Serial No. 07-0450
Docket No. 50-423

ATTACHMENT 4

LICENSE AMENDMENT REQUEST
STRETCH POWER UPRATE

MARK-UP OF ASSOCIATED TECHNICAL SPECIFICATIONS BASES PAGES

FOR INFORMATION ONLY

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset the effect of rod bow and any other DNB penalties that may occur. The remaining margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor, $F_Q(Z)$, is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions. Using the measured three dimensional power distributions, it is possible to derive $F_Q^M(Z)$, a computed value of $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations.

To account for these possible variations, the steady state limit of $F_Q(Z)$ is adjusted by an elevation dependent factor appropriate to either RAOC or base load operation, W(Z) or $W(Z)_{BL}$, that accounts for the calculated worst case transient conditions. The W(Z) and $W(Z)_{BL}$, factors described above for normal operation are specified in the COLR per Specification 6.9.1.6. Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion. Evaluation of the steady state $F_Q(Z)$ limit is performed in Specification 4.2.2.1.2.b and 4.2.2.1.4.b while evaluation nonequilibrium limits are performed in Specification 4.2.2.1.2.c and 4.2.2.1.4.c.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the Limiting Condition for Operation. Measurement errors of 2.4% for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been ~~allowed for~~ in determination of the design DNBR value. taken into account

INSERT 'Y'

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi will be added if venturis are not inspected and cleaned at least once for 18 months. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

Insert Y to Bases Page B 3/4 2-4

To perform the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated at least once per 18 months.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Positive Rate

positive reactivity insertion events

pressure

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of ~~a rupture of a control rod drive housing~~. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for all rod ejection accidents. This trip also complements the Pressurizer High trip, along with the Overtemperature ΔT and the Power Range Neutron Flux High Positive Rate trips, to ensure that the criteria are met for the rod withdrawal at power accidents trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer Water Level High trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Flow Low trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 38% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

50

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~~ 0.50 seconds. Since the seismic allowance should be removed when verifying the actual rod drop time, the acceptance criteria for surveillance testing is ~~2.19~~ 2.20 seconds (References 4 ~~and 5~~ 5).

Measuring rod drop times prior to reactor criticality, after reactor vessel head removal and installation, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. Any time the OPERABILITY of the control rods has been affected by a repair, maintenance, modification, or replacement activity, post maintenance testing in accordance with SR 4.0.1 is required to demonstrate OPERABILITY.

REACTIVITY CONTROL SYSTEMS

BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

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. 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.

Westinghouse Letter NEU-07-62, "MPS3 SPUP RCCA DROP TIME," dated June 4, 2007.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES — Replace with INSERT 'B'

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.

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. e

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:

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

e

BACKGROUND

The primary purpose of the main steam line Code safety valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to less than or equal to 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The design minimum total relieving capacity for all valves on all of the steam lines is 1.579×10^{7} lbs/hr which is 105 % of total secondary steam flow of 1.504×10^{7} lbs/h at 100 % RATED THERMAL POWER. The MSSV design includes staggered setpoints, according to Table 3.7-3 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip. Table 3.7-3 allows a $\pm$ 3 % setpoint tolerance (allowable value) on the lift setting for OPERABILITY to account for drift over an operating cycle.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to less than or equal to 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power turbine trip without steam dump is typically the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The FSAR Section 15.4 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. If the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs, it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions, unless it is demonstrated by analysis that a specified reactor power reduction alone is sufficient to prevent overpressurization of the steam system.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2, and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

ACTIONS are modified by a Note indicating that separate Condition entry is allowed for each MSSV.

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

a

In the case of only a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, ACTION a requires an appropriate reduction in reactor power within 4 hours. If the power reduction is not completed within the required time, the unit must be placed in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4 with an appropriate allowance for calorimetric power uncertainty.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined by the governing heat transfer relationship is the equation $q = \dot{m}\,\Delta h$, where q is the heat input from the primary side, $\dot{m}$ is the mass flow rate of the steam, and Δh is the increase in enthalpy that occurs in converting the secondary side water to steam. If it is conservatively assumed that the secondary side water is all saturated liquid (assuming no subcooled feedwater), then the Δh is the heat of vaporization (h_{fg}) at the steam relief pressure. For each steam generator, at a specified pressure, the maximum allowable power level is determined as follows:

Maximum Allowable Power Level $\leq (100/Q)\ (W_s h_{fg} N)\ /\ K$

Where:

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

K = Conversion factor, 947.82 (Btu/sec)/MWt

W_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest OPERABLE MSSV opening pressure including tolerance and accumulation, as appropriate, lb/sec.

h_{fg} = Heat of vaporization at the highest MSSV opening pressure including tolerance and accumulation as appropriate, Btu/lbm.

N = Number of loops in the plant.

