Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2746

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Proposed Amendment to Technical Specification 6.8.3.j for a One-Time
Extension of the Integrated Leakage Rate Test Interval to 20 Years

Reference: "Proposed Amendment to Technical Specification 6.8.3.j for a Change in 10CFR50 Appendix J Integrated Leakage Rate Test Interval," J. J. Sheppard to NRC Document Control Desk, dated August 2, 2001 (NOC-AE-01001115)

Pursuant to 10CFR50.90, the STP Nuclear Operating Company requests Nuclear Regulatory Commission review and approval of a proposed amendment to the South Texas Project Technical Specifications. The South Texas Project previously received Nuclear Regulatory Commission approval of a one-time extension of the ten-year period of the performance-based leakage rate test program to 15 years for Type A tests, as requested in the referenced correspondence. This application requests an additional five years, for a total of twenty years between integrated leakage rate tests as a one-time extension.

Justification for the extension is based on the following:

- Probabilistic risk assessment demonstrates the increase in Large Early Release Frequency is less than 1.0E-07.
- Previous test results show considerable margin.
- Ongoing inspections show no structural degradation has occurred.
- Future inspections will ensure identification of any structural degradation.
- Containment venting confirms the structure as being low leakage.

The Electric Power Research Institute previously submitted for NRC review a technical report on the risk impact of extended integrated leak rate testing intervals. The NRC subsequently issued a request for additional information. STPNOC provides as Attachment 4 responses to the questions as they relate specifically to the South Texas Project.

The South Texas Project Plant Operations Review Committee has reviewed the proposed amendment and recommended it for approval.

Rec'd 10/18/06

STI: 31976661

A017

In accordance with 10 CFR 50.91(b), the South Texas Project is providing the State of Texas with a copy of this proposed amendment.

There are no commitments included in this request.

The South Texas Project requests feedback from the Nuclear Regulatory Commission no later than May 31, 2006, as to whether the proposal is viable for approval to support the Unit 2 outage currently scheduled for March 2007. Although this request is neither exigent nor an emergency, prompt review by the Nuclear Regulatory Commission is requested.

If there are any questions, please contact either Mr. P. L. Walker at (361) 972-8392 or me at (361) 972-7902.

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

Executed on ________________.

T. J. Jordan
Vice President, Engineering

PLW

Attachments:
1) Proposed Amendment to Technical Specification 6.8.3.j for a One-Time Extension of the Integrated Leakage Rate Test Interval to 20 Years
2) Reactor Containment Design and Construction
3) Application of Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes To The Licensing Basis"
4) Response to Request for Additional Information (RAI) Regarding EPRI Technical Report 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals"
5) Proposed Technical Specification Changes

cc:
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ATTACHMENT 1

PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION 6.8.3.J FOR A ONE-TIME EXTENSION OF THE INTEGRATED LEAKAGE RATE TEST INTERVAL TO 20 YEARS

SOUTH TEXAS PROJECT
UNITS 1 AND 2

PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION 6.8.3.J FOR A ONE-TIME EXTENSION OF THE INTEGRATED LEAKAGE RATE TEST INTERVAL TO 20 YEARS

1.0 INTRODUCTION

Pursuant to 10CFR50.90, the STP Nuclear Operating Company requests an amendment to South Texas Project Technical Specification 6.8.3.J extending the interval between integrated leakage rate (Type A) tests of the reactor containment buildings. The proposed amendment will allow for a one-time extension of the interval between Type A tests from 15 years to 20 years for Unit 1 and Unit 2.

This one-time extension request is supported by the containment building construction characteristics and previous test history of the South Texas Project. The containment structure demonstrated leak-tightness in previous Type A tests, and an ongoing need to relieve pressure differences demonstrates that leak-tightness continues. The effects of aging on the containment structure are considered and found to be minor for the proposed extension, especially in light of ASME Section XI IWE and IWL inspection requirements that would identify indications of aging.

In addition, an extensive risk assessment has been performed to determine the extent of changes in risk levels as a result of extending the interval to 20 years. Conservative assumptions are made in the likelihood of occurrence of factors that could reduce the leak-tightness of containment. Baseline values are determined for accident frequencies and population dose rates for the original test interval of three in ten years and used for comparison against longer test intervals. Similar comparisons are made for Large Early Release Frequency and Conditional Containment Failure Probability. Containment liner corrosion is treated as an increase in LERF and the population dose rate; however, its contribution was found to be negligible over the 20-year interval. Even including the postulated occurrence of corrosion, the impact of the extension on large early release frequency and population dose rate is not significant.

This change will not affect any accident parameters discussed in the South Texas Project Updated Final Safety Analysis Report. Extending the schedule will not cause a significant change in risk, cause NRC safety limits to be exceeded, nor adversely impact the health and safety of the public.

2.0 BACKGROUND

10CFR50 Appendix J specifies the leakage rate test requirements for primary reactor containments. The test requirements ensure that: (a) leakage through containment or systems and components penetrating containment does not exceed allowable leakage rates specified in the Technical Specifications; and (b) integrity of the containment structure is maintained during its service life. The South Texas Project has adopted Option B of Appendix J, which requires that integrated leakage rate testing be performed at periodic intervals based on performance of the containment system.

3.0 PROPOSED CHANGE

3.1 Description

Technical Specification Section 6.8.3.j requires the following:

> A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995. The current ten-year interval between performance of the integrated leakage rate (Type A) test, beginning September 24, 1991, for Unit 2, and March 10, 1995, for Unit 1, has been extended to 15 years (a one-time change).

This proposed change in the current licensing basis is a one-time extension of the test interval from 15 years to 20 years. Approval of the one-time deferral of the integrated leakage rate test would be documented with the following replacement to Technical Specification 6.8.3.j:

> The current interval between performance of the integrated leakage rate (Type A) test, beginning September 24, 1991, for Unit 2 and March 10, 1995, for Unit 1, has been extended to 20 years (a one-time change).

3.2 Purpose

Approval of this request to defer the test will allow the South Texas Project to achieve a substantial cost savings by not performing the Type A test for an additional five years. Estimated cost savings for each ILRT not performed are:

- Test performance expense: $350,000
- Schedule impact expense: $3.15 million

  (The test requires three days to perform at an estimated cost of $1.05 million per day in power generation. This assumes that the Type A test is the sole reason for adding three days to the refueling outage duration.)

- Total expense for each Type A test: $3.5 million

4.0 TECHNICAL ANALYSIS

4.1 Definitions

An ILRT failure is defined in ANSI/ANS 56.8-2002 as one in which the Type A test performance requirements are not met. The criterion for successful Type A test performance is based on the sum of the measured Type A test upper confidence limit and the as-left minimum pathway leakage rate from all Type B and Type C pathways isolated during the Type A test.

- The upper confidence limit is a calculated value constructed from test data that places a statistical upper bound on the true leakage rate (%/24 hours). NOTE – the upper confidence limit in this standard is calculated at 95% confidence level.

- The minimum pathway leakage rate is the minimum leakage rate that can be attributed to a penetration leakage path; e.g., the smaller of either the inboard or outboard barrier's individual leakage rates.

This sum is required to be less than La for a Type A test to pass the performance criterion.

4.2 Containment Integrity

4.2.1 Containment Structural Design

The South Texas Project containment structure is a post-tensioned concrete cylinder with steel liner plates, hemispherical top and flat bottom. The cylindrical portion and the hemispherical dome of the Containment are pre-stressed by a post-tensioning system consisting of horizontal and vertical tendons. The purpose of the containment post-tensioning system is to provide additional strength to resist internal pressure during postulated design-basis accidents. Design details are provided in Attachment 2.

4.2.2 Margin of Safety

The South Texas Project containment structure includes a substantial design margin for pressure. The design pressure for the building is 56.5 psig, but the calculated maximum pressure that could occur following a design basis accident is 41.2 psig. The resulting design margin is 37% [56.5/41.2 = 1.37]. The design margin of safety exceeds the 10% design margin discussed in Chapter 6 of NUREG-0800, "Standard Review Plan."

4.2.3 Containment Venting During Operation

During power operation, leaks of instrument air from air-operated valves inside containment pressurize the containment building. Changes in atmospheric conditions can also result in changes in the pressure differential between the exterior and interior of the Containment building.

The increase in the building internal pressure is reduced by periodic operation of the supplementary purge system. Needing to cycle the purge system during operation is tantamount to a periodic integrated pressure test of the containment at a low differential pressure. With a large pre-existing leak, this pressurization will not occur and operation of the containment purge system would not be necessary. As experienced by the South Texas Project, the supplementary purge system relieves a pressure differential 20 to 30 times in a given month.

The fact that the containment can be pressurized by leakage from air-operated valves and pressure variation due to temperature and barometric changes implies a lack of 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.

4.2.4 Effects of Aging

Results from previous tendon examinations show that the progression of tendon pre-stress loss is close to the predicted behavior. The IWL-3221.1(b) limit for acceptability is 95% of the predicted value. The most recent surveillance was completed in 2004. Using regression analysis (NRC Information Notice 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," October 7, 1999), the trend lines for the four tendon groups indicate that pre-stress loss will remain in the acceptable range for the life of the plant.

Technical Specification Surveillance Requirement 4.6.1.6 requires that the structural integrity of the Containment tendons be verified in accordance with the Containment Post-Tensioning System Surveillance Program. The Program is in accordance with

ASME Code Section XI, Subsection IWL, 1992 Edition with 1992 Addenda, as supplemented by 10CFR50.55a(b)(2)(viii).

4.2.5 Plant Modifications

Implementation of plant modifications should not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR. Modifications altering the containment structure are infrequent and would be reviewed to ensure containment capabilities are not diminished. The South Texas Project design change control program and the 10CFR50.59 process provide assurance that safety-significant modifications are reviewed appropriately.

4.2.6 Corrosion

NRC Information Notice (IEN) 2004-09, "Corrosion of Steel Containment and Containment Liner," reported several events at other nuclear power plants in which the metal containment liner (or metal containment vessel) had experienced corrosion. The causes were varied, from wood inclusions behind the liner to standing water and degraded seals.

The NRC's discussions noted that 10 CFR 50.55a requires implementation of ASME Section XI subsections IWE and IWL. STP implements these requirements via 0PGP03-ZE-0075 (ASME Section XI IWE Inservice Inspection Program Unit 1 and Unit 2) and 0PGP03-ZE-0076 (ASME Section XI Visual Examination for IWE Containment Inspections):

- 0PGP03-ZE-0075 provides the overall programmatic direction, and
- 0PGP03-ZE-0076 provides specific implementation instructions. The 0PGP03-ZE-0076 instructions implement the IEN guidance.

The NRC in particular points out a concern associated with liner floor seal degradation. The seal is inspected at the South Texas Project. Additionally, the containment floor concrete is slightly raised at the point of the buried seal to resist water intrusion. IEN 97-10 addressed the same subject. The response to this indicated satisfactory compliance due to the ILRT process and 0PEP04-ZE-0001 (Structures Monitoring) which has been initiated for maintenance rule monitoring. The 0PGP03-ZE-0075/76 procedures were not yet implemented at that time but were in the development process. 0PGP03-ZE-0075 references IEN 97-10.

Reviewing the Corrective Action Program database for degradation events, only three such items were noted. All of these items were found as a result of inspections and did not affect function. One situation was noted as part of the review for IEN 97-10 in which a corrosion spot was found on a ceiling joint. The corrosion was minor. IEN 2004-09 does not appear to address new issues requiring additional types of inspection.

4.3 <u>Inspections and Tests</u>

The South Texas Project has implemented an on-going program of containment inspection and testing using the revised schedule criteria made available in 1995.

4.3.1 ASME Section XI, Subsections IWE/IWL

Inservice inspection of the South Texas Project Containment buildings is conducted in accordance with the requirements of Subsections IWE and IWL of the ASME Section XI code. 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.

There is no change to the schedule for these inspections as a result of this interval extension.

4.3.1.1 Identifying Visible Degradation

Examinations pursuant to the requirements of Subsection IWE of ASME Section XI and the plant protective coatings program ensure continued integrity of the South Texas Project containment liner. Under Subsection IWE, inservice inspection examinations are performed that require visual examination of essentially 100% of the accessible surface area of the containment liner once per ISI period (three in 10 years). This examination is performed and documented by qualified individuals per Subsection IWE during refueling outages. The examination is performed either directly or remotely, depending upon the accessibility of the various areas. There have been no recordable indications of liner plate degradation at the South Texas Project.

Coating condition assessments conducted as part of the structure monitoring program provide additional assurance that containment liner flaws are identified. The structure monitoring program covers the baseline inspection and subsequent inspections conducted once each period. The condition of the coatings inside containment is characterized by visual inspection. If localized areas of degraded coatings are identified, those areas are evaluated and scheduled for repair/replacement, as necessary.

The concrete surface, including coated areas, is visually examined under ASME Subsection IWL for evidence of conditions indicative of damage or degradation. Optical aids and artificial lighting may be used as necessary to ensure the assessment is sufficiently detailed. Examinations of the containment and the containment tendons were conducted at one, three, five, ten, and fifteen years following the initial post-tensioning operations for Unit 1, and for Unit 2. The containment concrete surface, including coated areas, is visually examined for areas of large spalling, severe scaling, D-cracking in an area of 25 ft^2 or more, other surface deterioration or disintegration, or significant grease leakage. No damage or degradation of the concrete surfaces was identified during the examinations.

4.3.1.2 Identifying Non-Visible Degradation

In some cases, a complete inspection of a section of the containment liner is not possible due to its inaccessibility. Approximately 16% of the containment liner inner surface is inaccessible for inspection because it is embedded in concrete. For these cases, the condition of the liner is characterized based on an inspection of areas that are reasonably accessible to be used as representative samples.

4.3.1.3 Augmented Inspections

Augmented inspections are required if containment surface areas are likely to experience accelerated degradation and aging. As described in IWE-1240, this includes areas subject to accelerated corrosion with no or minimal corrosion allowance, areas where absence or repeated loss of protective coatings has resulted in substantial corrosion or pitting, and areas subject to excessive wear from abrasion or erosion causing a loss of protective coatings, deformation, or material loss. The South Texas Project has not experienced such containment conditions that would require augmented inspection.

4.3.2 Previous Integrated Leakage Rate Tests

Results from previous Type A tests at the South Texas Project confirm that the reactor containment structure is extremely low leakage and represents minimal risk of increased leakage. Continued performance of Type B and Type C tests minimizes the risk for direct communication with containment atmosphere across containment penetrations.

To date, three Type A tests, preoperational and operational, have been performed on Unit 1, and two Type A tests, preoperational and operational, have been performed on Unit 2. There is considerable margin between these Type A test results and the Technical Specification 3.6.1.2 limit of 0.75 La, where La is equal to 0.3% by weight of the containment air per day at the peak accident pressure. These test results demonstrate that leakage for both containment buildings is low. Two different testing methods were employed in performing these tests: the total time leakage rate method and the mass point leakage rate method. The results of both test methods are shown in Table 1 for the Type A tests conducted to date at the South Texas Project.

Note that the test results do not exceed the acceptance limit of 0.225 weight % per day (0.75 La) used by the South Texas Project.

4.3.3 Local Leak Rate Testing

The existing Appendix J Type B and Type C testing programs will not be modified by this amendment. However, the South Texas Project has been exempted from special treatment 10CFR50 requirements (Reference 4) for low risk significant and non-risk significant components, including those for Appendix J Type C testing. No credit is taken for the Type A test in determining the risk significance of the excluded components. Type B and Type C tests of components not included in the exemption from 10 CFR 50 are performed in accordance with Appendix J and the associated Technical Specifications.

