Docket No. 50-423
B15094

Attachment 1

Millstone Unit No. 3
Proposed Revision to Technical Specifications

Reactor Coolant System Flow Rate
Marked Up Pages

January 1995

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| b. Three Loops Operating | | | | | |
| 1) Channels I, II | 10.0 | 6.80 | 1.71 + 1.33 (Temp + Press) | See Note 1 | See Note 2 |
| 2) Channels III, IV | 10.0 | 5.83 | 1.71 + 2.60 (Temp + Press) | See Note 1 | See Note 2 |
| 8. Overpower ΔT | 4.8 | 1.24 | 1.71 | See Note 3 | See Note 4 |
| 9. Pressurizer Pressure-Low | 5.0 | 1.77 | 3.3 | ≥ 1900 psia | ≥ 1890 psia |
| 10. Pressurizer Pressure-High | 5.0 | 1.77 | 3.3 | ≤ 2385 psia | ≤ 2395 psia |
| 11. Pressurizer Water Level-High | 8.0 | 5.13 | 2.7 | ≤ 89% of instrument span | ≤ 90.7% of instrument span |
| 12. Reactor Coolant Flow-Low | 2.5 | 1.52 | 0.78 | ≥ 90% of loop design flow* | ≥ 89.1% of loop design flow* |
| 13. Steam Generator Water Level Low-Low | 18.10 | 16.64 | 1.50 | ≥ 18.10% of narrow range instrument span | ≥ 17.11% of narrow range instrument span |
| 14. General Warning Alarm | N.A. | N.A. | N.A. | N.A. | N.A. |
| 15. Low Shaft Speed - Reactor Coolant Pumps | 3.8 | 0.5 | 0 | ≥ 95.8% of rated speed | ≥ 92.5% of rated speed |

*Minimum Measured Flow Per Loop = 96,870 gpm (Four Loops Operating); 101,066 gpm (Three Loops Operating)

1/4 of the RCS Flow Rate limit as listed in Section 3.2.3.1a

1/3 of the RCS Flow Rate limit as listed in Section 3.2.3.2a

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 387,480 gpm, and 371,920

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

Where:

1) $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the CORE OPERATING LIMITS REPORT (COLR), .

4) $PF_{\Delta H}$ - The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rat[illegible]r $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

Attachment 2

Millstone Unit No. 3
Proposed Revision to Technical Specifications

Reactor Coolant System Flow Rate
Retyped Pages

January 1995

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| b. Three Loops Operating | | | | | |
| 1) Channels I, II | 10.0 | 6.80 | 1.71 + 1.33 (Temp + Press) | See Note 1 | See Note 2 |
| 2) Channels III, IV | 10.0 | 5.83 | 1.71 + 2.60 (Temp + Press) | See Note 1 | See Note 2 |
| 8. Overpower ΔT | 4.8 | 1.24 | 1.71 | See Note 3 | See Note 4 |
| 9. Pressurizer Pressure-Low | 5.0 | 1.77 | 3.3 | ≥ 1900 psia | ≥ 1890 psia |
| 10. Pressurizer Pressure-High | 5.0 | 1.77 | 3.3 | ≤ 2385 psia | ≤ 2395 psia |
| 11. Pressurizer Water Level-High | 8.0 | 5.13 | 2.7 | ≤ 89% of instrument span | ≤ 90.7% of instrument span |
| 12. Reactor Coolant Flow-Low | 2.5 | 1.52 | 0.78 | ≥ 90% of loop design flow* | ≥ 89.1% of loop design flow* |
| 13. Steam Generator Water Level Low-Low | 18.10 | 16.64 | 1.50 | ≥ 18.10% of narrow range instrument span | ≥ 17.11% of narrow range instrument span |
| 14. General Warning Alarm | N.A. | N.A. | N.A. | N.A. | N.A. |
| 15. Low Shaft Speed - Reactor Coolant Pumps | 3.8 | 0.5 | 0 | ≥ 95.8% of rated speed | ≥ 92.5% of rated speed |

*Minimum Measured Flow Per Loop = 1/4 of the RCS Flow Rate Limit as listed in Section 3.2.3.1a (Four Loops Operating); 1/3 of the RCS Flow Rate Limit as listed in Section 3.2.3.2a (Three Loops Operating)

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

FOUR LOOPS OPERATING

LIMITING CONDITION FOR OPERATION

3.2.3.1 The indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows:

a. RCS total flow rate $\geq$ 371,920 gpm, and

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

Where:

1) $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$,

2) $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured value of $F^N_{\Delta H}$ should be used since Specification 3.2.3.1b. takes into consideration a measurement uncertainty of 4% for incore measurement,

3) $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER in the CORE OPERATING LIMITS REPORT (COLR),

4) $PF_{\Delta H}$ - The power factor multiplier for $F^N_{\Delta H}$ provided in the COLR, and

5) The measured value of RCS total flow rate shall be used since uncertainties of 2.4% for flow measurement have been included in Specification 3.2.3.1a.

