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Dominion

APR 2 2002

Docket No. 50-423
B18620

RE: 10 CFR 50.59

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

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

Dominion Nuclear Connecticut, Inc. (DNC) is providing the Nuclear Regulatory Commission Staff with changes to Millstone Unit No. 3 Technical Specifications Bases, Sections 3/4.1.3, 3/4.3.1, 3/4.3.2, 3/4.3.3.6, 3/4.4.1, 3/4.4.4, 3/4.4.6.1, 3/4.4.6.2, 3/4.5.2, 3/4.5.3, 3/4.6.2.1, 3/4.6.2.2, 3/4.6.3, 3/4.6.6.1, 3/4.7.1.2, 3/4.7.8, 3/4.7.9, 3/4.8.1, 3/4.8.2, 3/4.8.3, 3/4.9.1.1, and 3/4.9.12 for information only. The proposed changes to the Bases Sections were made in accordance with the provisions of 10 CFR 50.59. These changes were reviewed and approved by the Site Operations Review Committee.

Attachment 1 provides the retyped pages of the Technical Specification Bases for Millstone Unit No. 3.

There are no regulatory commitments contained in this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Attachment (1)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC Senior Resident Inspector, Millstone Unit No. 3

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Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Change to Technical Specifications Bases
Retyped Pages

Millstone Unit No. 3 Bases Pages

Section(s)	Page No.
3/4.1.3	B 3/4 1-4, 1-5, 1-6
3/4.3.1 and 3/4.3.2	B 3/4 3-2a, 3-2b
3/4.3.3.6	B 3/4 3-5, 3-5a
3/4.4.1	B 3/4 4-1b, 4-1c, 4-1d
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3/4.6.2.1 and 3/4.6.2.2	B 3/4 6-2, 6-2a
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3/4.7.8	B 3/4 7-17
3/4.7.9	B 3/4 7-23, 7-23a
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3/4.9.1.1	B 3/4 9-1
3/4.9.12	B 3/4 9-8, 9-9

MOVABLE CONTROL ASSEMBLIES (Continued)

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that the Digital Rod Position Indicator agrees with the demanded position within ±12 steps at 24, 48, 120, and fully withdrawn position for the Control Banks and 18, 210, and fully withdrawn position for the Shutdown Banks provides assururances that the Digital Rod Position Indicator is operating correctly over the full range of indication.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

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

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

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

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

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

With the above definition, LCO, 3.1.3.2, "ACTION a." is 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 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.

MOVABLE CONTROL ASSEMBLIES (Continued)

Additional surveillance is required to ensure the plant process computer indications are in agreement with those displayed on the DRPI. This additional SURVEILLANCE REQUIREMENT is as follows:

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

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

For Specification 3.1.3.3, the ACTION states to immediately open the reactor trip breakers if the LCO is not satisfied, however, it is appropriate to select Data A and Data B individually, to verify satisfactory rod position indication is not available before opening the reactor trip breakers.

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

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

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

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

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

MOVABLE CONTROL ASSEMBLIES (Continued)

References:

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

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

3. COLR Section 2.3

REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) quench spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start, (10) service water pumps start and automatic valves position, and (11) Control Room isolates.

For slave relays, or any auxiliary relays in ESFAS circuits that are of the type Potter & Brumfield MDR series relays, the SLAVE RELAY TEST is performed at an "R" frequency (at least once every 18 months) provided the relays meet the reliability assessment criteria presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR series relays," and WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals." The reliability assessments performed as part of the aforementioned WCAPs are relay specific and apply only to Potter and Brumfield MDR series relays. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.

REACTOR TRIP BREAKER

This trip function applies to the reactor trip breakers (RTBs) exclusive of individual trip mechanisms. The LCO requires two operable trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the control rod drive (CRD) system. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

These trip functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip functions must be OPERABLE when the RTBs or associated bypass breakers are closed, and the CRD system is capable of rod withdrawal.

BYPASSED CHANNEL* - Technical Specifications 3.3.1 and 3.3.2 often allow the bypassing of instrument channels in the case of an inoperable instrument or for surveillance testing.

A BYPASSED CHANNEL shall be a channel which is:

- Required to be in its accident or tripped condition, but is not presently in its accident or tripped condition using a method described below; or

REACTOR TRIP BREAKER (Continued)

- Prevented from tripping.

A channel may be bypassed by:

- Insertion of a simulated signal to the bistable; or
- Failing the transmitter or input device to the bypassed condition; or
- Returning a channel to service in a untripped condition; or
- An equivalent method, as determined by Engineering and I&C

*Bypass switches exist only for NIS source range, NIS intermediate range, and containment pressure Hi-3.

TRIPPED CHANNEL - Technical Specifications 3.3.1 and 3.3.2 often require the tripping of instrument channels in the case of an inoperable instrument or for surveillance testing.

A TRIPPED CHANNEL shall be a channel which is in its required accident or tripped condition.

A channel may be placed in trip by:

- The Bistable Trip Switches; or
- Insertion of a simulated signal to the bistable; or
- Failing the transmitter or input device to the tripped condition; or
- An equivalent method, as determined by Engineering and I&C

REMOTE SHUTDOWN INSTRUMENTATION (Continued)

instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The instrumentation included in this specification are those instruments provided to monitor key variables, designated as Category 1 instruments following the guidance for classification contained in Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident."

Action Statement "a":

The use of one main control board indicator and one computer point, total of two indicators per steam generator, meets the requirements for the total number of channels for Auxiliary Feedwater flow rate. The two channels used to satisfy this Technical Specification for each steam generator are as follows:

Steam Generator	Instrument	(MB5)	Instrument	(Computer)
S/G 1	FWA*FI51A1	(Orange)	FWA - F33A3	(Purple)
S/G 2	FWA*FI33B1	(Purple)	FWA - F51B3	(Orange)
S/G 3	FWA*FI33C1	(Purple)	FWA - F51C3	(Orange)
S/G 4	FWA*FI51D1	(Orange)	FWA - F33D3	(Purple)

The SPDS computer point for auxiliary feedwater flow will be lost 30 minutes following an LOP when the power supply for the plant computer is lost. However, this design configuration - one continuous main control board indicator and one indication via the SPDS/plant computer, total of two per steam generator - was submitted to the NRC via "Response to question 420.6" dated January 13, 1984, B11002. NRC review and approval was obtained with the acceptance of MP3, SSER 4 Appendix L, "Conformance to Regulatory Guide 1.97," Revision 2. (dated November 1985).

