## 1.1 Definitions

| | |
|---|---|
| MODE (continued) | vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. |
| OPERABLE-OPERABILITY | A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). |
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:<br>a. Described in FSAR Chapter 13, Initial Tests and Operations;<br>b. Authorized under the provisions of 10 CFR 50.59; or<br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| QUADRANT POWER TILT RATIO (QPTR) | QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. |
| RATED THERMAL POWER (RTP) | RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3216 MWt. |

(continued)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Vessel inlet temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlations.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

### 2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, 5, and in MODE 6 when the reactor vessel head is on, the RCS pressure shall be maintained $\leq$ 2735 psig.

---

## 2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6, restore compliance within 5 minutes.

Table 3.3.1-1 (page 1 of 8)
Reactor Protection System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 1,2 | 2 | B | SR 3.3.1.14 | NA |
| | 3[a], 4[a], 5[a] | 2 | C | SR 3.3.1.14 | NA |
| 2. Power Range Neutron Flux | | | | | |
| a. High | 1,2 | 4[j] | D | SR 3.3.1.1<br>SR 3.3.1.2<br>SR 3.3.1.7<br>SR 3.3.1.11 | ≤111% RTP |
| b. Low | 1[b],2 | 4[j] | E | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11 | ≤25% RTP |
| 3. Intermediate Range Neutron Flux | 1[b], 2[c] | 1 | F | SR 3.3.1.1<br>SR 3.3.1.8<br>SR 3.3.1.11 | NA |

(continued)

(a) With Rod Control System capable of rod withdrawal and one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(j) Only 3 channels required during Mode 2 Physics Tests, LCO 3.1.8

Table 3.3.1-1 (page 3 of 8)
Reactor Protection System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 7. Pressurizer Pressure | | | | | |
| a. Low | $1^{(e)}$ | 4 | H | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≥1900 psig |
| b. High | 1,2 | 3 | E | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≤2400 psig |
| 8. Pressurizer Water Level - High | $1^{(e)}$ | 3 | H | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≤97% |
| 9. Reactor Coolant Flow - Low | $1^{(e)}$ | 3 per loop | H | SR 3.3.1.1<br>SR 3.3.1.7<br>SR 3.3.1.10 | ≥90% |

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 7 of 8)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 2.8% of ΔT span :

$$\Delta T \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[ T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.
$\Delta T_0$ is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, $sec^{-1}$.
T is the measured RCS average temperature, °F.
T' is the nominal $T_{avg}$ at RTP, ≤ [ * ]°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≤ [ * ] psig

| | | |
|---|---|---|
| $K_1$ ≤ [ * ] | $K_2$ ≥ [ * ]/°F | $K_3$ ≥ [ * ]/psig |
| $\tau_1$ ≥ [ * ] sec | $\tau_2$ ≤ [ * ] sec | |

| | | |
|---|---|---|
| $f_1(\Delta I)$ = | [ * ] {[ * ] + ($q_t$ - $q_b$)} | when $q_t$ - $q_b$ ≤ - [ * ]% RTP |
| | 0% of RTP | when -[ * ]% RTP < $q_t$ - $q_b$ ≤ [ * ]% RTP |
| | -[ * ] {($q_t$ - $q_b$) - [ * ]} | when $q_t$ - $q_b$ > [ * ]% RTP |

Where $q_t$ and $q_b$ are percent RTP in the upper and lower halves of the core, respectively, and $q_t$ + $q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with [ * ] are specified in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Protection System Instrumentation

Note 2: Overpower $\Delta T$

The Overpower $\Delta T$ Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.8% of $\Delta T$ span:

$$\Delta T \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{(1+\tau_3 s)} T - K_6 \left(T - T''\right) - f_2(\Delta I) \right\}$$

Where: $\Delta T$ is measured RCS $\Delta T$, °F.
$\Delta T_0$ is the indicated $\Delta T$ at RTP, °F.
s is the Laplace transform operator, $sec^{-1}$.
T is the measured RCS average temperature, °F.
T″ is the nominal $T_{avg}$ at RTP, ≤ [ * ]°F.

$K_4 \leq$ [ * ]

$K_5 \geq$ [ * ] /°F for increasing $T_{avg}$
[ * ] /°F for decreasing $T_{avg}$

$K_6 \geq$ [ * ] /°F when $T > T''$
[ * ] /°F when $T \leq T''$

$\tau_3 \leq$ [ * ] sec

$f_2(\Delta I)$ = [ * ]

*The values denoted with [ * ] are specified in the COLR.

Table 3.3.2-1 (page 1 of 6)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Safety Injection | | | | | |
| a. Manual Initiation | 1,2,3,4 | 2 | B | SR 3.3.2.6 | NA |
| b. Automatic Actuation Logic and Actuation Relays | 1,2,3,4 | 2 trains | C | SR 3.3.2.2<br>SR 3.3.2.3<br>SR 3.3.2.5 | NA |
| c. Containment Pressure-Hi | 1,2,3 | 3 | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≤4.80 psig |
| d. Pressurizer Pressure-Low | 1,2,3[(b)] | 3 | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≥1710 psig |
| e. High Differential Pressure Between Steam Lines | 1,2,3 | 3 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | NA |
| f. High Steam Flow in Two Steam Lines | 1,2[(d)],3[(d)] | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | (c) |
| Coincident with $T_{avg}$- Low | 1,2[(d)],3[(d)] | 1 per loop | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≥540.5°F |

(continued)

(a) Not used

(b) Above the Pressurizer Pressure interlock.

