ATTACHMENT 1

AFFECTED TS PAGES

where:

$\Delta T_o$ = Indicated $\Delta T$ at rated thermal power, °F;
T = Average temperature, °F;
P = Pressurizer pressure, psig;
$K_1$ < 1.1365;
$K_2$ = 0.01228;
$K_3$ = 0.00089;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for $T_{avg}$ dynamic compensation;

$\tau_1$ & $\tau_2$ = Time constants utilized in the lead-lag controller for $T_{avg}$, $\tau_1$ = 20 seconds, $\tau_2$ = 3 seconds;

T' = 575.4°F Reference $T_{avg}$ at rated thermal power;

P' = 2235 psig (Nominal RCS Operating Pressure);

S = Laplace transform operator, $sec^{-1}$;

and f($\Delta$I) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

1) For ($q_t - q_b$) within +12% and -17%, where $q_t$ and $q_b$ are percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power (2300 Mwt), f($\Delta$I) = 0. For every 2.4% below rated power (2300 Mwt) level, permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent.

2) For each percent that the magnitude of ($q_t - q_b$) exceeds +12% in a positive direction, the $\Delta$T trip setpoint shall be automatically reduced by 2.4% of the value of $\Delta$T at rated power (2300 Mwt).

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,(2) is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to Specification 2.3.1.2.d.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed in Section 7.2.2 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.(2)

The setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figure 2.1-1.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.(4) The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 $ft^3$ of water corresponds to 92% of span. The specified setpoint allows margin for instrument error(2) and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[5]

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 45% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.17 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature Delta-T trip setpoint is adjusted to the value specified for three loop operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification, as these settings are not used in the transient and accident analysis. (FSAR Chapter 15)

References

(1) FSAR Section 15.4
(2) FSAR Section 15.0
(3) FSAR Section 15.6
(5) FSAR Section 15.3
(6) FSAR Section 15.2

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1a or 3.1-1b (as appropriate per 3.1.2.1).

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant

## ATTACHMENT 2: RTD RESPONSE TIME AND OPERATIONS DURING CYCLE 13

ANF 88-094 has two purposes for operation during Cycle 13. The analysis provides the basis for changing the Technical Specification and verifies the adequacy of the response time characteristics of the proposed thermowell mounted resistance temperature detectors. Basic assumptions used in the analysis include

(a) precluding the withdrawal function of automatic rod control,

(b) a non-positive moderator temperature coefficient above 50 percent power for the rod drop transient, and

(c) a K1 value in the OTΔT trip function of 1.24 including uncertainties.

Automatic Rod Control is an important factor because of its role in the rod drop event. The limiting case is inadvertent release of a Rod Cluster Control Assembly (RCCA) of low worth at a core location close to the N44 ex-core flux detector. As described in the discussion of this event in UFSAR Section 15.4.3, ("rod on bottom" or 5 percent decrease in core power in 2 seconds as) dropped rod indications actuate turbine runback from 100 percent to 70 percent load. Consequently, the associated RCS average temperature reference signal (i.e., T-ref) used as one of the inputs to the Automatic Rod Control system is also "runback" and 70 percent becomes the new target core power level. However, because automatic rod control depends exclusively on N44 for indication of core power and because "shadowing" of this one ex-core detector can result in an artifically low indication of average core power, its possible for Automatic Rod Control to erroneously insert positive reactivity instead of decreasing core power as intended. Although rod drop indications also actuate a block of control rod withdrawal, malfunction of the blocking action is postulated in the analysis as the worst single failure. The mismatch of power production and heat removal results in heatup of the RCS until OTΔT trip.

As an alternative to demonstrating acceptable consequences of automatic rod control's contribution to the severity of the rod drop event, operation during Cycle 13 will be based on preventing the potentially detrimental aspects of its operation. A plant modification will preclude action of the automatic rod control system to withdraw RCCA's. In other words, the limiting scenario will be prevented by making the "rod block" function invulnerable and continuous.

In a similar fashion, restricting the moderator temperature coefficient allowed in Technical Specification Section 3.1.3.1 to non-positive values above 50 percent power facilitates the accident analysis by insuring the heatup of the RCS will not (in itself) insert positive reactivity at these power levels. For heatup events this avoids a power "overshoot" prior to control rod insertion.

License Event Report 88-002-01 committed to a Technical Specification change to the K1 OTΔT setpoint as a result of corrected reanalysis of the rod drop event prior to startup of Cycle 13. ANF-88-094 is this corrected reanalysis and demonstrates that an analysis value of K1 = 1.24 (previous analysis used 1.26) is adequate to conservatively prevent DNB. Retaining the same uncertainty allowance in K1 gives a value of 1.1365 (previous value 1.1565) for Technical Specification Section 2.3.1.2. Because ANF-88-094 is intended to support operation during Cycle 13 and because elimination of the RTD bypass piping is planned for the upcoming outage, the analysis also reflects a RTD response time of a 4 second lag in combination with a 0.75 seconds delay (Reference Table 2.1-1 of WCAP-11889). Section 4.2 of WCAP-11889 verifies that the current allowance in K1 is sufficient to account for known instrument uncertainties.