2.0 SAFETY LIMITS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for WRB-1 DNB correlation for Vantage 5H (V5H) fuel assemblies, and ≥ 1.14 for WRB-2M DNB correlation for Robust Fuel Assemblies (RFA).

2.1.1.2 The peak fuel centerline temperature shall be maintained ≤ 4700°F.

APPLICABILITY: MODES 1 and 2.

ACTION:

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
7. Overtemperature ΔT	3	2	2	See Table Notation (A)	1, 2	7
8. Overpower ΔT	3	2	2	See Table Notation (B)	1, 2	7
9. Pressurizer Pressure-Low (Above P-7)	3	2	2	≥ 1941 psig	1, 2	7
10. Pressurizer Pressure-High	3	2	2	≤ 2389 psig	1, 2	7
11. Pressurizer Water Level-High (Above P-7)	3	2	2	≤ 92.5% of instrument span	1, 2	7
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	≥ 89.8% of indicated loop flow	1	7
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	≥ 89.8% of indicated loop flow	1	7
14. Steam Generator Water Level-Low-Low (Loop Stop Valves Open)	3/loop	2/loop	2/loop	≥ 19.1% of narrow range instrument span-each steam generator	1, 2	7

TABLE 3.3-1 (Continued)

TABLE NOTATION

(1) Trip function may be manually bypassed in this Mode above P-10.

(2) Trip function may be manually bypassed in this Mode above P-6.

(3) With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

(8) In this condition, source range Function does not provide reactor trip but does provide indication.

(A): Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.5% ΔT span for the T_{avg} channel, 0.5% ΔT span for the Pressurizer Pressure channel and 0.5% ΔT span for the f(ΔI) channel.

$$\Delta T \frac{1}{(1+\tau_4 S)} \leq \Delta T_0 \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)\left[T \frac{1}{(1+\tau_5 S)} - T'\right] + K_3 \left(P - P'\right) - f(\Delta I)\right]$$

where: ΔT is measured RCS ΔT, °F.

ΔT_0 is loop specific indicated ΔT at RATED THERMAL POWER, °F.

T is measured RCS average temperature, °F.

T' is T_{avg} at RATED THERMAL POWER specified in the COLR.

P is measured pressurizer pressure, psia.

P' is nominal pressurizer pressure specified in the COLR.

$\frac{1+\tau_1 S}{1+\tau_2 S}$ is the function generated by the lead-lag compensator for T_{avg}.

τ_1 & τ_2 are the time constants utilized in the lead-lag compensator for T_{avg} specified in the COLR.

$\frac{1}{(1+\tau_4 S)}$ is the function generated by the lag compensator for measured ΔT.

$\frac{1}{(1+\tau_5 S)}$ is the function generated by the lag compensator for measured T_{avg}.

τ_4 & τ_5 are the time constants utilized in the lag compensators for the ΔT and T_{avg}, respectively, specified in the COLR.

TABLE 3.3-1 (Continued)

TABLE NOTATION (Continued)

S is the Laplace transform operator, sec^{-1}.

K_1 is specified in the COLR.

K_2 is specified in the COLR.

K_3 is specified in the COLR.

f(ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers as specified in the COLR.

TABLE NOTATION (Continued)

(B): Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel and 0.5% ΔT span for the T_{avg} channel.

$$\Delta T \frac{1}{(1+\tau_4 S)} \leq \Delta T_0 \left[K_4 - K_5\left(\frac{\tau_3 S}{1+\tau_3 S}\right) T \frac{1}{(1+\tau_5 S)} - K_6 \left[T \frac{1}{(1+\tau_5 S)} - T''\right]\right]$$

where: ΔT is measured RCS ΔT, °F.

ΔT_0 is loop specific indicated ΔT at RATED THERMAL POWER, °F.

T is measured RCS average temperature, °F.

T" is T_{avg} at RATED THERMAL POWER specified in the COLR.

K_4 is specified in the COLR.

K_5 is specified in the COLR.

K_6 is specified in the COLR.

$\frac{\tau_3 S}{1+\tau_3 S}$ is the function generated by the rate lag compensator for T_{avg}.

τ_3 is the time constant utilized in the rate lag compensator for T_{avg} specified in the COLR.

$\frac{1}{(1+\tau_4 S)}$ is the function generated by the lag compensator for measured ΔT.

$\frac{1}{(1+\tau_5 S)}$ is the function generated by the lag compensator for measured T_{avg}.

τ_4 & τ_5 are the time constants utilized in the lag compensators for the ΔT and T_{avg}, respectively, specified in the COLR.

S is the Laplace transform operator, sec^{-1}.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1, provided the other channel is OPERABLE.

ACTION 2 - With one power range neutron flux channel inoperable,[4] perform one of the following, as applicable:

a. Power Range High Neutron Flux Channel

1. Place the inoperable channel in trip within 6 hours and reduce THERMAL POWER to less than or equal to 75 percent RATED THERMAL POWER within the next 6 hours and perform SR 4.2.4,[5] or

2. Place the inoperable channel in trip within 6 hours and perform SR 4.2.4,[5] or

3. Be in MODE 3 within 12 hours.

b. All other channels

1. Place the inoperable channel in trip within 6 hours, or

2. Be in MODE 3 within 12 hours.

---

(4) The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels.

