Docket No. 50-423
B18314

Attachment 4

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Retyped Pages

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE, with at least three reactor coolant loops in operation when the Control Rod Drive System is capable of rod withdrawal or with at least one reactor coolant loop in operation when the Control Rod Drive System is not capable of rod withdrawal:*

a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant 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 reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant 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.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 Either:*,**

a. With the Control Rod Drive System capable of rod withdrawal, at least two RCS loops shall be OPERABLE and in operation, or

b. With the Control Rod Drive System not capable of rod withdrawal, 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 less than the above required reactor coolant loops in operation and the Control Rod Drive System is capable of rod withdrawal, within 1 hour open the Reactor Trip System breakers.

c. 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.

** The first reactor coolant pump shall not be started when any RCS loop wide range cold leg temperature is ≤ 226°F unless:

a. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

b. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.3.3 The required loop(s) shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

a. One additional RHR loop shall be OPERABLE**, or

b. The 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.

***The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

COLD SHUTDOWN - LOOPS FILLED

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.

4.4.1.4.1.3 The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

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.

***The first reactor coolant pump shall not be started when:

a. Any RCS loop wide range cold leg temperature is > 150°F unless:

1. Two pressurizer PORVs are in service to meet the cold overpressure protection requirements of Technical Specification 3.4.9.3 and the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature; OR

2. The secondary side water temperature of each steam generator is at or below each RCS cold leg temperature.

b. All RCS loop wide range cold leg temperatures are ≤ 150°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature.

This restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 The required pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

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.

REACTOR COOLANT SYSTEM

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, and

b. The boron concentration of the isolated loop is greater than or equal to the boron concentration 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 isolated loop boron concentration shall be determined to be greater than or equal to the boron concentration 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 2 hours prior to opening the hot or cold leg stop valve.

3/4.4.2 SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2500 psia ± 3%.**

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 226°F.

ACTION:

With one pressurizer Code safety valve inoperable, restore the inoperable valve to OPERABLE status within 15 minutes. If the inoperable valve is not restored to OPERABLE status within 15 minutes, or if two or more pressurizer Code safety valves are inoperable, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with any RCS cold leg temperature ≤ 226°F within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**The lift setting shall be within ± 1% following pressurizer Code safety valve testing.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 Reactor Coolant System (except the pressurizer) temperature, pressure, and heatup and cooldown rates of ferritic materials shall be limited in accordance with the limits shown on Figures 3.4-2 and 3.4-3. In addition, a maximum of one reactor coolant pump can be in operation when the lowest unisolated Reactor Coolant System loop wide range cold leg temperature is $\leq$ 160°F.

APPLICABILITY: At all times.

ACTION:

a. With any of the above limits exceeded in MODES 1, 2, 3, or 4, perform the following:

   1. Restore the temperature and/or pressure to within limit within 30 minutes.

   AND

   2. Preform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation within 72 hours. Otherwise, be in at least MODE 3 within the next 6 hours and in MODE 5 with RCS pressure less than 500 psia within the following 30 hours.

b. With any of the above limits exceeded in other than MODES 1, 2, 3, or 4, perform the following:

   1. Immediately initiate action to restore the temperature and/or pressure to within limit.

   AND

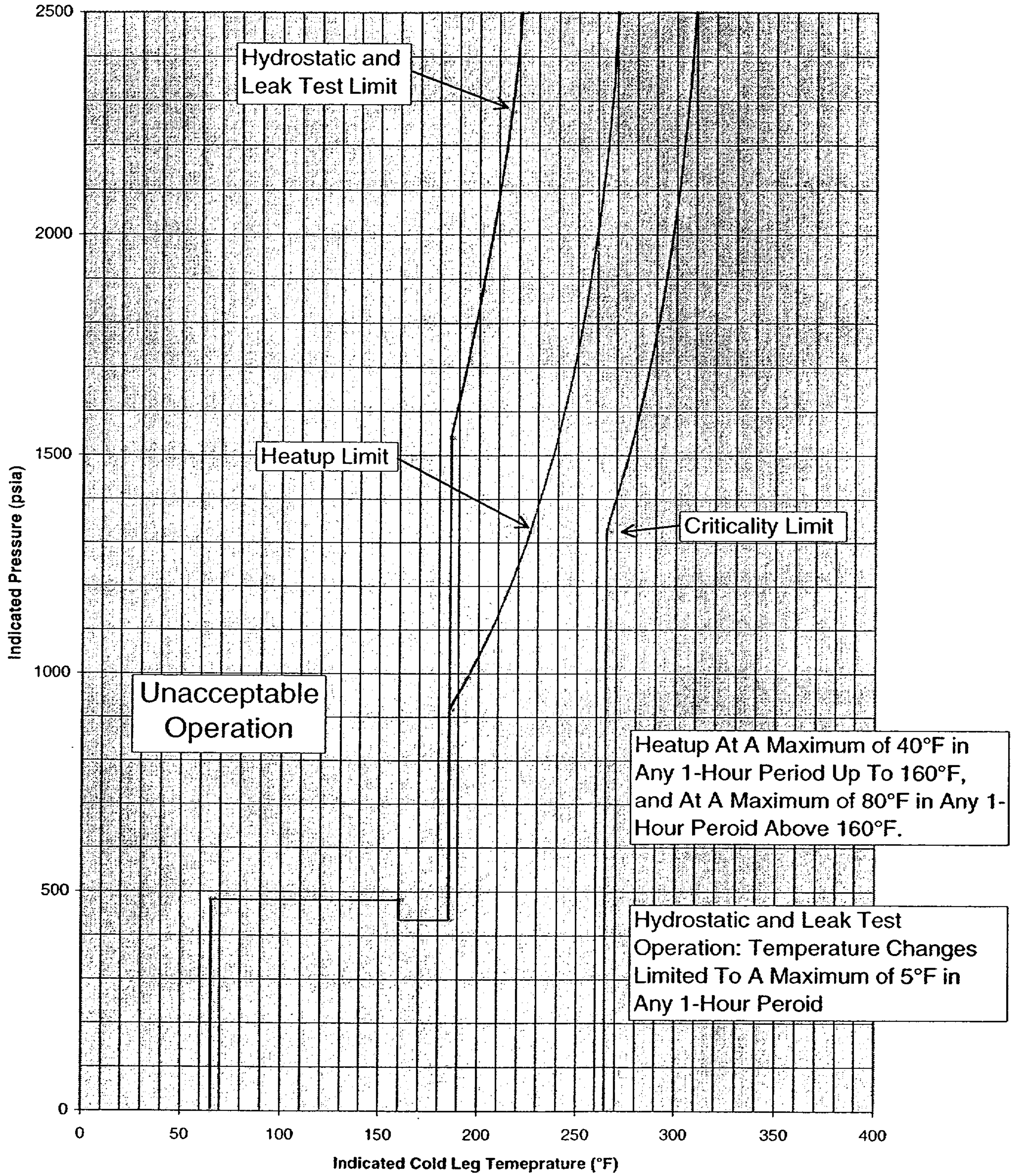
   2. Perform an engineering evaluation to determine the effects of the out of limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System is acceptable for continued operation prior to entering MODE 4.

