UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 12, 1998

50-423

Mr. Martin L. Bowling
Recovery Officer - Millstone
Unit No. 2
Northeast Nuclear Energy Company
c/o Ms. Patricia A. Loftus
Director - Regulatory Affairs
P. O. Box 128
Waterford, CT 06385

SUBJECT: ISSUANCE OF AMENDMENT - MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 (TAC NOS. M98035 AND M99502)

Dear Mr. Bowling:

The Commission has issued the enclosed Amendment No. 157 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3, in response to your application dated August 29, 1997, as supplemented by letters dated September 25 and November 14, 1997.

Based on a review and subsequent calculations of the cold overpressurization protection (COPS) enabling temperature and the emergency core cooling system (ECCS)/charging system mode 3 requirements, Northeast Nuclear Energy Company (NNECO) proposes to reduce the COPS enabling temperature. As a result, NNECO proposed the following Technical Specifications (TS) changes: add new heatup and cooldown pressure/temperature limit curves and their associated requirements; add new power operated relief valve (PORV) setpoint curves and their associated requirements; revise the reactor coolant loops and coolant circulation, ECCS, boration systems, and COPS to incorporate the lower enabling temperature and new restrictions for cold overpressure protection system, PORV undershoot, and residual heat removal (RHR) relief valve bellows; add a footnote to allow a reactor coolant pump to substitute for an RHR pump during heatup from Mode 5 to 4, which is consistent with the improved standard technical specification; reword TS 3/4.4.9.3 and its surveillance requirement section to be consistent with the improved STS; and revise the affected Bases sections to be consistent with the proposed changes.

By letter dated February 20, 1997, you submitted a change to TS Bases Section 3/4.4.9. As such, your TS contain updated TS Bases pages 3/4 4-14 and 3/4 4-15, and new TS Bases page 3/4 4-16. Since the NRC staff planned to act on the February 20, 1997, letter and this amendment request together, these three Bases pages have not been incorporated in the NRC's copy of the TS. In your August 29, 1997, TS amendment request, TS Bases Section 3/4.4.9 is replaced in its entirety; therefore, the NRC staff does not plan to act on your February 20, 1997, TS Bases Section 3/4.4.9 change.

DF01

CP-1

February 12, 1998

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

James W. Andersen, Project Manager
Special Projects Office - Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 157 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

James W. Andersen, Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 157 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 157
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated August 29, 1997, as supplemented by letters dated September 25 and November 14, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 157, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F McKee

Phillip F. McKee
Deputy Director for Licensing
Special Projects Office
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 12, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 157

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3/4 1-13	3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-15	3/4 1-15
3/4 1-16	3/4 1-16
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
-	3/4 4-5a
3/4 4-6	3/4 4-6
-	3/4 4-6a*
3/4 4-8	3/4 4-8
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-37	3/4 4-37
3/4 4-38	3/4 4-38
-	3/4 4-38a
3/4 4-39	3/4 4-39
3/4 4-40	3/4 4-40
3/4 4-41	3/4 4-41
3/4 5-7	3/4 5-7
3/4 5-8	3/4 5-8
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	B 3/4 4-12
B 3/4 4-13	B 3/4 4-13
B 3/4 4-14	B 3/4 4-14
B 3/4 4-15	B 3/4 4-15
-	B 3/4 4-16

Remove	Insert
-	B 3/4 4-17
-	B 3/4 4-18
-	B 3/4 4-19
-	B 3/4 4-20
-	B 3/4 4-21
-	B 3/4 4-22
-	B 3/4 4-23
-	B 3/4 4-24
-	B 3/4 4-25
-	B 3/4 4-26
-	B 3/4 4-27*
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
B 3/4 5-2a	B 3/4 5-2a*

*Overflow or new page - no change

Note: The NRC did not incorportate the licensee's February 20, 1997, Bases change in its copy of the TS, therefore, the NRC TS do not have a page B 3/4 4-16.

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 storage system via either a boric acid transfer pump or a gravity feed connection and a charging pump to the Reactor Coolant System if the boric acid storage system in Specification 3.1.2.5a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

a. With none of the above boron injection flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source in MODE 4, provide an OPERABLE flow path capable of being powered from an OPERABLE emergency power source within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With none of the above boron injection flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source in MODES 5 or 6, 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 Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°F when a flow path from the boric acid tanks 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage system via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits as shown in Figure 3.1-4 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°F when it is a required water source;

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,

c. At least once each REFUELING INTERVAL by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

d. At least once each REFUELING INTERVAL by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 33 gpm to the RCS.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging 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:

a. With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source in MODE 4, provide an OPERABLE charging pump capable of being powered from an OPERABLE emergency power source within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source in MODES 5 and 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that its developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limit as shown in Figure 3.1-4 at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump's developed head at the test flow point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

LIMITING CONDITION FOR OPERATION

3.4.1.3 Either:*,**

a. With Reactor Trip System breakers closed, at least two RCS loops shall be OPERABLE and in operation, or

b. With Reactor Trip System breakers open, at least two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops*** shall be OPERABLE, and at least one of these loops shall be in operation. For RCS loop(s) to be OPERABLE, at least one reactor coolant pump (RCP) shall be in operation.

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.

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*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 (RCP) shall not be started unless one of the following conditions is met:

a. At least one RCP is operating.
b. The secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
c. With a maximum of one RCS loop isolated and with the RHR relief valves isolated from the RCS, the secondary side water temperature of each steam generator, not isolated from the RCS, is less than or equal to 250°F.
d. All RCS wide range cold leg temperatures >275°F and no cold overpressure protection relief valves are in service as follows:
   1) COPPS is blocked or the PORV block valves are closed, and
   2) RHR relief valves are isolated from the RCS (3RHS*MV8701C or 3RHS*MV8701A is closed and 3RHS*MV8702B or 3RHS*MV8702C is closed).

***Prior to opening 3RHS*MV8701C and 3RHS*MV8701A, or 3RHS*MV8702B and 3RHS*MV8702C, all safety injection pumps and all but one centrifugal charging pump shall be incapable of injecting into the RCS. Surveillance Requirements 4.4.9.3.4 and 4.4.9.3.5 apply whenever any RHR relief valve is unisolated from the RCS.

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 secondary side water level of at least two steam generators shall be greater than 17%.

APPLICABILITY: MODE 5 with at least two reactor coolant loops filled***.

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*a. 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.

b. All RHR loops may be removed from operation during a planned heatup to MODE 4 when at least one RCS loop is OPERABLE and in operation and when two additional steam generators are OPERABLE as required by LCO 3.4.1.4.1.b.

**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. No reactor coolant pumps (RCPs) may be in operation below 160°F unless COPPS is blocked or unless the PORV block valves are closed.

b. An RCP shall not be started unless one of the following conditions is met:
1. At least one RCP is operating and the lowest RCS wide range cold leg temperature of the unisolated RCS loops is >160°F.
2. With two or more Reactor Coolant System (RCS) loops isolated, the first RCP shall not be started unless the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
3. With a maximum of one RCS loop isolated, with the RHR relief valves isolated from the RCS, and with the PORVs providing cold overpressure protection, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
4. With a maximum of one RCS loop isolated and with any RHR relief valve unisolated from the RCS, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 200°F and less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With less than the required RHR loop(s) OPERABLE or 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 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.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with less than two reactor coolant loops filled***.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status 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.

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

**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.

*** a. No reactor coolant pumps (RCPs) may be in operation below 160°F unless COPPS is blocked or unless the PORV block valves are closed.

b. An RCP shall not be started unless one of the following conditions is met:
1. At least one RCP is operating and the lowest RCS wide range cold leg temperature of the unisolated RCS loops is >160°F.
2. With two or more Reactor Coolant System (RCS) loops isolated, the first RCP shall not be started unless the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
3. With a maximum of one RCS loop isolated, with the RHR relief valves isolated from the RCS, and with the PORVs providing cold overpressure protection, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.
4. With a maximum of one RCS loop isolated and with any RHR relief valve unisolated from the RCS, the first RCP shall not be started until the secondary side water temperature of each steam generator not isolated from the RCS is less than or equal to 200°F and less than or equal to 50°F above the lowest RCS wide range cold leg temperature of the unisolated RCS loops.