LCO 3.3.3.6, Table 3.3.10, Item (17), requires 2 OPERABLE reactor vessel water level (heated junction thermocouples - HJTC) channels.
An OPERABLE reactor vessel water level channel shall be defined as:

1. Four or more total sensors operating.
2. At least one of two operating sensors in the upper head.
3. At least three of six operating sensors in the upper plenum.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

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

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

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

3/4.3.3.7 Deleted.

RCS Loops Filled/Not Filled:

In MODE 5, any RHR train with only one cold leg injection path is sufficient to provide adequate core cooling and prevent stratification of boron in the Reactor Coolant System.

The definition of operability states that the system or subsystem must be capable of performing its specified function(s). The reason for the operation of one reactor coolant pump (RCP) or one RHR pump is to:

- Provide sufficient decay heat removal capability
- Provide adequate flow to ensure mixing to:
    - Prevent stratification
    - Produce gradual reactivity changes due to boron concentration changes in the RCS

The definition of "Reactor coolant loops filled" includes a loop that is filled, swept, and vented, and capable of supporting natural circulation heat transfer. This allows the non-operating RHR loop to be removed from service while filling and unisolating loops as long as steam generators on the operable reactor coolant loops are available to support decay heat removal. Any loop being unisolated is not OPERABLE until the loop has been swept and vented. The process of sweep and vent will make the previously OPERABLE loops inoperable and the requirements of LCO 3.4.1.4.2, "Reactor Coolant System, Cold Shutdown -Loops Not Filled," are applicable. When the RCS has been filled, swept and vented using an approved procedure, all unisolated loops may be declared OPERABLE.

One cold leg injection isolation valve on an RHR train may be closed without considering the train to be inoperable, as long as the following conditions exist:

- CCP temperature is at or below 95°F
- Initial RHR temperature is below 184°F
- The single RHR cold leg injection flow path is not utilized until a minimum of 24 hours after reactor shutdown
- CCP flow is at least 6,600 gpm
- RHR flow is at least 2,000 gpm

In the above system lineup, total flow to the core is decreased compared to the flow when two cold legs are in service. This is acceptable due to the substantial margin between the flow required for cooling and the flow available, even through a slightly restricted RHR train.

The review concerning boron stratification with the utilization of the single injection point line, indicates there will not be a significant change in the flow rate or distribution through the core, so there is not an increased concern due to stratification.

Flow velocity, which is high, is not a concern from a flow erosion or pipe loading standpoint. There are no loads imposed on the piping system which would exceed those experienced in a seismic event. The temperature of the fluid is low and is not significant from a flow erosion standpoint.

The boron dilution accident analysis, for Millstone Unit 3 in MODE 5, assumes a full RHR System flow of approximately 4,000 gpm. Westinghouse analysis, Reference (1), for RHR flows down to 1,000 gpm, determined adequate mixing results. As the configuration will result in a RHR flow rate only slightly less then 4,000 gpm there is no concern in regards to a boron dilution accident.

The basis for the requirement of two RCS loops OPERABLE is to provide natural circulation heat sink in the event the operating RHR loop is lost. If the RHR loop were lost, with two loops swept and vented and two loops air bound, natural circulation would be established in the two swept loops.

Natural circulation would not be established in the air bound loops. Since there would be no circulation in the air bound loops, there would be no mechanism for the air in those loops to be carried to the vessel, and subsequently into the swept loops rendering them inoperable for heat sink requirements.

The LCO is met as long as at least two reactor coolant loops are OPERABLE and the following conditions are satisfied:

- One RHR loop is OPERABLE and in operation, with exceptions as allowed in Technical Specifications; and
  Either of the following:
- An additional RHR loop OPERABLE, with exceptions as allowed in Technical Specifications; or
- The secondary side water level of at least two steam generators shall be greater than 17% (These are assumed to be on OPERABLE reactor coolant loops)

When the reactor coolant loops are swept, the mechanism exists for air to be carried into previously OPERABLE loops. All previously OPERABLE loops are declared inoperable and an additional RHR loop is required OPERABLE as specified by LCO 3.4.1.4.2 for loops not filled. When the RCS has been filled, swept, and vented using an approved procedure, all unisolated loops may be declared OPERABLE.

ISOLATED LOOP STARTUP

The below requirements are for unisolating a loop with all four loops isolated while decay heat is being removed by RHR and to clarify prerequisites to meet T/S requirements for unisolating a loop at any time.

With no RCS loops operating, the two RHR loops referenced in Specification 3.4.1.4.2 are the operating loops. Starting in MODE 4 as referenced in Specification 3.4.1.3, the RHR loops are allowed to be used in place of an operating RCS loop. Specification 3.4.1.4.2 requires two RHR loops OPERABLE and at least one in operation. Ensuring the isolated cold leg temperature is within 20°F of the highest RHR outlet temperature for the operating RHR loops within 30 minutes prior to opening the cold leg stop valve is a conservative approach since the major concern is a positive reactivity addition.

SR 4.4.1.6.1: When in MODE 5 with all RCS loops isolated, the two RHR loops referenced in LCO 3.4.1.4.2 shall be considered the OPERABLE RCS loops.

ISOLATED LOOP STARTUP (Continued)

The isolated loop cold leg temperature shall be determined to be within 20°F of the highest RHR outlet temperature for the operating RHR loops within 30 minutes prior to opening the cold leg stop valve.

Surveillance requirement 4.4.1.6.2 is met when the following actions occur within 2 hours prior to opening the cold leg or hot leg stop valve:

- An RCS boron sample has been taken and analyzed to determine current boron concentration
- The SHUTDOWN MARGIN has been determined using OP 3209B, "Shutdown Margin" using the current boron concentration determined above
- For the isolated loop being restored, the power to both loop stop valves has been restored

Surveillance 4.4.1.6.2 indicates that the reactor shall be determined subcritical by at least the amount required by Specifications 3.1.1.1.2 or 3.1.1.2 for MODE 5 or Specification 3.9.1.1 for MODE 6 within 2 hours of opening the cold leg or hot leg stop valve.

