UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INSERVICE TESTING PLAN

REQUEST FOR RELIEF NO. RR-56 (RR-52)

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2 (STP)

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

The *Code of Federal Regulations*, 10 CFR 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (the Code) and applicable addenda, except where alternatives have been authorized or relief has been granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the licensee must demonstrate that (1) the proposed alternatives provide an acceptable level of quality and safety, (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (3) conformance is impractical for its facility. Section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The 1989 Edition of the ASME Code is the latest edition incorporated by reference in paragraph (b) of Section 50.55a. Subsection IWV of the 1989 Edition, which gives the requirements for IST of valves, references Part 10 of the American National Standards Institute/ASME *Operations and Maintenance Standards* (OM-10) as the rules for IST of valves. OM-10 replaces specific requirements in previous editions of Section XI, Subsection IWV of the ASME Code.

2.0 BACKGROUND

By letter dated November 16, 1998, STP Nuclear Operating Company (STPNOC), the licensee, submitted a new relief request RR-56 (RR-52) for the first 10-year interval for the STP IST Program. RR-56 (RR-52) requested relief from the ASME Section XI code requirement in order to extend testing of component cooling water (CCW) and safety injection (SI) check valves for closure exercise to the same frequency as 10 CFR Part 50, Appendix J, Option B, containment leak-rate test frequency. By letter dated February 23, 1999, STPNOC amended this request providing additional information in support of the proposed test interval.

The STP IST Program for the first 10-year interval began on August 25, 1988, and June 19, 1989, respectively. By letter dated March 15, 1999, the staff approved extension of the first

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120-month IST interval for both Unit 1 and Unit 2 through November 30, 2001. The STP IST Program for the first 10-year interval was developed to the 1983 Edition of the ASME Boiler and Pressure Vessel Code, Section XI through Summer 1983 Addenda.

In August 1998, during a review of the IST Program, the licensee determined that several of the valves associated with this relief request (CCW system containment isolation check valves) had not been tested to the closed position (as required to perform a specified safety function) within the required test periodicity of the Section XI ASME Code. By letter dated August 28, 1998, the licensee requested that the NRC exercise enforcement discretion not to enforce compliance with the action required in the STP Technical Specification Surveillance Requirement 4.0.5 as it applies to IST requirements for the identified containment isolation check valves in Unit 1. The requested duration of the discretion (for the Unit 1 CCW system check valves identified in this relief request) was for 14 days to allow on-line testing of the subject valves. By letter dated September 1, 1998, the NRC exercised the requested enforcement discretion.

3.0 CHECK VALVE RELIEF REQUEST RR-56 (RR-52)

RR-56 (RR-52) pertains to the following CCW and SI system containment isolation check valves.

Valve ID:	2R201(2)TCC0013	2R201(2)TCC0058	2R201(2)TCC0123
	2R201(2)TCC0138	2R201(2)TCC0183	2R201(2)TCC0198
	2N121(2)XSI0005A	2N121(2)XSI0005B	2N121(2)XSI0005C
	2N121(2)XSI0030A	2N121(2)XSI0030B	2N121(2)XSI0030C

These category A/C valves serve as containment isolation boundary valves, which close to maintain containment integrity during a design-basis accident. The licensee states that the closure function required by ASME IWV-3522 can be satisfied by the performance of the local leak-rate testing per Option B of 10 CFR Part 50, Appendix J.

The CCW check valves also open to provide cooling water to various safety-related components such as the reactor containment fan coolers, residual heat removal pumps, and heat exchangers.

The SI check valves also open to inject borated water from either the refueling water storage tank (RWST) or the containment sump to the reactor coolant system (RCS) cold legs during the cold leg injection phase of SI. Additionally, they open to recirculate borated water from the containment sump to the RCS hot legs during the hot leg recirculation phase of SI.

The licensee requests relief from the exercise test frequency requirement of IWV-3522. ASME Section XI Code, 1983 Edition, paragraph IWV-3522, requires check valves that have a safety function to close on cessation of flow, to be tested in a manner that proves that the disk travels to the seat promptly on cessation or reversal of flow on a quarterly basis, when practical. ASME Section XI Code, 1983 Edition, paragraph IWV-3521 and OM-10, paragraph 4.3.2.1 require check valves to be exercised to their safety position quarterly. Paragraph 4.3.2.2 allows check valves to be exercised at a refueling frequency where quarterly exercising is not practical.

