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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
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DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNITS 2 AND 3
CHANGES TO TECHNICAL SPECIFICATIONS BASES

In accordance with the requirements of Millstone Power Station Unit 2 (MPS2), Technical Specification 6.23.d, and Millstone Power Station Unit 3 (MPS3), Technical Specification 6.18.d, Dominion Nuclear Connecticut, Inc. (DNC) is providing the Nuclear Regulatory Commission Staff with changes to MPS2 and MPS3 Technical Specifications Bases Sections. MPS2 changes affect Technical Specifications Bases Sections 3/4.3, 3/4.4, 3/4.8, and 3/4.9. MPS3 changes affect Technical Specifications Bases Section 3/4.1, 3/4.4, 3/4.6, 3/4.7, and 3/4.9. 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.

Attachments 1 and 2 provide the revised pages of the Technical Specifications Bases for MPS2 and MPS3, respectively.

If you have any questions or require additional information, please contact Mr. Paul R. Willoughby at (804) 273-3572.

Very truly yours,

Eugene S. Grecheck
Vice President – Nuclear Support Services

Attachments:

1. Revised Bases Pages for Millstone Power Station Unit 2
2. Revised Bases Pages for Millstone Power Station Unit 3

Commitments made in this letter: None.

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ATTACHMENT 1

CHANGES TO TECHNICAL SPECIFICATIONS BASES

REVISED PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

Millstone Power Station Unit 2 Bases Pages

Page Changes

Section No.	Page No.
3/4.3 Instrumentation	B 3/4 3-1a B 3/4 3-1b
3/4.4 Reactor Coolant System	B 3/4 4-1b B 3/4 4-1c B 3/4 4-1d B 3/4 4-1e B 3/4 4-2a B 3/4 4-7 B 3/4 4-7a
3/4.8 Electrical Power Systems	B 3/4 8-1o B 3/4 8-1p
3/4.9 Refueling Operations	B 3/4 9-2 B 3/4 9-2a B 3/4 9-2b B 3/4 9-2c B 3/4 9-3b

Page Removals

The following pages should be removed from the MPS2 Technical Specification Bases.

3/4.9 Refueling Operations	B 3/4 9-3a

3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (continued)

declared inoperable, and ACTION Statement 2 of Technical Specification 3.3.1.1 entered. When testing the RPS logic (matrix testing), the individual RPS channels will not be affected. Each of the parameters within each RPS channel supplies three contacts to make up the 6 different logic ladders/ matrices (AB, AC, AD, BC, BD, and CD). During matrix testing, only one logic matrix is tested at a time. Since each RPS channel supplies 3 different logic ladders, testing one ladder matrix at a time will not remove an RPS channel from the overall logic matrix. Therefore, matrix testing will not remove an RPS channel from service or make the RPS channel inoperable. It is not necessary to enter an ACTION statement for any of the parameters associated with each RPS channel while performing matrix testing. This also applies when testing the reactor trip circuit breakers since this test will not remove an RPS channel from service or make the RPS channel inoperable.

ACTION statements for the RPS logic matrices and RPS logic matrix relays are required to be entered during matrix testing as these functional units become inoperable when the "HOLD" button is depressed during testing.

The ESFAS includes four sensor subsystems and two actuation subsystems for each of the functional units identified in Table 3.3-3. Each sensor subsystem includes measurement channels and bistable trip units. Each of the four sensor subsystem channels monitors redundant and independent process measurement channels. Each sensor is monitored by at least one bistable. The bistable associated with each ESFAS Function will trip when the monitored variable exceeds the trip setpoint. When tripped, the sensor subsystems provide outputs to the two actuation subsystems.

The two independent actuation subsystems each compare the four associated sensor subsystem outputs. If a trip occurs in two or more sensor subsystem channels, the two-out-of-four automatic actuation logic will initiate one train of ESFAS. An Automatic Test Inserter (ATI), for which the automatic actuation logic OPERABILITY requirements of this specification do not apply, provides automatic test capability for both the sensor subsystems and the actuation subsystems.

The provisions of Specification 4.0.4 are not applicable for the CHANNEL FUNCTIONAL TEST of the Engineered Safety Feature Actuation System automatic actuation logic associated with Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, Steam Generator Pressure Main Steam Line Isolation, and Pressurizer Pressure Enclosure Building Filtration for entry into MODE 3 or other specified conditions. After entering MODE 3, pressurizer pressure and steam generator pressure will be increased and the blocks of the ESF actuations on low pressurizer pressure and low steam generator pressure will be

3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (continued)

automatically removed. After the blocks have been removed, the CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic can be performed. The CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic must be performed within 12 hours after establishing the appropriate plant conditions, and prior to entry into MODE 2.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The Reactor Protective and Engineered Safety Feature response times are contained in the Millstone Unit No. 2 Technical Requirements Manual. Changes to the Technical Requirements Manual require a 10CFR50.59 review as well as a review by the Site Operations Review Committee.

