

ENCLOSURE 4

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT

TECHNICAL SPECIFICATION PAGES

8907070181 890630
PDR ADOCK 05000400
P PDC

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	
Reactor Coolant System.....	3/4 4-33
FIGURE 3.4-2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO ³ 4 EF PY.....	3/4 4-35
FIGURE 3.4-3 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO ³ 4 EF PY.....	3/4 4-36
TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE.....	3/4 4-37
TABLE 4.4-6 MAXIMUM HEATUP AND COOLDOWN RATES FOR MODES 4, 5 AND 6 (WITH REACTOR VESSEL HEAD ON).....	3/4 4-38
Pressurizer.....	3/4 4-39
Overpressure Protection Systems.....	3/4 4-40
FIGURE 3.4-4 MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE SYSTEM.....	3/4 4-41
3/4.4.10 STRUCTURAL INTEGRITY.....	3/4 4-43
3/4.4.11 REACTOR COOLANT SYSTEM VENTS.....	3/4 4-44
<u>3/4.5 EMERGENCY CORE COOLING SYSTEMS</u>	
3/4.5.1 ACCUMULATORS.....	3/4 5-1
3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F....	3/4 5-3
3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F.....	3/4 5-7
3/4.5.4 REFUELING WATER STORAGE TANK.....	3/4 5-9

10

11

12

13

14

15

16

17

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
TABLE B 3/4.4-1 REACTOR VESSEL TOUGHNESS.....	B 3/4 4-8
FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE ($E > 1\text{MeV}$) AS A FUNCTION OF FULL POWER SERVICE LIFE.....	B 3/4 4-9
FIGURE B 3/4.4-2 EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F..... (DELETED)	B 3/4 4-10
3/4.4.10 STRUCTURAL INTEGRITY.....	B 3/4 4-15
3/4.4.11 REACTOR COOLANT SYSTEM VENTS.....	B 3/4 4-15
<u>3/4.5 EMERGENCY CORE COOLING SYSTEMS</u>	
3/4.5.1 ACCUMULATORS.....	B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS.....	B 3/4 5-1
3/4.5.4 REFUELING WATER STORAGE TANK.....	B 3/4 5-2
<u>3/4.6 CONTAINMENT SYSTEMS</u>	
3/4.6.1 PRIMARY CONTAINMENT.....	B 3/4 6-1
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS.....	B 3/4 6-3
3/4.6.3 CONTAINMENT ISOLATION VALVES.....	B 3/4 6-4
3/4.6.4 COMBUSTIBLE GAS CONTROL.....	B 3/4 6-4
3/4.6.5 VACUUM RELIEF SYSTEM.....	B 3/4 6-4

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

- a. A flow path from the boric acid tank via either a boric acid transfer pump or a gravity feed connection and a charging/safety injection pump to the Reactor Coolant System if the boric acid tank in Specification 3.1.2.5a. or 3.1.2.6a. is OPERABLE, or
- b. The flow path from the refueling water storage tank via a charging/safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. or 3.1.2.6b. is OPERABLE.

APPLICABILITY: MODES 4*, 5*, and 6*.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

- a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header is greater than or equal to 65°F when a flow path from the boric acid tank is used, and
- b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

325

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging/safety injection pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4*, 5*#, and 6*#.

ACTION:

With no charging/safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

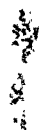
4.1.2.3.1 The above required charging/safety injection pump shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the reactor coolant system and reactor coolant pump seals, that a differential pressure across the pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging/safety injection pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable** at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position, ^{each pump's} prior to the temperature of one or more of the RCS cold legs decreasing below 325°F and at least once per 31 days thereafter, except when the reactor vessel head is removed.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

**An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

#For periods of no more than 1 hour, when swapping pumps, it is permitted that there be no OPERABLE charging/safety injection pump. No CORE ALTERATIONS or positive reactivity changes are permitted during this time.



REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

- a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**
- b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**
- c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**
- d. RHR Loop [A], or
- e. RHR Loop [B].

APPLICABILITY: MODE 4.

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; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.
- b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 38°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

325

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

- a. One additional RHR loop shall be OPERABLE**, or
- b. The narrow range secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

- a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.
- b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The narrow range secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures. (325)

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

- a. With one or more 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); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With one PORV inoperable as a result of causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve.
- c. With two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- d. With all three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.
- e. With one or more block valve(s) inoperable, within 1 hour:
(1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and
(2) apply the ACTION b., c. or d. above, as appropriate, for the isolated PORV(s).
- f. The provisions of Specification 3.0.4 are not applicable.

* MODE 4 when the temperature of all RCS cold legs is greater than 325°F.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, ~~criticality~~, and inservice leak and hydrostatic testing with:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period, and
- c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

if the pressure and temperature limit lines shown on Figures 3.4-2 and 3.4-3 were exceeded,

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, ~~criticality~~, and inservice leak and hydrostatic testing with:

- a. A maximum heatup rate as shown on Table 4.4-6.
- b. A maximum cooldown rate as shown on Table 4.4-6.
- c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

if the pressure and temperature limit lines shown on Figures 3.4-2 and 3.4-3 were exceeded,

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted.

