RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figures 3.2-4 and 3.2-5 for 4 and 3 loop operation, respectively.

For: Westinghouse Fuel , for: Exxon Nuclear Company Fuel

$$R = \frac{F^N_{\Delta H}}{1.48\,[1.0 + 0.2\,(1.0-P)]}\ , \quad R = \frac{F^N_{\Delta H}}{1.49\,[1.0 + 0.2\,(1.0 - P)]}$$

where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and $F^N_{\Delta H}$ = measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ and flow, without additional uncertainty allowance, shall be used.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-4 or 3.2-5 (as applicable):

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER Limit required by ACTION items a.2 and/or b above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation as shown on Figure 3.2-4 or 3.2-5 (as applicable) for RCS flow rate and R prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining $\geq$ 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-4 or 3.2-5 (as applicable):

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.4 The RCS total flow rate shall be determined by measurement at least once per 18 months.

The curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$F^N_{\Delta H}$ = 1.48 [1 + 0.2 (1-P)] (Westinghouse Fuel)

$F^N_{\Delta H}$ = 1.49 [1 + 0.2 (1-P)] (Exxon Nuclear Company Fuel)

where P is the fraction of RATED THERMAL POWER.

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 the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3411 MWt. The limit on $F_Q(Z)$ is 2.04. The limit on $F^N_{\Delta H}$ is 1.49. The analyses supporting the Exxon Nuclear Company limits are valid for an average steam generator tube plugging of up to 5% and a maximum plugging of one or more steam generators of up to 10%. In establishing the limits, a plant system description with improved accuracy was employed during the reflood portion of the LOCA Transient. With respect to the Westinghouse supplied fuel the minimum projected excess margin of at least 10% to ECCS limits will more than offset the impact of increase steam generator tube plugging.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is 1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.04 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200^{o}F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

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

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. As noted on Figures 3.2-4 and 3.2-5, RCS flow rate and $F^{N}_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^{N}_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. 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. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of the basis.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When RCS flow rate and $F^{N}_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 3.5% for RCS flow total flow rate and 4% for $F^{N}_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

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