Docket No. 50-423
B14653

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Pressurizer Safety Valves and Main Steam Safety Valves
Lift Setting Tolerance change

Marked Up Pages of Technical Specifications

December 1993

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3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.* **

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

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

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

** ±1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ±3% of the setpoint.

OPERATING

1

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%.* **

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 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.

** ±1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ±3% of the setpoint.

TABLE 3.7-3

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STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (± 1%)**	ORIFICE SIZE
LOOP 1		
RV22A	1185 psig	16.0 square inches
RV23A	1195 psig	16.0 square inches
RV24A	1205 psig	16.0 square inches
RV25A	1215 psig	16.0 square inches
RV26A	1225 psig	16.0 square inches
LOOP 2		
RV22B	1185 psig	16.0 square inches
RV23B	1195 psig	16.0 square inches
RV24B	1205 psig	16.0 square inches
RV25B	1215 psig	16.0 square inches
RV26B	1225 psig	16.0 square inches
LOOP 3		
RV22C	1185 psig	16.0 square inches
RV23C	1195 psig	16.0 square inches
RV24C	1205 psig	16.0 square inches
RV25C	1215 psig	16.0 square inches
RV26C	1225 psig	16.0 square inches
LOOP 4		
RV22D	1185 psig	16.0 square inches
RV23D	1195 psig	16.0 square inches
RV24D	1205 psig	16.0 square inches
RV25D	1215 psig	16.0 square inches
RV26D	1225 psig	16.0 square inches

** ±1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ±3% of the setpoint.

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

Docket No. 50-423
B14653

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Pressurizer Safety Valves and Main Steam Safety Valves
Lift Setting Tolerance change

Retyped Pages of Technical Specifications

December 1993

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2500 psia $\pm$1%.*, **

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 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.

** ± 1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ± 3% of the setpoint.

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2500 psia $\pm$1%.*, **

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

** ± 1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ± 3% of the setpoint.

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING (±1%)*,**	ORIFICE SIZE
LOOP 1		
RV22A	1185 psig	16.0 square inches
RV23A	1195 psig	16.0 square inches
RV24A	1205 psig	16.0 square inches
RV25A	1215 psig	16.0 square inches
RV26A	1225 psig	16.0 square inches
LOOP 2		
RV22B	1185 psig	16.0 square inches
RV23B	1195 psig	16.0 square inches
RV24B	1205 psig	16.0 square inches
RV25B	1215 psig	16.0 square inches
RV26B	1225 psig	16.0 square inches
LOOP 3		
RV22C	1185 psig	16.0 square inches
RV23C	1195 psig	16.0 square inches
RV24C	1205 psig	16.0 square inches
RV25C	1215 psig	16.0 square inches
RV26C	1225 psig	16.0 square inches
LOOP 4		
RV22D	1185 psig	16.0 square inches
RV23D	1195 psig	16.0 square inches
RV24D	1205 psig	16.0 square inches
RV25D	1215 psig	16.0 square inches
RV26D	1225 psig	16.0 square inches

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

** ± 1% applies when a safety valve is tested and set prior to declaring it operable. When a safety valve is tested in accordance with Specification 4.0.5, the as-found setting shall be considered to be acceptable when the setting is within ± 3% of the setpoint.

Docket No. 50-423
B14653

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Plant Safety Evaluation for Millstone Generating Station Unit 3
VANTAGE 5H Fuel Upgrade

December 1993

5. ACCIDENT ANALYSIS

5.0 INTRODUCTION

5.0.1 General

This section contains a discussion of the accident analyses which were performed to address the following:

- Transition from 17X17 Standard fuel to VANTAGE 5H fuel
- Implementation of fuel/core related changes
- Implementation of plant related changes
- Incorporation of contemplated plant changes into the analyses

The analyses presented in this section include:

- 5.1 Non-LOCA
- 5.2 LOCA
- 5.3 Mass and energy release and containment integrity
- 5.4 Radiological consequences
- 5.5 Margin to trip

These analyses address transition core effects as well as N-1 loop operation.