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION (continued)

In MODE 5, two OPERABLE SDC trains require 2 SDC pumps, 2 SDC heat exchangers, 2 RBCCW pumps, 2 RBCCW heat exchangers, and 2 SW pumps. In addition, 2 RBCCW headers are required to provide cooling to the SDC heat exchangers, but only 1 SW header is required to support the SDC trains. The equipment specified is sufficient to address a single active failure of the SDC System and associated support systems.

In addition, two SDC trains can be considered OPERABLE, with only one 125-volt D.C. bus train OPERABLE, in accordance with Limiting Condition for Operation (LCO) 3.8.2.4. 2-SI-306 and 2-SI-657 are both powered from the same 125-volt D.C. bus, on Facility 1. Should these valves reposition due to a loss of power, SDC would no longer be aligned to cool the RCS. However, a designated operator is assigned to reposition these valves as necessary in the event 125-volt D.C. power is lost. Consistent with the bases for LCO 3.8.2.4, the 125-volt D.C. support system operability requirements for both trains of SDC are satisfied in MODE 5 with at least one 125-volt D.C. bus train OPERABLE and the 125-volt D.C. buses cross-tied.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in MODE 4 with one or more RCS cold legs ≤ 275°F and in MODE 5 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by:

1. Restricting pressurizer water volume to ensure sufficient steam volume is available to accommodate the insurge;

2. Restricting pressurizer pressure to establish an initial pressure that will ensure system pressure does not exceed the limit; and

3. Restricting primary to secondary system delta-T to reduce the energy addition from the secondary system.

If these restrictions are met, the steam bubble in the pressurizer is sufficient to ensure the Appendix G limits will not be exceeded. No credit has been taken for PORV actuation to limit RCS pressure in the analysis of the energy addition transient.

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION (continued)

The limitations on pressurizer water level, pressurizer pressure, and primary to secondary delta-T are necessary to ensure the validity of the analysis of the energy addition due to starting an RCP. The values for pressurizer water level and pressure can be obtained from control room indications. The primary to secondary system delta-T can be obtained from Shutdown Cooling (SDC) System outlet temperature and the saturation temperature for indicated steam generator pressure. If there is no indicated steam generator pressure, the steam generator shell temperature indicators can be used. If these indications are not available, other appropriate instrumentation can be used.

The RCP starting criteria values for pressurizer water level, pressurizer pressure, and primary to secondary delta-T contained in Technical Specifications 3.4.1.3, 3.4.1.4 and 3.4.1.5 have not been adjusted for instrument uncertainty. The values for these parameters contained in the procedures that will be used to start an RCP have been adjusted to compensate for instrument uncertainty.

The value of RCS cold leg temperature (≤ 275°F) used to determine if the RCP start criteria applies, will be obtained from SDC return temperature if SDC is in service. If SDC is not in service, or natural circulation is occurring, RCS cold leg temperature will be used.

Average Coolant Temperature (T_{avg}) values are derived under the following 3 plant conditions, using the designated formula as appropriate for use in Unit 2 operating procedures.

- RCP Operation: $(T_{cold1} + T_{cold2} + T_{hot1} + T_{hot2}) / 4 = T_{avg}$
- Natural circulation only flow: $(T_{cold1} + T_{cold2} + T_{hot1} + T_{hot2}) / 4 = T_{avg}$
- SDC flow greater than 1000 gpm: $(SDC_{outlet} + SDC_{inlet}) / 2 = T_{avg}$
  (exception: T_{avg} is not expected to be calculated by this definition during the initial portion of the initiation phase of SDC. The transition point from loop temperature average to SDC system average during cooldowns is when T351Y decreases below Loop T_{cold})

During operation with one or more Reactor Coolant Pumps (RCPs) providing forced flow and during natural circulation conditions, the loop Resistance Temperature Detectors (RTDs) represent the inlet and outlet temperatures of the reactor and hence the average temperature of the water that the reactor is exposed to. This holds during concurrent RCP/SDC operation also.

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION (continued)

During Shutdown Cooling (SDC) only operation, there is no significant flow past the loop RTDs. Core inlet and outlet temperatures are accurately measured during those conditions by using T351Y, SDC return to RCS temperature indication, and T351X, RCS to SDC temperature indication. The average of these two indicators provides a temperature that is equivalent to the average RCS temperature in the core.

During the transition from Steam Generator (SG) and SDC heat removal to SDC only heat removal, actual core average temperature results from a mixture of both SDC flow and loop flow from natural circulation. This condition occurs from the time SDC cooling is initiated until SG steaming process stops removing heat. The temperature of this mixture cannot be measured or calculated. However, the average of the SDC temperatures is still appropriate for use. This provides a straightforward process for determining T_{avg}.

During some transient conditions, such as heatups on SDC, the value calculated by this average definition will be slightly higher than the actual core average. During other transients, such as cooldowns where SG heat removal is still taking place causing some natural circulation flow, the value calculated by the average definition will be slightly lower than actual core average conditions. For the purpose of determining MODE changes and technical specification applicability, these transient condition results are conservative.