(5) Only required to be performed when the power range high neutron flux channel input to QPTR is inoperable.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION						
a. Manual	2/steam line	1/steam line	2/operat-ing steam line	Not Applicable	1, 2, 3	18
b. Automatic Actuation Logic	2	1	2	Not Applicable	1, 2, 3	13
c. Containment Pressure Intermediate-High-High	3	2	2	≤ 7.33 psig	1, 2, 3	14
d. Steamline Pressure-Low	3/loop	2/loop any loop	2/loop any loop	≥ 495.8 psig steam line pressure	1, 2, 3(1)	14
e. Steamline Pressure Rate-High Negative	3/loop	2/loop any loop	2/operat-ing loop	≤ 104.2 psi with a time constant ≥ 50 seconds	3(2)	14
5. TURBINE TRIP & FEEDWATER ISOLATION						
a. Steam Generator Water Level--High-High, P-14	3/loop	2/loop in any operating loop	2/loop in each operating loop	≤ 90.2% of narrow range instrument span each steam generator	1, 2, 3	14

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ALLOWABLE VALUE	APPLICABLE MODES	ACTION
7. AUXILIARY FEEDWATER						
a. Steam Gen. Water Level-Low-Low (Loop Stop Valves Open)						
i. Start Turbine Driven Pump	3/stm. gen.	2/stm. gen. any stm. gen.	2/stm. gen.	≥ 19.1% of narrow range instrument span each steam generator	1, 2, 3	14
ii. Start Motor Driven Pumps	3/stm. gen. any 2 stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen.	≥ 19.1% of narrow range instrument span each steam generator	1, 2, 3	14
b. Undervoltage-RCP (Start Turbine Driven Pump)	(3)-1/bus	2	2	≥ 71.2% rated RCP bus voltage	1	14
c. S.I. (Start All Auxiliary Feedwater Pumps)	See 1 above (all S.I. initiating functions and requirements)					
d. (Deleted)						
e. Trip of Main Feedwater Pumps (Start Motor Driven Pumps)	1/pump	1	1	Not Applicable	1, 2, 3	18

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
c. Containment Pressure--Intermediate-High-High	S	R(2)(3)	Q(2)(3)	1, 2, 3
d. Steamline Pressure--Low	S	R	Q	1, 2, 3
e. Steamline Pressure Rate-High Negative	S	R	Q	1, 2, 3
5. TURBINE TRIP & FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R(2)(3)	Q(2)(3)	1, 2, 3
6. LOSS OF POWER				
a. 4.16kv Emergency Bus Under-voltage (Loss of Voltage) Trip Feed & Start Diesel	N.A.	R	Q	1, 2, 3, 4
b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	Q	1, 2, 3, 4

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operation involving a reduction in boron concentration of the Reactor Coolant system and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level greater than or equal to 28% narrow range at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50 percent of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and

SURVEILLANCE REQUIREMENTS (Continued)

2. Tubes in those areas where experience has indicated potential problems, and

3. A tube inspection pursuant to Specification 4.4.5.4.a.8 shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

SURVEILLANCE REQUIREMENTS (Continued)

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5 percent of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10 percent) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection of the Model 54F steam generators shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality following steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

Note: Inservice inspection is not required during the steam generator replacement outage.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steamline or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20 percent of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.

4. Percent Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to the 40 percent of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot-leg side) completely around the U-bend to the top support of the cold-leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be submitted in a Special Report in accordance with 10 CFR 50.4.

b. The complete results of the steam generator tube inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

SURVEILLANCE REQUIREMENTS (Continued)

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One (1)	One (2)

Table Notation:

(1) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9 percent of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

(2) The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in (1) above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S.G.</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.

Notification to NRC pursuant to Specification 6.6</td><td>All other S.G.s are C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S.G.s are C-2 but no additional S.G.s are C-3</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S.G. is C-3</td><td>Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Specification 6.6.</td><td>N/A</td><td>N/A</td></tr>
</table>

$S = \frac{9}{n}\%$ Where n is the number of steam generators inspected during an inspection.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between 6681 and 7645 gallons of usable borated water,

c. Between 2300 and 2600 ppm of boron, and

d. A nitrogen cover-pressure of between 611 and 685 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable due to boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 72 hours.

b. With one accumulator inoperable for reasons other than Action a, restore the inoperable accumulator to OPERABLE status within 24 hours.

c. With either Action a or b not being completed within the specified completion time, be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to ≤ 1000 psig within 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the usable borated water volume and nitrogen cover-pressure in the tanks are within limits, and

2. Verifying that each accumulator isolation valve is open.

---

* Pressurizer Pressure above 1000 psig.

3/4.5.5 SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.5 Reactor coolant pump seal injection flow shall be less than or equal to 28 gpm with the charging pump discharge pressure greater than or equal to 2457 psig and the seal injection flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure greater than or equal to 2457 psig and the seal injection flow control valve full open within 4 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 Verify at least once per 31 days that the valves are adjusted to give a flow within the limit with the charging pump discharge at greater than or equal to 2457 psig and the seal injection flow control valve full open.(1)

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(1) Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at greater than or equal to 2215 psig and less than or equal to 2255 psig.

CORE OPERATING LIMITS REPORT (Continued)

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:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1988 (Westinghouse Proprietary).

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.