to reduce the increase in the building internal pressure by periodic operation of the containment pressure relief, a large pre-existing leak would make it unnecessary to periodically operate the containment pressure relief. This change in operating pattern would be noticed by plant operators.

Although not as significant as pressure resulting from a Design Basis Accident, the fact that the containment can be pressurized by leakage from air-operated valves provides a degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure). This feature is a complement to visual inspection of the interior and exterior of the containment structure for those areas that may be inaccessible for visual examination.

4.2 Evaluation of Risk Impact

4.2.1 PRA Approach

10CFR50, Appendix J allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak testing. This report documents a risk-based evaluation of the proposed change of the integrated leak rate test (ILRT) test interval for the Comanche Peak Steam Electric Station (CPSES). The proposed change would impact testing associated with the current surveillance test for Type A leakage. No change to Type B or Type C testing is proposed at this time.

The evaluation for CPSES is consistent with similar assessments performed for the Indian Point 3 (IP3) plant, which was approved by the NRC (References 11 and 12) and for the Crystal River 3 plant (Reference 13). This assessment utilizes the guidelines set forth in NEI 94-01 (Reference 3), the methodology used in EPRI TR-104285 (Reference 14) and the regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings in support of a licensee request to change a plant's licensing basis, Regulatory Guide (RG) 1.174 (Reference 4)

This calculation evaluates the risk associated with various ILRT intervals as follows:

- 3 years - Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for CPSES.
- 15 years – Proposed extended test interval, similar to CPSES request.

The analysis utilizes the latest CPSES probabilistic risk assessment (PRA) results. The PRA was initially developed for the CPSES individual plant examination (IPE) (Reference 15) to estimate the baseline core damage and plant damage states. Several updates to the CPSES Level 1 analysis have been incorporated since the IPE, these updates also included an update to the Level 2 information. Therefore, this information represents the most recent analysis documented for CPSES.

The release category and person-rem information is based on design basis leakage evaluations and extrapolation of the release category information using a modeling framework that develops the person-rem estimates based on the relative release fractions of radionuclides. The framework is described in Appendix B of Enclosure 1.

4.2.2 Summary of Risk Results/Conclusions (similar to Indian Point 3 approach)

The discussion below summarizes the evaluation provided in Sections 1 through 5 of Enclosure 1, which is consistent with the Indian Point 3 template/methodology referred to in Section 4.2.1 above. However, recognizing that there is a weakness in that template, CPSES has also provided results correcting that weakness (see Appendix A of Enclosure 1). A review of the results of both approaches indicate the change is acceptable whether the original template or the modified template is used.

The specific results of the unmodified Indian Point 3 template are summarized in the table below. The Type A contribution to LERF is defined as the contribution from Class 3b (Class 3b is defined in Reference 14).

Summary of Risk Impact on Extending Type A ILRT Test Frequency *			
	Risk Impact for 3-years (baseline)	Risk Impact for 10-years (current requirement)	Risk Impact for 15-years
Total Integrated Risk (Person-Rem/yr)	89.247	89.258	89.263
Type A Testing Risk (Person-Rem/yr)	1.156E-1	1.272E-1	1.329E-1
% Total Risk (Type A / Total)	0.1295%	0.1425%	0.1489%
Type A LERF (Class 3b) (per year)	3.70E-7	4.07E-7	4.26E-7
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			5.42E-3
% Increase from current (Δ Risk / Total Risk)			0.006%
Δ LERF from current (per year)			1.85E-8
Δ CCFP from current			0.104%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			1.63E-2
% Increase from current (Δ Risk / Total Risk)			0.018%
Δ LERF from baseline (per year)			5.56E-8
Δ CCFP from baseline			0.312%

* Results of Evaluation using Indian Point 3 template methodology. For the results using the modified template see Appendix A of Enclosure 1.

Based on the evaluation the following conclusions are evident:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-10-year interval to once-per-fifteen years is 0.00542 person-rem/yr.

- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.85E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current once-per-10-year interval to once-per-15 years is 0.104%

- The change in Type A test frequency from once-per-10-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.006%. Also, the change in Type A test frequency from the original three-per-10-years to once-per-fifteen-years increases the risk only 0.018%. Therefore, the risk impact when compared to other severe accident risks is negligible.

- Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 1-per-10-years to 1- per-15-years is 1.85E-8. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from 3-per-10-years to 1- per-15-years is 5.56E-8/yr and is below the guidance.

- R.G. 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.104% for the proposed change and 0.312% for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

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

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

   Response: No

   The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493, "Performance-Based Containment System Leakage Testing Requirements," September 1995, has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. A high degree of assurance is provided through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last Type A test show leakage to be below acceptance criteria, indicating a very leak tight containment. Inspections required by the American Society of Mechanical Engineers (ASME) Code Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) are performed in order to identify indications of containment degradation that could affect that leak tightness. Type B and C testing required by Technical Specifications will identify any containment opening such as valves that would otherwise be detected by the Type A tests. These factors show that a Type A test extension will not represent a significant increase in the consequences of an accident.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing (10CFR50, Appendix J, Option B, Integrated Leak Rate Testing). The current test interval of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG 1493, "Performance-Based Containment System Leakage Testing Requirements," September 1995, generic study of the effects of extending containment leakage testing found that a 20 year extension in Type A leakage testing resulted 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 decrease in Type A testing frequency would have a minimal affect on this risk since 95% of the potential leakage paths are detected by Type C testing. Regular inspections required by the American Society of Mechanical Engineers (ASME) Code Section XI (Subsections IWE and IWL) and maintenance rule monitoring (10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants) will further reduce the risk of a containment leakage path going undetected.

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

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

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.54(o) - "Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under 50.82(a)(1) have been submitted, shall be subject to the requirements set forth in appendix J to this part."

10 CFR 50, Appendix A, General Design Criteria (GDC) 52 - "Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure."

GDC 53 - "Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows."

GDC 54 - "Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits."

10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performanced-Based Requirements."

Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995.

<u>Analysis</u>

The Containment Building, Containment penetrations, and Containment isolation barriers are designed to permit periodic leakage rate testing as required by General Design Criteria (GDC) 52, 53, and 54 of Title 10 Code of Federal Regulations, Part 50, Appendix A (Reference 16).

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." In License Amendment 51/37 (Reference 17), TXU Electric committed to testing as required by 10 CFR 50, Appendix J, Option B, and in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995."

Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of Nuclear Energy Institute (NEI) 94-01 and ANSI/ANS 56.8 - 1994 (Reference 18) 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 analyzes, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide."

Therefore, this application does not require an exemption to Option B.

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

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 REFERENCES

1. Comanche Peak Steam Electric Station Final Safety Analysis Report, Docket Nos. 50-445 and 50-446.

2. NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

3. NEI 94-01, "Nuclear Energy Institute Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

4. NRC 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.

5. NRC's Probabilistic Risk Assessment (PRA) Policy Statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Volume 60, p.42622, August 16, 1995.

6. NRC Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," August 1999.

7. NUREG-1493, " Performance- Based Containment Leak-Test Program," Final Report, September 1995.

8. Relief Requests Relief Requests E-1, "Metallic containment shell and penetration liners and their integral attachments" and L-1, "Concrete Containment Components" ASME Boiler and Pressure Vessel Code Section XI, Subsections IWE and IWL.

   a. TXU Electric letter logged TXX-98041, from C. L. Terry to USNRC dated February 20, 1998.
   b. TXU Electric letter logged TXX-99082, from C. L. Terry to USNRC dated March 26, 1999.
   c. TXU Electric letter logged TXX-99130, from C. L. Terry to USNRC dated June 8, 1999.
   d. TXU Electric letter logged TXX-99152, from C. L. Terry to USNRC dated June 15, 1998.
   e. NRC Evaluation of Relief Requests: Use of 1998 Edition of Subsections IWE and IWL of the ASME Code for Containment Inspection (TAC NOS. MA2038 and MA2039), dated July 23, 1999.

9. 10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," [the Maintenance Rule].

10. NRC Inspection Report 50-445/98-10; 50-446/98-10, dated July 31, 1998.

11. Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", Entergy, IPN-01-007, January 18, 2001.

12. Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MBO178), United States Nuclear Regulatory Commission, April 17, 2001.

13. Evaluation of Risk Significance of ILRT Extension, Revision 2, Florida Power Corporation, F-01-0001, June 2001.

14. Electric Power Research Institute, TR-104285, Gisclon, J. M., et al, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994.

15. TXU Electric letter TXX-92387, "Individual Plant Examination for Severe Accident Vulnerabilities," [Volume 1], dated August 28, 1992, and TXU Electric letter TXX-92490, "Individual Plant Examination for Severe Accident Vulnerabilities (IPE)," [Volume 2], TXX-92490, dated October 30, 1992 .

16. 10CFR50, Appendix A, General Design Criteria.

17. Comanche Peak Steam Electric Station, Issuance of License Amendment 51/37 re: Use of new Containment Leakage Rate Testing Program as required by 10 CFR 50, Appendix J, Option B for CPSES Units 1 and 2 (TAC NOS. M94992 and M94993), dated June 13, 1996.

