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

January 27, 1999

Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: UPDATE TO TECHNICAL SPECIFICATION BASES PAGES B 2-1 AND B 3/4 2-4, SOUTH TEXAS PROJECT, UNITS 1 AND 2 (STP) (TACS NOS. MA4402 AND MA4403)

Dear Mr. Cottle:

The purpose of the letter is to respond to STP Nuclear Operating Company's December 10, 1998, letter which provided an update to Technical Specification (TS) Bases Pages B 2-1 and B 3/4 2-4. The change adds a reference to TS Figure 2.1-2 to Bases Section 2.1.1 because this figure provides the departure from nucleate boiling heat flux ratio (DNBR) curves appropriate to the alternate departure from nucleate boiling (DNB) operating criteria. Also, a change was made to Bases Section 3/4.2.2 and 3/4.2.3 by referencing TS 3.2.5 rather than an explicit reactor coolant system (RCS) flow rate since the minimum flow rate required will be different if the alternate flow rate option is implemented. The letter did not request any approval or response from the Nuclear Regulatory Commission (NRC). However, this letter is provided to confirm that NRC agrees with the proposed change.

The NRC staff agrees with both of the proposed changes because the changes present a more complete discussion of the safety limits for the reactor core, by referencing both the primary and alternate "Reactor Core Safety Limit - Four Loops in Operation" curves and referencing TS 3.2.5 in place of a specific RCS flow rate. TS 3.2.5 provides for a reduced RCS flow rate with a reduced RCS $T_{avg}$. Referencing this Specification more accurately reflects the limits of the RCS flow rate parameter.

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Enclosed for your information, are revised TS Bases pages B 2-1 and B 3/4 2-4 that the NRC staff will use to update NRC's copy of the Bases. The revised pages contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness. If you have any additional questions regarding this issue, please contact me at (301) 415-1326.

Sincerely,

ORIGINAL SIGNED BY:

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure: Bases Pages B 2-1 and B 3/4 2-4

cc w/encls: See next page

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Thomas W. Alexion

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Division of Reactor Projects III/IV
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Enclosure: Bases Pages B 2-1 and B 3/4 2-4

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## 2.1 SAFETY LIMITS

## BASES

### 2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the WRB-1 correlation, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

The curves of Figure 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, and a reference cosine with a peak of 1.61 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} [1 + PF_{\Delta H} (1-P)]$$

where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specifc in the CORE OPERATING LIMITS REPORT (COLR);

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR; and,

P is the fraction of RTP.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming axial imbalance is within the limits of the $f_1$ (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

BASES

## 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

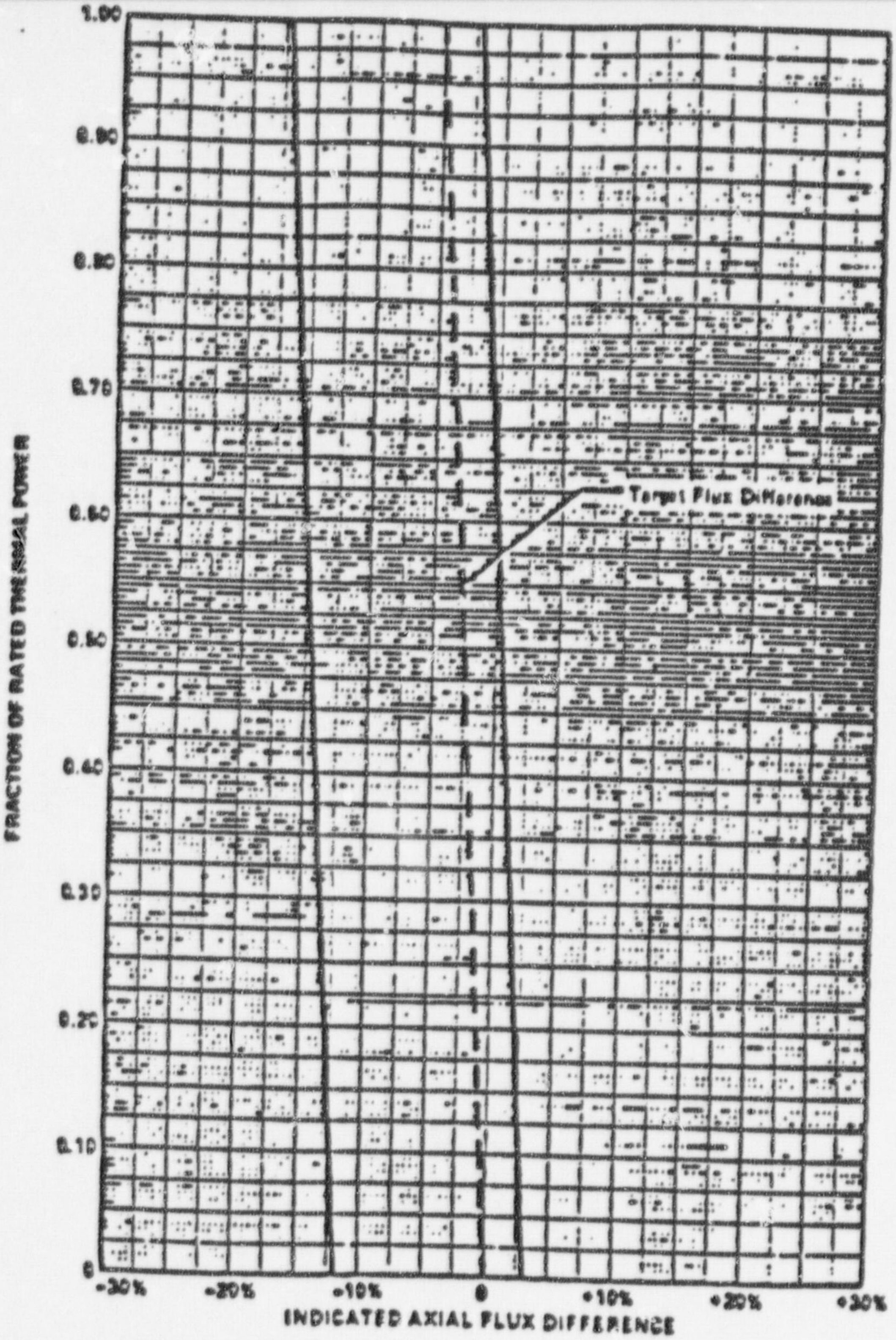


FIGURE B 3/4.2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement (TS 3.2.5) and the requirement on $F^N_{\Delta H}$ guarantees that the DNBR used in the safety analysis will be met. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limit. A measurement error of 4% for $F^N_{\Delta H}$ has been allowed for in the determination of the design DNBR value.

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses and the design limit to offset the rod bow penalty and other penalties which may apply.