COLD SHUTDOWN - LOOPS NOT FILLED

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 The required RHR loops shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

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

ISOLATED LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated with power removed from the associated RCS loop stop valve operators until:

a. The temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops,

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less,

c. All reactor coolant pumps are de-energized.

d. The isolated portion of the loop has been drained and is refilled, and

e. The reactor is subcritical by at least the value required by Specifications 3.1.1.1.2 or 3.1.1.2 for Mode 5 or Specification 3.9.1.1 for Mode 6.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the requirements of the above specification not satisfied, do not open the isolated loop stop valves.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least the value required by Specifications 3.1.1.1.2 or 3.1.1.2 for Mode 5 or Specification 3.9.1.1 for Mode 6 within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.3 Within 4 hours prior to opening the loop stop valves, the isolated loop shall be determined to:

a. Be drained and refilled, and

b. Have a boron concentration greater than or equal to the boron concentration of the operating loops, or greater than 2600 ppm whichever is less.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 The reactor coolant system (except the pressurizer) temperature and pressure shall be limited as follows:

a. During an RCS heatup, the heatup limits of Figure 3.4-2 apply with the additional restriction that only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is ≤160°F.

b. During an RCS cooldown, the limits of Figure 3.4-3 apply with the additional restriction that only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is ≤160°F and no reactor coolant pump may be operated when the lowest unisolated RCS loop wide range cold leg temperature is ≤120°F.

c. During steady state conditions, when the maximum temperature increase or decrease in any one hour period is <10°F and when the plant is not changing temperatures in accordance with a heatup or cooldown procedure, only one reactor coolant pump can be operating when the lowest unisolated RCS loop wide range cold leg temperature is ≤160°F. The limits of Figures 3.4-2 and 3.4-3 do not apply during steady state conditions.

d. During RCS inservice leak and hydrostatic testing operations, the Hydrostatic and Leak Test limit of Figure 3.4-2 apply with the additional restrictions that within a one-hour period prior to exceeding the heatup curve, and during each one-hour period above the heatup curve, a maximum temperature increase or decrease of 5°F in any one-hour period is allowed.

APPLICABILITY: At all times.

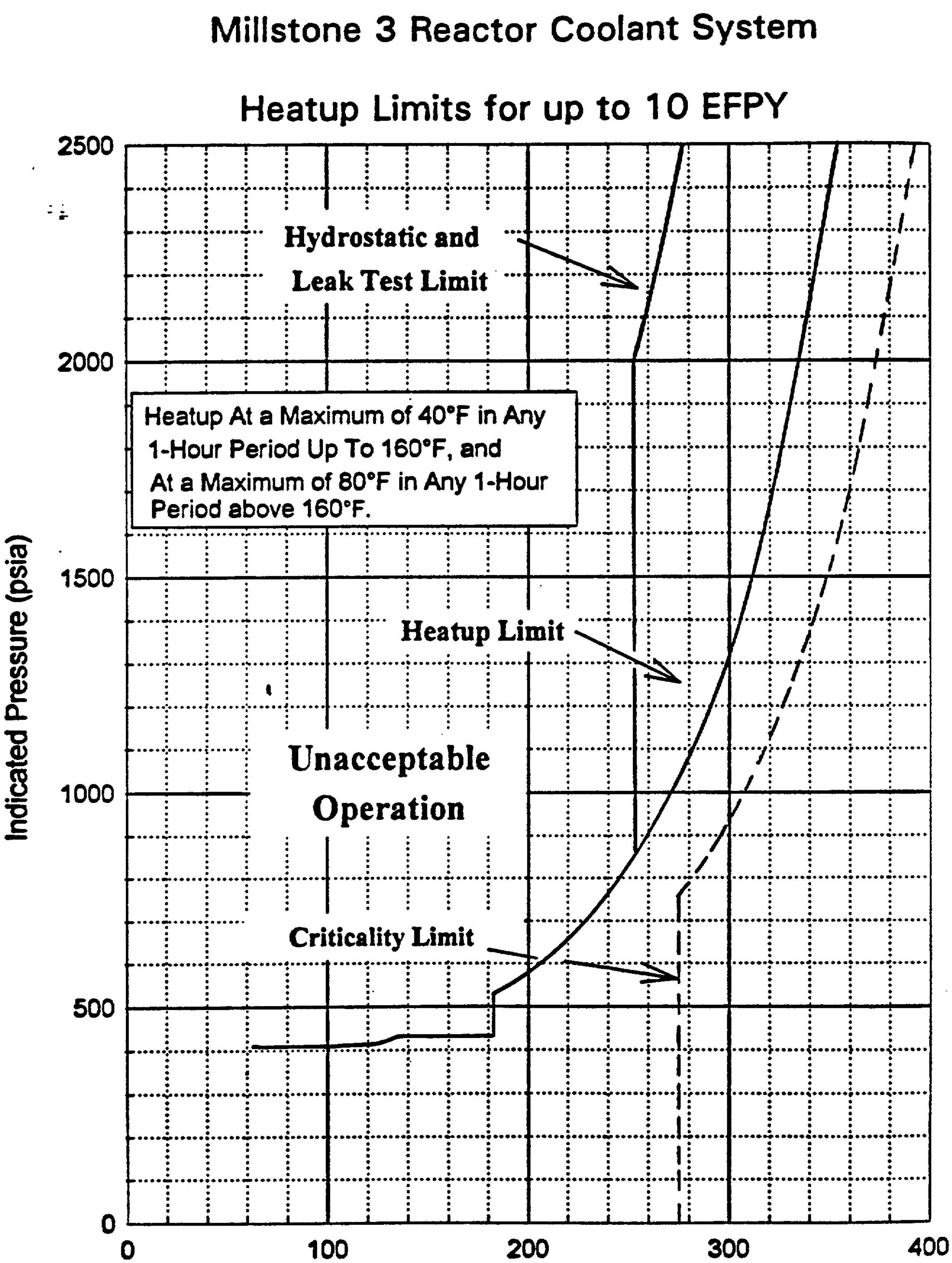
ACTION:

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 within 72 hours 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 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3 as required.



Millstone 3 Reactor Coolant System
Heatup Limits for up to 10 EFPY
Hydrostatic and Leak Test Limit
Heatup At a Maximum of 40°F in Any 1-Hour Period Up To 160°F, and At a Maximum of 80°F in Any 1-Hour Period above 160°F.
Heatup Limit
Unacceptable Operation
Criticality Limit
Indicated Pressure (psia)
Indicated Cold Leg Temperature (°F)
0
500
1000
1500
2000
2500
100
200
300
400

FIGURE 3.4-2

Millstone 3 Reactor Coolant System

Cooldown Limitations for up to 10 EFPY

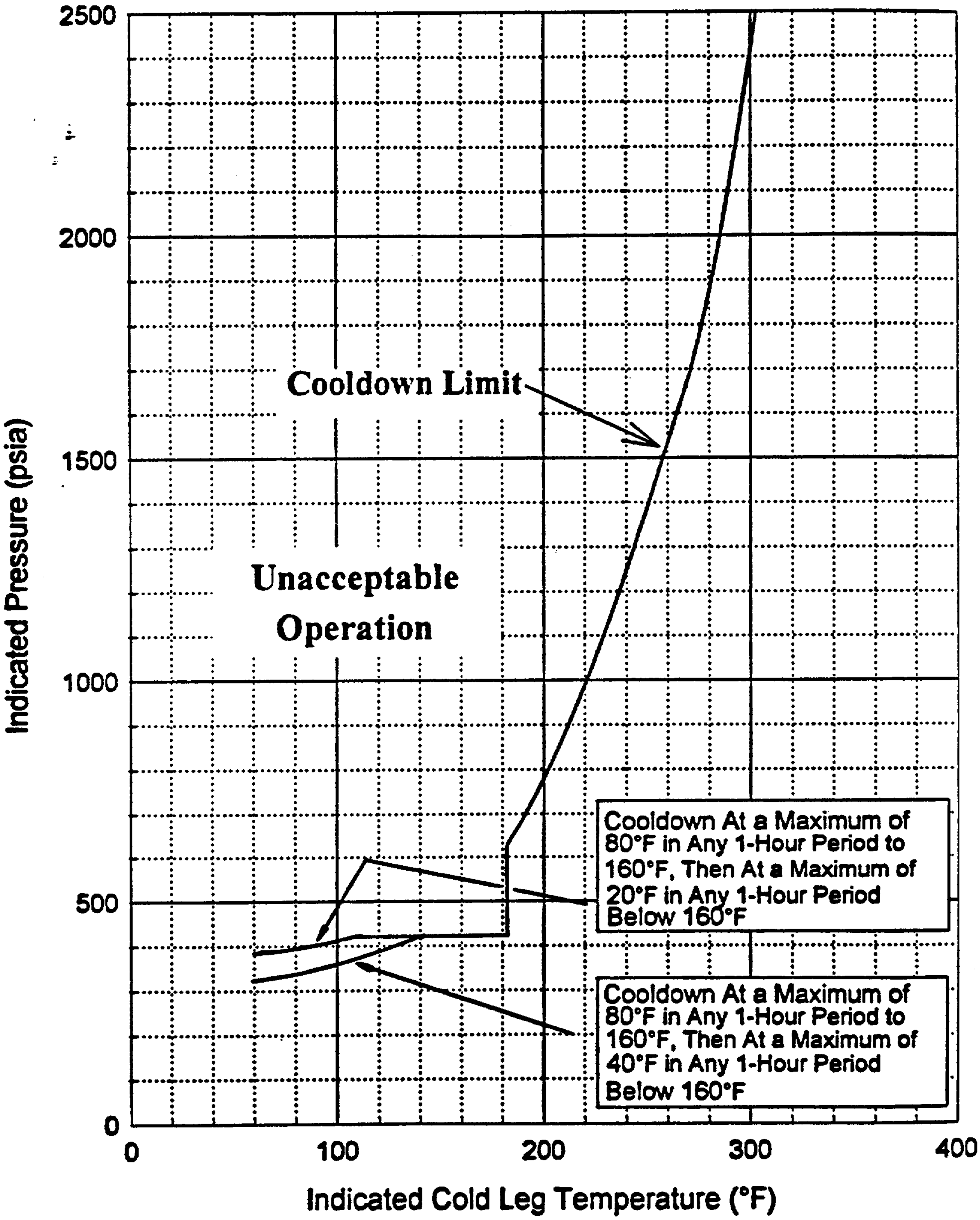


Figure 3.4-3

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	APPROXIMATE WITHDRAWAL TIME (EFPY)
U	58.5°	3.98(a)	First Refueling (1.3 EFPY actual)
Y	241°	3.74	9
V	61°	3.74	16
W	121.5°	4.01	STANDBY
X	238.5°	4.01	STANDBY
Z	301.5°	4.01	STANDBY