18. American National Standard ANSI/ANS - 56.8 - 1994, "Containment System Leakage Testing Requirements."

ATTACHMENT 2 to TXX-01187

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Page 5.0-27

5.5 Programs and Manuals (continued)

5.5.16. Containment Leakage Rate Testing Program

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

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

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

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

d. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

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

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

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

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

(continued)

ATTACHMENT 3 to TXX-01187

RETYPED TECHNICAL SPECIFICATION PAGES

Pages 5.0-27

5.5 Programs and Manuals (continued)

5.5.16. Containment Leakage Rate Testing Program

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

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

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

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

d. Leakage rate acceptance criteria are:

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

   2. Air lock testing acceptance criteria are:

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

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

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

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

(continued)

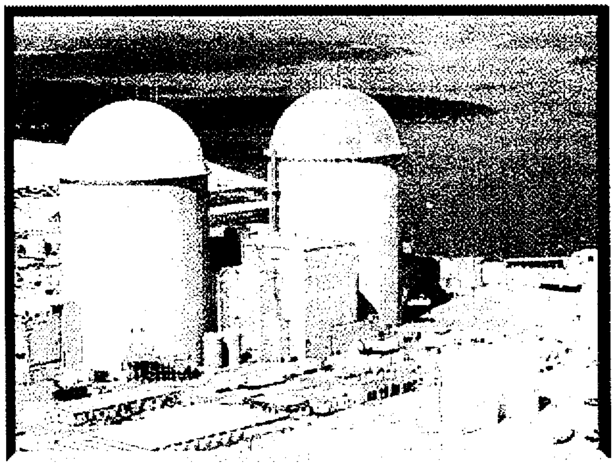
ENCLOSURE 3 to TXX-01187

Comanche Peak Steam Electric Station Probabilistic Safety Assessment, Evaluation of Risk Significance of ILRT Extension

[Non-Proprietary Version]

TXU Electric RSC 01-47/R&R-PN-110 Rev.0

Comanche Peak Steam Electric Station Probabilistic Safety Assessment



Evaluation of Risk Significance of ILRT Extension

Revision 0

November 2001

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1.0 PURPOSE

The purpose of this calculation is to evaluate the risk of extending the Type A Integrated Leak Rate Test (ILRT) interval beyond the current 10 years required by 10 CFR 50, Appendix J at the Comanche Peak Steam Electric Station for both unit 1 and unit 2.

1.1 SUMMARY OF THE ANALYSIS

10 CFR 50, Appendix J allows individual plants to extend Type A surveillance testing requirements and to provide for performance-based leak testing. This report documents a risk-based evaluation of the proposed change of the integrated leak rate test (ILRT) test interval for the Comanche Peak Steam Electric Station (CPSES). The proposed change would impact testing associated with the current surveillance test for Type A leakage (procedure PPT-S1-7014)[1]. No change to Type B or Type C testing is proposed at this time.

The evaluation for CPSES is consistent with similar assessments performed for the Indian Point 3 (IP3) plant, which was approved by the NRC[2,3] and for the Crystal River 3 plant[4]. This assessment utilizes the guidelines set forth in NEI 94-01[5], the methodology used in EPRI TR-104285[6] and the regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings in support of a licensee request to a plant's licensing basis, RG 1.174[7].

This calculation evaluates the risk associated with various ILRT intervals as follows:

- 3 years - Interval based on the original requirements of 3 tests per 10 years.
- 10 years – This is the current test interval required for CPSES.
- 15 years – Proposed extended test interval, similar to IP3 request.

The analysis utilizes the latest CPSES probabilistic risk assessment (PRA) results. The PRA was initially developed for the CPSES individual plant examination (IPE)[8] to estimate the baseline core damage and plant damage states. Several updates to the CPSES level 1 analysis have been incorporated since the IPE, these updates also included an update to the Level 2 information. Therefore, this information represents the most recent analysis documented for CPSES.

The release category and person-rem information is based on design basis leakage evaluations and extrapolation of the release category information using a modeling framework that develops the person-rem estimates based on the relative release fractions of radionuclides. The framework is described in Appendix B.

1.2 SUMMARY OF RESULTS/CONCLUSIONS

The specific results are summarized in Table 1 below. The Type A contribution to LERF is defined as the contribution from Class 3b.

Table 1
Summary of Risk Impact on Extending Type A ILRT Test Frequency

	Risk Impact for 3-years (baseline)	Risk Impact for 10-years (current requirement)	Risk Impact for 15-years
Total Integrated Risk (Person-Rem/yr)	89.247	89.258	89.263
Type A Testing Risk (Person-Rem/yr)	1.156E-1	1.272E-1	1.329E-1
% Total Risk (Type A / Total)	0.1295%	0.1425%	0.1489%
Type A LERF (Class 3b) (per year)	3.70E-7	4.07E-7	4.26E-7
Changes due to extension from 10 years (current)			
Δ Risk from current (Person-rem/yr)			5.42E-3
% Increase from current (Δ Risk / Total Risk)			0.006%
Δ LERF from current (per year)			1.85E-8
Δ CCFP from current			0.104%
Changes due to extension from 3 years (baseline)			
Δ Risk from baseline (Person-rem/yr)			1.63E-2
% Increase from baseline (Δ Risk / Total Risk)			0.018%
Δ LERF from baseline (per year)			5.56E-8
Δ CCFP from baseline			0.312%

Based on the analysis and available data the following is stated:

- The person-rem/year increase in risk contribution from extending the ILRT test frequency from the current once-per-ten-year interval to once-per-fifteen years is 0.00542 person-rem/yr.

- The risk increase in LERF from extending the ILRT test frequency from the current once-per-10-year interval to once-per-15 years is 1.85E-8/yr.

- The change in conditional containment failure probability (CCFP) from the current once-per-10-year interval to once-per-15 years is 0.104%

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the risk impact on the total integrated plant risk by only 0.006%. Also, the change in Type A test frequency from the original three-per-ten-years to once-per-fifteen-years increases the risk only 0.018% Therefore, the risk impact when compared to other severe accident risks is negligible.

- Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from an once-per-ten-years to an once per-fifteen-years is 1.85E-8. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr, increasing the ILRT interval from 10 to 15 years is therefore considered non-risk significant. In addition, the change in LERF resulting from a change in the Type A ILRT test interval from a three-per-ten-years to an once per-fifteen-years is 5.56E-8/yr, is below the guidance.

- R.G. 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with defense-in-depth philosophy is maintained by demonstrating that the balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in conditional containment failure probability was estimated to be 0.104% for the proposed change and 0.312% for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. These changes are small and demonstrate that the defense-in-depth philosophy is maintained.

2.0 DESIGN INPUTS

The CPSES PRA is a non-safety related tool and is intended to provide "best estimate" results that can be used as input when making risk informed decisions. The PRA provides the most recent results for the CPSES PRA. The PRA is not (Reference 8) considered as design basis information.

The inputs for this calculation come from the information documented in the CPSES PRA and the level 2 update (Reference 9). The CPSES key plant damage states are summarized in Table 2.

Table 2
CPSES Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
1H	Reactor Coolant System (RCS) breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter (includes stuck open PORVs and larger seal LOCAs), with early melting of the core. Fan coolers failed, CS failed	7.78E-08
1E	Reactor Coolant System (RCS) breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter (includes stuck open PORVs and larger seal LOCAs), with early melting of the core. Fan coolers failed, CS injection success only	4.38E-09
1F	Reactor Coolant System (RCS) breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter (includes stuck open PORVs and larger seal LOCAs), with early melting of the core. Fan coolers failed, CS injection and recirculation success	6.30E-09
2H	RCS breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter, (includes stuck open PORVs and larger seal LOCAs), with late melting of the core. Fan coolers failed, CS failed	5.52E-11
2E	RCS breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter, (includes stuck open PORVs and larger seal LOCAs), with late melting of the core. Fan coolers failed, CS injection success only	1.70E-10
2F	RCS breach with pressure and leakage rates associated with LOCAs of 0.6 to 2 inches in diameter, (includes stuck open PORVs and larger seal LOCAs), with late melting of the core. Fan coolers failed, CS injection and recirculation success	1.68E-09
3H	High RCS pressure. Leakage rates associated with boil-off of the reactor coolant through cycling pressurizer relief valves (not stuck open) or small seal LOCAs up to 60 GPM/PM (.06 inch diameter), with early melting of the core. Fan coolers failed, CS failed	6.43E-06

