TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

| | |
|---|---|
| Guaranteed Nuclear Steam Supply System thermal power output | 3471 MWt |
| The Engineered Safety Features design rating (maximum calculated turbine rating) | 3577 MWt |
| Thermal power generated by the reactor coolant pumps (nominal) | 12 MWt |
| Guaranteed Core Thermal Power | 3459 MWt |

TABLE 15.1-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

| Faults | Computer Codes Utilized | Reactivity Coefficients Assumed: Moderator Temperature(1) (Δk/°F) | Moderator Density(1) (Δk/gm/cc) | Doppler(2) | Initial NSSS Thermal Power Output Assumed (MWt) |
|---|---|---|---|---|---|
| CONDITION II | | | | | |
| Uncontrolled RCC Assembly Bank Withdrawal from a Subcritical Condition | TWINKLE, FACTRAN THINC | $+5 \times 10^{-5}$ | --- | Lower | 0 |
| Uncontrolled RCC Assembly Bank Withdrawal at Power | LOFTRAN | --- | 0 and 0.52 | Lower and upper | 3431 (3) 2058.6 and 343.1 |
| RCC Assembly Misalignment | THINC, ANC, LOFTRAN | --- | 0 | Upper | 3411 (4) |
| Uncontrolled Boron Dilution | NA | NA | NA | NA | 0 and 3423 |
| Partial Loss of Forced Reactor Coolant Flow | LOFTRAN THINC, FACTRAN | --- | 0 | Upper | 3431 |
| Start-up of an Inactive Reactor Coolant Loop | --- | --- | --- | --- | --- |
| Loss of External Electrical Load and/or Turbine Trip | LOFTRAN | --- | 0 and 0.52 | Lower and Upper | 3423 (DNB Cases) 3491.5 (Pressure Cases) (5) |
| Loss of Normal Feedwater | LOFTRAN | --- | NA | NA | 3491.5 (5) |

| Faults | Computer Codes Utilized | Reactivity Coefficients Assumed Moderator Temperature(1) ($\Delta k/°F$) | Moderator Density(1) ($\Delta k/gm/cc$) | Doppler(2) | Initial NSSS Thermal Power Output Assumed (MWt) |
|---|---|---|---|---|---|
| CONDITION II (cont'd) | | | | | |
| Loss of Offsite Power to the Plant Auxiliaries | LOFTRAN | --- | NA | NA | 3491.5 [(5)] |
| Excessive Heat Removal Due to Feedwater System Malfunctions | LOFTRAN | --- | 0.52 | Lower | 0 and 3411 [(6)] |
| Excessive Load Increase | LOFTRAN | --- | 0 and 0.52 | Lower | 3411 [(6)] |
| Accidental Depressurization of the Reactor Coolant System | LOFTRAN | --- | 0 | Upper | 3411 [(6)] |
| Accidental Depressurization of the Main Steam System | LOFTRAN | --- | Function of Moderator Density (See Sec 15.2.13) (Fig 15.2-41) | Fig. 15.4-49 | 0 (Subcritical) |
| Inadvertent Operation of ECCS During Power Operation | LOFTRAN | --- | 0 | Lower | 3491.5 [(5)] |

Reactivity Coefficients

| Faults | Computer Codes Utilized | Assumed Moderator Temperature(1) (Δk/°F) | Moderator Density(1) (Δk/gm/cc) | Doppler(2) | Initial NSSS Thermal Power Output Assumed (MWt) |
|---|---|---|---|---|---|
| CONDITION III | | | | | |
| Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipe which Actuate Emergency Core Cooling | NOTRUMP, SBLOCTA | | | | 3479 |
| Inadvertent Loading of a Fuel Assembly into an Improper Position | PHOENIX-P, ANC | --- | NA | NA | 3216-3563 (4) |
| Complete Loss of Forced Reactor Coolant Flow | LOFTRAN THINC, FACTRAN | --- | 0 | Upper | 3431 |
| Waste Gas Decay Tank Rupture | NA | --- | NA | NA | 3577 |
| Single RCC Assembly Withdrawal at Full Power | ANC, THINC PHOENIX-P | --- | NA | NA | 3423 |
| CONDITION IV | | | | | |
| Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the Reactor Coolant System (Loss of Coolant Accident) | SATAN BASH COCO LOCBART | Function of Moderator Density (See Section 15.4.1) | | Function of Fuel Temp. (See Section 15.4.1) | 3579 |