Specification 3.1.1.1.2 requires the SHUTDOWN MARGIN to be as shown in Figure 3.1-2 for three loop operation. Figure 3.1-2 is for three loop operation in MODE 3. The other figures, as used by this specification, require four loop operation, so cannot be used to determine the required SHUTDOWN MARGIN for the MODE 5 loops isolated condition.

Specification 3.1.1.2 requires the SHUTDOWN MARGIN to be as shown in Figure 3.1-5 or Figure 3.1-4 with CVCS aligned to preclude boron dilution. This specification is for loops not filled and therefore is applicable to an all loops isolated condition.

Specification 3.9.1.1 requires K_{eff} of 0.95 or less, or a boron concentration of greater than or equal to 2,600 ppm in MODE 6.

Specification 3.1.1.1.2 or 3.1.1.2 for MODE 5, both require boron concentration to be determined at least once each 24 hours. SR 4.1.1.1.2.1.b.2 and 4.1.1.2.1.b.1 satisfy the requirements of Specifications 3.1.1.1.2 and 3.1.1.2 respectfully. Specification 3.9.1.1 for MODE 6 requires boron concentration to be determined at least once each 72 hours. S.R.4.9.1.1.2 satisfy the requirements of Specification 3.9.1.1.

References:

1. Letter NEU-94-623, dated July 13, 1994; Mixing Evaluation for Boron Dilution Accident in Modes 4 and 5, Westinghouse HR-59782.

2. Memo No. MP3-E-93-821, dated October 7, 1993.

3/4.4.3 PRESSURIZER (cont'd.)

The 12-hour periodic surveillance requires that during MODE 3 operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The surveillance is performed by observing the indicated level. The 12-hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and to ensure that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

The basis for the pressurizer heater requirements is identical to MODES 1 and 2.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Requiring the PORVs to be OPERABLE ensures that the capability for depressurization during safety grade cold shutdown is met.

Action statements a, b, and c distinguishes the inoperability of the power operated relief valves (PORV). Specifically, a PORV may be designated inoperable but it may be able to automatically and manually open and close and therefore, able to perform its function. PORV inoperability may be due to seat leakage which does not prevent automatic or manual use and does not create the possibility for a small-break LOCA. For these reasons, the block valve may be closed but the action requires power to be maintained to the valve. This allows quick access to the PORV for pressure control. On the other hand if a PORV is inoperable and not capable of being automatically and manually cycled, it must be either restored or isolated by closing the associated block valve and removing power.

Note: PORV position indication does not affect the ability of the PORV to perform any of its safety functions. Therefore, the failure of PORV position indication does not cause the PORV to be inoperable. However, failed position indication of these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3.

Automatic operation of the PORVs is created to allow more time for operators to terminate an Inadvertent ECCS Actuation at Power. The PORVs and associated piping have been demonstrated to be qualified for water relief. Operation of the PORVs will prevent water relief from the pressurizer safety valves for which qualification for water relief has not been demonstrated. If the PORVs are capable of automatic operation but have been declared inoperable, closure of the PORV block valve is acceptable since the Emergency Operating Procedures provide guidance to assure that the PORVs would be available to mitigate the event. Operability and setpoint controls for the safety grade PORV opening logic are maintained in the Technical Requirements Manual.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

LCO 3.4.6.1.b. Containment Sump Drain Level or Pumped Capacity Monitoring System

The intent of LCO 3.4.6.1.b is to have a system able to monitor and detect leakage from the reactor coolant pressure boundary (RCPB). The system can use sump level, pump capacity or both as the LCO implies. It does not have to have two separate systems. The "Containment Drain Sump Level or Pumped Capacity Monitoring" System is defined as any one of the following three Systems:

A. 3DAS-P10, Unidentified Leakage Sump Pump, and associated local and main board annunciation.

B. 3DAS-P10, Unidentified Leakage Sump Pump, and computer point 3DAS-L39 and CVLKR2.

C. 3DAS-P2A or 3DAS-P2B, Containment Drains Sump Pump, and computer points 3DAS-L22 and CVLKR2 or CVLKR3I.

To meet Regulatory Guide 1.45 recommendations, the Containment Drain Sump Level or Pumped Capacity Monitoring System must meet the following five criteria:

1. Must monitor changes in sump water level, changes in flow rate or changes in the operating frequency of pumps.

2. Be able to detect an UNIDENTIFIED LEAKAGE rate of 1 gpm in less than one hour.

3. Remain operable following an Operating Basis Earthquake (OBE).

4. Provide indication and alarm in the Control Room.

5. Procedures for converting various indications to a common leakage equivalent must be available to the Operators.

The three Containment Drain Sump Level or Pumped Capacity Monitoring Systems identified above meet these five requirements as follows:

A. 3DAS-P10, Unidentified Leakage Sump Pump, and associated main board annunciation.

   1. Sump level is monitored at two locations by the starting and stopping of 3DAS-P10, Unidentified Leakage Sump Pump. Flow is measured as a function of time between pump starts/stops and the known sump levels at which these occur.

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

2. Two timer relays in the control circuitry of 3DAS-P10 are set to identify a 1 gpm leak rate within 1 hour.

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. This position has been reviewed by the NRC and documented as acceptable in the Safety Evaluation Report.

4. If the control circuitry of 3DAS-P10 identifies a 1 gpm leak rate within 1 hour, Liquid Radwaste Panel Annunciator LWS 4-5, CTMT UNIDENT LEAKAGE TROUBLE, and Main Board Annunciator MB1 B 4-3, RAD LIQUID WASTE SYS TROUBLE, will alarm. These control circuits and alarms operate independently from the plant process computer.

   If the computer is inoperable, these control circuits and alarms meet the Technical Specification requirements for the Containment Drain Sump Level or Pumped Capacity Monitoring System.

5. To convert the unidentified leakage sump pump run times to a leakage rate, use the following formula:

$$\frac{\text{(3DAS-P10 run times in minutes - [number of 3DAS-P10 starts x.5 minutes]) x 20 gpm}}{\text{Elapsed monitored Time in minutes}}$$

B. <u>3DAS-P10, Unidentified Leakage Sump Pump, and computer points 3DAS-L39 and CVLKR2.</u>

1. Sump level is monitored by 3DAS-LI39, the Unidentified Leakage Sump Level indicator. This level indicator provides an input to computer point 3DAS-L39.