Table 2
CPSES Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
3E	High RCS pressure. Leakage rates associated with boil-off of the reactor coolant through cycling pressurizer relief valves (not stuck open) or small seal LOCAs up to 60 GPM/PM (.06 inch diameter), with early melting of the core. Fan coolers failed, CS injection success only	4.75E-07
3F	High RCS pressure. Leakage rates associated with boil-off of the reactor coolant through cycling pressurizer relief valves (not stuck open) or small seal LOCAs up to 60 GPM/PM (.06 inch diameter), with early melting of the core. Fan coolers failed, CS injection and recirculation success	1.02E-06
4H	High RCS pressure and leakage rates associated with boil-off of the coolant through cycling relief vavles (not stuck open) or small seal LOCAs up to 60 GPM/PM (0.6 inch diameter), with late melting of the core. Fan coolers failed, CS failed	2.77E-08
4E	High RCS pressure and leakage rates associated with boil-off of the coolant through cycling relief vavles (not stuck open) or small seal LOCAs up to 60 GPM/PM (0.6 inch diameter), with late melting of the core. Fan coolers failed, CS injection success only	9.25E-09
4F	High RCS pressure and leakage rates associated with boil-off of the coolant through cycling relief vavles (not stuck open) or small seal LOCAs up to 60 GPM/PM (0.6 inch diameter), with late melting of the core. Fan coolers failed, CS injection and recirculation success	4.19E-08
5H	Large rates of leakage from the RCS and low pressures associated with LOCAs greater than 2 inches in diameter and failure of coolant injection, resulting in early melting of the core Fan coolers failed, CS failed	9.63E-10
5E	Large rates of leakage from the RCS and low pressures associated with LOCAs greater than 2 inches in diameter and failure of coolant injection, resulting in early melting of the core Fan coolers failed, CS injection success only	1.21E-10
5F	Large rates of leakage from the RCS and low pressures associated with LOCAs greater than 2 inches in diameter and failure of coolant injection, resulting in early melting of the core Fan coolers failed, CS injection and recirculation success	8.47E-10
6H	LOCA greater than 2 inches in diameter conditions, with failure of coolant recirculation and delayed melting. Fan coolers failed, CS failed	5.54E-10
6E	LOCA greater than 2 inches in diameter conditions, with failure of coolant recirculation and delayed melting. Fan coolers failed, CS injection success only	6.44E-10

Table 2
CPSES Plant Damage States

Plant Damage State	Representative Sequence	Frequency (/yr)
6F	LOCA greater than 2 inches in diameter conditions, with failure of coolant recirculation and delayed melting. Fan coolers failed, CS injection and recirculation success	3.80E-08
3S	Station Blackout sequences (or equivalent equipment failures), (y=3 early melt, y=4 late melt). Fan coolers failed, CS failed	9.33E-06
4S	Station Blackout sequences (or equivalent equipment failures), (y=3 early melt, y=4 late melt). Fan coolers failed, CS failed	1.75E-08
1CB	Bypass sequences with failure of coolant make up (x=1 interfacing systems LOCA, x=2 SGTR).	2.08E-07
2CB	Bypass sequences with failure of coolant make up (x=1 interfacing systems LOCA, x=2 SGTR).	1.20E-07
1CI	Any core melt sequence where the containment is also unisolated.	6.01E-10
TOTAL		1.78E-05

In order to develop the person-rem dose associated with each plant damage state it is necessary to associate each plant damage state with an associated release of radionuclides and from this information to calculate the associate dose.

The IP3 submittal (Reference 2) utilizes a multiplication factor to adjust the design basis leakage value (L_a) that is based on generic information that relates dose to leak size. The CR3 submittal (Reference 4) utilized plant-specific dose estimates based on the predicted level 2 analysis results.

The CPSES PRA (Reference 9) contains the necessary information to convert the plant damage states to release categories. Using this information the plant damage states are mapped to the ten release categories. In addition, the fraction of intact containment cases is determined using the split fraction information contained in Reference 8.

Since the CPSES PRA contains the necessary release fraction information, an approach similar to the CR3 submittal is utilized that better reflects the specific release conditions for CPSES. The CPSES PRA (Reference 8) release categories are defined by the release fraction of major radionuclides. These are extrapolated to dose using the approach presented in Appendix B. This approach has been presented in other licensing submittals (Reference 13) and is consistent with other similar to the method used in the CR3 submittal (Reference 4). The release category dose information is presented in Table 3.

Table 3
Release Category Radionuclide Fraction

Release Category	Frequency	Noble Gas[1]	Iodine[1]	Cesium[1]	Tellurium[1]	Strontium[1]	Total Dose
IC-1	1.50E-05	NA[2]	NA	NA	NA	NA	4.76E+03[3]
I	1.04E-08	8.00E-01	5.00E-02	4.00E-02	4.00E-02	1.00E-03	5.91E+06
II	1.31E-09	1.00E+00	6.00E-02	6.00E-02	0.00E+00	5.00E-04	7.50E+06
III	1.20E-07	2.00E-01	3.00E-03	2.00E-03	0.00E+00	1.00E-05	5.50E+05
IV	1.88E-08	5.00E-02	2.00E-04	2.00E-04	0.00E+00	5.00E-07	9.50E+04
V	5.99E-09	1.00E+00	2.00E-01	2.00E-01	7.00E-01	1.00E-03	2.50E+07
VI	1.19E-06	9.00E-01	3.00E-02	2.00E-02	8.00E-03	7.00E-06	3.89E+06
VII	5.37E-09	1.00E+00	7.00E-07	5.00E-06	1.00E-09	5.00E-13	1.50E+04
VIII	1.07E-06	8.00E-01	1.00E-10	1.00E-09	1.00E-09	2.00E-13	1.20E+06
IX	1.90E-08	9.00E-01	9.00E-03	9.00E-03	0.00E+00	3.00E-04	2.25E+06
X	0.00E+00	8.00E-01	2.00E-03	2.00E-03	0.00E+00	2.00E-05	1.40E+06
XI	2.04E-07	1.00E+00	8.00E-01	8.00E-01	9.00E-01	1.00E-01	8.65E+07
XII	1.20E-07	9.00E-01	8.00E-01	8.00E-01	3.00E-02	3.00E-03	8.15E+07
XIII	6.01E-10	1.00E+00	8.00E-01	8.00E-01	9.00E-01	1.00E-01	8.65E+07

1. Contributing fission product groups are discussed in Appendix B.
2. Release fractions not necessary for this calculation.
3. Intact containment representing design basis leakage (developed in Appendix C).

Other inputs to this calculation include ILRT test data from NUREG-1493[10] and the EPRI report (Reference 6) and are referenced in the body of the calculation.

3.0 ASSUMPTIONS

1. The maximum containment leakage for EPRI Class 1 (Reference 6) sequences is 1 L_a (Type A acceptable leakage) because a new Class 3 has been added to account for increased leakage due to Type A inspections.

2. The maximum containment leakage for Class 3a (References 2 and 4) sequences is 10 L_a based on the previously approved methodology (References 2 and 3).

3. The maximum containment leakage for Class 3b (References 2 and 4) sequences is 35 L_a

based on the previously approved methodology (References 2 and 3).

4. Class 3b is conservatively categorized LERF based on the previously approved methodology (References 2 and 3)

5. Containment leakage due to EPRI Classes 4 and 5 are considered negligible based on the previously approved methodology (References 2 and 3).

6. The containment releases are not impacted with time.

7. The containment releases for EPRI Classes 2, 6, 7 and 8 are not impacted by the ILRT Type A Test frequency. These classes already include containment failure with release consequences equal or greater than those impacted by Type A.

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

9. The information provided in Reference 9 does not identify any contribution for release category X. The CPSES IPE results do populate this release category. Since the total release category frequency presented in Reference 9 does equal the core damage presented in Reference 9, it is considered to be internally consistent and any former release category X contribution is assumed to be allocated to other populated release categories.

4.0 CALCULATIONS

This calculation applies the CPSES PRA release category information in terms of frequency and person-rem estimates to estimate the changes in risk due to increasing the ILRT test interval. The changes in risk are assessed consistent with the previously approved methodology used by Indian Point 3[2,3] and Crystal River 3[4]. This approach is similar to that presented in EPRI TR-104285[6] and NUREG-1493[10]. Namely, the analysis performed examined CPSES PRA plant specific results in which the containment integrity remains intact or the containment is impaired.

4.1 CALCULATIONAL STEPS

The analysis is based on guidance provided in Reference 6 and uses risk metrics presented in Reference 7 to evaluate the impact of a proposed change on plant risk. References 2 and 4 utilize several measures in their assessments. These measures are: change in release frequency, change in risk as defined by the change in person-rem, the change in LERF and the change in the conditional containment failure probability.

Reference 7 also lists the change in core damage frequency as a measure to be considered. Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The overall process is outlined below:

- Define baseline plant damage states and person-rem estimates
- Calculate baseline Type A leakage estimate to define the analysis baseline
- Modify Type A leakage estimate to address extension of the Type A test frequency
- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics

The first step in the analysis is to define the baseline plant damage states and person-rem dose measures. Plant damage state information is developed using the CPSES PRA (References 8 and 9) results. The plant damage state information and the results of the containment analysis are used to define the sequences. The population person-rem dose estimates for each key plant damage states are based on the application of the method described in Appedix B and design basis information[12].

The product of the person-rem for the key plant damage states by the frequency of the key plant damage state estimates the annual person-rem estimate for the plant damage state. Summing these estimates produces the annual person-rem dose based on the sequences defined in the PRA.

The PRA plant damage state definitions consider isolation failures due to Type B and Type C faults and examine containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into key plant damage states. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the plant damage state was then classified by release category definitions specified in Reference 6. With this information developed, the PRA baseline model is completed.

The second step expands the baseline model to address Type A leakage. The PRA did not directly address Type A (liner-related) faults and this contribution must be added to provide a complete baseline. In order to define leakage that can be linked directly to the Type A testing, it is important that only failures that would be identified by Type A testing exclusively be included.

Reference 6 provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the CPSES PRA to develop a baseline model including Type A faults.

The release, in terms of person-rem, is developed based on information contained in Reference 6 and is estimated as a leakage increase relative to allowable release L_a defined as part of the ILRT.

