Docket No. 50-336

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revisions to Technical Specifications

August, 1980

Technical Specification Changes - Cycle 4 Reload

P. 1-7	Revised shutdown margin in Mode 5.
P. 2-2	Delete words referring to Figure P 2.1-1.
P. 2-4, 5	Revised LSS Setpoints and Allowable Values.
P. B2-1	W-3 correlation, DNBR of 1.30. Deleted Reference to Figure B 2.1-1.
P. B2-2	Delete Figure B2.1-1.
P. B2-3	DNBR of 1.30.
P. B2-4	Revise Bases 2.2.1 Power Level High to Reflect the Revised Trip Setpoint.
P. B2-5	DNBR of 1.30.
P. B2-6	DNBR of 1.30.
P. B2-8	DNBR of 1.30.
P. 3/4 1-1	Revised Charging Pump Flow Rate.
P. 3/4 1-3	Shutdown Margin, Revised Charging Pump Flow Rate.
P. 3/4 1-5	Moderator Temperature Coefficient less negative than -2.4×10^{-4} $\Delta K/K^{\circ}F$.
P. 3/4 1-24a	Delete page.
P. 3/4 2-5	As shown.
P. 3/4 9-1	40 gpm. Revised Charging Pump Flow Rate.
P. 3/4 10-1	40 gpm. Revised Charging Pump Flow Rate.
P. B3/4 1-1	Shutdown Margin.
P. B3/4 1-3	Shutdown Margin.
P. B3/4 7-2	Automatic auxiliary feedwater system.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3. HOT STANDBY	< 0.99	0	$\geq 300°F$
4. HOT SHUTDOWN	< 0.99	0	$300°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	$< 0.99^{(1)}$ $< 0.98^{(2)}$	0 0	$\leq 200°F$ $\leq 200°F$
6. REFUELING**	≤ 0.90	0	$\leq 140°F$

*Excluding decay heat.

**Reactor vessel head unbolted or removed and fuel in the vessel.

(1) When the RCS is full or when the RCS is partially drained and the core is in the "ALL RODS IN" configuration.

(2) When the RCS is partially drained.