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 Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)



Millstone 3 Reactor Coolant System
Cooldown Limitations for Fluence up to 1.97E+19 n/cm (32 EFPY)

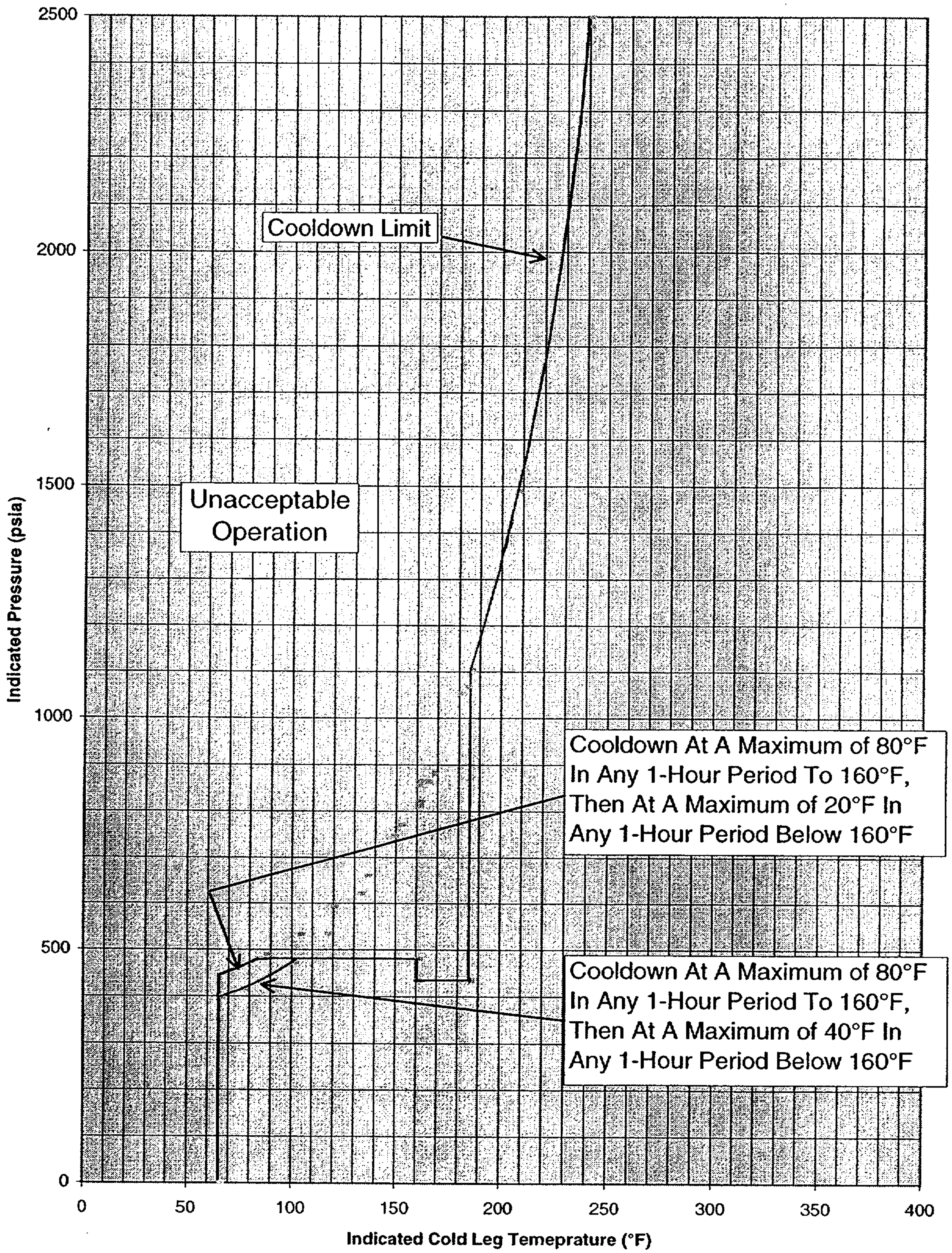


TABLE 4.4-5

Millstone Unit 3 Reactor Vessel Surveillance Capsule Withdrawal Schedule

CAPSULE	LOCATION	LEAD FACTOR[a]	REMOVAL TIME (EFPY)[b]	FLUENCE (n/cm^2, E>1.0MeV)[a]
U	58.5°	4.31	1.3	4.49×10^{18}(c)
X	238.5°	4.37	8.0	2.21×10^{19}(c)
W	121.5°	4.32	Approx. 14.0	3.76×10^{19}(c,d)
Y[e]	241°	4.11	Standby	
V[e]	61°	4.11	Standby	
Z[e]	301.5°	4.32	Standby	

(a) Updated in Capsule X dosimetry analysis.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) This projected fluence is not less than once or greater than twice the peak end of license EOL fluence, and is approximately equal to the peak vessel fluence at 54 EFPY.

(e) These capsules will be at the approximate 54 EFPY peak surface (i.e. clad/base metal interface) fluence when capsule W is withdrawn.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

