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

February 12, 2001
NOC-AE-000712
File No.: G20.02.01
G21.02.01
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 4.4.6.2.2 for
Reactor Coolant System Pressure Isolation Valve Surveillance

Pursuant to 10CFR50.90, the South Texas Project requests an amendment to the South Texas Project Unit 1 and Unit 2 Technical Specifications. This change applies to both units.

The proposed amendment will revise the Technical Specifications by deleting surveillance requirement 4.4.6.2.2.e, which refers to ASME Code Section XI, paragraph IWV-3427(b) as a requirement for demonstrating that each Reactor Coolant System Pressure Isolation Valve specified in Technical Specification Table 3.4-1 is operable. IWV-3427(b) specifies requirements for use of valve leak rate test data to determine test frequency and need for repair or replacement of affected valves. OMa-1988 is currently the applicable code for these valves and does not have these requirements.

The required affidavit, the Description and Assessment associated with the proposed Technical Specification change, and the proposed replacement Technical Specification pages are included with the attached request.

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 South Texas Project requests that this change be approved by the Nuclear Regulatory Commission by June 1, 2001, and that the effective date for this amendment be within 30 days following approval. Although this request is neither exigent nor an emergency, prompt review by the Nuclear Regulatory Commission is requested.

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

~~A001~~
A047

If there are any questions associated with this Technical Specification change, 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) Affidavit
2) Description and Assessment
3) Proposed Replacement Page For Technical Specification Surveillance Requirement 4.4.6.2.2
4) Revised Page for Technical Specification Surveillance Requirement 4.4.6.2.2

cc:

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

ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
STP Nuclear Operating Company, et al.,	)	Docket Nos. 50-498 50-499
	)	
South Texas Project Units 1 and 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 change; 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 12th day of February, 2001.

LINDA RITTENBERRY
Notary Public, State of Texas
My Commission Expires
OCT. 9, 2001

Linda Rittenberry
Notary Public in and for the
State of Texas

ATTACHMENT 2

DESCRIPTION AND ASSESSMENT

SOUTH TEXAS PROJECT
UNITS 1 & 2
PROPOSED AMENDMENT TO TECHNICAL SPECIFICATION
SURVEILLANCE REQUIREMENT 4.4.6.2.2
DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

1.1 This proposed Technical Specification change is to delete surveillance requirement 4.4.6.2.2.e for monitoring pressure isolation valve leakage.

1.2 Markup of existing Technical Specifications:

See Attachment 3.

1.3 Proposed Technical Specifications:

See Attachment 4.

1.4 Updated Final Safety Analysis Report:

The evaluations performed in support of this license amendment request do not result in any required changes to the Updated Final Safety Analysis Report.

2.0 BACKGROUND

Technical Specification 3/4.4.6.2 addresses the allowed limits for Reactor Coolant System leakage. Surveillance requirement 4.4.6.2.2.e states that verification of leakage from Reactor Coolant System Pressure Isolation Valves shall be as outlined in the ASME Code, Section XI, paragraph IWV-3427(b). IWV-3427(b) specifies limits on leakage rates to determine test frequency and requires projection of identified leakage rates for scheduling valve replacement or repair.

The South Texas Project Inservice Testing program for the first inspection interval was written in accordance with the ASME Section XI Code, 1983 edition with Summer 1983 addenda. As discussed in Reference 1, and approved by the Nuclear Regulatory Commission in Reference 2, the South Texas Project will begin the second 120-month interval using the current Inservice Testing program (1983 edition with Summer 1983 addenda). The South Texas Project intends to implement a risk-informed Inservice Testing program based on the 1987 edition with 1988 addenda to the ASME OM (OMa-1988) Code, as referenced in the 1989 ASME Section XI Code, for the second interval on a system-by-system/component-by-component approach with full implementation no later than the end of November 2001. As stated in Reference 1, as each system/component is brought into the new Code requirements, inservice testing for that

system/component will be in accordance with the risk-informed plan based on the OMa-1988 code.

The 1989 ASME Section XI Code has been applied to the Reactor Coolant System Pressure Isolation Valves. The ASME OMa-1988 Code, referenced in the Section XI Code, does not specify limits on leakage rates to determine test frequency and does not require projection of identified leakage rates for scheduling valve replacement or repair.

3.0 DESCRIPTION

Technical Specification surveillance requirement 4.4.6.2.2.e states that each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated operable by verifying leakage to be within its limit as outlined in the ASME Code, Section XI, paragraph IWV-3427(b). IWV-3427(b) requires the following:

> For valves 6 in. nominal pipe size and larger, if a leakage rate exceeds the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate by 50% or greater, the test frequency shall be doubled; the tests shall be scheduled to coincide with a cold shutdown until corrective action is taken, at which time the original test frequency shall be resumed. If tests show a leakage rate increasing with time, and a projection based on three or more tests indicates that the leakage rate of the next scheduled test will exceed the maximum permissible leakage rate by greater than 10%, the valve shall be replaced or repaired.

This surveillance requirement is to be deleted from the South Texas Project Technical Specifications as a permanent change.