a) Plant specific evaluation.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psia within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.9.3 Cold Overpressure Protection shall be OPERABLE with a maximum of one centrifugal charging pump* and no Safety Injection pumps capable of injecting into the Reactor Coolant System (RCS) and one of the following pressure relief capabilities:

1. One power operated relief valve (PORV) with a nominal lift setting established in Figure 3.4-4a and one PORV with a nominal lift setting established in Figure 3.4-4b, or

2. Two residual heat removal (RHR) suction relief valves with setpoints $\geq$ 426.8 psig and $\leq$ 453.2 psig, or

3. One PORV with a nominal lift setting established in Figure 3.4-4a or Figure 3.4-4b and one RHR suction relief valve with a setpoint $\geq$ 426.8 psig and $\leq$ 453.2 psig, or

4. RCS depressurized with an RCS vent of $\geq$ 5.4 square inches.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is $\leq$ 275°F**, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With two or more centrifugal charging pumps capable of injecting into the RCS, immediately initiate action to establish that a maximum of one centrifugal charging pump is capable of injecting into the RCS.

b. With any Safety Injection pump capable of injecting into the RCS, immediately initiate action to establish that no Safety Injection pumps are capable of injecting into the RCS.

c. With one required relief valve inoperable in MODE 4, restore the required relief valve to OPERABLE status within 7 days, or depressurize and vent the RCS through at least a 5.4 square inch vent within the next 8 hours.

---

*Two centrifugal charging pumps may be capable of injecting into the RCS for less than one hour, during pump swap operations. However, at no time will two charging pumps be simultaneously out of pull-to-lock during pump swap operations.

**When an RHR suction relief valve or a PORV, which is armed for COPPS, is unisolated from the RCS, and when the RCS cold leg temperature is greater than 275°F, a maximum of one centrifugal charging pump and no safety injection pumps shall be capable of injecting into the RCS.

LIMITING CONDITION FOR OPERATION

d. With one required relief valve inoperable in MODE 5 or 6, restore the required relief valve to OPERABLE status within 24 hours, or depressurize the RCS and establish an RCS vent of ≥ 5.4 square inches within the next 8 hours.

e. With two required relief valves inoperable and with no RCS vent ≥ 5.4 square inches, depressurize the RCS and establish an RCS vent of ≥ 5.4 square inches within 8 hours.

f. In the event the PORVs, the RHR suction relief valves, or the RCS vent 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, the RHR suction relief valves, or RCS vent on the transient, and any corrective action necessary to prevent recurrence.

g. Entry into an OPERATIONAL MODE is permitted while subject to these ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Demonstrate that each required PORV is 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 each REFUELING INTERVAL; and

c. Verifying the PORV block valve is open and the PORV Cold Overpressure Protection System (COPPS) is armed at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Demonstrate that each required RHR suction relief valve is OPERABLE by:

a. Verifying the isolation valves between the RCS and each required RHR suction relief valve are open at least once per 12 hours; and

b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway $\geq$ 5.4 square inches at least once per 31 days for a passive vent path and at least once per 12 hours for unlocked open vent valves.

4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS at least once per 12 hours.

4.4.9.3.5 Verify that a maximum of one centrifugal charging pump is capable of injecting into the RCS at least once per 12 hours.

