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

Dominion

JUL 31 2003

Docket No. 50-423
B18952

Re: 10 CFR 50.59

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

Millstone Power Station, Unit No. 3
Changes to Technical Specifications Bases

In accordance with the requirements of Technical Specification 6.18.d of Millstone Unit No. 3, Dominion Nuclear Connecticut, Inc. (DNC) is providing the Nuclear Regulatory Commission Staff with changes to Millstone Unit No. 3 Technical Specifications Bases Sections 3/4.1.3, 3/4.3.3.6, 3/4.5.2, 3/4.5.3 and 3/4.7.1.1. 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. 3.

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
V. Nerses, NRC Senior Project Manager, Unit No. 3
Millstone Senior Resident Inspector

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Attachment 1

Millstone Power Station, Unit No. 3

Changes to Technical Specifications Bases

Retyped Pages

Millstone Unit No. 3 Bases Pages

Section No.	Page No.
3/4.1.3	B 3/4 1-4, 1-5, 1-6
3/4.3.3.6	B 3/4 3-5a
3/4.5.2 and 3/4.5.3	B 3/4 5-2, 5-2a, 5-2b, 5-2c
3/4.7.1.1	B 3/4 7-1, 7-1a

MOVABLE CONTROL ASSEMBLIES (Continued)

rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and fully withdrawn position for the Control Banks and 18, 210, and fully withdrawn position for the Shutdown Banks provides assururances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

The required rod drop time of $\leq$ 2.7 seconds specified in Technical Specification 3.1.3.4 is used in the FSAR accident analysis. A rod drop time was calculated to validate the Technical Specification limit. This calculation accounted for all uncertainties, including a plant specific seismic allowance of 0.51 seconds. Since the seismic allowance should be removed when verifying the actual rod drop time, the acceptance criteria for surveillance testing is 2.19 seconds (References 4 and 5).

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The Digital Rod Position Indication (DRPI) System is defined as follows:

- Rod position indication as displayed on DRPI display panel (MB4), or
- Rod position indication as displayed by the Plant Process Computer System

With the above definition, LCO, 3.1.3.2, "ACTION a." is <u>not</u> applicable with either DRPI display panel or the plant process computer points OPERABLE.

The plant process computer may be utilized to satisfy DRPI System requirements which meets LCO 3.1.3.2, in requiring diversity for determining digital rod position indication.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable,

MOVABLE CONTROL ASSEMBLIES (Continued)

then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.

Technical Specification SR 4.1.3.2.1 determines each digital rod position indicator to be OPERABLE by verifying the Demand Position Indication System and the DRPI System agree within 12 steps at least once each 12 hours, except during the time when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the DRPI System at least once each 4 hours.

The Rod Deviation Monitor is generated only from the DRPI panel at MB4. Therefore, when rod position indication as displayed by the plant process computer is the only available indication, then perform SURVEILLANCE REQUIREMENTS every 4 hours.
Additional surveillance is required to ensure the plant process computer indications are in agreement with those displayed on the DRPI. This additional SURVEILLANCE REQUIREMENT is as follows:

> Each rod position indication as displayed by the plant process computer shall be determined to be OPERABLE by verifying the rod position indication as displayed on the DRPI display panel agrees with the rod position indication as displayed by the plant process computer at least once per 12 hours.

The rod position indication, as displayed by DRPI display panel (MB4), is a non-QA system, calibrated on a refueling interval, and used to implement T/S 3.1.3.2. Because the plant process computer receives field data from the same source as the DRPI System (MB4), and is also calibrated on a refueling interval, it fully meets all requirements specified in T/S 3.1.3.2 for rod position. Additionally, the plant process computer provides the same type and level of accuracy as the DRPI System (MB4). The plant process computer does not provide any alarm or rod position deviation monitoring as does DRPI display panel (MB4).

For Specification 3.1.3.1 ACTIONS b. and c., it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). Trippability is defined in Attachment C to a letter dated December 21, 1984, from E. P. Rahe (Westinghouse) to C. O. Thomas (NRC). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus falls under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately 4 hours for this verification.

For LCO 3.1.3.6 the control rods shall be limited in insertion as defined in the Core Operating Limits Report (COLR). The BASES for the Rod Insertion Limit (RIL) is located in the COLR (Reference 3.) and the current cycle reload 50.59 evaluation.

