Nuclear Operating Company

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

February 14, 2002
NOC-AE-02001242
10CFR50.90

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498 and STN 50-499
License Amendment Request -
Proposed Amendment to Technical Specification 3.4.2.2

Pursuant to 10CFR50.90, STP Nuclear Operating Company (STPNOC) hereby requests an amendment to Technical Specification (TS) 3.4.2.2, "Reactor Coolant System," to relax the lift setting tolerance of the pressurizer safety valves (PSVs) from ±2% to ±3%. The current TS requirement that the as left lift setting be within ±1% following valve testing will remain intact. STPNOC has determined that this proposed amendment to the operating licenses involves no significant hazards consideration.

Attachment 1 to this letter provides the No Significant Hazards Determination and Attachment 2 provides the TS page marked up with the proposed change. Attachment 3 provides the retyped TS page. There are no proposed changes to the Bases for TS 3.4.2.2, but they are provided in Attachment 4 for information.

The Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed the proposed change. STPNOC has notified the State of Texas in accordance with 10CFR50.91(b).

This proposed change is applicable to the Delta 94 replacement steam generators that have been installed in Unit 1 and will be installed in Unit 2 in late 2002. STPNOC requests approval of the proposed amendment by May 31, 2003. Once approved, the amendment shall be implemented within 30 days.

If there are any questions regarding this proposed amendment to TS 3.4.2.2, please contact Mr. Scott Head, Manager, Licensing at (361) 972-7136 or me at (361) 972-8757.

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

Executed on 2/14/02

J. J. Sheppard
Vice President,
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A001

STI: 31393035

Attachments:
1. Licensee's Evaluation
2. Proposed Technical Specification Change (Mark-up)
3. Proposed Technical Specification Page (Retyped)
4. Bases (For Information Only)

cc:

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Attachment 1

Licensee's Evaluation

LICENSEE'S EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses NPF-76 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed change would revise Technical Specification (TS) 3.4.2.2, "Reactor Coolant System," to relax the lift setting tolerance of the pressurizer safety valves (PSVs) from ±2% to ±3%. The current TS requirement that the as left lift setting shall be within ±1% following valve testing will remain intact.

This change is requested to reduce regulatory burden in complying with TS requirements that have little safety significance. During every refueling outage, the PSVs are removed from the plant and sent to a test facility. A Licensee Event Report (LER) is submitted to the NRC if the lift setting of more than one PSV is found to be outside the allowable tolerance. The test facility returns any out-of-tolerance PSV to within ± 1% of the required lift setting. Increasing the TS lift setting tolerance to the value assumed in the safety analyses would reduce the number of LERs prepared and reviewed. There would be no change in the margin to safety, and both licensee and NRC resources could be applied to issues of greater safety significance.

This proposed change is applicable to the Delta 94 replacement steam generators that have been installed in Unit 1 and will be installed in Unit 2 in late 2002. Approval of the proposed amendment is requested by May 31, 2003. Once approved, the amendment shall be implemented within 30 days.

2.0 PROPOSED CHANGE

Specifically, the proposed change would revise the following:

> 3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting [1] of 2485 psig ± 2%. [2]

to read

> 3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting [1] of 2485 psig ± 3%. [2]

3.0 BACKGROUND

In each STP unit, there are three PSVs set at 2485 psig (2500 psia) to provide overpressure protection for the reactor coolant system (RCS). The PSVs are Class 1 components, designed and manufactured to meet Section III of the ASME Code. They are of the totally enclosed pop type, spring-loaded, self-activated, with backpressure compensation.

The RCS design pressure is 2485 psig (2500 psia), which is also the setpoint for the PSVs. The PSVs prevent RCS pressure from exceeding 110% of system design pressure (approximately 2734 psig or 2750 psia) in compliance with the ASME Code, Section III.

South Texas Project owns nine PSVs; three are installed in each unit and the remaining three are replacement spares. During each refueling outage, all three valves installed in the unit are removed and replaced with the three spares, which have been previously tested and certified to lift at 2485 psig ± 1%. The three valves removed from the unit are sent offsite for testing and refurbishment in preparation for installation in the other unit during its next refueling outage. NWS Technologies in Spartanburg, South Carolina currently performs the PSV testing. STP Engineering monitors all testing and valve maintenance.

