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INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

SECTION PAGE

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL Unit	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Overpower N-16					
a. Unit 1	4.0	1.93	0	≤112% of RTP*	≤115.1% of RTP*
b. Unit 2	4.0	2.05	1.0+0.05(3)	≤112% of RTP*	≤114.5% of RTP*
9. Pressurizer Pressure-Low					
a. Unit 1	4.4	0.71	2.0	≥1880 psig	≥1863.6 psig
b. Unit 2	4.4	1.12	2.0	≥1880 psig	≥1863.6 psig
10. Pressurizer Pressure-High					
a. Unit 1	7.5	5.01	1.0	≤2385 psig	≤2400.8 psig
b. Unit 2	7.5	1.12	2.0	≤2385 psig	≤2401.4 psig
11. Pressurizer Water Level-High					
a. Unit 1	8.0	2.18	2.0	≤92% of instrument span	≤93.9% of instrument span
b. Unit 2	8.0	2.35	2.0	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low					
a. Unit 1	2.5	1.18	0.6	≥90% of loop design flow**	≥88.6% of loop design flow**
b. Unit 2	2.5	1.25	0.87	>90% of loop minimum measured flow***	≥ >88.8% of loop minimum measured flow***

(1)

(3) 1.0% span for N-16 power monitor and 0.05% for T_{cold} RTDs.

*RTP = RATED THERMAL POWER

**Loop design flow = 95,700 gpm.

***Loop minimum measured flow = 98,500 gpm

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BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. 1

The DNB design basis is that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence level that DNB will not occur when the minimum DNBR is at the DNBR limit. In meeting this design basis, uncertainties in plant operating ~~parameters, nuclear and thermal parameters and fuel fabrication~~ parameters are considered such that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses. 1

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis limit value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

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REACTOR CORE (continued)

(1) These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, and a reference for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

where: P = the fraction of RATED THERMAL POWER (RTP),

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RTP specified in the CORE OPERATING LIMITS REPORT (COLR), and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

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 N-16 trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature N-16 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 (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735) psig of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

(2)

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3[a], 4[a], 5[a]
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1[c], 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1[c], 2
6. Source Range, Neutron Flux	S	R(4, 13)	S/U(1), Q(9)	R(12)*	N.A.	2[b], 3, 4, 5
7. Overtemperature N-16	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
8. Overpower N-16	S	D(2, 4), R(4, 5)	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(8)	N.A.	N.A.	1[d]
10. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3 and for Unit 2 six months after initial criticality.

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation (Continued)								
c. Containment Vent Isolation								
1) Manual Initiation	See Item 3.a.1 and 2.a above. Containment vent isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiated.							1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
(2) e. Steam Line Pressure-Negative Rate-High	S	R	~~M~~ Q	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, [illegible]

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

(2) d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

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REACTOR COOLANT SYSTEM

RELIEF VALVES

SURVEILLANCE REQUIREMENTS (Continued)

(2) 4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION ~~b~~ 9b or c in Specification 3.4.4.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE R1108-1 (UNIT 1)
INTERMEDIATE SHELL PLATE R3807-2 (UNIT 2)

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INITIAL RT_{NDT}: 0°F (UNIT 1), 10°F (UNIT 2)

RT_{NDT} AFTER 16 EFPY: 1/4T, 85°F (UNIT 1), 81°F (UNIT 2)
3/4T, 70°F (UNIT 1), 62°F (UNIT 2)

CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

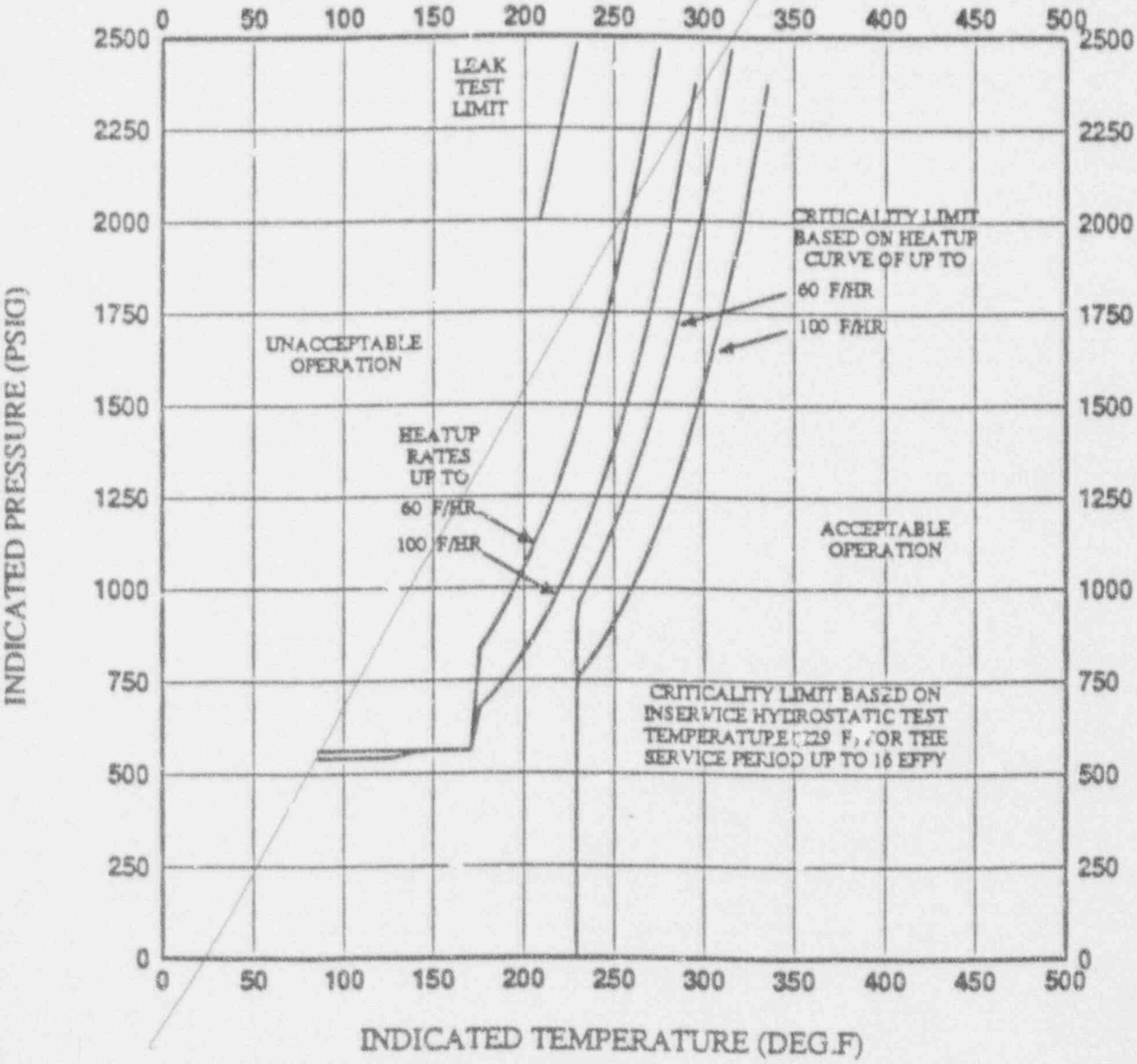


FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 16 EFPY

1

INSERT ATTACHED FIGURE

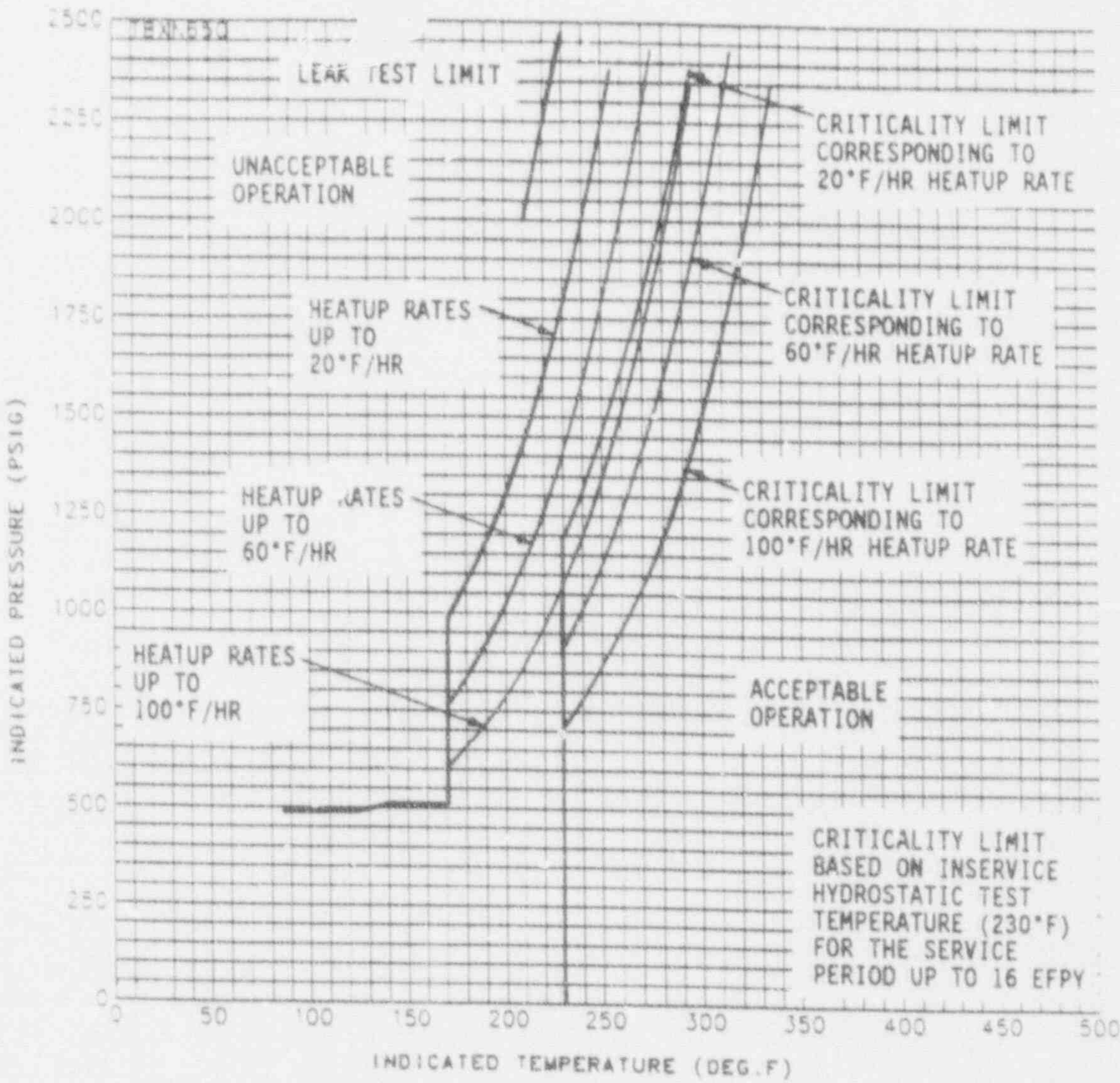
MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE R1108-1 (UNIT 1)
INTERMEDIATE SHELL PLATE R3807-2 (UNIT 2)

INITIAL RT_{NDT}: 0°F (UNIT 1), 10°F (UNIT 2)

ART AT 16 EFPY: 1/4T : 84°F (UNIT 1), 81°F (UNIT 2)
3/4T : 69°F (UNIT 1), 62°F (UNIT 2)

CURVES BOUNDING COMANCHE PEAK UNITS 1 AND 2. APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 110 PSIG FOR POSSIBLE INSTRUMENTATION ERRORS.