The Notes in LCOs 3.4.1.2, 3.4.1.3, 3.4.1.4, and 3.4.1.5 permit a limited period of operation without RCPs and shutdown cooling pumps. All RCPs and shutdown cooling pumps may be removed from operation for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1.1 is maintained is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Concerning TS 3.4.1.2, ACTION b.; 3.4.1.3, ACTION c.; 3.4.1.4, ACTION b.; and 3.4.1.5, ACTION b., if two required loops or trains are inoperable or a required loop or train is not in operation except during conditions permitted by the note in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1 must be suspended and action to restore one RCS loop or SDC train to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron

3/4.4.1 COOLANT LOOPS AND COOLANT CIRCULATION (continued)

concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate completion times reflect the importance of decay heat removal. The ACTION to restore must continue until one loop or train is restored to operation.

Technical Specification 3.4.1.6 limits the number of reactor coolant pumps that may be operational during MODE 5. This will limit the pressure drop across the core when the pumps are operated during low-temperature conditions. Controlling the pressure drop across the core will maintain maximum RCS pressure within the maximum allowable pressure as calculated in Code Case No. N-514. Limiting two reactor coolant pumps to operate when the RCS cold leg temperature is less than 120° F, will ensure that the requirements of 10 CFR 50 Appendix G are not exceeded. Surveillance 4.4.1.6 supports this requirement.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 296,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. If any pressurizer code safety valve is inoperable, and cannot be restored to OPERABLE status, the ACTION statement requires the plant to be shut down and cooled down such that Technical Specification 3.4.9.3 will become applicable and require the Low Temperature Overpressure Protection System to be placed in service to provide overpressure protection stuck open PORV at a time that the block valve is inoperable. This may be accomplished by various methods. These methods include, but are not limited to, placing the NORMAL/ISOLATE switch at the associated Bottle Up Panel in the "ISOLATE" position or pulling the control power fuses for the associated PORV control circuit.

Although the block valve may be designated inoperable, it may be able to be manually opened and closed and in this manner can be used to perform its function. Block valve inoperability may be due to seat leakage, instrumentation problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. This condition is only intended to permit operation of the plant for a limited period of time. The block valve should normally be available to allow PORV operation for automatic mitigation of overpressure events. The block valves must be returned to OPERABLE status prior to entering MODE 3 after a refueling outage.

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the completion time of 1 hour or isolate the flow path by closing and removing the power to the associated block valve and cooldown the RCS to MODE 4.

3/4.4.4 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the reactor coolant system during operations with both forced reactor coolant flow and with natural circulation flow. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish and maintain natural circulation.

The requirement for two groups of pressurizer heaters, each having a capacity of 130 kW, is met by verifying the capacity of the pressurizer proportional heater groups 1 and 2. Since the pressurizer proportional heater groups 1 and 2 are supplied from the emergency 480V electrical buses, there is reasonable assurance that these heaters can be energized during a loss of offsite power to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is

Included in this evaluation is consideration of flange protection in accordance with 10 CFR 50, Appendix G. The requirement makes the minimum temperature RT_{NDT} plus 90°F for hydrostatic test and RT_{NDT} plus 120°F for normal operation when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure. Since the flange region RT_{NDT} has been calculated to be 30°F, the minimum flange pressurization temperature during normal operation is 150°F (163°F with instrument uncertainty) when the pressure exceeds 20% of the preservice hydrostatic pressure. Operation of the RCS within the limits of the heatup and cooldown curves will ensure compliance with this requirement.

To establish the minimum boltup temperature, ASME Code Section XI, Appendix G, requires the temperature of the flange and adjacent shell and head regions shall be above the limiting RT_{NDT} temperature for the most limiting material of these regions. The RT_{NDT} temperature for that material is 30°F. Adding 13°F, for temperature measurement uncertainty, results in a minimum boltup temperature of 43°F. For additional conservatism, a minimum boltup temperature of 70°F is specified on the heatup and cooldown curves. The head and vessel flange region temperature must be greater than 70°F, whenever any reactor vessel stud is tensioned.

The Low Temperature Overpressure Protection (LTOP) System provides a physical barrier against exceeding the 10CFR50 Appendix G pressure/temperature limits during low temperature RCS operation either with a steam bubble in the pressurizer or during water solid conditions. This system consists of either two PORVs with a pressure setpoint ≤ 415 psia, or an RCS vent of sufficient size. Analysis has confirmed that the design basis mass addition transient discussed below will be mitigated by operation of the PORVs or by establishing an RCS vent of sufficient size.

The LTOP System is required to be OPERABLE when RCS cold leg temperature is at or below 275°F (Technical Specification 3.4.9.3). However, if the RCS is in MODE 6 and the reactor vessel head has been removed, a vent of sufficient size has been established such that RCS pressurization is not possible. Therefore, an LTOP System is not required (Technical Specification 3.4.9.3 is not applicable).

Adjusted Referenced Temperature (ART) is the RT_{NDT} adjusted for radiation effects plus a margin term required by Revision 2 of Regulatory Guide 1.99. The LTOP System is armed at a temperature which exceeds the limiting 1/4t ART plus 50°F as required by ASME Section XI, Appendix G. For the operating period up to 54 EFPY, the limiting 1/4t ART is 175°F which results in a minimum LTOP System enable temperature of at least 271°F when corrected for instrument uncertainty. The current value of 275°F will be retained.

