ATTACHMENT 1

REVISED TS PAGES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists.

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.

d. The safety limit is exceeded if the combination of Reactor Vessel inlet temperature and thermal power level is at any time above the appropriate pressure line in Figure 2.1-1 or if the thermal power level, coolant pressure, or Reactor Vessel inlet temperature violates the limits specified above.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure, have been related to DNB through the XNB DNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during normal operational transients and anticipated transients is limited to 1.17. A DNB ratio of 1.17 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.(1) The DNB ratio limit of 1.17 is a conservative design limit which is used as a basis for setting core safety limits. Based on rod bundle DNB tests, no fuel rod damage is expected at this DNB ratio or greater.

The curves of Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent the loci of points of thermal power, reactor vessel inlet temperature, and coolant system pressure for which the DNB ratio is not less than 1.17. The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.17 but

are set to preclude bulk boiling at the vessel exit. An arbitrary upper safety limit of 118% thermal power is shown. This limit is based on the high flux trip including all uncertainties.

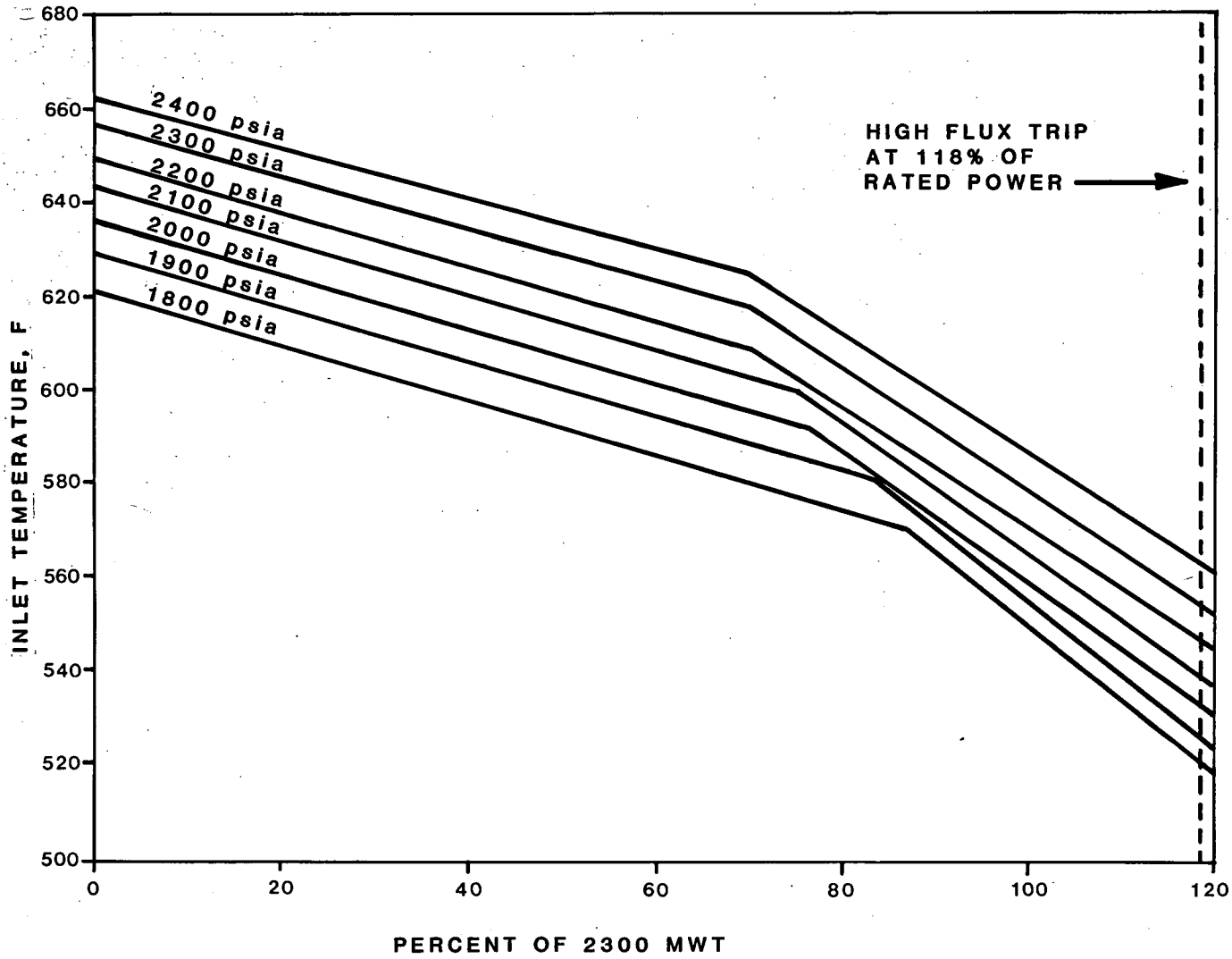
Radial power peaking factors consistent with the limit on $F_{\Delta H}$ given in Specification 3.10.2.1 have been employed in the generation of the curves in Figure 2.1-1. An additional heat flux factor of 1.03 has been included to account for fuel manufacturing tolerances and in-reactor densification of the fuel.