HIGH SETPOINT PORV CURVE
FOR THE COLD OVERPRESSURE PROTECTION SYSTEM

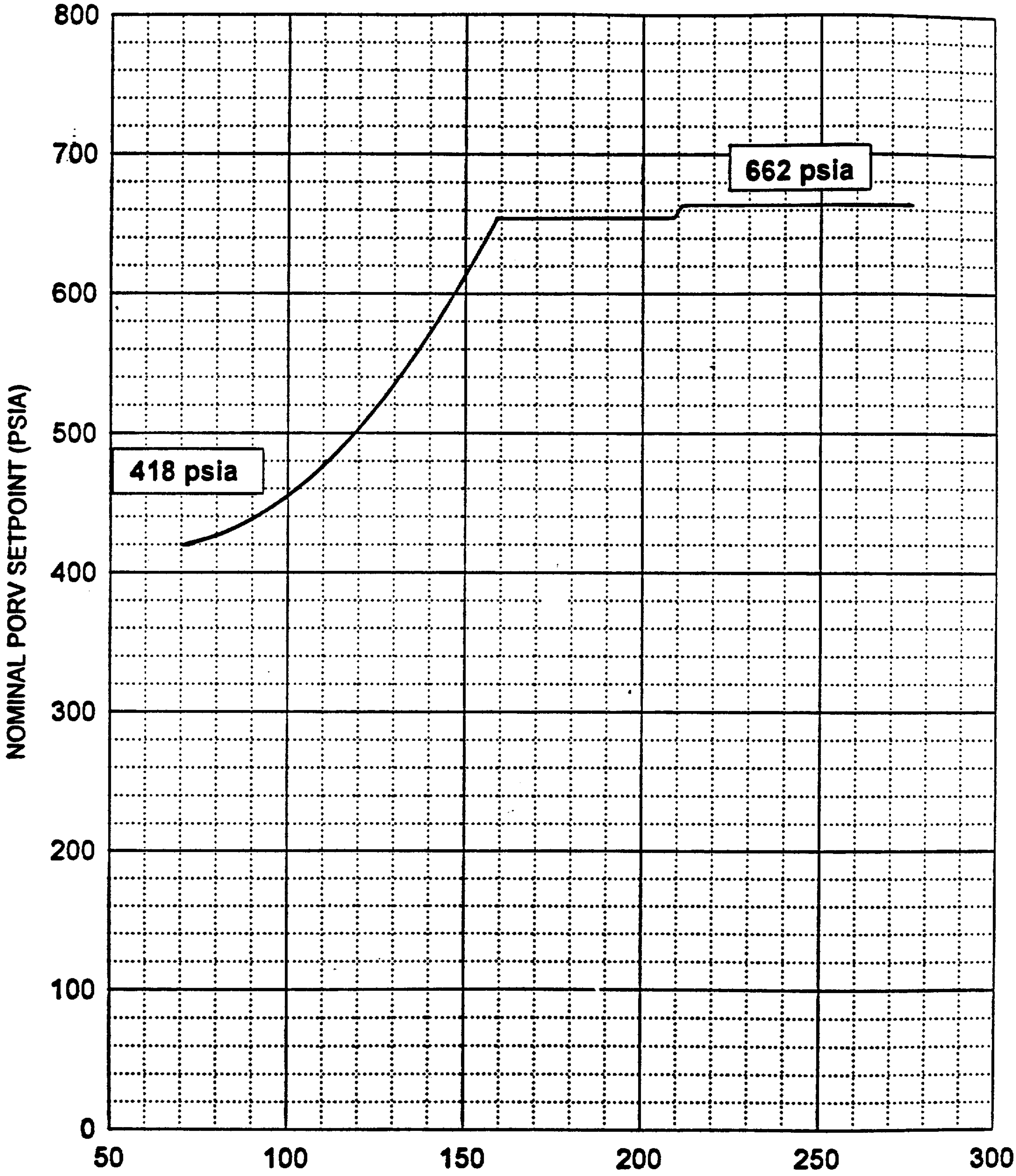


FIGURE 3.4-4a

LOW SETPOINT PORV CURV[illegible]
FOR THE COLD OVERPRESSURE PROTECTION SYSTEM

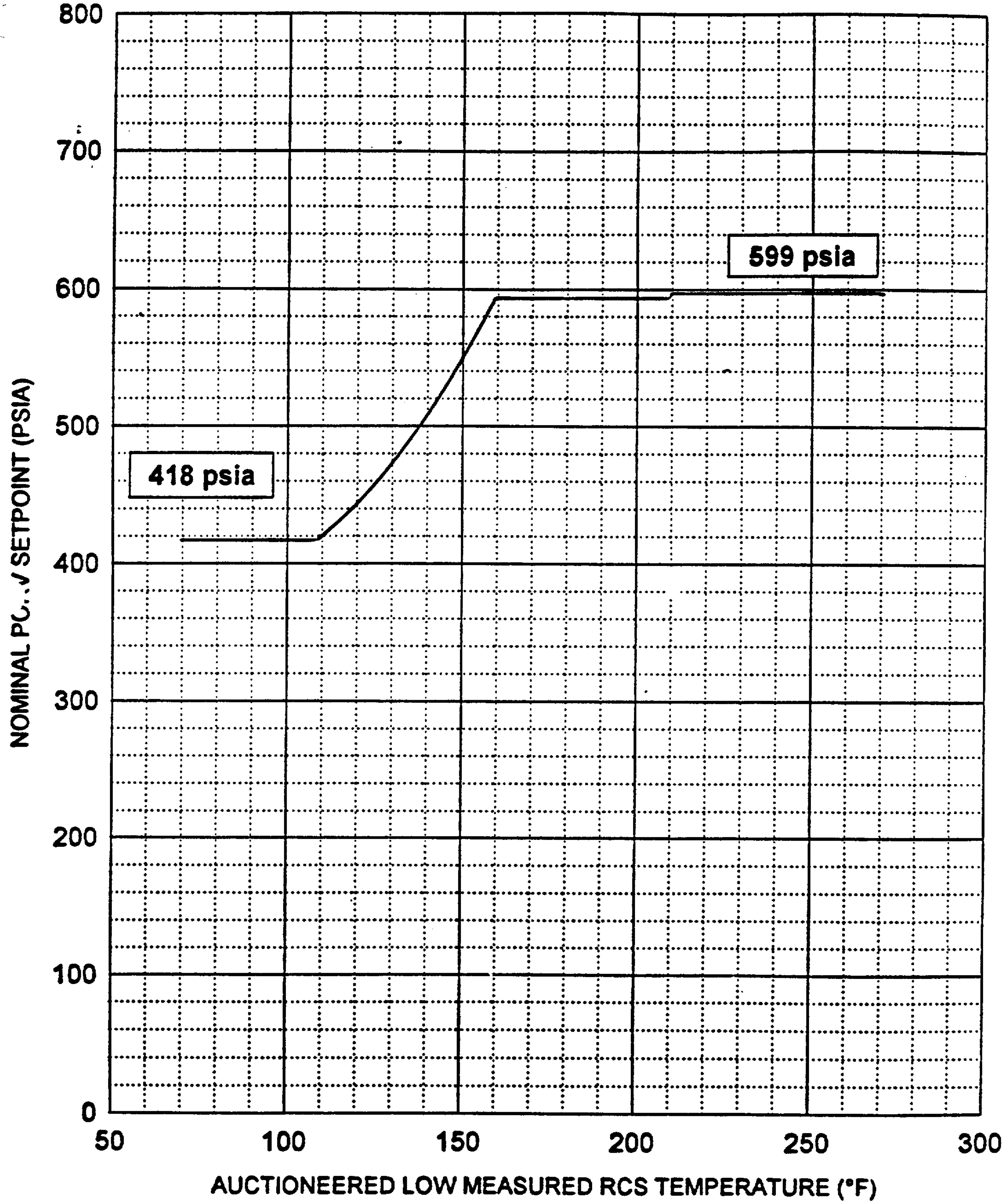


FIGURE 3.4-4b

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 centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump,

d. One OPERABLE containment recirculation heat exchanger,

e. One OPERABLE containment recirculation pump, and

f. An OPERABLE flow path which, with manual realignment of valves, is capable of discharging to the RCS, taking suction from the refueling water storage tank, 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 the centrifugal charging pump, the containment recirculation pump, the containment recirculation heat exchanger, the flow path from the refueling water storage tank, or the flow path capable of taking suction from the containment sump, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 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.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2, with the exception that valves may be out of alignment but capable of being manually realigned.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

These corrections involved: (1) a conversion of the MDC used in the FSAR safety analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR safety analyses into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative MTC value at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making corrections for burnup and soluble boron to the limiting EOL MTC value.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the P-12 interlock is above its setpoint, (4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (5) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the plant in MODES 1, 2, or 3, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions equivalent to that required by Figure 3.1-5 after xenon decay and cooldown to 200°F. The maximum boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The initial RCS boron concentration is based on a minimum expected hot full power or hot zero power condition (peak xenon). The final RCS boron concentration assumes that the most reactive control rod is not inserted into the core. This set of conditions requires a minimum usable volume of 21,802 gallons of 6600 ppm borated water from the boric acid storage tanks or 1,166,000 gallons of 2700 ppm borated water from the refueling water storage tank (RWST). A minimum RWST volume of 1,166,000 gallons is specified to be consistent with ECCS requirement.

With the plant in MODE 4, one boron injection flowpath is acceptable without single failure consideration for emergency boration requirements on the basis of the stable reactivity condition of the reactor, the emergency power supply requirement for the OPERABLE charging pump, and the fact that the plant is administratively borated to at least MODE 5 requirements prior to cooldown to MODE 4. Also, the primary grade water addition path to the charging pumps is surveilled to be locked closed to prevent a direct dilution accident in MODE 4.

With the plant in MODES 5 and 6, 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 one centrifugal charging pump to be OPERABLE, when cold overpressure protection is in service, provides assurance that a mass addition pressure transient can be relieved by operation of a single PORV or RHR suction relief valve.

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 a usable volume of 4100 gallons of 6600 ppm borated water from the boric acid storage tanks or 250,000 gallons of 2700 ppm borated water from the RWST. The unusable volume in each boric acid storage tank is 1300 gallons.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 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.

The minimum RWST solution temperature for MODES 5 and 6 is based on analysis assumptions in addition to freeze protection considerations. The minimum/maximum RWST solution temperatures for MODES 1, 2, 3 and 4 are based on analysis assumptions.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate in MODES 1 and 2 with three or four reactor coolant loops in operation and maintain DNBR greater than the design limit during all normal operations and anticipated transients. With less than the required reactor coolant loops in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, three reactor coolant loops, and in Mode 4, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, in MODE 3 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.

In MODE 4, if a bank withdrawal accident can be prevented, 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 (any combination of RHR or RCS) be OPERABLE.

In MODE 5, with reactor coolant loops filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two RHR loops or at least one RHR loop and two steam generators 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.

In MODE 5, during a planned heatup to MODE 4 with all RHR loops removed from operation, an RCS loop, OPERABLE and in operation, meets the requirements of an OPERABLE and operating RHR loop to circulate reactor coolant. During the heatup there is no requirement for heat removal capability so the OPERABLE and operating RCS loop meets all of the required functions for the heatup condition. Since failure of the RCS loop, which is OPERABLE and operating, could also cause the associated steam generator to be inoperable, the associated steam generator cannot be used as one of the steam generators used to meet the requirement of LCO 3.4.1.4.1.b.

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.

The restrictions on starting an RCP 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 or which could cause pressure excursions within the RHR system which would exceed the design pressure of the system. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs based upon the secondary water temperature of each steam generator and the RCS wide-range cold leg temperatures.

The requirement to maintain the isolated loop stop valves shut with power removed ensures that no reactivity addition to the core could occur due to the startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. The 2600 ppm is sufficient to bound shutdown margin requirements and provide for boron concentration measurement uncertainty between the loop and the RWST. Draining and refilling the isolated loop within 4 hours prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

The requirement to have all reactor coolant pumps de-energized, prior to unisolating a loop, insures that the heat from the secondary side of the steam generator, in the loop being unisolated, does not result in an energy addition transient during the return of the loop to service.

SPECIFIC ACTIVITY (Continued)

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 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

REACTOR COOLANT SYSTEM (EXCEPT THE PRESSURIZER)

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The LCO and Figures 3.4-2 and 3.4-3 contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational requirements during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. A heatup or cooldown is defined as a temperature increase or decrease of greater than or equal to 10°F in any one hour period. This definition of heatup and cooldown is based upon the ASME definition of isothermal conditions described in ASME, Section XI, Appendix E.

PRESSURE/TEMPERATURE LIMITS (continued)

Steady state thermal conditions exist when temperature increases or decreases are <10°F in any one hour period and when the plant is not performing a planned heatup or cooldown in accordance with a procedure. During steady state thermal conditions, the limits of the heatup and cooldown curves do not apply. Cold overpressure protection is adequate to protect the reactor coolant system.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the Pressurizer, which has different design characteristics and operating functions which are addressed by LCO 3.4.9.2, "Pressurizer".