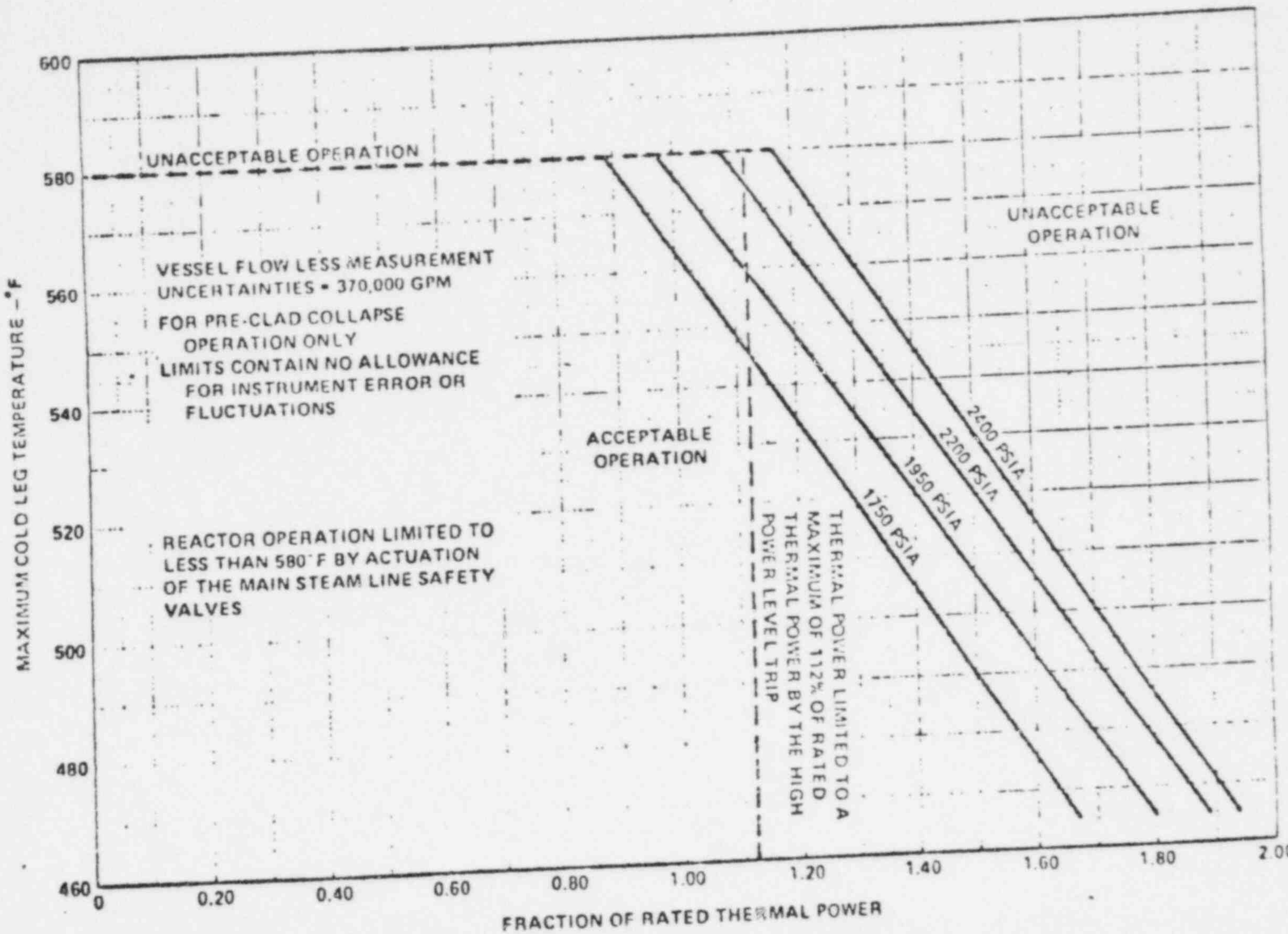

FIGURE 2.1-1 Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level-High		
Four Reactor Coolant Pumps Operating	≤ 9.60% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.1% of reactor coolant flow with 4 pumps operating*.
4. Reactor Coolant Pump Speed - Low	≥ 830 rpm	≥ 823 rpm
5. Pressurizer Pressure - High	≤ 2400 psia	≤ 2408 psia
6. Containment Pressure - High	≤ 4.75 psig	≤ 5.23 psig
7. Steam Generator Pressure - Low (2) (5)	≥ 500 psia	≥ 492 psia
8. Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 35.2% Water Level - each steam generator

* Design Reactor Coolant flow with 4 pumps operating is 370,000 gpm.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High (3)	Trip setpoint adjusted to not (6) exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip setpoint adjusted to not (6) exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not (6) exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not (6) exceed the limit lines of Figures 2.2-3 and 2.2-4.
11. Loss of Turbine -- Hydraulic Fluid (3) Pressure - Low	$\geq$ 500 psig	$\geq$ 500 psig

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

(6) Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to 40
predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 40
percent confidence level that DNB will not occur and is chosen as as appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.30. 40
The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperatures is not 40
a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip set point specified in Table 40
2.2-1. The area of safe operation is below and to the left of these lines.

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The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.30 and preclude the existence of flow instabilities. 40

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.60% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

Reactor Coolant Flow-Low (Continued)

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation 40 and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational 40 transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 815 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low Trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.30. 40

LIMITING SAFETY SYSTEM SETTINGS

BASES

Underspeed - Reactor Coolant Pumps

The Underspeed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on all four reactor 40
coolant pumps. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above 1.30 following a 4 pump loss of flow event.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3.20% $\Delta k/k$. 32

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

ACTION:

With the SHUTDOWN MARGIN < 3.20% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm until the required SHUTDOWN MARGIN is restored. 32

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.20% $\Delta k/k$: 32

a. Immediately upon detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, the SHUTDOWN MARGIN, required by Specification 3.1.1.1, shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. When in MODES 1 or 2, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. Prior to initial operation above 5% RATED THERMAL POWER after each refueling, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. $\geq$ 1.0% $\Delta k/k$ when the reactor coolant system is full.

b. $\geq$ 2.0% $\Delta k/k$ or $\geq$ 1% $\Delta k/k$ and in the All Rods in configuration when the system is partially drained.

