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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NOS. 97 AND 84 TO

# FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

# STP NUCLEAR OPERATING COMPANY

# DOCKET NOS. 50-498 AND 50-499

# SOUTH TEXAS PROJECT, UNITS 1 AND 2

## 1.0 INTRODUCTION

By application dated December 31, 1997, as supplemented by letters dated June 30, August 6, August 18, and August 27, 1998, STP Nuclear Operating Company, et.al., (STPNOC, the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License Nos. NPF-76 and NPF-80) for the South Texas Project, Units 1 and 2 (STP). The proposed changes would revise TS 2.1 (Safety Limits), 2.2 (Limiting Safety System Settings), and 3/4.2.5 (Departure from Nucleate Boiling Parameters) by including alternate operating criteria to allow continued plant operation with a reduced measured reactor coolant system flow rate, if necessary.

The June 30, August 6, August 18, and August 27, 1998, suppleme[illegible] provided clarifying information and did not change the initial no significant hazards consid[illegible]tion determination.

## 2.0 DISCUSSION

The following proposed TS changes are related to the reduction in RCS flow rate for alternate operation:

1. Addition of Figure 2.1-2, Reactor Core Safety Limit-Four Loops in Operation (Alternate) to account for reduction in TS RCS flow.

2. Change to Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints as follows because of reduction of TS RCS flow rate:

   a. For Functional Unit 12, Reactor Coolant Flow - Low trip setpoint, by noting that the applicable Reactor Coolant System loop design flow is 92,500 gpm;

   b. For Functional Unit 7, Overtemperature $\Delta T$ (OT$\Delta T$), the values in Note 1 are modified for $K_1$, T', and $f_1(\Delta I)$ offset and gain;

   c. [illegible]or Functional Unit 8, Overpower $\Delta T$ (OP$\Delta T$), the values in Note 3 are modified for $K_4$ and T";

3. The Limiting Condition for Operation 3.2.5.(a) is revised by adding 595 °F as an alternate lower Reactor Coolant system $T_{avg}$ limit; and

4. The Limiting Condition for Operation 3.2.5.(c) is revised by adding 380,500 gpm as an alternate lower Reactor Coolant System flow.

TS Bases Section 3/4.2.5 is also to be revised to reflect the alternate departure from nucleate boiling (DNB) criteria.

## 3.0 BACKGROUND

TS 3.2.5 for STP requires that RCS flow be maintained at or above the minimum measured flow of 392,300 gpm, which is equivalent to the thermal design flow of 381,600 gpm plus 2.8% to account for flow measurement uncertainties. Units 1 and 2 have the potential to fall below this minimum flow limit following a worst case scenario of 10% steam generator tube plugging (current licensing basis limit) combined with the effect of hot leg streaming on the precision calorimetric flow measurement. If this minimum measured flow cannot be met within two hours, the TS requires that the unit be in Mode 2 within the next four hours. In order to minimize the impact on continued plant operation for this scenario, STPNOC proposes to allow a 3% reduction in thermal design flow to 370,000 gpm, or a minimum measured flow of 380,500 gpm (with 2.8% added to account for flow measurement uncertainties), on an as needed basis, while maintaining a full power level of 3817 megawatts-thermal (MWt).

Power capability parameters for a Nuclear Steam Supply System (NSSS) power level of 3817 MWt, based on a 3% reduction in thermal design flow, were evaluated for four cases, using RCS $T_{avg}$ values of 583.2°F (proposed lower bound) and 590 °F (proposed upper bound), and steam generator tube plugging levels of zero and 10%. The increase of 0.9°F in the lower bound from 582.3°F to 583.2°F allows the minimum core inlet design temperature from the T-hot Reduction Program to be maintained at 549.0 °F. These parameters were used to evaluate the operation of Units 1 and 2 at 3817 MWt with the proposed 3% reduction in minimum measured RCS flow. All of the accident analyses previously evaluated in the Updated Final Safety Analysis Report (UFSAR) for STP were reevaluated to determine the effect of this 3% flow reduction.

