Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUL 31 2003

Docket No. 50-336
B18951

Re: 10 CFR 50.59

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Power Station, Unit No. 2
Changes to Technical Specifications Bases

In accordance with the requirements of Technical Specification 6.23.d of Millstone Unit No. 2, Dominion Nuclear Connecticut, Inc. (DNC) is providing the Nuclear Regulatory Commission Staff with changes to Millstone Unit No. 2 Technical Specifications Bases Sections 2.2.1, 3/4.6.3 and 3/4.7.1.2. These changes are provided for information only. The changes to the Bases Sections were made in accordance with the provisions of 10 CFR 50.59. These changes have been reviewed and approved by the Site Operations Review Committee.

Attachment 1 provides the retyped pages of the Technical Specifications Bases for Millstone Unit No. 2.

There are no regulatory commitments contained within this letter.

A001

If you should have any questions regarding this submittal, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Attachment

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Unit No. 2
Millstone Senior Resident Inspector

Docket No. 50-336
B18951

Attachment 1

Millstone Power Station, Unit No. 2

Changes to Technical Specifications Bases

Retyped Pages

Millstone Unit No. 2 Bases Pages

Section No.	Page No.
2.2.1	B 2-6, 2-7
3/4.6.3	B 3/4 6-3b
3/4.7.1.2	B 3/4 7-2, 7-2a, 7-2b

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.

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. The trip is automatically bypassed below 15 percent power as sensed by the power range nuclear instrument Level 1 bistable.

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 below the 95/95 limit for the DNB correlation.

February 20, 2003
LBDCR 2-21-02

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with 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 trip setpoints are derived from the core safety limits. A safety margin is provided which includes allowances for equipment response times, core power, RCS temperature, and pressurizer pressure measurement uncertainties, processing errors, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER as sensed by the power range nuclear instrument Level 1 bistable. This trip provides turbine protection, reduces the severity of the ensuring transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

(3) assuring that environmental conditions will not preclude access to close the valve and that this action will prevent the release of radioactivity outside the containment.

The appropriate administrative controls, based on the above considerations, to allow locked or sealed closed containment isolation valves to be opened are contained in the procedures that will be used to operate the valves. Entries should be placed in the Shift Manager Log when these valves are opened and closed. However, it is not necessary to log into any Technical Specification Action Statement for these valves, provided the appropriate administrative controls have been established.

If a locked or sealed closed containment isolation valve is opened while operating in accordance with Abnormal or Emergency Operating Procedures (AOPs and EOPs), it is not necessary to establish a dedicated operator. The AOPs and EOPs provide sufficient procedural control over the operation of the containment isolation valves.

Opening a locked or sealed closed containment isolation valve bypasses a plant design feature that prevents the release of radioactivity outside the containment. Therefore, this should not be done frequently, and the time the valve is opened should be minimized. As a general guideline, a locked or sealed closed containment isolation valve should not be opened longer than the time allowed to restore the valve to OPERABLE status, as stated in the action statement for LCO 3.6.3.1 "Containment Isolation Valves."

A discussion of the appropriate administrative controls for the containment isolation valves, that are expected to be opened during operation in MODES 1 through 4, is presented below.

Manual containment isolation valve 2-SI-463, safety injection tank (SIT) recirculation header stop valve, is opened to fill or drain the SITs and for Shutdown Cooling System (SDC) boron equalization. While 2-SI-463 is open, a dedicated operator, in continuous communication with the control room, is required.

When SDC is initiated, SDC suction isolation remotely operated valves 2-SI-652 and 2-SI-651 (inside containment isolation valve) and manual valve 2-SI-709 (outside containment isolation valve) are opened. 2-SI-651 is normally operated from the control room. While in Modes 1, 2 or 3, 2-SI-651 is closed with manual disconnect switch NSI651 locked open to satisfy Appendix R requirements.It does not receive an automatic containment isolation closure signal, but is interlocked to prevent opening if Reactor Coolant System (RCS) pressure is greater than approximately 275 psia. When 2-SI-651 is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

When valve 2-SI-709 is opened locally, a separate dedicated operator is not required to remain at the valve. 2-SI-709 is opened before 2-SI-651. Therefore, opening 2-SI-709 will not establish a connection between the RCS and the SDC System. Opening 2-SI-651 will connect the RCS and SDC System. If a problem then develops, 2-SI-651 can be closed from the control room.

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.

The FSAR Chapter 14 Loss of Normal Feedwater (LONF) analysis evaluates the event occurring with and without offsite power available, and a single active failure. This analysis has determined that one motor driven AFW pump is not sufficient to meet the acceptance criteria. Therefore, two AFW pumps (two motor-driven AFW pumps, or one motor-driven AFW pump and the steam-driven AFW pump) are required to meet the acceptance criteria for this moderate frequency event. To meet the requirement of two AFW pumps available for mitigation, all three pumps must be OPERABLE to accommodate the failure of one pump. This is consistent with the limiting condition for operation and action statements of Technical Specification 3.7.1.2.

Although not part of the bases of Technical Specification 3.7.1.2, the less conservative FSAR Chapter 10 Best Estimate Analysis of the LONF event was performed to demonstrate that one motor-driven AFW pump is adequate to remove decay heat, prevent steam generator dryout, maintain Reactor Coolant System (RCS) subcooling, and prevent pressurizer level from exceeding acceptable limits. From this best estimate analysis of the LONF event, an evaluation was performed to demonstrate that a single motor-driven AFW pump has sufficient capacity to reduce the RCS temperature to 300°F (in addition to decay heat removal) where the Shutdown Cooling System may be placed into operation for continued cooldown. As a result of these evaluations, one motor-driven AFW pump (or the steam-driven AFW pump which has twice the capacity of a motor-driven AFW pump) can meet the requirements to remove decay heat, prevent steam generator dryout, maintain RCS subcooling, prevent the pressurizer from exceeding acceptable limits, and reduce RCS temperature to 300°F.