2. The plant process computer calculates a leakage rate every 30 seconds when 3DASP10 indicates stop. This leakage rate is displayed via computer point CVLKR2. When pump P10 does run, the leakage rate calculation is stopped and resumes 10 minutes after pump P10 stops. If it cannot provide a value of the leakage rate within any 54 minute interval, CVDASP10NC (UNDNT LKG RT NOT CALC) alarms which alerts the Operator that Unidentified Leakage cannot be determined.

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. A priority computer alarm (CVLKR2) is 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

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

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 unidentified leakage sump. The rate of identified leakage may be determined by either measurement or 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 set point 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 identified leakage sump level monitoring system alarm operating range which has a maximum set point of 2.3 gallons per minute.

5. To convert unidentified leakage sump level changes to leakage rate, use the following formula:

Note: Wait 10 minutes after 3DAS-P10 stops before taking level readings.

$$\frac{1.08315 \text{ gallons}}{1\%} \quad x \quad \frac{\text{\% change in level from 3DAS-L39}}{\text{time between level readings in minutes}}$$

C. 3DAS-P2A or 3DAS-P2B, Containment Drains Sump Pump, and computer points 3DAS-L22 and CVLKR2 or CVLKR3I.

1. Sump level is monitored by 3DAS-LI22, the Containment Drains Sump Level Indicator. This level indicator provides an input to computer point 3DAS-L22.

This method can be used to monitor Unidentified Leakage when Pump P10 and its associated equipment is inoperable provided Pump P10 is out of service and 3DAS-LI39 indicates that the unidentified leakage sump is overflowing to the containment drains sump (approximately 36% level on 3DAS-LI39). In this case, CVLKR2 and CVLKR3I monitor flow rate by comparing level indications on the containment drains sump when Pumps P10, P2A, P2B and P1 are not running.

2. The plant process computer calculates a leakage rate every 30 seconds when 3DAS-P10, 3DAS-P1, 3DAS-P2A and 3DAS-P2B indicate stop. This leakage rate is displayed via computer points CVLKR3I and CVLKR2 when 3DAS-P10 is off and when the unidentified leakage sump is overflowing to the containment drains sump. When one of these pumps does run, the leakage rate calculation is stopped and resumes 10 minutes after all pumps stop. If it cannot provide a value of the leakage rate within any 54 minute interval, two computer point alarms (CVDASP2NC, UNDNT LKG RT NOT CALC and CVDASP2NC, SMP 3 LKG RT NT CALC) are generated which alerts the Operator that Unidentified Leakage cannot be determined.

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS (Continued)

3. This monitoring system is not seismic Category 1, 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 set point 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 set point 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 a small fraction of 10 CFR Part 100 dose guideline 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

A Limit of 40 gpm is placed on CONTROLLED LEAKAGE. CONTROLLED LEAKAGE is determined under a set of reference conditions, listed below:

a. One Charging Pump in operation.
b. RCS pressure at 2250 +/- 20 psia.

By limiting CONTROLLED leakage to 40 gpm during normal operation, we can be assured that during an SI with only one charging pump injecting, RCP seal injection flow will continue to remain less than 80 gpm as assumed in accident analysis. When the seal injection throttle valves are set with a normal charging line up, the throttle valve position bounds conditions where higher charging header pressures could exist. Therefore, conditions which create higher charging header pressures such as an isolated charging line, or two pumps in service are bounded by the single pump - normal system lineup surveillance configuration. Basic accident analysis assumptions are that 80 gpm flow is provided to the seals by a single pump in a runout condition.

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

References:

1. Letter FSD/SS-NEU-3713, dated March 25, 1985.
2. Letter NEU-89-639, dated December 4, 1989.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values 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

ECCS SUBSYSTEMS (Continued)

- the RSS pumps, since this equipment is laid-up dry during plant operation.
- the RSS heat exchangers, since this equipment is laid-up dry during plant operation.
- the RSS piping that is not maintained filled with water during plant operation.

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

ECCS Subsystems: Auxiliary Building RPCCW Ventilation Area Temperature Maintenance:

In MODES 1, 2, 3 and 4, two trains of 4 heaters each, powered from class 1E power supplies, are required to support charging pump OPERABILITY during cold weather conditions. These heaters are required whenever outside temperature is less than or equal to 17°F.

When outside air temperature is below 17°F, if both trains of heaters in the RPCCW Ventilation Area are available to maintain at least 65°F in the Charging Pump and Reactor Component Cooling Water Pump areas of the Auxiliary Building, both charging pumps are OPERABLE for MODES 1, 2 and 3.

When outside air temperature is below 17°F, if one train of heaters in the RPCCW Ventilation Area is available to maintain at least 32°F in the Charging Pump and Reactor Component Cooling Water Pump areas of the Auxiliary Building, the operating charging pump is OPERABLE, for MODE 4.

With less than 4 operable heaters in either train, the corresponding train of charging is inoperable. This condition will require entry into the applicable action statement for LCO's 3.5.2 and 3.5.3.

LCO 3.5.2 action statement "b", and LCO 3.5.3 action statement "c" address special reporting requirements in response to ECCS actuation with water injection to the RCS. The special report completion is not a requirement for logging out of the action statements that require the reports.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following a large break (LB) LOCA, assuming mixing of the RWST, RCS, ECCS water, and other sources of water that may eventually reside in the sump, with all control rods assumed to be out. These assumptions are consistent with the LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 7.5 for the solution recirculated within containment after a LOCA.. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum/minimum solution temperatures for the RWST in MODES 1, 2, 3 and 4 are based on analysis assumptions.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 60 psia in the event of a LOCA. A visual inspection, in accordance with the Containment Leakage Rate Testing Program, is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be locked closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The Type C testing frequency required by 4.6.1.2 is acceptable, provided that the resilient seats of these valves are replaced every other refueling outage.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH SPRAY SYSTEM and RECIRCULATION SPRAY SYSTEM

The OPERABILITY of the Containment Spray Systems ensures that containment depressurization and iodine removal will occur in the event of a LOCA. The pressure reduction, iodine removal capabilities and resultant containment leakage are consistent with the assumptions used in the safety analyses.