APPLICABILITY: MODE 1.

ACTION:

With the RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore the RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

Attachment 3

Millstone Unit No. 3
Proposed Revision to Technical Specifications

Reactor Coolant System Flow Rate
Safety Assessment

January 1995

Millstone Unit No. 3
Proposed Revision to Technical Specifications
Reactor Coolant System Flow Rate

**Safety Assessment**

In support of the requested amendment, NNECO hereby provides a safety assessment which addresses a reduction in minimum measured flow (MMF) and thermal design flow (TDF) by 4%. The MMF is used in the technical specifications whereas the TDF is used in those accident analyses using the revised thermal design procedure. The MMF is equal to TDF plus flow uncertainties. The assessment addresses four-loop operation only.

**TRANSIENT ANALYSIS LOSS OF COOLANT ACCIDENT EVALUATION**

Large Break Loss of Coolant Accident

The current licensing basis large break loss of coolant accident (LOCA) analysis in Section 15.6.5 of the Millstone Unit No. 3 Final Safety Analysis Report (FSAR) was performed using the 1981 Evaluation Model with the BASH code. The current analysis assumed a TDF of 94,600 gpm/loop. A predicted peak cladding temperature (PCT) of 1974°F (analysis of record) resulted for the double-ended guillotine break with a discharge coefficient of 0.6 under minimum safeguards conditions. Decreasing the TDF impacts both the RCS loop flow rate and the operating temperature of the RCS. These two effects were evaluated for the large break LOCA analysis.

The reduction in RCS flow rate would have a negligible impact on the analysis results, since the RCS flow rate is dominated by the break after the initiation of a large break LOCA. Therefore, there is no reason to further assess the change due to the reduction in RCS loop flow rate.

The large break LOCA evaluation conservatively assumed the largest temperature change which occurred as a result of the reduced TDF. Applying a conservative PCT sensitivity to RCS temperature resulted in a 12°F penalty. In a letter dated April 28, 1994,[1] pursuant to 10CFR50.46(a)(ii), NNECO reported the changes to, and errors in, the emergency core cooling system (ECCS) evaluation model or

(1) J. F. Opeka letter to the U.S. Nuclear Regulatory Commission, "Millstone Unit No. 3 Reporting of Changes to, and Errors in, Emergency Core Cooling System Models or Applications," dated April 28, 1994.

application of the model for Millstone Unit No. 3. This letter included the penalty due to the proposed change in the RCS flow rate. It is concluded that the 10CFR50.46 acceptance criteria continue to be met.

Small Break LOCA

The current licensing basis small break LOCA analysis in Section 15.6.5 of the Millstone Unit No. 3 FSAR was performed using the NOTRUMP evaluation model. The analysis assumed a TDF of 94,600 gpm/loop. A predicted PCT of 1891°F resulted for the 3 inch break under minimum safeguards condition. Decreasing the TDF impacts both the RCS loop flow rate and the operating temperature of the RCS. These two effects were evaluated for small break LOCA analysis.

The reduction in RCS flow rate would have a negligible impact on the analysis results since the RCS flow rate is dominated by the break after the initiation of the small break LOCA. Therefore, there is no reason to further assess the change due to the reduction in the RCS loop flow rate.

The small break LOCA evaluation conservatively assumed the largest temperature change which occurred as a result of the reduced TDF. Applying a conservative PCT sensitivity to RCS temperature results in a 12°F penalty. In a letter dated April 28, 1994,[2] pursuant to 10CFR50.46(a)(3)(ii), NNECO reported the changes to, and errors in, the ECCS evaluation model or application of the model for Millstone Unit No. 3. This letter included the penalty due to the proposed change in the RCS flow rate. It is concluded that the 10CFR50.46 acceptance criteria continue to meet.

Hot Leg Switchover to Prevent Boron Precipitation

Post-LOCA hot leg switchover time is dependent upon the power level and the RCS, refueling water storage tank (RWST), and accumulator water volumes and boron concentrations. The reduced TDF will not increase the power level or affect the maximum boron concentration or water volumes assumed in the RCS, RWST, or accumulator. The reduction in TDF will have no adverse affect on the post-LOCA hot leg switchover time for Millstone Unit No. 3.