For use in determining the %RTP in ACTION a, the Maximum NSSS Power calculated above is reduced by 2% RTP to account for calorimetric power uncertainty.

b and c

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour completion time to reduce reactor power is consistent with ACTION a. An additional 32 hours is allowed to reduce the Power Range Neutron Flux High reactor setpoint. The total completion time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time to perform the power reduction, operating experience to reset all channels of a protection function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. If the required action is not completed within the associated time, the unit must be placed in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4, with an appropriate allowance for nuclear instrumentation system trip channel uncertainties.

To determine the Table 3.7-1 Maximum Allowable Power for Required ACTIONS b and c (%RTP), the calculated Maximum NSSS Power is reduced by 9% RTP to account for Nuclear Instrumentation System trip channel uncertainties.

ACTIONS b and c are modified by a Note. The Note states that the Power Range Neutron Flux High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 2.2.1, "Reactor Trip System Instrumentation Setpoints," provide sufficient protection.

The allowed completion times are reasonable based on operating experience to accomplish the ACTIONS in an orderly manner without challenging unit systems.

d

If one or more steam generators have four or more inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS (SR) 4.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint (Table 3.7-3) in accordance with the Inservice Testing Program. During this testing, the MSSVs are OPERABLE provided the actual lift settings are within ± 3 % of the required lift setting. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7-3 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for 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 setting, the MSSVs remain OPERABLE provided the actual lift setting is within ± 3 % of the required setting. The lift settings, according to Table 3.7-3, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.3.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, 1971 edition.

3. FSAR, Section 15.2.

4. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES (Continued)

where:

$Hi\phi$ = 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

SAFETY VALVES (Continued)

w_s = Minimum total steam flow rate capability of the OPERABLE MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the OPERABLE MSSVs at the highest OPERABLE MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the OPERABLE MSSVs at the highest OPERABLE MSSV operating pressure, excluding the three highest capacity MSSVs. The following plant specific safety valve flow rates were used:

SG Safety Valve Number (Bank No.)	Main Steam System	
	Set Pressure (psia)	Flow (lbm/hr per loop)
1	1200	893,160
2	1210	900,607
3	1220	908,055
4	1230	915,502
5	1240	922,950

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater (AFW) System ensures a makeup water supply to the steam generators (SGs) to support decay heat removal from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply, assuming the worst case single failure. The AFW System consists of two motor driven AFW pumps and one steam turbine driven AFW pump. Each motor driven AFW pump provides at least 50% of the AFW flow capacity assumed in the accident analysis. After reactor shutdown, decay heat eventually decreases so that one motor driven AFW pump can provide sufficient SG makeup flow. The steam driven AFW pump has a rated capacity approximately double that of a motor driven AFW pump and is thus defined as a 100% capacity pump.

Given the worst case single failure, the AFW System is designed to mitigate the consequences of numerous design basis accidents, including Feedwater Line Break, Loss of Normal Feedwater, Steam Generator Tube Rupture, Main Steam Line Break, and Small Break Loss of Coolant Accident.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK

7

The OPERABILITY of the demineralized water storage tank (DWST) with a 334,000 gallon minimum measured water volume ensures that sufficient water is available to maintain the reactor coolant system at HOT STANDBY conditions for $\boxed{10}$ hours with steam discharge to the atmosphere, concurrent with a total loss-of-offsite power, and with an additional 6-hour cooldown period to reduce reactor coolant temperature to 350°F. The 334,000 gallon required water volume contains an allowance for tank inventory not usable because of tank discharge line location, other tank physical characteristics, and surveillance measurement uncertainty considerations. The inventory requirement is conservatively based on 120°F water temperature which maximizes inventory required to remove RCS decay heat. In the event of a feedline break, this inventory requirement includes an allowance for 30 minutes of spillage before operator action is credited to isolate flow to the line break.

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)

LIMITING CONDITION FOR OPERATION

Two independent control room emergency air filtration systems are required to be OPERABLE to ensure that at least one is available in the event the other system is disabled.

A control room emergency air filtration system is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions; and

c. moisture separator, heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The integrity of the control room habitability boundary (i.e., walls, floors, ceilings, ductwork, and access doors) must be maintained such that the control building habitability zone can be maintained at its design positive pressure if required to be aligned in the filtration pressurization mode. However, the LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, 5, and 6.
During fuel movement within containment or the spent fuel pool.