LCO 3.6.2.2

One Recirculation Spray System consists of:

- Two OPERABLE containment recirculation heat exchangers
- Two OPERABLE containment recirculation pumps

The Containment Recirculation Spray System (RSS) consists of two parallel redundant subsystems which feed two parallel 360 degree spray headers. Each subsystem consists of two pumps and two heat exchangers. Train A consists of 3RSS*P1A and 3RSS*P1C. Tain B consists of 3RSS*P1B and 3RSS*P1D.

The design of the Containment RSS is sufficiently independent so that an active failure in the recirculation spray mode, cold leg recirculation mode, or hot leg recirculation mode of the ECCS has no effect on its ability to perform its engineered safety function. In other words, the failure in one subsystem does not affect the capability of the other subsystem to perform its designated safety function of assuring adequate core cooling in the event of a design basis LOCA. As long as one subsystem is OPERABLE, with one pump capable of assuring core cooling and the other pump capable of removing heat from containment, the RSS system meets its design requirements.

The LCO 3.6.2.2. ACTION applies when any of the RSS pumps, heat exchangers, or associated components are declared inoperable. All four RSS pumps are required to be OPERABLE to meet the requirements of this LCO 3.6.2.2. During the injection phase of a Loss Of Coolant Accident all four RSS pumps would inject into containment to perform their containment heat removal function. The minimum requirement for the RSS to adequately perform this function is to have at least one subsystem available. Meeting the requirements of LCO 3.6.2.2. ensures the minimum RSS requirements are satisfied.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. FSAR Table 6.2-65 lists all containment isolation valves. The addition of deletion of any containment isolation valve shall be made in accordance with Section 50.59 of 10CFR50 and approved by the committee(s) as described in the NUQAP Topical Report.

For the purposes of meeting this LCO, the safety function of the containment isolation valves is to shut within the time limits assumed in the accident analyses. As long as the valves can shut within the time limits assumed in the accident analyses, the valves are OPERABLE. Where the valve position indication does not affect the operation of the valve, the indication is not required for valve operability under this LCO. Position indication for containment isolation valves is covered by Technical Specification 6.8.4.e., Accident Monitoring Instrumentation. Failed position indication on these valves must be restored "as soon as practicable" as required by Technical Specification 6.8.4.e.3. Maintaining the valves OPERABLE, when position indication fails, facilitates troubleshooting and correction of the failure, allowing the indication to be restored "as soon as practicable."

If the containment isolation valve on a closed system becomes inoperable, the remaining barrier is a closed system since a closed system is an acceptable alternative to an automatic valve. However, actions must still be taken to meet Technical Specification ACTION 3.6.3.c and the valve, not normally considered as a containment isolation valve, and closest to the containment wall should be put into the locked closed position. No leak testing of the alternate valve is necessary to satisfy the action statement. Placing the manual valve in the locked closed position sufficiently deactivates the penetration for Technical Specification compliance.

For the purposes of meeting this LCO, neither the containment isolation valve, nor any alternate valve on a closed system have a leakage limit associated with valve operability.

3/4.6.4 COMBUSTIBLE GAS CONTROL

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

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the Mechanical Vacuum Pumps are capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen

3/4.6.4 COMBUSTIBLE GAS CONTROL (Continued)

Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Post-LOCA performance of the hydrogen recombiner blowers is based on a series of equations supplied by the blower manufacturer. These equations are also the basis of the acceptance criteria used in the surveillance procedure. The required performance was based on starting containment conditions before the LOCA of 10.59 psia (total pressure), 120°F and 100% relative humidiy.

The surveillance procedure shall use the following methods to verify acceptable blower flow rate:

1. Definitions and constants

   CFM = cubic feet per minute

   RPM = revolutions per minute

   Blower RPM = 3550

   Blower ft^3/revolution = .028 ft^3

   Standard CFM = gas volume converted to conditions of 68°F and 14.7 psia.

2. Measure and record the following information:

   Pcontainment--Average of 3LMS*P934, 935, 936, and 937 (psia)

   Pout--From 3HCS*PI1A or B (psia)

   Tc--Containment temperature (°F)

   Pin--Measure with a new inlet gauge or calculate from Equation 3a below (psia)

   scfm measured--See Procedure/Form 3613A.3-1

   ΔP_f--From Table 2 (psi)

   A--As found Slip Constant

   Accuracy--Instrument accuracy range from Table 1.

3. Calculate as found slip constant (A)

   a. Pin = Pcontainment - ΔP_f

   b.

$$A = \frac{3550 - \left(\left[\dfrac{scfm_{measured} - Accuracy}{0.028 * 0.95}\right] * \left[\dfrac{14.7}{Pin} * \dfrac{Tc + 460}{528}\right]\right)}{\left(\left[\dfrac{Pout}{Pin} * 14.7\right] - 14.7\right)^{\frac{1}{2}} * \left(\dfrac{14.7}{Pin} * \dfrac{Tc + 460}{528}\right)^{\frac{1}{2}}}$$

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

Background

The OPERABILITY of the Supplementary Leak Collection and Release System (SLCRS) ensures that radioactive materials that leak from the primary containment into the Secondary Containment following a Design Basis Accident (DBA) are filtered out and adsorbed prior to any release to the environment.

SLCRS Ductwork Integrity:

The Supplementary Leak Collection and Release System (SLCRS) remains OPERABLE with the following bolting configuration:

a. For 3HVR*DMPF44:

- Eight bolts properly installed on the ductwork access panels.
- At least one bolt must be installed in each corner area.
- The remaining bolts should be installed in the center area of each side.

b. For 3HVR*DMPF29:

- 12 bolts properly installed on the ductwork access panel.
- At least one bolt must be installed in each corner area.
- The remaining bolts should be approximately equally spaced along each side with two bolts per side.

With the above bolting specified for 3HVR*DMPF44 and 3HVR*DMPF29, reference (1) concluded the following:

- Any leakage around the plates is minimal and causes negligible effect on the performance of the SLCRS system.
- Assures the gasket will not be extruded from between the plate and duct flange when the SLCRS fans are started.
- The remaining bolts may be installed with the fans running.
- Provides adequate structural integrity in the seismic event based on engineering analysis.

Applicable Safety Analyses

The SLCRS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis assumes that only one train of the SLCRS and one train of the auxiliary building filter system is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction of the airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from the containment is determined for a LOCA.