REPLACE WITH ATTACHED INSERT

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLAIN METAL
COPPER CONTENT : 0.10%
PHOSPHORUS CONTENT : 0.005%
NOT INITIAL : 50°F
NOT AFTER 4 EFPY : 1/4T, 15%
3/4T, 13%

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR
THE SERVICE PERIOD UP TO 4 EFPY AND CONTAINS MARGINS
OF 10% AND 50 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

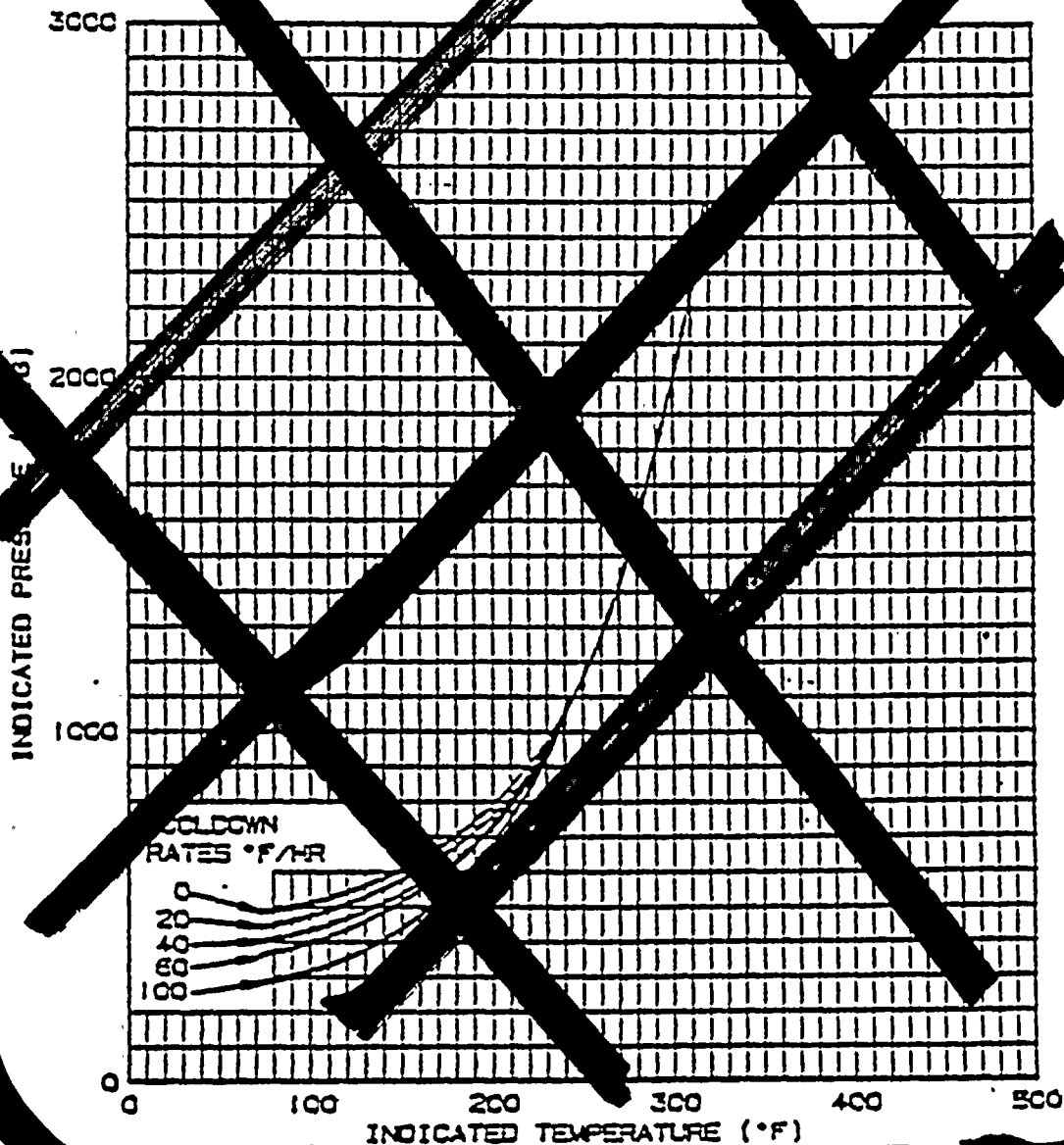


FIGURE 3.4-2

REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS - APPLICABLE UP TO 4 EFPY