3.1 Licensee's Basis for Request

The licensee provided the following basis for the relief request:

> Pursuant to 10CFR50.55a(a)(3)(i), the South Texas Project requests relief from ASME Section XI Code, 1983 Edition, paragraph IWV-3522. These check valves have a safety function in the closed direction as containment isolation valves. There are no intra- or inter-system cross-ties downstream of these valves that would cause a diversion of flow from another pump if the check valve did not close.
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> Due to the fact that there are no cross-ties downstream of the valves, they lack design provisions for system testing to verify closure capability in any plant condition.
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> Leak rate testing verifies valve closure by validating the valve seats properly and is leak tight, and provides more information about the closed position than a simple backflow test. Leak rate testing is performed in accordance with the requirements of 10CFR50 Appendix J, Option B. The interval between leak rate testing for each valve can be set to a maximum of five years based on completion of two as-found tests within the administrative limits set for the valve. The test interval shall be set at the original value of 30 months if local leak rate test results exceed the administrative limit. Additionally, a cause determination is performed and corrective action identified to reduce the leakage rate. The condition reporting process is used to identify inspection and rework requirements for the valve. Other valves with similar designs will be evaluated based on the failure mechanism identified in the results of the condition investigation.
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> NUREG-1482 ["Guidelines for Inservice Testing at Nuclear Power Plants"], Section 4.1.4, allows the extension of the test interval to refueling outage frequency for check valves where the only practical means of verifying check valve closure is by performing the Appendix J Leak Test. The South Texas Project has adopted Option B of Appendix J that allows these check valves to be leak tested on a frequency not to exceed once every five years.
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> Disassembly provides limited information on a check valve's ability to seat properly on cessation of flow. Following reassembly, the Code requires a post-assembly test, which would reopen the check valve without providing assurance the disk would return to the closed position. Disassembly of these check valves is not practical due to the design complexity of the check valves, the increased probability of human error during valve reassembly, foreign material exclusion concerns, and ALARA [as low as reasonably achievable] considerations.
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> The subject valves have exhibited a history of satisfactory operation. Based on their performance history, it is believed that the current Probabilistic Risk Assessment (PRA) modeling of the failure rates for these valves is still accurate. Irrespective of the failure rate modeling, the current South Texas Project Probabilistic Risk Assessment model indicates that the potential failure of these

valves to close has no impact on core damage frequency. In addition, the impact of these valves *[assuming complete failure]* from a Large Early Release standpoint is minimal.

Based on the above, it is evident that in the event that containment isolation is necessary, the subject valves will have a high probability of performing their intended safety function. Therefore, the South Texas Project believes that the safety significance and potential consequences of the proposed relief are extremely small.

Limitations:

The South Texas Project is unique in having a three independent train design. With no intra- or inter-system cross-tie downstream of these valves, provisions for performing closure exercise testing are not available.

Specific Considerations:

A performance history of these valves indicates that there is no evidence that would indicate that these valves would fail to close on a cessation [or reversal of] flow. A probabilistic failure model was developed based on this maintenance history. The results of this analysis indicate that the mean time to failure for the Component Cooling Water check valves is 6.0 - 8.6 years. The mean time to failure for the Safety Injection check valves is greater than 20 years.

The attached Table 1 [to STPNOC letter dated February 23, 1999] is a reliability report for the subject Component Cooling Water check valves and Safety Injection check valves.

Justification for the Granting of Relief:

The containment isolation function is designed to limit the leakage of radioactive materials through lines penetrating the Reactor Containment Building (RCB) so that the site boundary dose guidelines specified in 10CFR100 are not exceeded following a Loss-of-Coolant Accident (LOCA) or other design basis accident. Upon receipt of the appropriate signals, isolation of the RCB is accomplished by automatic isolation of all non-essential fluid systems that penetrate the RCB.