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 with no more than one isolated RCS loop, 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 with no more than one isolated RCS loop 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 $\geq$ 2.0 square inches.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is $\leq$ 226°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 2.0 square inch vent within the next 12 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.

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 $\geq$ 2.0 square inches within the next 12 hours.

e. With two required relief valves inoperable, depressurize the RCS and establish an RCS vent of $\geq$ 2.0 square inches within 12 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

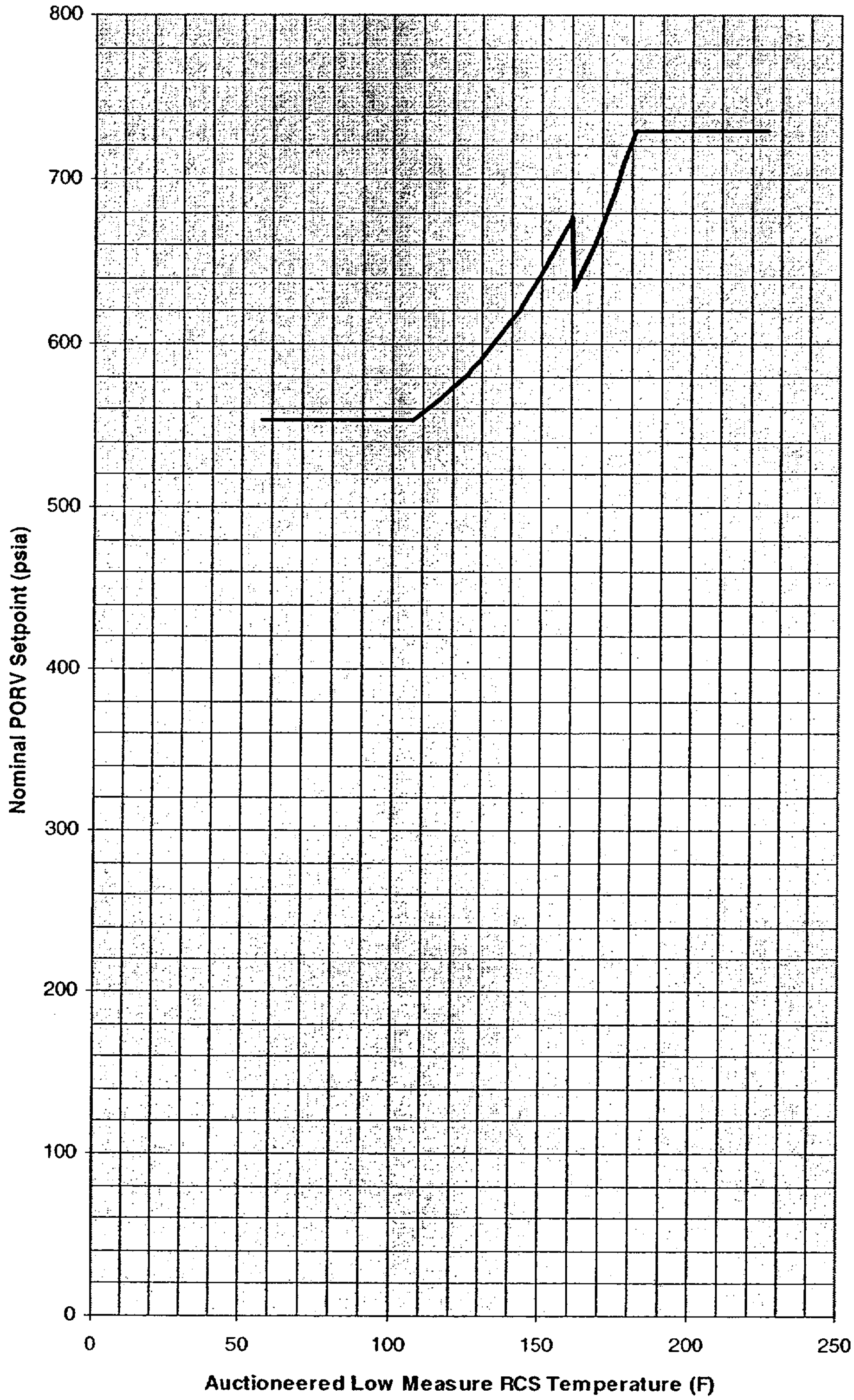
b. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 When complying with 3.4.9.3.4, verify that the RCS is vented through a vent pathway $\geq$ 2.0 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.