The impact of the following items have been incorporated into the above analyses; but, separate sections have been devoted to them to discuss their impact more fully:

- 5.6 RWST boron concentration increase
- 5.7 RCCA parked position change

Section 5.8 provides a summary of the instrumentation uncertainty calculations which were performed in support of the change to the Revised Thermal Design Procedure.

Section 5.9 provides an integrated safety evaluation which specifically confirms the acceptability of these changes using the criteria of 10CFR50.59.

5.0.2 VANTAGE 5H Design Features

The design features of the VANTAGE 5H fuel assembly that were considered in the safety analysis effort include the following:

- Fuel Rod Dimensions
- Intermediate Flow Mixer (IFM) Grids
- Axial Blankets
- Integral Fuel Burnable Absorbers (IFBAs)

- Debris Filter Bottom Nozzle (DFBN)
- Reconstitutable Top Nozzle (RTN)
- Zircaloy Grids

A brief description of each of these items and its treatment in the safety analyses follows.

Fuel Rod Dimensions

The fuel rod dimensions which determine the temperature versus linear power density relationship include rod diameter, pellet diameter, initial pellet-to-clad gap size, and stack height. As shown in Section 2, there are no changes in these parameters for the VANTAGE 5H design versus the standard design. The fuel temperature and rod geometry assumptions used in the safety analysis bound both the 17X17 STANDARD and VANTAGE 5H fuel.

IFM Grids

The IFM Grid used in the VANTAGE 5H fuel assembly design provides an increase in DNB margin over standard fuel. As a result, the VANTAGE 5H fuel safety analysis limit DNBR values contain significant DNB margin (see Section 4). For the transition cycles the core limits dictated by the 17X17 STANDARD fuel without IFM grids are more restrictive than those for the VANTAGE 5H fuel. Any transition core penalty resulting from the presence of the two fuel types is accounted for with the available DNBR margin.

The IFM grid feature of the VANTAGE 5H fuel design increases the core pressure drop. One result is that the control rod scram time to the dashpot has been increased from 2.2 to 2.7 seconds. This increased drop time primarily affects the fast reactivity transients, all of which were explicitly reanalyzed for this report. The increased scram time was modeled in all reanalyzed events and the remaining transients have been evaluated.

Axial Blankets and IFBAs

Axial blankets reduce power at the top and the bottom of the rod, thereby increasing axial power peaking at the center of the rod. This effect is offset by the presence of the part length IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects of axial blankets and IFBAs on the reload safety analysis parameters are taken into account in the reload design process. The axial power distribution assumptions in the safety analysis kinetics calculations for this report are sufficiently bounding to accommodate the presence of IFBAs and axial blankets. Although both these features are part of the VANTAGE 5H fuel product for Millstone Unit 3, they are not new features for this plant since they were introduced for Cycle 3 operation.

RTNs and DFBNs

RTNs and DFBNs are features that have been used extensively in Westinghouse designs. Analyses and tests have been performed that confirm the hydraulic compatibility of these particular designs to existing designs so that these components do not impact any parameters important to the safety analysis. Similar nozzle designs have been previously used at Millstone Unit 3 and were introduced for Cycle 3 operation (see Section 2).

Zircaloy Grids

Zircaloy structural grids have replaced Inconel grids in the VANTAGE 5H fuel assembly with the exception of the top and bottom grids which remain Inconel. The effects of Zircaloy grids have been incorporated into the safety analysis.

5.0.3 Fuel/Core Related Features and Assumptions

The following core related features and assumptions have been incorporated into the safety analysis.