4.0 TECHNICAL ANALYSIS

The proposed change would revise TS 3.4.2.2 to relax the lift setting tolerance of the PSVs from ±2% to ±3%. The current TS requirement that the as left lift setting be within ±1% following valve testing will remain intact. Table 1 depicts the various RCS nominal pressure settings.

This relaxation is applicable for drift of PSV lift settings that occurs during the operating cycle and is consistent with ASME Section III, Subarticles NB-7512.1 and NB-7512.2.

The 1989 edition of the ASME Code Section III, Subarticle NB-7410/NC-7410 specifies:

> The set pressure of at least one of the pressure relief devices connected to the system shall not be greater than the design pressure of any component within the pressure retaining boundary of the protected system.

Additionally, the 1989 edition of the ASME Code, Section XI, requires that the PSVs be tested in accordance with ASME/ANSI OM-1987, Part 1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." This standard allows a testing lift pressure to vary from the stamped pressure by no more than ±3% before declaring a test failure. This standard also includes guidelines for testing additional valves when a valve exceeds the ±3% tolerance. Therefore, increasing the PSV setpoint tolerance to ±3% for the "as-found" test acceptance criterion is in compliance with the 1989 ASME Code, Section XI requirements.

The proposed change has been reviewed to determine the impact on the PSV inlet and discharge piping. The components reviewed meet the applicable requirements of ASME Section III,

Subarticle NB-3000. The calculated stress intensity levels do not exceed the service limits specified in the component specification for all system service loading.

Increasing the tolerance of the PSV lift setpoint from +2% to +3% (i.e., from 2550 to 2575 psia) would affect reactor coolant system (RCS) overpressure events. The maximum calculated RCS pressure was for the turbine trip event, which is the limiting overpressure event (Reference 1). The turbine trip analysis for the Delta 94 steam generators assumed a PSV lift setpoint of 2575 psia and an initial NSSS power level of 3897 MWt, which bounds the 1.4% power uprate currently proposed (Reference 2). The resulting maximum calculated pressure is 2743.5 psia, which is 5 psi less than the maximum allowable RCS pressure of 2748.5 psia. These results were found acceptable by the NRC staff (Reference 3).

Increasing the allowed tolerance from -2% to -3% (i.e., from 2450 to 2425 psia) would affect DNBR for events that lift the PSVs. A lower PSV lift setpoint will result in lower RCS pressure and a consequent lower DNBR.

Some overpower events may result in an RCS pressure increase that exceeds the capacity of the pressurizer spray and pressurizer PORVs, and could result in lifting the PSVs. The limiting overpower event is rod withdrawal at power, which is analyzed for several power levels to identify the limiting DNB case.

The DNB analysis assumes that pressurizer spray and the pressurizer PORVs operate as designed. This is a conservative assumption because it lowers RCS pressure and the DNB ratio. The DNB analysis was revised as part of the power uprate proposal (Reference 2), using the Delta 94 steam generator design and assuming that the PSVs lifted at 2425 psia. The analysis also used the Revised Thermal Design Procedure, which assumes nominal initial conditions and accounts for the uncertainties using an increased DNB ratio limit of 1.52 (minimum). The results of the analysis showed that the minimum DNB ratio stayed above 1.57.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

STPNOC has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92 as discussed below:

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

   Response: No

   The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR.

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

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

Response: No

The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR. Those analyses demonstrated that 1) the fuel design limits were maintained by the reactor protection system since the DNBR was maintained above the limit value, and 2) the plant design is such that a turbine trip presents no hazard to the integrity of the RCS or the main steam system pressure boundary. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10CFR50.92 and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for TS 3.4.2.2 is to ensure that overpressure protection is operable such that reactor coolant system pressure does not exceed 110% of the design pressure. Overpressure protection for the RCS is accomplished by the utilization of safety valves along with the reactor protection system and associated equipment. Combinations of these systems provide compliance with the overpressure requirements of the ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-7300. (UFSAR 5.2.2)