| Faults | Computer Codes Utilized | Reactivity Coefficients Assumed: Moderator Temperature(1) (Δk/°F) | Reactivity Coefficients Assumed: Moderator Density(1) (Δk/gm/cc) | Doppler(2) | Initial NSSS Thermal Power Output Assumed (MWt) |
|---|---|---|---|---|---|
| CONDITION IV (cont) | | | | | |
| Major Secondary System Pipe Rupture, up to and Including Double-Ended Rupture (Rupture of a Steam Pipe) | LOFTRAN, THINC | Function of Moderator Density (See Section 15.2.13) (Fig. 15.4-50 Unit 1) (Fig. 15.4-48 Unit 2) | | Fig. 15.4-49 | 0 (Subcritical) |
| Steam Generator Tube Rupture | NA | NA | NA | NA | 3577 |
| Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break | LOFTRAN THINC, FACTRAN | --- | 0 | Upper | 3431 |
| Fuel Handling Accident | NA | NA | NA | | 3600 |
| Rupture of a Control Rod Mechanism Housing (RCCA Ejection) | TWINKLE, FACTRAN PHOENIX-P | -0 pcm/°F BOL -26 pcm/°F EOL | --- | Consistent with lower limit shown on Fig 15.1-5 | 0 and 3479 (7) |

NOTES:

(1) Only one is used in an analysis, i.e., either moderator temperature or moderator density coefficient.
(2) Reference Figure 15.1-5 for Doppler power coefficients.
See UFSAR Section 4.5 for the applicable station reload analysis.
(3) Cases are considered at 3 different initial power levels – 100%, 60%, and 10%.
(4) Core power is assumed in the analysis.
(5) Analysis is performed at 102% of an NSSS power of 3423 MWt which is equivalent to 100.6% of 3471 MWt.
(6) No pump heat is assumed in the analysis.
(7) Analysis is performed at 102% of a core power of 3411 MWt which is equivalent to 100.6% of 3459 MWt.

TABLE 15.1-3

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSIS

| Trip Function | Limiting Trip Point Assumed In Analyses | Time Delay (sec) |
|---|---|---|
| Power range high neutron flux, high setting | 118 percent | 0.5 |
| Power range high neutron flux, low setting | 35 percent | 0.5 |
| Overtemperature ΔT | Variable, see Figure 15.1-1 | 7.0(1) (Ref.21)(2) |
| Overpower ΔT | Variable, see Figure 15.1-1 | 7.0(1) (Ref.21) |
| High pressurizer pressure | 2425 psig | 2.0 |
| Low pressurizer pressure | 1825 psig | 2.0 |
| Low reactor coolant flow (from loop flow detectors) | 87-percent loop flow | 1.0 |
| Undervoltage trip | 68 percent nominal | 1.5 |
| Turbine trip | Not Applicable | 1.0 |
| Low-low steam generator level | 0 percent of Narrow Range Level Span | 2.0 |
| High steam generator level trip of feedwater pumps and closure of feedwater system valves, and turbine trip | 73 percent of Narrow Range Level Span | 2.0 |
| Underfrequency trip | 53.9 Hz | 0.6 |
| Loss of offsite power time delay | Not Applicable | 1.5 (3) |

NOTES:

(1) Total time delay (including RTD response time and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint at the channel sensor until the rods begin to drop.

(2) See Reference 21, Section 15.1.10.

(3) From rod motion

TABLE 15.1-4

DETERMINATION OF MAXIMUM OVERPOWER TRIP POINT - POWER RANGE NEUTRON FLUX CHANNEL - BASED ON NOMINAL SETPOINT CONSIDERING INHERENT INSTRUMENTATION ERRORS

Nominal Setpoint (percent of rated power) 109

Calorimetric Errors in the Measurement of Secondary System Thermal Power:

| Variable | Accuracy of Measurement of Variable (Percent Error) | Estimated Effect on Thermal Power Determination (Percent Error) |
|---|---|---|
| Feedwater temperature | ±0.5 | |
| Feedwater pressure (small correction on enthalpy) | ±0.5 | 0.3 |
| Steam pressure (small correction on enthalpy) | ±2 | |
| Feedwater flow | ±1.25 | 1.25 |

Assumed calorimetric error (percent of rated power) 2

Axial power distribution effects on total ion chamber current

Estimated error (percent of rated power) 3
Assumed error (percent of rated power) 5

Instrumentation channel drift and setpoint reproducibility

Estimated error (percent of rated power) 1
Assumed error (percent of rated power) 2

Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction (percent of rated power) 118

TABLE 15.1-5

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TABLE 15.1-6

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