- Increase in FdH and Fq
- Reactivity coefficient changes
- Revised thermal design procedure
- Revised non-RTDP uncertainties
- Relaxed axial offset control
- Reduced shutdown margin
- Thimble plug removal
- Modified OTΔT and OPΔT trip setpoints
- RCCA parked position change
- RWST boron concentration increase
- Deletion of clad temperature acceptance criterion for RCCA ejection event

A discussion of the effect which these items have on the safety analysis is given below.

Increased Power Distribution Peaking Factors ($F_{\Delta H}$ & F_Q)

The non-LOCA safety analyses have conservatively modeled a full power $F_{\Delta H}$ for both the 17X17 STANDARD and VANTAGE 5H fuel of 1.64 for RTDP applications and 1.70 for non-RTDP. A value of $F_{\Delta H}$ = 1.7 has been used in the LOCA analyses.

The Technical Specification maximum F_Q for four loop operation has been increased from 2.32 to 2.60 for both the 17X17 STANDARD and the VANTAGE 5H fuel. The three loop maximum value of F_Q has increased from 2.25 to 3.0 (75% power).

Reactivity Coefficient Changes

To accommodate longer fuel cycles and extended fuel burnup, a moderator density coefficient of .50 Δρ/gm/cc corresponding to end-of-cycle, full power conditions, was conservatively incorporated into the safety analyses performed for this report.

The minimum feedback Doppler power coefficient as a function of power (see the lower bound case in Figure 5.1.1-1) has been reduced (i.e. smaller absolute value) in the non-LOCA analysis. The Doppler power coefficient remained unchanged for the LOCA analysis.

The limiting case for the complete loss of flow event is analyzed using a 0.0 pcm/°F full power moderator temperature coefficient. The reference licensing basis analysis for this event used an overly conservative value of +5.0 pcm/°F at full power.

Revised Thermal Design Procedure (RTDP)

The calculational method utilized to meet the DNB design basis is the RTDP (see Section 5.8), which is described in Reference 10. Conservative uncertainties in the plant operating parameters are statistically incorporated in the design limit DNBR value. The conservative values are +6.0/-6.6°F for reactor coolant system average temperature, +50/-53 psi for pressurizer pressure, ±2.0% of rated thermal power (RTP) for core power (N loop analysis), ±2.3% RTP for core power (N-1 loop analysis), ±2.4% flow for reactor coolant system flow (N loop analysis), and ±2.8% flow for reactor coolant system flow (N-1 loop analysis). The calculated uncertainties are reported in Reference 43. Since the parameter uncertainties are considered in determining the design DNBR value, the new methodology permits the associated plant safety analyses to be performed using nominal input parameters without uncertainties.

Revised Non-RTDP Uncertainties

The use of RTDP represents a change in the methodology for applying uncertainties for those events used to show the DNB design basis is met. However, the change in power, pressurizer pressure, reactor coolant system average temperature, and RCS flow uncertainty allowances from previously used values must be evaluated for those events using Standard Thermal Design Procedure (i.e., non-RTDP). The core power uncertainty used for three loop operation is ±2.3% RTP as compared to the ±2.0% RTP in the current licensing basis (the core power uncertainty used for N loop operation remains ±2.0% of rated thermal power). The LOCA analysis continued using ± 2.0% RTP as the core power uncertainty for three loop operation. The reactor coolant system average temperature uncertainty allowance is +6.0/-6.6°F as compared to the previous ±6.1°F. The conservative pressurizer pressure uncertainty is +50/-53 psi as compared to the ±45 psi noted currently in the FSAR. The higher pressurizer pressure uncertainty resulted in using some of the existing analysis PCT margin. Specifically, 0.2°F of large break margin and 13.4°F of small break margin was utilized. Finally, the conservative value of the uncertainty for reactor coolant system flow for N loop operation is ±2.4% flow

versus the previous ± 1.8% flow and for N-1 loop operation is ± 2.8% flow versus ± 2.0% flow.

The values of these uncertainties used in the analysis increased in order to conservatively bound plant data which has recently been obtained.