Leakage rate test measurements of South Texas Project Reactor Coolant System isolation valves will continue to be taken pursuant to the surveillance requirements of Technical Specification 4.4.6.2.2, which is consistent with the requirements of code OMa-1988, paragraph 4.2.2.3.e for analysis of leakage rates. Records of tests are maintained in accordance with the requirements of code OMa-1988, paragraph 6.3, which includes analysis of deviations in test values.

4.0 TECHNICAL ANALYSIS

The limiting accident from overpressurization of low pressure systems is the interfacing system Loss of Coolant Accident. This event results from failure of the isolation valves between the Reactor Coolant System and a lower pressure interfacing system, leading to a primary coolant leak that bypasses containment. Only the Residual Heat Removal and Safety Injection systems have large enough pipes to produce a line break that can lead to a rapid loss of Reactor Coolant System inventory and Refueling Water Storage Tank contents outside the containment. Should a low pressure system component rupture

inside containment, coolant would be retained so that long-term cooling would be available via Emergency Core Cooling System recirculation.

The higher operating pressure of the Reactor Coolant System is isolated from the lower pressures of the following interfacing systems to prevent overpressurization and rupture of low pressure piping:

- High Head Safety Injection System
- Low Head Safety Injection System
- Residual Heat Removal System
- Accumulators

Isolation of these systems from the Reactor Coolant System, as listed in Table 3.4-1 of the South Texas Project Technical Specifications, is provided by:

- High Head Safety Injection (HHSI) Cold Leg Injection Check Valves (XSI0007)
- HHSI Hot Leg Recirculation Check Valves (XSI0009)
- Low Head Safety Injection (LHSI) Hot Leg Recirculation Check Valves (XSI0010)
- LHSI/HHSI Hot Leg Recirculation Check Valves (XRH0020)
- LHSI/RHR Cold Leg Injection Check Valves (XRH0032)
- LHSI/HHSI/RHR/Accumulator Cold Leg Injection Check Valves (XSI0038)
- Accumulator Cold Leg Injection Check Valves (XSI0046)
- Residual Heat Removal Suction Isolation Valves (XRH0060, 61)

4.1 Affected Systems

High Head Safety Injection System

Each of the three high head safety injection trains outside the reactor containment are normally isolated from the Reactor Coolant System hot leg by three check valves (including pressure isolation valves XSI0009 and XSI0010) and one normally closed motor-operated valve, all in series. Three check valves in series (including pressure isolation valves XSI0038 and XSI0007) isolate the high head safety injection pump from the Reactor Coolant System cold leg. All such valves in the piping to either the hot leg or cold leg would have to fail in order to expose the high head safety injection pump and associated piping to Reactor Coolant System pressure.

Low Head Safety Injection System

Each of the three low head safety injection trains outside the reactor containment are normally isolated from the Reactor Coolant System hot leg by three check valves (including pressure isolation valves XSI0010 and XRH0020) and one normally closed

motor-operated valve, all in series. Three check valves in series (including pressure isolation valves XSI0038 and XSI0007) isolate the low head safety injection pump from the Reactor Coolant System cold leg.

The Refueling Water Storage Tank provides coolant inventory to the high head and low head safety injection pumps. Isolation from Reactor Coolant System pressure is provided by the three check valves and motor-operated valve downstream of each safety injection pump as described above. In addition to the isolation valves separating the safety injection pumps from the Reactor Coolant System, an additional check valve is located between the tank and the safety injection pumps.

Residual Heat Removal System

Each of the three Residual Heat Removal system trains is normally isolated from the Reactor Coolant System hot leg by two normally closed motor-operated valves (including pressure isolation valves XRH0060 and XRH0061) upstream of the Residual Heat Removal pump. They are located entirely within the reactor containment building. Both valves have a pressure interlock to prevent overpressurization of the Residual Heat Removal system from the Reactor Coolant System. Isolation from the Reactor Coolant System cold leg is provided by two check valves in series (including pressure isolation valves XSI0038 and XRH0032).

The Residual Heat Removal heat exchanger is located downstream of the Residual Heat Removal pump. The heat exchanger is normally isolated from the Reactor Coolant System cold leg by two check valves in series (including pressure isolation valves XSI0038 and XRH0032), from the hot leg discharge by two check valves in series (including pressure isolation valves XRH0020 and XSI0010), and from the hot leg suction by the motor-operated valves identified previously.

The shell side of an RHR heat exchanger is supplied by one of the three Component Cooling Water (CCW) system trains. In the event of an interfacing LOCA from the Reactor Coolant System to the Residual Heat Removal system, consequential ruptured tubes in the heat exchanger would result in a LOCA through the CCW system. The Component Cooling Water System is located primarily outside containment.

The CCW system supply line to the RHR heat exchanger includes one motor-operated valve outside containment and one check valve inside containment. The return line from the heat exchanger includes two motor-operated valves, one inside containment and one outside containment. These valves are not included under Surveillance Requirement 4.4.6.2.2.e. Operator action is required to close the motor-operated valves in the CCW supply and return lines. These valves will enable operators to isolate a faulted RHR heat exchanger from the CCW system.