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 5).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations may be more restrictive, and thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The P/T limits include uncertainty margins to ensure that the calculated limits are not inadvertently exceeded. These margins include gauge and system loop uncertainties, elevation differences, containment pressure conditions and system pressure drops between the beltline region of the vessel and the pressure gauge or relief valve location. In an effort to minimize the system frictional losses, additional restrictions on RCP operation below 160°F are provided in the LCO. These restrictions result in increased acceptable system pressures enabling greater operator flexibility during heatup and cooldown in MODE 5.

PRESSURE/TEMPERATURE LIMITS (continued)

The criticality limit curve includes the Reference 1 requirement that it be $\geq$ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSIS

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 2 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the Pressurizer. These limits define allowable operating regions while providing margin against nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curve. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

PRESSURE/TEMPERATURE LIMITS (continued)

Violating the LCO limits places the reactor vessel outside of the bounds of the analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10CFR50, Appendix G (Ref. 1). The P/T limits were developed to provide requirements for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, in keeping with the concern for nonductile failure. The limits do not apply to the Pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.2.5, "DNB Parameters"; LCO 3.2.3.1 and 3.2.3.2, "RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - Four Loops Operating/Three Loops Operating"; LCO 3.1.1.4, "Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The 30 minute Allowed Outage Time (AOT) reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

PRESSURE/TEMPERATURE LIMITS (continued)

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour AOT is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

This evaluation must be completed whenever a limit is exceeded. Restoration within 30 minutes alone is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, action must be implemented to reduce pressure and temperature as specified in the ACTION statement.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in the Action statement. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psia within the next 30 hours.

The AOTs are reasonable, based on operating experience to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Verification that operation is within the LCO limits as well as the limits of Figures 3.4-2 and 3.4-3 is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This frequency is considered reasonable in view of the control room indication available to monitor RCS status.

PRESSURE/TEMPERATURE LIMITS (continued)

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This Surveillance Requirement is only required to be performed during system heatup, cooldown, and ISLH testing. No Surveillance Requirement is given for criticality operations because LCO 3.1.1.4 contains a more restrictive requirement.

The Surveillance Requirement to remove and examine the reactor vessel material irradiation surveillance specimens is in accordance with the requirements of 10CFR50, Appendix H.

REFERENCES

1. 10CFR50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10CFR50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

PRESSURIZER

BACKGROUND

The Pressurizer is part of the RCPB, but is not subject to the same restrictions as the rest of the RCS. This LCO limits the temperature changes of the Pressurizer and allowable temperature differentials, within the design assumptions and the stress limits for cyclic operation.

PRESSURIZER (continued)

The LCO contains the Pressurizer limits for heatup, cooldown, and spray water temperature differential. Each temperature limit defines an acceptable region for normal operation. The limits that apply to the Pressurizer are as follows: The Pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the Pressurizer and the spray fluid is greater than 320°F.

The heatup limit represents a different set of restrictions than the cooldown limit because the directions of the thermal gradients through the Pressurizer wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The consequence of violating the LCO limits is that the Pressurizer has been operated under conditions that can result in failure, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the Pressurizer.

APPLICABLE SAFETY ANALYSIS

The Pressurizer temperature limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering temperature and temperature rate of change conditions that might cause the initiation/propagation of undetected cracks and cause failure of the pressure boundary.

LCO

The two elements of this LCO are:

a. Limits on the rate of change of temperature; and

b. Limits on the spray water differential temperature.

The LCO limits apply to the Pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the Pressurizer wall and, therefore, restricts stresses caused by thermal gradients.

Violating the LCO limits places the Pressurizer outside of the bounds of the stress analyses. The consequences depend on several factors, as follow:

a. The severity of the rate of change of temperature;

PRESSURIZER (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the Pressurizer walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the Pressurizer material.

APPLICABILITY

The Pressurizer temperature limits LCO provides a definition of acceptable operation for prevention of failure. The temperature limits were developed to provide requirements for operation during heatup or cooldown, and their Applicability is at all times in keeping with the concern for failure.

ACTIONS

Operation outside the temperature limits must be corrected so that the Pressurizer is returned to a condition that has been verified by stress analyses. The 30 minute AOT reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if Pressurizer operation can continue. The evaluation must verify the Pressurizer integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour AOT is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

This evaluation must be completed whenever a limit is exceeded. Restoration within 30 minutes alone is insufficient because higher than analyzed stresses may have occurred and may have affected the Pressurizer integrity.

If the required remedial actions are not completed within the allowed times, the plant must be placed in a lower MODE because a sufficiently severe event may have caused entry into an unacceptable region. This possibility indicates a need for more careful examination of the event, best accomplished with the Pressurizer at reduced pressure. In reduced pressure conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, action must be implemented to reduce pressure as specified in the ACTION statement.

PRESSURIZER (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure as specified in the Action statement. A favorable evaluation must be completed and documented before returning to operating pressure conditions.

Pressure is reduced by bringing the plant to MODE 3 within 6 hours. Pressure is further reduced by bringing the plant to MODE 4 or 5 and reducing Pressurizer pressure < 500 psia within the next 30 hours.

The AOTs are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Verification that operation is within the LCO heatup and cooldown limits is required every 30 minutes when Pressurizer temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor Pressurizer status. Surveillance for heatup or cooldown may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied. The Surveillance Requirement for heatup or cooldown is only required to be performed during system heatup and cooldown.

Verification that operation is within the LCO spray water temperature differential limit is required every 12 hours when auxiliary spray is in operation. This Frequency is considered reasonable in view of the control room indication available to monitor Pressurizer status.

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The Cold Overpressure Protection System limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10CFR50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection.

Cold Overpressure Protection consists of two PORVs with nominal lift setting as specified in Figures 3.4-4a and 3.4-4b, or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two relief valves are required for redundancy. One relief valve has adequate relieving capability to prevent overpressurization of the RCS for the required mass input capability.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause nonductile cracking of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the limits provided in Figures 3.4-2 and 3.4-3.

This LCO provides RCS overpressure protection by limiting mass input capability and requiring adequate pressure relief capacity. Limiting mass input capability requires all Safety Injection (SIH) pumps and all but one centrifugal charging pump to be incapable of injection into the RCS. The pressure relief capacity requires either two redundant relief valves or a depressurized RCS and an RCS vent of sufficient size. One relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum mass input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the Cold Overpressure Protection MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve.

PORV Requirements

As designed, the PORV Cold Overpressure Protection (COPPS) is signaled to open if the RCS pressure approaches a limit determined by the COPPS actuation logic. The COPPS actuation logic monitors both RCS temperature and RCS pressure and determines when the nominal setpoint of Figure 3.4-4a or Figure 3.4-4b is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure setpoint for that temperature. The calculated pressure setpoint is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

Figure 3.4-4a and Figure 3.4-4b present the PORV setpoints for COPPS. Above 110°F, the setpoints are staggered so only one valve opens during a low

OVERPRESSURE PROTECTION SYSTEMS (continued)

temperature overpressure transient. Setting both valves to the values of Figure 3.4-4a and Figure 3.4-4b within the tolerance allowed for the calibration accuracy, ensures that the Reference 1 limits will not be exceeded for the analyzed isothermal events.

When a PORV is opened, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

The isolation valves between the RCS and the RHR suction relief valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valves with setpoint tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

When the RHR system is operated for decay heat removal or low pressure letdown control, the isolation valves between the RCS and the RHR suction relief valves are open, and the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting mass or heat input transient, and maintaining pressure below the P/T limits for the analyzed isothermal events.

For an RCS vent to meet the flow capacity requirement, it requires removing a Pressurizer safety valve, removing a PORV and disabling its block valve in the open position, removing a Pressurizer manway, or similarly establishing a vent by opening an RCS vent valve provided that the opening meets the size requirements. The vent path must be above the level of reactor coolant, so as not to drain the RCS when open.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABLE SAFETY ANALYSIS

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits for the analyzed isothermal events. In MODES 1, 2, AND 3, and in MODE 4, with RCS cold leg temperature exceeding 275°F, the pressurizer safety valves will provide RCS overpressure protection in the ductile region. At 275°F and below, overpressure prevention is provided by two means: (1) two OPERABLE relief valves, or (2) a depressurized RCS with a sufficiently sized RCS vent, as required by NUREG-0800, RSB 5-2 for temperatures less than RT_{NDT} + 90°F. Each of these means has a limited overpressure relief capability.

The required RCS temperature for a given pressure increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the Technical Specification curves are revised, the cold overpressure protection must be re-evaluated to ensure its functional requirements continue to be met using the RCS relief valve method or the depressurized and vented RCS condition.

Transients capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch

Heat Input Transients

a. Inadvertent actuation of Pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The Technical Specifications ensure that mass input transients beyond the operability of the cold overpressure protection means do not occur by rendering all Safety Injection Pumps and all but one centrifugal charging pump incapable of injecting into the RCS whenever an RHR suction relief valve is unisolated from the RCS or whenever any PORV has COPPS armed and its block valve open.