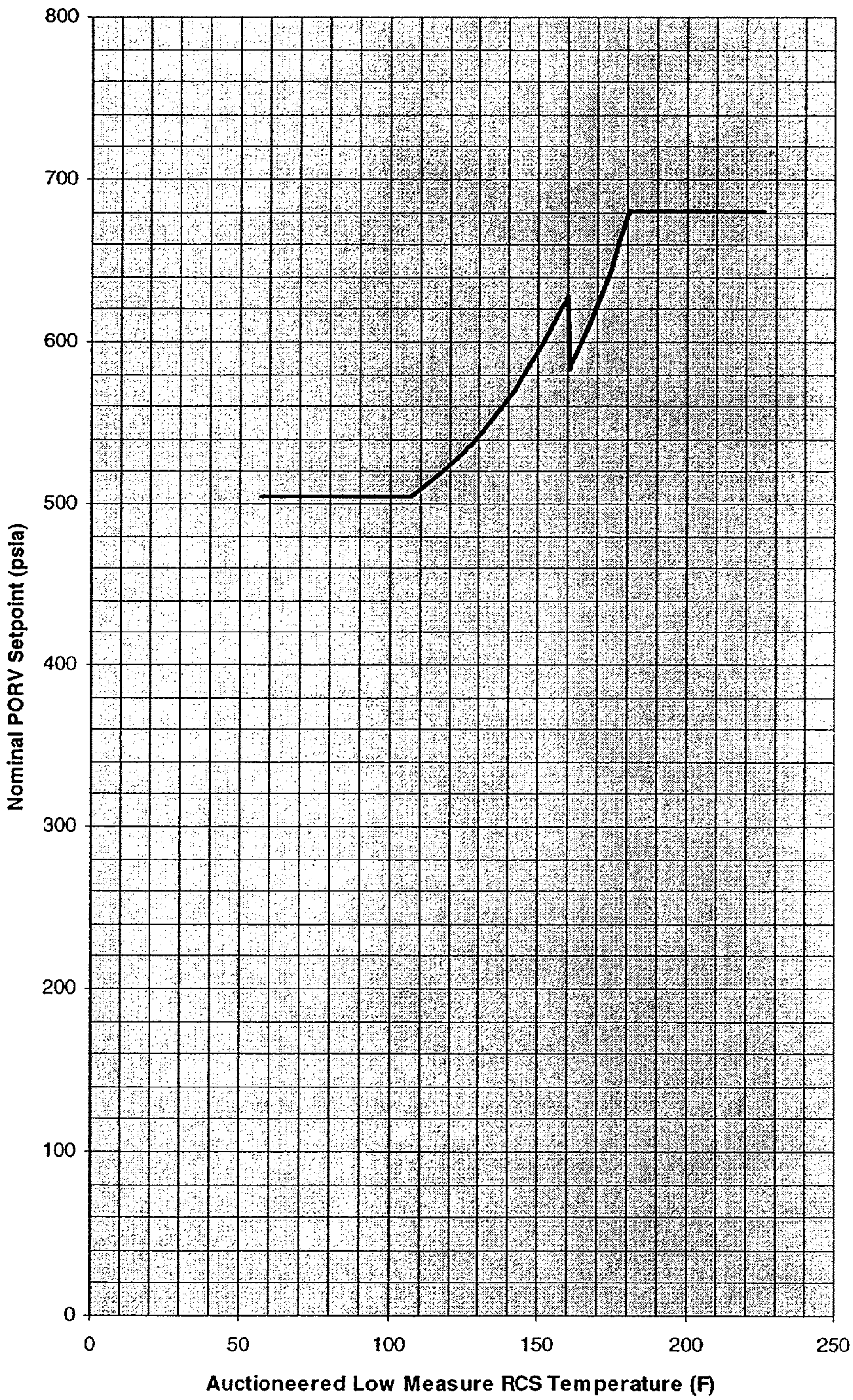
4.4.9.3.4 Verify that no Safety Injection pumps are capable of injecting into the RCS at least once per 12 hours.

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



Low Setpoint PORV Curve
For the Cold Overpressure Protection System



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 DELETED

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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

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., the Control Rod Drive System is not capable of rod withdrawal.

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 the first RCP in MODE 4 below the cold overpressure protection enable temperature (226°F), and in MODE 5 are provided to prevent RCS pressure transients. These transients, energy additions due to the differential temperature between the steam generator secondary side and the RCS, can result in pressure excursions which could challenge the P/T limits. The RCS will be protected against overpressure transients and will not exceed the reactor vessel isothermal beltline P/T limit by restricting RCP starts based on the differential water temperature between the secondary side of each steam generator and the RCS cold legs. The restrictions on starting the first RCP only apply to RCPs in RCS loops that are not isolated. The restoration of isolated RCS loops is normally accomplished with all RCPs secured. If an isolated RCS loop is to be restored when an RCP is operating, the appropriate temperature differential limit between the secondary side of the isolated loop steam generator and the in service RCS cold legs is applicable, and shall be met prior to opening the loop isolation valves.