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(2) Ibid

Post-LOCA Long-Term Core Cooling

The initial RCS temperature impacts the RCS mass. However, since the reduced TDF impacts the temperature difference between T-hot and T-cold while $T_{avg}$ remains constant, there will not be a significant change to the RCS mass. The Millstone Unit No. 3 licensing basis post-LOCA long-term core cooling calculations are not impacted by the reduced TDF.

LOCA Evaluation Conclusions

For the calculations considered, a reduced TDF of 90,800 gpm/loop) will not result in any regulatory limit being exceeded for Millstone Unit No. 3.

**TRANSIENT ANALYSIS (NON-LOCA) EVALUATION**

The current Millstone Unit No. 3 non-LOCA safety analyses assume a total RCS TDF of 378,400 gpm.

The following discussion supports a 4% TDF reduction. Assumed in this evaluation is an unchanged $T_{avg}$ and a slightly lower steam generator (SG) steam pressure.

Excessive Load Increase Incident

The analysis presented in Section 15.1.3 of the Millstone Unit No. 3 FSAR describes the plant response to a 10% step increase in load from nominal full power conditions. Four cases are analyzed for this ANS Condition II event based on automatic versus manual rod control and minimum versus maximum reactivity feedback parameters. Reactor protection against an excessive load increase transient is provided by the power range high neutron flux, overpower ΔT and overtemperature ΔT reactor protection system signals. Each case showed that the minimum departure from nucleate boiling ratio (DNBR) remained above the safety analysis limit value. The reduction in RCS flow rate would impact the minimum DNBR. However, it has been determined that the core limits used in the calculation of the minimum DNBR remain valid for the 4% TDF reduction. Setpoints also remain valid. Since the transient conditions are not significantly altered by the TDF reduction, the conclusion that the departure from nucleate boiling (DNB) design basis is met remains valid.

Excessive Heat Removal Due to Feedwater System Malfunctions

Two cases are analyzed and described for this ANS Condition II event in Section 15.1.2 of the Millstone Unit No. 3 FSAR. A full

power case is used to determine the plant response to a large step increase in the feedwater flow to one steam generator; a zero power case examines a step increase in feedwater flow from zero to nominal full-load flow in one steam generator. For the full power case, the minimum DNBR is shown to remain above the safety analysis limit value. The zero power case demonstrates that the reactivity transient and, hence, the minimum DNBR is bounded by the rod withdrawal from subcritical event. The reduction in RCS flow rate would impact the minimum DNBR. However, it has been determined that the core limits used in the calculation of the minimum DNBR remain valid for the 4% TDF reduction. Since the transient conditions are not significantly altered by the TDF reduction, the conclusion that the DNB design basis is met remains valid.

The reactivity insertion of the zero power feedwater malfunction event would decrease slightly as the TDF decreases because the primary-to-secondary heat transfer capability will be slightly decreased. The results is less of a primary cooldown and less of a reactivity transient. The maximum reactivity insertion will remain bounded by the rod withdrawal from subcritical analysis.

The feedwater temperature reduction transient resulting from a failed fully open feedwater control valve will continue to be bounded by the excessive load increase event since the steam pressure and temperature will be maintained at essentially the same value; i.e., the enthalpies will remain unchanged.

Accidental Depressurization of the Main Steam System

This ANS Condition II event is initiated by the full opening of a single steam dump, relief, or safety valve from zero power conditions; the current analysis is described in Section 15.1.4 of the Millstone Unit No. 3 FSAR. The analysis confirms that the minimum DNBR remains above the safety analysis limit value.

Reductions in the RCS TDF potentially decrease the minimum DNBR calculated during the event. However, reduced flow rates result in less primary-to-secondary heat transfer and, consequently, less of a power increase. Sufficient DNB margin is available to account for the reduction in RCS flow rate such that the DNBR values calculated in the analysis of record remain unaffected.

Major Secondary-Side Pipe Rupture

For this ANS Condition IV event, the transient is assumed to be initiated by the instantaneous double-ended rupture of a main steam line while at hot zero power conditions. The current analysis is

described in Section 15.1.5 of the Millstone Unit No. 3 FSAR. Two cases (with and without offsite power) are considered.

The analysis demonstrates the minimum DNBR remains above the licensing limit value in each case. Reductions in the RCS TDF potentially decrease the minimum DNBR calculated for the event. This DNB penalty would be partially offset because the lower flow rate would lead to reduced primary-to-secondary heat transfer and, subsequently, less of a power increase. For the DNB evaluation, sufficient DNB margin is available to offset the effects of a reduced TDF on DNB calculations without crediting the effect of a reduced flow rate on the peak power achieved. Thus, the DNB design basis is met, and the results and conclusions presented in Section 15.1.5 of the Millstone Unit No 3 FSAR remain valid.