The Technical Specifications ensure that energy addition transients beyond the operability of the cold overpressure protection means do not occur by limiting reactor coolant pump starts. LCO 3.4.1.4.1, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Filled," LCO 3.4.1.4.2, "Reactor Coolant

OVERPRESSURE PROTECTION SYSTEMS (continued)

Loops and Coolant Circulation - Cold Shutdown - Loops Not Filled," and LCO 3.4.1.3, "Reactor Coolant Loops and Coolant Circulation - Hot Shutdown" limit reactor coolant pump starts to one of the following plant conditions:

a. An RCP is running, and
The wide range cold leg temperature of any unisolated RCS loop is >160°F, or

b. Two or more RCS loops are isolated, and
An RCP is not running, and
The secondary side water temperature of any steam generator in an unisolated loop is equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or

c. No more than one RCS loop is isolated, and
An RCP is not running, and
Any RHR suction relief valve is unisolated from the RCS, and
The secondary side water temperature of any steam generator in an unisolated loop is either:

- >200°F and equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or
- ≤200°F and ≤50°F hotter than the wide range cold leg temperature of any unisolated RCS loop. (Note: Reactor coolant pumps cannot be run with the wide range cold leg temperature of any unisolated RCS loop <160°F if any PORV has COPPS armed and has its block valve open.), or

d. No more than one RCS loop is isolated, and
An RCP is not running, and
The RHR suction relief valves are isolated from the RCS, and
The wide range cold leg temperature of any unisolated RCS loop ≥160°F, and
Any PORV has COPPS armed and has its block valve open, and
The secondary side water temperature of any steam generator in an unisolated loop is either:

- equal to or less than the wide range cold leg temperature of any unisolated RCS loop, or
- <250°F and ≤50°F hotter than the wide range cold leg temperature of any unisolated RCS loop,

or

e. The RHR suction relief valves are isolated from the RCS, and
Both PORVs are isolated or COPPS is blocked, and
The wide range cold leg temperature of any unisolated RCS loop is >275°F.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The cold overpressure transient analyses demonstrate that either one relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when RCS letdown is isolated and only one centrifugal charging pump is operating. Thus, the LCO allows only one centrifugal charging pump capable of injecting when cold overpressure protection is required.

The cold overpressure protection enabling temperature is conservatively established at a value $\geq$ 275°F based on the criteria described in Branch Technical Position RSB 5-2 provided in the Standard Review Plan (NUREG-0800).

PORV Performance

The 10CFR50 Appendix G analyses show that the vessel is protected against non-ductile failure when the PORVs are set to open at the values shown in Figures 3.4-4a and 3.4-4b within the tolerance allowed for the calibration accuracy. The curves are derived by analyses that model the performance of the PORV cold overpressure protection system (COPPS), assuming the limiting mass and heat transients of one centrifugal charging pump injecting into the RCS, or the energy addition as a result of starting an RCP with temperature asymmetry between the RCS and the steam generators. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times.

The PORV setpoints in Figures 3.4-4a and 3.4-4b will be updated when the P/T limits conflict with the cold overpressure analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement. Revised limits are determined using neutron fluence projections and the results of testing of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.9.1, "Pressure/Temperature Limits - Reactor Coolant System (Except the Pressurizer)," discuss these evaluations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints as do the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 426.8 psig and 453.2 psig will pass flow greater than that required for the limiting cold overpressure transient while maintaining RCS pressure less than the isothermal P/T limit curve. Assuming maximum relief flow requirements during the limiting cold overpressure event, an RHR suction relief valve will maintain RCS pressure to $\leq$ 110% of the nominal lift setpoint.

Although each RHR suction relief valve is a passive spring loaded device, which meets single failure criteria, its location within the RHR System precludes meeting single failure criteria when spurious RHR suction isolation valve or RHR suction valve closure is postulated. Thus the loss of an RHR suction relief

OVERPRESSURE PROTECTION SYSTEMS (continued)

valve is the worst case single failure. Also, as the RCS P/T limits are revised to reflect change in toughness in the reactor vessel materials, the RHR suction relief valve's analyses must be re-evaluated to ensure continued accommodation of the design bases cold overpressure transients.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of ≥ 5.4 square inches is capable of mitigating the allowed cold overpressure transient. The capacity of this vent size is greater than the flow of the limiting transient, while maintaining RCS pressure less than the maximum pressure on the isothermal P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the isothermal P/T limit curves are revised.

The RCS vent is a passive device and is not subject to active failure.

The RCS vent satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

RCP Seal Protection

As described above, the analyses of the cold overpressure transients result in pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The valve overshoots are considered in the generation of the PORV setpoints presented in Figures 3.4-4a and 3.4-4b.

The valve undershoots are also evaluated in terms of potential damage to the RCP #1 seal. The minimum pressure, considering valve undershoot, must be higher than that required to maintain the RCP #1 seal as a film riding seal. This requirement resulted in restrictions on the operation of pumps when the cold overpressure protection is being provided by one or two PORVs. Specifically,

a. When the RCS cold leg temperature of any unisolated loop is less than 160 degrees F, the PORV block valves are open, and the PORV's Cold Overpressure Protection System (COPPS) is armed, no RCPs may be in operation. LCO 3.4.1.4.1, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Filled," and LCO 3.4.1.4.2, "Reactor Coolant Loops and Coolant Circulation - Cold Shutdown - Loops Not Filled," provide this protection.

b. When COPPS is armed, with the steam generator secondary side ≥ 250°F, heat injection transients due to the start of the first RCP with a temperature asymmetry between the RCS and the steam generators is prohibited. LCO 3.4.1.3, "Reactor Coolant System - Hot Shutdown," provides this protection for MODE 4.

OVERPRESSURE PROTECTION SYSTEMS (continued)

c. When COPPS is armed, PORV undershoot is analyzed for mass injection transients limited to one charging pump. LCO 3.4.9.3, "Reactor Coolant System - Overpressure Protection Systems," provides this protection by requiring both safety injection pumps and all but one charging pump to be incapable of injection into the RCS.

In order to provide protection for the RCP #1 seal, a PORV setpoint of ≥ 595 psia for temperatures ≥ 160 degrees F must be met. This minimum setpoint is derived by adding the applicable train uncertainty and valve undershoot to the required minimum RCS pressure required for seal integrity. Due to the differing instrument uncertainties for the two trains of PORV COPPS, the train with the highest uncertainty is paired to the high setpoint curve.

LCO

This LCO requires that cold overpressure protection be OPERABLE and the maximum mass input be limited to one charging pump. Failure to meet this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 isothermal limits as a result of an operational transient.

To limit the mass input capability, the LCO requires a maximum of one centrifugal charging pump capable of injecting into the RCS.

The elements of the LCO that provides low temperature overpressure mitigation through pressure relief are:

1. Two OPERABLE PORVs; or

   A PORV is OPERABLE for cold overpressure protection when its block valve is open, its lift setpoint is set to the nominal setpoints provided by Figure 3.4-4a or 3.4-4b and when the surveillance requirements are met.

2. Two OPERABLE RHR suction relief valves; or

   An RHR suction relief valve is OPERABLE for cold overpressure protection when its isolation valves from the RCS are open and when its setpoint is at or between 426.8 psig and 453.2 psig, as verified by required testing.

3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or

4. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 5.4 square inches.

Each of these methods of ovepressure prevention is capable of mitigating the limiting cold overpressure transient.

OVERPRESSURE PROTECTION SYSTEMS (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 275°F, in MODE 5, and in MODE 6 when the head is on the reactor vessel. The Pressurizer safety valves provide RCS overpressure protection in the ductile region (i.e. > 275°F). When the reactor head is off, overpressurization cannot occur.

LCO 3.4.9.1 "Pressure/Temperature Limits" provides the operational P/T limits for all MODES. LCO 3.4.2.2, "Safety Valves - Operating," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and LCO 3.4.2.1, "Safety Valves - Shutdown," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODE 4.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a rapid increase in RCS pressure when little or no time exists for operator action to mitigate the event.

ACTIONS

a. and b.

With two or more centrifugal charging pumps capable of injecting into the RCS, or with any SIH pump capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted mass input capability to the RCS reflects the urgency of removing the RCS from this condition.

Required Action a. is modified by a Note that permits two centrifugal charging pumps capable of RCS injection for ≤ 1 hour to allow for pump swaps. This is a controlled evolution of short duration and the procedure prevents having two charging pumps simultaneously out of pull-to-lock while both charging pumps are capable of injecting into the RCS.

c.

In MODE 4 when any RCS cold leg temperature is ≤ 275°F, with one required relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within an allowed outage time (AOT) of 7 days. Two relief valves in any combination of the PORVs and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

OVERPRESSURE PROTECTION SYSTEMS (continued)

The AOT in MODE 4 considers the facts that only one of the relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. The RCS must be depressurized and a vent must be established within the following 8 hours if the required relief valve is not restored to OPERABLE within the required AOT of 7 days.

d.

