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

Dominion

MAY 7 2002

Docket No. 50-336
B18638

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 2
License Basis Document Change Request (LBDCR) 2-1-02
Limiting Safety System Settings and Instrumentation

Introduction

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications. DNC is proposing to change Technical Specification 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," 3.3.1.1, "Instrumentation - Reactor Protective Instrumentation," Technical Specification 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation," Technical Specification 3.3.2.2, "Instrumentation - Engineered Safety Feature Actuation System Sensor Cabinet Power Supply Drawers," Technical Specification 3.3.3.1, "Monitoring Instrumentation - Radiation Monitoring," Technical Specification 3.3.3.5, "Instrumentation - Remote Shutdown Instrumentation," Technical Specification 3.3.3.8, "Installation, Accident Monitoring," and Technical Specification 3.3.4, "Instrumentation - Containment Purge Valve Isolation Signal." The Bases for these Technical Specifications will also be modified to reflect these changes as applicable.

Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications.

Environmental Considerations

DNC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.22. DNC has determined that the proposed changes meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This

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determination is based on the fact that the changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, or that changes a surveillance requirement, and that the amendment request meets the following specific criteria.

(i) The proposed changes involve no Significant Hazards Consideration.

As demonstrated in Attachment 2, the proposed changes do not involve a Significant Hazards Consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released off site.

The proposed changes will remove the Reactor Coolant Pump Speed - Low functional unit from the Technical Specification. However, this functional unit is not credited with performing an accident mitigation function in the facility accident analysis. The proposed changes are consistent with the design basis of the plant. The proposed change will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed change will not increase the type and amounts of effluents that may be released off site.

The proposed changes to the Reactor Protection System Instrumentation Technical Specification provide for completeness and consistency within these sections of the Technical Specifications. The proposed changes will not change the design basis of the plant. The proposed changes will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not increase the type and amounts of effluents that may be released off site.

The proposed changes for the Engineered Safety Feature Actuation System technical specification provide OPERABILITY, trip setpoint, and surveillance requirements for the Automatic Actuation Logic for each functional unit, as applicable and also provides for completeness and consistency through this technical specification. The proposed changes will not change the design basis of the plant. The proposed changes will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not increase the type and amounts of effluents that may be released off site.

The proposed changes for the Radiation Monitoring Instrumentation technical specification for inclusion of a response time surveillance requirement, the

proposed changes for relocation within the applicable technical specification for the Noble Gas Effluent Monitor, and the proposed changes for the Remote Shutdown Instrumentation technical specification will not change the design basis of the plant. The proposed changes will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not increase the type and amounts of effluents that may be released off site.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the configuration of the facility. There will be no change in the level of controls or methodology used for processing radioactive effluents or the handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed change.

Conclusions

The proposed changes were evaluated and we have concluded that they are safe. The proposed changes do not involve an adverse impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2).

Site Operations Review Committee and Nuclear Safety Assessment Board

The Site Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

We request issuance of this amendment for Millstone Unit No. 2 prior to June 1, 2003, with the amendment to be implemented within 90 days of issuance.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained within 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

Sworn to and subscribed before me

this 7th day of May, 2002

Notary Public

My Commission expires

SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (4)

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

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Docket No. 50-336
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Attachment 1

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-1-02
Limiting Safety System Settings and Instrumentation
Discussion of Proposed Changes and Safety Summary

License Basis Document Change Request (LBDCR) 2-1-02
Limiting Safety System Settings and Instrumentation
Discussion of Proposed Changes and Safety Summary

Introduction

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications. DNC is proposing to change Technical Specification 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," 3.3.1.1, "Instrumentation - Reactor Protective Instrumentation," Technical Specification 3.3.2.1, "Instrumentation - Engineered Safety Feature Actuation System Instrumentation," Technical Specification 3.3.2.2, "Instrumentation - Engineered Safety Feature Actuation System Sensor Cabinet Power Supply Drawers," Technical Specification 3.3.3.1, "Monitoring Instrumentation - Radiation Monitoring," Technical Specification 3.3.3.5, "Instrumentation - Remote Shutdown Instrumentation," Technical Specification 3.3.3.8, "Installation, Accident Monitoring," and Technical Specification 3.3.4, "Instrumentation - Containment Purge Valve Isolation Signal." The Bases for these Technical Specifications will also be modified to reflect these changes as applicable. Each proposed change will be discussed.