MATERIAL PROPERTY BASIS

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CONTROLLING MATERIAL: LOWER SHELL PLATE R1108-1 (UNIT 1)
INTERMEDIATE SHELL PLATE R3807-2 (UNIT 2)

INITIAL RT_{NDT}: 0°F (UNIT 1), 10°F (UNIT 2)

RT_{NDT} AFTER 16 EFPY: 1/4T, 85°F (UNIT 1), 81°F (UNIT 2)
3/4T, 70°F (UNIT 1), 62°F (UNIT 2)

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

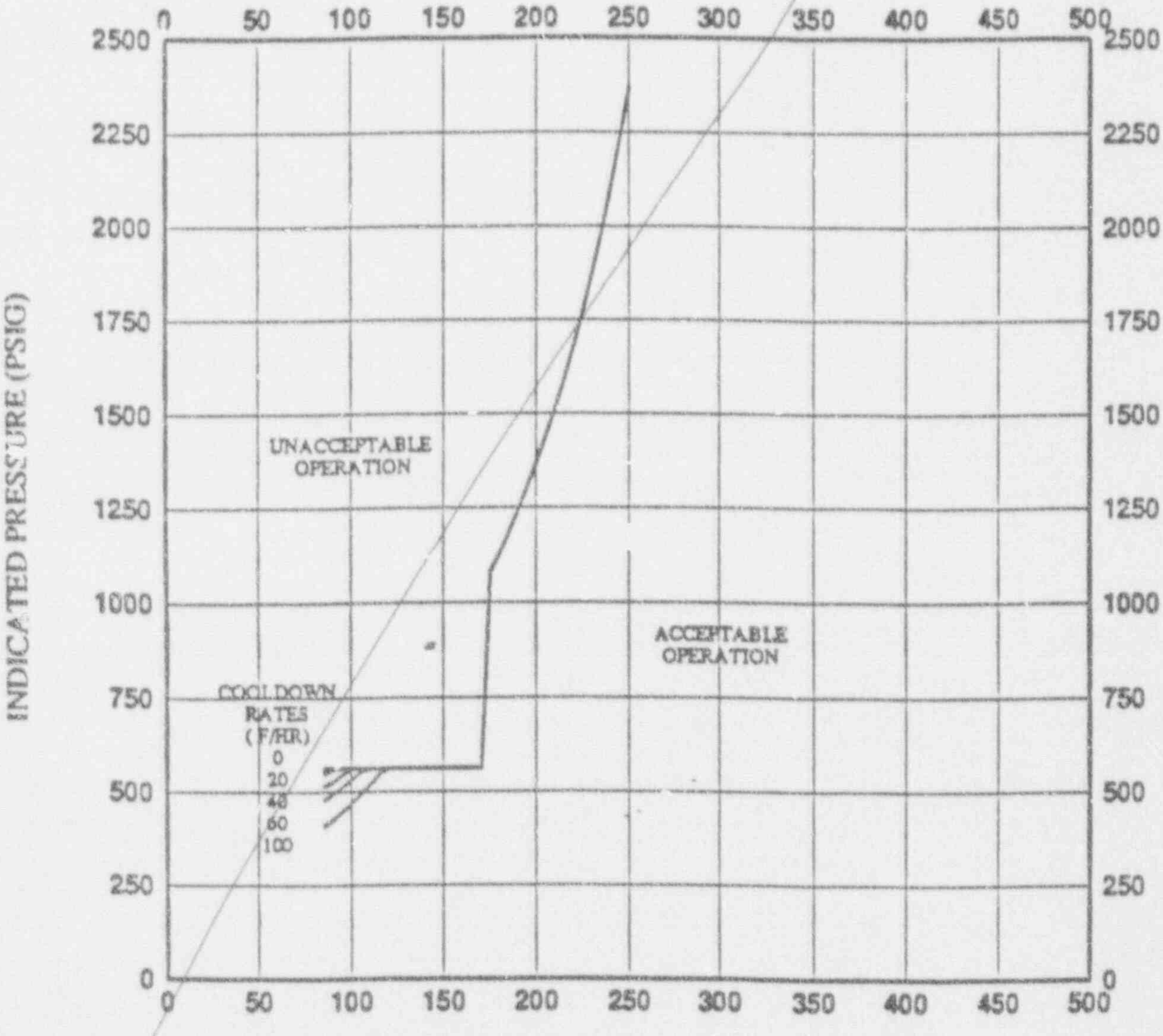


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 16 EFPY