Relaxed Axial Offset Control (RAOC)

For four loop conditions, RAOC operation with a + 12%, -17% ΔI band at 100% Rated Thermal Power (RTP) formed the basis of the non-LOCA safety evaluations. For three loop conditions, a RAOC ΔI band of + 10%, -15% at 75% RTP formed the basis for the non-LOCA safety evaluations. The LOCA analyses used a + 13%, -20% ΔI band for both 4 and 3 loop operation.

Reduced Shutdown Margin

The shutdown margin considered in the non-LOCA safety analysis was reduced from 1.6% ΔK to 1.3% ΔK.

Thimble Plug Removal

Thimble plug removal affects the core pressure drops and increases core bypass flow. These effects have been conservatively incorporated into the safety analyses performed for this report. The analyses are actually bounding for either the thimble plugs in place or removed. There are no restrictions in placement of the remaining thimble plugs in the core in this situation.

Modified Over-Temperature ΔT and Over Power ΔT Trip Setpoints

The implementation of Vantage 5H fuel and the inclusion of more conservative uncertainty values in temperature and pressure cause the DNB core limits to change. With the core limit change and implementation of RAOC, the over-temperature and over-power ΔT reactor trip setpoints are changed in the analysis. The values of the setpoints are changed in the analysis. The values of the setpoints are given in Table 5.1.1-4.

RCCA Parked Position Change

The safety analysis performed for this report included consideration of the RCCA full-out position being in the range from 222 to 231 steps.

RWST Boron Concentration Increase

The non-LOCA safety analysis provided in this report considered a range in the RWST boron concentration of from 2500 to 2900 ppm. The LOCA analysis is considered a range in RWST boron concentration of 2700 to 2900 ppm. These values will support the expected Technical Specification limits for this parameter.

Deletion of Clad Temperature Acceptance Criterion for the RCCA Ejection Event

The analysis of the RCCA ejection event for the VANTAGE 5H transition does not include consideration of the 2700°F clad average temperature criterion previously applied to this accident, as described in FSAR Section 15.4.8.1. The deletion of this criterion is not related to the fuel change. Rather the analysis required to support the fuel change provided an opportunity to implement the deletion which was actually explained and justified in Reference 28.

5.0.4 Plant/Tech Spec Related Changes to be Implemented

The following plant changes have been addressed completely in this report and can be implemented:

- Expanded accumulator level band
- Revised pressurizer level program
- Response time assumed for the high pressurizer water level reactor trip

A discussion of the effect of these changes on the safety analysis is given below.

Expanded Accumulator Level Band

A nominal accumulator water volume of 912.5 ft^3 was considered in the non-LOCA safety analysis. The LOCA analysis used 900 ft^3 per accumulator and have evaluated the impact of 912.5 ft^3.

Revised Pressurizer Level Program

The non-LOCA safety analysis used a revised pressurizer level program that specified nominal levels of 61.5% and 28% indicated span for full and zero power conditions, respectively. Previously, the nominal zero power level had been 25%. The LOCA analysis used a value of 62% for full power conditions.

Response Time Assumed for the High Pressurizer Water Level Reactor Trip

The safety analysis performed for the VANTAGE 5H transition assumed a response time of 2.0 seconds for the high pressurizer water level reactor trip. The previous licensing basis analysis did not require a response time for this function and the current Technical Specification indicate "N.A." as the associated response time for this trip function. The assumption of the response time in the analysis presented in this report dictates that a corresponding Technical Specification change be made to require the 2.0 second response.

5.0.5 Additional Assumptions Incorporated into Safety Analysis

The impact of the following items on the results of the safety analysis has been specifically determined. These additional assumptions are conservative and require no Tech Spec changes for the analysis to be valid. However, additional work is required before these Tech Spec changes can be implemented at the plant.