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN conditions are not met, immediately initiate and continue boration at $\geq$ 40 gpm until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be in accordance with LIMITING CONDITION FOR OPERATION statement 3.1.1.2a or 3.1.1.2b (as applicable):

a. Within one hour after detection of an inoperable CEA. If the inoperable CEA is immovable or untrippable, within one hour the SHUTDOWN MARGIN required by Specification 3.1.1.2 shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA.

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration,
6. Samarium concentration, and
7. Reactor coolant system fluid volume.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT (MTC)

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.4×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours. 32

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the predicted values. 32

* With $K_{eff} \geq 1.0$.

\# See Special Test Exemption 3.10.2.

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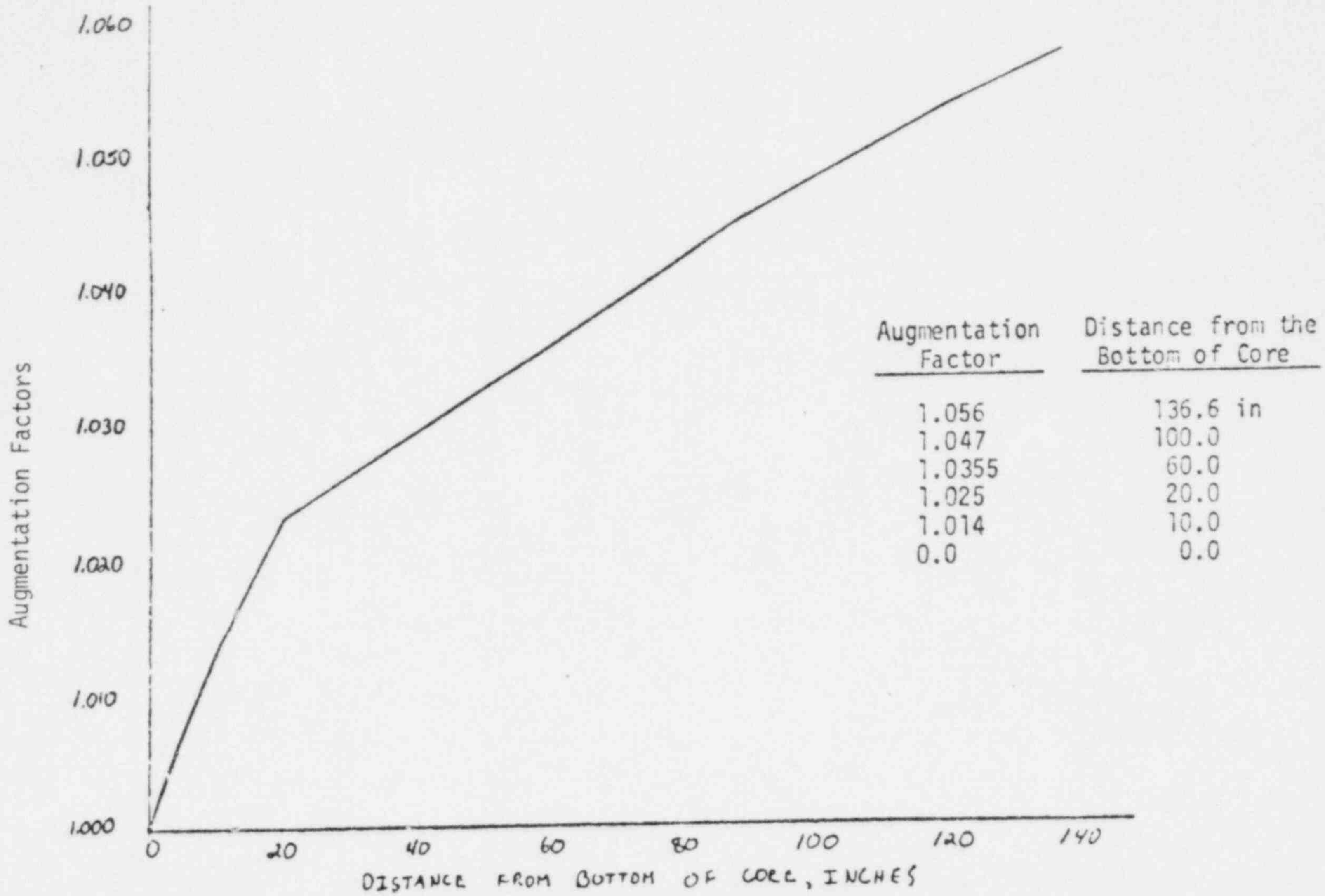