The SLCRS is not normally in operation. The SLCRS starts on a SIS signal. The modeled SLCRS actuation in the safety analysis (the Millstone 3

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

FSAR Chapter 15, Section 15.6) is based upon a worst-case response time following an SI initiated at the limiting setpoint. One train of the SLCRS in conjunction with the Auxiliary Building Filter (ABF) system is capable of drawing a negative pressure (0.4 inches water gauge at the auxiliary building 24'6" elevation) within 120 seconds after a LOCA. This time includes diesel generator startup and sequencing time, system startup time, and time for the system to attain the required negative pressure after starting.

LCO

In the event of a DBA, one SLCRS is required to provide the minimum postulated iodine removal assumed in the safety analysis. Two trains of the SLCRS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single-active failure. The SLCRS works in conjunction with the ABF system. Inoperability of one train of the ABF system also results in inoperability of the corresponding train of the SLCRS. Therefore, whenever LCO 3.7.9 is entered due to the ABF train A (B) being inoperable, LCO 3.6.6.1 must be entered due to the SLCRS train A (B) being inoperable.

When a SLCRS LCO is not met, it is not necessary to declare the secondary containment inoperable. However, in this event, it is necessary to determine that a loss of safety function does not exist. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed.

Applicability

In MODES 1, 2, 3, and 4, a DBA could lead to a fission product release to containment that leaks to the secondary containment. The large break LOCA, on which this system's design is based, is a full-power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and reactor coolant system pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the SLCRS is not required to be OPERABLE.

ACTIONS

With one SLCRS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The operable train is capable of providing 100 percent of the iodine removal needs for a DBA. The 7-day Completion Time is based on consideration of such factors as the reliability of the OPERABLE redundant SLCRS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs. If the SLCRS cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and MODE 5 within the following 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full-power conditions in an orderly manner and without challenging plant systems.

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

Surveillance Requirements

a

Cumulative operation of the SLCRS with heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The 31-day frequency was developed in consideration of the known reliability of fan motors and controls. This test is performed on a STAGGERED TEST BASIS once per 31-days.

b, c, e, and f

These surveillances verify that the required SLCRS filter testing is performed in accordance with Regulatory Guide 1.52, Revision 2. ANSI N510-1980 shall be used in place of ANSI N510-1975 referenced in Regulatory Guide 1.52, Revision 2. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. The surveillances include testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). 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.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system."

This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits, as well as providing trend data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and the sample canisters are typically removed due to the 18 month criteria.

d

The automatic startup ensures that each SLCRS train responds properly. The REFUELING INTERVAL frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance was performed with the reactor at power. The surveillance verifies that the SLCRS starts on a SIS test signal. It also includes the automatic functions to isolate the other ventilation systems that are not part of the safety-related postaccident operating configuration and to start up and to align the ventilation systems

3/4.6.6.1 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (Continued)

that flow through the secondary containment to the accident condition.

- The main steam valve building ventilation system isolates.
- Auxiliary building ventilation (normal) system isolates.
- Charging pump/reactor plant component cooling water pump area cooling subsystem aligns and discharges to the auxiliary building filters and a filter fan starts.
- Hydrogen recombiner ventilation system aligns to the postaccident configuration.
- The engineered safety features building ventilation system aligns to the postaccident configuration.

References:

1. Engineering analysis, Memo MP3-DE-94-539, "Bolting Requirements for Access Panels on Dampers 3HVR*DMPF29 & 44, dated June 16, 1994.

AUXILIARY FEEDWATER SYSTEM (Continued)

In addition, given the worst case failure, the AFW is designed to supply sufficient makeup water to replace SG inventory loss as the RCS is cooled to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

Surveillance Requirement 4.7.1.2.1 verifies that each AFW pump's total head at a recirculation flow test point is greater than or equal to the required total head. This surveillance ensures that the AFW pump performance has not degraded during the operating cycle. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed with recirculation flow. This test confirms one point on the pump curve and is indicative of overall performance. This test confirms component OPERABILITY is used to trend performance and to detect incipient failures by indicating abnormal performance. The total head specified in Surveillance Requirement 4.7.1.2.1 does not include a margin for test measurement uncertainty. This consideration shall be addressed at the implementing procedure level.

Motor driven auxiliary feedwater pumps and associated flow paths are OPERABLE in the following alignment during normal operation below 10% RATED THERMAL POWER.

- Motor operated isolation valves (3FWA*MOV35A/B/C/D) are open in MODE 1, 2 and 3,
- Control valves (3FWA*HV31A/B/C/D) may be throttled or closed during alignment, operation and restoration of the associated motor driven AFW pump for steam generator inventory control.

The motor operated isolation valves must remain fully open due to single failure criteria (the valves and associated pump are powered from the opposite electrical trains).

The Turbine Driven Auxiliary Feedwater (TDAFW) pump and associated flow paths are OPERABLE with all control and isolation valves fully open in MODE 1, 2 and 3. Due to High Energy Line Break analysis, the TDAFW pump cannot be used for steam generator inventory control during normal operation below 10% RATED THERMAL POWER.

3/4.7.1.3 DEMINERALIZED WATER STORAGE TANK

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 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.

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 a small fraction of 10 CFR Part 100 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)

SURVEILLANCE REQUIREMENTS (Continued)

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 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM

BACKGROUND

The control room envelope pressurization system provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity.

The control room envelope pressurization system consists of two banks of air bottles with its associated piping, instrumentation, and controls. Each bank is capable of providing the control room area with one-hour of air following any event with the potential for radioactive releases.

Control Room Envelope OPERABILITY is satisfied while:

- Door 352 (C-49-1) is closed (East door)
- Door 351 (C-47-1) is closed, but C-47-1A, ATD/Missile Shield, is not closed (West doors)

Normal Operation

During normal operations, the control room envelope pressurization system is required to be on standby.

Post Accident Operation

The control room envelope pressurization system is required to operate during post-accident operations to ensure the control room will remain habitable during and following accident conditions.

The sequence of events which 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 is described in Bases Section 3/4.7.7.