A leakage rate determined by integrated leakage rate tests is the total of containment leakage and Type B and C test results. Penetrations exempted from Type C Appendix J testing are assigned a leak rate value of 'zero' although some leakage is permitted. Treating the previous results of these Type C test as 'zero' artificially increases the assigned containment leakage rate to compensate. Consequently, the Type A leakage is actually lower than that reported.

4.4 <u>Maintenance Rule</u>

10 CFR 50.65(a)(1) requires, in part, that the holders of an operating license monitor the performance or condition of structures, systems, or components, as defined in 10 CFR 50.65(b), against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, or components are capable of fulfilling their intended functions. Appropriate corrective action shall be taken when the performance or condition of a structure, system, or component does not meet established goals.

Maintenance Rule baseline inspections were performed in March 1998. The inspection results indicate that the program meets the requirements of 10CFR50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” [the Maintenance Rule]. The inspection determined that the program for monitoring the condition and effectiveness of structures is appropriate and meets the intent of the Maintenance Rule. The results of the inspection are documented in NRC Inspection Report 50-498/98-01; 50-499/98-01, dated June 16, 1998.

4.5 Defense-in-Depth

Defense-in-depth is maintained by the robust containment design (which is not affected by the proposed change), on-going performance monitoring, and inspection activities. The proposed change only affects the inspection interval and has no effect on containment design margins or isolation system capability. On-going performance monitoring and inspection is described in Section 4.3.

Calculations of radionuclide release do not consider re-establishment of containment cooling following core damage. Re-establishing containment cooling would both decrease the actual release rate and scrub fission products from the containment atmosphere.

4.6 Industry Operating Experience

The South Texas Project has in place a group tasked with review and assessment of documentation of events that have occurred at other nuclear power plants. Documentation of industry events issued by the Institute for Nuclear Operations and the Nuclear Regulatory Commission is included under this program. Type A, Type B, and Type C test and maintenance programs at the South Texas Project can be modified as necessary to implement changes resulting from reported industry experience.

4.7 Risk Assessment

4.7.1 Background

The South Texas Project Probabilistic Risk Assessment (PRA) is a full scope Level 2 analysis of Core Damage Frequency (CDF) and Containment Response including Large Early Release Frequency (LERF) and Small Early Release Frequency. The STP PRA satisfies current industry PRA standards (ASME RA-Sa-2003) and generally meets these requirements at a capability category 2 rating. The scope of STP's PRA is sufficient for assessing the risk impact of extending the Type A test interval; LERF is the key figure-of-merit relative to containment performance under severe accidents. STP does not have a Level 3 Probabilistic Safety Assessment and therefore does not have plant-specific information on the population dose associated with the various EPRI accident classes. Consequently, dose analyses are determined from estimates from NUREG-1150.

A template approach developed by the Nuclear Energy Institute (NEI), as documented in NEI correspondence dated November 13, 2001, was originally used to extend the test interval for the Type A test from 10 years to 15 years. This application presents the results of using South Texas Project Level 2 PRA results for the proposed five-year test interval extension from 15 years to 20 years.

Containment failure sequences in the South Texas Project PRA leading to radiation release are classified across six major categories. These categories are defined in Table 3 of this report. Included in Table 3 are the frequencies for these release categories as determined in the South Texas Project PRA.

4.7.2 NUREG-1493

Determination of risk associated with containment leakage and extended inspection intervals is addressed by NUREG-1493, "Performance-Based Containment Leak-Test Program." NUREG-1493 includes the results of a sensitivity study performed to explore the risk impact of several alternate leak-rate test schedules. Alternative 6 from this study examined relaxing the Integrated Leakage Rate Test frequency from 3 in 10 years to 1 in

20 years. NUREG-1493 concludes that the increased risk of population dose attributable to extending the test interval to 20 years would be imperceptible.

The estimated increase in risk is small because Type A tests identify only a few potential leakage paths that cannot otherwise be identified by Type B and C testing. Using best estimate data, NUREG-1493 concludes that the increase in population exposure risk to those in the vicinity of the five representative plants ranges from 0.02 to 0.16 %. This low impact on risk is attributable to:

- Effectiveness of Type B and C tests in identifying potential leak paths;
- Low likelihood of ILRT-identified leakage exceeding twice the allowable; and
- Insensitivity of risk to containment leak rate.

Given the insensitivity of population risk to containment leakage rates, and the small fraction of overall leakage that can be detected solely by Type A tests, the interval between integrated leakage rate tests can be increased without adverse effect on public risk.

4.7.3 Accident Classification

Accidents are classified according to the EPRI methodology for use in the ILRT evaluation:

- **Class 1 Sequence:** All core damage accident progression bins are included for which the containment remains intact with negligible leakage. Containment isolation is successful, and long-term containment heat removal capability is available via containment sprays or fan coolers.
- **Class 2 Sequence:** All core damage accident progression bins are included for which a pre-existing leak is present due to failure to isolate the containment. These sequences are dominated by failure to close containment isolation valves that are greater than 2 inches in diameter.
- **Class 3 Sequence:** All core damage accident progression bins are included for which there is pre-existing leakage in the containment structure that exceeds normal leakage. These are then grouped into two categories of leakage: small (3A) and large (3B).
- **Class 4 Sequence:** All core damage accident progression bins are included for which a failure-to-seal containment isolation failure occurs for Type B test components. This failure is detected by Type B tests, and their frequency of occurrence is very low compared with the other classes. No further separate evaluation will be performed.
- **Class 5 Sequence:** All core damage accident progression bins are included for which a failure-to-seal containment isolation failure occurs for Type C test components. This failure is detected by Type C tests, and their frequency of occurrence is very low compared with the other classes. No further separate evaluation will be performed.
- **Class 6 Sequence:** All core damage accident progression bins are included for which a failure-to-seal containment leakage occurs due to failure to isolate the containment. This group is similar to Class 2. This sequence is dominated by misalignment of containment isolation valves following a test/maintenance activity. This is typically failure to close a smaller containment isolation valve.

- **Class 7 Sequence:** All core damage accident progression bins are included for which containment failure occurs due to severe accident phenomena. The containment bypass contribution is determined by each plant's PRA.
- **Class 8 Sequence:** All core damage accident progression bins are included for containment bypass occurrences.

Table 2 provides a summary description of the accident classes as well as correlation of the equivalent STP major release categories to the applicable EPRI accident class number. Note that only accident classes 3A and 3B are affected by changes in Type A test frequency.

4.7.4 Large and Small Early Release Frequencies

Large Early Release Frequency (LERF) is defined in the PSA Applications Guide (reference 7) as the exchange of a single containment volume before effective implementation of the offsite emergency response and public protective actions. In turn, public protective actions are generally assumed to be taken 2 to 4 hours following a core damage event. Given the response measures in place after 4 hours, dose consequences of the release are assumed to terminate after 4 hours, instead of extending the full 24 hours used to establish the rate of the release. Exchange of a single containment volume within four hours corresponds to the equivalent of six full exchanges (600%) of the containment volume in 24 hours. The South Texas Project has a Type A test acceptance criterion of 0.225% per day. Conservatively setting the ILRT acceptance criterion at 0.10% per day, the leakage rate becomes 6000 La per day, or 1000 La over four hours.

The potential for a Large Early Release lies in large containment isolation failures and phenomenological effects associated with severe accidents. The predominant mechanical contributor to the Large Early Release Frequency (LERF) is failure to isolate the large supplemental purge penetrations in the unlikely event that a containment atmosphere purge is in progress during the accident. This potential contributor would not be impacted by the proposed increased interval between tests.

Large pre-existing leaks in the reactor containment building are not modeled in the South Texas Project PRA. (See also the discussion in Section 4.2.3.)

The results show that the major contributors for LERF at STP, as well as at other pressurized water reactors, are dominated by sequences resulting from a phenomenon called Induced Steam Generator Tube Rupture. This occurs when the secondary side of the steam generators dries out after a core damage event while the primary side of the reactor coolant system remains intact at high pressure. High temperature coolant circulates through the reactor coolant system, heating up the steam generator tubes to the point of failure. Core damage scenarios that involve loss of all station AC power (Station Blackout) are the primary cause of Induced Steam Generator Tube Rupture sequences. This pathway through the steam generators is not covered by the ILRT.

The Small Early Release Frequency represents the potential for a small pre-existing leak. A small containment leakage pathway present prior to core damage is the most important contributor to small early releases. The primary contribution is from station blackout scenarios, but a significant portion derives from pre-existing leaks in the containment building. The impact of this change in the test interval would be a negligible increase in the contribution for a Small Early Release Frequency.

4.7.5 Calculations

4.7.5.1 Assumptions

1) The risk associated with extending the ILRT interval has no impact on the core damage frequency of the plant.
2) The risk associated with extending the ILRT interval is affected only by the likelihood for pre-existing leaks in containment. There is no impact on containment failure modes such as containment bypass due to interfacing system LOCA and SGTR events, and failure of containment isolation function.
3) EPRI Accident Classes 4, 5, and 6 are not affected by the change in the ILRT interval and are not included in this evaluation.
4) The probability of pre-existing leak in containment is a function of the time between the ILRTs.
5) The maximum containment leak rate for EPRI Class 1 sequences is La (Type A acceptable leakage) since a new Accident Class 3 has been included to account for increase leakage due to increase in Type A inspection interval.
6) The maximum containment leakage for Accident Class 3A is 25 La.
7) The maximum containment leakage for Accident Class 3B is 1000 La.
8) The population dose results from NUREG/CR-4551 for Zion (reference 6) are applicable to STP by scaling the parameter based on the thermal output of the plants and the plant-specific population densities in the surrounding areas of the plants.
9) The rate of occurrence of Type A leakage sources is constant.

4.7.5.2 Accident Class Frequency

Class 1

These sequences apply to core damage events in which containment isolation is successful, and long term containment heat removal capability is available through containment spray or reactor containment fan coolers. The event frequency with an intact containment is based on the South Texas Project PRA. Class 1 represents containment leakage less than La.

The frequencies of Class 1, Class 3A, and Class 3B comprise release category REL IV. Therefore, the frequency for Class 1 events is related to the intact containment core damage frequency as follows:

(Frequency Class 1) = (REL IV) – (Freq Class 3A) – (Freq Class 3B)

For the South Texas Project, REL IV = 4.88E-06. Frequencies of Class 3A and Class 3B are addressed under Class 3 below.

Class 2

The frequency of occurrence for a Class 2 event is determined from the South Texas Project PRA. The frequency is not affected by changes in the test interval. No further assessment is performed.

Class 3

This group is separated into Class 3A for small containment leaks, and Class 3B for large containment leaks. Their probabilities are based on data from industry Type A test data.

Probability Class 3A is the probability of small pre-existing containment leakage in excess of design allowable but less than 100 La. See Section 4.7.5.3.

Probability Class 3B is the probability of large pre-existing containment leakage exceeding 100 La. See Section 4.7.5.3.

Class 4

These failures are detected by Type B tests. The frequency is very low and its contribution is insignificant relative to the other probabilities. The frequency is not affected by changes in the test interval. No further assessment is performed.

Class 5

These failures are detected by Type C tests. The frequency is very low and its contribution is insignificant relative to the other probabilities. The frequency is not affected by changes in the test interval. No further assessment is performed.

Class 6

This is primarily misalignment of containment isolation valves following a test or maintenance. Other failures of this type are covered in Class 2. Probability for a Class 6 event is the probability of random failure of containment to isolate due to valve misalignment (failure modes not otherwise included in Class 2). The frequency is very low and its contribution is insignificant relative to the other probabilities. The frequency is not affected by changes in the test interval. No further assessment is performed.

Class 7

Containment failure can also occur as a result of severe accident phenomena. The frequency for this group is the sum of frequencies for accident sequences that lead to early containment failure and to late containment failure. The frequency is not affected by changes in the test interval. No further assessment is performed.

Class 8

The contribution from core damage sequences in which containment bypass occurs is determined using the plant-specific PRA. The frequency for this group is the sum of frequencies for interfacing loss of coolant accidents and for steam generator tube rupture with no isolation. The magnitude of a Class 8 release is also plant-specific and is expected to be much larger than a release from a leakage event. Class 8 releases are not affected by the condition of the containment structure. The frequency is not affected by changes in the test interval. No further assessment is performed.

4.7.5.3 Baseline Frequency Determination

Baseline frequencies for the various accident classes were determined from the STP-specific PRA. These values are listed under "Frequency" in Table 4. The leakage rates associated with EPRI Accident Classes 3A and 3B are taken from WCAP-15691, Revision 5 (Reference 1).

Class 3A releases are based on Type A test history which indicates that approximately 3% of leaks identified by Type A tests may not be detected by local leak rate tests.

Class 3A releases are conservatively estimated based on a leakage rate of 25 La. Class 3B release frequencies are approximated as containment leakage greater than 100 La. In computing public dose, Class 3B radiological releases are conservatively based on leakage of 1000 La.

The probabilities for leakage for Class 3A and Class 3B are based on results from previous Type A tests. For Class 3A, 5 tests out of 182 encountered containment leakage. Consequently, a mean estimate for the probability of leakage for Class 3A is given by:

Leakage Probability Class 3A = 5/182

= 2.75E-02

For Class 3B, there is no Type A test result that could be categorized as a large release (i.e., LERF). To estimate the leakage probability for Class 3B, Jeffery's non-informative prior distribution is applied. The leakage probability for Class 3B is given by:

Leakage Probability Class 3B = (n + A)/(N + B)

n = the number of failure events (0)

N = the number of tests (182)

A = distribution parameter (0.5)

B = distribution parameter (1.0)

Leakage Probability Class 3B = 2.73E-03

The resulting frequency for small leaks is:

Frequency Class 3A = (Probability Class 3A) * (REL IV)

= (2.75E-02) * (4.88E-06)

= 1.34E-07

The resulting frequency for large leaks is:

Frequency Class 3B = (Probability Class 3B) * (REL IV)

= (2.73E-03) * (4.88E-06)

= 1.33E-08

Using these values, the adjusted frequency for a Class 1 leakage event is:

Frequency Class 1 = (REL IV) – (Freq Class 3A) – (Freq Class 3B)

= 4.88E-06 – 1.34E-07 – 1.33E-08

= 4.73E-06

4.7.5.4 Effect on Frequency

Calculating the impact of an extension assumes a constant rate of occurrence of Type A leakage points, and the potential for leakage is evenly distributed across the period of interest. Relaxing the Type A test interval increases the average time that a leak detectable only by the Type A test can be present without having been detected. The increase is proportional to the increase in duration between containment tests. The historical data is based on tests performed three times in ten years. Although this equates to a mean time between tests of 3.3 years (40 months), Type A tests are performed during refueling outages, which occur at 18-month intervals. Consequently, the test interval is assumed to be 3 years (36 months). Under Option B criteria, the test interval is 10 years, although currently the interval at the South Texas Project has a one-time extension out to 15 years.

Only Class 1, Class 3A, and Class 3B probabilities are affected by changes in the test interval. The increases in leak likelihood are directly proportional to the increase in time between Type A tests. Consequently, the factors for the various intervals are:

10 years: 120/36 = 3.33

15 years: 180/36 = 5.00

20 years: 240/36 = 6.67

The revised risk level is:

Revised Frequency = Baseline Frequency * Time Ratio

For Class 3A as an example, the respective risk levels are:

10 years: Revised Frequency = 1.34E-07 * 3.33 = 4.46E-07

15 years: Revised Frequency = 1.34E-07 * 5.00 = 6.70E-07

20 years: Revised Frequency = 1.34E-07 * 6.67 = 8.94E-07

The calculated results for each accident class for each interval are given in Table 5. The baseline results are included for comparison.

4.7.5.5 Baseline Population Dose Rate Determination

Because a Level 3 Probabilistic Safety Assessment has not been prepared for STP, STP-specific information on the population dose associated with the various accident classes is not available. The population dose for STP was estimated from the results given in NUREG/CR-4551 (reference 6) for Zion.

The population dose is weighted for each group by multiplying according to the frequency of the assigned accident class. The dose for each accident class is found by adding up the assigned frequency-weighted population dose contributions. The corresponding population dose for STP for each of the accident classes (except 3A and 3B) is derived using a conversion factor based on the Zion and STP population demographics within a 50-mile radius and thermal power level of the plants.

The baseline doses for Accident Classes 3A and 3B are found by multiplying the Class 1 dose by the La for the associated accident class:

Class 3A dose = Class 1 dose * Class 3A leakage

= 1.57E+01 * 25 La

= 3.93E+02 person-Rem

Class 3B dose = Class 1 dose * Class 3B leakage

= 1.57E+01 * 1000 La

= 1.57E+04 person-Rem

For Class 3A as an example, the respective dose rate is:

Population Dose Rate = Class 3A Dose * Class 3A Frequency

= 3.93E+02 * 1.34E-07

= 5.27E-05 person-rem/reactor-year

The baseline population dose for each accident class is given in Table 4.

4.7.5.6 Effect on Population Dose Rate

Population dose rate is the product of the dose associated with a given event class and the frequency at which it is expected to occur.

Dose Rate = Dose * Event Class Frequency

For Class 3A as an example, the respective dose rates are:

10 years: **Population Dose Rate = 3.93E+02 * 4.47E-07 = 1.76E-04**

15 years: **Population Dose Rate = 3.93E+02 * 6.71E-07 = 2.63E-04**

20 years: **Population Dose Rate = 3.93E+02 * 8.95E-07 = 3.51E-04**

The change in population dose rate for a given frequency can be expressed as:

Δ Population Dose Rate = New Dose Rate - Baseline Dose Rate

For Class 3A as an example, the respective changes in population dose rate (person-rem per reactor-year) are:

10 years: **Δ Population Dose Rate = 1.76E-04 – 5.27E-05 = 1.23E-04**

15 years: **Δ Population Dose Rate = 2.63E-04 – 5.27E-05 = 2.10E-04**

20 years: **Δ Population Dose Rate = 3.51E-04 – 5.27E-05 = 2.98E-04**

The total frequency-weighted dose rate for a given interval is found by adding the results for each accident class. The results of calculations for each accident type for the various intervals are given in Table 5. Included are assessments of the change from the baseline case in both real and relative terms. Note that changing the interval has no effect on dose rate for accident classes 2, 4, 5, 6, 7, and 8.

Table 5 shows the relative contribution of dose rate due to ILRT-affected accident classes (1, 3A, and 3B) compared to the total population dose rate. The change in the percentage contribution for each Type A test interval is included.

4.7.5.7 Effect on Large Early Release Frequency (LERF)

Extending the Type A test interval may affect the risk that a core damage scenario, normally resulting in a small radioactive release from containment, could result in a large release due to an undetected leak path. Only Class 3 sequences (pre-existing leaks) have the potential to result in early releases. The frequencies of Class 3B sequences are used as a measure of LERF, and the change in LERF is determined by the change in Class 3B frequency. Therefore, the change in LERF is expressed as:

ΔLERF = (Frequency Class 3B) – (Frequency Class 3B baseline)

The baseline frequency for Class 3B events is 1.33E-08 as shown in 4.7.5.3. As the Type A test interval is extended, the probability for a large early release changes as follows:

10 years: **ΔLERF = 4.44E-08 – 1.33E-08 = 3.11E-08**

15 years: **ΔLERF = 6.66E-08 – 1.33E-08 = 5.33E-08**

20 years: **ΔLERF = 8.88E-08 – 1.33E-08 = 7.55E-08**

Details are shown in Table 5.

4.7.5.8 Effect on Conditional Containment Failure Probability (CCFP)

CCFP is defined as the probability of containment failure given the occurrence of core damage. CCFP is calculated for a given Type A test interval by determining the ratio of probability of NO containment failure to the overall core damage frequency. CCFP is found for an interval by subtracting the calculated ratio from 1.0 (unity).

CCFP = 1 – (Frequency Class 1) / (Total CDF)

The baseline value is found using the baseline Class 1 frequency of 4.73E-6 and the total core damage frequency of 9.02E-06. The resulting baseline CCFP is 0.475.

CCFP is a measure of the likelihood of an intact containment being present at the time of an event. The likelihood decreases as the Type A test interval increases. The change from the baseline CCFP value for each extended Type A test interval is:

10 years: **ΔCCFP = [1 – 4.39E-06 / 9.02E-06] – 0.475 = 3.83E-02**

15 years: **ΔCCFP = [1 – 4.14E-06 / 9.02E-06] – 0.475 = 6.60E-02**

20 years: **ΔCCFP = [1 – 3.90E-06 / 9.02E-06] – 0.475 = 9.26E-02**

Relative to the baseline CCFP of 0.475, the change in CCFP at 20 years is 19.5%. The progression of increase is nearly linear with the increasing interval. There is substantial margin remaining before the intact containment contribution to containment failure probability is no longer significant.

Details are shown in Table 5.

4.7.5.9 Assessment of Containment Liner Corrosion

Detectable Flaw

Likelihood of a flaw developing in the containment liner as a result of corrosion is determined by dividing the number of observed flaws by the number of applicable plant years. The likelihood for occurrence of a liner flaw is determined by the following:

Containment Flaw Likelihood = (observed flaws) / (containment-years)

Over a period of 5.5 years following implementation of the containment inspection criteria beginning in 1996, visual inspection of 70 containment liners identified two instances of liner flaws. Consequently:

Containment Flaw Likelihood = 5.19E-03 (containment cylinder and dome)

This value is taken as the year 0, or base value for the South Texas Project. This result does not reflect differences that may result from the differing ages of the various plants reviewed. The South Texas Project is relatively young compared to other plants; consequently, this value is conservative for the South Texas Project.

Zero failures have been observed by the industry for the containment basemat area. To estimate the failure likelihood for this area, Jeffery's non-informative prior distribution is applied. The leakage likelihood is given by:

Basemat Leakage Likelihood = [(n + A) / (N + B)] / T

n = the number of failure events (0)

N = the number of tests (70)

A = distribution parameter (0.5)

B = distribution parameter (1.0)

T = Elapsed Time (5.5 years)

Basemat Leakage Likelihood = 1.28E-03

For the 20-year interval, the number of observed flaws is assumed to double every five years. Doubling every five years means an average annual increase in the failure occurrence of 14.9%.

This results in progressive increases in failure probability for the containment cylinder and dome sections and the containment basemat as follows:

INTERVAL	CUMULATIVE FAILURE LIKELIHOOD	
	DOME AND CYLINDER	BASEMAT
Start	5.19E-03	1.28E-03
3 years	9.05E-03	1.94E-03
5 years	1.04E-02	2.56E-03
10 years	2.08E-02	5.12E-03
15 years	4.15E-02	1.02E-02
20 years	8.30E-02	2.05E-02

The increase in flaw likelihood from 3 years to 20 years is the difference in the listed values. Consequently, for the dome and cylinder:

Δ Likelihood of detectable flaw = 8.30E-02 – 9.05E-03

= 7.40E-02

For the basemat:

Δ Likelihood of detectable flaw = 2.05E-02 – 1.94E-03

= 1.88E-02

Undetectable Flaw

The increase in likelihood for non-detected, corrosion-induced containment leakage is the product of:

- The increase in flaw likelihood during the extended inspection interval;
- The likelihood of failure of visual inspection to detect the flaw; and
- The likelihood of a containment breach during a pressure increase given a liner flaw.

The increase in flaw likelihood was determined as above.

Previous instances of corrosion were discovered through visual inspection. This evaluation assumes that for the cylinder and dome the probability of failure to detect an existing flaw is 10%. This includes 5% for failure to identify visible flaws and 5% to address non-visible flaws. Non-visible flaws are those that have not completely penetrated through the containment dome/cylinder liner, but would be detected by a Type A test. For the containment basemat, the probability of failure to detect an existing flaw is 100%.

In determining the likelihood of a containment breach, the probability is assumed to increase logarithmically to 100% at the design pressure capacity of the containment, 141 psig (155.7 psia). The upper end pressure is consistent with the South Texas Project Level 2 PRA analysis. The likelihood for the lower end is assumed to be 0.1% at 5.3 psig (20 psia). Intermediate failure likelihoods are determined through logarithmic interpolation. The breach is assumed to occur at the Type A test pressure of 41.2 psig (55.9 psia), and the probability of failure at that pressure found through interpolation. The likelihood values for the basemat are assumed to be one-tenth of those for the cylinder/dome analysis.

The change in likelihood is expressed as:

Δ Likelihood (Non-Detected) = F * V * B

F = Δ Likelihood of detectable flaw

V = Likelihood of failure of visual inspection to detect the flaw

B = Likelihood of breach at ILRT pressure

Using the change in the dome/cylinder value from 3 years to 20 years as an example:

F = 7.40E-02

V = 1.00E-01

B = 6.5E-03

Δ Likelihood (Non-Detected) = 4.81E-05

Similarly, for the basemat:

F = 1.88E-02

V = 1.00E-00

B = 6.5E-04

Δ Likelihood (Non-Detected) = 1.22E-05

The total increase in likelihood is the sum of the calculated increase for the containment cylinder/dome and the basemat.

Likelihood (Non-Detected Leakage) = (Dome/Cylinder) + (Basemat)

10 years:	**= 7.67E-06 + 2.07E-06 = 9.74E-06**
15 years:	**= 2.11E-05 + 5.37E-06 = 2.65E-05**
20 years:	**= 4.81E-05 + 1.22E-05 = 6.03E-05**

Table 6 presents the results of the analysis for the likelihood of non-detected containment leakage due to liner corrosion. Results for 10 years and 15 years are included for comparison.

The risk from extending the Type A test interval from 3 in 10 years to one in 20 years is determined using the following considerations:

- The risk associated with failure of the containment due to a pre-existing containment breach at the time of core damage (Class 3A and Class 3B).
- The risk associated with liner corrosion that could increase the likelihood a containment over-pressurization event becomes a large early release event.

The non-LERF containment over-pressurization failure frequency is the sum of frequencies for Class 1, Class 3A, and the late release contribution from Class 7 (REL II).

Non-LERF = (Freq Class 1) + (Freq Class 3A) + (Freq REL II)

10 years: Non-LERF = 4.39E-06 + 4.46E-07 + 1.38E-08 = 4.85E-06

15 years: Non-LERF = 4.14E-06 + 6.70E-07 + 1.38E-08 = 4.82E-06

20 years: Non-LERF = 3.90E-06 + 8.94E-07 + 1.38E-08 = 4.81E-06

The increase in LERF due to corrosion-induced leakage becomes:

Δ LERF = (Likelihood Total Non-Detected) * (Non-LERF)

10 years: Δ LERF = 9.74E-06 * 4.85E-06 = 4.73E-11

15 years: Δ LERF = 2.65E-05 * 4.82E-06 = 1.28E-10

20 years: Δ LERF = 6.03E-05 * 4.81E-06 = 2.90E-10

EFFECT OF EXTENDING TYPE A TEST INTERVAL FROM 3 YEARS TO 20 YEARS		
Containment Liner Condition	**Change in Large Early Release Frequency**	**Change in Person-rem per year**
Without Corrosion	8.88E-08 - 1.33E-08 = 7.55E-08	1.39E-03 – 2.09E-04 = 1.18E-03
With Corrosion	7.55E-08 + 2.90E-10 = 7.58E-08	1.18E-03 + 4.55E-06 = 1.18E-03

Detailed results are provided in Table 6.

4.7.6 Conclusion

The effect of extending the Type A test interval to 20 years has been assessed in the areas of risk, population dose, large early release frequency, and conditional containment failure probability, with the addition implications of corrosion. Although approved to follow a ten-year test interval, with the one-time use of 15 years currently in effect, this assessment takes the conservative approach of comparing the effects of the 20-year interval against those of the previous requirement of three tests in 10 years.

Using substantial conservatisms in the bases for the calculated values:

- The calculated change in LERF as a result of the interval extension to 20 years, including the effects of postulated corrosion, is 7.58E-08. This compares to the 1.0E-07 used as the limit for significance of changes in risk. Changes less than 1.0E-7 are considered small per Regulatory Guide 1.174.
- The calculated change in the population dose rate due to LERF as a result of the interval extension to 20 years, including the effects of postulated corrosion, is 1.18E-03 person-rem / reactor-year. This compares to the base case total dose rate of 1.42E-02. The relative increase from the base case total dose rate is 8.31%.
- The calculated conditional containment failure probability as a result of the interval extension to 20 years increases from a probability of 0.475 to 0.567, a change of 9.26E-02, or 19.5%. While a failed containment concurrent with an event is more likely with a 20-year interval, the effect on the likelihood of an event, and on the consequences of an event, is insignificant.

The proposed changes will not alter assumptions relative to the mitigation of an accident or transient event, and will not adversely affect normal plant operation and testing. The proposed changes are consistent with the current safety analysis assumptions and with the Technical Specifications. The effects of lengthening the time between Type A tests are summarized in Table 5.

Overall plant risk due to containment leakage is relatively small, given the small probability of containment leakage itself. The predominant mechanical contributor to the Large Early Release Frequency is failure to isolate the large supplemental purge penetrations in the unlikely event that a purge is in progress during the accident. This contributor would not be impacted by this proposed change.

Test histories, structural capability of the containments, and risk assessments have established that:

- Containment leakage rates are acceptable with considerable margin;
- The structural integrity of containment is assured; and
- Risk impact in extending the Type A test interval to 20 years is negligible.

Postponement of Type A testing could increase the probability of occurrence of a small containment leak. This increased probability has been shown to result in a very small increase in the calculated population dose for the South Texas Project. However, extension would not cause a significant change in risk, nor cause NRC safety goals to be exceeded.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed Technical Specification revision extends the current interval for Type A testing. The current test interval would be extended on a one-time basis to 20 years from the preceding Type A test. Pursuant to 10CFR50.91, this analysis provides a determination that the proposed change to the Technical Specifications for a one-time extension of the interval for Integrated Leakage Rate Testing does not involve any significant hazards consideration as defined in 10CFR50.92.

- **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

  The proposed extension to the Type A test interval will not increase the probability of an accident previously evaluated. The Type A test interval extension is not a modification and the test interval extension is not of a type that could lead to equipment failure or accident initiation.

  NUREG-1493 concluded that reducing the Type A test frequency to one per twenty years leads to a small increase in risk. Testing and inspection provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. Previous Type A tests at the South Texas Project demonstrate leakage does not exceed acceptance criteria, indicating a very leak-tight containment. Inspections required by the Maintenance Rule and ASME code are expected to identify indications of containment degradation that could affect leak tightness. The on-going need for pressure relief by periodic venting through the supplementary purge system confirms that containment integrity continues.

  Consequently, extending the Type A test interval to 20 years does not involve a significant increase in the probability of occurrence or consequences of an accident.

- **The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.**

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

- **The proposed change does not involve a significant reduction in the margin of safety.**

  The proposed extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing results in an imperceptible increase in risk to the public. NUREG-1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the increase in Type A test interval would have a minimal effect on this risk because 95% of the potential leakage paths are detected by Type B and C testing. The increase in LERF has been found to be less than 1.0E-07.

  Deferral of Type A testing for the South Texas Project does not increase the level of public risk due to loss of capability to detect and measure containment leakage or loss of containment structural capability. Other containment testing methods and inspections will assure all limiting conditions of operation will continue to be met. The potential for development of corrosion during the extended interval, but was determined to have an insignificant impact. The margin of safety inherent in existing accident analyses is maintained.

Based on the evaluation provided above, the South Texas Project concludes that the proposed change does not involve a significant hazards consideration and will not have a significant effect on safe operation of the plant. Therefore, there is reasonable assurance that operation of the South Texas Project in accordance with the proposed revision to the Technical Specifications will not endanger the public health and safety.

5.2 Applicable Requirements

- 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"
- Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program"

5.3 Analysis

The testing requirements of 10CFR50, Appendix J, provide assurance that containment leak paths, including systems and components that penetrate the containment, do not exceed the allowable leakage values specified in the Technical Specifications.

Effective October 26, 1995, 10CFR50, Appendix J, was revised to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." The South Texas Project complies with the criteria of Option B – Performance-Based Requirements. 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; however, it did alter the frequency of measuring primary containment leakage in Type A, B, and C tests.

Regulatory Guide 1.163 endorses NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995 and prepared by the Nuclear Energy Institute. Exceptions to the requirements of Regulatory Guide 1.163 are allowed by 10CFR50, Appendix J, Option B, Section V.B, "Implementation," which states:

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

Consequently, extending the interval requires a license change only and not an exemption from Option B.

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 Revision 0 includes the criterion that Option B Type A testing be performed at a frequency of 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 1.0 La) and consideration of the performance factors in NEI 94-01, Section 11.3. The test frequency is based upon a generic evaluation documented in NUREG-1493 (Reference 3).

NUREG-1493 states that reducing the Type A (ILRT) test frequency to 1 per 20 years leads to an imperceptible increase in risk. The estimated increase in risk is small because a Type A test will identify only a few potential leakage paths that cannot otherwise be identified by Type B and C testing. Given the insensitivity of population risk to containment leakage rates, and the same fraction of leakage detected solely by Type A tests, the interval between integrated leakage rate tests can be increased with negligible effect on public risk.

6.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(b) specifies the criteria for categorical exclusion from the requirements for a specific environmental assessment per 10 CFR 51.21. The South Texas Project has evaluated the proposed amendment and determined that:

- The proposed amendment does not involve a significant hazards consideration.

  As demonstrated in the No Significant Hazards Consideration Determination, the requested license amendment does not involve any significant hazards consideration.

- There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

  The proposed amendment involves no change to the facility and does not involve any change in the manner of operation of any plant systems involving the generation, collection or processing of radioactive materials or other types of effluents. Therefore, no increase in the amounts of effluents or new types of effluents would be created.

- There is no significant increase in individual or cumulative occupational radiation exposure.

  The requested license amendment involves no change to the facility and will not increase the radiation dose resulting from the operation of any plant system. Furthermore, implementation of this proposed change will not involve work activities that could contribute to occupational radiation exposure. Therefore, there will be no increase in individual or cumulative occupational radiation exposure associated with this proposed change.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required to be prepared in connection with these proposed changes.

7.0 IMPLEMENTATION

The South Texas Project requests feedback from the Nuclear Regulatory Commission no later than May 31, 2006, as to whether the proposal is viable for approval to support the Unit 2 outage currently scheduled for March 2007. Although this request is neither exigent nor an emergency, prompt review by the Nuclear Regulatory Commission is requested.

8.0 REFERENCES

1. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995
2. American National Standard ANSI/ANS - 56.8 - 1994, "Containment System Leakage Testing Requirements"
3. NUREG-1493, "Performance-Based Containment Leak-Test Program"
4. "Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100 (TAC Nos. MA6057 and MA6058)," dated August 3, 2001

5. WCAP-15691, "Joint Applications Report for Containment Integrated Leakage Rate Test Interval Extension," Revision 5 (March 2004)
6. NUREG/CR-4551, Volume 7, Revision 1, "Evaluation of Severe Accident Risks: Zion, Unit 1," March 1993
7. "PSA Applications Guide," EPRI, Palo Alto, CA (1995) (TR 105396)

TABLE 1

PREVIOUS TEST RESULTS

Unit	Date	Test Pressure (psig)	Acceptance Limit (%)	Mass Point Leakage (%)	Total Time Leakage (%)
1	03/25/87	37.4	0.225	0.0320	0.0321
	01/10/91	39.5	0.225	0.0668	0.1336
	03/10/95	44.5	0.225	0.020	0.0139
2	09/27/88	38.3	0.225	0.034	0.034
	09/23/91	44.6	0.225	0.0765	0.0681

TABLE 2

ACCIDENT CLASS INFORMATION

Class No.	Description	STP Mapping
Class 1	Containment intact (not affected by ILRT frequency)	REL IV
Class 2	Large Containment Isolation failures (not affected by ILRT frequency)	REL I
Class 3A	Small pre-existing leak in containment structure or liner, identifiable by ILRT (affected by ILRT frequency)	-
Class 3B	Large pre-existing leak in containment structure or liner, identifiable by ILRT (affected by ILRT frequency)	-
Class 4	Small Isolation Failure – failure to seal - Type B test (not affected by ILRT frequencies)	REL IIA
Class 5	Small Isolation Failure – failure to seal - Type C test (not affected by ILRT frequencies)	REL IIA
Class 6	Containment Isolation Failure - dependent failures personnel errors (not affected by ILRT frequency, low frequency)	REL IIA
Class 7	Severe accident phenomena induces failures – early and late containment failures (not affected by ILRT frequency)	REL IB, II and III
Class 8	Containment Bypass – SGTR and ISLOCA (not affected by ILRT frequency)	REL IA

TABLE 3

MAJOR RELEASE CATEGORIES

Name	Definition	Frequency (per year)
REL I	Large early failure of containment – Containment Isolation Failure	1.59E-09
REL IA	Large early failure of containment – ISGTR and ISLOCA	4.81E-07
REL IB	Large early failure of containment – Due to Severe Accident Phenomenon	2.99E-08
REL II	Small early failure of containment	1.38E-08
REL IIA	Small early failure of containment - Containment Isolation Failure	1.43E-06
REL III	Late failure of containment – Due to Severe Accident Phenomenon	2.18E-06
REL IV	No containment failure	4.88E-06
Total (Frequency per year)		9.02E-06

TABLE 4
BASELINE ACCIDENT CLASS INFORMATION

Class No.	Description	STP Mapping	Frequency	Leakage	Population Dose, person-rem	Population Dose Rate, person-rem/Rx-Year
Class 1	No Containment Failure (not affected by changes to ILRT frequencies)	REL IV	4.73E-6 (REL IV Freq) minus (F3A + F3B)	La	1.57E+01 Value from Zion	7.43E-05 Dose1 * Frequency1
Class 2	Large Containment Isolation Failures (failure-to-close) (not affected by changes to ILRT frequencies)	REL I	1.59E-09 Value from Plant PRA	Value from Plant PRA	1.61E+02 Value from Zion	2.56E-7 Dose2 * Frequency2
Class 3A	Small Pre-existing Containment Leak (identifiable by ILRT; affected by ILRT frequency)	--	1.34E-07 0.0275* (REL IV Freq)	25 La	3.93E+02 (Class 1 dose for La) * 25 La	5.27E-05 Dose3A * Frequency3A
Class 3B	Large Pre-existing Containment Leak (identifiable by ILRT; affected by ILRT frequency)	--	1.33E-08 0.00273* (REL IV Freq)	1000 La	1.57E+04 (Class 1 dose for La) *1000 La	2.09E-04 Dose3B * Frequency3B
Class 4	Small Isolation Failure (failure-to-seal) (Type B) (not affected by ILRT frequencies)	--	1.43E-06 Value from Plant PRA	NA	NA	NA
Class 5	Small Isolation Failure (failure-to-seal) (Type C) (not affected by ILRT frequencies)	--		NA	NA	NA
Class 6	Containment Isolation Failures (affected by ILRT frequency, low probability)	--		NA	NA	NA
Class 7	Severe Accident Phenomena (not affected by ILRT frequency)	REL IB, II, and III	2.22E-06 Value from Plant PRA	Value from Plant PRA	5.90E+03 Value from Zion	1.31E-02 Dose7 * Frequency7
Class 8	Containment Bypassed (not affected by ILRT frequency)	REL IA	4.81E-07 Value from Plant PRA	Value from Plant PRA	1.57E+03 Value from Zion	7.55E-04 Dose8 * Frequency8
	Total	**CDF**	9.02E-06	**Total Dose Rate**		1.42E-02

TABLE 5
EFFECTS OF INTERVAL CHANGE ON EVENT FREQUENCY AND POPULATION DOSE RATE

Class	Population Dose**	Base Case (3 in 10 years)		10-Year Interval		15-Year Interval		20-Year Interval	
		Event Frequency	Population Dose Rate***	Event Frequency	Population Dose Rate***	Event Frequency	Population Dose Rate***	Event Frequency	Population Dose Rate***
1	1.57E+01	4.73E-06	7.43E-05	4.39E-06	6.89E-05	4.14E-06	6.50E-05	3.90E-06	6.12E-05
2	1.61E+02	1.59E-09	2.56E-07	1.59E-09	2.56E-07	1.59E-09	2.56E-07	1.59E-09	2.56E-07
3A	3.93E+02	1.34E-07	5.27E-05	4.47E-07	1.76E-04	6.71E-07	2.63E-04	8.95E-07	3.51E-04
3B	1.57E+04	1.33E-08	2.09E-04	4.44E-08	6.79E-04	6.66E-08	1.05E-03	8.88E-08	1.39E-03
4	NA	1.43E-06	NA	1.43E-06	NA	1.43E-06	NA	1.43E-06	NA
5	NA		NA		NA		NA		NA
6	NA		NA		NA		NA		NA
7	5.90E+03	2.22E-06	1.31E-02	2.22E-06	1.31E-02	2.22E-06	1.31E-02	2.22E-06	1.31E-02
8	1.57E+03	4.81E-07	7.55E-04	4.81E-07	7.55E-04	4.81E-07	7.55E-04	4.81E-07	7.55E-04
Total CDF	**Total Dose Rate**	9.02E-06	1.42E-02	9.02E-06	1.48E-02	9.02E-06	1.52E-02	9.02E-06	1.57E-02
Change in Total Dose Rate from Base (rem)					6.06E-04		1.04E-03		1.47E-03
ILRT Fractional Contribution to Total Dose Rate			2.37E-02		6.36E-02		9.01E-02		1.15E-01
Change in ILRT Contribution to Total Dose Rate *					3.99E-02		6.65E-02		9.16E-02
LERF **[Class 3B only]**			1.33E-08		4.44E-08		6.66E-08		8.88E-08
Change in LERF from base					3.11E-08		5.33E-08		7.55E-08
Conditional Containment Failure Probability			0.475		0.513		0.541		0.568
Change in CCFP** from base					0.0382		0.0654		0.0927

* This value is the change in the fraction of the total dose attributable to Classes 3A and 3B (those accident classes affected by change in ILRT surveillance interval) from the base dose.
**Person-rem
***Person-rem / Reactor-year

TABLE 6

LINER CORROSION CASE COMPARISON

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Cylinder and Dome

86 %</th><th colspan="2">Containment Basemat

14 %</th></tr>
<tr><td>1</td><td>Historical Liner Flaw Likelihood
Failure Data:
Success Data:
Based on results from periodic visual inspections of containment surfaces for 70 steel-lined Containments over a 5.5-year period.</td><td colspan="2">Events: 2
2/(70*5.5) = 5.19E-03</td><td colspan="2">Events: 0
0.5/((70+1)*5.5) = 1.28E-03</td></tr>
<tr><td rowspan="6">2</td><td rowspan="6">Age-Adjusted Liner Flaw Likelihood
During a 20-year interval, the assumed failure rate doubles every five years (increasing 14.9% per year).</td><td>Year</td><td>Failure Rate</td><td>Year</td><td>Failure Rate</td></tr>
<tr><td>3</td><td>9.05E-03</td><td>3</td><td>1.94E-03</td></tr>
<tr><td>5</td><td>1.04E-02</td><td>5</td><td>2.56E-03</td></tr>
<tr><td>10</td><td>2.08E-02</td><td>10</td><td>5.12E-03</td></tr>
<tr><td>15</td><td>4.15E-02</td><td>15</td><td>1.02E-02</td></tr>
<tr><td>20</td><td>8.30E-02</td><td>20</td><td>2.05E-02</td></tr>
<tr><td>3</td><td>Change in Flaw Likelihood
Between 3 and 10 Years
Between 3 and 15 Years
Between 3 and 20 Years
Uses age-adjusted liner flaw likelihood (Step 2).
Assumes failure rate doubles every five years.</td><td colspan="2">
1.18E-02
3.25E-02
7.40E-02</td><td colspan="2">
3.18E-03
8.26E-03
1.88E-02</td></tr>
</table>

4	**Likelihood of Breach in Containment Given Liner Flaw**	**Pressure (psig)**	**Likelihood of Breach**	**Pressure (psig)**	**Likelihood of Breach**
	The upper end pressure is consistent with the South Texas Project Level 2 PRA analysis. The likelihood for the lower end is assumed to be 0.1%. Intermediate failure likelihoods are determined through logarithmic interpolation. The likelihood values for the basemat are assumed to be one-tenth of those for the cylinder/dome analysis. (Note 1)	20 58.9 (ILRT) 100 120 150	0.1% ≈0.65% 12% 34% 100	20 58.9 (ILRT) 100 120 150	.01% ≈0.065% 1.2% 3.4% 10%
5	**Likelihood of Visual Inspection Detection Failure**	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT)		**100%** Basemat cannot be visually inspected.	
6	**Change in Likelihood of Not Detecting Containment Leakage**				
	3 Years to 10 Years	7.67E-06		2.07E-06	
	3 Years to 15 Years	2.11E-05		5.37E-06	
	3 Years to 20 Years	4.81E-05		1.22E-05	
	Product of results from steps 3, 4, and 5				

Note 1: From the STP IPE, the median failure pressure for hoop failure is 162.8 psig with a β_c (standard deviation for a log-normal distribution) of 0.14. The median failure pressure for liner tear is 112.8 psig with a β_c of 0.2.

ATTACHMENT 2

REACTOR CONTAINMENT DESIGN AND CONSTRUCTION

REACTOR CONTAINMENT DESIGN AND CONSTRUCTION

Each South Texas Project containment building is a fully continuous, steel-lined, post-tensioned, reinforced concrete structure consisting of a vertical cylinder with a hemispherical dome, supported on a flat foundation mat. The cylinder and dome are post-tensioned with high-strength unbonded wire tendons. The dimensions of the containment are:

- 150-foot inside diameter
  - 239-1/4-foot inside height to the top of the dome
  - 4-foot cylinder wall thickness
  - 3-foot dome thickness.
- 18-foot mat thickness.
  - The top of the foundation mat is 41-1/4 feet below grade.

The entire inside face of the containment is a continuous welded steel liner plate provided to limit release of radioactive material into the environment. The nominal thickness of the liner is 3/8-inch. A 3/8-inch-thick plate is used on top of the foundation mat and is covered with a 24-inch concrete fill slab. Increased plate thickness up to two inches is provided around all penetrations and for the crane girder brackets.

An anchorage system is provided to prevent instability of the liner. For the dome, the anchorage system consists of meridional structural tees, circumferential angles, and plates. A system of vertical and circumferential stiffeners is provided for the cylinder, using structural angles, channels, and plates.

Leak chase channels and angles are provided at the bottom liner seams which, after construction, are inaccessible for leak tightness examination due to the two-foot interior fill slab.

The cylindrical wall is reinforced with conventional steel reinforcing bars throughout the structure. The bars are placed in a horizontal and vertical pattern in each face of the cylinder wall. Additional bars are provided around penetrations and in the buttresses to resist local stress concentrations. Radial shear reinforcement is provided throughout, and tangential shear reinforcement is provided where required. Reinforcement in the dome is provided in a meridional and circumferential pattern up to 45 degrees from the spring line, with the remaining area reinforced using a grid pattern. Reinforcement is provided on both faces of the dome wall. Radial ties are provided both to resist radial shear and prevent de-lamination of the dome under pre-stressing.

The cylindrical portion and the hemispherical dome of the Containment are pre-stressed by a post-tensioning system consisting of horizontal and vertical tendons. Three buttresses equally spaced around the Containment provide anchor points for the horizontal tendons. The cylinder and the lower half of the dome are pre-stressed by horizontal tendons anchored 360 degrees apart, bypassing the intermediate buttresses. Each successive hoop is progressively offset 120 degrees from the one beneath it. The vertical U-shaped tendons are continuous over the dome, forming a two-way post-tensioning system for the dome. These tendons are anchored in a continuous gallery beneath the base slab which provides for installation and inspection of the vertical tendons. The reinforced concrete containment structure is designed to resist loads imposed by external events such as wind, seismic activity, or tornado.

ATTACHMENT 3

APPLICATION OF REGULATORY GUIDE 1.174, "AN APPROACH FOR USING PROBABILISTIC RISK ASSESSMENT IN RISK-INFORMED DECISIONS ON PLANT-SPECIFIC CHANGES TO THE LICENSING BASIS"

APPLICATION OF REGULATORY GUIDE 1.174, "AN APPROACH FOR USING PROBABILISTIC RISK ASSESSMENT IN RISK-INFORMED DECISIONS ON PLANT-SPECIFIC CHANGES TO THE LICENSING BASIS"

1.0 INTRODUCTION

The South Texas Project has completed a risk assessment of the proposed one-time extension of the containment Type A test interval to 20 years. The risk assessment follows the applicable guidelines of Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," on the use of Probabilistic Risk Assessment (PRA) findings and risk insights. The approach combined the Individual Plant Examination (IPE) results and findings to estimate plant risk on specific accident sequences impacted by Type A testing.

In implementing risk-informed decision-making, changes are expected to meet a key set of principles. These principles are:

- Regulatory compliance;
- Defense-in-depth;
- Sufficient safety margins;
- Proposed increases in risk are small and their cumulative effects do not cause the NRC Safety Goals to be exceeded; and
- Performance-based implementation and monitoring strategies address uncertainties in analysis models and data and provide for timely feedback and corrective action.

The change in plant risk was evaluated based on the changes in the predicted person-rem/year frequency and Large Early Release Frequency (LERF).

The analysis examined plant-specific accident sequences in which the containment remains intact or the containment integrity is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact initially and in the long term (Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components; e.g., liner breach, or steam generator manway leakage (Class 3 sequences).
- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left open following a maintenance test; e.g., valve failing to close following a valve stroke test (Class 6 sequences).
- Accident sequences involving containment failure induced by severe accident phenomena (Class 7 sequences), containment bypass (Class 8 sequences), large containment isolation failures (Class 2 sequences) and small containment isolation 'failure-to-seal' events (Class 4 and 5 sequences) are not included in this evaluation. These sequences are impacted by changes in Type B and C test intervals, not changes in the Type A test interval.

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

- Quantification of the base-lined risk in terms of frequency per reactor-year for each of the eight accident classes presented.
- Development of plant-specific person-rem dose (population dose) per reactor-year for each of the eight accident classes.
- Evaluation of the risk impact of extending the Type A test interval one time to 20 years.
- Determination of the change in risk in terms of LERF in accordance with Regulatory Guide 1.174.

2.0 SUBMITTAL CONTENT

The following is provided as confirmation that the proposed licensing basis change is consistent with the key principles of risk-informed regulation in accordance with Regulatory Guide 1.174:

- A description of how the proposed change will impact the current licensing basis.

  See Attachment 1, Section 3.1.

- A description of the components and systems affected by the change, the types of proposed changes, the reason for the changes, and results and insights from an analysis of available data on equipment performance.

  See Attachment 1, Sections 3.1, 3.2, and 4.7.

- A tabulation of the current licensing basis accident parameters affected by the change and an assessment of the expected changes.

  This change does not result in a change to the current licensing basis accident parameters.

- A reevaluation of the licensing basis accident analysis and the provisions of 10 CFR Parts 20 and 100, if appropriate.

  See Attachment 1, Section 4.7.5.

- An evaluation of the impact of the change in licensing bases on the breadth or depth of defense-in-depth attributes of the plant.

  See Attachment 1, Section 4.5.

- Identification of how and where the proposed change will be documented as part of the plant's licensing basis (e.g., FSAR, TS, and license conditions).

  See Attachment 1, Section 3.1

- Section 2.2 of this attachment identifies assumptions developed to support this amendment request. No key assumptions (per RG 1.200T) were made in the PRA to perform this assessment.

- Structures, systems, and components for which requirements should be increased.

  Increased requirements are not necessary.

- A description of the information to be provided as part of the plant's licensing basis (e.g., FSAR, TS, licensing condition).

  See Attachment 1, Section 3.1.

- The top events included in the STP Level 2 Containment Event Tree are presented in Table 1 of this attachment.

- Components included in the STP PRA include all active components for the function modeled, and most passive components where failure could affect the system function. See Table 2 of this attachment.

- A list of initiators considered and their frequencies, as well as the basis for excluding any initiators from the risk assessment.

  See Attachment 1, Section 4.7.3.

A "Living PRA" program has been implemented at the South Texas Project to maintain the PRA in a state of readiness to provide a current, up-to-date assessment of the hardware and procedures that affect safety. Procedures, guidelines, tools, and processes are put in place to allow the PRA to assess the current plant safety status. The PRA is maintained by periodically updating it to reflect all relevant plant changes, new data, or improved understanding, thus monitoring over time the plant safety level and the relative importance of the various components supporting that level.

2.1 Appendix B

The South Texas Project has developed a quality assurance plan for application to the PRA. Included in the plan is provision for a qualified reviewer to perform technical reviews of PRA documents. The reviewer checks the PRA document to assure technical adequacy of the work performed and to assure proper documentation. Records of review comments and associated resolutions that are technical in nature and result in material changes to the PRA shall be maintained for the entire operating life of the plant.

2.2 Risk Assessment Methods

The probabilistic safety assessment and its updates follow the procedures described in NUREG/CR-2300, "PRA Procedures Guide," January 1983, and approved plant procedures. NUREG-1150 identifies containment and severe accident issues, such as direct containment heating and induced steam generator rupture, that have been considered in the development and quantification of the containment event tree.

The analysis assumes:

- The ILRT has no effect on the quantification of core damage frequency and large early release frequency.
- The ILRT will only find "small holes" in containment, and therefore can be evaluated via Top Event 10 for small containment failure Level 2 analysis.
- All other assumptions in STP_RV42 are unaffected by this analysis and remain valid.

No credit is taken for operator action in the PRA model for releases from a failed containment. Associated risk is dominated by other event sequences which do require operator action.

Table 1: Model Name: STP_RV42

Top Events for Event Tree: CET

Top Event Name	Description
CDSQ	TOGGLE FOR CORE DAMAGE SEQUENCES
LBY	BYPASS PRIOR TO VESSEL BREECH IS LARGE
L1	LARGE CONTAINMENT FAILURE PRIOR TO VESSEL BREECH
LS	INDUCED PORV LOCA
PSC	REACTOR COOLANT PUMP SEAL LOCA
IS	INDUCED STEAM GENERATOR TUBE RUPTURE
IP	INDUCED RCS HOT LEG OR SURGE LINE FAILURE
BY	CONTAINMENT BYPASSED PRIOR TO CORE DAMAGE
C1	CONTAINMENT FAIL PRIOR TO VESSEL BREECH
CV	CORE DAMAGE ARRESTED IN VESSEL
RP	RCS PRESSURE AT VESSEL BREACH
AP	ALPHA MODE FAILURE FOR CONTAINMENT
HME	HIGH PRESSURE MELT EJECTION (HPME)
C2	CONTAINMENT FAIL AT VESSEL BREECH
L2	LARGE CONTAINMENT FAILURE AT VESSEL BREECH
X2	FAN COOLERS FAIL AT VESSEL BREECH
DBC	DEBRIS COOLED EX-VESSEL
HE	HYDROGEN BURN WITHIN 4 HOURS OF VESSEL BREECH
CE	CONTAINMENT FAILURE DUE TO EARLY BURN
H3	LATE BURN OF COMBUSTIBLE GASES
C3	LATE CONTAINMENT FAILURE DUE TO BURN
XR	FAN COOLERS FAIL AFTER LATE H2 BURN
C4	CNTMT FAILURE DUE TO LONG TERM OVERPRESSURIZATION
L4	LARGE LATE CONTAINMENT FAILURE

Table 2: Model: STP_RV42
Systems and Components

System	Top Event Name	Description	Major Components
Class 1E 120VAC	INST	120V VITAL AC	Instrument Inverter, distribution panel, regulating transformer, supply breakers
Channel 1	AC1	INSTRUMENT CHANNEL I	
Channel 2	AC2	INSTRUMENT CHANNEL II	
Channel 3	AC3	INSTRUMENT CHANNEL III	
Channel 4	AC4	INSTRUMENT CHANNEL IV	
Class 1E 125VDC	DA	125 VDC POWER TRAIN A - E1A11	Battery, 2 battery chargers, distribution panels, supply breakers
Class 1E 125VDC	DB	125 VDC POWER TRAIN B - E1B11	
Class 1E 125VDC	DC	125 VDC POWER TRAIN C - E1C11	
Class 1E 125VDC	DD	125 VDC POWER TRAIN D - E1D11	
Auxiliary Feedwater	AFWS	AUXILIARY FEEDWATER SYSTEM	Pump, flow control, isolation automatic valves, manual valves, AFWST, room cooling (MD only)
Train A	AFA	AFW TRAIN A	MD pump
Train B	AFB	AFW TRAIN B	MD pump
Train C	AFC	AFW TRAIN C	MD pump
Train D	AFD	AFW TRAIN D	TD pump, steam supply. No room cooling
Component Cooling Water	CCWS	COMPONENT COOLING WATER SYSTEM	Pumps, heat exchangers, manual valvs in flow path, automatic valves with cooled system (
Train A	KA	CCW TRAIN A	
Train B	KB	CCW TRAIN B	
Train C	KC	CCW TRAIN C	
CCW to Charging Pumps	CHCL	CCW COOLING TO CCPS	MOVs, manual valves in flow path
Train A	CLA	CCW COOLING TO CCP 1A	
Train B	CLB	CCW COOLING TO CCP 1B	
CCW non-essential Isolation	IN	ISOLATION OF NON-ESSENTIAL CCW LOADS	MOVs
Containment Isolation	CI	CONTAINMENT ISOLATION SYSTEM	Active MOVs, AOVs, SOV, check valves.
Containment Purge Isolation	CP	CONTAINMENT SUPPLEMENTAL PURGE ISOLATION	AOVs, MOVs
Containment Event Tree	AP	ALPHA MODE FAILURE FOR CONTAINMENT	
Containment Event Tree	BY	CONTAINMENT BYPASSED PRIOR TO CORE DAMAGE	
Containment Event Tree	C1	CONTAINMENT FAIL PRIOR TO VESSEL BREECH	

System	Top Event Name	Description	Major Components
Containment Event Tree	C2	CONTAINMENT FAIL AT VESSEL BREECH	
Containment Event Tree	C3	LATE CONTAINMENT FAILURE DUE TO BURN	
Containment Event Tree	C4	CNTMT FAILURE DUE TO LONG TERM OVERPRESSURIZATION	
Containment Event Tree	CDSQ	TOGGLE FOR CORE DAMAGE SEQUENCES	
Containment Event Tree	CE	CONTAINMENT FAILURE DUE TO EARLY BURN	
Containment Event Tree	CV	CORE DAMAGE ARRESTED IN VESSEL	
Containment Event Tree	DBC	DEBRIS COOLED EX-VESSEL	
Containment Event Tree	H3	LATE BURN OF COMBUSTIBLE GASES	
Containment Event Tree	HE	HYDROGEN BURN WITHIN 4 HOURS OF VESSEL BREECH	
Containment Event Tree	HME	HIGH PRESSURE MELT EJECTION (HPME)	
Containment Event Tree	IP	INDUCED RCS HOT LEG OR SURGE LINE FAILURE	
Containment Event Tree	IS	INDUCED STEAM GENERATOR TUBE RUPTURE	
Containment Event Tree	L1	LARGE CONTAINMENT FAILURE PRIOR TO VESSEL BREECH	
Containment Event Tree	L2	LARGE CONTAINMENT FAILURE AT VESSEL BREECH	
Containment Event Tree	L4	LARGE LATE CONTAINMENT FAILURE	
Containment Event Tree	LBY	BYPASS PRIOR TO VESSEL BREECH IS LARGE	
Containment Event Tree	LS	INDUCED PORV LOCA	
Containment Event Tree	PSC	REACTOR COOLANT PUMP SEAL LOCA	
Containment Event Tree	RP	RCS PRESSURE AT VESSEL BREACH	
Containment Event Tree	X2	FAN COOLERS FAIL AT VESSEL BREECH	
Containment Event Tree	XR	FAN COOLERS FAIL AFTER LATE H2 BURN	
Chemical and Volume Control	CH	CHEMICAL VOLUME AND CONTROL SYSTEM (CVCS)	Centrifugal charging pumps, room cooling, valves,
Letdown Isolation	LI	CVCS LETDOWN AND SEAL RETURN LINES ISOLATION	MOVs and AOV
Letdown Isolation	LO	LETDOWN LINE AND CONTAINMENT ISOLATION	MOVs
Positive Displacement Pump	PD	POSITIVE DISPLACEMENT CHARGING PUMP	PDP, TSC diesel and electric power distribution
Essential Chilled Water	ECHS	ESSENTIAL CHILLED WATER SYSTEM	Chillers, pumps, room cooling, manual valves in flow path
Train A	ECA	ECH TRAIN A	
Train B	ECB	ECH TRAIN B	
Train C	ECC	ECH TRAIN C	
Essential Service Water	ECWS	ESSENTIAL COOLING WATER SYSTEM	Pumps, MOVs, strainers, traveling screens, room cooing
Train A	WA	ECW TRAIN A	
Train B	WB	ECW TRAIN B	