Technical Specification Changes

1. Technical Specification 2.2.1, Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," Item 4, "Reactor Coolant Pump Speed - Low" will be deleted and replaced with the word "DELETED."

   The Reactor Coolant Pump Speed - Low functional unit is not credited with performing a safety-related function in the Millstone Unit No. 2 accident analysis. However, the Reactor Coolant Pump Speed - Low functional unit does provide a backup to the Reactor Coolant Flow - Low functional unit, thereby providing an additional level of protection beyond what is credited in the facility accident analysis. Therefore, the design features associated with the Reactor Coolant Pump Speed - Low functional unit will be retained and its functionality will be administratively controlled and verified.

2. Technical Specification 2.2.1, Table 2.2-1, Item 12, "Wide Range Logarithmic Neutron Flux Monitor - Shutdown" will be added for consistency with Technical Specification 3.3.1.1, Table 3.3-1, "Reactor Protective Instrumentation," Item 11, "Wide Range Logarithmic Neutron Flux Monitor - Shutdown." The trip setpoint and allowable values will both be marked as "Not Applicable" since the Wide Range Logarithmic Neutron Flux Monitor - Shutdown functional unit does not have a specific trip or allowable value, similar to the manual reactor trip functional unit, Table 2.2-1, Item 1. Additionally, Technical Specification 3.3.1.1, Table 4.3-1, "Reactor Protective Instrumentation Surveillance Requirements," Item 11, "Wide Range Logarithmic Neutron Flux Monitor" will be renamed to

"Wide Range Logarithmic Neutron Flux Monitor - Shutdown" consistent with the aforementioned change. These are editorial changes.

3. Technical Specification 2.2.1, Table 2.2-1, will be revised for completeness and consistency with the proposed changes for the Reactor Protection System (RPS) Logic requirements of Technical Specification 3.3.1.1 (see items 6, 7, and 11).

   The RPS Logic consists of both matrix and initiation logic and employs a scheme that provides a reactor trip when bistables in any two out of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. Bistable relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two bistable channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same function in the two channels being monitored by the logic matrix, all four matrix relays de-energize. The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized Reactor Trip Circuit Breakers (RTCB) control relays (K1, K2, K3, and K4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, opening the RTCBs if any one or more of the six logic matrices indicate a coincidence condition. Millstone Unit No. 2 Final Safety Analysis Report (FSAR) Figure 7.2-1 provides a simplified block diagram of the RPS Logic.

   Therefore, two new functional units, the RPS Logic Matrices and the RPS Logic Matrix Relays, will be added to the Millstone Unit No. 2 Technical Specifications to more accurately reflect the existing plant design for the RPS Logic, thereby focusing the facility technical specifications on those key functional units which determine RPS Logic OPERABILITY.

   The two new functional units will be added as Technical Specification 2.2.1, Table 2.2-1, Item 13, "Reactor Protection System Logic Matrices" and Technical Specification 2.2.1, Table 2.2-1, Item 14, "Reactor Protection System Logic Matrix Relays." The trip setpoint and allowable values will both be marked as "Not Applicable" since the RPS Logic Matrices and the RPS Matrix Relays functional units do not have a specific trip or allowable value, similar to the manual reactor trip functional unit, Table 2.2-1, Item 1.