ACTIONS a., b., and c. of this specification are applicable at all times during plant operation in MODES 1, 2, 3, and 4. ACTIONS d. and e. are applicable in MODES 5 and 6, and whenever fuel is being moved within containment or the spent fuel pool. The fuel handling accident analyses assume that during a fuel handling accident some of the fuel that is dropped and some of the fuel impacted upon is damaged. Therefore, the movement of either new or irradiated fuel (assemblies or individual fuel rods) can cause a fuel handling accident, and this specification is applicable whenever new or irradiated fuel is moved within the containment or the storage pool.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

→ INSERT 1

ACTIONS

MODES 1, 2, 3, and 4

a. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. In this condition, the remaining control room emergency air filtration system is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE train could result in a loss of the control room emergency air filtration system function. The 7-day completion time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

   If the inoperable train cannot be restored to an OPERABLE status within 7 days, the unit must be placed in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit condition from full power conditions in an orderly manner and without challenging unit systems.

b. With both control room emergency air filtration systems inoperable, except due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. At least one control room emergency air filtration system must be restored to OPERABLE status within 1 hour, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. These completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

c. With both control room emergency air filtration systems inoperable due to an inoperable control room boundary, the movement of fuel within the spent fuel pool must be immediately suspended. The control room boundary must be restored to OPERABLE status within 24 hours, or the unit must be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the control room emergency air filtration systems cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be

<u>Insert '1' To Bases Page B 3/4 7-13</u>

An analysis was completed that analyzed a bounding drop of a non-spent fuel component. The analysis showed that the amount of fuel damage from this drop resulted in control room dose less than 5 rem TEDE without operation of the control room ventilation system.

PLANT SYSTEMS

BASES

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

ACTIONS (Continued)

available to address these concerns for intentional and unintentional entry in to this condition. The 24 hour allowed outage time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour allowed outage time is a typically reasonable time to diagnose, plan, and possibly repair, and test most problems with the control room boundary.

MODES 5 and 6, and fuel movement within containment or the spent fuel pool

d. With one control room emergency air filtration system inoperable, action must be taken to restore the inoperable system to an OPERABLE status within 7 days. After 7 days, either initiate and maintain operation of the remaining OPERABLE control room emergency air filtration system in the recirculation mode or suspend the movement of fuel. Initiating and maintaining operation of the OPERABLE train in the recirculation mode ensures: (i) OPERABILITY of the train will not be compromised by a failure of the automatic actuation logic; and (ii) active failures will be readily detected.

e. With both control room emergency air filtration systems inoperable, or with the train required by ACTION 'd' not capable of being powered by an OPERABLE emergency power source, actions must be taken to suspend all operations involving the movement of fuel. This action places the unit in a condition that minimizes risk. This action does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

4.7.7.a

The control room environment should be checked periodically to ensure that the control room temperature control system is functioning properly. Verifying that the control room air temperature is less than or equal to 95°F at least once per 12 hours is sufficient. It is not necessary to cycle the control room ventilation chillers. The control room is manned during operations covered by the technical specifications. Typically, temperature aberrations will be readily apparent.

4.7.7.b

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing the trains once every 31 days on a STAGGERED TEST BASIS provides an adequate check of this system. This surveillance requirement verifies a system flow rate of 1,120 cfm ± 20%. Additionally, the system is required to operate for at least 10 continuous hours with the heaters energized. These operations are sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters due to the humidity in the ambient air.

3/4.9.13 SPENT FUEL POOL - REACTIVITY

During normal Spent Fuel Pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than 0.95 in an unborated water environment.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 3-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, a maximum nominal 5 weight percent fuel enrichment, and the use of blocking devices in certain fuel storage locations, as specified by the interface requirements shown in Figure 3.9-2.

Maintaining K_{eff} at less than or equal to 0.95 is accomplished in Region 1 4-OUT-OF-4 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup specified in Figure 3.9-1.

Maintaining Keff at less than or equal to 0.95 is accomplished in Region 2 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in the racks, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-3.

Maintaining Keff at less than or equal to 0.95 is accomplished in Region 3 storage racks by the combination of geometry of the rack spacing, and the limits on fuel enrichment/fuel burnup and fuel decay time specified in Figure 3.9-4 for assemblies used exclusively in the pre-uprate (3411 MWt) cores and Figure 3.9-5 for assemblies used in the post-uprate (3650 MWt) cores. Fixed neutron absorbers are not credited in the Region 3 fuel storage racks.

The limitations described by Figures 3.9-1, 3.9-2, 3.9-3 ~~and 3.9-4~~, 3.9-4, and 3.9.5 ensure that the reactivity of the fuel assemblies stored in the spent fuel pool are conservatively within the assumptions of the safety analysis .

Administrative controls have been developed and instituted to verify that the fuel enrichment, fuel burnup, fuel decay times, and fuel interface restrictions specified in Figures 3.9-1, 3.9-2, 3.9-3 ~~and 3.9-4~~, 3.9-4, and 3.9.5 are complied with.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region 1 3-OUT-OF-4 storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region 1 3-OUT-OF-4 storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region 1 storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region 1 3-OUT-OF-4 boundary to other storage regions and minimize the number of boundaries where a fuel misplacement incident can occur.