System	Top Event Name	Description	Major Components
Train C	WC	ECW TRAIN C	
Emergency Transformer	EX	EMERGENCY TRANSFORMER FEEDS AN E1 BUS	Transformer, breakers and bus to 1E buses
Standby Transformer	SXA	UNIT 1 STANDBY TRANSFORMER	Transformer, breakers to 1E buses
Standby Transformer	SXB	UNIT 2 STANDBY TRANSFORMER (SHUTDOWN)	Transformer, breakers to 1E buses
Unit Auxiliary Transformer	UA	UNIT AUXILIARY TRANSFORMER	Transformer, breakers to non-1E standby buses
13.8kV Bus F	BF	13.8 KV BUS 1F	Standby bus to E1A
13.8kV Bus G	BG	13.8 KV BUS 1G	Standby bus to E1B
13.8kV Bus H	BH	13.8 KV BUS 1H	Standby bus to E1C
Class 1E AC Power	EP4KV	ESSENTIAL 4160V BUSES E1A, E1B, E1C	4160Vswitchgear, 480V load centers, 480V MCCs, breakers, transformers
Train A	EA	4.16 KV BUS E1A	
Train B	EB	4.16 KV BUS E1B	
Train C	EC	4.16 KV BUS E1C	
Offsite Grid	OG	LOOP AFTER START OF EVENT	Data
Offsite Grid	OGR	RECOVERY OF OFFSITE POWER WITHIN 1 HOUR	Data
Condensate	CND	CONDENSATE SYSTEM - POST TRIP	
Feedwater	FWS	MAIN FEEDWATER SYSTEM - ATWS	
Feedwater	MFS	MAIN FEEDWATER SYSTEM - POST TRIP	
EAB HVAC	DM	SMOKE PURGE DAMPERS	Air-operated dampers
EAB HVAC	EABHV	EAB HVAC SYSTEM FAILS	Fans, dampers
Train A	FA	EAB HVAC TRAIN A FANS	
Train B	FB	EAB HVAC TRAIN B FANS	
Train C	FC	EAB HVAC TRAIN C FANS	
Instrument Air	IAS	INSTRUMENT AIR SYSTEM	Air compressors, driers, receivers.
Loss of DC Train A	DCAIN	125 VDC POWER TRAIN A - E1A11 INITIATOR	Initiating Event
Loss of DC Train B	DCBIN	125 VDC POWER TRAIN B - E1B11 INITIATOR	Initiating Event
Loss of CCW - 1 Train	LCCW1A	LOCCW - ABRUN, CCWB=F, CCWC=F	Initiating Event
Loss of CCW - 1 Train	LCCW1B	LOCCW - BCRUN, CCWA=F, CCWC=F	Initiating Event
Loss of CCW - 1 Train	LCCW1C	LOCCW - ACRUN, CCWA=F, CCWB=F	Initiating Event
Loss of CCW - 2 Trains	LCCW2A	LOCCW - ABRUN, CCWC=F	Initiating Event
Loss of CCW - 2 Trains	LCCW2B	LOCCW - BCRUN, CCWA=F	Initiating Event
Loss of CCW - 2 Trains	LCCW2C	LOCCW - ACRUN, CCWB=F	Initiating Event
Loss of CCW - 3 Trains	LOCCWA	LOSS OF CCW - BCRUN, ALL SUPPORT	Initiating Event

System	Top Event Name	Description	Major Components
Loss of CCW - 3 Trains	LOCCWB	LOSS OF CCW - ACRUN, ALL SUPPORT	Initiating Event
Loss of CCW - 3 Trains	LOCCWC	LOSS OF CCW - ABRUN, ALL SUPPORT	Initiating Event
Loss of CR HVAC - 1 Train	LCRV1A	LOCRV - ABRUN, CRVB=F, CRVC=F	Initiating Event
Loss of CR HVAC - 1 Train	LCRV1B	LOCRV - BCRUN, CRVA=F, CRVC=F	Initiating Event
Loss of CR HVAC - 1 Train	LCRV1C	LOCRV - ACRUN, CRVA=F, CRVB=F	Initiating Event
Loss of CR HVAC - 2 Trains	LCRV2A	LOCRV - ABRUN, CRVC=F	Initiating Event
Loss of CR HVAC - 2 Trains	LCRV2B	LOCRV - BCRUN, CRVA=F	Initiating Event
Loss of CR HVAC - 2 Trains	LCRV2C	LOCRV - ACRUN, CRVB=F	Initiating Event
Loss of CR HVAC - 3 Trains	LOCRVA	LOSS OF CRE HVAC - BCRUN, ALL SUPPORT	Initiating Event
Loss of CR HVAC - 3 Trains	LOCRVB	LOSS OF CRE HVAC - ACRUN, ALL SUPPORT	Initiating Event
Loss of CR HVAC - 3 Trains	LOCRVC	LOSS OF CRE HVAC - ABRUN, ALL SUPPORT	Initiating Event
Loss of EAB HVAC - 1 Train	LEAB1A	LOEAB - ABRUN, EABB=F EABC=F	Initiating Event
Loss of EAB HVAC - 1 Train	LEAB1B	LOEAB - BCRUN, EABA=F EABC=F	Initiating Event
Loss of EAB HVAC - 1 Train	LEAB1C	LOEAB - ACRUN, EABA=F EABB=F	Initiating Event
Loss of EAB HVAC - 2 Trains	LEAB2A	LOEAB - ABRUN, EABC=F	Initiating Event
Loss of EAB HVAC - 2 Trains	LEAB2B	LOEAB - BCRUN, EABA=F	Initiating Event
Loss of EAB HVAC - 2 Trains	LEAB2C	LOEAB - BCRUN, EABB=F	Initiating Event
Loss of EAB HVAC - 3 Trains	LOEABA	LOSS OF EAB HVAC - BCRUN, ALL SUPPORT	Initiating Event
Loss of EAB HVAC - 3 Trains	LOEABB	LOSS OF EAB HVAC - ACRUN, ALL SUPPORT	Initiating Event
Loss of EAB HVAC - 3 Trains	LOEABC	LOSS OF EAB HVAC - ABRUN, ALL SUPPORT	Initiating Event
Loss of ECW - 1 Train	LECW1A	LOECW - ABRUN, ECWB=F, ECWC=F	Initiating Event
Loss of ECW - 1 Train	LECW1B	LOECW - BCRUN, ECWA=F, ECWC=F	Initiating Event
Loss of ECW - 1 Train	LECW1C	LOECW - ACRUN, ECWA=F, ECWB=F	Initiating Event
Loss of ECW - 2 Trains	LECW2A	LOECW - ABRUN, ECWC=F	Initiating Event
Loss of ECW - 2 Trains	LECW2B	LOECW - BCRUN, ECWA=F	Initiating Event
Loss of ECW - 2 Trains	LECW2C	LOECW - ACRUN, ECWB=F	Initiating Event
Loss of ECW - 3 Trains	LOECWA	LOSS OF ECW - BCRUN, ALL SUPPORT	Initiating Event
Loss of ECW - 3 Trains	LOECWB	LOSS OF ECW - ACRUN, ALL SUPPORT	Initiating Event
Loss of ECW - 3 Trains	LOECWC	LOSS OF ECW - ABRUN, ALL SUPPORT	Initiating Event
ISLOCA	VSEQS	INTERFACING LOCA THROUGH SI	
MAINT	GENST	GENERIC PLANNED MAINTENANCE	
MAINT	STATE	MAINTENANCE STATES	

System	Top Event Name	Description	Major Components
MAINT	TYPE	TYPE OF QUANTIFICATION, AVG CDF OR MAINT	
MAPPNG	BI	TOP EVENT FOR PDS BINNING	
MAPPNG	BRKS	STEAM LINE BREAK FRACTION	
MAPPNG	N1	SUMP RECIRCULATION	
MAPPNG	N2	EARLY CORE DAMAGE	
MAPPNG	N3	ISOLATION OF SECONDARY SIDE	
MAPPNG	N4	RECOVERABLE SGTR SEQUENCES	
MSS	MSIV	MAIN STEAM ISOLATION	MSIVs, turbine trip, MS PORVs, MSSVs
MSS	PORV	SG PORVS	
MSS	S1	ONE OF FIVE SG SRVS OPERATES	
MSS	S2	FOUR OR MORE SG SRVS	
MSS	SL	STEAM GENERATOR ISOLATION (SGTR)	
MSS	TT	TURBINE TRIP FUNCTION	
Operator Actions	OD	OPERATOR DEPRESSURIZES RCS	
Operator Actions	OF	OPERATOR ACTION TO CONTROL MFW - POST TRIP	
Operator Actions	OL	OPER DEPRESSURIZE BY BLOWING DOWN SGS	
Operator Actions	OT	MANUAL TRIP OF REACTOR	
Operator Actions	OX	OPERATOR ALIGNS EMERGENCY TRANSFORMER	
QPDS	QDPS	QDPS SYSTEM	APCs, power supplies
Train A	QA	QDPS TRAIN A	
Train B	QB	QDPS TRAIN B	
Train C	QC	QDPS TRAIN C	
Train D	QD	QDPS TRAIN D	
RCFC	CF	REACTOR CONTAINMENT FAN COOLERS	Fans, dampers
Trains A and B Run	CFAB	RCFCS - STATE ABRUN	
Trains A and C Run	CFAC	RCFCS - STATE ACRUN	
Trains B and C Run	CFBC	RCFCS - STATE BCRUN	
Reactor Coolant System	OB	OPERATOR ACTION FOR BLEED AND FEED	Primary PORVs, MOVs
Reactor Coolant System	PO	PRIMARY PRESSURE RELIEF	Primary PORVs, MOVs, Pzr PSVs
Reactor Coolant System	PPV1	ONE OF TWO PZR PORVS FAILED - ATWS	Primary PORVs, MOVs, Pzr PSVs
Reactor Coolant System	PPV2	TWO OF TWO PZR PORVS FAILED - ATWS	Primary PORVs, MOVs, Pzr PSVs
Reactor Coolant System	PR	PRIMARY PRESSURE RELIEF (FAILS) OPEN	Primary PORVs, MOVs, Pzr PSVs

System	Top Event Name	Description	Major Components
Reactor Coolant System	PSV1	ONE OR MORE PZR PSV FAIL - ATWS	Pzr PSVs
Reactor Coolant System	PSV2	TWO OR MORE PZR PSV FAIL - ATWS	Pzr PSVs
Reactor Coolant System	SE	RCPS SEAL INJECTION AND BARRIER COOLING	Movs, CVs, manual valves
Reactor Coolant System	SP	PRESSURIZER SPRAY	SOVs
Reactor Coolant System	VI	VESSEL INTEGRITY	
Recovery Actions	OM	DIESEL GENERATOR RECOVERY FACTOR	
Recovery Actions	OR	OPERATOR STARTS TRAIN WITH NO SIGNAL	
Recovery Actions	OS	HVAC RECOVERY - START SMOKE PURGE	
Recovery Actions	RE	RECOVERY FACTORS BASED ON SPLIT FRACTIONS	
Recovery Actions	REAF	RECOVERY OF THE TURBINE DRIVEN AFW PUMP	
Recovery Actions	RPDS	RECOVERY OF PDS	
Residual Heat Removal	OC	RESIDUAL HEAT REMOVAL SYSTEM	Pumps, heat exchangers, fow control valves, manual valves
Residual Heat Removal	RX	RHR SYSTEM HEAT EXCHANGERS	Pumps, heat exchangers, fow control valves, manual valves
Reactor Protection System	ESFAS	ESF ACTUATION SYSTEM	Master and slave relays
Train A	IA	ESFAS ACTUATION TRAIN A	
Train B	IB	ESFAS ACTUATION TRAIN B	
Train C	IC	ESFAS ACTUATION TRAIN C	
Reactor Protection System	SSPS	SOLID STATE PROTECTION SYSTEM	Input signals, logic, power supplies
Train R	SPR	SSPS TRAIN R	
Train S	SPS	SSPS TRAIN S	
Reactor Protection System	AM	AMSAC	AMSAC System
Reactor Protection System	RT	REACTOR TRIP SYSTEM	Breakers
Reactor Protection System	UET	UNFAVORABLE EXPOSURE TIME - ATWS	
Standby Diesel Generator	DGX	STANDBY DIESEL GENERATOR SYSTEM	Diesel generators and support equipment
SDG 11	GA	STANDBY DIESEL GENERATOR 11	
SDG 12	GB	STANDBY DIESEL GENERATOR 12	
SDG 13	GC	STANDBY DIESEL GENERATOR 13	
SEISMC	SAC	SEISMIC FAILURE 4160 VAC POWER	
SEISMC	SAF	SEISMIC FAILURE OF AFW	
SEISMC	SCL	SEISMIC FAILURE OF ECH	
SEISMC	SCW	SEISMIC FAILURE OF CCW	

System	Top Event Name	Description	Major Components
SEISMC	SDC	SIESMIC FAILURE OF 125 VDC POWER	
SEISMC	SDG	SEISMIC FAILURE OF SDGS	
SEISMC	SEW	SEISMIC FAILURE OF ECW	
SEISMC	SIV	SEISMIC FAILURE OF INVERTERS/BATTERY CHARGERS	
SEISMC	SOG	SEISMIC FAILURE OF THE OFFSITE GRID	
SEISMC	SSS	SEISMIC FAILURE OF SSPS/ESFAS	
Spet Fuel Pool Cooling	SFC	SPENT FUEL POOL COOLING	SFP Pumps, heat exchangers
Accumulators	AI	ACCUMULATOR SAFETY INJECTION	Accumulators, MOVs, CVs
Safety Injection System	HI	HIGH HEAD SAFETY INJECTION SYSTEM	Pumps, MOVs, CVs
HHSI Train A	HA	HIGH HEAD SAFETY INJECTION TRAIN A	
HHSI Train B	HB	HIGH HEAD SAFETY INJECTION TRAIN B	
HHSI Train C	HC	HIGH HEAD SAFETY INJECTION TRAIN C	
Safety Injection System	HLEG	SI HOT LEG RECIRCULATION	MOVs, CVs
Safety Injection System	LHSI	LOW HEAD SAFETY INJECTION SYSTEM	Pumps, MOVs, CVs, RHR heat exchanger (flow path)
LHSI Train A	LA	LOW HEAD SAFETY INJECTION TRAIN A	
LHSI Train B	LB	LOW HEAD SAFETY INJECTION TRAIN B	
LHSI Train C	LC	LOW HEAD SAFETY INJECTION TRAIN C	
Safety Injection System	SICOM	SAFETY INJECTION COMMON	RWST, common valves
Train A	PA	SI COMMON TRAIN A	Train MOVs, CVs, room coolers
Train B	PB	SI COMMON TRAIN B	Train MOVs, CVs, room coolers
Train C	PZ	SI COMMON TRAIN C	Train MOVs, CVs, room coolers
Safety Injection System	SIREC	SI RECIRCULATION COMMON SYSTEM	Signal, RWST Level transmitters
Train A	RA	SI RECIRCULATION TRAIN A	Sump Isolation MOVs, pump miniflow Ovs
Train B	RB	SI RECIRCULATION TRAIN B	Sump Isolation MOVs, pump miniflow Ovs
Train C	RC	SI RECIRCULATION TRAIN C	Sump Isolation MOVs, pump miniflow Ovs
Safety Injection System	SI38	SI COLD LEG INJECTION PATH CHECK VALVES	CVs
Train A	SI38A	SI38 PATH A	
Train B	SI38B	SI38 PATH B	
Train C	SI38C	SI38 PATH C	
Containment Spray system	CSR	CONTAINMENT SPRAY - RECIRCULATION	Pumps, MOVs, manual valves, CVs, nozzles
Containment Spray system	WI	CONTAINMENT SPRAY - INJECTION	Pumps, MOVs, manual valves, CVs, nozzles
Fire Protection Pumps	FPD	Fire Protection Supply	Pumps, FWSTs