The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the auxiliary feedwater pumps differential pressure tests on recirculation, Surveillance Requirements 4.7.1.2.a.2.a and 4.7.1.2.a.2.b, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the motor driven auxiliary feedwater pumps was developed assuming a 5% degraded pump from the actual pump curves. The surveillance requirement acceptance criteria for the turbine driven auxiliary feedwater pump was developed from high flow test data extrapolated to minimum recirculation flow, and can be adjusted to account for the affect on pump performance of variations in pump speed. Flow and pressure measurement instrument inaccuracies have not been accounted for in the design basis hydraulic analysis for the motor driven auxiliary feedwater pumps. Flow, pressure, and speed measurement instrument inaccuracies have not been accounted for in the design basis hydraulic analysis for the turbine driven auxiliary feedwater pump. Corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies will be applied to test data taken when verifying pump

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS (Continued)

performance in the flow ranges credited in the accident analyses. No corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies will be applied to minimum recirculation flow type test data since this portion of the curve is not credited in the accident analyses. Corrections for flow, pressure, and speed (turbine driven pump only) measurement instrument inaccuracies are not reflected in the Technical Specification acceptance criteria.

The Auxiliary Feed Water (AFW) system is OPERABLE when the AFW pumps and flow paths required to provide AFW to the steam generators are OPERABLE. Technical Specification 3.7.1.2 requires three AFW pumps to be OPERABLE and provides ACTIONS to address inoperable AFW pumps. The AFW flow path requirements are separated into AFW pump suction flow path requirements, AFW pump discharge flow path to the common discharge header requirements, and common discharge header to the steam generators flow path requirements.

There are two AFW pump suction flow paths from the Condensate Storage Tank to the AFW pumps. One flow path to the turbine driven AFW pump, and one flow path to both motor driven AFW pumps. There are three AFW pump discharge flow paths to the common discharge header, one flow path from each of the three AFW pumps. There are two AFW discharge flow paths from the common discharge header to the steam generators, one flow path to each steam generator. With 2-FW-44 open (normal position), the discharge from any AFW pump will be supplied to both steam generators through the associated AFW regulating valves.

2-FW-44 should remain open when the AFW system is required to be OPERABLE (MODES 1, 2, and 3). Closing 2-FW-44 places the plant in a configuration not considered as an initial condition in the Chapter 14 accident analyses. Therefore, if 2-FW-44 is closed while the plant is operating in MODES 1, 2, or 3, two AFW pumps should be considered inoperable and the appropriate action requirement of Technical Specification 3.7.1.2 entered to limit plant operation in this configuration.

A flow path may be considered inoperable as the result of closing a manual valve, failure of an automatic valve to respond correctly to an actuation signal, or failure of the piping. In the case of an inoperable automatic AFW regulating valve (2-FW-43A or B), flow path OPERABILITY can be restored by use of a dedicated operator stationed at the associated bypass valve (2-FW-56A or B) as directed by OP 2322. Failure of the common discharge header piping will cause both discharge flow paths to the steam generators to be inoperable.

An inoperable suction flow path to the turbine driven AFW pump will result in one inoperable AFW pump. An inoperable suction flow path to the motor driven AFW pumps will result in two inoperable AFW pumps. The ACTION requirements of Technical Specification 3.7.1.2 are applicable based on the number of inoperable AFW pumps.

An inoperable pump discharge flow path from an AFW pump to the common discharge header will cause the associated AFW pump to be inoperable. The ACTION requirements of Technical Specification 3.7.1.2 for one AFW pump are applicable for each affected pump discharge flow path.

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS (Continued)

AFW must be capable of being delivered to both steam generators for design basis accident mitigation. Certain design basis events, such as a main steam line break or steam generator tube rupture, require that the affected steam generator be isolated, and the RCS decay heat removal safety function be satisfied by feeding and steaming the unaffected steam generator. If a failure in an AFW discharge flow path from the common discharge header to a steam generator prevents delivery of AFW to a steam generator, then the design basis events may not be effectively mitigated. In this situation, the ACTION requirements of Technical Specification 3.0.3 are applicable and an immediate plant shutdown is appropriate.

Two inoperable AFW System discharge flow paths from the common discharge header to both steam generators will result in a complete loss of the ability to supply AFW flow to the steam generators. In this situation, all three AFW pumps are inoperable and the ACTION requirements of Technical Specification 3.7.1.2 are applicable. Immediate corrective action is required. However, a plant shutdown is not appropriate until a discharge flow path from the common discharge header to one steam generator is restored.

During quarterly surveillance testing of the turbine driven AFW pump, valve 2-CN-27A is closed and valve 2-CN-28 is opened to prevent overheating the water being circulated. In this configuration, the suction of the turbine driven AFW pump is aligned to the Condensate Storage Tank via the motor driven AFW pump suction flow path, and the pump minimum flow is directed to the Condensate Storage Tank by the turbine driven AFW pump suction path upstream of 2-CN-27A in the reverse direction. During this surveillance, the suction path to the motor driven AFW pump suction path remains OPERABLE, and the turbine driven AFW suction path is inoperable. In this situation, the ACTION requirements of Technical Specification 3.7.1.2 for one AFW pump are applicable.

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. The contained water volume limit includes an allowance for water not usable due to discharge nozzle pipe elevation above tank bottom, plus an allowance for vortex formation.

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