INSERT ATTACHED FIGURE 1

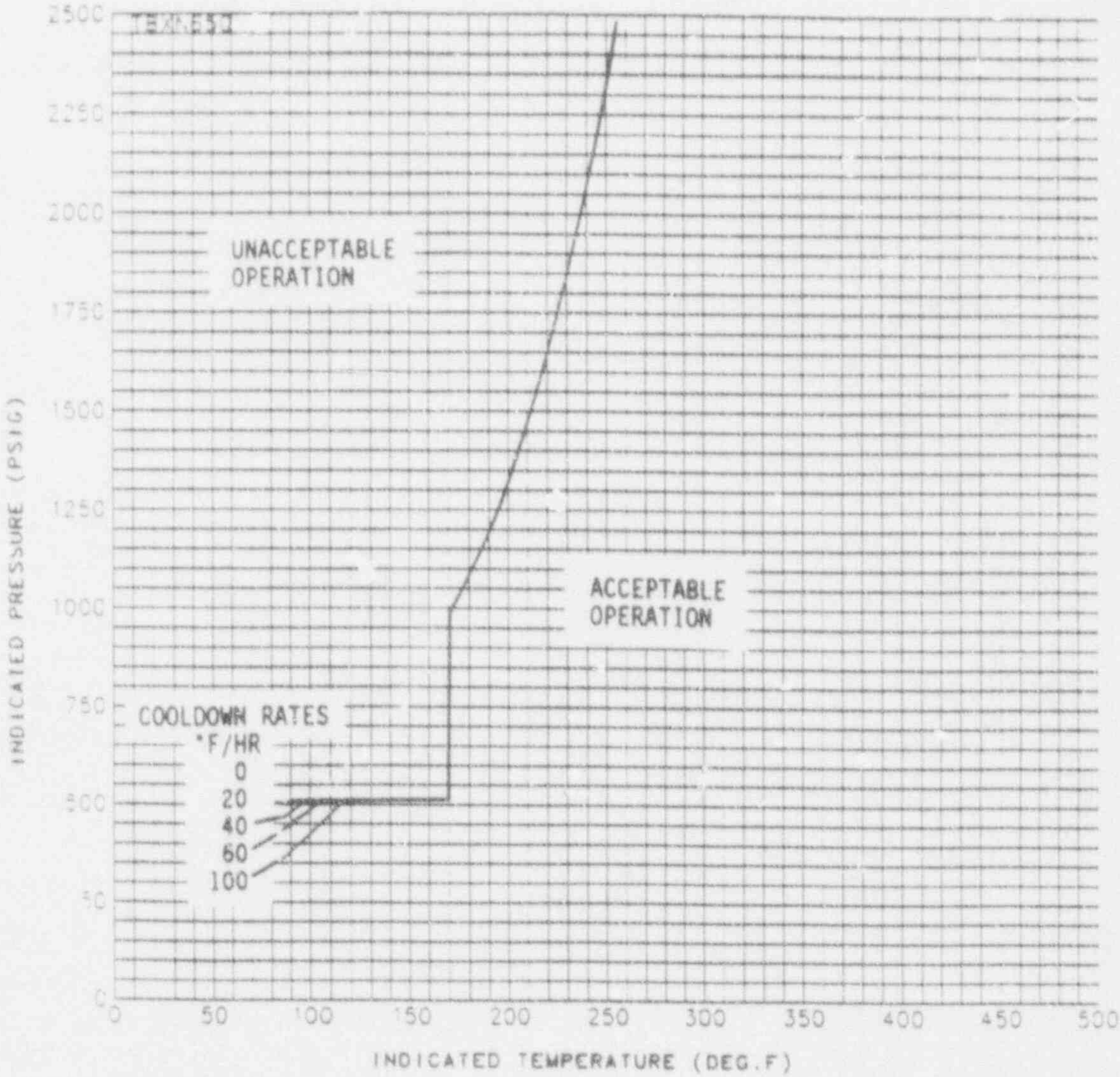
MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: LOWER SHELL PLATE R1108-1 (UNIT 1)
INTERMEDIATE SHELL PLATE R3807-2 (UNIT 2)

INITIAL RT_{NDT}: 0°F (UNIT 1), 10°F (UNIT 2)

ART AT 16 EFPY: 1/4T : 84°F (UNIT 1), 81°F (UNIT 2)
3/4T : 69°F (UNIT 1), 62°F (UNIT 2)

CURVES BOUNDING COMANCHE PEAK UNITS 1 AND 2. APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 110 PSIG FOR POSSIBLE INSTRUMENTATION ERRORS.