The plant-specific Probabilistic Risk Assessment (PRA) analyzes containment isolation for mitigating releases to the environment. The index of interest is the Large Early Release Frequency (LERF). This measures the frequency of an event that will exceed 10CFR100 in a short period of time. This function is documented in the *STPEGS [South Texas Project Electric Generating Station] Probabilistic Safety Assessment Containment Isolation Function Package.* The only valves of interest in terms of a LERF are the ones providing Component Cooling Water (CCW) flow to the Residual Heat Removal (RHR) heat exchanger.

CCW flow to the RHR heat exchangers is required for long-term decay heat removal after a design basis initiating event. Valves CCW-0013, 0123 and 0183 are required to open and remain open to provide this function. In the unlikely event of a tube rupture in one of the RHR heat exchangers, radioactive materials would be entrained in the CCW system. The operator in the control room would be made aware of this condition by the CCW radiation monitor. Additional indications of this condition would be given by the CCW flow indications, the high and low flow alarms, surge tank level indications, and tank high level alarms. The isolation valves in this system are designed to be operated remote-manually from the control room.

The CCW supply check valves inside reactor containment are included in the analysis of an Interfacing Systems LOCA. In the event of an Interfacing Systems LOCA, the RHR heat exchanger tubes could fail resulting in a release path through the CCW lines. The CCW supply line contains one motor-operated valve and the CCW supply check valve; the CCW return line contains two motor-operated valves. Operator action is required to close the MOVs in the CCW supply and return lines. An analysis of the change in Interfacing Systems LOCA frequency was made by assuming that all three CCW check valves fail to close on demand. The contribution to LERF from Interfacing Systems LOCA changed from 3.0×10^{-08} per year to 7.8×10^{-08} per year which is less than the 1.0×10^{-07} per year change identified in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis." A copy of the Interfacing System LOCA analysis is provided as Addendum 1 to this Relief Request [reference STPNOC letter dated February 23, 1999]. A similar analysis has been performed for the Low Head Safety Injection check valves and is provided as Addendum 2 to this relief request [reference STPNOC letter dated February 23, 1999].

Based on the analysis in the PRA, these valves have low safety significance for preventing an Interfacing Systems LOCA and/or containment isolation.

3.2 Proposed Alternate Testing

The licensee proposed the following:

> Closure verification of these check valves will be performed by leak testing in accordance with 10CFR50, Appendix J leak test on a frequency specified by Option B of Appendix J.

3.3 Evaluation

In NRC Generic Letter 89-04 ("Guidance on Developing Acceptable Inservice Testing Programs"), Position 10, the staff has determined that testing of containment isolation valves (CIVs) in accordance with Appendix J and the analysis of leakage rates and corrective action requirements of paragraph IWV-3426 and 3427(a) are acceptable. The use of either Option A or B is allowed by this position and relief is not required. Use of this position must be documented in the licensee's IST program. CIVs that have other leak tight functions (such as the valves addressed in this relief request) must, however, meet the requirements of IWV (or Part 10 as discussed in NUREG-1482, Section 4.4.5) unless relief is granted or an alternative test is authorized by the NRC.

Specifically, the Code requires both a leak-rate test and an exercise test for active Category A valves. The licensee will continue to full-stroke exercise the subject check valves to the open position each refueling outage, not to exceed once every 2 years. The licensee proposes to combine the exercise test to the closed position (currently performed each refueling outage) with the leak-rate test for these CIVs on a schedule in accordance with Option B of 10 CFR Part 50, Appendix J. Option B of Appendix J requires that Type C containment leak-rate tests be "conducted...periodically...at intervals based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release...."

Regulatory Guide (RG) 1.163, which is referenced in 10 CFR Part 50, Appendix J, provides guidance on an acceptable performance-based leak-test program. This RG references NEI 94-01, Revision 0, dated July 26, 1995, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50 Appendix J."

The guidance provided by these two documents states that Type C tests shall be performed at a frequency of at least once per 30 months, until adequate performance has been established. Upon completion of two consecutive successful as-found Type C tests, the interval between tests may be extended to no greater than 60 months. If the test results are not acceptable, then the following corrective actions apply: (1) test frequency shall be returned to 30 months, and (2) a cause determination shall be performed and corrective actions identified. Once the cause determination and corrective actions have been completed, acceptable performance may be reestablished and the testing frequency returned to the extended interval after completion of two consecutive successful as-found tests, as described earlier.