The mass input analysis performed to ensure the LTOP System is capable of protecting the reactor vessel assumes that all pumps capable of injecting into the RCS start, and then one PORV fails to actuate (single active failure). Since the PORVs have limited relief capability, certain administrative restrictions have been implemented to ensure that the mass input transient will not exceed the relief capacity of a PORV. The analysis has determined two PORVs (assuming one PORV fails) are sufficient if the mass addition transient is limited to the inadvertent start of one high pressure safety injection (HPSI) pump and two charging pumps when RCS temperature is at or below 275°F and above 190°F, and the inadvertent start of one charging pump when RCS temperature is at or below 190°F.

The LTOP analysis assumes only one PORV open due to single active failure of the other to open. Analysis has shown that one PORV is sufficient to prevent exceeding the 10CFR Appendix G pressure/temperature limits during low temperature operation. If the RCS is depressurized and vented through at least a 2.2 square inch vent, the peak RCS pressure, resulting from the maximum mass input transient allowed by Technical Specification 3.4.9.3, will not exceed 300 psig (SDC System suction side design pressure).

When the RCS is at or below 190°F, additional pumping capacity can be made capable of injecting into the RCS by establishing an RCS vent of at least 2.2 square inches. Removing the pressurizer manway cover, pressurizer vent port cover or a pressurizer safety relief valve will result in a passive vent of at least 2.2 square inches. Additional methods to establish the required RCS vent are acceptable, provided the proposed vent has been evaluated to ensure the flow characteristics are equivalent to one of these.

Establishing a pressurizer steam bubble of sufficient size will be sufficient to protect the reactor vessel from the energy addition transient associated with the start of an RCP, provided the restrictions contained in Technical Specification 3.4.1.3 are met. These restrictions limit the heat input from the secondary system. They also ensure sufficient steam volume exists in the pressurizer to accommodate the insurge. No credit for PORV actuation was assumed in the LTOP analysis of the energy addition transient.

The restrictions apply only to the start of the first RCP. Once at least one RCP is running, equilibrium is achieved between the primary and secondary temperatures, eliminating any significant energy addition associated with the start of the second RCP.

The LTOP restrictions are based on RCS cold leg temperature. This temperature will be determined by using RCS cold leg temperature indication when RCPs are running, or natural circulation if it is occurring. Otherwise, SDC return temperature indication will be used.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or REFUELING condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status. If the required power sources or distribution systems are not OPERABLE in MODES 5 and 6, operations involving CORE ALTERATIONS, positive reactivity additions, or movement of irradiated fuel assemblies are required to be suspended. Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power source or distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Each 125-volt D.C. bus train consists of its associated 125-volt D.C. bus, a 125-volt D.C. battery bank, and a battery charger with at least 400 ampere charging capacity. To demonstrate OPERABILITY of a 125-volt D.C. bus train, these components must be energized and capable of performing their required safety functions. Additionally, in MODES 1 through 4 at least one tie breaker between the 125-volt D.C. bus trains must be open for a 125-volt D.C. bus train to be considered OPERABLE.

For MODES 5 and 6, each battery is sized to supply the total connected vital loads (one battery connected to both buses) for one hour without charger support. Therefore, in MODES 5 and 6 with at least one 125-volt D.C. bus train OPERABLE and the 125-volt D.C. buses cross-tied, the 125-volt D.C. support system operability requirements for both buses are satisfied.

Footnote (a) to Technical Specification Tables 4.8-1 and 4.8-2 permits the electrolyte level to be above the specified maximum level for the Category A limits during equalizing charge, provided it is not overflowing. Because of the internal gas generation during the performance of an equalizing charge, specific gravity gradients and artificially elevated electrolyte levels are produced which may exist for several days following completion of the equalizing charge. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. In accordance with the recommendations of IEEE 450-1980, electrolyte level readings should be taken only after the battery has been at float charge for at least 72 hours.

Based on vendor recommendations and past operating experience, seven (7) days has been determined a reasonable time frame for the 125-volt D.C. batteries electrolyte level to stabilize and to provide sufficient time to verify battery electrolyte levels are with in the Category A limits.

Footnote (b) to Technical Specification Tables 4.8-1 and 4.8-2 requires that level correction is not required when battery charging current is < 5 amps on float charge. This current provides, in general, an indication of overall battery condition.

Footnote (c) to Technical Specification Tables 4.8-1 and 4.8-2 states that level correction is not required when battery charging current is < 5 amps on float charge. This current provides, in general, an indication of overall battery condition. Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity measurement for determining the state of charge. This footnote allows the float charge current to be used as an alternative to specific gravity to show OPERABILITY of a battery for up to seven (7) days following the completion of a battery equalizing charge. Each connected cells specific gravity must be measured prior to expiration of the 7 day allowance.