Steam system piping failures are also analyzed to ensure that the containment pressure and temperature remain within acceptable limits during a steam line break. This analysis is performed to predict the amount of mass and energy released during a steam line break. The mass and energy release rates are then used to calculate a containment response. A reduction in RCS flow rate would reduce the primary-to-secondary heat transfer and would, therefore, reduce the amount of energy available to be released to containment. Therefore, the conclusions presented in the Millstone Unit No. 3 FSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip

The analyses presented in Sections 15.2.2 and 15.2.3 of the Millstone Unit No. 3 FSAR represent a complete loss of steam load from full power without a direct reactor trip. Four cases are analyzed which are based on two different primary-side pressure control strategies (automatic versus none) and two sets of core physics characteristics (maximum versus minimum reactivity feedback). The analysis demonstrates that, with the power mismatch between the core and turbine, the primary and secondary system pressures remain below 110% of the design values and the minimum DNBR remains above the safety analysis limit value. Automatic reactor trip signals which may be generated during this event include high pressurizer pressure and overtemperature ΔT. The reduction in RCS flow rate can potentially impact the results of this analysis with respect to the minimum DNBR calculated. However, it has been determined that the core limits used in the calculation of the minimum DNBR continue to be valid while still accounting for a 4% TDF reduction. The overtemperature ΔT setpoint remains valid. Since the transient conditions are not significantly altered by the TDF reduction, the conclusion that the DNB design basis is met remains valid.

In all four cases, there is margin to the primary/secondary-side pressure limits. Furthermore, sensitivity studies have shown that the maximum pressures reached during this transient are insensitive to a change in reactor coolant flow rate of the magnitude specified in this evaluation. A reduction in RCS flow rate of about 4% results in the licensing basis criteria remaining valid following a loss of load event.

Loss of Offsite Power to the Station Auxiliaries

The analysis presented in Section 15.2.6 of the Millstone Unit No. 3 FSAR represents a complete loss of power to the plant auxiliaries (i.e., the RCPs, condensate pumps, etc) from full power. The low of power results in a heatup and pressurization of the primary and secondary systems. The analysis demonstrates that adequate auxiliary feedwater flow is delivered to the steam generators to remove decay heat such that DNB will not occur, overpressurization of the primary and secondary systems will not occur, and the pressurizer will not become water solid. Reductions in the RCS TDF decrease the minimum DNBR, increase the peak RCS pressure, and potentially lead to increased coolant expansion and a reduction in the margin to pressurizer filling.

When offsite power is lost, the RCPs coastdown and the RCS TDF will eventually reduce to natural circulation flow. The dominant driving force for natural circulation is the density difference between the fluid in the reactor vessel downcomer and the fluid within the core barrel (in the core and upper core plenum). This driving force will act to force flow through all of the reactor coolant loops. The reduction in TDF has an insignificant impact on this interaction.

The results of the complete loss of forced reactor coolant flow analysis and the loss of normal feedwater analysis continue to show that for a loss of all non-emergency AC power, no adverse conditions occur in the reactor core. As a result, the DNBR remains above the safety analysis limit value and the primary and secondary peak pressure licensing basis design criteria continue to be met. Pressurizer filling also will not occur; therefore, the conclusions for the loss of offsite power event which are documented in Section 15.2.6 of the Millstone Unit No. 3 FSAR remain valid.

Loss of Normal Feedwater

The loss of normal feedwater analysis in Section 15.2.7 of the Millstone Unit No. 3 FSAR presents the consequences of a complete loss of normal feedwater flow simultaneous to all four steam

generators. The loss of AC power event is similar except that the loss of offsite power also results in all four RCPs coasting down. These transients are analyzed to demonstrate that neither the primary nor secondary sides are overpressurized, that the core is not adversely affected, and the pressurizer does not fill.

The loss of normal feedwater event is sensitive to initial steam generator mass, as well as the mass in the steam generators at the time of reactor trip. Following the loss of normal feedwater, the reactor continues to operate until, due to the rapid loss of steam generator inventory and the continued heat transfer to the secondary side, it is tripped on a low-low steam generator level signal.

The effect of reducing the RCS flow rate would be an increase in the heatup of the RCS during the initial phase of the transient. The increased heatup results in a decrease in the coolant density which, in turn, would increase the pressurizer insurge during this heatup. However, considerable margin exists to filling the pressurizer in the initial portion of the transient such that pressurizer filling is not expected to occur. Thus, the primary- and secondary-side peak pressure licensing basis design criteria will continue to be met, the pressurizer will not go solid, and the conclusions made in Section 15.2.7 of the Millstone Unit No. 3 FSAR for the loss of normal feedwater event remain valid.