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① INSERT ATTACHED FIGURE

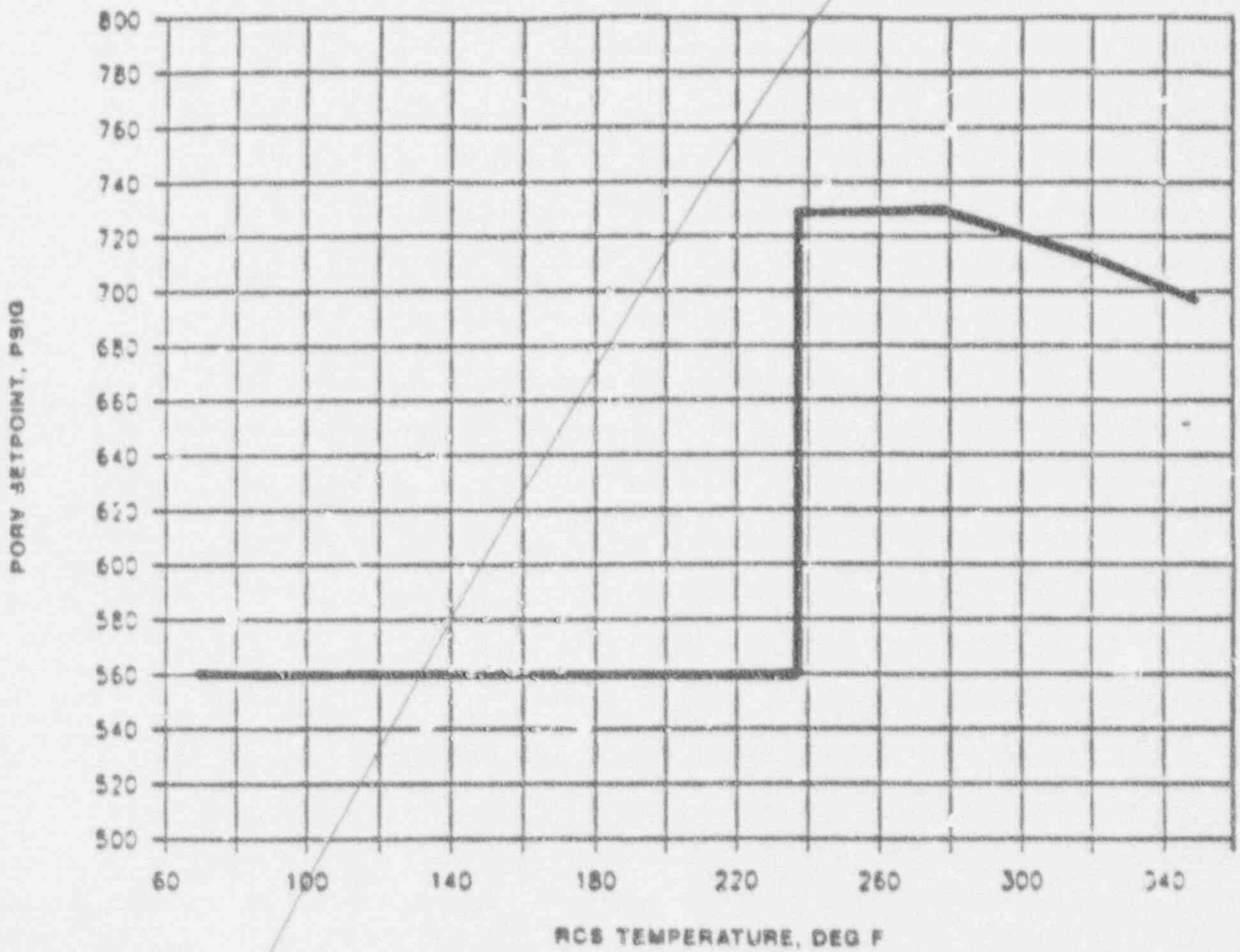

FIGURE 3.4-4 PORV SETPOINTS FOR OVERPRESSURE MITIGATION
APPLICABLE UP TO 16 EFPY

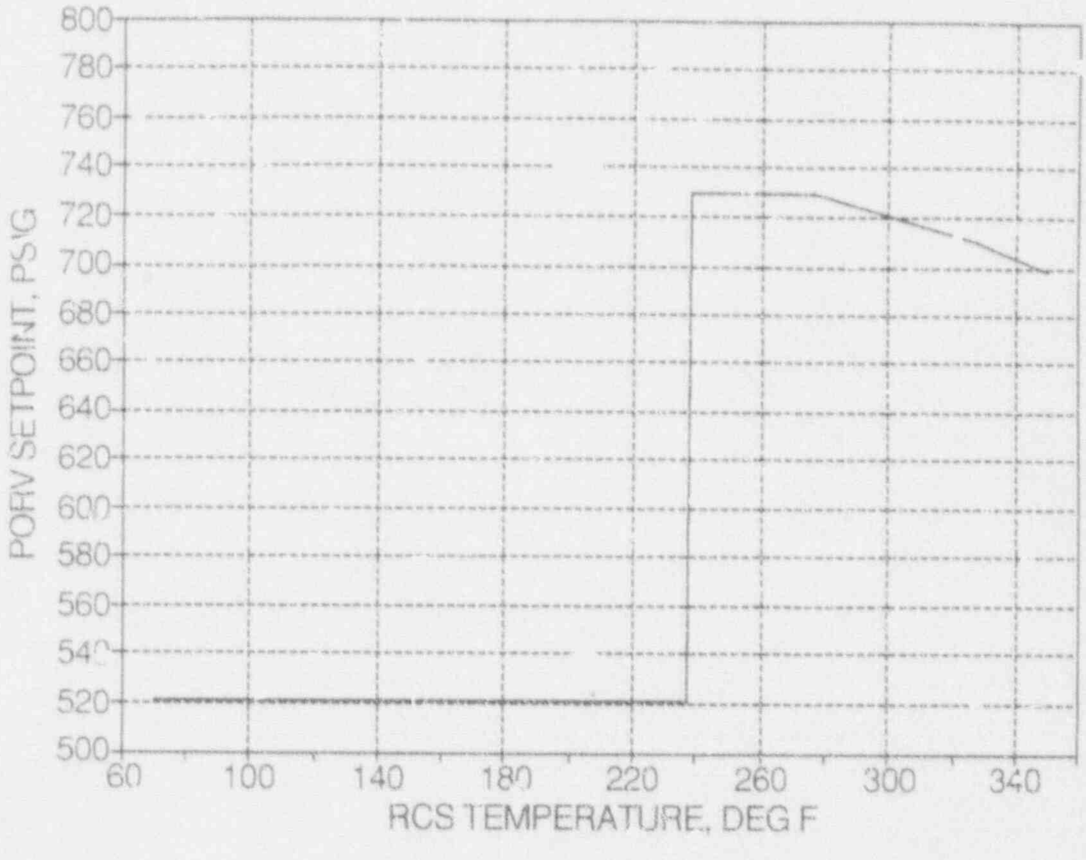

FIGURE 3.4-4

PORV SETPOINTS FOR OVERPRESSURE MITIGATION -
APPLICABLE FOR UP TO 16 EFPY

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3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

INSERT A

INSERT A

Note: Separate Technical Specifications provide specific temperature/pressure limitations for specific components (e.g., 3/4.4.8.1 for the Reactor Coolant System, 3/4.7.2 for the Steam Generators, 3/4.7.13 for the Main Feedwater Isolation Valves, etc.)

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3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open with power removed,

b. An indicated borated water level of between 39% and 61%

c. A boron concentration of between 1900 and 2200 ppm, and

d. An indicated cover-pressure of between 623 and 644 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or the boron concentration outside the required values, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With the boron concentration of one cold leg injection accumulator outside the required limit, restore the boron concentration to within the required limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the indicated borated water level ~~volume~~ and nitrogen cover-pressure in the tanks, and (2)

*Pressurizer pressure above 1000 psig.

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PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with an indicated water level of at least 53%.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the Station Service Water (SSW) system as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

level

4.7.1.3.1 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying the indicated water ~~volume~~ is within its limits when the tank is the supply source for the auxiliary feedwater pumps. 2

4.7.1.3.2 The SSW system shall be demonstrated OPERABLE at least once per 12 hours whenever the SSW system is being used as an alternate supply source to the auxiliary feedwater pumps by verifying the SSW system OPERABLE and each motor operated valve between the SSW system and each OPERABLE auxiliary feedwater pump is OPERABLE.

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3/4.7.4 STATION SERVICE WATER SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent station service water loops per unit and the cross-connect between the Station Service Water Systems of each unit shall be OPERABLE.

APPLICABILITY: Units 1 and 2 in MODES 1, 2, 3, and 4.