Surveillance Requirements 4.8.2.3.2.c.1 and 4.8.2.5.2.c.1 provide for visual inspection of the battery cells, cell plates, and battery racks to detect any indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The non-safety grade 125V D.C. Turbine Battery is required for accident mitigation for a main steam line break within containment with a coincident loss of a vital D.C. bus. The Turbine Battery provides the alternate source of power for Inverters 1 & 2 respectively via non-safety grade Inverters 5 & 6. For the loss of a D.C. event with a coincident steam line break within containment, the feedwater regulating valves are required to close to ensure containment design pressure is not exceeded.

The Turbine Battery D.C. electrical power subsystem consists of 125-volt D.C. bus 201D and 125-volt D.C. battery bank 201D. To demonstrate OPERABILITY of this subsystem, these components must be energized and capable of performing their required safety functions.

3/4.9.6 DELETED

3/4.9.7 DELETED

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

In MODE 6 the shutdown cooling trains are the primary means of heat removal. One SDC train provides sufficient heat removal capability. However, to provide redundant paths for heat removal either two SDC trains are required to be OPERABLE and one SDC train must be in operation, or one SDC train is required to be OPERABLE and in operation with the refueling cavity water level ≥ 23 feet above the reactor vessel flange. This volume of water in the refueling cavity will provide a large heat sink in the event of a failure of the operating SDC train. Any exception to these requirements are contained in the LCO Notes.

An OPERABLE SDC train, for plant operation in MODE 6, includes a pump, heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine RCS temperature. In addition, sufficient portions of the Reactor Building Closed Cooling Water (RBCCW) and Service Water (SW) Systems shall be OPERABLE as required to provide cooling to the SDC heat exchanger. The flow path starts at the RCS hot leg and is returned to the RCS cold legs. An OPERABLE SDC train consists of the following equipment:

1. An OPERABLE SDC pump (low pressure safety injection pump);
2. The associated SDC heat exchanger from the same facility as the SDC pump;
3. An RBCCW pump, powered from the same facility as the SDC pump, and RBCCW heat exchanger capable of cooling the associated SDC heat exchanger;
4. A SW pump, powered from the same facility as the SDC pump, capable of supplying cooling water to the associated RBCCW heat exchanger; and
5. All valves required to support SDC System operation are in the required position or are capable of being placed in the required position.

In MODE 6, two OPERABLE SDC trains require 2 SDC pumps, 2 SDC heat exchangers, 2 RBCCW pumps, 2 RBCCW heat exchangers, and 2 SW pumps. In addition, 2 RBCCW headers are required to provide cooling to the SDC heat exchangers, but only 1 SW header is required to support the SDC trains. The equipment specified is sufficient to address a single active failure of the SDC System and associated support systems.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (Continued)

In addition, two SDC trains can be considered OPERABLE, with only one 125-volt D.C. bus train OPERABLE, in accordance with Limiting Condition for Operation (LCO) 3.8.2.4. 2-SI-306 and 2-SI-657 are both powered from the same 125-volt D.C. bus, on Facility 1. Should these valves reposition due to a loss of power, SDC would no longer be aligned to cool the RCS. However, a designated operator is assigned to reposition these valves as necessary in the event 125-volt D.C. power is lost. Consistent with the bases for LCO 3.8.2.4, the 125-volt D.C. support system operability requirements for both trains of SDC are satisfied in MODE 6 with at least one 125-volt D.C. bus train OPERABLE and the 125-volt D.C. buses cross-tied.

Either SDC pump may be aligned to the refueling water storage tank (RWST) to support filling the fueling cavity or for performance of required testing. A SDC pump may also be used to transfer water from the refueling cavity to the RWST. In addition, either SDC pump may be aligned to draw a suction on the spent fuel pool (SFP) through 2-RW-11 and 2-SI-442 instead of the normal SDC suction flow path, provided the SFP transfer canal gate valve 2-RW-280 is open under administrative control (e.g., caution tagged). When using this alternate SDC flow path, it will be necessary to secure the SFP cooling pumps, and limit SDC flow as specified in the appropriate procedure, to prevent vortexing in the suction piping. The evaluation of this alternate SDC flow path assumed that this flow path will not be used during a refueling outage until after the completion of the fuel shuffle such that approximately one third of the reactor core will contain new fuel. By waiting until the completion of the fuel shuffle, sufficient time (at least 14 days from reactor shutdown) will have elapsed to ensure the limited SDC flow rate specified for this alternate lineup will be adequate for decay heat removal from the reactor core and the spent fuel pool. In addition, CORE ALTERATIONS shall be suspended when using this alternate flow path, and this flow path should only be used for short time periods, approximately 12 hours. If the alternate flow path is expected to be used for greater than 24 hours, or the decay heat load will not be bounded as previously discussed, further evaluation is required to ensure that this alternate flow path is acceptable.

These alternate lineups do not affect the OPERABILITY of the SDC train. In addition, these alternate lineups will satisfy the requirement for a SDC train to be in operation if the minimum required SDC flow through the reactor core is maintained.