3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM

The OPERABILITY of the Auxiliary Building Filter System ensures that radioactive materials leaking from the equipment within the charging pump, component cooling water pump and heat exchanger areas following a LOCA are filtered prior to reaching the environment. The charging pump/reactor plant component cooling water pump ventilation system must be operational to ensure operability of the auxiliary building filter system and the supplementary leak collection and release system. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. 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.

LCO 3.7.9 Action Statement:

With one Auxiliary Building Filter System inoperable, restoration to OPERABLE status within 7 days is required.

The 7 days restoration time requirement is based on the following: The risk contribution is less for an inoperable Auxiliary Building Filter System, than for the charging pump or reactor plant component cooling water (RPCCW) systems, which have a 72 hour restoration time requirement. The Auxiliary Building Filter System is not a direct support system for the charging pumps or RPCCW pumps. Because the pump area is a common area, and as long as the other train of the Auxiliary Building Filter System remains OPERABLE, the 7 day restoration time limit is acceptable based on the low probability of a DBA occurring during the time period and the ability of the remaining train to provide the required capability. A concurrent failure of both trains would require entry into LCO 3.0.3 due to the loss of functional capability. The Auxiliary Building Filter System does support the Supplementary Leak Collection and Release System (SLCRS) and the LCO Action statement time of 7 days is consistent with that specified for SLCRS (See LCO 3.6.6.1).

Surveillance Requirement 4.7.9.c

Surveillance requirement 4.7.9.c requires that after 720 hours of operation a charcoal sample must be taken and the sample must be analyzed within 31 days after removal.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system." This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing

Surveillance Requirement 4.7.9.c (Continued)

trending data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and sample canisters are typically removed due to the 18 month criteria.

3/4.7.10 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. For the purpose of declaring the affected system OPERABLE with the inoperable snubber(s), an engineering evaluation may be performed, in accordance with Section 50.59 of 10 CFR Part 50.

Snubbers are classified and grouped by design and manufacturer but not by size. Snubbers of the same manufacturer but having different internal mechanisms are classified as different types. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g.,

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

LCO 3.8.1.1.a

LCO 3.8.1.1.a requires two independent offsite power sources. With both the RSST and the NSST available, either power source may supply power to the vital busses to meet the intent of Technical Specification 3.8.1.1. The FSAR, and Regulatory Guide 1.32, 1.6, and 1.93 provide the basis for requirements concerning off-site power sources. The basic requirement is to have two independent offsite power sources. The requirement to have a fast transfer is not specifically stated. An automatic fast transfer is required for plants without a generator output trip breaker, where power from the NSST is lost on a turbine trip. The surveillance requirement for transfer from the normal circuit to the alternate circuit is required for a transfer from the NSST to the RSST in the event of an electrical failure. There is no specific requirement to have an automatic transfer from the RSST to the NSST.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

Action Statement 'b'

Required Action Statement 'b' provides an allowance to avoid unnecessary testing of the other operable diesel generator. If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generator, Surveillance Requirement 4.8.1.1.2.a.5 does not have to be performed. If the cause of inoperability exists on the other OPERABLE diesel generator, the

Action Statement 'b' (Continued)

other OPERABLE diesel generator would be declared inoperable upon discovery and ACTION Statement 'f' would be entered and appropriate actions will be completed per ACTION Statement 'f'. Once the failure is repaired, the common cause failure no longer exists, and the required ACTION 'b' will be satisfied. If the cause of the initial inoperable diesel generator can not be confirmed not to exist on the remaining diesel generator, performance of Surveillance Requirement 4.8.1.1.2.a.5 (within 24 hours of entering ACTION Statement 'b') suffices to provide assurance of continued OPERABILITY of the other diesel generator. In the event the inoperable diesel generator is restored to OPERABLE status prior to determination of the cause of the inoperability of the diesel generator, the licensee will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in ACTION Statement 'b'.

According to Generic Letter 84-15, 24 hours is reasonable to confirm that the OPERABLE diesel generator is not affected by the same problem as the inoperable diesel generator.

Action Statement 'c'

Required ACTION Statement 'c' provides an allowance to avoid unnecessary testing of the other OPERABLE diesel generator. If it can be determined that the cause of the inoperable diesel generator does not exist on the operable diesel generator, Surveillance Requirement 4.8.1.1.2.a.5 does not have to be performed. If the cause of inoperability exists on the other OPERABLE diesel generator, the other OPERABLE diesel generator would be declared inoperable upon discovery and ACTION Statement 'f' would be entered and appropriate actions will be completed per ACTION Statement 'f'. Once the failure is repaired, the common cause failure no longer exists, and the required ACTION 'c' will be satisfied. If the cause of the initial inoperable diesel generator can not be confirmed not to exist on the remaining diesel generator, performance of Surveillance Requirement 4.8.1.1.2.a.5 (within 8 hours of entering ACTION Statement 'c') suffices to provide assurance of continued OPERABILITY of the other diesel generator.

In the event, the inoperable diesel generator is restored to operable status prior to determination of the cause of the inoperability of the diesel generator, the licensee will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under eight hours constraint imposed while in ACTION Statement 'c'.

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 unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

LCO 3.8.1.1 Action statement 'd.1'

Required ACTION Statement 'd.1' requires that all systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel as a source of emergency power be verified OPERABLE. The Fuel Building Exhaust Filter System is not considered to be a required system under ACTION 3.8.1.1.d based on the following:

The Fuel Building Exhaust Filter System is described in Technical Specification 3.9.12. This system is not required as long as there is no evolution involving the movement of fuel within the storage pool or crane operations with loads above the fuel pool. With one diesel inoperable in MODES 1-4 operation in the fuel building may continue as long as there is one remaining diesel OPERABLE and OPERABILITY requirements for the Fuel Building Exhaust Filter System are met. Technical Specification Bases section 3/4.9.12 states that the limitations on the Fuel Building Exhaust Filter System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA and charcoal filters prior to discharge. This establishes the fact that the fuel building filters are installed to support a specialized fuel handling accident rather than a DBA LOCA.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

Technical Specification 3.8.1.1.b.1 requires a minimum volume of 278 gallons be contained in each of the diesel generator day tanks. Technical Specification 3.8.1.2.b.1 requires a minimum volume of 278 gallons be contained in the required diesel generator day tank. This capacity ensures that a minimum usable volume of 189 gallons is available to permit operation of each of the diesel generators for approximately 27 minutes with the diesel generators loaded to the 2,000 hour rating of 5335 kW. The shutoff level for the (two) fuel oil transfer pumps is 493 gallons (413 gallons usable volume) which corresponds to approximately 60 minutes of engine operation at the 2,000 hour rating. The first pump has a make-up setpoint of 372 gallons (284 gallons usable volume) which corresponds to approximately 42 minutes of operation at the 2,000 hour rating. The 278 gallon day tank low level value corresponds to the auto make-up setpoint of the second pump and is therefore the lowest value of fuel oil with auto make-up capability. Loss of the two redundant pumps would cause day tank level to drop below the minimum value.