The predicted probability of Type A leakage is then modified to address the expanded time between testing. This is accomplished by a ratio of the existing testing interval and the proposed test interval. This assumes a constant failure rate and that the failures are randomly dispersed during the interval between the test.

The change due to the expanded interval is calculated and reported in terms of the change in release due to the expanded testing interval, the change in the population person-rem and the change in large early release frequency. The change in the conditional containment failure probability is also developed. From these comparisons, a conclusion is drawn as to the risk significance of the proposed change.

Using this process, the following were performed:

1. Map the Level 3 release categories into the 8 release classes defined by the EPRI Report (Reference 6)

2. Calculate the Type A leakage estimate to define the analysis baseline

3. Calculate the Type A leakage estimate to address the current inspection frequency

4. Modify the Type A leakage estimates to address extension of the Type A test interval

5. Calculate increase in risk due to extending Type A inspection intervals

6. Estimate the change in LERF due to the Type A testing.

7. Estimate the change in conditional containment failure probability due to the Type A testing.

4.2 SUPPORTING CALCULATIONS

Step 1: Map the Level 3 release categories into the 8 release classes defined by the EPRI Report

EPRI Report TR-104285 defines eight (8) release classes as presented in Table 4.

Table 4
Containment Failure Classifications (from Reference 6)

Failure Classification	Description	Interpretation for Assigning CPSES Release Category
1	Containment remains intact with containment initially isolated	Intact containment bins
2	Dependent failure modes or common cause failures	Isolation faults that are related to a loss of power or other isolation failure mode that is not a direct failure of an isolation component
3	Independent containment isolation failures due to Type A related failures	Isolation failures identified by Type A testing
4	Independent containment isolation failures due to Type B related failures	Isolation failures identified by Type B testing
5	Independent containment isolation failures due to Type C related failures	Isolation failures identified by Type C testing
6	Other penetration failures	Other faults not previously identified
7	Induced by severe accident phenomena	Early containment failure sequences as a result of hydrogen burn or other early phenomena
8	Bypass	Bypass sequence or SGTR

Table 5 presents the CPSES release category mapping for these eight accident classes. Person-rem per year is the product of the frequency and the person-rem.

Table 5
CPSES PRA Release Category Grouping to EPRI Classes (as described in Reference 6)

Class	Description	Release Category	Frequency	Person-Rem	Person-Rem/yr
1	No Containment Failure	IC-1	1.50E-05	4.76E+03	7.16E-02
2	Large Containment Isolation Failures	None	ε		
3a	Small Isolation Failures (Liner breach)	None	Not Addressed		0.00E+00
3b	Large Isolation Failures (Liner breach)	None	Not Addressed		0.00E+00
4	Small isolation failures - failure to seal (type B)	None	ε		
5	Small isolation failures - failure to seal (type C)	None	ε		
6	Containment Isolation Failures (dependent failure, personnel errors)	XIII	6.01E-10	8.15E+07	4.90E-02
7	Severe Accident Phenomena Induce Failure (Early and Late)	All other Release Categories	2.44E-06	2.50E+07	6.10E+01
8	Containment Bypass	XI, XII	3.24E-07	8.65E+07	2.81E+01
		Total	1.78E-05		8.91E+1

Step 2: Calculate the Type A leakage estimate to define the analysis baseline (3 year test interval)

As displayed in Table 5 the CPSES PRA did not identify any release categories specifically associated with EPRI Classes 3, 4, or 5. Therefore each of these classes must be evaluated for applicability to this study.

Class 3:

Containment failures in this class are due to leaks such as liner breaches that could only be detected by performing a Type A ILRT.

Reference 3 states that a review of experience data finds that Type A testing identified only 4 leakage events of the 144 events identified. Thus about 3% (0.028) of containment leakage events are identified by the ILRT. The remaining events were identified by LLRT (Type B and C testing) and are not included in the analysis. This probability, however, is based on three tests over a 10-year period and not the one per ten-year frequency currently employed at CPSES (Reference 1). The probability (0.028) must be adjusted to reflect this difference.

For this estimation, the question on containment isolation was modified consistent with the previously approved methodology (References 2 and 3), to include the probability of a liner breach (due to excessive leakage) at the time of core damage.

Class 3 is divided into two classes using this approach. Class 3a is defined as a small liner breach and Class 3b is defined as a large liner breach.

Calculation of Class 3b Probability

To calculate the probability that a liner leak will be large (Class 3b), use was made of the data presented in NUREG-1493 (Reference 10). One data set found in NUREG-1493 reviewed 144 ILRTs. 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, 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 χ^2 distribution. This is consistent with the Indian Point 3 (Reference 2 and Crystal River 3 (Reference 4) templates. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests (See Reference 11). 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, ν. 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 using the following equation:

$$p(\alpha) = \frac{\chi^2(2F+2,\alpha)}{2N}$$

where: N is the number of events, F is the number of events (faults) of interest, α is the percentile distribution (typically assumed to be the 95%-tile). The result of 2F+2 defines the degree of freedom.

Given that there have been no large leaks (n = 0, therefore ν =2) in 144 events (N = 144) the value of $\chi^2(2, 0.05)$ is equal to 5.99. Solving for the 95th percentile estimate of the probability of a large leak yields 0.021 as presented below:

$$p_{Class3B} = \frac{\chi^2(2,0.05)}{2 \bullet 144} = \frac{5.99}{288} = 0.021$$

Calculation of Class 3a Probability

The data presented in NUREG-1493 (Reference 10) is also used to calculate the probability that a liner leak will be small (Class 3a). 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 $1.0L_a$. However, of the 23 events that exceeded the test requirements, only 4 were found by an ILRT, the others were found by Type B and C testing or errors in test alignments.

Therefore, a best estimate for the probability of leakage is ~0.03 (4-of-144). However, the Class 3a probability is estimated using the conservative χ^2 distribution approach described previously. This is consistent with the approach taken in References 2, 3 and 4.

The χ^2 distribution is calculated by F=4 (number of small leaks) and N=144 (number of events) which yields a solution as shown below:

$$p_{Class3A} = \frac{\chi^2(10,0.05)}{2 \bullet 144} = \frac{18.307}{288} = 0.064$$

Therefore, the 95th percentile estimate of the probability of a small leak (Class 3a) is calculated as 0.064.

The probability of liner failures must then be multiplied by an appropriate accident frequency to determine the Class 3A and Class 3B frequencies. The IP3 (Reference 2) and CR3 (Reference 4) submittals utilized the entire core damage frequency when developing the contributions for Classes 3A and 3B and then adjusted the Class 1 contribution.

This is somewhat conservative since it does provide the maximum possible contributions due to the extension of the ILRT testing interval. This approach is maintained for the CPSES analysis, in order to be consistent with the approved methodology.

Therefore the frequency of a Class 3b failure is calculated as:

$FREQ_{class3b} = PROB_{class3b}$ x CDF = 0.021 x 1.78E-5/yr = 3.70E-7/yr

Therefore the frequency of a Class 3a failure is calculated as:

$FREQ_{class3a} = PROB_{class3a}$ x CDF = 0.064 x 1.78E-5 = 1.13E-6/yr

Class 4:

This group consists of all core damage accident accidents for which a failure-to-seal containment isolation failure of Type B test components occurs. By definition, these failures are dependent on Type B testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

Class 5:

This group consists of all core damage accident accidents for which a failure-to-seal containment isolation failure of Type C test components occurs. By definition, these failures are dependent on Type C testing, and Type A testing will not impact the probability. Therefore this group is not evaluated any further, consistent with the approved methodology.

Class 6:

The Class 6 group is comprised of isolation faults that occur as a result of the accident sequence progression. The leakage rate is not considered large by the PRA definition and therefore it is placed into Class 6 to represent a small isolation failure and identified in Table 6 as Class 6.

$FREQ_{class6.}$=6.01E-10/yr

Class 1:

Although the frequency of this class is not directly impacted by Type A testing, the PRA did not model Class 3 failures, and the frequency for Class 1 should be reduced by the estimated frequencies in the new Class 3a and Class 3b in order to preserve the total CDF. The revised Class 1 frequency is therefore:

$FREQ_{class1} = FREQ_{class1} – (FREQ_{class3a} + FREQ_{class3b})$

$FREQ_{class1}$ = 1.50E-5/yr – (1.1319E-6/yr + 3.7045E-7/yr) = 1.35E-5/yr

Class 2:

The CPSES PRA did not identify any contribution to this group above the quantification truncation.

Class 7:

The frequency of Class 7 is the sum of those release categories identified in Table 6 as Class 7.

$FREQ_{class7}$ = 2.44E-6/yr

Class 8:

The frequency of Class 8 is the sum of those release categories identified in Table 6 as Class 8.

$FREQ_{class8}$ = 3.24E-7/yr

Table 6 summarizes the above information by the EPRI defined classes. This table also presents dose exposures calculated using the methodology described in Reference 9. For Class 1, 3a and 3b, the person-rem is developed based on the design basis assessment of the intact containment. This assumes operation of the standby gas treatment system and any losses from the primary containment being transported to the environment through the stack. The Class 3a and 3b doses are represented as $10L_a$ and 35 L_a respectively. Table 6 also presents the person-rem frequency data determined by multiplying the failure class frequency by the corresponding exposure.