MOVABLE CONTROL ASSEMBLIES (Continued)

The applicable I&C calibration procedure (Reference 1.) being current indicates the associated circuitry is OPERABLE.

There are conditions when the Lo-Lo and Lo alarms of the RIL Monitor are limited below the RIL, as indicated in COLR Section 2.3, Control Rod Insertion Limits. The RIL Monitor remains OPERABLE because the lead control rod bank still has the Lo and Lo-Lo alarms greater than or equal to the RIL.

When rods are at the top of the core, the Lo-Lo alarm is limited below the RIL to prevent spurious alarms. The RIL is equal to the Lo-Lo alarm until the adjustable upper limit setpoint on the RIL Monitor is reached, then the alarm remains at the adjustable upper limit setpoint. When the RIL is in the region above the adjustable upper limit setpoint, the Lo-Lo alarm is below the RIL.

References:

1. IC 3469N08, Rod Control Speed, Insertion Limit, and Control TAVE Auctioneered/Deviation Alarms.

2. Letter NS-OPLS-OPL-1-91-226, (Westinghouse Letter NEU-91-563), dated April 24, 1991.

3. COLR Section 2.3

4. Westinghouse Letter NEU-97-298, "Millstone Unit 3 - RCCA Drop Time," dated November 13, 1997.

5. Westinghouse Letter 98NEU-G-0060, "Millstone Unit 3 - Robust Fuel Assembly (Design Report) and Generic SECL," dated October 2, 1998.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

A channel is operable if four or more sensors, half or more in the upper head region and half or more in the upper plenum region, are OPERABLE.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, at least one channel shall be restored to OPERABLE status in the nearest refueling outage.

The Reactor Coolant System Subcooling Margin Monitor, Core Exit Thermocouples, and Reactor Vessel Water Level instruments are processed by two separate trains of ICC (Inadequate Core Cooling) and HJTC (Heated Junction ThermoCouple) processors. The preferred indication for these parameters is the Safety Parameter Display System (SPDS) via the non-qualified PPC (Plant Process Computer) but qualified indication is provided in the instrument rack room. When the PPC data links cease to transmit data, the processors must be reset in order to restore the flow of data to the PPC. During reset, the qualified indication in the instrument rack room is lost. These instruments are OPERABLE during this reset since the indication is only briefly interrupted while the processors reset and the indication is promptly restored. The sensors are not removed from service during this reset. The train should be considered inoperable only if the qualified indication fails to be restored following reset. Except for the non-qualified PPC display, the instruments operate as required.

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. Containment hydrogen concentration is also important in verifying the adequacy of mitigating actions. The requirement to perform a hydrogen sensor calibration at least once every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two point calibration, utilizing gas containing approximately one percent hydrogen gas for one of the calibration points, and gas containing approximately four percent hydrogen gas for the other calibration point.

3/4.3.3.7 Deleted.

ECCS SUBSYSTEMS (Continued)

The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System is required to be available to support charging pump operation. The Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation System consists of two redundant trains, each capable of providing 100% of the required flow. Each train has a two position, "Off" and "Auto," remote control switch. With the remote control switches for each train in the "Auto" position, the system is capable of automatically transferring operation to the redundant train in the event of a low flow condition in the operating train. The associated fans do not receive any safety related automatic start signals (e.g., Safety Injection Signal).

Placing the remote control switch for a Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train in the "Off" position to start the redundant train or to perform post maintenance testing to verify availability of the redundant train will not affect the availability of that train, provided appropriate administrative controls have been established to ensure the remote control switch is immediately returned to the "Auto" position after the completion of the specified activities or in response to plant conditions. These administrative controls include the use of an approved procedure and a designated individual at the control switch for the respective Charging Pump/Reactor Plant Component Cooling Water Pump Ventilation Train who can rapidly respond to instructions from procedures, or control room personnel, based on plant conditions.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Surveillance Requirement 4.5.2.b.1 requires verifying that the ECCS piping is full of water. The ECCS pumps are normally in a standby, nonoperating mode, with the exception of the operating centrifugal charging pump(s). As such, the ECCS flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly when required to inject into the RCS. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gases (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling.