The Refueling Water Storage Tank receives coolant inventory from trains B and C of the Residual Heat Removal System. In addition to the valves blocking Reactor Coolant System operating pressure from the Residual Heat Removal System, two locked closed gate valves, one inside and one outside containment, provide separation for the Refueling Water Storage Tank from the RHR System.

Accumulators

The three accumulators are pressure vessels located inside containment and are partially filled with borated water and pressurized with nitrogen gas. The accumulators inject borated water into the reactor pressure vessel in the event Reactor Coolant System pressure drops below a set level. The accumulators are protected from Reactor Coolant System operating pressure by two check valves in series (pressure isolation valves XSI0038 and XSI0046).

There is no risk of inventory release outside containment associated with an interfacing system LOCA with the accumulators. They are located entirely within containment.

4.2 Leakage Rate Analysis

Adverse test results will be addressed under the corrective action program and by application of the Maintenance Rule. Engineering analysis of test results can take into account special circumstances associated with a test that would affect the conclusions.

Leakage rate test measurements of South Texas Project Reactor Coolant System isolation valves will continue to be taken pursuant to the surveillance requirements of Technical Specification 4.4.6.2.2, which is consistent with the requirements of code OMa-1988, paragraph 4.2.2.3.e for analysis of leakage rates. Code OMa-1988, paragraph 6.3, requires records of tests, including analysis of deviations in test values.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration Determination

Pursuant to 10CFR50.91, this analysis provides a determination that the proposed change to the Technical Specifications described previously does not involve any significant hazards consideration as defined in 10CFR50.92:

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

   This Technical Specification change only affects trending of valve leakage rate test results to anticipate the expected leakage rate performance of Reactor Coolant System

pressure isolation valves. Redundant pressure isolation valves are included in the plant to ensure continued protection of lower pressure systems from exposure to the higher pressure of the Reactor Coolant System in the event that excessive leakage develops in an isolation valve. In addition, leakage rate tests of Reactor Coolant System pressure isolation valves will continue to be performed with no change in the accepted amount of leakage or frequency. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The limiting event associated with these valves is a Loss of Coolant Accident. This has already been reviewed as part of the South Texas Project Updated Final Safety Analysis Report. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   This proposed change only removes a requirement for trending of pressure isolation valve leakage rates. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   There is no change in the design of the plant associated with this proposed license amendment. The only impact of this change is in the prediction of when a particular pressure isolation valve may have a leakage rate higher than what is allowed. Adverse test results will be addressed under the corrective action program and by application of the Maintenance Rule. Engineering analysis of test results can take into account special circumstances associated with a test that would affect the conclusions.

   Leakage rate test measurements of South Texas Project Reactor Coolant System isolation valves will continue to be taken pursuant to the surveillance requirements of Technical Specification 4.4.6.2.2, which is consistent with the requirements of code OMa-1988, paragraph 4.2.2.3.e for analysis of leakage rates. Code OMa-1988, paragraph 6.3, requires records of tests, including analysis of deviations in test values. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

5.2 Regulatory Safety Analysis

<u>Applicable Regulatory Requirements/Criteria</u>

10CFR50, Appendix A, General Design Criterion 14: Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected,

and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

10CFR50, Appendix A, General Design Criterion 54: Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

10CFR50, Appendix A, General Design Criterion 55: Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

5.3 Analysis

The proposed change removes a requirement to trend leakage rates of containment isolation valves to determine inspection frequencies. This change does not change the compliance with any of the above General Design Criteria and is consistent with the current ASME Section XI inspection standards.

5.4 Conclusion

Based upon the above, deleting the surveillance requirement for trending Reactor Coolant System pressure isolation valve leakage test results from Technical Specifications does not have a significant impact on safe operation of the South Texas Project.

6.0 ENVIRONMENTAL EVALUATION

The South Texas Project has reviewed the attached proposed license 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.

7.0 IMPLEMENTATION

The South Texas Project requests that this change be approved by the Nuclear Regulatory Commission by June 1, 2001, and that the effective date for this amendment be within 30 days following approval.

8.0 REFERENCES

1) Correspondence from T. J. Jordan, South Texas Project, to NRC Document Control Desk, dated February 1, 1999 (NOC-AE-000391)

2) Correspondence from Nuclear Regulatory Commission to William T. Cottle, STP Nuclear Operating Company, dated March 15, 1999

ATTACHMENT 3

PROPOSED REPLACEMENT PAGE FOR

TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.4.6.2.2

Delete Surveillance Requirement 4.4.6.2.2.e

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous radioactivity and particulate radioactivity channels at least once per 12 hours;

b. Monitoring the containment normal sump inventory and discharge at least once per 12 hours;

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

~~e. As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).~~

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

ATTACHMENT 4

REVISED PAGE FOR TECHNICAL SPECIFICATION

SURVEILLANCE REQUIREMENT 4.4.6.2.2

Delete Surveillance Requirement 4.4.6.2.2.e

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous radioactivity and particulate radioactivity channels at least once per 12 hours;

b. Monitoring the containment normal sump inventory and discharge at least once per 12 hours;

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.