ACTION:

a. With only one station service water loop in a ~~per~~ unit OPERABLE, restore at least two loops per unit to OPERABLE status within 72 hours or for the unit(s) with the inoperable station service water loop be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (1)

b. With one or more of the cross-connects inoperable, within 7 days, restore the cross-connect(s) to OPERABLE status, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1.1 Each station service water loop shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months ~~during shutdown~~, by verifying that each station service water pump starts automatically on a Safety Injection test signal.

4.7.4.1.2 At least once per 92 days the cross-connects shall be demonstrated OPERABLE by cycling the cross-connect valves in the flow path or verifying that ~~the~~ these valves are locked open.

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PLANT SYSTEMS

STATION SERVICE WATER SYSTEM

ONE UNIT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.4.2 At least two independent station service water loops in the operating unit*, at least one station service water pump in the shutdown unit** and the cross-connects from the OPERABLE station service water pump(s) in the shutdown unit to the station service water loops of the operating unit shall be OPERABLE.

APPLICABILITY: Unit 1 (Unit 2) in MODES 1, 2, 3 and 4
Unit 2 (Unit 1) in MODES 5, 6 and Defueled

(1)

ACTION:

a. With one station service water loop in the operating unit inoperable, restore two loops in the operating unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more of the cross-connects between the OPERABLE station service water pump(s) in the shutdown unit and the station service water loops in the operating unit inoperable, within 7 days restore the ~~cross-connects~~ to OPERABLE status. Otherwise, place the operating unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

inoperable valves

c. If neither station service water pump in the shutdown unit is OPERABLE, restore at least one pump to OPERABLE status within 7 days or place the operating unit in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.2.1 Each station service water loop in the operating unit shall be demonstrated OPERABLE per the requirements of Specification 4.7.4.1.1.

4.7.4.2.2 At least once per 92 days the cross-connect(s) between the OPERABLE station service water pump(s) in the shutdown unit and the station service water loops in the operating unit shall be demonstrated OPERABLE by cycling the cross-connect valves in the flow path or verifying that these valves are locked open.

T

* A Unit in MODE 1, 2, 3 or 4 is designated as the "Operating unit".
** A unit is MODE 5, 6 or Defueled is designated as the "shutdown Unit".

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3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

a. A minimum water level at or above elevation 770 feet Mean Sea Level, USGS datum,

(3) b. A station service water intake temperature of less than or equal to (102)°F, and

c. A maximum average sediment depth of less than or equal to 1.5 feet in the service water intake channel.

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

ACTION: (Units 1 and 2)

a. With the above requirements for water level and intake temperature not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the average sediment depth in the service water intake channel greater than 1.5 feet, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of all surveillances performed pursuant to Specification 4.7.5c and specify what measures will be employed to remove sediment from the service water intake channel.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the station service water intake temperature and UHS water level to be within their limits,

b. At least once per 12 months by visually inspecting the dam and verifying no abnormal degradation or erosion, and

c. At least once per 12 months by verifying that the average sediment depth in the service water intake channel is less than or equal to 1.5 feet.

New Specification

(1)

PLANT SYSTEMS

3/4.7.13 MAIN FEEDWATER ISOLATION VALVE PRESSURE/TEMPERATURE LIMIT

LIMITING CONDITION FOR OPERATION

3.7.13 The valve body and neck of each main feedwater isolation valve shall be greater than or equal to 90°F, when feedwater line pressure is greater than 675 psig.

APPLICABILITY: MODES 1, 2, 3 and during pressure testing of the steam generator or main feedwater line.

ACTION:

With one or more main feedwater isolation valves outside of the above limits, restore main feedwater isolation valve pressure and/or temperature to within the limits within one hour, and perform an engineering evaluation to determine the effect of the overpressure on the structural integrity of the main feedwater isolation valve(s) and determine that the main feedwater isolation valve(s) remains acceptable for continued operation within 72 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.13 Each main feedwater isolation valve shall be determined to be greater than or equal to 90°F at least once per 12 hours.*

* Except in MODE 1 with the main feedwater isolation valve open.

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BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of two charging pumps to be OPERABLE and the requirement to verify one charging pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The limitation for minimum solution temperature of the borated water sources are sufficient to prevent boric acid crystallization with the highest allowable boron concentration.

(3) The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either [1,100] gallons of 7000 ppm borated water from the boric acid storage tanks or [7,113] gallons of 2000 ppm borated water from the RWST.

As listed below, the required indicated levels for the boric acid storage tanks and the RWST include allowances for required/analytical volume, unusable volume, measurement uncertainties (which include instrument error and tank tolerances, as applicable), ~~system configuration requirements~~ margin, and other required volume.

(4)

Tank	MODES	Ind. Level	Unusable Volume (gal)	Required Volume (gal)	Measurement Uncertainty	~~System Config.~~ Margin (gal)	Other (gal)
RWST	5,6	24%	~~45,494~~ 98,900	7,113	4% of span	~~57,857~~ 10,293	N/A
	1,2,3,4	95%	45,494	70,702	4% of span	N/A	357,535*
Boric	5,6	10%	3,221	1,100	6% of span	N/A	N/A
Acid	5,6	20%	3,221	1,100	6% of span	3,679	N/A
Storage	(gravity	feed)					
Tank	1,2,3,4	50%	3,221	15,700	6% of span	N/A	N/A

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

*Additional volume required to meet Specification 3.5.4.