Technical Specification 3.8.1.1.b.2 requires a minimum volume of 32,760 gallons be contained in each of the diesel generator's fuel storage systems. Technical Specification 3.8.1.2.b.2 requires a minimum volume of 32,760 gallons be contained in the required diesel generator's fuel storage system. This capacity ensures that a minimum usable volume (29,180 gallons) is available to permit operation of each of the diesel generators for approximately three days with the diesel generators loaded to the 2,000 hour rating of 5335 kW. The ability to cross-tie the diesel generator fuel oil supply tanks ensures that one diesel generator may operate up to approximately six days. Additional fuel oil can be supplied to the site within twenty-four hours after contacting a fuel oil supplier.

Surveillance Requirements 4.8.1.1.2.a.6 (monthly) and 4.8.1.1.2.b.2 (once per 184 days) and 4.8.1.1.2.j (18 months test)

The Surveillances 4.8.1.1.2.a.6 and 4.8.1.1.2.b.2 verify that the diesel generators are capable of synchronizing with the offsite electrical system and loaded to greater than or equal to continuous rating of the machine. A minimum time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the diesel generator is connected to the offsite source. Surveillance Requirement 4.8.1.1.2.j requires demonstration once per 18 months that the diesel generator can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which are at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the diesel generator. The load band is provided to avoid routine overloading of the diesel generator. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain diesel generator operability. The load band specified accounts for instrumentation inaccuracies using plant computer and for the operational control capabilities and human factor characteristics. The note (*) acknowledges that momentary transient outside the load range shall not invalidate the test.

Surveillance Requirements 4.8.1.1.2.a.5 (Monthly), 4.8.1.1.2.b.1 (Once per 184 Days), 4.8.1.1.2.g.4.b (18 Month Test), 4.8.1.1.2.g.5 (18 Month Test) and 4.8.1.1.2.g.6.b (18 Month Test)

Several diesel generator surveillance requirements specify that the emergency diesel generators are started from a standby condition. Standby conditions for a diesel generator means that the EDG system is aligned for automatic start and loading, diesel engine coolant and lubricating oil are being circulated and temperatures are maintained within design ranges. Design ranges for standby temperatures are greater than or equal to the low temperature alarm setpoints and less than or equal to the standby "keep-warm" heater shutoff temperatures for each respective sub-system.

Surveillance Requirement 4.8.1.1.2.j (18 Month Test)

The existing "standby condition" stipulation contained in specification 4.8.1.1.2.a.5 is superseded when performing the hot restart demonstration required by 4.8.1.1.2.j.

3/4.9.1.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2600 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The 2600 ppm provides for boron concentration measurement uncertainty between the spent fuel pool and the RWST. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

MODE ZERO shall be the Operational MODE where all fuel assemblies have been removed from containment to the Spent Fuel Pool. Technical Specification Table 1.2 defines MODE 6 as "Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed." With no fuel in the vessel the definition for MODE 6 no longer applies. The transition from MODE 6 to MODE ZERO occurs when the last fuel assembly of a full core off load has been transferred to the Spent Fuel Pool and has cleared the transfer canal while in transit to a storage location. This will:

- Ensure Technical Specifications regarding sampling the transfer canal boron concentration are observed (4.9.1.1.2);
- Ensure that MODE 6 Technical Specification requirements are not relaxed prematurely during fuel movement in containment.

3/4.9.1.2 Boron Concentration in Spent Fuel Pool

During normal spent fuel pool operation, the spent fuel racks are capable of maintaining K_{eff} at less than or equal to 0.95 in an unborated water environment. This is accomplished in Region 1, 2, and 3 storage racks by the combination of geometry of the rack spacing, the use of fixed neutron absorbers in some fuel storage regions, the limits on fuel burnup, fuel enrichment and minimum fuel decay time, and the use of blocking devices in certain fuel storage locations.

The boron requirement in the spent fuel pool specified in 3.9.1.2 ensures that in the event of a fuel assembly handling accident involving either a single dropped or misplaced fuel assembly, the K_{eff} of the spent fuel storage racks will remain less than or equal to 0.95.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

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 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 safety analysis.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive iodine released from an irradiated fuel assembly and storage pool water will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Laboratory testing of methyl iodide penetration shall be performed in accordance with ASTM D3803-89 and Millstone Unit 3 specific parameters. 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. The filtration system removes radioiodine following a fuel handing or heavy load drop accident. Noble gases would not be removed by the system. Other radionuclides would be scrubbed by the storage pool water. Iodine-131 has the longest half-life: ~8 days. After 60 days decay time, there is essentially negligible iodine and filtration is unnecessary.

Surveillance Requirement 4.9.12.1.c

Surveillance requirement 4.9.12.1.c requires that after 720 hours of operation, a charcoal sample must be taken and the sample must be analyzed within 31 days after removal.

The 720 hours of operation requirement originates from Regulatory Guide 1.52, Revision 2, March 1978, Table 2, Note "c", which states that "Testing should be performed (1) initially, (2) at least once per 18 months thereafter for systems maintained in a standby status or after 720 hours of system operations, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system." This testing ensures that the charcoal adsorbency capacity has not degraded below acceptable limits as well as providing trending data. The 720 hour figure is an arbitrary number which is equivalent to a 30 day period. This criteria is directed to filter systems that are normally in operation and also provide emergency air cleaning functions in the event of a Design Basis Accident. The applicable filter units are not normally in operation and sample canisters are typically removed due to the 18 month criteria.

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 or equal to 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

3/4.9.13 SPENT FUEL POOL - REACTIVITY (continued)

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 K_{eff} 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 specified in Figure 3.9-3.

Maintaining K_{eff} 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. 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 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 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.