The temperature differential limit between the secondary side of the steam generators and the RCS cold legs is based on the equipment providing cold overpressure protection as required by Technical Specification 3.4.9.3. If the pressurizer PORVs are providing cold overpressure protection, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50°F. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is above 150°F, the steam generator secondary water temperature must be at or below RCS cold leg water temperature. If any RHR relief valve is providing cold overpressure protection and RCS cold leg temperature is at or below 150°F, the steam generator secondary to RCS cold leg water temperature differential is limited to a maximum of 50°F.

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 isolated loop prior to opening the first stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. If any pressurizer Code safety valve is inoperable, and cannot be restored to OPERABLE status, the action statement requires the plant to be shut down and cooled down such that Technical Specification 3.4.9.3 will become applicable and require cold overpressure protection to be placed in service.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The pressurizer provides a point in the RCS when liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during load transients.

MODES 1 AND 2

The requirement for the pressurizer to be OPERABLE, with pressurizer level maintained at programmed level within ± 6% of full scale is consistent with the accident analysis in Chapter 15 of the FSAR. The accident analysis assumes that pressurizer level is being maintained at the programmed level by the automatic control system, and when in manual control, similar limits are established. The programmed level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure and pressurizer overfill transients. A pressurizer level control error based upon automatic level control has been taken into account for those transients where pressurizer overfill is a concern (e.g., loss of feedwater, feedwater line break, and inadvertent ECCS actuation at power). When in manual control, the goal is to maintain pressurizer level at the program level value. The ±6 % of full scale acceptance criterion in the Technical Specification establishes a band for operation to accommodate variations between level measurements. This value is bounded by the margin applied to the pressurizer overfill events.

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.

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.

The LCO establishes operating limits that provide a margin to brittle failure, applicable to the ferritic material 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".

The P/T limits have been established for the ferritic materials of the RCS considering ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2), and the additional requirements of 10 CFR 50 Appendix G (Reference 3). Implementation of the specific requirements provide adequate margin to brittle fracture of ferritic materials during normal operation, anticipated operational occurrences, and system leak and hydrostatic tests.

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. 4) and Appendix H of 10 CFR 50 (Ref. 5). 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. 6).

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.

PRESSURE/TEMPERATURE LIMITS (continued)

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than 160°F above the minimum permissible temperature for ISLH testing. This limit provides the required margin relative to brittle fracture. 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 ferritic RCPB materials, 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. 7) 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 1, as modified by Reference 2, combined with the additional requirements of Reference 3 provide 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 LCO limits apply to the ferritic components of the RCS, except the Pressurizer. These limits define allowable operating regions while providing margin against nonductile failure for the controlling ferritic component.

The limitations imposed on the rate of change of temperature have been established to ensure consistency with the resultant heatup, cooldown, and ISLH testing P/T limit curves. These limits control the thermal gradients (stresses) within the reactor vessel beltline (the limiting component). Note that while these limits are to provide protection to ferritic components within the reactor coolant pressure boundary, a limit of 100°F/hr applies to the reactor coolant pressure boundary (except the pressurizer) to ensure that operation is maintained within the ASME Section III design loadings, stresses, and fatigue analyses for heatup and cooldown.

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 of ferritic RCS components using ASME Section XI, Appendix G, as modified by Code Case N-640 and the additional requirements of 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 Allowed Outage Times (AOTs) 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. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour AOT when operating in MODES 1 through 4 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 the AOT 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 or not allowed to enter MODE 4 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 evaluation for continued operation in MODES 1 through 4 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.

Completion of the required evaluation following limit violation in other than MODES 1 through 4 is required before plant startup to MODE 4 can proceed.

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.

It is not necessary to perform Surveillance Requirement 4.4.9.1.1 to verify compliance with Figures 3.4-2 and 3.4-3 when the reactor vessel is fully detensioned. During refueling, with the head fully detensioned or off the reactor vessel, the RCS is not capable of being pressurized to any significant value. The limiting thermal stresses which could be encountered during this time would be limited to flood-up using RWST water as low as 40°F. It is not possible to cause crack growth of postulated flaws in the reactor vessel at normal refueling temperatures even injecting 40°F Water.

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. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.

2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.

3. 10 CFR 50 Appendix G, "Fracture Toughness Requirements."

4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706."

5. 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6. Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events," 1995 Edition.

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)

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 isothermal beltline pressure and temperature (P/T) limits developed using the guidance of ASME Section XI, Appendix G (Reference 1) as modified by ASME Code Case N-640 (Reference 2). 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 use of a PORV for Cold Overpressure Protection is limited to those conditions when no more than one RCS loop is isolated from the reactor vessel. When two or more loops are isolated, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized and vented RCS.

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. 3). 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.

If a loss of RCS inventory or reduction in shutdown margin event occurs, the appropriate response will be to correct the situation by starting RCS makeup pumps. If the loss of inventory or shutdown margin is significant, this may necessitate the use of additional RCS makeup pumps that are being maintained not capable of injecting into the RCS in accordance with Technical Specification 3.4.9.3. The use of these additional pumps to restore RCS inventory or shutdown margin will require entry into the associated action statement. The action statement requires immediate action to comply with the specification. The restoration of RCS inventory or shutdown margin can be considered to be part of the immediate action to restore the additional RCS makeup pumps to a not capable of injecting status. While recovering RCS inventory or shutdown margin, RCS pressure will be maintained below the P/T limits. After RCS inventory or shutdown margin has been restored, the additional pumps should be immediately made not capable of injecting and the action statement exited.