4. Technical Specification 2.2.1, Table 2.2-1, Item 15, "Reactor Trip Breakers" will be added for consistency with Technical Specification 3.3.1.1, Table 4.3-1, Item 14. The trip setpoint and allowable values will both be marked as "Not Applicable" since the Reactor Trip Breakers functional units do not have a specific trip or allowable value, similar to the manual reactor trip functional unit, Table 2.2-1, Item 1. This is an editorial change.

5. Technical Specification 3.3.1.1, Table 3.3-1, "Reactor Protective Instrumentation," Item 12, "Underspeed - Reactor Coolant Pumps" will be deleted and replaced with the word "DELETED." As stated in item 1, the design features associated with the Underspeed - Reactor Coolant Pumps (also known as the Reactor Coolant Pump Speed - Low) functional unit will be retained and its functionality will be administratively controlled and verified.

6. Technical Specification 3.3.1.1, Table 3.3-1, will be revised to include a new functional unit, Item 13, the "Reactor Protection System Logic Matrices" consistent with the proposed changes for the RPS Logic discussed in Item 3. The total number of channels will be "6," the channels to trip "1," the minimum channels operable "6," the applicable modes "1, 2 and *" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as "Action 5," a new action to be added to Table 3.3-1.

   Action 5 will state, "With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1."

   The channel must be restored to OPERABLE status within 48 hours. The completion time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, shutdown requirements are provided consistent with the existing Millstone Unit No. 2 Technical Specification shutdown requirements for an inoperable RPS manual trip functional unit (see Technical Specification 3.3.1.1, Table 3.3-1, Item 1 and associated Action 1). The proposed restoration period provides for sufficient time to assess the condition of the affected RPS logic matrix and determine if the affected unit can be restored to OPERABILITY within a minimal timeframe (48 hours), thereby avoiding the risk associated with a reactor shutdown.

   These changes are consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," for the RPS Logic, although the RPS Logic has been expanded to more accurately reflect the key functional units which ensure OPERABILITY of the RPS Logic.

7. Technical Specification 3.3.1.1, Table 3.3-1, will be revised to include a new functional unit, Item 14, the "Reactor Protection System Logic Matrix Relays." The total number of channels will be "4/Matrix," the channels to trip "3/Matrix," the minimum channels operable "4/Matrix," the applicable modes "1, 2 and *" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as "Action 6," a new action to be added to Table 3.3-1.

Action 6 will state, "With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1."

These changes are consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," for the RPS Logic, although the RPS Logic has been expanded to more accurately reflect the key functional units which ensure OPERABILITY of the RPS Logic.

8. Technical Specification 3.3.1.1, Table 3.3-1 will be revised to include a new functional unit, Item 15, the "Reactor Trip Breakers." This change provides for the inclusion of OPERABILITY requirements consistent with the existing surveillance requirements for the Reactor Trip Breakers, as shown in Technical Specification 3.3.3.1, Table 4.3-1, Item 14. The total number of channels will be "4," the channels to trip "3," the minimum channels operable "4," the applicable modes "1, 2 and *" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as "Action 6" above. These changes are also consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Revision 2 although the OPERABILITY requirements for this functional unit will be expressed in channels, as opposed to NUREG-0212 which identifies OPERABILITY requirements based on trains.

9. The "and *" mode applicability will be deleted from the surveillance requirements for Technical Specification 3.3.1.1, Table 4.3-1 (page 3/4 3-8) Item 11, "Wide Range Logarithmic Neutron Flux Monitor." License Amendment Number 38,[1] dated April 19, 1978, removed the Wide Range Logarithmic Neutron Flux Monitor Startup and Operating Rate of Change of Power - High functional unit from Technical Specifications. This functional unit was applicable in Modes "1, 2 and *." The Wide Range Logarithmic Neutron Flux Monitor - Shutdown functional unit, which is applicable in Modes 3, 4, and 5, was retained.

