Nuclear Operating Company

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

August 2, 2001
NOC-AE-01001115
File No.: G21.02
10CFR50, Appendix J
10CFR50.90

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

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
Change in 10CFR50, Appendix J, Integrated Leakage Rate Test Interval

Reference: "Proposed Amendment to Technical Specification 6.8.3.j for a One-time Change in 10CFR50 Appendix J Integrated Leakage Rate Test Interval," J. J. Sheppard to NRC Document Control Desk, dated August 31, 2000 (NOC-AE-000692)

Pursuant to 10CFR50.90, the South Texas Project requests Nuclear Regulatory Commission review and approval of a proposed amendment to the Technical Specifications. This amendment proposes a one-time extension of the ten-year period of the performance-based leakage rate testing program for Type A tests as prescribed by NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50, Appendix J," and applied by 10CFR50, Appendix J, Option B. The ten-year interval between integrated leakage rate tests is to be extended to 15 years from the previous integrated leakage rate tests, which were completed on September 24, 1991 (Unit 2) and March 10, 1995 (Unit 1).

A similar request was submitted for Indian Point 3 on September 6, 2000, and January 18, 2001, and approved by the Nuclear Regulatory Commission on April 17, 2001.

This application represents a risk-informed, cost beneficial licensing change. The integrated leakage rate test imposes significant expense on the station while the safety benefit of performing it at a ten-year interval, rather than a 15-year interval, is minimal. The results from previous integrated leakage rate tests support deferral of the test. The proposed change meets the criteria of Regulatory Guide 1.174 for risk-informed changes described in Attachment 3.

The reactor containment buildings will continue to be inspected under the requirements of ASME Section XI Subsections IWE and IWL. The existing Type B and C testing programs are not being modified by this request.

The South Texas Project has reviewed the attached proposed amendment pursuant to 10CFR50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10CFR51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment.

A017

The South Texas Project Plant Operations Review Committee has reviewed the proposed amendment and recommended it for approval. The South Texas Project Nuclear Safety Review Board has reviewed and approved the proposed change.

The required affidavit, along with a Safety Evaluation and No Significant Hazards Consideration Determination associated with the proposed change, and the proposed replacement pages of the Technical Specifications are included as attachments to this letter.

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

The South Texas Project previously submitted a request for a change in the integrated leakage rate test schedule (referenced). That request was subsequently withdrawn in response to NRC reviewer comments. This re-submittal incorporates additional information to support that previously provided.

The South Texas Project requests that this amendment be approved no later than January 31, 2002, in order to facilitate scheduling for the Unit 2 outage currently scheduled for October 2002. Date of effectiveness would occur within 30 days following approval. 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. M. S. Lashley at (361) 972-7523 or me at (361) 972-8757.

J. J. Sheppard
Vice President,
Engineering & Technical Services

PLW

Attachments:

1) Proposed Amendment to Technical Specification 6.8.3.j for a One-time Change in 10CFR50 Appendix J Integrated Leakage Rate Test Interval
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 Current Licensing Basis"
4) Proposed Technical Specification Changes
5) Revised Technical Specifications

cc:

Ellis W. Merschoff
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

John A. Nakoski
Addressee Only
U. S. Nuclear Regulatory Commission
Project Manager, Mail Stop OWFN/7-D-1
Washington, DC 20555-0001

Mohan C. Thadani
Addressee Only
U. S. Nuclear Regulatory Commission
Project Manager, Mail Stop OWFN/7-D-1
Washington, DC 20555

Cornelius F. O'Keefe
c/o U. S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, TX 77404-0910

A. H. Gutterman, Esquire
Morgan, Lewis & Bockius
1800 M. Street, N.W.
Washington, DC 20036-5869

M. T. Hardt/W. C. Gunst
City Public Service
P. O. Box 1771
San Antonio, TX 78296

A. Ramirez/C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

Jon C. Wood
Matthews & Branscomb
112 East Pecan, Suite 1100
San Antonio, Texas 78205-3692

Institute of Nuclear Power
Operations - Records Center
700 Galleria Parkway
Atlanta, GA 30339-5957

Richard A. Ratliff
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

R. L. Balcom/D. G. Tees
Houston Lighting & Power Co.
P. O. Box 1700
Houston, TX 77251

C. A. Johnson/R. P. Powers
AEP - Central Power and Light Company
P. O. Box 289, Mail Code: N5012
Wadsworth, TX 77483

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

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
STP Nuclear Operating Company	)	Docket Nos. STN 50-498 STN 50-499
	)	
	)	
South Texas Project Units 1 & 2	)	

AFFIDAVIT

I, J. J. Sheppard, being duly sworn, hereby depose and say that I am Vice President, Engineering & Technical Services, of STP Nuclear Operating Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached Technical Specification amendment request; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

J. J. Sheppard
Vice President,
Engineering & Technical Services

STATE OF TEXAS	)
	)
COUNTY OF MATAGORDA	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 2nd day of August, 2001.