Feedwater System Pipe Break

For this ANS Condition IV event, the double-ended rupture of a main feedwater pipe initially results in a cooldown of the RCS due to the heat removal of the steam generator blowdown. The current analysis is provided in Section 15.2.8 of the Millstone Unit No. 3 FSAR. This cooldown period is followed by a heatup as the high levels of decay heat and the lack of inventory on the secondary side result in inadequate heat transfer. The event is analyzed to show that adequate auxiliary feedwater flow exists to remove core decay heat and stored energy following a reactor trip from full power and that the core remains in a coolable geometry and covered with water. For ease of interpreting the transient, the more restrictive criterion of no bulk boiling in the primary coolant system following a feedwater pipe break, prior to the time that the heat removal capacity of the steam generators being fed auxiliary feedwater exceeds NSSS generation, has been applied. This is determined by verifying that the RCS coolant remains subcooled.

Sensitivity studies performed for Millstone Unit No. 3 with a 4% TDF reduction show that the feedwater pipe break event is insensitive to reductions in RCS flow rates with respect to peak

hot leg temperatures. Other sensitivities performed with respect to initial steam generator masses show that a reduction in TDF results in a small decrease in the steam generator mass. The effect of the reduced mass is a small decrease in the minimum margin to hot leg saturation. The current Millstone Unit No. 3 licensing analysis basis contains sufficient margin to accommodate the penalty associated with a 4% lower TDF.

Partial Loss of Forced Reactor Coolant Flow

The partial loss of forced reactor coolant flow transient is a Condition II event which is analyzed under full power conditions assuming that 1 of 4 operating RCPs coasts down. The reactor is promptly tripped on the RCS low flow reactor trip. The current analysis provided in Section 15.3.1 of the Millstone Unit No. 3 FSAR demonstrates that the minimum DNBR remains above the safety analysis limit value.

The reduced RCS TDF will have a negative impact on the calculated DNBR because it is a critical parameter in DNBR determination. The reduction of RCS TDF may also produce an increase in the RCS moderator temperature, both of which tend to reduce the margin to the licensing basis DNB limit for this event. Statepoints used in the DNBR calculations are a fraction of nominal values and, since transient conditions will not be altered, these would not be impacted. It has been determined that the DNB design basis is met with some allocation of generic DNB margin. As a result, the reduced TDF does not alter the conclusions presented in the FSAR for the partial loss of forced reactor coolant flow event.

Complete Loss of Forced Reactor Coolant Flow

This Condition III event is analyzed under full power conditions assuming that 4-out of-4 operating RCPs coast down. The current analysis is provided in Section 15.3.2 of the Millstone Unit No. 3 FSAR. The reactor is assumed to trip on an RCP underspeed signal. The analysis demonstrates that the minimum DNBR remains above the safety analysis limit value. A decrease in the RCS flow rate potentially decreases the minimum DNBR calculated during the event.

The reduction in RCS flow would impact the minimum DNBR. Statepoints used in the DNBR calculations are a fraction of nominal values and, since transient conditions will not be altered, these would not be impacted. It has been determined that the DNB design basis is met with some allocation of generic DNB margin. Thus, the DNBR remains above the safety analysis limit value and the conclusions from the FSAR analysis also remain valid.

Reactor Coolant Pump Shaft Seizure (Locked Rotor)

This Condition IV event is analyzed under full power conditions assuming the instantaneous seizure of one RCP rotor using the LOFTRAN and FACTRAN computer codes. This results in a rapid RCS flow reduction and pressure rise which may lead to DNB. The reactor is promptly tripped on a low flow signal. The analysis demonstrates that the maximum reactor coolant system pressure is less than the limit value, the maximum fuel clad temperature is less than 2700°F, and the amount of zirconium-water reaction is small. The current analysis is provided in Section 15.3.3 of the Millstone Unit No. 3 FSAR.

The lower RCS flow rate will result in slightly higher system pressures than those calculated in the current FSAR analysis. The PCT analysis performed for the locked rotor event conservatively assumed that DNB occurs upon the initiation of the event. DNB significantly decreases fuel-to-clad heat transfer. This assumption maximizes the calculated PCT and minimizes the impact of a flow reduction since fuel-to-clad heat transfer is already substantially degraded.

Since the low flow setpoint is a fraction of the initial loop flow, a reduction in RCS flow rate will not impact the time of trip, and thus, the nuclear power and heat flux transients would not be adversely affected.

Generic DNB margin has been allocated to account for the adverse effects of a 4% TDF reduction on the current locked rotor analysis. There is significant margin to the system pressure limits, as well as to the PCT limit, to offset the negative effects of the proposed RCS flow reduction. Therefore, it can be concluded that the results presented in the Millstone Unit No. 3 FSAR pertaining to the locked rotor event remain valid.