3.3.1 Engineering Analysis In Support of the Proposed Relief Request

The licensee indicated that the only plausible test strategy for verifying closure of these containment isolation check valves, other than utilizing the 10 CFR Part 50, Appendix J, Option B, is a differential pressure test across the check valve seating surface. This involves running the CCW or SI pumps against a closed downstream isolation valve, installing temporary gauges upstream of the check valves, and relieving the pressure in the upstream penetration space. The licensee stated that the consequences of this test strategy are possible damage to safety-related equipment due to hydraulic transients and pump bearing fatigue.

3.3.2 Safety Significance of the Proposed Alternative

From a risk analysis perspective, nine of the twelve valves affected by this relief request may contribute to the LERF since these valves perform a containment isolation function and their failure could increase the probability of a radionuclide release that bypasses the containment. In PRAs, these valves play a role in preventing the interfacing-systems loss-of-coolant accident (ISLOCA) sequences. These sequences usually involve the failure of the RCS pressure boundary into a low-pressure piping system. In the case of this relief request, the two postulated ISLOCA scenarios of interest are a LOCA outside containment through the RHR heat exchanger to the CCW system, and LOCA outside containment through the low head safety injection (LHSI) penetration. Each of these scenarios is discussed in more detail below.

3.3.2.1 ISLOCA Through the RHR Heat Exchanger to the CCW System

Valves CCW-0013, CCW-0123, and CCW-0183 (all of which are part of this relief request) are required to open and remain open to provide CCW flow to the RHR heat exchangers for long-term decay heat removal after a design basis initiating event. In the unlikely event of a tube rupture in one of the RHR heat exchangers, radioactive materials would be entrained in the CCW system. The operator in the control room would be made aware of this condition by the CCW radiation monitor. Additional indications of this condition would be given by the CCW flow indications, the high and low flow alarms, surge tank level indications, and tank high level alarms. Isolation valves in the CCW system (one motor-operated valve in the CCW supply line to each RHR heat exchanger and two motor-operated valves in each CCW return line) are designed to be operated remote-manually from the control room.

In evaluating the risk increase from the test interval extension of the CCW inside reactor containment supply check valves from ISLOCAs, the licensee's analysis assumed that all three CCW check valves (CCW-13, -123, and -183) fail to close on demand. This resulted in a LERF increase from ISLOCAs to 7.8×10^{-08} per year, a change of 4.8×10^{-08} per year when compared to the base case. A copy of the interfacing system LOCA analysis was provided as Addendum 1 to the licensee's revised Relief Request [reference STPNOC letter dated February 23, 1999].

3.3.2.2 ISLOCA Through the LHSI Penetration

In Section 3.3.6 of the STP Individual Plant Examination (IPE) for Severe Accident Vulnerabilities (submitted to the NRC by letter dated August 28, 1992, and as supplemented by letter dated November 17, 1994), the following were ISLOCA scenarios identified for the safety injection systems:

> Each low head safety injection pump and the associated piping outside the reactor containment are isolated from the RCS hot leg by three normally closed check valves and one normally closed MOV, all in series. All four valves must fail in the open position to expose the LHSI pump and associated piping to the RCS pressure. Similarly, all three normally closed check valves that isolate the LHSI pump from the RCS cold leg via the RHR heat exchanger must fail open in order to expose the pump to normal RCS pressure. It is, however, expected that the RHR heat exchanger will fail first when exposed to normal RCS pressure.
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> Each HHSI [high head safety injection] and the associated piping outside the reactor containment are also isolated from the RCS cold leg and the RCS hot leg by three normally closed check valves and four normally closed valves (three check valves and one MOV), respectively. All valves in the piping to either the hot leg or cold leg would need to fail to expose the HHSI pump and associated piping to the RCS pressure. In addition, this piping is designed for high pressure.