OVERPRESSURE PROTECTION SYSTEMS (continued)

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 RCS pressure measured by a wide range pressure channel. If the measured pressure meets or exceeds the calculated value, a PORV is signaled to open.

The use of the PORVs is restricted to three and four RCS loops unisolated: for a loop to be considered isolated, both RCS loop stop valves must be closed. If more than one loop is isolated, then the PORVs must have their block valves closed or COPPS must be blocked. For these cases, Cold Overpressure Protection must be provided by either the two RHR suction relief valves or a depressurized RCS and an RCS vent. This is necessary because the PORV mass and heat injection transients have only been analyzed for a maximum of one loop isolated, the use of the PORVs is restricted to three and four RCS loops unisolated.

The RHR suction relief valves have been qualified for all mass injection transients for any combination of isolated loops. In addition, the heat injection transients not prohibited by the Technical Specifications have also been considered in the qualification of the RHR suction relief valves.

Figure 3.4-4a and Figure 3.4-4b present the PORV setpoints for COPPS. The setpoints are staggered so only one valve opens during a low 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 isothermal P/T 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.

OVERPRESSURE PROTECTION SYSTEMS

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. 4) 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 acceptable pressure levels 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 Pressurizer manway, or similarly establishing a vent by opening an RCS vent valve provided that the opening meets the relieving capacity 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. 5) 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 226°F, the pressurizer safety valves will provide RCS overpressure protection in the ductile region. At 226°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, consistent with ASME Section XI, Appendix G for temperatures less than RT_{NDT} + 50°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 any RCS cold leg is ≤ 226°F.

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 starting the first reactor coolant pump such that it shall not be started when any RCS loop wide range cold leg temperature is ≤ 226°F unless the secondary side water temperature of each steam generator is < 50°F above each RCS cold leg temperature. The restrictions ensure the potential energy addition to the RCS from the secondary side of the steam generators will not result in an RCS overpressurization event beyond the capability of the COPPS to mitigate. The COPPS utilizes the pressurizer PORVs and the RHR relief valves to mitigate the limiting mass and energy addition events, thereby protecting the isothermal reactor vessel beltline P/T limits. The restrictions will ensure the reactor vessel will be protected from a cold overpressure event when starting the first RCP. If at least one RCP is operating, no restrictions are necessary to start additional RCPs for reactor vessel protection. In addition, this restriction only applies to RCS loops and associated components that are not isolated from the reactor vessel.

The RCP starting criteria are based on the equipment used to provide cold overpressure protection. A maximum temperature differential of 50°F between the steam generator secondary sides and RCS cold legs will limit the potential energy addition to within the capability of the pressurizer PORVs to mitigate the transient. The RHR relief valve are also adequate to mitigate energy addition transients constrained by this temperature differential limit, provided all RCS cold leg temperature are at or below 150°F. The ability of the RHR relief valves to mitigate energy addition transients when RCS cold leg temperature is above 150°F has not been analyzed. As a result, the temperature of the steam generator secondary sides must be at or below the RCS cold leg temperature if the RHR relief valves are providing cold overpressure protection and the RCS cold leg temperature is above 150°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 ≤ 226°F based on the criteria provided by ASME Section XI, Appendix G.

PORV Performance

The 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 for both three and four RCS loops unisolated 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 beyond the PORV opening setpoint 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 ≤ 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 $\geq$ 2.0 square inches is capable of mitigating the limiting 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).

OVERPRESSURE PROTECTION SYSTEMS (continued)

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 reactor vessel isothermal P/T 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 for both three and four loops unisolated 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 $\geq$ 2.0 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 ≤ 226°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. > 226°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, "Safety Valves," requires the OPERABILITY of the Pressurizer safety valves that provide overpressure protection during MODES 1, 2, 3, and 4 when all RCS cold leg temperatures are > 226°F.

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 ≤ 226°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 12 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 overpressurize the RCS are more severe at lower temperatures (Ref. 8). 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 12 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 12 hours when both required Cold Overpressure Protection relief valves are inoperable.

The vent must be sized $\geq$ 2.0 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. 9), 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 ≥ 2.0 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. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness for Protection Against Failure," 1995 Edition.
2. ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.
3. Generic Letter 88-11
4. ASME, Boiler and Pressure Vessel Code, Section III
5. FSAR, Chapter 15
6. 10CFR50, Section 50.46
7. 10CFR50, Appendix K
8. Generic Letter 90-06
9. ASME, Boiler and Pressure Vessel Code, Section XI

Attachment 5

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Exemption Request from 10 CFR 50.60(a) and 10 CFR 50 Appendix G

Technical Specifications Change Request 3-11-00
Reactor Coolant System Heatup and Cooldown Curves
Exemption Request from 10 CFR 50.60(a) and 10 CFR 50 Appendix G

Purpose

Dominion Nuclear Connecticut, Inc. (DNC), is requesting, on behalf of Millstone Unit No. 3 an exemption from application of specific requirements of 10 CFR 50.60(a) and 10 CFR 50 Appendix G, and substitute use of American Society of Mechanical Engineers (ASME) Code Case N-460.(1) Code Case N-640 permits the use of an alternate reference fracture toughness for reactor vessel materials in determining pressure/temperature (P/T) limits.