- Increased safety valve drift (Pressurizer and Steam Generator)
- Revised auxiliary feedwater flow rates
- Negative flux rate trip deletion
- 10% steam generator tube plugging
- Degraded safety injection and charging parameters
- Modified N-1 loop conditions
- Modified safety injection system flow source for the steamline break event
- Modified setpoints

A discussion of these items is given below.

Increased Safety Valve Drift (Pressurizer & Steam Generator)

The non-LOCA safety analyses performed for the VANTAGE 5H transition effort include revisions in the treatment of both the pressurizer and steam generator safety valves. With regard to the pressurizer safety valves, they were modeled as not opening until pressurizer pressure reached 2575 psia. The flow through the pressurizer safety valves was modeled with 3% accumulation, i.e., it ramps up from zero to full rated flow over the range of 2575 to 2652 psia (2500 x 1.03 x 1.03). Previously the pressurizer safety valves had been modeled as opening with the pressurizer pressure at 2500 psia and full rated flow being reached at 2575 psia.

For events that may challenge the secondary system pressure protection, the non-LOCA analysis in this report employed a model that allowed steam generator pressure to rise to 1320 psia, at which point it was assumed that sufficient safety valve capacity exists to stabilize the pressure and prevent further increases. The 1320 psia figure was selected as a conservative bounding value since it is equal to 110% of the steam generator design pressure. Previous analyses had used a model that allowed steam generator pressure to rise to 1236 psia, which represented the design pressure with a 3% allowance for accumulation.

The current FSAR analysis auxiliary flow rates correspond to a steam generator pressure of 1236 psia which reflects the assumed capability of the steam generator safety valves. However, as noted above, the analyses for the VANTAGE 5H transition use a more conservative model that allows the steam generator pressure to reach 1320 psia.

The LOCA analyses are not affected by the increased pressurized safety valve drift. The 3% increase in the steam generator safety valve drift was considered in both the large and small break LOCA analysis.

Revised Auxiliary Feedwater Flow Rates

For the loss of non-emergency AC power to the station auxiliaries (Section 5.1.4.6), loss of normal feedwater (Section 5.1.4.7), and feedwater system pipe break (Section 5.1.4.8) events, the cases analyzed for the VANTAGE 5H transition assume auxiliary feedwater flow rates similar to the current licensing basis FSAR analysis. Specifically, the four loop loss of normal feedwater analysis uses an auxiliary feedwater flow rate of 510 gpm delivered to four steam generators, while the four loop feedwater system pipe break assumed 480 gpm delivered to three steam generators. The 510 gpm four loop value represents a slight reduction from the 520 gpm used in the current loss of normal feedwater FSAR analysis. The three loop feedwater system pipe break assumes 300 gpm delivered to two steam generators.

In addition, the analyses performed for steam system piping failure (FSAR Section 15.1.5) and the inadvertent opening of a steam generator relief or safety valve (FSAR Section 15.1.4) assumed a reduction in the maximum auxiliary feedwater flow rate that can be delivered to the steam generators. The current licensing basis steam system piping failure analysis assumes a total auxiliary feedwater flow rate of 2850 gpm delivered to the faulted steam generator for four-loop operation and 2529 gpm to two steam generators (2229 gpm to the faulted) for three-loop operation. The VANTAGE 5H analyses assume 1200 gpm and 900 gpm delivered to one steam generator for four-loop and three-loop operation, respectively. The current licensing basis analysis for inadvertent opening of a steam generator relief or safety valve assumes a total of 2850 gpm delivered to three steam generators (2229 gpm to the faulted) for four-loop operation and 2529 gpm to two steam generators (2229 gpm to the faulted) for three-loop operation. The VANTAGE 5H analyses assume 1200 gpm delivered to four steam generators and 900 gpm delivered to three steam generators for four-loop and three-loop operation, respectively.

Negative Flux Rate Trip Deletion

The analysis for the dropped RCCA event (FSAR Section 15.4.3) did not credit the intervention of the power range negative flux rate trip function. Previous Millstone Unit 3 licensing basis analysis had taken credit for this trip.