One LHSI containment penetration check valve per pump train (XSI-0030A, XSI-0030B, or XSI-0030C) is part of this relief request, and one HHSI containment penetration check valve per pump train (XSI-0005A, XSI-0005B, or XSI-0005C) is part of this relief request. In quantifying the risk increase from changes to test intervals for these CIVs, the licensee assumed that this increase is bounded by failure of the valves in the LHSI line (valves XSI-0030A/B/C) and that the failure of the HHSI lines outside containment are less likely since these lines are designed for high pressure. This analysis, provided as Addendum 2 to the licensee's revised relief request [reference STPNOC letter dated February 23, 1999] shows that, assuming failure of a single LHSI check valve will result in an increase of LERF from 3.7×10^{-8} per year to 6.0×10^{-8} per year, a change of 2.3×10^{-8} per year. Assuming the failure of all three LHSI check valves will increase the LERF to 1.1×10^{-07} per year, an increase of 7×10^{-8} per year.

3.3.2.3 Overall Safety Significance of the Relief Request

Based on the bounding risk analysis provided by the licensee in the relief request which assumed failure to close (or re-close) of the CCW check valves and the LHSI check valves, the increase in LERF is estimated to be on the order of 1×10^{-7} per year. The increase in core damage frequency (CDF) from failure of these valves is negligible. These increases are consistent with the guidelines proposed in RG 1.174 and the staff finds these increases to be acceptable. In addition to the guidance from RG. 1.174, the staff bases this finding on reviews of the models, inputs and assumptions of the risk analysis found in the relief request, and in the PRA model as described in the licensee's IPE submittal. Staff review comments, requests for additional information (RAIs) on the IPE submittal, and the licensee responses to the RAIs were also considered. The staff finds that the ISLOCA analysis is of adequate quality to support the conclusions made in the licensee's relief request.

3.3.3 Performance Information to Justify the Test Interval Extension

The licensee indicated that the subject valves have exhibited a history of satisfactory operation. Specifically,

> A performance history of these valves indicates that there is no evidence that would indicate that these valves would fail to close on a cessation [or reversal of] flow. A probabilistic failure model was developed based on this maintenance history. The results of this analysis indicate that the mean time to failure for the Component Cooling Water check valves is 6.0 - 8.6 years. The mean time to failure for the Safety Injection check valves is greater than 20 years.

Table 1, "Reliability Report for Component Cooling Water Check Valves and Safety Injection Check Valves," of the Attachment to STPNOC's February 23, 1999, letter describes the "failure" events in more detail:

> The performance history of these valves indicates several cases of seat leakage exceeding the local leakage rate program limits [either guideline limits or administrative limits]; however, no cases were found indicating that the valves failed to seat promptly on cessation or reversal of flow. A search of the equipment history and corrective action program databases identified no events that could be attributed to failure of these check valves to maintain their ability to close.

The licensee stated that, based on the performance history, the current PRA modeling of the failure rates for the subject check valves is accurate and representative. The relief request also noted that, irrespective of the failure rate modeling, the calculated change in risk is small even when assuming complete failure of the CIVs.

3.3.4 Feedback and Corrective Actions

As stated by the licensee, the interval between leak-rate testing for each valve can be set to a maximum of 5 years based on completion of two as-found tests within the administrative limits set for the valve. The test interval will be set at the original value of 30 months if local leak test results exceed the administrative limit. Additionally, a cause determination will be performed and corrective action identified to reduce the leakage rate. The licensee's condition reporting process will be used to identify inspection and rework requirements for the valve. The licensee will evaluate other valves with similar designs based on the failure mechanism identified in the results of the condition investigation. The staff finds these feedback and corrective action activities to be consistent with the guidance provided in RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," and, therefore, acceptable.

4.0 CONCLUSION

The engineering analyses used by the licensee in support of this limited-scope relief request in RR-56 (RR-52) adequately assesses the safety and risk significance of the proposed alternative. Furthermore, sufficient component performance information is provided to justify the test interval extension. In addition, a process for feedback and corrective action is proposed so that, should unexpected age-related degradation of these containment isolation check valves occurs, the test intervals will be shortened. Therefore, the staff concludes that the licensee's proposed alternative will provide an acceptable level of quality and safety.

The licensee's request to extend the Code required closure verification test from once each refueling outage to once every 5 years for the subject CCW and SI system containment isolation check valves is authorized for the life of STP pursuant to 10 CFR 50.55a(a)(3)(i) based on the alternative providing an acceptable level of quality and safety.

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