Rod Withdrawal from a Subcritical Condition

For this Condition II event, rod withdrawal results in a rapid reactivity insertion and increase in core power potentially leading to high local fuel temperatures and heat fluxes and a reduction in the minimum DNBR. The current analysis is provided in Section 15.4.1 of the Millstone Unit No. 3 FSAR. The power excursion is terminated by Doppler feedback and then the transient is promptly terminated by a reactor trip on the Power Range High Neutron Flux - low setpoint. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow.

The reduction in TDF would result in a reduction to fuel-to-coolant

heat transfer and an increase in PCT. It has been determined that the DNB design basis is met with some allocation of generic DNB margin. The transient conditions will not be significantly altered. Thus, the conclusions for this event, as presented in the FSAR, remain valid.

Rod Withdrawal at Power

For this Condition II event, three initial power levels (i.e., 100%, 60%, and 10%) and a range of reactivity insertion rates assuming two (minimum and maximum) reactivity feedback conditions are analyzed in Section 15.4.2 of the Millstone Unit No. 3 FSAR. The resulting power excursion produced by a rod cluster control assembly (RCCA) withdrawal at power results in high local fuel temperatures and an increase in the core heat flux. Since the heat extraction capability of the steam generator lags behind the core power generation, a net increase in the moderator temperature occurs. The resulting power mismatch and increase in reactor coolant temperature can result in DNB unless the transient is terminated by either manual or automatic means.

Automatic reactor protection is provided via the high neutron flux reactor trip for a rapid activity insertion and the overtemperature ΔT reactor trip for slower reactivity insertion cases. The licensing basis analysis presented in Section 15.4.2 of the Millstone Unit No. 3 FSAR ensures that fuel damage will not occur by demonstrating that the minimum DNBR remains above the safety analysis limit value.

The reduction in RCS flow rate would impact the minimum DNBR. However, it has been determined that the core limits used in the calculation of the minimum DNBR for a 4% TDF reduction remain valid. Since the transient conditions are not significantly altered by the TDF reduction and the overtemperature ΔT setpoint remains valid, the conclusion that the DNB design basis is met remains valid.

Rod Cluster Control Assembly Misalignment

This Condition II event is analyzed to demonstrate that following various RCCA misoperation events, such as dropped rod(s)/bank or statically misaligned rods, the minimum DNBR remains above the safety analysis limit value. The reduction in RCS flow potentially impacts the RCCA misoperation events by changing the initial condition assumptions used in this analysis. Reductions in the RCS flow rate serve to reduce the margin to the DNB licensing basis limit.

Allocation of the generic DNB margin ensures that the DNB licensing basis criteria will continue to be met and the conclusions in Section 15.4.3 of the Millstone Unit No. 3 FSAR remain valid.

Uncontrolled Boron Dilution

This Condition II event is analyzed for five modes of plant operation. The current analysis is provided in Section 15.4.6 of the Millstone Unit No. 3 FSAR. The analysis demonstrates that sufficient shutdown margin exists, such that should a dilution event occur, there is sufficient time following the start of dilution to allow operator detection and termination of the event prior to a complete loss of shutdown margin. This event is analyzed for operating Modes 1 through 5. Mode 6 is not analyzed due to implementation of administrative procedures which prevent the accident from occurring in this mode of operation.

The TDF is not an input to the boron dilution analysis. Therefore, the conclusions presented in Section 15.4.6 of the Millstone Unit No. 3 FSAR remain unaltered with the reduced RCS flow rate.

Rupture of a Control Rod Drive Mechanism Housing

For this Condition IV event, a rapid reactivity insertion and increase in core power leads to high local fuel and clad temperatures and possible fuel and/or clad damage. The current analysis is provided in Section 15.4.8 of the Millstone Unit No. 3 FSAR. The RCCA ejection analysis is analyzed at four conditions: beginning and end-of-life core physics characteristics, at hot zero power, and full power. The analysis demonstrates that gross fuel damage will not occur, that the core will remain in a coolable geometry, and that the RCS will remain intact. In order to demonstrate that these criteria are met, the following more restrictive criteria are applied.

1. The average full pellet enthalpy at the hot spot is less than 200 cal/gm (360 Btu/lbm).

2. Fuel melt at the hot spot is limited to less than the innermost 10% of the fuel pellet.

3. Peak RCS pressure is less than which would cause stresses to exceed the faulted condition stress limits.

The rod ejection event is characterized by a rapid power excursion terminated by Doppler feedback. The reactor is tripped on high neutron flux (low setting for the zero power case, high setting for the full power case). A reduction in RCS flow rate will result in

a reduction in the fuel rod-to-coolant heat transfer. This may result in an increase in the calculated fuel and clad temperatures, as well as the fuel stored energy during an RCCA ejection.