## 4.0 REACTOR CORE SAFETY LIMITS, REACTOR TRIP SETPOINTS, AND NON-LOSS OF COOLANT ACCIDENT (LOCA) AND LOCA SAFETY ANALYSIS

### 4.1 Evaluation

Core safety limits and axial offset limits were recalculated based on alternative operation with a 3% reduction in minimum measured RCS flow. These were incorporated into an alternate TS Figure 2.1-2 for reactor core safety limits when operating under the reduced flow conditions. Available DNB margin and the reduction in the upper end of the nominal $T_{avg}$ range (593 °F to 590 °F) were used in the recalculation of the OTΔT and OPΔT reactor trip setpoints due to the revised safety limits, resulting in a reduction of approximately 1% in the margin to trip. However, the margin between the nominal and maximum values of the reference setpoints (i.e.,

$K_1$ and $K_4$ for the OTΔT and the OPΔT setpoints, respectively) were maintained. Therefore, the instrument accuracy uncertainties are unaffected by the change. The revised OTΔT and OPΔT setpoints were calculated using methods reviewed and approved by the NRC as documented in WCAP-8745-P-A.

The f(ΔI) function is used in the OTΔT trip equation and is currently defined for STP such that when the percent power difference between the top and bottom halves of the core is less than +8%, f(ΔI) is zero. If the percent power difference between the top and bottom halves of the core exceeds +8%, the trip setpoint is automatically reduced by 2.65% of its value at rated thermal power. In order to reflect the setpoint changes, the onset of the OTΔT setpoint reduction was lowered to an axial offset of greater than +6% for alternate operation with reduced RCS flow. Since the axial offset is not expected to be as high as +6% during normal full power operation, this reduction should not affect plant operability.

The UFSAR transient and accident analyses were evaluated by the licensee using the 3.0% reduction in flow for the events below.

| Event | UFSAR Section |
|---|---|
| Feedwater Malfunction at Hot Full Power Conditions | 15.1.1 - 15.1.2 |
| Excessive Load Increase | 15.1.3 |
| Loss of Load/Turbine Trip | 15.2.2 - 15.2.5 |
| Reactor Coolant System Depressurization | 15.6.1 |
| Rod Withdrawal at Power | 15.4.2 |
| Steamline Break | 15.1.5 |
| Loss of Flow | 15.3.1 - 15.3.2 |
| Locked Rotor | 15.3.3 - 15.3.4 |
| Rod Withdrawal from Subcritical | 15.4.1 |
| Dropped Rod | 15.4.3 |
| Steamline Depressurization | 15.1.4 |
| Feedwater Malfunction at Hot Zero Power Conditions | 15.1.2 |
| Loss of Normal Feedwater/Station Blackout | 15.2.6 - 15.2.7 |
| Feedline Break | 15.2.8 |
| Startup of an Inactive Reactor Coolant Loop | 15.4.4 |
| Chemical & Volume Control System Malfunction | 15.5.2 |
| Boron Dilution | 15.4.6 |
| Rod Ejection | 15.4.8 |
| Inadvertent Emergency Core Cooling System Actuation | 15.5.1 |
| Large Break LOCA | 15.6.5 |
| Small Break LOCA | 15.6.5 |
| Hot Leg Switchover | 6.3.2 |
| Post-LOCA Long-Term Core Cooling | Cycle specific calculation |
| LOCA Hydraulic Forces | Used as input into analyses contained in Chapter 3 |
| Rod Ejection Mass & Energy Releases | 15.4.8 |
| Steam Generator Tube Rupture | 15.6.3 |

The evaluation by the licensee of the events listed above was by recalculation, sensitivity studies or engineering judgement. For those events where the Departure from Nucleate Boiling Ratio (DNBR) or Pressure Limit was the limiting parameter, these requirements were found to be within the acceptable limits.

The limiting DNBR accident is the complete loss of forced reactor coolant flow from simultaneous loss of electrical power to all reactor coolant pumps. The licensee noted that the DNBR is a function of reactor core design, and varies somewhat with the specific parameters of each new core load. The required DNBR value for this application for 3% flow reduction is 1.26 for typical cells and 1.24 for thimble cells. This is based on the Revised Thermal Design Procedure (RTDP) and the WRB-1 DNB correlation; and for both 17x17XL STD and V5H fuel. The safety analysis limit DNBR values of 1.43 for typical cells and 1.38 for thimble cells are used, conservatively increasing the DNBR limit to provide for effects of rod bow and other DNB penalties that may occur. For the limiting accident, assuming four loops in operation, the DNBR is currently 1.514 for the full flow case. With 3% flow reduction, the DNBR is 1.497 and is acceptable. The THINC code is used to calculate DNBR for the transient, based on heat flux from the FACTRAN code and flow from the LOFTRAN code.