NOTARY PUBLIC STATE OF TEXAS
LOIS J. MILLS
Notary Public, State of Texas
My Commission Expires
JULY 27, 2003

Lois J. Mills
Notary Public in and for the State of Texas

ATTACHMENT 1

PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION 6.8.3.J
FOR A CHANGE IN 10CFR50 APPENDIX J
INTEGRATED LEAKAGE RATE TEST INTERVAL

SOUTH TEXAS PROJECT
UNITS 1 & 2
PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION 6.8.3.J
FOR A CHANGE IN 10CFR50 APPENDIX J
INTEGRATED LEAKAGE RATE TEST INTERVAL

1.0 INTRODUCTION

1.1 Proposed Change

Pursuant to 10CFR50.90, the South Texas Project requests an amendment to the Technical Specification requirement to perform integrated leakage rate (Type A) tests of the reactor containment buildings. The proposed amendment will allow for a one-time extension of the current interval between the Type A tests from ten years to 15 years.

This one-time extension request is supported by the containment building construction characteristics and previous test history. 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, nor cause NRC safety goals to be exceeded. The impact on the health and safety of the public is minimal.

1.2 Proposed Technical Specification Changes

See Attachment 4.

1.3 Revised Technical Specifications

See Attachment 5.

1.4 Updated Final Safety Analysis Report

Implementation of this proposed change will not require revision of the South Texas Project Updated Final Analysis Report.

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.

This request does not modify the existing Appendix J Type B and Type C testing programs. However, the South Texas Project has requested exemption of low safety/risk significant and non-risk significant components from special treatment 10CFR50 requirements. "Revised Request for Exemption to Exclude Certain Components From The Scope of Special Treatment Requirements Required by Regulations," was submitted August 31, 2000 (NOC-AE-00000918), requesting exemption from requirements for local leakage rate testing (Type C) of low safety significant and non-risk significant isolation valves.

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.

Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," endorses NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995 (Reference 1) and prepared by the Nuclear Energy Institute. NEI 94-01 provides methods acceptable to the NRC staff for complying with the provisions of Option B as described in Regulatory Guide 1.163. NEI 94-01 includes the criterion that Option B Type A testing be performed at a frequency of at least once per ten years.

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

> 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).

3.2 Purpose

This application represents a cost beneficial licensing action. The integrated leakage rate test imposes significant expense on the station while the safety benefit of performing it within ten years, rather than 15 years, is minimal.

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 Unit resulting from this deferral are:

- Test performance expense: $350,000
- Schedule impact expense: $1,680,000

  [Three days @ $560,000 per day]

In addition, if the Unit 2 Type A test is conducted during its upcoming refueling outage in accordance with the ten-year interval, the test will occur during the outage when the Unit 2 steam generators are to be replaced. This would cause a significant complication in logistics and undue hardship for the outage.

4.0 REGULATORY ANALYSIS

4.1 Applicability

The testing requirements of 10CFR50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the Technical Specifications. Limiting containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant design basis accident.

10CFR50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." The South Texas Project previously selected Option B. Regulatory Guide 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8 - 1994 (Reference 3), subject to several regulatory positions in the guide.

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.

Therefore, the change proposed by this application does not require an exemption from Option B.

4.2 Test Frequency

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3. Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B, and C tests.

Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained. The changes to Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493 (Reference 4). NUREG-1493 made the following observations with regard to decreasing the test frequency:

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

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Because leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

4.3 Continuation of Type B and C Tests

The existing Appendix J Type B and Type C testing programs will not be modified by this change. However, the South Texas Project has requested exemption of low safety/risk significant and non-risk significant components from special treatment 10CFR50 requirements (Reference 5), including Appendix J Type C testing. One criterion for excluding a penetration from Type C testing is that the associated piping and components are designed for greater than the containment design pressure. No credit is taken for the ILRT in determining the risk significance of the excluded components.

Given the minimal safety/risk significance of the components in question, normal commercial and industrial practices are sufficient to ensure that the components will satisfy their functional requirements and have sufficient safety margin. Furthermore, performance of the systems/trains containing safety-related low safety/risk significant and non-risk significant components will be monitored to ensure that these systems/trains maintain satisfactory levels of performance. Type B and Type C tests that are not exempted under this request will continue to be performed in accordance with Appendix J and the associated Technical Specifications.