In MODE 6, with the refueling cavity filled to ≥ 23 feet above the reactor vessel flange, both SDC trains may not be in operation for up to 1 hour in each 8 hour period, provided no operations are permitted that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (Continued)

uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and RCS to SDC isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling pool.

In MODE 6, with the refueling cavity filled to ≥ 23 feet above the reactor vessel flange, both SDC trains may also not be in operation for local leak rate testing of the SDC cooling suction line (containment penetration number 10) or to permit maintenance on valves located in the common SDC suction line. This will allow the performance of required maintenance and testing that otherwise may require a full core offload. In addition to the requirement prohibiting operations that would dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1, CORE ALTERATIONS are suspended and all containment penetrations providing direct access from the containment atmosphere to outside atmosphere must be closed. The containment purge valves are containment penetrations and must satisfy all requirements specified for a containment penetration. No time limit is specified to operate in this configuration. However, factors such as scope of the work, decay heat load/heatup rate, and RCS temperature should be considered to determine if it is feasible to perform the work. Prior to using this provision, a review and approval of the evolution by the SORC is required. This review will evaluate current plant conditions and the proposed work to determine if this provision should be used, and to establish the termination criteria and appropriate contingency plans. During this period, decay heat is removed by natural convection to the large mass of water in the refueling pool.

The requirement that at least one shutdown cooling loop be in operation at ≥ 1000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) is consistent with boron dilution analysis assumptions. The 1000 gpm shutdown cooling flow limit is the minimum analytical limit. Plant operating procedures maintain the minimum shutdown cooling flow at a higher value to accommodate flow measurement uncertainties.

Average Coolant Temperature (T_{avg}) values are derived under shutdown cooling conditions, using the designated formula for use in Unit 2 operating procedures.

- SDC flow greater than 1000 gpm: $(SDC_{outlet} + SDC_{inlet}) / 2 = T_{avg}$

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION (Continued)

During SDC only operation, there is no significant flow past the loop RTDs. Core inlet and outlet temperatures are accurately measured during those conditions by using T351Y, SDC return to RCS temperature indication, and T351X, RCS to SDC temperature indication. The average of these two indicators provides a temperature that is equivalent to the average RCS temperature in the core.

T351X will not be available when using the alternate SDC suction flow path from the SFP. Substitute temperature monitoring capability shall be established to provide indication of reactor core outlet temperature. A portable temperature device can be used to indicate reactor core outlet temperature. Indication of reactor core outlet temperature from this temporary device shall be readily available to the control room personnel. A remote television camera or an assigned individual are acceptable alternative methods to provide this indication to control room personnel.

3/4.9.9 AND 3/4.9.10 DELETED

3/4.9.11 AND 3/4.9.12 WATER LEVEL-REACTOR VESSEL AND STORAGE POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.16 SHIELDED CASK

The limitations of this specification ensure that in the event of a shielded cask drop accident the doses from ruptured fuel assemblies will be within the assumptions of the safety analyses.

3/4.9.17 SPENT FUEL POOL BORON CONCENTRATION

The limitations of this specification ensures that sufficient boron is present to maintain spent fuel pool $K_{eff} \leq 0.95$ under accident conditions.

Postulated accident conditions which could cause an increase in spent fuel pool reactivity are: a single dropped or mis-loaded fuel assembly, a single dropped or mis-loaded Consolidated Fuel Storage Box, or a shielded cask drop onto the storage racks. A spent fuel pool soluble boron concentration of 1400 ppm is sufficient to ensure $K_{eff} \leq 0.95$ under these postulated accident conditions. The required spent fuel pool soluble boron concentration of ≥ 1720 ppm conservatively bounds the required 1400 ppm. The ACTION statement ensure that if the soluble boron concentration falls below the required amount, that fuel movement or shielded cask movement is stopped, until the boron concentration is restored to within limits.

An additional basis of this LCO is to establish 1720 ppm as the minimum spent fuel pool soluble boron concentration which is sufficient to ensure that the design basis value of 600 ppm soluble boron is not reached due to a postulated spent fuel pool boron dilution event. As part of the spent fuel pool criticality design, a spent fuel soluble boron concentration of 600 ppm is sufficient to ensure $K_{eff} \leq 0.95$, provided all fuel is stored consistent with LCO requirements. By maintaining the spent fuel pool soluble boron concentration ≥ 1720 ppm, sufficient time is provided to allow the operators to detect a boron dilution event, and terminate the event, prior to the spent fuel pool being diluted below 600 ppm. In the unlikely event that the spent fuel pool soluble boron concentration is decreased to 0 ppm, K_{eff} will be maintained <1.00, provided all fuel is stored consistent with LCO requirements. The ACTION statement ensures that if the soluble boron concentration falls below the required amount, that immediate action is taken to restore the soluble boron concentration to within limits, and that fuel movement or shielded cask movement is stopped. Fuel movement and shielded cask movement is stopped to prevent the possibility of creating an accident condition at the same time that the minimum soluble boron is below limits for a potential boron dilution event.

The surveillance of the spent fuel pool boron concentration within 24 hours of fuel movement, consolidated fuel movement, or cask movement over the cask layout area, verifies that the boron concentration is within limits just prior to the movement. The 7 day surveillance interval frequency is sufficient since no deliberate major replenishment of pool water is expected to take place over this short period of time.