As shown in the FSAR, the full power cases result in the highest fuel pellet temperatures and are the most limiting with respect to criteria 1 and 2. Examination of these cases reveals that, due to the rapid power and fuel temperature rise, coupled with the thermal lag in the fuel pellet itself, the time at which the maximum pellet enthalpy and fuel melt are calculated to occur in before any significant amount of heat has reached the coolant. A sensitivity analysis, which used methods consistent with WCAP-7588, Rev. 1, demonstrated that for a 2% reduction in thermal design flow, there was only a minor change (~2 cal/gm) to the maximum pellet enthalpy and fuel melt results for the full power rod ejection cases. This sensitivity showed only an 8°F increase in the maximum fuel center temperature. There is sufficient margin in the Millstone Unit No. 3 analysis to absorb the differences in results from the sensitivity. Therefore, the 4% reduction in TDF satisfies the safety criteria for the full power cases.

The zero power rod ejection cases are characterized by a sharp increase in the clad average temperature. However, the sensitivity which addresses a 2% reduction in TDF showed only a slight increase (< 1%) in the maximum clad average temperature. There exists enough margin in the Millstone Unit No. 3 analyses to absorb the differences created from the reduction in TDF. Therefore, the 4% reduction in TDF would not produce a significant increase in the Millstone Unit No. 3 maximum PCT.

The analysis of the peak pressure transient for the RCCA ejection event is discussed in WCAP-7588, Rev. 1. A reduction in RCS flow could increase the primary-side pressurization by reducing primary-to-secondary side heat transfer. However, due to the rapid nature of this event, any second-side heat removal will lag well behind the heat addition to the primary side.

Thus, a 4% flow reduction will only have a minimal impact on the primary-side peak pressure. There is more than sufficient margin to the faulted condition stress limits to accommodate a 4% reduction in the RCS flow.

Based upon the preceding discussions, a 4% reduction in RCS flow rate satisfies the licensing basis criteria following a RCCA ejection event, and the conclusions of the FSAR remain valid.

Spurious Operation of Safety Injection System at Power

A spurious safety injection system (SIS) signal is an ANS Condition II event which is assumed to be initiated at full power. The current analysis is provided in Section 15.5.1 of the Millstone Unit No. 3 FSAR. The injection of highly concentrated borated water into the RCS reduces core power, temperature, and pressure until the reactor trips on low pressurizer pressure.

The RCS power and temperature reductions produce a similar reduction in pressure on the second side of the plant. The current Millstone Unit No. 3 analysis demonstrates that the minimum DNBR remains above the safety analysis limit value. The reduction in RCS flow rate would impact the minimum DNBR. However, it has been determined that the core limits used at the calculation of the minimum DNBR remain valid even after accounting for a 4% TDF reduction. Since the transient conditions are not significantly altered by the TDF reduction, the conclusion that the DNB design basis is met remains valid.

Accidental Depressurization of the Reactor Coolant System

For this ANS Condition II event, the transient is initiated by the opening of a single pressure relief or safety valve while the reactor is at full power. The current analysis is presented in Section 15.6.1 of the Millstone Unit No. 3 FSAR. Initially, the RCS pressure drops rapidly until a reactor trip occurs on either the pressurizer low pressure or overtemperature ΔT reactor protection signals. At this time, the pressure decrease continues, but at a much slower rate. The analysis demonstrates that the minimum DNBR remains above the safety analysis limit value.

The reduction in RCS flow rate would impact the minimum DNBR. However, it has been determined that the core limits used in the calculation of the minimum DNBR remain valid even after accounting for a 4% TDF. Since the transient conditions are not significantly altered by the TDF reduction and the overtemperature ΔT setpoint is unchanged, the conclusions that the DNB design basis is met remains valid. Thus, the conclusions in the FSAR for this event remain valid.

Steam Line Break Mass/Energy Release - Inside/Outside Containment

Various steam line break cases are analyzed in Section 15.1.5 of the Millstone Unit No. 3 FSAR for the purposes of generating mass and energy release rates which are then applied to containment response or compartment environmental analyses. Cases are performed assuming various break sizes and initial power levels.

Four major factors influence the release of mass and energy following a steam line break. These are steam generator fluid inventory, protection system operation, state of the secondary fluid blowdown, and primary-to-secondary heat transfer. A decrease in RCS flow rate would tend to reduce the primary-to-secondary heat transfer, thereby reducing the steam pressure and temperature during normal operation. Any reduction in the secondary-side temperature and pressure would tend to lessen the mass and energy released during a steam line break event. As a result, a 4% reduction in RCS flow rate would not adversely affect the steam line break mass/energy releases. These statements are supported by the discussion in WCAP-1091, Rev. 1.