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The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the safety analysis limit value during normal operation and in short-term transients, and (2) 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; and

$F^N_{\Delta H}$ 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The 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 target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

1

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density

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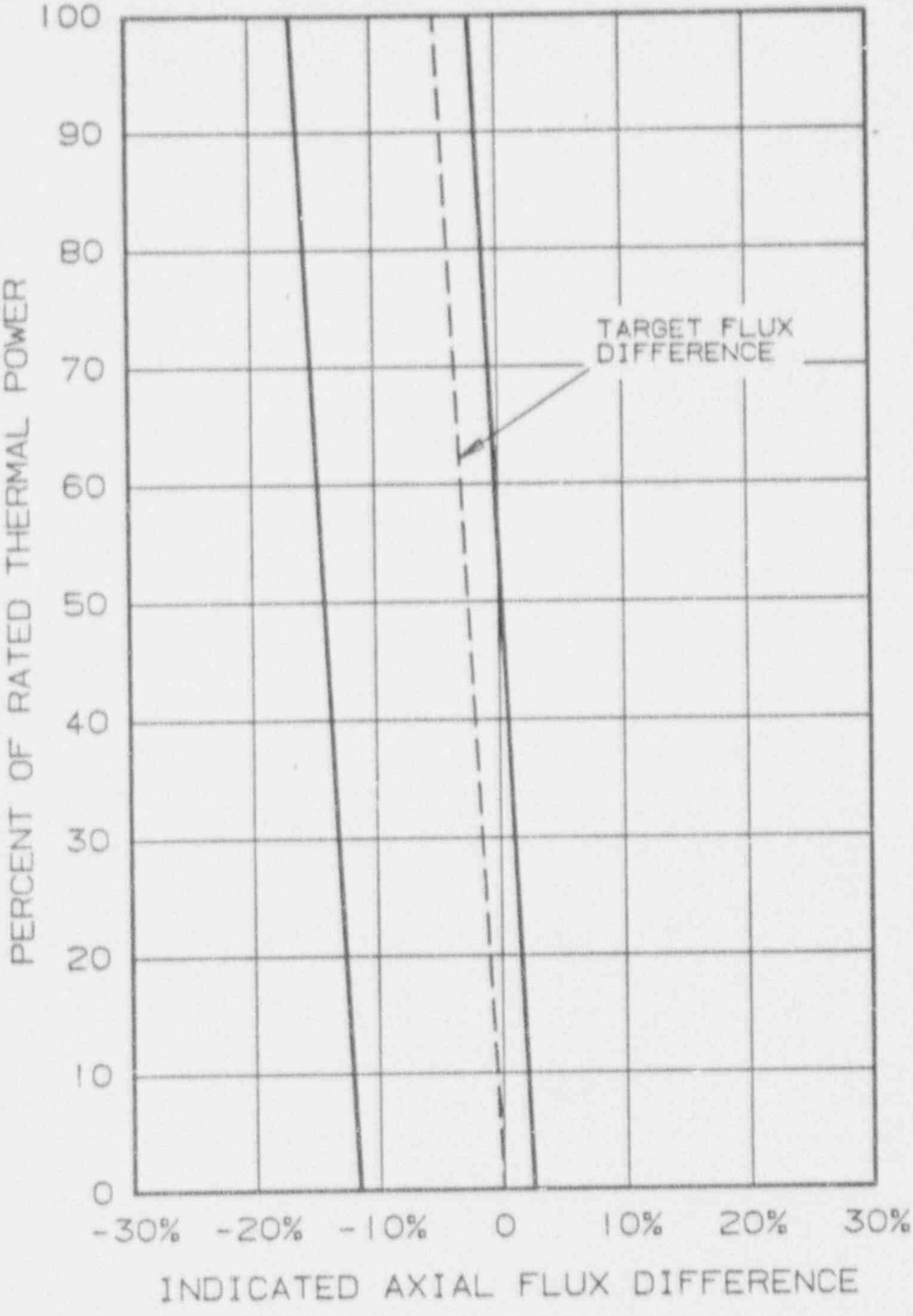

FIGURE B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°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, indicated, 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; and

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. 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.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the generic margin. The DNBR generic margin, totaling 9.1% for Unit 1 and 10.1% for typical cells and 9.5% for thimble cells for Unit 2 completely offset any rod bow penalties. This margin includes the following for Unit 1:

(1) for DNBR

a. ~~Design limit~~ DNBR of 1.30 vs 1.28,

b. Grid Spacing (K_5) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.051,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The margin for Unit 2 is included by establishing a fixed difference between the safety analysis limit DNBR and the design limit DNBR equal to the percent margin of the safety analysis limit DNBR.
The applicable values of rod bow penalties are referenced in the FSAR.

(1)

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HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to Constant Axial Offset Control (CAOC) operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the range of burnup conditions in the core. The W(Z) function is provided in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.6.

When $F^N_{\Delta H}$ is measured, an adjustment for measurement uncertainty must be included for a full-core flux map taken with the Incore Detector Flux Mapping System.

$F_Q(Z)$ should be measured with the reactor core at, or near, equilibrium conditions. Therefore, the effects of transient maneuvers, such as power increases, should be permitted to decay to the extent possible while assuring that flux maps are taken in accordance with the specified surveillance schedules.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

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3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR at or above the safety analysis limit value throughout each analyzed transient. The Unit 1 indicated T_{avg} value of 592.7°F (conservatively rounded to 592°F) and the Unit 1 indicated pressurizer pressure value of 2207 psig correspond to analytical limits of 594.7°F and 2193 psig respectively, with allowance for measurement uncertainty. The Unit 2 indicated T_{avg} value of 592.8°F (conservatively rounded to 592°F) and the Unit 2 indicated pressurizer pressure value of 2219 psig correspond to analytical limits of 595.16°F and 2205 psig respectively, with allowance for measurement uncertainty. The indicated uncertainties assume that the reading from four channels will be averaged before comparing with the required limit.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation, and to detect any significant flow degradation of the Reactor Coolant System (RCS).

The additional surveillance requirements associated with the RCS total flow rate are sufficient to ensure that the measurement uncertainties are limited to 1.8% as assumed in the Improved Thermal Design Procedure Report for CPSES.

Performance of a precision secondary calorimetric is required to precisely determine the RCS temperature. The transit time flow meter, which uses the N-16 system signals, is then used to accurately measure the RCS flow. Subsequently, the RCS flow detectors (elbow tap differential pressure detectors) are normalized to this flow determination and used throughout the cycle.