Piping, valves, and associated equipment used for overpressure protection are classified in accordance with ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." (UFSAR 5.2.2.6)

NUREG-0800, Section 5.2.2, "Overpressure Protection" states:

> The acceptance criteria for the overpressure protection system are based on meeting the relevant portions of the following regulations:
>
> > General Design Criterion 15 - "Reactor coolant system design."
> >
> > The reactor coolant system and associated auxiliary control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.
> >
> > General Design Criterion 31 – "Fracture prevention of reactor coolant pressure boundary."
> >
> > The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.
>
> Licensees must meet the recommendations of Task Action Plan item II.D.1 of NUREG-0737, which required that licensees and applicants conduct testing to qualify the RCS safety valves under expected operating conditions for design basis transients and accidents.
>
> Other specific acceptance criteria necessary to meet the requirements of GDC 15 and 31 are as follows:
>
> > Safety valves shall be designed with sufficient capacity to limit the pressure to less than 110% of the reactor coolant pressure boundary design pressure during the most severe abnormal operational transient with reactor scram.

Reference 4 submitted reports that show the valves, piping arrangements, and fluid inlet conditions for STP Units 1 and 2 are bounded by the values and test parameters of the EPRI Safety and Relief Test Program. The EPRI tests confirmed the ability of the safety valves to open and close under the expected operating fluid conditions. The NRC confirmed the conclusions drawn by STP (Reference 5).

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

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter, S.E. Thomas to NRC, "Replacement Steam Generator Reactor Coolant Flow Differences," May 20, 1999 (NOC-AE-000540)

2. Letter, J.J. Sheppard to NRC, "Technical Specifications Associated with a 1.4% Core Power Uprate," August 22, 2001 (NOC-AE-01001162)

3. Letter, T.W. Alexion to W.T. Cottle, "South Texas Project, Units 1 and 2 – Issuance of Amendments re: Replacement Steam Generator Reactor Coolant Flow Differences," November 15, 1999

4. Letter, HL&P to NRC, "Responses to DSER/FSAR Items Regarding Chapter 7A, Item II.D.1," October 31, 1985 (ST-HL-AE-1466)

5. NUREG-0781, Supplement 4, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2," Appendix W, July 1987

TABLE 1

REACTOR COOLANT SYSTEM NOMINAL PRESSURE SETTINGS

	psig
Hydrostatic Test Pressure	3,107
110% Design Pressure	2,734
Pressurizer Safety Valves Begin to Open (with + 3% tolerance)	**2,560**
Pressurizer Safety Valves Begin to Open (with + 2%tolerance)	2,535
Design Pressure and Pressurizer Safety Valve Set Pressure	2, 485
Pressurizer Safety Valves Begin to Open (with - 2% tolerance)	2,435
Pressurizer Safety Valves Begin to Open (with - 3% tolerance)	**2,410**
High Pressure Reactor Trip	2,380
Power Operated Relief Valves Begin to Open	2,335
High Pressure Alarm / Pressurizer Spray Valves Full Open	2,310
Pressurizer Spray Valves Begin to Open	2,260
Pressurizer Proportional Heaters Begin to Operate	2,250
Pressurizer Proportional Heaters Full Operation	2,220
Low Pressure Alarm / Pressurizer Backup Heaters Energize	2,210
Low Pressure Reactor Trip	1,870

Attachment 2

Proposed Technical Specification Changes

(Mark-up)

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting[1] of 2485 psig ~~±2%~~[2] **± 3%.**[2]

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

[1]The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

[2]The as left lift setting shall be within ±1% following valve testing.

Attachment 3

Proposed Technical Specification Page
(Retyped)

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting[1] of 2485 psig ± 3%. [2]

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

[1]The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

[2]The as left lift setting shall be within ±1% following valve testing.

Attachment 4

Bases

(For Information Only)

FOR INFORMATION ONLY

REACTOR COOLANT SYSTEM BASES

SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 504,950 lbs. per hour of saturated steam at the valve setpoint of 2500 psia.

During Modes 1, 2, and 3, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the turbine trip resulting from loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.