The safety limit curves given in Figure 2.1-1 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.17[2] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

References

(1) XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."
(2) FSAR Section 15.



CORE PROTECTION BOUNDARIES FOR 3-LOOP OPERATION

Figure 2.1-1

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17% in the negative direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

e. Overpower Δ T

$$\leq \Delta T_o \{ K_4 - K_5 \left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 (T - T') - f(\Delta I) \}$$

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;

T = Average temperature, °F;

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

K_4 < 1.07;

K_5 = 0.0 for decreasing average temperature, 0.02 sec/°F for increasing average temperature;

K_6 = 0.00277 for T > T' and 0 for T ≤ T';

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

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, τ_3 = 10 seconds;

f(ΔI) = As defined in d. above

f. Low reactor coolant loop flow ≥ 90% of normal indicated flow.

g. Low reactor coolant pump frequency ≥ 57.5 Hz.

h. Undervoltage ≥ 70% of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level ≤ 92% of span.

b. Low-low steam generator water level ≥ 14% of narrow range instrument span.

3.0 LIMITING CONDITIONS FOR OPERATION

Except as otherwise provided for in each specification, if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in hot shutdown within eight hours and in COLD SHUTDOWN within the next 30 hours unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. With reactor power less than 2% of rated thermal power and less than two reactor coolant pumps in operation, one of the following actions shall be taken:

1. maintain a shutdown margin of at least 4% $\Delta k/k$, or
2. open the lift disconnect switches for all control rods not fully withdrawn, or
3. open reactor trip breakers.

b. Power operation with less than three loops in service is prohibited.

c. At least one reactor coolant pump or residual heat removal pump shall be in operation when $T_{avg} > 200°F$ and reactor power is less than 2% of rated thermal power. In the event this condition cannot be satisfied, the following actions shall be taken:

   1. Proceed to establish a boron concentration in the reactor coolant equal to or greater than that concentration needed to maintain a shutdown margin of 1% $\Delta k/k$ at 200°F, and

   2. Restore at least one reactor coolant pump or residual heat removal pump to operation within one hour, or prepare and submit a Special Report to the NRC within 30 days.

d. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

Basis

Specification 3.1.1.1.a contains requirements designed to limit the consequences of the uncontrolled bank withdrawal at low or subcritical power conditions as analyzed in the safety analysis. The requirement of two reactor coolant pumps in operation below 2% power is consistent with the assumptions utilized in the bounding transient that was analyzed. The specification makes allowance for less than two pumps in operation by specifying either of three actions that must be taken. Either maintaining the specified shutdown margin, opening the lift disconnect switches on the control rods or opening the reactor trip breakers will prevent the occurrence of the postulated uncontrolled bank withdrawal transient, therefore allowing the two pump requirement to be lifted.

Maintaining a shutdown margin of 4% $\Delta k/k$ is sufficient to prevent a return to criticality if the worth of the two most reactive control rod banks are simultaneously withdrawn as is the assumption of the postulated transient.

5.3 REACTOR

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 68 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods which are all pre-pressurized. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rod locations occupied by rods consisting of natural or slightly enriched uranium pellets, solid inert materials, or a combination of the aforementioned.(1)

5.3.1.2 Deleted.

5.3.1.3 Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.5 weight percent of U-235.

5.3.1.4 Deleted.

5.3.1.5 There are 45 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain 144 inch segments of silver-indium-cadmium alloy clad with the stainless steel.(2)

5.3.1.6 Up to 10 grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the Reactor Coolant System complies with the Code requirements.(3)

ATTACHMENT 2

REPLACEMENT TS PAGES

REPLACMENT INSTRUCTIONS

Delete	Insert
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-4	2.1-4
2.1-5	-
2.3-3	2.3-3
3.1-1	3.1-1
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5.3-1	5.3-1

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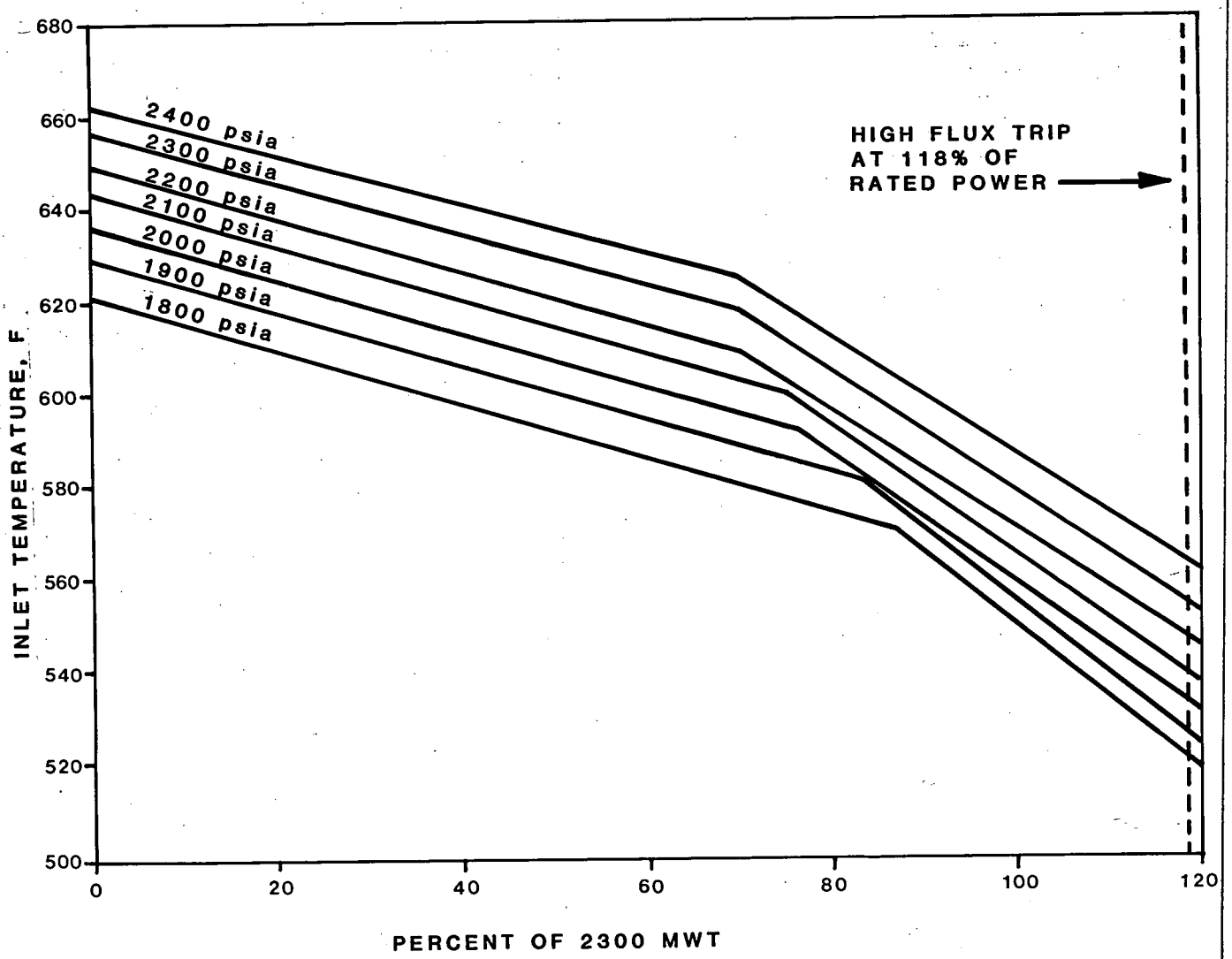

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