Augmentation Factor	Distance from the Bottom of Core
1.056	136.6 in
1.047	100.0
1.0355	60.0
1.025	20.0
1.014	10.0
0.0	0.0

Figure 4.2-1
Augmentation Factors Vs. Distance
From Bottom of Core

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure a K_{eff} equivalent to no greater than 0.90 with all full length CEAs (shutdown and regulating) fully inserted.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend CORE ALTERATIONS and initiate and continue boration at $\geq$ 40 gpm until K_{eff} is reduced to $\leq$ 0.90. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1 The boron concentration of the refueling pool shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

*See Special Test Exception 3.10.4.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 40 gpm of $\geq$ 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored. 40

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 40 gpm of $\geq$ 1720 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored. 40

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.2% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection when the RCS is full.

If the RCS is drained for maintenance work (approximately 4828 cubic feet) the reactor shall be maintained at 2% $\Delta k/k$ SHUTDOWN MARGIN or in the All Rods In configuration.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 10,060 + 700/-0 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 1% Δ k/k SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn and the primary system full. (approximately 9260 cubic feet without the pressurizer) This condition requires either 5,050 gallons of 6.25% boric acid solution from the boric acid tanks or 57,000 gallons of 1720 ppm borated water from the refueling water storage tank.

If the system is drained for maintenance work (approximately 4828 cubic feet) the reactor shall be maintained at 2% Δk/k SHUTDOWN MARGIN or in the All Rods In configuration.

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 a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment (≥ 20 steps) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a immovable or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 20 steps) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 20 steps) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

107 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour = 6.35×10^6 lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour = 7.94×10^5 lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of off-site power.

Either motor driven pump or the steam driven pump have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to ≤ 300°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 300°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 10 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction

Docket No. 50-336

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Discussion of Proposed

Revisions to Technical Specifications

August, 1980

Proposed Technical Specification Changes

The purpose of this Attachment is to provide an explanation for each of the changes proposed in support of Cycle 4 operation of Millstone Unit No. 2.

(1) In Reference (11), NNECO reported to the NRC Staff that the calculated shutdown margin required by Technical Specifications in Mode 5 at Millstone Unit No. 2 was based on a Reactor Coolant System (RCS) Volume of 9,500 ft^3. However, during Mode 5, the RCS can be drained to the centerline of the hot legs which correspond to an RCS volume of 4,828 ft^3. This smaller volume requires a larger shutdown margin to result in acceptable consequences during a postulated boron dilution event in Mode 5.

A reanalysis of the boron dilution event in Mode 5 using a conservative RCS volume of 4,828 ft^3 was performed. The results of this analysis illustrate that a shutdown margin of 2% $\frac{\Delta K}{K}$ with "All Rods Out" of the core will adequately provide the required time of 15 minutes for operator action to mitigate the dilution event. The shutdown margin in Mode 5 with the RCS partially drained can be reduced to 1% $\frac{\Delta K}{K}$ with the core in the "All Rods In" configuration. The "All Rods In" configuration assures a shutdown margin of 2% $\frac{\Delta K}{K}$.

The analysis of the boron dilution event in Mode 5 for Cycle 4 is provided in References (6) and (7).

Technical Specification pages 1-7, 3/4 1-3, B 3/4 1-1, and B 3/4 1-3 have been revised accordingly to reflect the results of the reanalysis of the boron dilution event in Mode 5.

(2) The infomation presented on Figure 2.1-1 on page 2-2 has been revised to reflect the deletion of Figure B 2.1-1 on page B 2-2. The axial power distributions illustrated in Figure B 2.1-1 were valid for the previous vendor's analyses. The Westinghouse axial power distribution

methodology is utilized to produce Figure 2.1-1. It is therefore, inappropriate to include Figure B 2.1-1 in the Cycle 4 Technical Specifications.