5.0 TECHNICAL ANALYSIS

5.1 Inspections

5.1.1 IWE/IWL Inservice Inspection

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.

5.1.2 Maintenance Rule Inspection

Maintenance Rule baseline inspections were performed in March 1998. The inspection results indicated that an appropriate program had been developed and implemented to meet the requirements of 10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [the Maintenance Rule]. The inspection determined that the program for monitoring the condition and effectiveness of structures was appropriate and met the intent of the Maintenance Rule. The results were documented in NRC Inspection Report 50-498/98-01; 50-499/98-01, dated June 16, 1998.

5.2 Previous Integrated Leakage Rate Test Results

Previous Type A test results confirmed that the reactor containment structure is extremely low leakage and represents minimal risk of increased leakage. The risk is minimized by continued Type B and Type C testing for direct communication with containment atmosphere. Also, the In-Service Inspection Program (IWE/IWL) and Maintenance Rule inspections provide confidence in containment integrity.

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 L_a, where L_a is equal to 0.3% by weight of the containment air per day at the peak accident pressure. These test results demonstrate that both units have a low leakage containment. 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 below for each of the Type A tests conducted to date at the South Texas Project.

Unit (date)	**Mass Point Leakage (%)**	**Total Time Leakage (%)**	**Acceptance Limit (%)**	**Test Pressure (psig)**
1 (03/25/1987)	0.0320	0.0321	0.225	37.4
1 (01/10/1991)	0.0668	0.1336	0.225	39.5
1 (03/10/1995)	0.020	0.0139	0.225	44.5
2 (09/27/1988)	0.034	0.034	0.225	38.3
2 (09/23/1991)	0.0765	0.0681	0.225	44.6

These results are below the acceptance limit of 0.225 weight % per day (0.75 L_a).

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

5.3 Operational Containment Venting

During power operation, instrument air leaks from air-operated valves inside containment and pressurizes the containment building. Containment pressure is monitored and conditions approaching the limits allowed by the Technical Specifications are annunciated. The increase in the building internal pressure is reduced by periodic operation of the supplementary purge system. This cycling of the containment pressure during operation amounts to a periodic integrated pressure test of the containment at a low differential pressure. With a large pre-existing leak, operation of the containment purge system would not be necessary, and would be noticed by plant operators.

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

5.4 Risk Assessment

5.4.1 Background

Plant-specific probabilistic risk assessments were not available and therefore were not considered when the regulation requiring compliance with Appendix J was adopted. Overall plant risk due to containment leakage is relatively small, given the small probability of containment leakage occurring. 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 purge is in progress during the accident. This contributor would not be impacted by the proposed change.

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

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

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

5.4.2 NUREG-1493, "Performance-Based Containment Leak-Test Program"

Based on information provided in NUREG-1493, the increased risk of population dose attributable to this extension of the test interval would be extremely small. NUREG-1493 includes the results of a sensitivity study performed to explore the risk impact of several alternate leak rate testing schedules. Alternative 6 from this study examines relaxing the Integrated Leakage Rate Test frequency from 3 in 10 years to 1 in 20 years. Using best estimate data, the NUREG 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:

- the effectiveness of Type B and C tests in identifying potential leak paths (about 97%);
- a low likelihood of Integrated Leakage Rate Test-identified leakage exceeding twice the allowable; and
- the insensitivity of risk to containment leak rate.

NUREG-1493 concludes that even increasing the Integrated Leakage Rate Test interval to once per 20 years would "lead to an imperceptible increase in risk." The proposed extension of the test interval is concluded to be bounded by the analyses of NUREG-1493. The associated increase in risk is not significant. By comparison, the probability of occupational dose associated with performance of an integrated leakage rate test is 100%.

5.4.3 Large and Small Early Release Frequencies

The most important causes for a Large Early Release were found to be large containment isolation failures and phenomenological effects associated with severe accidents. The large containment isolation contributor includes failure to isolate the large supplemental purge penetrations in the unlikely event that a purge is in progress during the accident. Large pre-existing leaks in the reactor containment building are not modeled in the South Texas Project Probabilistic Risk Assessment.

The results for LERF are dominated by sequences caused by a phenomenon called Induced Steam Generator Tube Rupture which occurs when the secondary side of the steam generators dries out after a core damage event with the reactor coolant system intact at high pressure. High temperature coolant circulates through the Reactor Coolant System, heating up the steam generator tubes to the point of failure. The Induced Steam Generator Tube Rupture sequences are primarily caused by core damage scenarios that involve loss of all station AC power (Station Blackout). The Integrated Leakage Rate Test does not test this pathway through the steam generators.

The dominant cause of containment bypass is failure of the supplementary containment purge to isolate during an accident sequence. This sequence is not affected by Integrated Leakage Rate Testing.

The Small Early Release Frequency evaluates the potential for a small pre-existing leak. A small containment failure existing prior to core damage is the most important contributor to small early releases. Most of this contribution is from steam generator tube ruptures, 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.

Implementation of the proposed change will have no effect on the South Texas Project Probabilistic Risk Assessment for core damage frequency or large early release frequency, and consequently meets the requirements of Regulatory Guides 1.174 and 1.177. However, the proposed change would affect the results of a PRA analysis for a small containment isolation failure.

5.4.4 Risk Assessment Results

The dominant cause of early containment failure in the South Texas Project PRA is not affected by ILRT. The purpose of the ILRT is to confirm that there is no pre-existing "small hole" through the containment. Therefore, this analysis assumes that the "small hole" is present prior to occurrence of core damage, and there is no large early release.

A set of sensitivity analyses was performed using the current South Texas Project Level II PRA, STP_1997. The assumptions used in the analyses:

- The ILRT has no effect on the quantification of core damage frequency and large early release frequency.
- The ILRT will only find small openings (no larger than three inches) in containment, and therefore can be evaluated for small containment failure Level 2 analysis.
- All other assumptions in STP_1997 are unaffected and remain valid.

Under normal conditions (Base Case), the likelihood of a Small Early Release in the current South Texas Project PRA is:

5.67 E-07 per reactor year

When the containment isolation failure factor is set at 0.1, the small early release frequency rises to:

2.34 E-06 per reactor year

If containment isolation is assumed to fail (Containment isolation failure factor is set to 1.0), the small early release frequency rises to:

9.10 E-06 per reactor year

The respective population dose results for the three sensitivity cases are summarized as:

- Base Case 20.8 person-rem/reactor year at fifty miles
- Failure Factor at 0.1 20.8 person-rem/reactor year at fifty miles
- Failure Factor at 1.0 22.3 person-rem/reactor year at fifty miles

The primary conclusion is that the population risk is dominated by scenarios in which the containment is bypassed or in which the containment itself fails and not by containment leakage rates several orders of magnitude above the current requirements. Assuming a failure in a containment penetration equivalent to a three-inch diameter opening, the increase in population risk is only 1.5 person-rem per reactor year at 50 miles and varies less than 7% from the base case. This is because the population dose from an assumed core damage event is dominated by the Large Early Release Frequency, which in turn is dominated by the Induced Steam Generator Tube Rupture sequences.

5.4.5 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. This is a proposed change to the inspection interval and does not have any effect on containment design margins or isolation system capability. On-going performance monitoring and inspection is described in Section 4.3 and 5.1.

6.0 Statement of No Significant Hazards Consideration

The proposed Technical Specification revision extends the current interval for Type A testing. The current test interval of ten years would be extended on a one-time basis to 15 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.

Criterion 1: 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 testing interval will not increase the probability of an accident previously evaluated. The containment Type A testing interval extension is not a modification and the testing interval extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. Research documented in NUREG-1493 has determined that Type B and C tests can identify the vast majority (more than 95%) of all potential leakage paths.

NUREG-1493 concluded that reducing the Type A test frequency to one per twenty years leads to an imperceptible 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 show leakage does not exceed acceptance criteria, indicating a very leak-tight containment. Inspections required by the Maintenance Rule and ASME code are performed in order to identify indications of containment degradation that could affect leak tightness.

Experience at the South Texas Project demonstrates that excessive containment leakage paths are detected by Type B and C Local Leakage Rate Tests. Type B and C testing will identify any containment opening, such as a valve, that would otherwise be detected by the Type A tests. These factors show that a Type A test interval extension will not involve a significant increase in the consequences of an accident.

Criterion 2: 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 testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no physical changes being made to the plant and there are no changes in operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

Criterion 3: 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 decrease in Type A testing frequency would have a minimal effect on this risk because 95% of the potential leakage paths are detected by Type B and C testing.

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 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 revised Technical Specifications will not endanger the public health and safety.

7.0 Environmental Assessment

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupations radiation exposure.

Based on the above, the South Texas Project concludes that the proposed change meets the criteria specified in 10CFR51.22 for a categorical exclusion from the requirements of 10CFR51.22 relative to requiring a specific environmental assessment by the Commission.

8.0 CONCLUSION

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.

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 exemption request and Technical Specification changes.

Postponement of Type A testing could increase the probability of a Small Containment Leakage Failure. This increased probability has been shown to result in a very small increase in the calculated population dose for the South Texas Project. However, the increased calculated population dose is bounded by 10CFR100 limits. Therefore, the extension would not cause a significant change in risk, nor cause the NRC Safety Goals to be exceeded. Sufficient safety margins are maintained.

9.0 IMPLEMENTATION OF THE PROPOSED CHANGE

This amendment request is neither an emergency nor exigent. However, the South Texas Project requests that the Nuclear Regulatory Commission approve the proposed amendment by January 31, 2002, to support scheduling activities of the Unit 2 refueling outage currently scheduled for October 2002. Date of effectiveness would occur within 30 days following approval.

10.0 REFERENCES

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

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

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

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

5. “Request for Exemption to Exclude Certain Components From The Scope of Special Treatment Requirements Required by Regulations,” dated August 31, 2000 (NOC-AE-00000918).

ATTACHMENT 2

REACTOR CONTAINMENT DESIGN AND CONSTRUCTION

SOUTH TEXAS PROJECT
UNITS 1 & 2
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, with four-foot cylinder wall thickness, three-foot dome thickness, and 18-foot mat thickness. The top of the foundation mat is 41-1/4 feet below grade.

A continuous welded steel liner plate is provided on the entire inside face of the containment to limit release of radioactive materials into the environment. The nominal thickness of the liner in the wall and dome 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. An 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.

The reinforcement in the dome is provided in a meridional and circumferential pattern up to 45 degrees from the spring line, with the remaining area being reinforced using a grid pattern. Reinforcement is provided on both faces of the dome wall. Radial ties are provided to both resist radial shear and prevent de-lamination of the dome under pre-stressing.

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 CURRENT LICENSING BASIS"

SOUTH TEXAS PROJECT
UNITS 1 & 2
APPLICATION OF REGULATORY GUIDE 1.174, "AN APPROACH FOR USING PROBABILISTIC RISK ASSESSMENT IN RISK-INFORMED DECISIONS ON PLANT-SPECIFIC CHANGES TO THE CURRENT LICENSING BASIS"

1.0 INTRODUCTION

The South Texas Project has completed a risk assessment of the proposed one-time extension of the ten-year containment Type A test interval to fifteen years. The risk assessment followed 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 Current 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:

- The proposed change meets the current regulations.
- Defense-in-depth is maintained.
- Sufficient safety margins are maintained.
- Proposed increases in risk, and their cumulative effect, are small and do not cause the NRC Safety Goals to be exceeded.
- Performance-based implementation and monitoring strategies are proposed that 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. For example, 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 'opened' following a plant post-maintenance test. (For example, 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 bypassed (Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences) and small containment isolation 'failure-to-seal' events (Class 4 and 5 sequences) were not accounted for 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 evaluated in EPRI TR-104285.

- Evaluation of the risk impact of extending the Type A test interval from ten to 15 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 changes proposed, the reason for the changes, and results and insights from an analysis of available data on equipment performance.

  See Attachment 1, Sections 3.2, 4.1, and 5.1.

- A tabulation of the current licensing basis accident parameters that are 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 5.4.

- 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 5.4.4.

- Identification of how and where the proposed change will be documented as part of the plants licensing basis (e.g., FSAR, TS, licensing conditions). This should include proposed changes and/or enhancements to the regulatory controls for high risk-significant structures, systems, and components not subject to any requirements, or where the requirements are not commensurate with the risk significance.

See Attachment 4.

- Those key assumptions in the PRA, elements of the monitoring program, and commitments made to support the application.

  See Attachment 1, Section 5.4.

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

  It is not necessary to increase requirements.

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

  See Attachment 4.

- A list describing all events included in the risk assessment.

  See Attachment 1, Section 5.4.3.

- A list of systems and components addressed in the risk assessment, the failures considered for each and the basis for excluding failures, and the dependencies between systems and components

  See Attachment 1, Section 5.4.2.

- 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 5.4.2.

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. Containment and severe accident issues identified in NUREG-1150, such as direct containment heating and induced steam generator rupture, have been considered in the development and quantification of the containment event tree.

The applicable analysis from the South Texas Project Probabilistic Safety Assessment is Top Event 10 in which the reactor pressure vessel is failed or unisolated prior to vessel breach. 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_1997 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.

ATTACHMENT 4

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 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).

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, L_a, is 0.3% of primary containment air weight per day.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ 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.

ATTACHMENT 5

REVISED TECHNICAL SPECIFICATION PAGES

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 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).

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, L_a, is 0.3% of primary containment air weight per day.

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria for the overall air lock leakage rate is $\leq 0.05\ L_a$ 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.