This Surveillance Requirement is met by:

- venting ECCS pump casings and the accessible discharge piping high points including the ECCS pump suction crossover piping (i.e., downstream of valves 3RSS*MV8837A/B and 3RSS*MV8838A/B to safety injection and charging pump suction).

ECCS SUBSYSTEMS (Continued)

- venting of the nonoperating centrifugal charging pumps at the suction line test connection. The nonoperating centrifugal charging pumps do not have casing vent connections and venting the suction pipe will assure that the pump casing does not contain voids and pockets of entrained gases.

- using an external water level detection method for the water filled portions of the RSS piping upstream of valves 3RSS*MV8837A/B and 3RSS*MV8838A/B. When deemed necessary by an external water level detection method, filling and venting to reestablish the acceptable water levels may be performed after entering LCO action statement 3.6.2.2 since venting without isolation of the affected train would result in a breach of the containment pressure boundary.

The following ECCS subsections are exempt from this Surveillance:

- the operating centrifugal charging pump(s) and associated piping - as an operating pump is self venting and cannot develop voids and pockets of entrained gases.

- the RSS pumps, since this equipment is left partially dewatered during plant operation.

- the RSS heat exchangers, since this equipment is laid-up dry during plant operation.

- the RSS piping that is not maintained filled with water during plant operation.

Surveillance Requirement 4.5.2.C.2 requires that the visual inspection of the containment be performed at least once daily if the containment has been entered that day and when the final containment entry is made. This will reduce the number of unnecessary inspections and also reduce personnel exposure.

The Emergency Core Cooling System (ECCS) has several piping cross connection points for use during the post-LOCA recirculation phase of operation. These cross-connection points allow the Recirculation Spray System (RSS) to supply water from the containment sump to the safety injection and charging pumps. The RSS has the capability to supply both Train A and B safety injection pumps and both Train A and B charging pumps. Operator action is required to position valves to establish flow from the containment sump through the RSS subsystems to the safety injection and charging pumps since the valves are not automatically repositioned. The quarterly stroke testing (Technical Specification 4.0.5) of the ECC/RSS recirculation flowpath valves discussed below will not result in subsystem inoperability (except due to other equipment manipulations to support valve testing) since these valves are manually aligned in accordance with the Emergency Operating Procedures (EOPs) to establish the recirculation flowpaths. It is expected the valves will be returned to the normal pre-test position following termination of the surveillance testing in response to the accident. Failure to restore any valve to the normal pre-test position will be indicated to the Control Room

ECCS SUBSYSTEMS (Continued)

Operators when the ESF status panels are checked, as directed by the EOPs. The EOPs direct the Control Room Operators to check the ESF status panels early in the event to ensure proper equipment alignment. Sufficient time before the recirculation flowpath is required is expected to be available for operator action to position any valves that have not been restored to the pre-test position, including local manual valve operation. Even if the valves are not restored to the pre-test position, sufficient capability will remain to meet ECCS post-LOCA recirculation requirements. As a result, stroke testing of the ECCS recirculation valves discussed below will not result in a loss of system independence or redundancy, and both ECCS subsystems will remain OPERABLE.

When performing the quarterly stroke test of 3SIH*MV8923A, the control switch for safety injection pump 3SIH*P1A is placed in the pull-to-lock position to prevent an automatic pump start with the suction valve closed. With the control switch for 3SIH*P1A in pull-to-lock, the Train A ECCS subsystem is inoperable and Technical Specification 3.5.2, Action a., applies. This action statement is sufficient to administratively control the plant configuration with the automatic start of 3SIH*P1A defeated to allow stroke testing of 3SIH*MV8923A. In addition, the EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, and an additional single failure does not occur (an acceptable assumption since the Technical Specification action statement limits the plant configuration time such that no additional equipment failure need be postulated). During the injection phase the redundant subsystem (Train B) is fully functional, as is a significant portion of the Train A subsystem. During the recirculation phase, the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps, and the Train B RSS subsystem can supply water from the containment sump to the B safety injection pump.