(3) The Reactor Protection System (RPS) Setpoints listed in Table 2.2-1 on pages 24 and 2-5 have been updated and the allowable values have been revised to include the maximum expected drift assumed to occur (between surveillance intervals) for each trip in the accident analyses. The revised Allowable Values are consistent with Section 2.2.1 of the Limiting Safety System Settings (LSSS) Bases. The LSSS Bases on page B 2-4 have also been revised to reflect changes made to Table 2.2-1.

a. Upon review of the bases for the uncertainties utilized in the Power Level-High trip by the NSSS vendor, it was determined that Technical Specification Table 2.2-1 required revising to be consistent with these bases. Specifically, while the uncertainties remain the same (5%) for the Power Level-High trip, the uncertainty shall be combined multiplicatively rather than additively as was the case in previous cycles.

b. The Reactor Coolant Pump (RCP) Speed-Low trip setpoint has been updated to 830 RPM. The present trip setpoint provided to the Staff in Reference (12), as supplemented by Reference (13), was calculated by utilizing a nominal RCP operating speed which was derived from measurements of RCS flow rate. This was required as no data on RCP nominal operating speed existed at the time of the Reference (13) submittal.

During Cycle 3 operation, measurements of actual RCP speed at nominal conditions were made. The measured value of the RCP Speed was slightly greater than that derived from the RCS flow rate. This slight deviation in the measured vs. calculated speed of the RCP's resulted in the revised trip setpoint for the Reactor Coolant Pump Speed-Low function.

It is noted that although the Millstone Unit No. 2 Safety Analysis requires the Reactor Coolant Pump Speed-Low trip setpoint be 830 RPM, the actual trip setpoint for Cycle 3 was significantly more conservative (nominally 845 rpm) as will be the case for Cycle 4.

(4) The Safety Limit Bases on pages B 2-1 and B 2-3 and the Limiting Safety System Settings Bases on pages B 2-5, B 2-6, and B 2-8 have been revised to reflect the use of the W-3 correlation in the Cycle 4 Safety Analyses. The W-3 correlation relates reactor power, reactor coolant temperature and pressure to DNBR and limits the calculated minimum DNBR to 1.30 which provides a 95/95 probability/ confidence level that DNB will not occur.

(5) In Reference (14), Northeast Nuclear Energy Company reported to the NRC Staff that in-service inspections at Millstone Unit No. 2 had shown that under test conditions, the positive displacement charging pumps deliver approximately 43 gpm. These pumps were previously reported to deliver 44 gpm in the FSAR.

The ECCS/LOCA analyses performed in support of Cycle 4 subsequently utilized a conservative value of 40 gpm on the flow delivered by these pumps and acceptable results were obtained.

Pages 3/4 1-1, 3/4 9-1, and 3/4 10-1 have been revised to reflect the conservative charging pump flow of 40 gpm as used in the safety analyses.

(6) The Moderator Temperature Coefficient (MTC) has been recalculated based on Cycle 4 core kinetics characteristics. Page 3/4 1-5 of the Millstone Unit No. 2 Technical Specifications are proposed to be revised accordingly.

(7) Page 3/4 1-24a is proposed to be deleted as the requirements of this specification no longer apply to Millstone Unit No. 2 and the Technical Specifications are clarified accordingly.

(8) Figure 4.2-1 on page 3/4 2-5 has been revised to reflect the augmentation factors which will apply to the Westinghouse Batch F fuel during Cycle 4. The proposed augmentation factors presented in the attached Figure 4.2-1 envelop the previous cycles' factors and will therefore be utilized for the entire Cycle 4 core.

(9) In Reference (15) and (16), NNECO provided the Staff with the proposed design for the control grade automatic auxiliary feedwater system to be installed at Millstone Unit No. 2. Analyses supporting the design have illustrated that either electric motor driven auxiliar feedwater pump on the steam driven auxiliary feedwater pump will provide adequate feedwater to the steam generators to remove reactor decay heat. The Plant Systems Bases Section 3/4 7.1.2 referring to the Auxiliary Feedwater Pumps on page B 3/4 7-2 are proposed to be modified accordingly.