The limiting pressure event is the Loss of Load/Turbine Trip (Without Immediate Reactor Trip). In this transient there is a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System. The sensitivity study for Loss of Load/Turbine Trip event was performed with the LOFTRAN code which was approved by the NRC in WCAP-79007-P-A. The associated pressure for the limiting high pressure event is 2748 psia which is 110% of the RCS design pressure of 2485 psig. The sensitivity study for the reduced RCS flow calculated a peak pressure of 2747 psia which is 1 psi lower and therefore acceptable.

The current maximum peak clad temperature (PCT) for the large break LOCA is 2132 °F for Unit 1 and 2136 °F for Unit 2. These values are acceptable as they are below the 2,200 °F requirement in 10 CFR 50.46. The licensee stated (letter dated August 27, 1998) that a recalculation of the large break LOCA with updated information, including the effects of a 3% reduction in RCS flow rate indicated a PCT of 2090 °F for Unit 1 and 2. This confirmed that the predicted peak clad temperature remained bounded by the current maximum PCT for the lower loop temperatures and flow rate.

The current PCT temperature for the small break LOCA is 1849 °F for a 1.5 inch break for Unit 1 and Unit 2. This value is acceptable as it is below the acceptance limit of 2,200 °F. The licensee's evaluation confirmed that the predicted peak clad temperature remained bounded by the current PCT for the lower loop temperatures and flow rate.

4.2 <u>Summary</u>

The proposed alternative operating conditions for STP with a 3% RCS flow reduction were evaluated. Based on the staff's evaluation, full power operation with the proposed alternate TS for the reactor core safety limits with a reduction in the RCS $T_{avg}$ range and the OTΔT and

OPΔT reactor trip setpoints remains bounded by the current licensing basis analysis and the proposed changes are acceptable. In addition, non-LOCA and LOCA safety analyses and evaluations were performed to confirm the acceptability of a 3.0% reduction in thermal design flow. The evaluation by the licensee was by recalculation, sensitivity studies or engineering judgement. Approved codes were used in the recalculations. All the applicable acceptance criteria for each event were found to be met. Therefore, the inclusion of the provisional TSs described above to allow full power operation with minimum measured flow and thermal design flow reduced by up to 3% is acceptable.

## 5.0 STRUCTURAL INTEGRITY OF THE RCS AND RCS COMPONENTS

### 5.1 Evaluation

As part of the justification to support a 3% reduction in the measured RCS flow, the licensee performed an evaluation of the structural integrity for the RCS and components. The components evaluated include the reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), reactor coolant pumps, pressurizer, steam generators, auxiliary system equipment, and reactor coolant piping, components and their supports.

The licensee indicated that Westinghouse had previously performed bounding analyses for STP over the ranges of operation parameters provided in Table 1 of its June 30, 1998, letter. The bounding analyses consisted of a postulated limiting case associated with a 10°F reduction in vessel hot leg temperature. The postulated T-hot reduction resulted in higher stresses and cumulative fatigue usage factors (CUFs) than the original licensing basis values for certain components. The affected components are the reactor vessel (CRDM housing and rotolock inserts), pressurizer (spray nozzle) and steam generators (auxiliary feedwater nozzle). Other components evaluated were not affected by the hot leg temperature reduction. In its August 18, 1998, letter, the licensee provided a summary table of maximum stresses and CUFs for the limiting RCS components resulting from the current licensing basis bounding analyses. The stress ratios and CUF values provided in the table were found to be within the code allowable limits. The bounding analyses include STP plant operation at an RCS pressure of 2250 psia, a range of steam pressure from 975 psia to 1100 psia, a range of hot leg temperature from 615.6° F to 625.6° F, and a range of RCS average temperature from 549.0° F to 560.4°F, as documented in the STP UFSAR.

In its December 31, 1997, application, the licensee evaluated the effects of a 3% reduction in the RCS flow rate on the structural integrity of the reactor coolant systems and components, by comparing operational conditions used in the existing analyses against the plant new operating conditions at a proposed thermal design flow of 380,500 gpm. The input parameters for the proposed new operating condition are provided in Table 1 of the December 31, 1997, application. The licensee indicated that STP will operate with the proposed 3% RCS flow reduction, at an RCS pressure of 2250 psia, within a range of steam pressure from 979 psia to 1067 psia, a range of hot leg temperature from 617.4°F to 623.8°F, and a range of RCS average temperature from 549.0°F to 556.2°F.

The staff finds that the operational conditions used in the previous evaluation are bounding for the proposed operation under the new conditions with a 3% reduction in RCS flow rates, and