The consequences of operational events that will overpressure the RCS are more severe at lower temperatures (Ref. 7). Thus, with one of the two required relief valves inoperable in MODE 5 or in MODE 6 with the head on, the AOT to restore two valves to OPERABLE status is 24 hours.

The AOT represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE relief valve to protect against overpressure events. The RCS must be depressurized and a vent must be established within the following 8 hours if the required relief valve is not restored to OPERABLE within the required AOT of 24 hours.

e.

The RCS must be depressurized and a vent must be established within 8 hours when both required Cold Overpressure Protection relief valves are inoperable.

The vent must be sized $\geq$ 5.4 square inches to ensure that the flow capacity is greater than that required for the worst case cold overpressure transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible non-ductile failure of the reactor vessel.

The time required to place the plant in this Condition is based on the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1

Performance of an ANALOG CHANNEL OPERATIONAL TEST is required within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The ANALOG CHANNEL OPERATIONAL TEST will verify the setpoint in accordance with the nominal values given in Figures 3.4-4a and 3.4-4b. PORV actuation could depressurize the RCS; therefore, valve operation is not required.

OVERPRESSURE PROTECTION SYSTEMS (continued)

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required once each REFUELING INTERVAL to adjust the channel so that it responds and the valve opens within the required range and accuracy to a known input.

The PORV block valve must be verified open and COPPS must be verified armed every 72 hours to provide a flow path and a cold overpressure protection actuation circuit for each required PORV to perform its function when required. The valve is remotely verified open in the main control room. This Surveillance is performed if credit is being taken for the PORV to satisfy the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the open position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure transient.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify the PORV block valve remains open.

4.4.9.3.2

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying the RHR suction valves, 3RHS*MV8701A and 3RHS*M8701C, are open when suction relief valve 3RHS*RV8708A is being used to meet the LCO and by verifying the RHR suction valves, 3RHS*MV8702B and 3RHS*MV8702C, are open when suction relief valve 3RHS*RV8708B is being used to meet the LCO. Each required RHR suction relief valve shall also be demonstrated OPERABLE by testing it in accordance with 4.0.5. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction valves are verified to be open every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

The ASME Code, Section XI (Ref. 8), test per 4.0.5 verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

OVERPRESSURE PROTECTION SYSTEMS (continued)

4.4.9.3.3

The RCS vent of ≥ 5.4 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent valve that cannot be locked open.

b. Once every 31 days for a valve that is locked, sealed, or secured in position or any other passive vent path. A removed Pressurizer safety valve fits this category.

This passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO.

4.4.9.3.4 and 4.4.9.3.5

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SIH pumps and all but one centrifugal charging pump are verified incapable of injecting into the RCS.

The SIH pumps and charging pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. Alternate methods of control may be employed using at least two independent means to prevent an injection into the RCS. This may be accomplished through any of the following methods: 1) placing the pump in pull to lock (PTL) and pulling its UC fuses, 2) placing the pump in pull to lock (PTL) and closing the pump discharge valve(s) to the injection line, 3) closing the pump discharge valve(s) to the injection line and either removing power from the valve operator(s) or locking manual valves closed, and 4) closing the valve(s) from the injection source and either removing power from the valve operator(s) or locking manual valves closed.

An SIH pump may be energized for testing or for filling the Accumulators provided it is incapable of injecting into the RCS.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

REFERENCES

1. 10CFR50, Appendix G
2. Generic Letter 88-11
3. ASME, Boiler and Pressure Vessel Code, Section III
4. FSAR, Chapter 15
5. 10CFR50, Section 50.46
6. 10CFR50, Appendix K
7. Generic Letter 90-06
8. ASME, Boiler and Pressure Vessel Code, Section XI

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).

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, 80 Edition and Addenda through Winter.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible 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 reactor vessel head vent path consists of two parallel flow paths with redundant isolation valves (3RCS*SV8095A, 3RCS*SV8096A and 3RCS*SV8095B, 3RCS*SV80965B) in each flow path. The pressurizer steam space vent path consists of two parallel paths with a power operated relief valve (PORV) and PORV block valve in series (3RCS*PCV455A, 3RCS*MV800A and 3RCS*PCV456, 3RCS*MV8000B).

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.

3/4.5.1 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 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 and with some valves out of normal injection lineup, on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

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.

Surveillance Requirement 4.5.2.b.1 requires verifying that the ECCS piping is full of water except for the operating centrifugal charging pump(s) and associated piping, and the RSS pump, the RSS heat exchanger and associated RSS piping. The ECCS pumps are normally in a standby, nonoperating mode, with the exception of the operating centrifugal charging pump(s). As such, the ECCS flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly when required to inject into the RCS. This Surveillance Requirement is met by venting ECCS pump casings and the accessible discharge piping high points except for: (1) the RSS pump, RSS heat exchanger and associated RSS piping that is not maintained filled with water during plant operation is excluded from the Surveillance Requirement, (2) the operating centrifugal charging pump(s) and associated piping is also excluded as an operating pump is self venting and cannot develop voids and pockets of entrained gases, and (3) the nonoperating centrifugal charging pumps do not have casing vent connections and the pump manufacturer indicates that venting the suction pipe will assure that the pump casing does not contain voids and pockets of entrained gases. The venting of the nonoperating centrifugal charging pumps is accomplished by venting with the suction line test connection.

Surveillance Requirement 4.5.2.C.2 requires that the visual inspection of the containment be performed at least once daily if the containment has been entered that day and when the final containment entry is made. This will reduce the number of unnecessary inspections and also reduce personnel exposure.

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, and (2) 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. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum/minimum solution temperatures for the RWST in MODES 1, 2, 3 and 4 are based on analysis assumptions.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated August 29, 1997, as supplemented by letters dated September 25 and November 14, 1997, the Northeast Nuclear Energy Company, et al. (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 3 Technical Specifications (TS). Based on a review and subsequent calculations of the cold overpressurization protection (COPS) enabling temperature and the emergency core cooling system (ECCS)/charging system mode 3 requirements, the licensee proposes to reduce the COPS enabling temperature. As a result, the licensee proposed the following TS changes: add new heatup and cooldown pressure/temperature (P-T) limit curves and their associated requirements; add new power operated relief valve (PORV) setpoint curves and their associated requirements; revise the reactor coolant loops and coolant circulation, ECCS, boration systems, and COPS to incorporate the lower enabling temperature and new restrictions for cold overpressure protection system, PORV undershoot, and residual heat removal (RHR) relief valve bellows; add a footnote to allow an reactor coolant pump (RCP) to substitute for an RHR pump during heatup from Mode 5 to 4 which is consistent with the improved standard technical specification (STS); reword TS 3/4.4.9.3 and its surveillance requirement to be consistent with the improved STS; and revise the affected Bases sections to be consistent with the proposed changes. The September 25 and November 14, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 PRESSURE/TEMPERATURE LIMITS

2.1 Background

The August 29, 1997, license amendment request was intended to update the Millstone Unit 3 P-T curves for 10 effective full power years (EFPY) using the latest vessel beltline material and fluence data. In addition, the licensee proposed more restrictions to the P-T limit limiting condition for operation (LCO), and proposed changes to the surveillance requirements in the TS for better clarity.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan

(SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit submittals, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limits for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}) and the Charpy USE at the maximum postulated flaw depth. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term. The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

2.2 Evaluation

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessel of Millstone Unit 3. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 10 EFPY is the intermediate shell B9805-1, with 0.05% copper (Cu), 0.64% nickel (Ni), and an initial RT_{NDT} of 60 °F. The ART calculated by the staff is 111.8 °F for the limiting material. This ART was calculated at 1/4T at 10 EFPY with corresponding neutron fluence of 0.556E19 n/cm^2. The ART calculated by the licensee, using the Chemistry Factor Table in Section 1.1 of RG 1.99, Rev. 2, is 111.9 °F. Both the staff and the licensee did not include the cladding thickness in calculating the attenuation of the fluence through the vessel wall.

The ARTs calculated by the staff and the licensee are almost identical. However, substituting the ART of 111.9 °F into equations in SRP 5.3.2, the staff could not verify the proposed P-T limit curves. In its response to the staff's request, the licensee, in a letter dated September 25, 1997, provided details of the methodology used. This information revealed that Millstone Unit 3 has some unique features built into the P-T limits: (1) the proposed P-T limits considered indicator uncertainties, 22 °F for temperature and 129 psia for pressure; (2) it considered a pressure drop of 28.3 psi between the pressure transmitter and the reactor vessel beltline for one pump operation and a pressure drop of 74 psi for four pump operation; (3) the cooldown curve is made of the conservative portions of the P-T limits at a cooldown rate of 0 °F/hr and the P-T limits at a cooldown rate of 80 °F/hr to 160 °F and 40 °F/hr to 60 °F; and (4) the pressure was expressed in "psia," with a value of 10 added to the gage pressure to account for the primary containment pressure. After considering the extra conservatism in (1) (2), and (3), the staff confirmed the proposed P-T limits (for 10 EFPY) for heatup, cooldown, and hydrotest and concluded that they meet the beltline material requirements in Appendix G of 10 CFR Part 50. The staff found that all equations involving the conversion of the pressure in psig to psia in Reference 2 had a sign error. However, from the intermediate results presented, the staff verified that the correct sign was used in actual calculations.