Serial No. 07-0251
Docket No. 50-423

ATTACHMENT 2

CHANGES TO TECHNICAL SPECIFICATIONS BASES

REVISED PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Millstone Power Station Unit 3 Bases Pages

Page Changes

Section No.	Page No.
3/4.1 Reactivity Control Systems	B 3/4 1-3 B 3/4 1-4 B 3/4 1-5 B 3/4 1-6
3/4.4 Reactor Coolant System	B 3/4 4-4c B 3/4 4-5
3/4.6 Containment Systems	B 3/4 6-1 B 3/4 6-7
3/4.7 Plant Systems	B 3/4 7-2b B 3/4 7-11 B 3/4 7-15 B 3/4 7-16 B 3/4 7-17 B 3/4 7-18 B 3/4 7-19 B 3/4 7-20 B 3/4 7-21 B 3/4 7-22
3/4.9 Refueling Operations	B 3/4 9-8

Page Removals

The following pages should be removed from the MPS3 Technical Specification Bases.

3/4.1 Reactivity Control Systems	B 3/4 1-3a
3/4.7 Plant Systems	B 3/4 7-20a

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control 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 assurances 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 500°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 ≤ 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

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 not 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, 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

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 bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). These insertion limits are the initial assumptions in safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions, assumptions of available SHUTDOWN MARGIN, and initial reactivity insertion rate.

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 specified in the COLR. 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

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. Millstone Unit 3 Technical Requirements Manual, Appendix 8.1, "CORE OPERATING LIMITS REPORT".

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.

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

3. This monitoring system is not seismic Category I, but is expected to remain OPERABLE during an OBE. If the monitoring system is not OPERABLE following a seismic event, the appropriate ACTION according to Technical Specifications will be taken.

4. Two priority computer alarms (CVLKR2 and CVLKR3I) are generated if the calculated leakage rate is greater than a value specified on the Priority Alarm Point Log. This alarm value should be set to alert the Operators to a possible RCS leak rate in excess of the Technical Specification maximum allowed UNIDENTIFIED LEAKAGE. The alarm value may be set at one gallon per minute or less above the rate of IDENTIFIED LEAKAGE, from the reactor coolant or auxiliary systems, into the containment drains sump. The rate of IDENTIFIED LEAKAGE may be determined by either measurement or by analysis. If the Priority Alarm Point Log is adjusted, the high leakage rate alarm will be bounded by the IDENTIFIED LEAKAGE rate and the low leakage rate alarm will be set to notify the operator that a decrease in leakage may require the high leakage rate alarm to be reset. The priority alarm setpoint shall be no greater than 2 gallons per minute. This ensures that the IDENTIFIED LEAKAGE will not mask a small increase in UNIDENTIFIED LEAKAGE that is of concern. The 2 gallons per minute limit is also within the containment drains sump level monitoring system alarm operating range which has a maximum setpoint of 2.5 gallons per minute.

5. To convert containment drains sump run times to a leakage rate, refer to procedure SP3670.1 for guidance on the conversion method.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to 10 CFR 50.67 and Regulatory Guide 1.183 dose values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

3/4.4.7 DELETED

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting EAB, LPZ and control room doses will not exceed 10 CFR 50.67 and Regulatory Guide 1.183 dose criteria following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR 50.67 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

Primary CONTAINMENT INTEGRITY is required in MODES 1 ~~through~~ 4. This requires an OPERABLE containment automatic isolation valve system. In MODES 1, 2 and 3 this is satisfied by the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure. In MODE 4 the automatic containment isolation signals generated by high containment pressure, low pressurizer pressure and low steamline pressure are not required to be OPERABLE. Automatic actuation of the containment isolation system in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. Since the manual actuation pushbuttons portion of the containment isolation system is required to be OPERABLE in MODE 4, the plant operators can use the manual pushbuttons to rapidly position all automatic containment isolation valves to the required accident position. Therefore, the containment isolation actuation pushbuttons satisfy the requirement for an OPERABLE containment automatic isolation valve system in MODE 4.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates, as specified in the Containment Leakage Rate Testing Program, ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The Limiting Condition for Operation defines the limitations on containment leakage. The leakage rates are verified by surveillance testing as specified in the Containment Leakage Rate Testing Program, in accordance with the requirements of Appendix J. Although the LCO specifies the leakage rates at accident pressure, P_a, it is not feasible to perform a test at such an exact value for pressure. Consequently, the surveillance testing is performed at a pressure greater than or equal to P_a to account for test instrument uncertainties and stabilization changes. This conservative test pressure ensures that the measured leakage rates

3/4.6.6.2 SECONDARY CONTAINMENT

The Secondary Containment is comprised of the containment enclosure building and all contiguous buildings (main steam valve building [partially], engineering safety features building [partially], hydrogen recombiner building [partially], and auxiliary building). The Secondary Containment shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

b. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

Secondary Containment ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Supplementary Leak Collection and Release System, and Auxiliary Building Filter System will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR 50.67 during accident conditions.