ATTACHMENT 4

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)

REGARDING EPRI TECHNICAL REPORT 1009325,

"RISK IMPACT ASSESSMENT OF EXTENDED INTEGRATED LEAK RATE TESTING INTERVALS"

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING EPRI TECHNICAL REPORT 1009325, "RISK IMPACT ASSESSMENT OF EXTENDED INTEGRATED LEAK RATE TESTING INTERVALS"

1. **Please address the issues described below associated with the determination of the probability for an extremely large leak. Provide the entire community uncertainty distribution for the probability, including the mean and 95th percentile values for this distribution.**

Section 6.0: Issues associated with the determination of probability for an extremely large leak have been identified by NRC staff based on independent calculations using a methodology that addresses these issues and uses the estimates of the four experts from the report. Using an alternate methodology, the staff obtained a mean value of 8.8E-3 for the probability of an extremely large leak (A > 100 La) in a large containment. Even though the staff estimate did not include the fatigue failure mode, the staff estimate is a factor of over 30 higher than the estimate given in the report (2.47E-4 from Table 6-1 of the EPRI report). Using the estimates of all six experts and including the fatigue failure mode may further increase the discrepancy between the estimate obtained and the estimate given in Table 6-1. The issues in the approach used in the EPRI report are described below:

a. Since one leak size range for which expert opinion elicitation was performed was the extremely large leak size A > 100 La, the results of the expert elicitation should be used for that leak size only, instead of attempting a fit of the probability of a leak versus leak size to a Weibull distribution (Section 6.3). In other words, **uncertainty distributions should be generated for the probability of the leak size range A > 100 La for each expert, and arithmetically average these uncertainty distributions.** This becomes the community distribution, or aggregated distribution.

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

b. The results for leaks greater than 100 La are based on a Weibull model (page 6-4). Because the Weibull is fitted to the estimated frequencies for small and medium leaks as well as large leaks, the estimated frequencies for large leaks are sensitive to the results for smaller leaks. In effect, the results for small and medium size leaks are extrapolated to large leaks. In view of the fact that there is no known relation between leak frequency and leak size, **please provide the justification for using the Weibull distribution. In addition, use other distribution shapes to analyze the sensitivity of the results to the different distribution shapes.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

c. Each expert was asked to provide his low, best, and high estimates (Sections 1 and 6. 2, pages 5-7 and 6-5). It is generally accepted that, in expert elicitation, the expert's "best estimate" refers to the median of the distribution. However, it is clear (page 6-5) that the expert's best estimate was treated as a mean. **Please provide the guidance given to the experts as to the definition of the terms low, best and high estimates. Please describe the methods used to insure that the responses of the various experts had a common meaning. Please explain how the experts' inputs were interpreted in calculating the results.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

d. Tchebycheff's theorem (Section 6. 2, page 6-5) is used to derive a standard deviation from the low, best, and high estimates. However, even if the best estimate were the mean value, this theorem would give only a lower bound on the standard deviation. When Tchebycheff's theorem is used to estimate tail areas given the standard deviation, it is conservative, but when estimating the standard deviation given the tail areas, it is non-conservative. For example, for a normal distribution estimating the variance from the upper and lower 5 percent tails, the variance is estimated to be 0.27 times the true variance, and the estimated standard deviation is about half of the true standard deviation. **Please justify the use of the Tchebycheff approximation.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

e. The EPRI report reduced the problem of fitting the probability of the leak rate size range to the leak rate to a linear regression problem. The estimates of the exceedance frequencies for the probabilities of leak rates of different sizes are dependent variables. The procedure of linearizing the problem requires approximations to be made in the treatment of the dependence of the variables. Equations 6-19 (Section 6.3, page 6-7) are only valid if the errors in the y_1 are linear functions of the errors in the P_k (i.e., it is assumed that a first order Taylor series expansion is an adequate approximation). These errors are in addition to the errors made by using Tchebycheff's theorem to estimate the variances of the P_k, and the errors made by assuming that the expert's best estimate is the mean of his uncertainty distribution.

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this issue is not relevant to the South Texas Project application.

f. A logit transformation of Q(a) is introduced, X=logit (Q), and X is assumed to be normally distributed (Section 6.3.4, page 6-8). It is difficult to see the justification for using a normal distribution for logit (Q). The minimization of D^2, given in Equation 6-15 (Section 6.3.3, page 6-7) is consistent with assuming that the y_1 have a joint normal distribution, with covariance matrix as given by Equation 6-18. Then the most logically consistent approach would be to assume that the estimates b1, b2 of Equation 6-22 are jointly normally distributed, not that logit (Q) is normally distributed. Then ln(lnQ(a)) is normally distributed. **Please justify the assumption that logit(Q) is normally distributed.**

Moreover, the uncertainty analysis approach is questionable because information on the uncertainty distributions for each expert are not included, except incorrectly by the use of Tchebycheff's theorem (Section 6, page 6-5, Equation 6-7). In addition, if the expert's best estimate corresponds, as is generally true, to the median and not mean, then the treatment of the best estimate as a mean is questionable.

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

g. The EPRI report (Section 6. 6, page 6-10), notes that one expert used zero several times in the assignment of the probability of ILRT failure, and that zeros are difficult to treat in the statistical evaluation of expert opinion. This expert was not included in the community distribution, and therefore it was considered prudent to not include the highest expert as well. Although it seems that the expert who assigned zero to the probability of ILRT failure exhibited an extraordinary amount of overconfidence, there is no difficulty mathematically in including his results, when one uses only the estimates for the extremely large leak size range, and does not fit the probability of leak rates of different sizes to a Weibull distribution in the way done in the EPRI report. The reason for including this expert is that then there is no basis for discarding the estimates of the highest expert in the development of the community distribution.

Alternate Methodology: To eliminate these issues, the staff determined each expert's uncertainty distribution for the probability of an extremely large leak in the following way:

The expert's low, best estimate, and high probability estimates for the extremely large leak were fitted to a distribution, where the best estimate was taken as the median value of the distribution. A split lognormal distribution, truncated at the 99.9th percentile, was used and the remaining 0.1 percent of the distribution mass was placed at the 99.9th percentile. To define the bottom half of the distribution, the best estimate was used as the median value and the low estimate, treated as a 5 percent lower bound, was used for the second parameter defining the lognormal distribution. Similarly, the distribution for the upper part of the lognormal distribution was defined using the best estimate for the median value and the high estimate, treated as a 95 percent bound, was used for the other parameter. Then, the non-conservative errors from using the Tchebycheff distribution to estimate the variance are eliminated (Item d, above), and the expert's best estimate is the median instead of the mean (Item g, above). The low, best, and high estimates of the leak rate probability were given for only four experts, and the staff derived uncertainty distributions for only these four experts. **The distributions for all six experts should be used (Item g, above).** The results were aggregated using Equation 6-29 of the EPRI report.

In principle, the issues in Section 6.3 of the EPRI report (Statistical Analysis of the Expert Elicitation Input) could be at least partially removed by using a non-linear least squares regression technique with a variety of functional forms of probability of leak rate versus leak rate size. Uncertainty distributions based on the split lognormal distribution with truncation could be used for the uncertainty in the expert's estimate of the leak rate of a specific size. However, when a variety of functional forms for the probability of leak rate vs. leak rate size are used, the uncertainties will likely be so large that there is no benefit from using this approach in comparison to the approach used by the staff (i.e., where the aggregate estimate of the probability of an extremely large leak is estimated by using only the experts estimates of the probability of an extremely large leak and no attempt is made to use a fit to a probability of leak rate vs leak rate size curve).

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

2. Documentation is a key aspect of the expert elicitation process (per References 11 and 12 on EPRI report page 10-1). Only through documentation can the analysis and results be reviewed by others and provide a credible, defensible basis for the purposes of licensing. There is insufficient information provided in this report to conclude that the expert elicitation process is acceptable as conducted. As an example, the staff attempted to reproduce the EPRI results using the stated method in the report. The staff obtained different values in many cases. **Please provide additional documentation regarding the basis for selection of the experts, the actual elicitation process, training provided to the experts, each individual expert's judgments and reasoning bases, and detailed calculations used to obtain the results documented in the report.** (See Items 4, 5, 8, 10, 13, 18, 19, 32, and 33 below for additional details where documentation is requested.)

   RESPONSE

   The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

<u>Report Summary</u>

3. <u>Page v:</u> The document is intended to be a demonstration that the "generic risk impact assessment for optimized ILRT intervals of up to 20 years" is very small for all containment designs and sites. However, the number of example calculations is very limited and biased and there is not enough evidence to support a conclusion that the assessments described are either representative or bounding for all designs and sites. **Provide additional justification and analyses to support a conclusion that the risk impact is very small for all containment types and sites.**

   RESPONSE

   The risk impact assessment performed in support of this application is specific to the South Texas Project. Therefore, this request is not relevant to the South Texas Project application.

4. <u>Pages v and vi:</u> It is stated that "there is a very small risk associated with the extension, provided that the performance bases and defense-in-depth are maintained." On page vi, it is stated that the expert panel also considered defense-in-depth approaches, such as alternative inspections that supplement the testing programs. Alternative means of leakage detection are briefly mentioned in Section 3.5, but do not appear to have been further pursued as part of the risk study or expert elicitation process. **Provide the following in this regard:**

(1) A description of the performance bases and the defense-in-depth measures that are considered essential to assuring the very small risk associated with the extension;

(2) Further details on the deliberations and conclusions of the expert panel in this area; and

(3) An explanation of how safety principle 3 of Regulatory Guide 1.174 (related to the use of performance measurement strategies) would continue to be met if these bases/measures are not expected to be enhanced as part of the proposed extension of the ILRT interval (e.g., through use of indirect containment monitoring techniques or enhanced inspections as a condition of receiving the 20-year extension).

RESPONSE

(1) Performance bases are as described in Attachment 1, Section 4.3. Defense-in-depth against unidentified sources of containment leakage is addressed in Attachment 1, Section 4.5.

(2) The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

(3) Safety principle 3 of Regulatory Guide 1.174 ensures that a proposed change will maintain sufficient safety margins. Tests and inspections as adjuncts to the Type A tests to ensure performance requirements are satisfied are addressed in Attachment 1, Section 4.3.

Section 2: Problem Statement

5. Section 2.2 (page 2-2): **Please show the mathematical expressions for ΔLERF and Δ population-dose as a frequency-weighted sum over all accident classes rather than a simple product.**

RESPONSE

Determination of ΔLERF is addressed in Attachment 1, Section 4.7.5.7.

Determination of Δ population dose rate is addressed in Attachment 1, Section 4.7.5.6.

6. Section 2.2 (page 2-2): Footnote 1 defines an ILRT failure. A practical definition of ILRT failure is one which does not meet the Type A test performance criterion as defined in ANSI/ANS 56.2002. This definition should be used for risk-informed analysis. **Please provide justification for using the less conservative definition of Footnote 1.**

RESPONSE

The definition of "ILRT failure" provided in ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements," is stated in Attachment 1, Section 4.1.

Section 3: ILRT Data Applicability

7. Section 3 (page 3-1): The report states, "In fact, no failures that would result in a large early release have been found." There is no direct connection between ILRT failures and large early release. Early release is likely to occur under accident conditions which would generate high pressures or if the containment is not properly isolated. The ILRTs are performed at relatively low pressures and the duration of a test is articulated to

ensure an essentially leak-tight containment. If ILRTs at DBA pressure indicate a leakage rate greater than 1 La to 10 La, it can be considered as a precursor to an early release under challenging accident pressures. (See Sandia Report SAND90-0119, "Insight into the Behavior of Nuclear Power Plant Containments During Severe Accidents.") **This concept should be incorporated into the elicitation report.**

RESPONSE

Results of previous Type A tests at the South Texas Project are addressed in Attachment 1, Section 4.3.2. There is no correlation made between ILRT failures and large early release. Large early release frequency is addressed in Section 4.7.4.

8. Section 3 (page 3-1): The report states that "the testing data alone does not, without expert opinion, support the development of realistic values for the probability of a large containment leakage event."

a. Please explain why alternative statistical models beyond those identified in Table 3-1 can not be used to derive more realistic probability values (e.g., mean or 50 percentile values at some lower confidence level) in lieu of an expert elicitation process.

b. Also, provide a comparison of the probability values (for Class 3a and 3b leakage) obtained from the expert elicitation process with values derived from alternative statistical models or model assumptions (e.g., in the form of an expanded version of Table 3-1).

RESPONSE

a. The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

b. The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

9. Section 3.1 (page 3-1): It is misleading to state: "The first ILRT survey was performed in early 1994-NUMARC/NEI," when NUREG/CR-4220 (June 1985): "Reliability Analysis of Containment Isolation System" provides an extensive database of isolation failures that occurred between 1965 and 1983 based on licensee event reports (LERs). NUREG-1273, "Technical Findings and Regulatory Analysis for GSI II.E.4.3, Containment Integrity Check," subsequently moderated the conservative bias of NUREG/CR-4220, and analyzed potential alternatives to ILRTs. **Please provide justification for not using the combined database (old and new) and appropriate ILRT failure criterion (See Item 5) in the elicitation process and eventually in the risk-informed analysis.**

RESPONSE

Test results from previous Type A tests at the South Texas Project are provided in Attachment 1, Section 4.3.2. Test data from other plants are not used in South Texas Project containment leak rate assessments. Therefore, this request is not relevant to the South Texas Project application.

Section 4: Expert Elicitation Process

10. Section 4.1 (page 4-3), 4.4. and Table 4: Based on the credentials of the experts (Table 4.2 of the EPRI report), there was one expert in probabilistic risk assessment, three in-service inspection (ISI) and testing experts, and two containment leakage rate testing experts (one of which co-authored the report). Since most of the experts had little

knowledge or experience in the field of containment leakage rate testing, training was provided during the meeting that may have influenced their proffered leakage rate probabilities. It is also stated (page 4-3) that "experts will be chosen for their knowledge of the mechanisms that can result in containment leakage events, and therefore provide additional assurance that their judgment is only moderately significant to the overall result." Except for one expert from the NRC, the experts all have associations with industry rather than including experts from the National Laboratories or academic institutions. **Please explain why there were not more containment leakage rate testing experts chosen for expert elicitation. Provide additional information and credentials to justify that the experts have expertise regarding the mechanisms that can result in containment leakage events (including relevant containment degradation mechanisms). Justify why the expert choices that were made would not lead to biased results, especially since the input from only four experts was used to determine the leakage rate probabilities.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

11. Section 4.3.2.1 (page 4-3): The EPRI report notes that "the results of the expert elicitation process are very significant to the results of the analysis necessitating an assignment of a Degree III" importance (i.e., "highly contentious issue; very significant to the overall result of the analysis; and/or highly complex), but that this is mitigated by the availability of significant amounts of data. As a result, the issue was assigned a Degree II importance. Assigning the issue a Degree II importance appears to have enabled the use of a "technical integrator" approach, wherein the technical integrator plays the role of "evaluator." In contrast, assigning the issue a Degree III importance would require the use of a "technical facilitator/integrator" approach, wherein multiple evaluators would be involved. One could argue the converse with regard to the availability of significant amounts of data, i.e., that there is insufficient data regarding large leakage events. **Please discuss the implications on the expert elicitation process and results if the issue were assigned a Degree III rather than Degree II degree of importance.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

Section 5: Expert Elicitation Input

12. Section 5.3 (page 5-5): It is stated that the database of found degradations was included in the presentation, and that the effects of aging on potential containment failure modes was emphasized in the expert elicitation. **Please provide a description of the technical content of the presentation, a copy of the slides/handouts from the meeting, and a characterization of the expert inputs regarding the significance of various degradation mechanisms to the probability of containment leakage derived from the expert elicitation process.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach, and was not a participant in the subject process. Therefore, this request is not relevant to the South Texas Project application.

13. Section 5.5 (pages 5-6 and 5-7) and Section 6.2 (page 6-3): It is stated that the actual frequencies elicited from the experts were all relative to available data or to frequencies of smaller leaks. However, all of the results presented in the report are in terms of absolute frequencies. **Please provide the raw elicitation inputs of the relative values provided by the experts. If "best," "low," and "high" relative values were elicited, please provide the guidance given to the experts as to the definition of these terms. Please describe how the relative values provided by the experts were used to calculate the absolute frequencies.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

14. Section 5.1 (page 5-7): **Please explain why a multiplier of 2.5 (i.e., 1000/400) was used rather than a multiplier of 5.5 (i.e., 1000 /182).**

RESPONSE

The South Texas Project position is not dependent upon test results from other nuclear power plants. Use of a multiplier to normalize Type A test data from the industry in general is not necessary. Therefore, this request is not relevant to the South Texas Project application.

Section 6: Expert Elicitation Results and Analysis

15. Section 6.0 (and other sections): It is well-known that experts tend to underestimate the uncertainties of their subjective assessments. For example, a rule of thumb is that the actual coverage of an uncertainty interval is about a factor of two smaller than the nominal coverage. Thus, for almanac-type questions where the answers are known, only about half of the nominal 90% coverage intervals supplied by a panel of experts actually contain the correct answers. There is no discussion of this phenomenon in the report. **Please explain why no adjustment was made to the expert responses to compensate for possible overconfidence.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach, and consequently is not influenced by subjective assessments of uncertainties. Therefore, this request is not relevant to the South Texas Project application.

16. Section 6.1 (page 6-1): Separate input was collected for small containments and large containments. **Please explain the decision for delineating based on containment size rather than containment type, especially since the applicability of various corrosion mechanisms or failure modes may be more a function of containment type than size. Please discuss the potential impact on results if the inputs were grouped based on containment type.**

RESPONSE

The South Texas Project application is specific to its containment size and type. Therefore, this request is not relevant to the South Texas Project application.

17. Section 6.1 (page 6-2): The EPRI report states that "significant changes to failure modes were made by the experts." It is not quite clear which failure modes were considered in the expert elicitation process. One of the major failure modes that is relevant to the risk analysis is the corrosion of the bottom liner plate of concrete containments, and

corrosion of the bottom head or plates of steel containments. These areas are not conducive to ISI but an ILRT failure could indicate if any gross degradation is occurring. The staff recognizes that no data is available to characterize the probability of such a failure. However, the elicitation process needs to consider the feasibility of such a failure mode in estimating the leakage rates. **Please provide a summary of failure modes considered in the elicitation process.**

RESPONSE

A summary of the failure modes considered in the review process for the South Texas Project is included in Attachment 1, Section 4.7.3. The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

18. Section 6.2 (page 6-2): Significant areas for deliberation were said to include: (1) the effects of aging on the containments and the resulting failure modes; (2) the fact that not all potential containment failure modes may appear in the current data; and (3) different containment types having the potential for different failure modes with potentially different failure rates. **Please provide a discussion of the key points deliberated in each of these areas, and the view(s) of the expert group with regard to each key point. Include any views expressed by dissenting experts, including the two experts whose inputs were not used in developing the community distribution.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

19. Section 6.2 (page 6-2): Expert input was solicited for each of five containment failure modes: (1) construction errors or deficiency; (2) human error associated with testing or maintenance; (3) human error, design error, or other deficiency associated with modifications; (4) corrosion; and (5) fatigue. Based on the information provided in Section 2 and Appendix C, it is not possible to discern the key issues considered by each expert for each of the five failure modes, and ultimately, whether the experts were able to provide meaningful input for each of these failure modes. **Please provide a brief description of the key issues considered by each expert for each of the five failure modes, and a characterization of their views regarding the relative importance of each failure mode to the overall failure probability.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

20. Section 6.2 (page 6-3): Corrosion is mentioned only briefly as a potential failure mode to be considered by the experts. In support of one-time, 15-year ILRT test interval extensions, licensees were requested to evaluate the potential contribution to large early release frequency (LERF) from shell/liner degradation mechanisms. Although the contribution from corrosion was typically less than 1 E-8 per year in these analyses, under alternative assumptions (e.g., relating to the increase in flaw likelihood, the likelihood of breach given a flaw, and likelihood of failure to detect a flaw developing from the liner back side) this contribution could be in the range of 5E-8 per year to 2E-7 per year, and would be even greater for a 20-year interval extension. In this regard, **please provide a quantitative evaluation of the potential impact of corrosion on**

LERF based on a 20-year interval extension and a methodology similar to that used in support of the one-time, 15-year test interval extensions.

RESPONSE

Attachment 1, Section 4.7.5.9, provides the results of a quantitative evaluation of the potential impact of corrosion on LERF.

Section 7: Technical Approach

21. Section 7.1 (page 7-4): **Please further describe containment endstates/release modes that contribute to "Intact CDF."** All CDF that does not lead to LERF/bypass (e.g., intermediate and late containment failures) should be considered for inclusion in this accident class. **Additional guidance is also needed on whether/how events with containment sprays (for PWRs and BWRs) and suppression pools (for BWRs) are to be handled.**

RESPONSE

"Intact CDF" is the frequency of an event with core damage consequence that does not have an associated leak route from containment. Containment endstates/release modes that contribute to "Intact CDF" are described in Attachment 1, Section 4.7.3. The South Texas Project application does not include special consideration for containment spray.

22. Section 7.1 (page 7-4): The last sentence of the second bullet seems to indicate that the existing empirical data for small leaks has been disregarded and replaced by the probability developed by the expert elicitation process. **Please explain why it is appropriate to supplant empirical data with the results of expert elicitation.**

RESPONSE

The South Texas Project application does not use the expert elicitation approach. Therefore, this request is not relevant to the South Texas Project application.

23. Section 7.2 (page 7-4): The plant IPE or PRA are identified as possible sources for offsite dose estimates. **Please expand this discussion to recognize that population dose estimates may have also been developed subsequent to the IPE, e.g., as part of the severe accident mitigation alternative analysis performed in support of license renewal.** Plant-specific values can also be approximated based on offsite consequence analyses for similar plants, and appropriate scaling to account for site-to-site differences in meteorology and demography.

RESPONSE

A discussion of offsite dose estimates is provided in Attachment 1, Sections 4.7.5.5 and 4.7.5.6..

24. Section 7.2 (page 7-4): A leak rate for Class 3b of 100 La is said to be conservative. While 100 La may be conservative as a threshold for when a leak is considered to be large, it is non-conservative as the basis for quantifying the magnitude of a large leak. As stated on page 3-2, a leakage of 600 %/day (or 600 to 6000 La) is more representative of a large early release. Accordingly, the doses for Class 3b should be assigned a value closer to 1000 La. **Please justify the use of 100 La as conservative for Class 3b.**

RESPONSE

The South Texas Project has selected a leak rate of 1000 La for Class 3B events when quantifying the magnitude of a large leak. See Attachment 1, Section 4.7.4.

25. Section 7.5 (page 7-5): The change in probability of leakage detectable only by ILRT was determined by multiplying the baseline probability by the ratio of the new to the old test interval. **Please justify this approach, since it presumes a constant containment degradation rate, which may not be reasonable over a 20-year test interval.**

RESPONSE

The ratio of the new to the old test interval only considers the increase in risk due to the longer time that a flaw may go undetected. Attachment 1 Section 4.7.5.9 discusses the presumed containment degradation rate and assumes its progression is non-linear.

Section 8: Application of Technical Approach

26. Section 8.1 (page 8-1): The hypothetical PWR CDF characteristics (i.e., breakdown of CDF by accident class) presented in Table 8-1 is not very representative of PWRs. Specifically, in the example, 53% of the CDF is associated with a large release (Class 7 and 8) and only 47% of the CDF is associated with an intact containment. This will tend to understate the relative impact of the ILRT interval extension. (A more typical CDF breakdown would involve less than 10% early/bypass and 70 to 80% intact containment.) **Please provide a justification for the basis of the example calculation or provide a calculation more representative of CDF characteristics to avoid biasing the results.**

RESPONSE

The calculation results for the South Texas Project are based on CDF characteristics specifically for the South Texas Project.

27. Sections 8.2 (pages 8-11): The doses presented in Table 8-1 for a hypothetical PWR are extremely large and not representative of values typically derived from PRA methods (as claimed on page 7-2). In general, dose values are expected to be on the order of 1 E+3 person-rem per event for Class 1, and on the order of 1 E+6 to 1 E+7 person-rem per event for Class 7 and 8. The dose values in Table 8-1 result in an annual population dose for the hypothetical plant of 68,000 person-rem per year, which is about three to four orders of magnitude greater than obtained from recent risk analyses submitted in support of license renewal, and is also 4 orders of magnitude greater than the value for the hypothetical BWR in Section 8.2. **The example calculation should be based upon more representative dose values to avoid biasing the results.** Finally, the sum of the population dose estimates for each accident class (presented as the "total" dose in Tables 8-1, 8-3, and 8-10) is not meaningful. The total dose is only meaningful if it is a probabilistically-weighted sum. **Please delete reference to the "total doses".**

RESPONSE

Dose values used in the calculations for the results in Attachment 1 are expected to be typical of the South Texas Project. Therefore, this issue is not relevant to the South Texas Project application.

The population dose estimates generated for Attachment 1, Section 4.7.5, are weighted by probability of occurrence for the selected Type A test intervals. Deletion of the reference should not be necessary.

28. Section 8 5 (page 8-6): The equation for Δ population dose does not yield the increase in population dose. Rather it yields the annual population dose (rate) corresponding to the extended test interval. **Please express the Δ population dose as:**

Δ population dose rate$_{3a}$ = population dose$_{3a}$ x [frequency$_{3a, 1/10}$ - frequency$_{3a, 3/10}$].

RESPONSE

The change in population dose for the South Texas Project is determined for each accident class as described in Attachment 1, Section 4.7.5.6. The results are given in Attachment 1, Table 5.

29. Section 8.6 (page 8-7): The information displayed in Table 8-6 (percentage values for each accident class) is not very useful for showing the impact of the ILRT extension on total population dose. **Rather than reporting percentage values, please consider showing the population dose (per year) for each accident class, and the sum of these values over all accident classes in the table.**

RESPONSE

Table 8-6 of the EPRI submittal gives the relative contributions of population dose over the accident classes for the various Type A test intervals. The South Texas Project application includes a table showing the projected population dose (per year) for each accident class, and the sum of the change in doses over all the accident classes in the table. See Attachment 1, Section 4.7.5.6 and Table 5.

Section 9: Results Summary and Conclusions

30. Section 9 (page 9-1): The role of the EPRI report is unclear, i.e., whether it is intended to establish a methodology that can be used by licensees to support plant-specific license amendment requests to extend the ILRT test interval to 20 years, or whether it is intended to be a generic demonstration that the risk impact of extending the ILRT test interval to 20 years is very small for all containment designs and sites. The document appears to be oriented toward the former objective, and does not meet the latter intent for numerous reasons, including a very limited and biased number of example calculations, and the lack of any rationale or logic to support a conclusion that the assessments described are either representative or bounding for all designs and sites. **Please provide a detailed discussion of how the EPRI document is to be used by individual licensees and the NRC to support a permanent extension of the ILRT test interval to 20 years, including a description of any plant-specific analyses that would be required as part of plant-specific license amendment requests.**

RESPONSE

The EPRI report is used only for background information in developing the South Texas Project application.

B: Expert Elicitation Input Data

31. Table B-2 (page B-3): **If the experts supplied rationale for the probability estimates, please supply the documentation giving this rationale. If there is no documentation of the rationale, please justify the decision not to require such rationale to be documented.**

RESPONSE

Table B-3 of the EPRI report provides results of expert elicitation for large containment with medium leakage pathway. The South Texas Project application does not use the expert elicitation approach. Therefore, this question is not relevant to the South Texas Project application.

C: Expert Elicitation Results

32. Table C-5 (page C-7): In Table C-5, the aggregate upper bounds for La $\geq$ 5 are all zero while the corresponding upper bounds for Experts A, D and E are all greater than zero. **Please explain how the entries in the tables in Section C were calculated.**

RESPONSE

Table C-5 of the EPRI report provides results of statistical analysis of expert elicitation values for fatigue failure in large containments. The South Texas Project application does not use the expert elicitation approach. Therefore, this question is not relevant to the South Texas Project application.

33. Figures C-1, C-2, and C-3 (pages C-15 and C-16): Page 7-3 indicates that the data used to develop these curves was "Detectable by ILRT Only (Failures)," but the data set used to produce Table 6-1 is not identified. **Please identify whether the set of data used to produce these figures was based on "Total Degraded ILRTs" or "Detectable by ILRT Only (Failures)."**

RESPONSE

Table 6-1 of the EPRI report presents expert elicitation results for leak size versus probability. The South Texas Project application does not use the expert elicitation approach. Therefore, this question is not relevant to the South Texas Project application.

ATTACHMENT 5

PROPOSED TECHNICAL SPECIFICATION CHANGES

PROCEDURES AND PROGRAMS (Continued)

j) Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program", dated September 1995. The current interval between performance of the integrated leakage rate (Type A) test, beginning September 24, 1991, for Unit 2 and March 10, 1995, for Unit 1, has been extended to 20 years (a one-time change).

Peak calculated primary containment internal pressure for the design basis loss of coolant accident (LOCA), P_a is 41.2 psig.

The maximum allowable primary containment leakage rate, La, is 0.3% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq$ 1.0 La. During the first unit start-up following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 La for the combined Type B and Type C tests, and $\leq$ 0.75 La as-left and $\leq$ 1.0 La as-found for Type A tests.

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq$ 0.05 La when tested at $\geq P_a$.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test intervals specified in the Containment Leakage Rate Testing Program.

The provisions of Surveillance Requirement 4.0.3 apply to the Containment Leakage Rate Testing Program.

k) Configuration Risk Management Program (CRMP)

A program to assess changes in core damage frequency and cumulative core damage probability resulting from applicable plant configurations. The program should include the following:

1) training of personnel,

2) procedures for identifying plant configurations, the generation of risk profiles and the evaluation of risk against established thresholds; and

3) provisions for evaluating changes in risk resulting from unplanned maintenance activities.