Therefore, the conclusions of the current steam break mass/energy release calculations are considered to be applicable for the reduced RCS flow rate scenario.

Overpower ΔT and Overtemperature ΔT Setpoint Impact

The impact of reduced TDF on the Millstone Unit No. 3 overpower and overtemperature reactor protection functions has been addressed, and it has been determined that the current overpower ΔT/ overtemperature ΔT setpoint equations continue to protect the core under the reduced TDF conditions.

Transient Analysis Evaluation Conclusion

Based on the discussion above, it is concluded that the Millstone Unit No. 3 licensing basis non-LOCA safety analyses are not adversely affected by a 4% reduction in the RCS thermal design flow. The conclusions presented in the Millstone Unit No. 3 FSAR remain valid.

**STEAM GENERATOR TUBE RUPTURE EVALUATION**

The FSAR steam generator tube rupture (SGTR) analysis to determine the offsite radiological consequences used the LOFTRAN computer code. The current analysis is provided in Section 15.6.3 of the Millstone Unit No. 3 FSAR. In 1991, a design basis SGTR analysis was performed for Millstone Unit No. 3 to demonstrate margin to steam generator overfill, using the LOFTTR2 computer code.[3][4]

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(3) J. L. Stackhouse, "Margin to Overfill Analysis for a Steam Generator Tube Rupture for Millstone Nuclear Power Station, Unit 3, Four-Loop Operation, WCAP-13002," dated August 1991.

The SGTR evaluation for a 4% reduction in TDF was performed for both the FSAR analysis and for the margin to steam generator overfill analysis. This evaluation included the changes to other RCS and secondary-side parameters associated with the 4% reduction in TDF.

The reduction in the TDF is expected to reduce the reactor trip time on the overtemperature ΔT trip which is a penalty for the SGTR analyses. The effect of the reduced trip time was also evaluated.

The margin to steam generator overfill will be reduced by the TDF reduction. Still, the conclusion in WCAP-13002 that there is margin to steam generator overfill for Millstone Unit No. 3 will not change.

The results of the FSAR SGTR analysis for primary-to-secondary break flow and steam released to the atmosphere will increase slightly, but the conclusion that the radiological dose limits are less than a small percent of the 10CFR100 guidelines remains valid.

Based on the results of the SGTR evaluation for the 4% reduction in TDF for Millstone Unit No. 3 four loop operation, it is determined that the conclusion in Reference 4 and the FSAR remain unchanged.

**CORE DESIGN EVALUATION**

Core design has evaluated the effects of up to a 4.5% reduction in the RCS flow rate for Millstone Unit No. 3, Cycle 6.

**MECHANICAL SYSTEMS AND COMPONENTS EVALUATION**

A change in TDF of 4% is equivalent to approximately a 1°F change in RCS ΔT at full power. This is equivalent to less than a 1.5°F change in T-cold and T-hot. Based on Westinghouse experience with numerous rerating and T-hot reduction programs which included flow reduction of up to 5%, it is judged that such a small change in these temperature and flow parameters will have an insignificant effect on the design transients. Thermal/hydraulic parameter changes of a significantly higher magnitude (due to tube plugging, uprating, T-hot reduction, etc.) have been defined on other plants. In no case have unacceptable fatigue stress results been obtained. Previous studies addressed an assessment of the nuclear steam supply system primary components, including the reactor pressure

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(4) J. F. Opeka letter to the U.S. Nuclear Regulatory Commission, "Steam Generator Tube Rupture Analysis (TAC No. 67054)," dated April 28, 1992.

vessel, reactor vessel internals, reactor coolant pump, steam generator, pressurizer, loop isolation valves, control rod drive mechanisms, and the RCS piping and supports.

Based on the above, it is judged continued operation will remain in compliance with all applicable requirements for a reduced TDF of 4%.

**Conclusion**

There is sufficient margin to the system pressure, PCT and DNB limits to offset the negative effects of the change. Therefore, it is concluded that the limits for the FSAR accident analysis results are still met.

In summary, the proposed 4% decrease in the Technical Specification limit for total RCS flow rate will not adversely affect the results of the FSAR accident analysis and it is concluded that this change is safe.

The change does not adversely affect any equipment credited in the safety analysis. Also, the change has a negligible impact on the calculated peak clad temperature (PCT) and it does not increase the offsite doses or decrease the DNB below its acceptance limit. Therefore, there is no impact on the margin of safety as specified in the technical specifications.