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PRESSURE/TEMPERATURE LIMITS (Continued)

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1986 Edition to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can ~~cause~~ an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, (ART) based upon the fluence, and the chemical content of the material in question, has been predicted using Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". The fluence values for 16 EFPY is taken from the 26.5 degree plot in Figure B 3/4.4-1. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} up to 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments. (1)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-2. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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PRESSURE/TEMPERATURE LIMITS (Continued)

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50 Appendix G, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the Linear Elastic Fracture Mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the ~~nil-ductility~~ adjusted reference temperature, RT_{NDT}, is used ART (1)
and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

ART

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

ART

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal ~~nil-ductility~~ adjusted reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows: (1)

$$C\, K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material, (1)

ART

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PRESSURE/TEMPERATURE LIMITS (Continued)

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, ART and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated. 1

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

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ECCS SUBSYSTEMS (Continued)

to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The requirement to remove power from certain valve operators is in accordance with Branch Technical Position ICSB-18 for valves that fail to meet single failure considerations. Power is removed via key-lock switches on the control board.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) for small break LOCA and steam line breaks, the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly, and (3) for large break LOCAs, the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all shutdown and control rods fully withdrawn, and (4) sufficient time is available for the operator to take manual action and complete switchover of ECCS and containment spray suction to the containment sump without emptying the RWST or losing suction.

(4) The required indicated level includes a 4-percent measurement uncertainty, an unusable volume of 45,494 gallons and a required water volume of ~~428,437~~ 428,237 gallons.

The limits on indicated water volume and boron concentration of the RWST also ensure a long-term pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 430 gpm to two steam generators at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 860 gpm to four steam generators at a pressure of 1221 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The Auxiliary Feedwater System is capable of delivering a total feedwater flow of 430 gpm at a pressure of 1221 psig to the entrance of at least two steam generators while allowing for: (1) any possible spillage through the design worst case break of the main feedwater line; (2) the design worst case single failure; and (3) recirculation flow. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed in operation. The test flow for the steam-driven auxiliary feedwater pump at a pressure of greater than or equal to 1450 psid ensures this capability.

The auxiliary feedwater flow path is a passive flow path based on the fact that valve actuation is not required in order to supply flow to the steam generators. The automatic valves tested in the flow path are the Feedwater Split Flow Bypass which are required to be shut upon initiation of the Auxiliary Feedwater System to meet the requirements of the accident analysis.

Both steam supplies for the turbine-driven auxiliary feedwater pump must be OPERABLE in order to meet the design bases for the complete range of accident analyses. The allowed outage time for one inoperable steam source is consistent with the lower probability of the worst case steam or feedwater line break accident.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 18 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power or 4 hours at HOT STANDBY followed by a cooldown to 350°F at a rate of 50°F/hr for 5 hours. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The required indicated level includes a 3.5-percent measurement uncertainty, an unusable volume of ~~12,100~~ 12,900 gallons
(4) and a required usable volume of ~~249,900~~ 249,100 gallons.

NUREG-0737, Item II.E.1.1 requires a backup source to the CST which is the CPSES Station Service Water System, which can be manually aligned, if required in lieu of CST minimum water volume.

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 STATION SERVICE WATER SYSTEM

The OPERABILITY of the Station Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses. A unit in MODE 1, 2, 3 or 4 will be designated as operating and a unit in MODE 5, 6 or Defueled will be designated as shutdown with respect to the Station Service Water System.

Train isolation by two normally closed valves in series or one locked closed valve is provided to satisfy ~~GDC-44~~. Unit isolation by one locked closed valve is provided to satisfy ~~GDC-5~~. A pump for an operating unit is
1 inoperable when its associated cross-connect is open.
GDC-5

In the event of a total loss of Station Service Water in one unit at Comanche Peak, backup cooling capacity is available via a cross-connect between the two units. The OPERABLE pump(s) is manually realigned and flow balanced to provide cooling to essential heat loads. The OPERABILITY of the unit cross-connect along with a Station Service Water pump in the shutdown unit ensures the availability of sufficient redundant cooling capacity for the operating unit. The Limiting Condition of Operation will ensure a significant risk reduction as indicated by the analyses of a loss of Station Service Water System event. The surveillance requirements ensure the short and long-term OPERABILITY of the Station Service Water System and cross-connect between the two units.

The Station Service Water System cross-connect between the two units consists of appropriate piping, and cross-connect valves connecting the discharge of the Station Service Water pumps of the two units. By aligning the cross-connect flow paths, additional redundant cooling capacity from one unit is available to the Station Service Water System of the other unit.

A cross-connect valve is OPERABLE if it can be cycled or is locked open. A valve that cannot be demonstrated OPERABLE by cycling is considered inoperable until the valve is surveilled in the locked open position. However, at least one cross-connect valve between units is required to be maintained closed in accordance with GDC-5 unless required for flushing or due to total loss of Station Service Water pumps for either unit.

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3/4.7.11 UPS HVAC SYSTEM

The OPERABILITY of the UPS HVAC System ensures that the uninteruptible power supplv and distribution rooms ambient air temperatures do not exceed the allowable temperatures per Specification 3/4.7.10 for continuous-duty rating for the equipment and instrumentation cooled by this equipment.

3/4.7.12 SAFETY CHILLED WATER SYSTEM

The OPERABILITY of the Safety Chilled Water System ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

1 INSERT B

INSERT B

3/4.7.13 MAIN FEEDWATER ISOLATION VALVE PRESSURE/TEMPERATURE LIMIT

The fracture toughness requirements are satisfied with a metal temperature of 90°F for the main feedwater isolation valve body and neck, therefore, these portions will be maintained at or above this temperature prior to pressurization of these valves above 675 psig. Minimum temperature limitations are imposed on the valve body and neck of main feedwater isolation valves HV-2134, HV-2135, HV-2136 and HV-2137. These valves do not need to be verified at or above 90°F when in MODES 4, 5, or 6 (except during special pressure testing) since Tavg < 350°F which corresponds to a pressure at the valves of 140-150 psig or less. The maximum pressurization during cold conditions (valve temperature < 90°F) should be limited to no more than 20% of the valve hydrostatic test pressure (3375 psig x 20% = 675 psig).