The SLCRS and the ABF fans and filtration units are located in the auxiliary building. The SLCRS is described in the Millstone Unit No. 3 FSAR, Section 6.2.3.

In order to ensure a negative pressure in all areas within the Secondary Containment under most meteorological conditions, the negative pressure acceptance criterion at the measured location (i.e., 24' 6" elevation in the auxiliary building) is 0.4 inches water gauge.

LCO

The Secondary Containment OPERABILITY must be maintained to ensure proper operation of the SLCRS and the auxiliary building filter system and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

Applicability

Maintaining Secondary Containment OPERABILITY prevents leakage of radioactive material from the Secondary Containment. Radioactive material may enter the Secondary Containment from the containment following a LOCA. Therefore, Secondary Containment is required in MODES 1, 2, 3, and 4 when a design basis accident such as a LOCA could release radioactive material to the containment atmosphere.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK (Continued)

If the combined condensate storage tank (CST) and DWST inventory is being credited, there are 50,000 gallons of unusable CST inventory due to tank discharge line location, other physical characteristics, level measurement uncertainty and potential measurement bias error due to the CST nitrogen blanket. To obtain the Surveillance Requirement 4.7.1.3.2's DWST and CST combined volume, this 50,000 gallons of unusable CST inventory has been added to the 334,000 gallon DWST water volume specified in LCO 3.7.1.3 resulting in a 384,000 gallons requirement (334,000 + 50,000 = 384,000 gallons).

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to 10 CFR 50.67 and Regulatory Guide 1.183 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

BACKGROUND (Continued)

Post Accident Operation

The control room emergency ventilation system is required to operate during post-accident operations to ensure the temperature of the control room is maintained and to ensure the control room will remain habitable during and following accident conditions.

The following sequence of events occurs upon receipt of a control building isolation (CBI) signal or a signal indicating high radiation in the air supply duct to the control room envelope.

1. The control room boundary is isolated to prevent outside air from entering the control room to prevent the operators from being exposed to the radiological conditions that may exist outside the control room. The analysis for a loss of coolant accident assumes that the highest releases occur in the first hour after a loss of coolant accident.

2. After one hour, the control room emergency ventilation system will be placed in service in the filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). To run the control room emergency air filtration system in the filtered pressurization mode, the air supply line must be manually opened.

APPLICABLE SAFETY ANALYSIS

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room. For all postulated design basis accidents, the radiation exposure to personnel occupying the control room shall be 5 rem TEDE or less, consistent with the requirements of 10 CFR 50.67. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

The laboratory analysis is required to be performed within 31 days after removal of the sample. ANSI N510-1980 is used in lieu of ANSI N510-1975 referenced in Revision 2 of Regulatory Guide 1.52.

The maximum surveillance interval is 900 hours, per Surveillance Requirement 4.0.2. The 720 hours of operation requirement originates from Nuclear Regulatory Guide 1.52, Table 2, Note C. This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data.

4.7.7.e.1

This surveillance verifies that the pressure drop across the combined HEPA filters and charcoal adsorbers banks at less than 6.75 inches water gauge when the system is operated at a flow rate of 1,120 cfm ± 20%. The frequency is at least once per 24 months.

4.7.7.e.2

This surveillance verifies that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch water gauge at less than or equal to a pressurization flow of 230 cfm relative to adjacent areas and outside atmosphere during positive pressure system operation. The frequency is at least once per 24 months.

The intent of this surveillance is to verify the ability of the control room emergency air filtration system to maintain a positive pressure while running in the filtered pressurization mode.

A CBI signal will automatically align an operating filtration system into the recirculation mode of operation due to the isolation of the air supply line to the filter.

After the first hour of an event with the potential for a radiological release, the control room emergency ventilation system will be aligned in the filtered pressurization mode (outside air is diverted through the filters to the control room envelope to maintain a positive pressure). Alignment to the filtered pressurization mode requires manual operator action to open the air supply line.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.e.3

This surveillance verifies that the heaters can dissipate 9.4 ± 1 kW at 480V when tested in accordance with ANSI N510-1980. The frequency is at least once per 24 months. The heater kW measured must be corrected to its nameplate rating. Variations in system voltage can lead to measurements of kW which cannot be compared to the nameplate rating because the output kW is proportional to the square of the voltage.

4.7.7.f

Following the complete or partial replacement of a HEPA filter bank, the OPERABILITY of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 1,120 cfm ± 20%.

4.7.7.g

Following the complete or partial replacement of a charcoal adsorber bank, the OPERABILITY of the cleanup system should be confirmed. This is accomplished by verifying that the cleanup system satisfied the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow of 1,120 cfm ± 20%.

References:

(1) Nuclear Regulatory Guide 1.52, Revision 2

(2) MP3 UFSAR, Table 1.8-1, NRC Regulatory Guide 1.52

(3) NRC Generic Letter 91-04

(4) Condition Report (CR) #M3-99-0271

3/4.7.8 DELETED

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3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove at least 99% of the assumed iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.