In addition to beltline materials, Appendix G of 10 CFR Part 50 imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the limiting nozzle shell reference temperatures of 40 °F, the minimum allowable temperature of this region is 160 °F. As mentioned previously, the licensee added 22 °F to 160 °F to account for indicator uncertainty. These limits are shown as straight-line segments on Figures 3.4-2 and 3.4-3 of the submittal, and the staff has determined that the proposed P-T limits satisfy the requirements in Section IV.A.2 of Appendix G.

In the TS, the licensee proposed to further restrict the LCO under "3/4.4.9 Pressure/Temperature Limits" by limiting one pump in operation when the cold leg temperature is less than 160 °F and no pumps in operation when the cold leg temperature is less than 120 °F. The staff considered these measures useful in preventing overpressurization of the RPV at low temperatures. The proposed changes to the surveillance requirements in the TS are for more clarity, and are also acceptable.

Appendix G further requires that the predicted Charpy USE at end-of-license (EOL) for vessel beltline materials be above 50 ft-lb or that licensees demonstrate that lower values of Charpy USE will provide margins of safety equivalent to those required by Appendix G of Section XI of the ASME Code. This USE requirement is satisfied because all beltline materials have EOL USEs above 50 ft-lb.

2.3 Overall

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid as indicated on the curves. The P-T limits satisfy the requirements of Appendix G of 10 CFR Part 50 for 10 EFPY. The proposed P-T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the Millstone Unit 3 TS. The proposed changes to a more restrictive P-T limit LCO and to the surveillance requirements in the TS are also acceptable.

3.0 COLD OVERPRESSURE PROTECTION SYSTEM

3.1 Background

The COPS mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating the 10 CFR Part 50, Appendix G, P-T limits under steady state operating conditions. Millstone Unit 3 COPS uses the combination of PORV and the RHR suction relief valves or an RCS vent with the reactor depressurized to accomplish this function. The system is manually enabled by operators and uses variable setpoints as the lift pressure for the PORV and a single setpoint for RHR suction relief valves. The design basis of Millstone Unit 3 COPS considers both mass-addition and heat-addition transients. The mass-addition analyses account for the injection from one centrifugal charging pump to the RCS. The heat-addition analyses accounts for heat input from the secondary side of the steam generators into the RCS upon starting a single RCP under RCS operational configurations and temperatures.

In October 1996 and May 1997, the licensee identified inconsistencies and nonconservatisms in the current TS with respect to COPS at Millstone Unit 3. In response to its commitments made in Licensee Event Reports 96-038 and 97-030, the licensee has evaluated the TS and COPS at Millstone Unit 3 and by letter dated August 29, 1997, proposed changes to TS 3.1.2, 3.4.1, 3.4.9, 3.5.3, and their associated Bases in the areas that are affected by the modified COPS at Millstone Unit 3. The proposed changes will ensure that the TS with its associated bases are consistent and conservative with respect to COPS for Millstone Unit 3 up to 10 EFPY of reactor operation. The proposed TS provided restrictions in plant operation within the configuration assumed in the analysis for COPS design.

The Millstone Unit 3 proposed COPS enable temperature and the PORV actuation setpoint were established using a plant-specific methodology similar to that presented in WCAP-14040, Revision 1. However, the licensee for Millstone Unit 3 chose to keep its COPS setpoints in its TS.

3.2 Evaluation

The proposed LCO in TS 3.4.9.3 requires that a COPS shall be operable with a maximum of one centrifugal charging pump and no safety injection pumps capable of injecting into the RCS and (1) any combination of two operable relief valves (PORVs or RHR suction relief valves), or (2) RCS depressurized with an RCS vent of $\geq$ 5.4 square inches. This LCO is applicable when any RCS cold leg temperature is $\leq$ 275 °F when the head is on the reactor vessel. Consistent with this LCO, the licensee proposed modifications to TS 3.1.2, 3.4.1, 3.4.9.1, 3.4.9.2, and TS 3.5.3 to restrict the plant operational configuration consistent with the design of the COPS. The staff's evaluation of the COPS setpoints is presented below.

3.2.1 Enable Temperature

The COPS enable temperature is the temperature below which the COPS system is required to be operable. The licensee proposed to establish an COPS enable temperature methodology to: (1) account for instrument uncertainties associated with the instrumentation used to enable the COPS system and, (2) implement the NRC branch technical position (BTP) RSB 5-2 of using an enable RCS water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 °F at the belt line location (1/4T or 3/4T). Therefore, the licensee proposed to calculate the enable temperature as RT_{NDT} + 90 °F + temperature difference between RCS and metal + instrument uncertainties. Using the above equation, the calculated minimum enable temperature is 244 °F. The licensee proposed an enable temperature of 275 °F that includes an additional margin of 31 °F.

The staff finds that this proposed COPS enable temperature is conservative with respect to the enable temperature allowed by NRC (BTP) RSB 5-2 and therefore, is acceptable.

3.2.2 COPS Actuation Setpoint

COPS is designed to mitigate overpressure transients at low temperatures to prevent violating 10 CFR Part 50, Appendix G P-T limits. Additionally, since overpressure events most likely occur during isothermal conditions in the RCS, the NRC has accepted the use of the steady-state Appendix G limits for the design of COPS. The COPS actuation setpoint is the pressure at which the pressure relief valves will lift, when the COPS is enabled, to limit the peak RCS pressure during a pressurization transient.

Millstone Unit 3 uses PORVs and RHR suction relief valves to provide pressure relief capacity for COPS. The methodology used for determining the PORV actuation setpoint is consistent with the methodology presented in WCAP-14040, Revision 1.

The licensee proposed PORV actuation setpoints in TS Figures 3.4-4a and 3.4-4b were calculated in accordance with the proposed methodology. In response to the staff request, the licensee, in its letter dated November 14, 1997, provided a tabulation to list PORV setpoint, transient pressure overshoot, instrumentation uncertainties for temperature and pressure and corresponding P-T limit under various temperature conditions below the COPS enable temperature. The data presented in this tabulation confirms that the proposed PORV setpoints will provide adequate protection to the 10 CFR Part 50, Appendix G, P-T limits under steady state conditions during a design basis overpressure transient (mass-addition or heat-addition) as described in Section 3.1 of this safety evaluation. Based on the preceding discussion, the staff finds the proposed PORV setpoint acceptable.

Millstone Unit 3 uses the RHR suction relief valves as an alternative to PORVs for COPS. The RHR suction relief valves are designed with setpoint of ≤440 psig +/-3%. The licensee has performed an evaluation to verify the adequacy of this relief valve setpoint. The results of its calculation confirmed that the Millstone Unit 3 RHR suction relief valve with the proposed setpoint will adequately mitigate the most limiting overpressure transient and protect the 10 CFR Part 50, Appendix G P-T limits under steady state operating conditions. Also, the design pressure of the RHR discharge piping (600 psig) would not be exceeded during the overpressure transient. The staff finds the RHR suction relief valve setpoint acceptable.

3.2.3 RCS Vent Size

With the RCS depressurized, the results of the licensee's analysis show that a vent size of 5.4 square inches is capable of mitigating the most limiting cold overpressure transient. The capability of a vent this size is greater than the flow resulting from this limiting mass-addition transient. The staff finds this vent size acceptable.

3.3 Overall

The staff has reviewed the licensee's proposed TS 3.4.9.3 for the COPS enable temperature and actuation setpoint. The staff also reviewed the licensee's analyses related to the proposed enable temperature of 275 °F and actuation setpoints as discussed in Sections 3.1 and 3.2 above. The licensee has considered instrument uncertainties in its setpoint calculation using ISA S67.04-1994. The staff finds that the licensee's analyses were performed in a manner consistent with the approved methodology and that the results of the analyses conservatively demonstrated that the 10 CFR Part 50, Appendix G, P-T limits will be adequately protected with these setpoints and, therefore, finds the proposed TS 3.4.9.3 with its associated Bases regarding COPS acceptable.

The staff has also reviewed the licensee's proposed TS 3.1.2, 3.4.1, 3.4.9.1, 3.4.9.2, and 3.5.3 and finds that these proposed TS provide adequate restrictions of plant operation to support the design of COPS at Millstone Unit 3, and therefore, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, an attempt to contact the Connecticut State official was made; however, the official could not be contacted.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 52583 dated December 8, 1997). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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