Table 6
Baseline Risk Profile

Class	Description	Frequency (/yr)	Person-rem (calculated)[1]	Person-rem (from L_a factors)	Person-rem (/yr)
1	No Containment Failure	1.35E-5		4.76E+3[2]	6.45E-2
2	Large Containment Isolation Failures	ε			
3a	Small Isolation Failures (Liner breach)	1.13E-6		4.76E+4[3]	5.39E-2
3b	Large Isolation Failures (Liner breach)	3.7E-7		1.67E+5[4]	6.17E-2
4	Small isolation failures - failure to seal (type B)	ε			
5	Small isolation failures - failure to seal (type C)	ε			
6	Containment Isolation Failures (dependent failure, personnel errors)	6.01E-10	8.15E+7		4.90E-2
7	Severe Accident Phenomena Induce Failure (Early and Late)	2.44E-6	2.50E+7[5]		6.10E+1
8	Containment Bypass	3.24E-7	8.65E+7		2.81E+01
	Totals	1.78E-5			8.92E+1

1. From Table 3 using the method presented in Appendix B.
2. 1 L_a dose value calculated in Appendix C.
3. 10 times L_a
4. 35 times L_a
5. Maximum dose from contributing release categories.

The percent risk contribution due to Type A testing is as follows:

$\%Risk_{BASE} = [(Class3a_{BASE} + Class3b_{BASE}) / Total_{BASE}] \times 100$

Where:

$Class3a_{BASE}$ = Class 3a person-rem/year = 5.39E-2 person-rem/year

$Class3b_{BASE}$ = Class 3b person-rem/year = 6.17E-2 person-rem/year

$Total_{BASE}$ = total person-rem year for baseline interval = 89.247 person-rem/year (Table 6)

$\%Risk_{BASE}$ = [(5.39E-2 + 6.17E-2) / 89.247] x 100 = **0.1295%**

Step 3: Calculate the Type A leakage estimate to address the current inspection interval

The current surveillance testing requirements as proposed in NEI 94-01 (Reference 5) for Type A testing and allowed by 10 CFR 50, Appendix J 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 $1.0L_a$).

According to NUREG-1493 (Reference 10), extending the Type A ILRT interval from 3-in-l0 years to 1-in-l0 years will increase the average time that a leak detectable only if an ILRT goes undetected from 18 to 60 months. Multiplying the testing interval by 0.5 and multiplying by 12 to convert from "years" to "months" calculates the average time for an undetected condition to exist.

Since ILRTs only detect about 3% of leaks (4/144) that are not detected by other local tests, the increase for a 10-yr ILRT interval is the ratio of the average time for a failure to detect for the increased ILRT test interval (60 months) to the baseline average time for a failure to detect of 18 months (i.e., 0.03 x 60/18 = 0.10). References 2 and 4 indicate this is a 10% increase in the likelihood of a Type A leak.

Risk Impact due to 10-year Test Interval

Based on the previously approved methodology (References 2 and 3), the increased probability of not detecting excessive leakage due to Type A tests directly impacts the frequency of the Class 3 sequences. Consistent with Reference 2 and 4 the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in the probability of leakage (1.1 x Class 3 baseline). The results of this calculation are presented in Table 7 below.

Table 7
Risk Profile for Once in Ten Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No Containment Failure[1]	1.34E-5	4.76E+3	6.37E-2
2	Large Containment Isolation Failures	ε		
3a	Small Isolation Failures (Liner breach)	1.25E-6	4.76E+4	5.93E-2
3b	Large Isolation Failures (Liner breach)	4.07E-7	1.67E+5	6.79E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment Isolation Failures (dependent failure, personnel errors)	6.01E-10	8.15E+7	4.90E-02
7	Severe Accident Phenomena Induce Failure (Early and Late)	2.44E-6	2.50E+7	6.10E+01
8	Containment Bypass	3.24E-7	8.65E+7	2.81E+1
	Total	1.78E-5		8.93E+01

1. The IPE frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.

Using the same methods as for the baseline, and the data in Table 7 the percent risk contribution due to Type A testing is as follows:

$\%Risk_{10} = [(Class3a_{10} + Class3b_{10}) / Total_{10}] \times 100$

Where:

$Class3a_{10}$ = Class 3a person-rem/year = 5.927E-2 person-rem/year

$Class3b_{10}$ = Class 3b person-rem/year = 6.789E-2 person-rem/year

$Total_{10}$ = total person-rem year for current 10-year interval = 89.258 person-rem/year (Table 7)

$\%Risk_{10}$ = [(5.927E-2 + 6.789E-2) / 89.258] x 100 = **0.1425%**

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-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 person-rem/year for baseline interval = 89.247 person-rem/year (Table 6)

$Total_{10}$ = total person-rem/year for 10-year interval = 89.258 person-rem/year (Table 7)

$\Delta\%Risk_{10} = [(89.258 – 89.247) / 89.247] \times 100.0 =$ **0.012%**

Step 4: Calculate the Type A leakage estimate to address extended inspection intervals

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 (0.5 x 15 x 12). For a 15-yr-test interval, the result is the ratio (0.03 x 90/18) of the exposure times. Thus, increasing the ILRT test interval from 10 years to 15 years results in a proportional increase in the overall probability of leakage. The approach for developing the risk contribution for a 15-year interval is the same as that for the 10-year interval. References 2 and 4 indicate that the increase is a 50% increase from that for the 10-year interval or a 15% increase from the baseline. Different values are provided for the probability of leakage. In addition, the containment leakage used for the 10-year test interval for Class 3 is used in the 15-year interval evaluation (1.15 x Class 3 baseline). The results for this calculation are presented in Table 8.

Table 8
Risk Profile for Once in Fifteen Year Testing

Class	Description	Frequency (/yr)	Person-rem[2]	Person-rem (/yr)
1	No Containment Failure[1]	1.33E-5	4.76E+3	6.34E-2
2	Large Containment Isolation Failures	ε		
3a	Small Isolation Failures (Liner breach)	1.30E-6	4.76E+4	6.20E-2
3b	Large Isolation Failures (Liner breach)	4.26E-7	1.67E+5	7.10E-2
4	Small isolation failures - failure to seal (type B)	ε		
5	Small isolation failures - failure to seal (type C)	ε		
6	Containment Isolation Failures (dependent failure, personnel errors)	6.01E-10	8.15E+7	4.90E-2
7	Severe Accident Phenomena Induce Failure (Early and Late)	2.44E-6	2.50E+7	6.10E+1
8	Containment Bypass	3.24E-7	8.65E+7	2.81E+1
	Total	1.78E-5		8.93E+1

1. The IPE frequency of Class 1 has been reduced by the frequency of Class 3a and Class 3b in order to preserve total CDF.
2. From Table 6.

Using the same methods as for the baseline, and the data in Table 10 the percent risk contribution due to Type A testing is as follows:

$\%Risk_{15} = [(Class3a_{15} + Class3b_{15}) / Total_{15}] \times 100$

Where:

$Class3a_{15}$ = Class 3a person-rem/year = 6.196E-2 person-rem/year

$Class3b_{15}$ = Class 3b person-rem/year = 7.097E-2 person-rem/year

$Total_{15}$ = total person-rem year for 15-year interval = 89.263 person-rem/year (Table 8)

$\%Risk_{15}$ = [(7.097E-2 + 6.196E-2) / 89.263] x 100 = **0.1489%**

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

$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0$

Where:

$Total_{BASE}$ = total person-rem/year for baseline (3 per 10 years) interval = 89.247 person-rem/year (Table 6)

$Total_{15}$ = total person-rem/year for 15-year interval = 89.263 person-rem/year (Table 8)

$\Delta\%Risk_{15} = [(89.263 - 89.247) / 89.247] \times 100.0 =$ **0.018%**

Step 5: Calculate increase in risk due to extending Type A inspection intervals

Based on the previously approved methodology (Reference 2 and 4), the percent increase in risk (in terms of person-rem/yr) of these associated specific sequences is computed as follows.

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

Where:

$PER\text{-}REM_{10}$ = person-rem/year of ten years interval (see Table 7, classes 1, 3a and 3b)

= 0.1909 person-rem/yr

$PER\text{-}REM_{15}$ = person-rem/year of fifteen years interval (see Table 8, classes 1, 3a and 3b)

= 0.1963 person-rem/yr

$\%Risk_{10\text{-}15} = [(0.1963 - 0.1909) / 0.1909] \times 100 =$ **2.84%**

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

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

Where:

$Total_{10}$ = total person-rem/year for 10-year interval

= 89.258 person-rem/year (Table 7)

$Total_{15}$ = total person-rem/year for 15-year interval

= 89.263 person-rem/year (Table 8)

$\% Total_{10\text{-}15} = [(89.263 - 89.258) / 89.258] \times 100 =$ **0.006%**

Step 6: Calculate 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 larger release due to failure to detect a pre-existing leak during the relaxation period.

From references 2 and 4, the Class 3A dose is assumed to be 10 times the allowable intact containment leakage, L_a (or 46,700 person-rem) and the Class 3B dose is assumed to be 35 times L_a (or 167,000 person-rem). The dose equivalent for allowable leakage (L_a) is developed in Appendix C. This compares to a historical observed average of twice L_a. Therefore, the estimate is somewhat conservative.