10% Steam Generator Tube Plugging

All of the events reanalyzed for this report have incorporated modeling assumptions that bound up to a maximum of 10% steam generator tube plugging (or the hydraulic equivalent of plugs and sleeves) in each steam generator. It is assumed that no one steam generator exceeds 10% tube plugging.

Degraded Safety Injection (SI) and Charging Parameters

The non-LOCA analyses performed for steam system piping failure (FSAR Section 15.1.5), the inadvertent opening of a steam generator relief or safety valve (FSAR Section 15.1.4),

and feedwater system pipe break (FSAR Section 15.2.8) assume a 42.0 second delay in SI initiation. This delay time was conservatively assumed for the cases with offsite power available as well as for those considering the loss of offsite power. Diesel generator starting and sequence loading delays are included in the 42.0 seconds. The large break LOCA analysis incorporated a safety injection delay time of 40 seconds. The small break LOCA analysis incorporated a safety injection delay time of 45 seconds.

A 10% reduction in SI and charging flow was incorporated in the LOCA and non-LOCA analysis as well as a 10 gpm SI and charging pump flow imbalance.

Modified N-1 Loop Condition

For all safety analyses performed for this report, 75% of the four loop Rated Thermal Power was the assumed nominal "full power" for three loop operation. This is a change from the current licensing basis for Millstone Unit 3, where all three loop non-LOCA analyses used this assumption, with the exception of the rods in DNB evaluation for the locked rotor event that assumed a nominal three loop power of 65%. The previous LOCA analysis had also used a three loop power of 65%.

A decrease in the reactor coolant system thermal design flow from 99,600 to 98,300 gpm per loop was considered in the safety analyses. With the use of RTDP for the VANTAGE 5H transition analysis, the non-LOCA analyses performed to confirm the DNB design basis use minimum measured flow, which for three loop operation at Millstone Unit 3 is 101,066 gpm per loop. These events are unaffected by the decrease in thermal design flow. For all three loop non-LOCA analyses that consider thermal design flow, the reduction in flow was either addressed explicitly via reanalysis or evaluated.

Modified Safety Injection System Flow Source for the Steamline Break Event

Previous steamline break analysis had taken credit for minimum safety injection flow as provided by a single high head centrifugal charging pump. The analyses performed for this report instead credit the minimum safety injection flow as provided by a single high head safety injection pump. Using either of these assumed water sources represents a conservative analysis model.

Modified Setpoints

The high steam generator water level setpoint for the safety analysis was conservatively assumed to be at 100% narrow range span (NRS), in contrast to the previous safety analysis setpoint of 85.7% NRS. Using the 100% NRS figure in the analysis was intended to provide maximum operational flexibility for the plant.

A reduced setpoint for safety injection initiation and steamline isolation on low-steam pressure of 435 psia was used in the safety analyses for this report as compared to the previous 444 psia setpoint.

The setpoint for safety injection on low pressurizer pressure was reduced from 1680 psia to 1600 psia for the non-LOCA safety analysis. The large break LOCA analysis used 1600 psia for the reactor trip and safety injection setpoint. The small break LOCA analysis used 1600 psia as the low pressurizer pressure safety injection setpoint. The small break LOCA analysis of the 3 and 4 inch break used 1600 psia as the reactor trip setpoint. However, in the small break LOCA analysis of the 2 inch break, a reactor trip setpoint of 1700 psia was required to achieve acceptable results.

A revised trip setpoint for low reactor coolant flow has been incorporated in the analysis (see Table 5.1.1-4). This value was used in the analysis to conservatively account for the anticipated flow measurement uncertainty. Based on the final uncertainty allowance reported in Reference 43, no change to the Technical Specification values associated with this reactor trip are required as a result of the VANTAGE 5H transition analysis.