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 120% full steam flow at 100% load, and corresponding to 120% full steam flow above 100% load. Time delay for SI ≤6 seconds.

(d) Except when all MSIVs are closed.

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

| FUNCTION | APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS | REQUIRED CHANNELS | CONDITIONS | SURVEILLANCE REQUIREMENTS | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 4. Steam Line Isolation | | | | | |
| a. Manual Initiation | 1,2(d),3(d) | 2 per steam line | F | SR 3.3.2.6 | NA |
| b. Automatic Actuation Logic and Actuation Relays | 1,2(d),3(d) | 2 trains | G | SR 3.3.2.2<br>SR 3.3.2.3<br>SR 3.3.2.5 | NA |
| c. Containment Pressure (Hi-Hi) | 1,2(d), 3(d) | 2 sets of 3 | E | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≤24 psig |
| d. High Steam Flow in Two Steam Lines | 1,2(d), 3(d) | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | (c) |
| Coincident with $T_{avg}$-Low | 1,2(d), 3(d) | 1 per loop | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≥540.5°F |
| e. High Steam Flow in Two Steam Lines | 1,2(d), 3(d) | 2 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | (c) |
| Coincident with Steam Line Pressure-Low | 1,2(d), 3(d) | 1 per steam line | D | SR 3.3.2.1<br>SR 3.3.2.4<br>SR 3.3.2.7 | ≥500 psig |

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 120% full steam flow at 100% load, and corresponding to 120% full steam flow above 100% load. Time delay for SI ≤6 seconds.

(d) Except when all MSIVs are closed.

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;

b. RCS average loop temperature is less than or equal to the limit specified in the COLR; and

c. RCS total flow rate ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

------------------------------NOTE------------------------

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

-------------------------------------------------------

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more RCS DNB parameters not within limits. | A.1 Restore RCS DNB parameter(s) to within limits. | 2 hours |
| B. Required action and associated Completion Time not met. | B.1 Be in MODE 2. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.1.1 | Verify pressurizer pressure is greater than or equal to the limit specified in the COLR. | 12 hours |
| SR 3.4.1.2 | Verify RCS average loop temperature is less than or equal to the limit specified in the COLR. | 12 hours |
| SR 3.4.1.3 | Verify RCS total flow rate is ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR. | 12 hours |
| SR 3.4.1.4 | -------------------NOTE----------------<br>Not required to be performed until 24 hours after ≥ 90% RTP.<br>---------------------------------------<br><br>Verify by precision heat balance that RCS total flow rate is ≥ 354,400 gpm and greater than or equal to the limit specified in the COLR. | 24 months |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 54.3% in MODES 1 and 2 or ≤ 90% in MODE 3; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Pressurizer water level not within limit. | A.1 Be in MODE 3 with reactor trip breakers open. | 6 hours |
| | <u>AND</u> | |
| | A.2 Be in MODE 4. | 12 hours |
| B. One required group of pressurizer heaters inoperable. | B.1 Restore required group of pressurizer heaters to OPERABLE status. | 72 hours |
| C. Required Action and associated Completion Time of Condition B not met. | C.1 Be in MODE 3. | 6 hours |
| | <u>AND</u> | |
| | C.2 Be in MODE 4. | 12 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.9.1 | Verify pressurizer water level is ≤ 54.3% in MODES 1 and 2 OR ≤ 90% in MODE 3. | 12 hours |
| SR 3.4.9.2 | Verify capacity of each required group of pressurizer heaters is ≥ 150 kW. | 24 months |

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Neutron Flux Trip Setpoint in Percent of RATED THERMAL POWER

| MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE | APPLICABLE Neutron Flux Trip Setpoint (% RTP) |
|---|---|
| 4 | ≤ 57 |
| 3 | ≤ 38 |
| 2 | ≤ 20 |

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

cooler unit when pressurized at ≥ 1.1 Pa. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 42.0 psig. The containment design pressure is 47 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 0.1% of primary containment air weight per day.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Specification 2.1, Safety Limits (SL);
2. Specification 3.1.1, Shutdown Margin;
3. Specification 3.1.3, Moderator Temperature Coefficient;
4. Specification 3.1.5, Shutdown Bank Insertion Limits;
5. Specification 3.1.6, Control Bank Insertion Limits;
6. Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));
7. Specification 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor;
8. Specification 3.2.3, AXIAL FLUX DIFFERENCE (AFD);
9. Specification 3.3.1, Reactor Protection System Instrumentation;
10. Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits; and
11. Specification 3.9.1, Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Specifications 3.1.5, Shutdown Bank Insertion Limits, 3.1.6, Control Bank Insertion Limits, and 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor);

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES, TOPICAL REPORT," September 1974 (W Proprietary). (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control);

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Specification 3.2.3, Axial Flux Difference (AFD) (Constant Axial Offset Control));

3a. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss-of-Coolant-Accident Analysis," March 1998 (Westinghouse Proprietary);

3b. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989 (Specification 2.1, Safety Limits (SL)) and Specification 3.4.1, (RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits);

3c. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986 (Specification 2.1, Safety Limits (SL));

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z));

3e. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code; Safety Injection into the Broken Loop and Cosi Condensation Model," July 1997 (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)));

3f. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor (FQ(Z))); and

3g. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary). (Specification 3.2.1, Heat Flux Hot Channel Factor).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided for each reload cycle to the NRC.

5.6.6 NOT USED

(continued)