In addition to this request, DNC is requesting a revision to the Millstone Unit No. 3 Technical Specifications 3.4.9.1, "Reactor Coolant System - Pressure/Temperature Limits;" and 3.4.9.3, "Reactor Coolant Systems - Overpressure Protection Systems." The proposed changes to the Millstone Unit No. 3 Technical Specifications (refer to Attachment 1 of this submittal) will revise the P/T limits related to the heatup and cooldown of the Reactor Coolant System (RCS) and the Cold Overpressure Protection System (COPPS) setpoints for operation up to 32 effective full-power years (EFPY). The proposed Technical Specification changes are based in part on this exemption request.

Background

The current Technical Specification heatup and cooldown limitations utilize a peak vessel fluence for predicting irradiation damage to the reactor vessel beltline materials previously associated with 10 EFPY of operation. The fluence used as the basis was the result of conservative fluence analyses subsequent to the first capsule evaluation (capsule U). Based on current plant performance, 10 EFPY is expected to be reached late Summer, 2001. The current P/T curves utilize the guidance of ASME Section XI Appendix G, as currently required by 10 CFR 50 Appendix G.

The results of the most recent capsule removal and evaluation (capsule X) provide the most accurate results available, and consider the complete power history and core loading patterns to provide revised projections of accumulated vessel fluence. The results of this analysis was documented in WCAP-15405, Rev. 0,(2) which was submitted to the NRC by a letter dated May 17, 2000.(3) The projected fluence for 32

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(1) ASME Section XI, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," dated February 26, 1999.

(2) WCAP-15405, Rev. 0, "Analysis of Capsule X from the Northeast Nuclear Energy Company Millstone Unit 3 Reactor Vessel Radiation Surveillance Program," May 2000.

(3) S. E. Scace letter to the NRC, "Millstone Nuclear Power Station, Unit No. 3 Submittal of Second Reactor Vessel Surveillance Capsule Report," dated May 17, 2000.

EFPY was chosen as the basis for the proposed heatup and cooldown limitations. Consistent with the current curves and current licensing basis, Regulatory Guide 1.99, Rev. 2,[4] was used to predict the beltline material degradation. However, the development of the beltline P/T limits was established using ASME Section XI, Appendix G, as modified by ASME Code Case N-640. This code case, which is not currently endorsed for use by the NRC in the regulations, provides an alternate reference fracture toughness curve (K_{IC}) for establishment of the beltline P/T limits. The additional requirements of 10 CFR 50 Appendix G were considered in the establishment of the P/T limits. Clarification regarding applicablility of P/T limits to ferritic materials was also provided. The proposed changes result in less restrictive P/T limits.

Directly related to the P/T limits are the COPPS requirements. The current COPPS (also referred to as low temperature overpressure protection or LTOP) setpoint curves are affected since the basis for the setpoint curves, the isothermal beltline P/T limit, has been modified. The setpoint curves were established to ensure the applicable limit would not be exceeded by placing operational restrictions consistent with the COPPS analyses assumptions.

The proposed COPPS setpoint curves have been established to protect the 32 EFPY isothermal reactor vessel beltline P/T curve and the power operated relief valve (PORV) discharge piping design pressure of 800 psia. When applying ASME Code Case N-640, the COPPS shall limit the maximum pressure in the vessel beltline to 100% of the isothermal beltline P/T limit curve. This condition was satisfied in the development of the COPPS setpoint curves. The limiting design basis transients associated with the COPPS setpoint curves have not changed. The limiting energy addition event remains the start of a reactor coolant pump (RCP) with the RCS at a temperature no higher than 250°F, and the secondary side 50°F higher than the RCS. The limiting mass addition event remains the injection of a single charging pump (unthrottled). A staggered high and low setpoint philosophy was maintained over the entire temperature range to minimize the potential of both valves opening concurrently. Both setpoint curves are capable of protecting the P/T limit curve, which satisfies single failure assumptions. The analyses, which assume water solid operation, are conservative with respect to plant operation with a steam bubble in the pressurizer.

Discussion

The proposed Technical Specification changes to modify the P/T limits and COPPS setpoints rely in part on the requested exemption to use ASME Code Case N-640. The revised P/T limits, as specified in ASME Code Case N-640, use a higher allowable

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(4) Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

stress intensity factor, K_{IC} instead of K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor, based on the lower bound values of K_{IC} and K_{IA}.

The use of Code Case N-640 has been approved by the NRC for the Shearon Harris Nuclear Power Plant (Docket No. 400).[5] The following discussion was contained in the NRC approval of the exemption request submitted by Shearon Harris.

> Use of the K_{IC} in determining the lower bound fracture toughness in the development of P/T operating limits curves and LTOP setpoints is more technically correct than the K_{IA} curve since the rate of loading during heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the conservatism of the K_{IA} curve since 1974, when the curve was adopted by the ASME Code. This conservatism was necessary due to the limited knowledge of the fracture toughness of RPV materials at that time. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{IA} curve greatly exceeds the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P/T curves and LTOP setpoints based on the K_{IC} curve will enhance overall plant safety by opening the P/T operating window, with the greatest safety benefit in the region of low temperature operations.
>
> Since an unnecessarily reduced P/T operating window can reduce operator flexibility without just basis, implementation of the proposed P/T curves and LTOP setpoints as allowed by ASME Code Case N-640 may result in enhanced safety during critical plant operational periods, specifically heatup and cooldown conditions.