Based on the previously approved methodology (References 2 and 4), 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 $2L_a$). A larger 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 CPSES PRA (Reference 9) that result in large releases, are not impacted because a LERF will occur regardless of the presence of a pre-existing leak. Therefore, the frequency of Class 3b sequences is used as the increase in LERF for CPSES, and the change in LERF can be determined by the differences. References 2 and 4 identify that Class 3B is considered to be a contributor to LERF. The assumed dose for this class is compared to other LERF sequences to determine if it truly represents an increase in LERF. In order to be a LERF sequence, it must be both early in time and large in population dose. The first condition is met since the failure represents an existing isolation failure. However, the dose is small compared to other early sequences. Table 9 compares the doses for this and several other cases.

Table 9
Comparisons of Release Class Doses

Release Class	Population Dose (Person-rem)
Class 3B (Table 6)	167,000
Class 8 (Table 6)	86,500,000
Class 7 (Table 6)	25,000,000

The table shows that even a conservative estimate for the release (person-rem) is found to be on the order of 1% of that obtained from other early release classes. On a best-estimate basis the average expected leakage would be less than 10,000 person-rem and would be much less than one percent of the other classes associated with large early release. The conclusion can be drawn from this data that the potential consequence of a Type A leakage event is not large and the proposed change has no impact on LERF. However, conservatively Class 3B is considered to be an estimate for the change in LERF to be consistent with the accepted methodology (References 2 and 4). Table 10 summarizes the results of the LERF evaluation assuming that Type 3B is indicative of a LERF sequence.

Table 10
Impact on LERF due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
Class 3B (Type A LERF)	3.7E-7 (see Table 6)	4.07E-7 (see Table 7)	4.26E-7 (see Table 8)
ΔLERF (3 year baseline)		3.70E-8	5.56E-8
ΔLERF (10 year baseline)			1.85E-8

Reg. Guide 1.174 (Reference 7) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-6/yr and increases in LERF below 1E-7/yr. Since the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0E-7/yr, increasing the ILRT interval to 15 years (**1.85E-8/yr**) is non-risk significant. It should be noted that if the risk increase is measured from the original 3-in-10-year interval, the increase in LERF is **5.56E-8/yr**, which is also below the 1.0E-07/yr screening criterion in Reg.Guide 1.174.

Step 7: Calculate the change in Conditional Containment Failure Probability (CCFP)

The conditional containment failure probability (CCFP) is defined as the probability of containment failure given the occurrence of an accident. This probability can be expressed using the following equation:

$$CCFP = 1 - \left[\frac{f(ncf)}{CDF}\right]$$

Where *f(ncf)* is the frequency of those sequences which result in no containment failure. This frequency is determined by summing the Class 1 and Class 3a results, and CDF is the total frequency of all core damage sequences.

Therefore the change in CCFP for this analysis is the CCFP using the results for 15 years ($CCFP_{15}$) minus the CCFP using the results for 10 years ($CCFP_{10}$). This can be expressed by the following:

$$\Delta CCFP_{10-15} = CCFP_{15} - CCFP_{10}$$

Using the data previously developed the change in CCFP from the current testing interval is calculated and presented in Table 11.

Table 11
Impact on Conditional Containment Failure Probability due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf)	1.4673E-5	1.4636E-5	1.4618E-5
f(ncf)/CDF	0.824	0.822	0.821
CCFP	0.176	0.178	0.179
ΔCCFP (3 year baseline)		0.208%	0.312%
ΔCCFP (10 year baseline)			0.104%

5.0 REFERENCES

1. Comanche Peak Steam Electric Station Unit 1, Containment Integrated Leak Rate Test, Rev. 0, TX, PPT-S1-7014, October 1993.

2. Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", Entergy, IPN-01-007, January 18, 2001.

3. Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MBO178), United States Nuclear Regulatory Commission, April 17, 2001.

4. Evaluation of Risk Significance of ILRT Extension, Revision 2, Florida Power Corporation, F-01-0001, June 2001.

5. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.

6. Gisclon, J. M., et al, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, Electric Power Research Institute, TR-104285, August 1994.

7. An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

8. Individual Plant Examination Submittal Comanche Peak Steam Electric Station, Volumes 1 and 2, TU Electric, October 1992.

9. Tirsun, D., Comanche Peak Steam Electric Station, Level 2 PRA Release Category Quantification, Rev. 2, TXU Electric, R&R-PN-036, May 2001.

10. Performance-Based Containment Leak-Test Program, USNRC, NUREG-1493, July 1995.

11. PRA Procedures Guide- A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, American Nuclear Society and the National Institute of Electrical and Electronic Engineers, NUREG/CR-2300, January 1983.

12. Comanche Peak Steam Electric Station FSAR, Section 15.6 and Section 2.1.3, TXU Electric, Amendment 96, August 2,1999.

13. Summitt, R., Assessment of Safety Benefit for Installation of a Generator Disconnect Switch at Robinson, RSC, Inc., RSC 98-19, June 1998.

Appendix A

Alternative Calculation for ILRT Evaluation

A.0 INTRODUCTION

In addition to the approved template, other analyses approaches are possible that rely on a best-estimated approach and that avoid some potential weakness identified below:

- The calculation of the increase in the probability of Type A leakage is incorrect and equates a probability with a percentage increase.

- The approach utilizes upper bound (95%-tile) values and not best-estimate mean values when developing the estimates for the likelihood of Type A leakage events although adequate data exists.

- The approach arbitrarily increases the probability of leakage for the intact containment cases by a factor of two although the Type A leakage is assessed separately and the intact containment cases would still be isolated.

- The approach increases the intact containment source term by factors of 1.1 and 1.15 for the 10-and 15-year cases although the source term is a physical process that would not be altered by changing the statistic of testing.

- Although historical data indicates that the mean value for leakage is on the order of twice allowable (2 x L_a) the analysis arbitrarily utilizes factors of $10L_a$ and $35L_a$ which tends to overestimate the initial dose and thereby mask some of the predicted increase in risk when considered in terms of delta change.

- The analysis assumes that Type 3A and Type 3B are independent when developing the upper bound estimates and utilizes the 144 events as two separate populations. Actually the Type 3B leakage events are a proportion of the Type 3A events and should be calculated in a dependent manner. The embedded independence assumption results in an arbitrary increase in the likelihood of leakage by over a factor of two when compared to actual historical evidence. This overestimation inflates the baseline calculation and reduces the net change predicted in a non-conservative manner.

The approach documented in this appendix addresses these weaknesses by providing a best-estimate approach. Although the approach is similar to the template there are key differences that are highlighted as appropriate. The same input data (References 6, 8, 9 and 10) are used to generate release category frequency information.

The man-rem information is based on the approach found in Appendix B.

Section A.1 of the document presents a summary of the analysis steps. Section A.2 presents the baseline analysis. Sections A.3 and A.4 develop the impact of the increased testing interval on the analysis metrics. Section A.5 presents a summary of the analysis.

A.1 ANALYSIS APPROACH

The analysis is based on guidance provided in Reference 6 and uses risk metrics presented in Reference 10 to evaluate the impact of a proposed change on plant risk. Finally, Reference 7 suggests two measures be utilized in the assessment, core damage frequency and LERF. It is these two metrics that are assessed first.

These measures are considered in the determination of the impact of testing extension. The Comanche Peak Steam Electric Station (CPSES) is currently considering an extension from 10 years to 15 years.

Since the testing addresses the ability of the containment to maintain its function, the proposed change has no measurable impact on core damage frequency. Therefore, this attribute remains constant and has no risk significance.

The change in testing interval could impact the ability of the containment to perform its function and this could impact the LERF attribute. Therefore, the estimated change in LERF is addressed.

The change in risk, as defined by the change in annual population man-rem dose is calculated. This metric provides a means to identify the increased risk posed by the change in testing interval.

The basic analysis steps are outlined below:

- Define baseline plant damage states and man-rem estimates
- Calculate baseline Type A leakage estimate to define the analysis baseline
- Modify Type A leakage estimate to address extension of the Type A test frequency
- Compare analysis metrics to estimate the impact and significance of the increase related to those metrics

The first step in the analysis is to define the baseline plant damage states and man-rem dose measures. Plant damage state information is developed in Section 2.0 of the main report. The plant damage state information and the release fraction information (Table 3 of main report) are used to develop the population man-rem dose estimates.

The CPSES PRA (Reference 8 and 9) plant damage state definitions include isolation failures due to Type B and Type C faults and examine containment challenges occurring after core damage and/or reactor vessel failure. These sequences are grouped into plant damage states. Using the plant damage state information, bypass, isolation failures and phenomena-related containment failures are identified. Once identified, the sequence was then classified by release category definitions specified in Reference 6 and summarized in Table 5 of the main report.

The second step expands the baseline model to address Type A leakage. The CPSES PRA did not explicitly include Type A (liner-related) faults and this contribution must be added to provide a complete baseline. In order to define leakage that can be linked directly to the Type A testing, it is important that only failures that would be identified by Type A testing exclusively be included.

Reference 10 provides the estimate for the probability of a leakage contribution that could only be identified by Type A testing based on industry experience. This probability is then used to adjust the intact containment category of the CPSES PRA to develop a baseline model including Type A faults.

The release, in terms of man-rem, is developed based on information contained in Reference 10 and is estimated as a leakage increase relative to allowable release La defined as part of the ILRT.