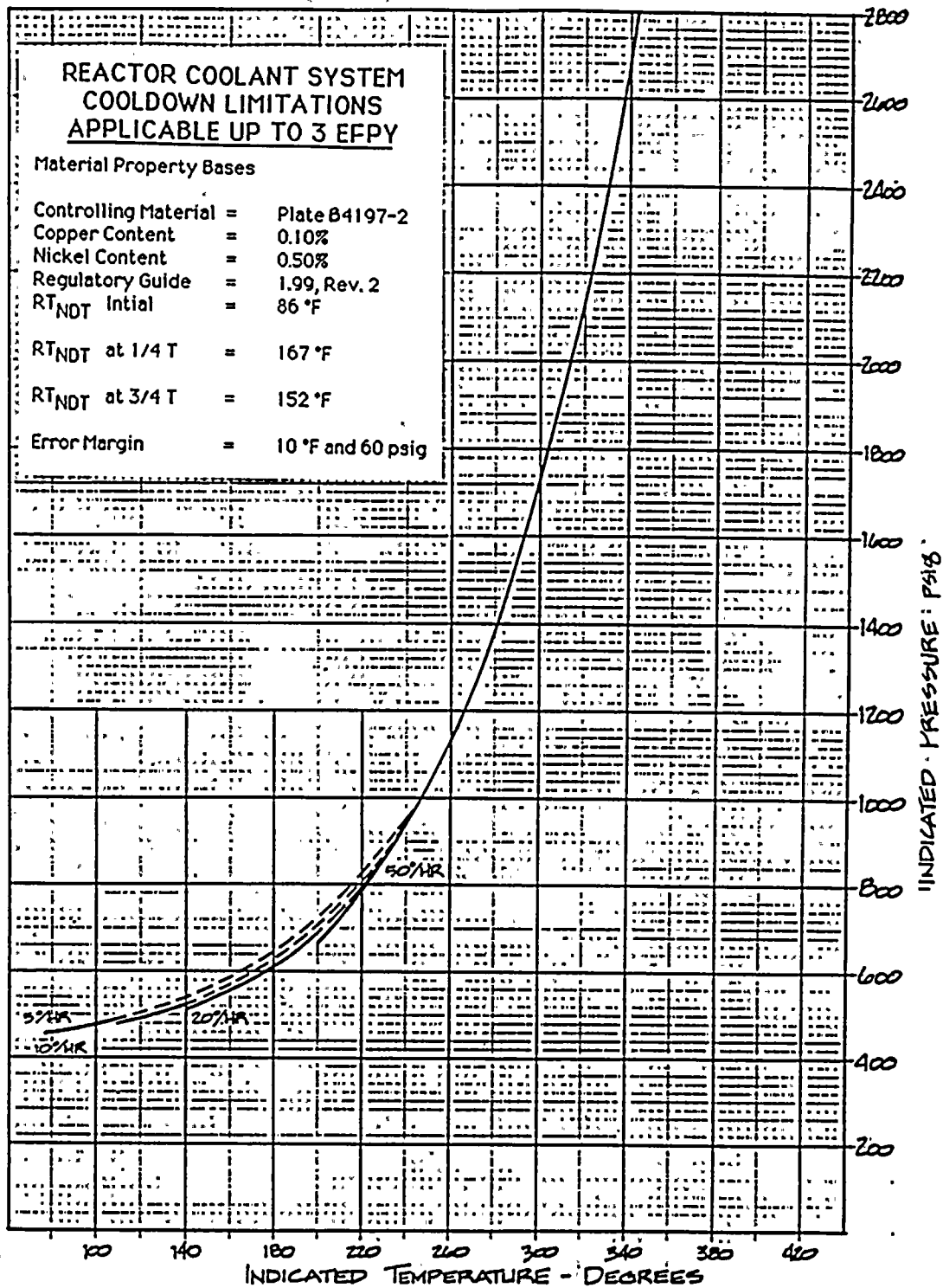


FIGURE 3.4-2

REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS - APPLICABLE UP TO 3 EFY

100

100

100

100

100

REPLACE WITH ATTACHED INSERT

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: PLATE METAL
 COPPER CONTENT: 0.10 WT%
 PHOSPHORUS CONTENT: 0.006 WT%
 RT NOT INITIAL: 1/2
 RT NOT AFTER 4 EPY: 3/4

CURVE APPLICABLE FOR HEATUP RATE UP TO 100°F/HR FOR
 SERVICE PERIOD UP TO 4 EPY. CONTAINS MARGINS
 OF 10°F AND 60 PSIG FOR POSSIBLE MEASUREMENT ERRORS

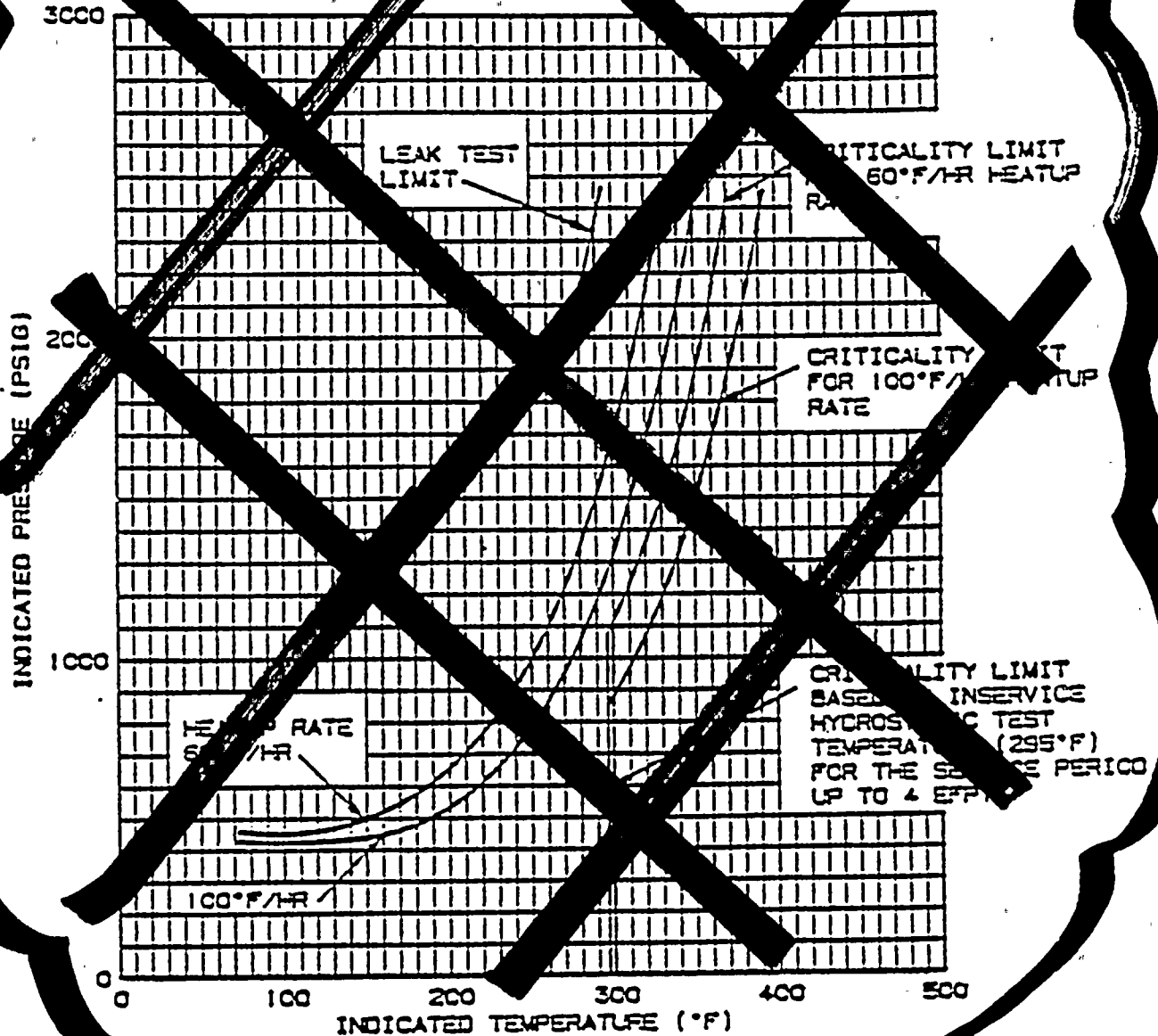


FIGURE 3.4-3

REACTOR COOLANT SYSTEM
 HEATUP LIMITATIONS - APPLICABLE UP TO 4 EPY

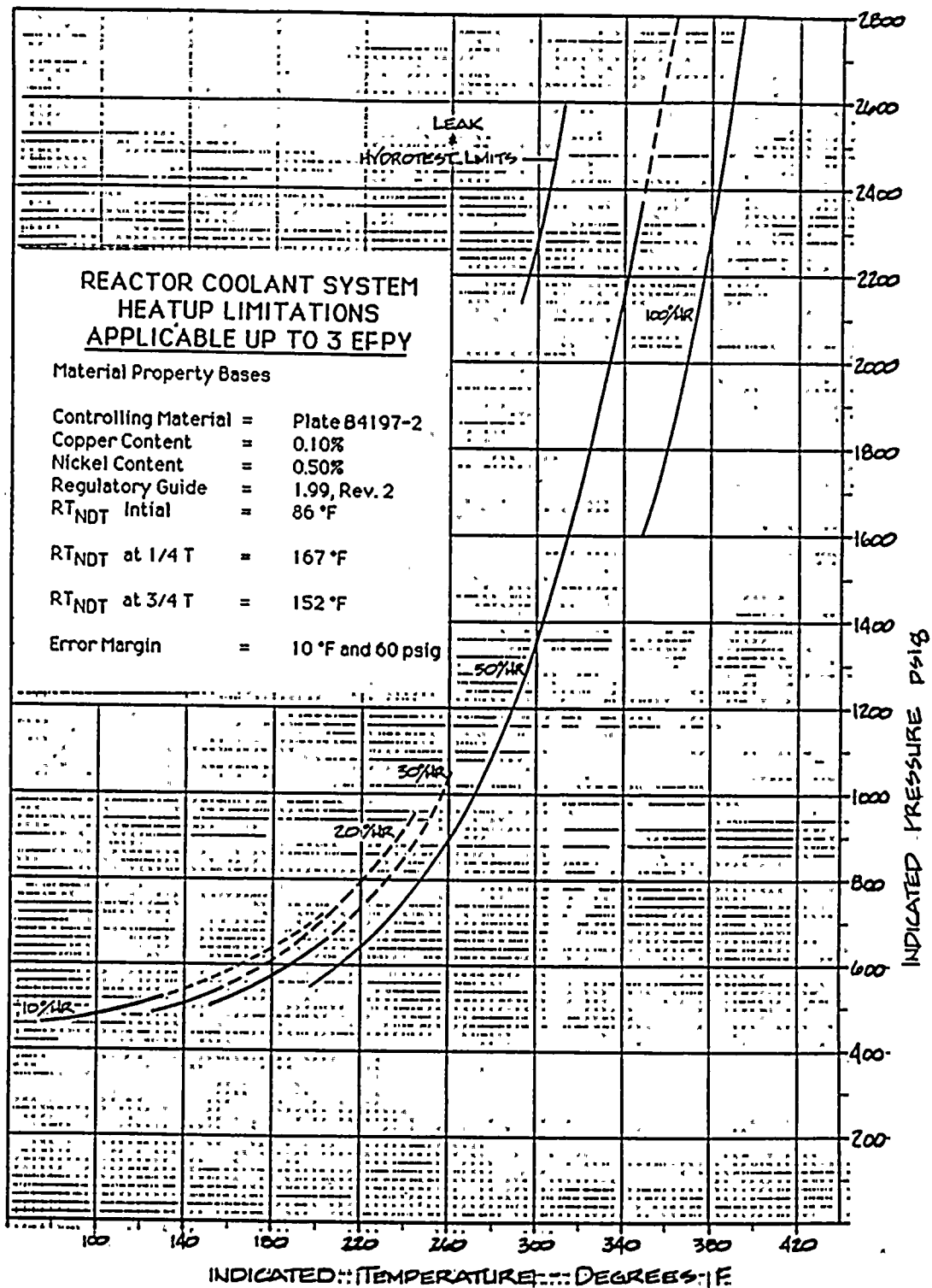


FIGURE 3.4-3

REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS - APPLICABLE UP TO 3 EFY

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

<u>TEMPERATURE*</u>	<u>COOLDOWN IN ANY 1 HOUR PERIOD</u> **
350-200°F	50°F
200-140°F (140)	20°F
< 140°F (110)	5°F
140-110°F	10°F

HEATUP RATES

<u>TEMPERATURE*</u>	<u>HEATUP IN ANY 1 HOUR PERIOD</u> **
< 135°F (135)	10°F
135-155°F	20°F
155-200°F	30°F
200-350°F	50°F

** Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

*Temperature range used should be based on the lowest RCS cold leg value.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

- a. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4, or
- b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

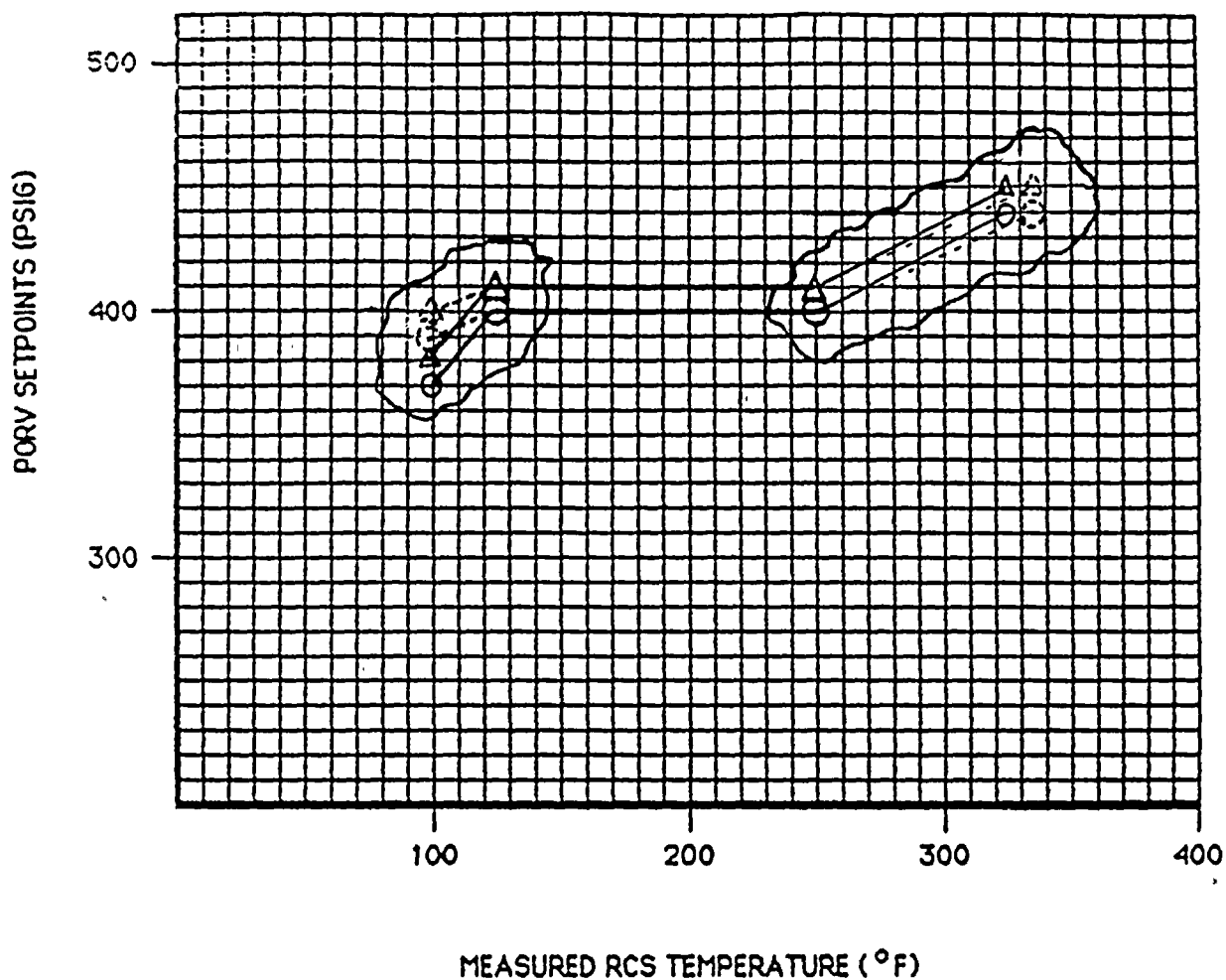
ACTION:

- a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.
- b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.
- c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.



RCS TEMP OF	LOW PORV * PSIG ○	HIGH PORV * PSIG △
< 100	390 370	400 380
125	400	410
250	400	410
300	400 427	425 437
335 325	440	450

* VALUES BASED ON ³EFPPY REACTOR VESSEL DATA AND
CONTAINS MARGINS OF ⁻¹⁶ -16°F AND +60 PSIG FOR POSSIBLE
INSTRUMENT ERROR

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW
TEMPERATURE OVERPRESSURE SYSTEM

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

- a. One OPERABLE charging/safety injection pump,*
- b. One OPERABLE RHR heat exchanger,
- c. One OPERABLE RHR pump, and
- d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

- a. With no ECCS subsystem OPERABLE because of the inoperability of either the charging/safety injection pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.
- b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.
- c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to

325°F.
325

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 ^(Deleted) ~~All charging/safety injection pumps, except the above allowed OPERABLE pump, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position prior to the temperature of one or more of the RCS cold legs decreasing below 335°F and at least once per 21 days thereafter.~~

~~*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with the power removed from the valve operator or by a manual valve secured in the closed position.~~

REACTIVITY CONTROL SYSTEMS

BASES

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below ~~255~~³²⁵°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7100 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2000-2200 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified value, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values. The specified percent level and gallons differ by less than 0.3%.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 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.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e.; by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

325 The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 335°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

REACTOR COOLANT SYSTEM

BASES

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and ^{and H}cooldown are limited to be consistent with the requirements given in the ASME (Boiler and Pressure Vessel Code, Section III, Appendix G, and 10 CFR 50 Appendix G). 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 cooldown and heatup limitations shown in Figures 3.4-2 and 3.4-3 and Table 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:
 - a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

- b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
2. These limit lines shall be calculated periodically using methods provided below,
 3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,
 4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and
 5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} , at the end of ⁽³⁾ effective full power years (EFPY) of service life. The ~~4 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, based upon the fluence, copper content, and ^(nickel)phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by ~~either~~ Regulatory Guide 1.99, Revision 2, ⁽²⁾"Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" ~~or the Westinghouse~~.

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS

							<u>AVG. SHELF ENERGY</u>	
<u>COMPONENT</u>	<u>GRADE</u>	<u>HEAT NO</u>	<u>CU %</u>	<u>P (%)</u>	<u>T_{NDT} (°F)</u>	<u>RT_{NDT} (°F)</u>	<u>MWD FT-LB</u>	<u>MMWD FT-LB</u>
Closure Hld. Dome	A533, D, CL1	A9213-1	-	-	-10	8	-	114
Head Flange	A508, CL2	5302-V2	-	-	0	0	-	135
Vessel Flange	"	5302-V1	-	-	-10	-8	-	110
Inlet Nozzle	"	438B-4	-	-	-20	-20	-	169
" "	"	438B-5	-	-	0	0	-	128
" "	"	438B-6	-	-	-20	-20	-	149
Outlet Nozzle	"	439B-4	-	-	-10	-10	-	151
" "	"	439B-5	-	-	-10	-10	-	152
" "	"	439B-6	-	-	-10	-10	-	150
Nozzle Shell	A533B, CL1	C0224-1	.12	.008	-20	-1	-	90
" "	"	C0123-1	.12	.006	0	42	-	84
Inter. Shell	"	A9153-1	.09	.007	-10	60	106	83
" "	"	B4197-2	.10	.006	-10	108 36	112	74
Lower Shell	"	C9924-1	.08	.005	-10	54	147	98
" "	"	C9924-2	.08	.005	-20	57	148	88
Bottom Hld. Torus	"	A9249-2	-	-	-40	14	-	94
" " Dome		A9213-2	-	-	-40	-8	-	125
Weld (Inter & Lower Shell Vertical Weld Seams)			.06	.013	-20	-20	-	>94
Weld (Inter. to Lower Shell Girth Seam)			.04	.013	-20	-20	-	88

REPLACE WITH ATTACHED INSERT

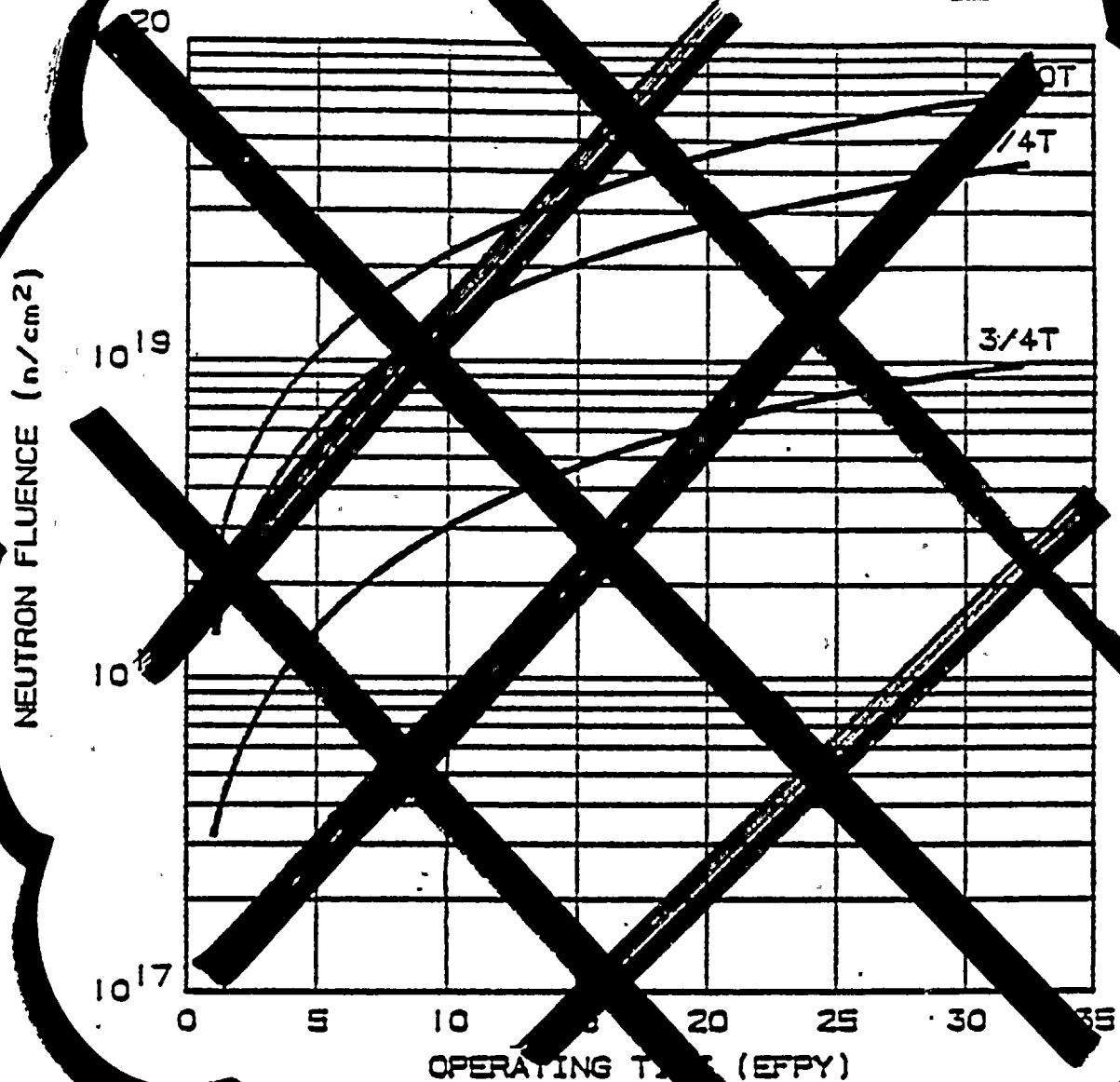


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE ($E > 1\text{MeV}$) AS A FUNCTION OF FULL POWER SERVICE LIFE

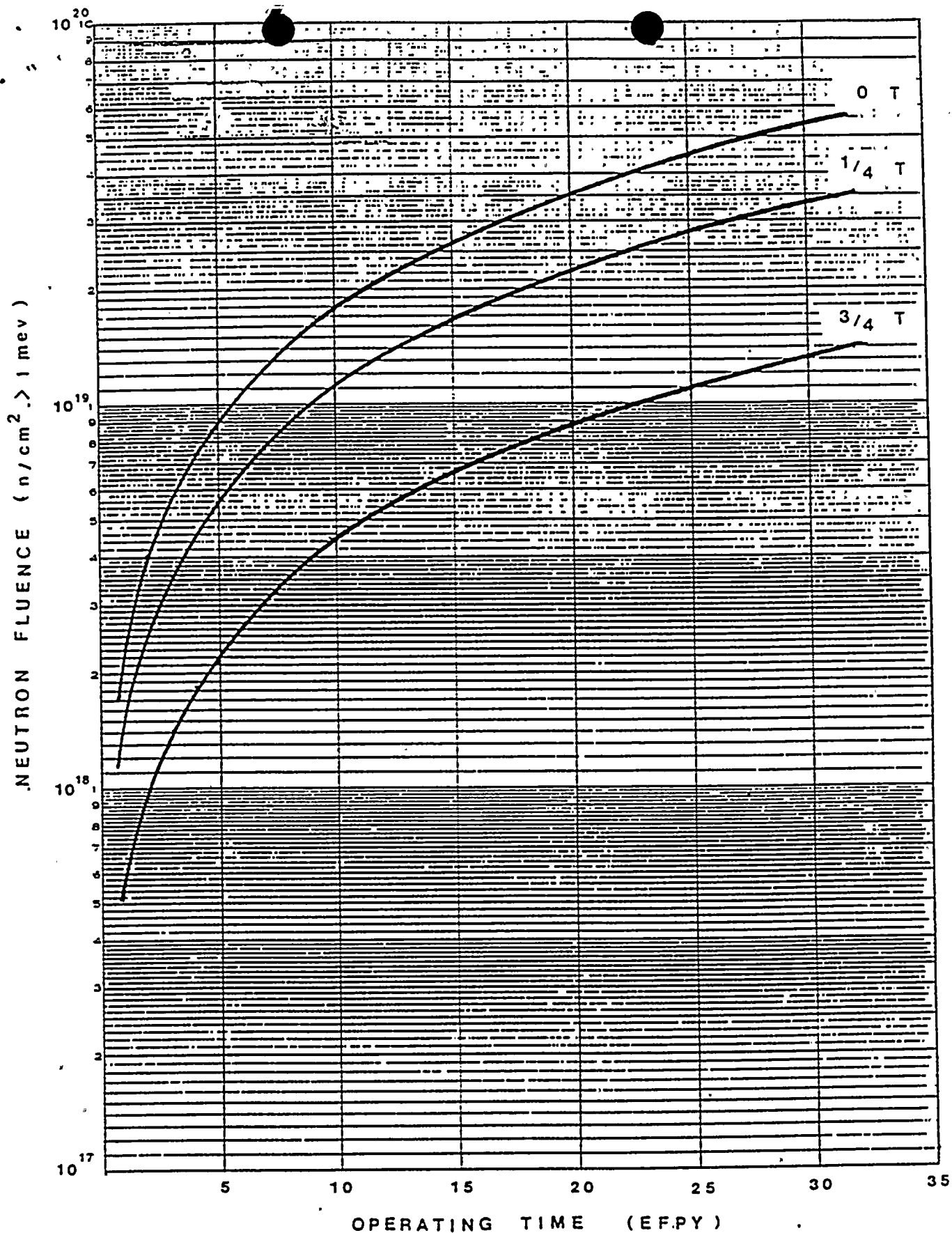


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE ($E > 1\text{MeV}$) AS A FUNCTION OF FULL POWER SERVICE LIFE

11

11

11

11

11

11

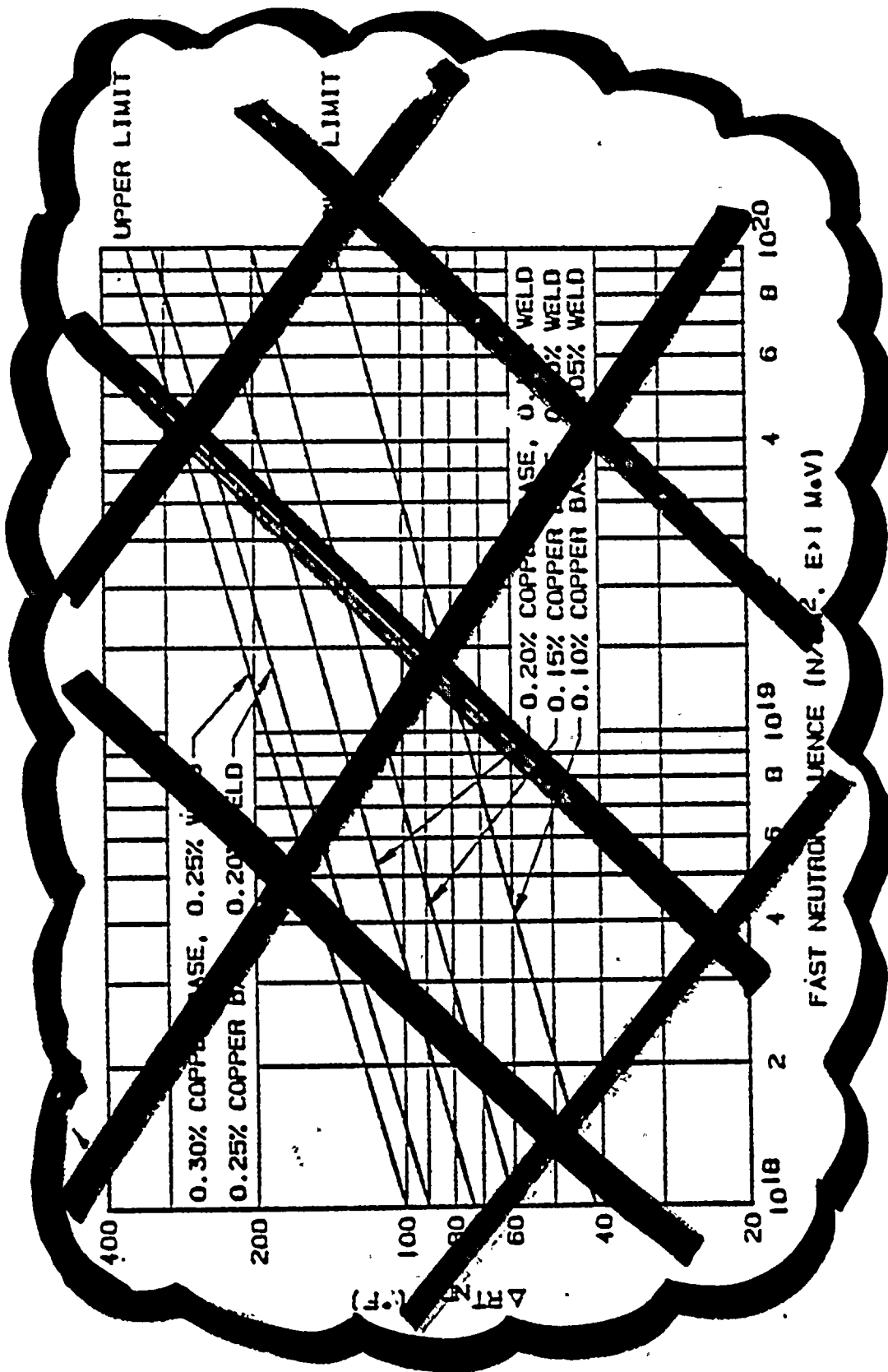


FIGURE B 3/4.4-2 (DELETED)

~~EFFECT OF TEMPERATURE AND COPPER ON CHANGE IN TEMPERATURE FOR REACTION VESSELS EXPOSED TO IRRADIATION AT 500°F~~

2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100

101

102

103
104
105
106
107
108
109
110
111
112
113
114
115
116
117
118
119
120
121
122
123
124
125
126
127
128
129
130
131
132
133
134
135
136
137
138
139
140
141
142
143
144
145
146
147
148
149
150
151
152
153
154
155
156
157
158
159
160
161
162
163
164
165
166
167
168
169
170
171
172
173
174
175
176
177
178
179
180
181
182
183
184
185
186
187
188
189
190
191
192
193
194
195
196
197
198
199
200

201

202

203
204
205
206
207
208
209
210
211
212
213
214
215
216
217
218
219
220
221
222
223
224
225
226
227
228
229
230
231
232
233
234
235
236
237
238
239
240
241
242
243
244
245
246
247
248
249
250
251
252
253
254
255
256
257
258
259
260
261
262
263
264
265
266
267
268
269
270
271
272
273
274
275
276
277
278
279
280
281
282
283
284
285
286
287
288
289
290
291
292
293
294
295
296
297
298
299
300

301

302

303
304
305
306
307
308
309
310
311
312
313
314
315
316
317
318
319
320
321
322
323
324
325
326
327
328
329
330
331
332
333
334
335
336
337
338
339
340
341
342
343
344
345
346
347
348
349
350
351
352
353
354
355
356
357
358
359
360
361
362
363
364
365
366
367
368
369
370
371
372
373
374
375
376
377
378
379
380
381
382
383
384
385
386
387
388
389
390
391
392
393
394
395
396
397
398
399
400

401

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

Copper Trend Curves shown in Figure B 3/4-4-2. The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 and Table 4.4-6 include predicted adjustments for this shift in RT_{NDT} at the end of 4 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

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 and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. 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 cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, 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 reference temperature, RT_{NDT} , is used and this includes the radiation-induced shift, ΔRT_{NDT} , corresponding to the end of the period for which cooldown and heatup 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

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.9 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 335°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of a charging/safety injection pump and its injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those

REACTOR COOLANT SYSTEM

BASES

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

assumed cannot occur, Technical Specifications ^(below 325°F) require lockout of all but one charging/safety injection pump while in MODES 4, 5, and 6 ~~(below 335°F)~~ with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensable gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The value of 66% indicated level ensures that a minimum of 7440 gallons is maintained in the accumulators. The maximum indicated level of 96% ensures that an adequate volume exists for nitrogen pressurization.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 AND 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one charging/safety injection pump to be OPERABLE and the Surveillance Requirement to verify one charging/safety injection pump OPERABLE below 335°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

325