The primary safety benefits in opening the low temperature operating window are a reduction in the challenges to pressurizer PORVs, and additional margin to maintain RCP net positive suction head (NPSH) requirements. In addition, the pressure undershoot due to the relief capacity of one PORV and the time delay for the valve to close after opening for pressure relief due to a COPPS event can result in damage to the RCP seals due to inadequate seal differential pressure. Damage to the RCP seals can require an unplanned shutdown to replace the seals. By raising the COPPS setpoints at low RCS temperatures, the likelihood of challenging the pressurizer PORVs will be reduced, and operation at higher pressures to provide additional margin for RCP seal protection will be allowed.

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[5] R. J. Laufer (NRC) letter to J. Scarola, "Exemption from the Requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G - Shearon Harris Nuclear Power Plant, Unit 1 (TAC No. MA8643)," dated July 26, 2000.

The P/T limits determined using ASME Code Case N-640 are less restrictive than the requirements of 10 CFR 50 Appendix G, Section IV.A.2.b, which requires the use of methods equivalent to those provided by Appendix G to ASME Section XI. Since ASME Section XI Code Case N-640 was employed in the development of the reactor vessel beltline P/T limits, an exemption to 10 CFR 50.60(a), based on ASME Code Case N-640, is required to support the proposed Technical Specification changes.

Justification

10 CFR 50.12(a)(1) states that the Commission may grant exemptions from the regulations in 10 CFR 50 provided they are "authorized by law, will not present an undue risk to the public health and safety, and consistent with the common defense and security."

1. The Requested Exemption is authorized by law.

   10 CFR 50.60(a) and 10 CFR 50 Appendix G specify that the requirements for the P/T limits are the ASME Code XI, Appendix G limits. Both 10 CFR 50 Appendix G and the ASME Code requires that the effects of neutron irradiation on the material properties of the reactor pressure vessel (RPV) be considered. However, Millstone Unit No. 3 will be using ASME Code Case N-640. Code Case N-460 permits the use of an alternate reference fracture toughness for reactor vessel materials in determining the P/T limits. 10 CFR 50.60(b) permits the Commission to grant exemptions in accordance with 10 CFR 50.12. Therefore, the Commission is authorized by law to grant this exemption.

2. The Requested Exemption Does Not Present an Undue Risk to the Public Health and Safety.

   The proposed P/T limit curves have been developed using the K_{IC} reference fracture toughness curve in lieu of the K_{IA} reference fracture toughness curve. The primary difference between the two curves is that the K_{IA} reference fracture toughness curve represents a lower bound considering limited static and dynamic test results. The K_{IC} curve represents consideration of static initiation representative of reactor pressure vessel behavior.

   The K_{IA} curve was codified in 1974 and included necessary conservatism due to limited experience and knowledge of fracture toughness of reactor pressure vessel materials. As noted by its parent document WRC 175,[(6)] and ASME Section XI, Working Group on Operating Plant Criteria, the resulting curve is very conservative.

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(6) WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," August 1972.

Since that time, significant additional information has been attained regarding fracture toughness of reactor pressure vessel steels. The current K_{IC} curve is based on over 1500 data points eliminating uncertainties previously postulated. Based on the current information available, use of the K_{IC} curve will provide an adequate margin of safety to protect the public health and safety from potential reactor pressure vessel failure.

Use of this code case as providing adequate margins of safety has previously been recognized by the NRC granting exemptions from 10 CFR 50.60(a).

The proposed exemption from 10 CFR 50.60(a) and 10 CFR 50 Appendix G does not present an undue risk to the public health and safety because the implementation of the proposed P/T curves, as allowed by ASME Code Case N-640, may result in enhanced safety during critical plant operational periods, specifically heatup and cooldown conditions. In addition, the underlying purpose of the regulation will continue to be achieved as use of the K_{IC} curve is technically appropriate.

3. The Requested Exemption Will Not Endanger the Common Defense and Security.

The activity, implementation of the proposed P/T curves and COPPS setpoints as allowed by ASME Code Case N-640, is not considered in the common defense and security of the nation. Therefore, this exemption will not impact the common defense and security.

Special Circumstances

Additionally, 10 CFR 50.12(a)(2) states that "the Commission will not consider granting an exemption unless special circumstances are present." It then provides a list of special circumstances. In this instance, a special circumstance is applicable. It is 10 CFR 50.12(a)(2)(ii): "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

Since an unnecessarily reduced P/T operating window can reduce operator flexibility without just basis, implementation of the proposed P/T curves as allowed by the ASME Code Case N-640 may result in enhanced safety during critical plant operational periods, specifically heatup and cooldown conditions. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of 10 CFR 50.60 and 10 CFR 50 Appendix G will continue to be served.

Conclusion

In summary, the ASME Section XI Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effect of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. Therefore, DNC concludes that this increased knowledge permits relaxation of the ASME XI Appendix G requirements by application of ASME Code Case N-640, and the request for an exemption from 10 CFR 50.60(a) and 10 CFR 50 Appendix G is justified pursuant to 10 CFR 50.12(a)(1) and 10 CFR 50.12(a)(2)(ii).