The predicted probability of Type A leakage is then modified to address the expanded time between testing. This is accomplished by a ratio of the existing testing interval and the proposed test interval. This assumes a constant failure rate and that the failures are randomly dispersed during the interval between the test.

The change due to the expanded interval is calculated and reported in terms of the change in population man-rem. In addition, the change in large early release frequency is predicted and compared to the acceptance criteria presented in Reference 7.

From these comparisons, a conclusion is drawn as to the risk significance of the proposed change.

A.2 DEFINITION OF ACCIDENT SEQUENCES

The CPSES PRA (Reference 8 and 9) provides the baseline core damage bin frequency information for the contributing accident sequences. The assessment includes internal initiating events and the total core damage frequency is estimated to be 1.78E-5/yr. Table 2 in the main report presents a summary of this information.

A.3 CALCULATION OF INCREASE IN TYPE-A RELATED LEAKAGE

In order to determine the impact of the change in testing interval, it is first necessary to define a baseline probability for Type A leakage events and then to adjust this probability to account for the proposed change in testing interval.

Reference 10 states that a review of experience data finds that Type A testing identified only 4 leakage events of the 144 events identified. Thus about 3% (0.028) of containment leakage events are identified by the ILRT and that the remaining events are identified by the LLRT that is not being evaluated for change. This probability, however, is based on three tests over a 10 year period and not the one per ten year frequency currently employed at CPSES (Reference 1). The probability (0.028) must be adjusted to reflect this difference.

The impact of relaxing the Type A penetration test interval will increase the average time that a leak that could only be detected by the Type A test would possibly be present. The increase in risk is proportional to the increase in the duration between containment tests. The historical data is based on testing three times per 10 years (120 months). This equates to a mean time between tests of 3.3 years or 40 months. The CPSES testing interval is once per 10 years (120 months). The increase in the exposure time will influence the probability of leakage.

To calculate this impact, two assumptions are made consistent with standard practice and are listed below:

- A constant rate for Type A leakage events;
- The potential for leakage is equally distributed across the period of interest such that the exposure time is reduced by one-half the period.

With these assumptions, the increase can be determined by a ratio of the proposed to the prior exposure times multiplied by the known rate for the prior probability of failure. The equation is shown below:

$$p_{10/1} = p_{10/3} \bullet \frac{0.5 \bullet Exp_{10/1}}{0.5 \bullet Exp_{10/3}}$$

Substituting the values for p10/3 (0.028) and the exposure times (Exp10/1 = 120, Exp10/3 =40) yields a value for the probability of leakage of 0.0833. This value serves as the baseline probability of Type A leakage for the analysis.

The proposed change would increase the duration between tests by decreasing the number of tests from once per 10 years to once per 15 years. Therefore, the total time between Type A testing will increase from ten years (120 months) to 15 years (180 months). The same equation is again utilized with the variables altered to reflect the specific bounds as shown below:

$$p_{16} = p_{10} \bullet \frac{0.5 \bullet Exp_{15}}{0.5 \bullet Exp_{10}}$$

Substituting yields a value for the probability of Type A (ILRT) detectible leakage events for the relaxed testing interval of 15 years. This probability represents the probability of a leakage path that could only be identified by a Type A test. The results are summarized below in Table A.1.

Table A.1
Probability of Type A Leakage Given a Testing Interval

Case	Probability of Leakage
Baseline (once per 10 years)	0.0833
15 year testing	0.125

The baseline analysis must include the consideration being assessed in order to preclude biasing the results. The existing analysis does not account for Type A faults. The model is expanded to encompass Type A faults. The intact containment cases are adjusted to include these isolation failures. Since other sequences represent failure they are not adjusted.

The intact containment frequency for the baseline (1.50E-5) is multiplied by the potential for Type A leakage (0.0833) for the baseline case to generate the frequency contribution associated with Type A leakage (1.88E-6/yr). The intact containment contribution is then reduced by this value to maintain the overall frequency (1.31E-5). Table A.2 summarizes the results for the extended case.

Table A.2
Type A Leakage Frequency

Variable	10 year period case	15 year period case
Testing Interval (years)	10	15
Intact Containment Frequency (class 1)	1.50E-5	1.50E-5
Baseline Probability Type A Fault	8.33%	8.33%
Extension (years/test)	0	5
Modified Probability of Type A Faults	8.33%	12.5%
Contribution from Type A faults	1.25E-6	1.88E-6
Revised Intact Containment Freq (class 1)	1.38E-5	1.31E-5

In order to develop the estimate for man-rem, it is necessary to determine the magnitude of the expected release for Type A leakage. Information in Reference 10 indicates that the typical leakage from a Type A failure is on the order of 2La where La represents the allowable leakage rate.

Information in Appendix C estimates the intact containment contribution as 4.76E+3 rem. The dose rate from this release category is selected and then doubled to define the expected release for the Type A leakage of 9.52E+3 man-rem (2 x 4.67E+3).

This new release category must be compared to other sequences to determine if it represents an increase in LERF. In order to be a LERF sequence, it must be both early in time and large in population dose. The first condition is met since the failure represents an existing isolation failure. However, the dose is small compared to other early sequences. Table A.3 compares the doses for this and several other cases.

Table A.3
Comparison of Release Class Doses

Release Class	Population Dose (Man-Rem)
Class 7 (Section 4.2, Table 6)	25,000,000
Class 8 (Section 4.2, Table 6)	86,500,000
Type A (Class 3B	9,520

The Type A man-rem estimate is shown to be substantially (less than 1%) of the other early releases. Therefore, the potential consequence is not a LERF sequence and the proposed change has no impact on LERF.

No examples of Type A leakage sufficient to be considered LERF are identified in the historical data. However, an estimate for LERF is calculated for comparison using the Reg. Guide 1.174 risk criterion for change in LERF and is developed. The information provided in References 6 and 10 indicate that the historical leakage rates are not sufficient to result in a situation defined as large-early release. Therefore, the current data supports the supposition that the relaxed testing interval will not have a measurable impact on large early release fraction (LERF).

However, the data does not preclude events that may occur at a lower probability than would be supported by the data collected to date. As a sensitivity study, the probability of larger leakage rates is conservatively estimated. Using the estimated probability and the plant damage state information presented in the CPSES PRA an estimate of LERF is defined for both the baseline model and the 15-year frequency.

The LERF contribution is based on estimation of the frequency through the use of a chi-square distribution to develop an upper estimate as defined by Reference 11. The chi-square distribution can provide an upper bound given no events. The general equation is presented below:

$$p(\alpha) = \frac{\chi^2(2F+2,\alpha)}{2N}$$

where: N is the number of events, F is the number of events of interest, a is the percentile distribution (assumed to be the 95%-tile).

This equation is used to estimate the probability that, given a leak, it will be sufficiently large to represent a LERF contributor. For this estimate, the following is supplied: N=144 events, F=0 LERF events.

Substitution yields the following:

$$p(\alpha) = \frac{\chi^2(2,0.05)}{288}$$

Solving for the probability yields a value of 0.02 (5.99/288). This probability represents a probability that given a leak event, it will be sufficiently large to contribute to LERF. The probability of a Type A failure sufficient to contribute to LERF is found by multiplying the probability of a Type A leak (0.0833) and the probability that the leak will be sufficiently large to generate a LERF contributor (0.02). This equates to a probability of a Type A LERF contributor of 1.67E-3 for the baseline case. For the case involving a 15-year interval, the probability of a leak increases to 0.125. Thus, the probability of a Type A LERF contributor is estimated as 2.50E-3 for the 15-year case.

With the LERF contributor estimated, the next section defines the baseline LERF frequency.

The baseline LERF frequency is defined by collecting any frequency for KPDSs that:

- Involve an early containment isolation failure
- Involve a bypass failure
- Involve early containment failure at or near reactor vessel failure.

Since the potential for containment challenges is addressed in the containment event tree (CET) the last item must be addressed later in the definition of release categories. However, the LERF frequency attributed to the first two conditions can be defined by the PDSs.

Isolation failures must be sufficiently large to preclude any future challenges. For example, an isolation failure must be of sufficient size to mitigate pressure challenges that could occur at reactor vessel failure, e.g., loads from high pressure melt ejection. If this condition is not met, the failure is not sufficient large to meet the definition of a LERF contributor. Reference 9 define a baseline LERF Frequency of 5.31E-7/yr.

A.4 ESTIMATION OF IMPACT ON LERF

The LERF contribution defined in the CPSES PRA does not include the impact of Type A leakage events. This contribution must be added to provide a basis comparison.

The Type A LERF contribution is determined by multiplying the probability of a LERF sequence by the intact containment contribution in the same manner as the Type A sequence contribution was developed. The LERF probability (or split fraction) is determined by multiplying the probability of Type A leakage (0.0833 for the 10 year case) by the probability that the leakage will be large (0.13). The result, 1.085E-2, represents the LERF fraction. This value is then multiplied by the intact containment frequency to obtain the result (1.5E-5 x 1.085E-2) Table A.4 summarizes the calculations for the 10- and 15-year cases.

Table A.4
Type A LERF Contribution

Variable	Baseline	15 year period case
Testing Interval (years)	10	15
Intact Containment Frequency (class 1)	1.50E-5	1.50E+5
Probability Type A LERF Fault, p(LERF)	0.13	0.13
Probability of Type A Leakage, p(TYPEA)	0.0833	0.125
LERF Fraction, LF = p(LERF) x p(TYPEA)	0.01085	0.01625
Type A LERF Frequency, Class 1 x LF	1.63E-7	2.44E-7

The calculated Type A LERF Frequency is added for the baseline and the 15-year case and the difference calculated. This represents the increase in LERF due to the relaxation of the testing interval. The results are presented in Table A.5.

Table A.5
Calculation in the Change in LERF

Variable	Baseline	15 year period case
Baseline LERF	5.31E-7	2.31E-7
Type A LERF Frequency	1.63E-7	2.44E-7
Total LERF	6.94E-7	7.75E-7
Delta LERF		8.15E-8

Reference 3 defines a set of risk significance criteria. The following summarizes the criteria:

- If the calculated increase is very small, which is taken as being less than 10-7 per reactor year, the change is typically considered to be an insignificant increase in risk.

- If the increase is in the range of 10-7 per reactor year to 10-6 per reactor year, proposed change will be considered only if it can be reasonably shown that the total LERF is less than 10-5 per reactor year.

- If the result shows an increase above 10-6 per reactor year, the proposed change would not normally be considered

A comparison of the results to these criteria indicate that the change in LERF frequency is below the level differentiating risk significance and the net change is not risk significant. This result is based on assuming a conservative estimate for the potential for a Type A failure resulting in a LERF sequence. A more realistic value would most likely result in a further reduction in the change in LERF and further support this conclusion.

A.5 MODIFIED MODEL EVALUATION

The information provided in Section 4.0 develops an estimate of the increase in the likelihood of a containment isolation failure given that the Type A ILRT testing interval is extended. An increase in the testing interval to once per 15 years increases the probability of a Type A detectible leakage by 0.04. This increase is used to adjust the baseline model to determine the estimated man-rem.

The baseline model results must be adjusted to address this increase likelihood of increased containment leakage. Only certain sequences would be impacted by this increase since many sequences already involve an impaired containment or isolation failure. Basically only intact containment scenarios need be addressed.

Sequences that already represent release sequences are excluded. The intact containment sequences are combined with the increased probability of leakage (0.04) to define a new contribution to increased leakage. The resulting change in man-rem is summarized below in Table A.6.

Table A.6
Man-Rem Comparisons

Case	Man-Rem/yr	Delta Man-Rem	Percent Increase
Baseline with Type A	89.258		
15-year	89.264	0.006	0.0067%

The net increase in man-rem per year is estimated to be 0.0067% (0.000067) for the 15-year case. The results indicate an increase in integrated man-rem of approximately 6.0E-2 man-rem if the 5-year interval is adopted. The integrated results are presented for all cases in Table A.7.

Table A.7
Integrated Man-Rem Estimates

Case	Man-Rem/yr	Delta Man-Rem	Integrated Man-Rem Increase[1]
Baseline with Type A	89.258		
15-year	89.264	0.006	8.95E-2

1. Net increase relative to the period being assessed for extension.

A.6 REFERENCES

1. Comanche Peak Steam Electric Station Unit 1, Containment Integrated Leak Rate Test, Rev. 0, TX, PPT-S1-7014, October 1993.

2. Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", Entergy, IPN-01-007, January 18, 2001.

3. Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MBO178), United States Nuclear Regulatory Commission, April 17, 2001.

4. Evaluation of Risk Significance of ILRT Extension, Revision 2, Florida Power Corporation, F-01-0001, June 2001.

5. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.

6. Gisclon, J. M., et al, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, Electric Power Research Institute, TR-104285, August 1994.

7. An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

8. Individual Plant Examination Submittal Comanche Peak Steam Electric Station, Volumes 1 and 2, TU Electric, October 1992.

9. Tirsun, D., Comanche Peak Steam Electric Station, Level 2 PRA Release Category Quantification, Rev. 2, TXU Electric, R&R-PN-036, May 2001.

10. Performance-Based Containment Leak-Test Program, USNRC, NUREG-1493, July 1995.

11. PRA Procedures Guide- A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, American Nuclear Society and the National Institute of Electrical and Electronic Engineers, NUREG/CR-2300, January 1983.

12. Comanche Peak Steam Electric Station FSAR, Section 15.6 and Section 2.1.3, TXU Electric, Amendment 96, August 2,1999.

Appendix B

Surrogate Person-Rem Methodology
(RSC 01-44)

[This appendix contains proprietary information]

Appendix C

Development of Intact Containment Person-Rem Estimate

C.0 ESTIMATION OF INTACT CONTAINMENT PERSON-REM

This appendix documents the development of an estimate for whole person man-rem given that although the core has melted, the containment is intact with safeguards functioning. This is the typical licensing basis assessment used for judging the acceptability of the containment.

C.1 LICENSING BASIS INFORMATION

The information contained in Reference 1 provides the person-rem exposure given a single individual at the exclusion area boundary (EAB). This dose is given as 1.4 rem whole body. In addition, the dose within the low population zone (LPZ) would be 1.1 rem whole body. This assumes a 30-day exposure to a single individual.

Reference 1 also provides the population data surrounding the Comanche Peak Steam Electric Plant (CPSES). This data indicates that there are no persons within the EAB and approximately 500 persons within the LPZ. The LPZ is defined as an area within a four-mile radius of the plant site. Reference 1 also indicates that the population within 50 miles is approximately 2.5E+6 if Dallas county is included.

C.2 POPULATION DOSE RELATIONSHIP

The population dose d_{pop} is developed by the following equation:

$$d_{pop} = d_{ind} \bullet p \qquad \text{(eq. 1)}$$

where: d_{ind} is the dose calculated for a single individual and p is the population density.

Solving for the dose yields a value of 550 rem (500 persons x 1.1 person-rem). This dose represents the expected dose to the population given design basis leakage assumptions including the TID 14844 source term and an exposure of 30 days at a distance of 4 miles.

This information is extrapolated to 50 miles to calculate a population dose, it is important to estimate the dose out to 50 miles (although dose rates decrease significantly with distance, the population is much greater as distance increases) in order to account for the total exposed population. In addition, this is consistent with the IP3 submittal (Reference 2). The extrapolation equation is based on a ratio of the LPZ dose to the EAB dose. The equation is shown below:

$$Y = X \bullet \left(\frac{d_{LPZ}}{d_{EAB}}\right)^{C} \qquad \text{(eq. 2)}$$

This equation assumes a dose linear relation and a uniform population density (that is appropriate for the two dose values). Solving for the equation (letting Y being the LPZ dose, X be the EAB dose, d_{LPZ} and d_{EAB} are the respective distances) yields a value of 0.17 for the constant.

C.3 CALCULATION OF POPULATION DOSE

Based on the linear dose extrapolation the value at 25 miles can be used to represent an average dose for the 50-mile distance. Additionally, it is usually assumed that 95% of the population will be evacuated prior to release such that only 5% of the population would be involved. Given a total population estimate of approximately 2.5E+6 persons, this equates to an exposed population of 125,000 persons. Solving equation 2 for the 25-mile dose (Y) yields a value of 0.038 person-rem. This equates to a total population whole body dose of 4.76E+3 rem.

This value compares favorably with that presented in Reference 2 (1.41E+6 rem) and Reference 4 (987 rem). The value is lower than the IPE value due to the lower population density. It is higher than that developed for CR3 due to the impact of including the high population present in Dallas county.

It is also comparable to the released calculated for accident sequences involving impaired containments. The long-term released from CCI overpressure (release category VII) represent approximately an order of magnitude higher value then predicted for the intact containment case. Some of this difference can be attributed to the use of the TID 14844 source term in developing the intact containment case.

C.4 REFERENCES

1. Comanche Peak Steam Electric Station Unit 1, Containment Integrated Leak Rate Test, Rev. 0, TXU Electric, PPT-S1-7014, October 1993.

2. Indian Point 3 Nuclear Power Plant, "Supplemental Information Regarding Proposed Change to Section 6.14 of the Administrative Section of the Technical Specification", Entergy, IPN-01-007, January 18, 2001.

3. Indian Point Nuclear Generating Unit No.3 – Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC NO. MBO178), United States Nuclear Regulatory Commission, April 17, 2001.

4. Evaluation of Risk Significance of ILRT Extension, Revision 2, Florida Power Corporation, F-01-0001, June 2001.

5. Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, Nuclear Energy Institute, NEI 94-01, July 26, 1995.

6. Gisclon, J. M., et al, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, Electric Power Research Institute, TR-104285, August 1994.

7. An Approach for Using Probabilistic Risk Assessment in Risk-Informed decisions on Plant-Specific Changes to the Licensing Basis, U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.174, July 1998.

8. Individual Plant Examination Submittal Comanche Peak Steam Electric Station, Volumes 1 and 2, TU Electric, October 1992.

9. Tirsun, D., Comanche Peak Steam Electric Station, Level 2 PRA Release Category Quantification, Rev. 2, TXU Electric, R&R-PN-036, May 2001.

10. Performance-Based Containment Leak-Test Program, USNRC, NUREG-1493, July 1995.

11. PRA Procedures Guide- A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, American Nuclear Society and the National Institute of Electrical and Electronic Engineers, NUREG/CR-2300, January 1983.

12. Comanche Peak Steam Electric Station FSAR, Section 15.6 and Section 2.1.3, TXU Electric, Amendment 96, August 2,1999