When performing the quarterly stroke test of 3SIH*MV8923B, the control switch for safety injection pump 3SIH*P1B is placed in the pull-to-lock position to prevent an automatic pump start with the suction valve closed. With the control switch for 3SIH*P1B in pull-to-lock, the Train B ECCS subsystem is inoperable and Technical Specification 3.5.2, Action a., applies. This action statement is sufficient to administratively control the plant configuration with the automatic start of 3SIH*P1B defeated to allow stroke testing of 3SIH*MV8923B. In addition, the EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, and an additional single failure does not occur (an acceptable assumption since the Technical Specification action statement limits the plant configuration time such that no additional equipment failure need be postulated). During the injection

ECCS SUBSYSTEMS (Continued)

phase the redundant subsystem (Train A) is fully functional, as is a significant portion of the Train B subsystem. During the recirculation phase, the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps and the Train A safety injection pump. The Train B RSS subsystem cannot supply water from the containment sump to any of the remaining pumps.

When performing the quarterly stroke test of 3SIH*MV8807A or 3SIH*MV8807B, 3SIH*MV8924 is closed first to prevent the potential injection of RWST water into the RCS through the operating charging pump. When 3SIH*MV8924 is closed, it is not necessary to declare either ECCS subsystem inoperable. Although expected to be open for post-LOCA recirculation, sufficient time is expected to be available post-LOCA to identify and open 3SIH*MV8924 either from the Control Room or locally at valve. The EOPs and the ESF status panels will identify this abnormal plant configuration, if not corrected following the termination of the surveillance testing, to the plant operators to allow restoration of the normal post-LOCA recirculation flowpath. Even if system restoration is not accomplished, sufficient equipment will be available to perform all ECCS and RSS injection and recirculation functions, provided no additional ECCS or RSS equipment is inoperable, even if a single failure is postulated. The failure to open 3SIH*MV8924 due to mechanical binding or the loss of power to ECCS Train A could be the single failure. If a different single failure is postulated, restoration of 3SIH*MV8924 can be accomplished. The closure of 3SIH*MV8924 has no affect on the injection phase. During the recirculation phase, assuming 3SIH*MV8924 remains closed (i.e., the single failure), the Train A RSS subsystem can supply water from the containment sump to the Train A and B charging pumps, and the Train B RSS subsystem can supply water from the containment sump to the Train A and B safety injection pumps. If power is lost to ECCS Train A and 3SIH*MV8924 is not opened locally (i.e., the single failure), cold leg recirculation can be accomplished by using RSS Train B to supply containment sump water via 3SIH*P1B to the RCS cold legs and 3SIL*MV8809B can be opened to supply containment sump water via RSS Train B to the RCS cold legs. Hot leg recirculation can be accomplished by using RSS Train B to supply containment sump water via 3SIH*P1B to the RCS hot legs and maintaining 3SIL*MV8809B open to supply containment sump water via RSS Train B to the RCS cold legs.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The design minimum total relieving capacity for all valves on all of the steam lines is 1.579×10^7 lbs/h which is 105% of the total secondary steam flow of 1.504×10^7 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

The OPERABILITY of the main steam Code safety valves is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The lift settings for the main steam Code safety valves are listed in Table 3.7-3. This table allows a ± 3% setpoint tolerance (allowable value) on the lift setting for OPERABILITY to account for drift over an operating cycle.

Each main steam Code safety valve is demonstrated OPERABLE with lift settings as shown in Table 3.7-3, in accordance with Technical Specification 4.0.5. During this testing, the main steam Code safety valves are OPERABLE provided the actual lift settings are within ± 3% of the required lift setting. A footnote to Table 3.7-3 requires that the lift setting be restored to within ± 1% of the required lift setting following testing to allow drift during the next operating cycle. However, if the testing is done at the end of the operating cycle when the plant is being shut down for refueling, restoration to ± 1% of the specified lift setting is not required for valves that will not be used (e.g., replaced) for the next operating cycle. While the lift settings are being restored to within ± 1% of the required lift setting, the main steam Code safety valves remain OPERABLE provided the actual lift setting is within ± 3% of the required lift setting.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$Hi\ \phi = (100/Q)\,\frac{(w_s h_{fg} N)}{K}$$

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES (Continued)

where:

Hiϕ = Safety Analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$

h_{fg} = heat of vaporization for steam at the highest MSSV opening pressure including tolerance (± 3%) and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant