Bases - Unit 3

The safety limits presented for Oconee Unit 3 have been generated using BAW-2 critical heat flux correlation[1] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (131.32 x 10^6 lbs/hr for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate.[2]

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation[1]. The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1C represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 139.86x 10^6 lbs/hr.). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects: $F_q^N = 2.565$; $F_{\Delta H}^N = 1.71$[3]; $F_z^N = 1.50$. The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

The curves of Figure 2.1-2C are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing.

1. The 1.30 DNBR limit produced by a nuclear peaking factor of $F_q^N = 2.565$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 3.

Power peaking is not a directly observable quantity, and, therefore, limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2 and 3 of Figure 2.1-2C correspond to the expected minimum flow rates with four pumps, three pumps and one pump in each loop, respectively.

The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055= 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3C a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1C is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3C.

References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August, 1977.

(3) Amendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

level trip and associated reactor power/reactor power-imbalance boundaries by 1.055% for 1% flow reduction.

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation. The reactor trip upon loss of one pump during 4-pump operation above 80% FP is specified for Unit 1 in order to provide a minimum of 11.2% DNBR margin in the flux/flow trip setpoint to accommodate the possible reduction in thermal margin due to rod bowing. For unit 2, loss of one pump trip is not required because of thermal credits from excess RC flow, i.e., by maintaining a minimum RC flow of 109.5%. For unit 3, the required DNBR margin for rod bowing is included in the analysis of the flux/flow trip setpoint.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure (11.14 T_{out}-4706) trip
(1800) psig (11.14 T_{out}-4706)
(1800) psig (11.14 T_{out}-4706)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of (11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.