   An error was introduced in retyped page 3/4 3-8 when the "Wide Range Logarithmic Neutron Flux Monitor, Startup and Operating - Rate of Change of Power - High" functional unit was deleted. The error was the addition of the "and *" mode applicability to the Wide Range Logarithmic Neutron Flux Monitor Shutdown functional unit surveillance requirement, (Table 4.3-1, Item 11). The current surveillance requirement for the Wide Range Logarithmic Neutron Flux Monitor Shutdown functional unit (Table 4.3-1, Item 11) shows the mode applicability for the Wide Range Logarithmic Neutron Flux Monitor Shutdown functional unit surveillance requirement as Modes "3, 4, 5 and *" when the correct mode applicability for performance of the surveillance requirements should read Modes "3, 4, 5" as shown in Table 3.3-1, Item 11. This deficiency

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[1] Robert W. Reid, U.S. Nuclear Regulatory Commission to Donald C. Switzer, Northeast Nuclear Energy Company, Amendment No. 38 to Docket No. 50-336, dated April 19, 1978.

was reported in Millstone Unit No. 2 Licensee Event Report 2000-004-00,[2] dated March 15, 2000. This is an editorial change.

10. Technical Specification 3.3.1.1, Table 4.3-1, Item 12, "Underspeed - Reactor Coolant Pumps" will be deleted and replaced with the word "DELETED." As stated in item 1, the design features associated with the Underspeed - Reactor Coolant Pumps (or Reactor Coolant Pump Speed - Low) functional unit will be retained and its functionality will be administratively controlled and verified.

11. Technical Specification 3.3.1.1, Table 4.3-1, Item 13, the "Reactor Protection System Logic" will be revised to provide additional detail to more accurately reflect the key functional units which ensure OPERABILITY of the RPS Logic as discussed in Items 6 and 7.

    Table 4.3-1, Item 13, the "Reactor Protection System Logic" will be replaced with two new functional units, Table 4.3-1, Item 13, the "Reactor Protection System Logic Matrices" and Table 4.3-1, Item 14, the "Reactor Protection System Logic Matrix Relays." For both functional units, the channel check will be marked as "N.A.," the channel calibration as "N.A.," the channel functional test as "M and S/U(1)" and the applicable modes as "1, 2 and *," consistent with the existing surveillance requirements for the Reactor Protection System Logic. However, the mode applicability has been expanded to require the applicable surveillances to be performed whenever the RTCBs are closed "and *," consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Revision 2.

    These changes are consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Revision 2 for the RPS Logic, although the RPS Logic has been expanded to more accurately reflect the key functional units which ensure OPERABILITY of the RPS Logic.

12. Table 4.3-1, Item 14, "Reactor Trip Breakers," will be renumbered as Item 15, consistent with the changes discussed in Item 11. This is an editorial change.

13. Technical Specification 3.3.2.1, Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," will be revised to include OPERABILITY requirements for the Automatic Actuation Logic (AAL) associated with the Safety Injection, Containment Spray, Containment Isolation, Main Steam Isolation, Enclosure Building Filtration, Containment Sump Recirculation, and Auxiliary Feedwater functional units as identified in Table 4.3-2, "Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements." These changes are consistent with NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Revision 2.

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[2] C. J. Schwarz to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Licensee Event Report 2000-004-00, Failure to Perform Channel Check for Wide Range Logarithmic Neutron Flux Monitor," dated March 15, 2000.

Additionally, Technical Specification 3.3.2.1, Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," will be revised to include OPERABILITY requirements for the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One and Level Two functional units AAL, which is not explicitly addressed by NUREG-0212, Revision 2.

a. For the Safety Injection functional unit, Table 3.3-3 Item 1, the AAL will be added as Item 1.d, with the total number of channels as "2," the channels to trip as "1," the minimum channels operable as "2," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5," a new action to be added to Table 3.3-3.

   Action 5 will state, "With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE."

   Action 5 provides for up to 48 hours for restoration of an inoperable channel of AAL for the applicable Engineered Safety Feature Actuation System (ESFAS) functional unit(s). The 48 hour completion time for restoration of an inoperable channel of AAL is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel of AAL. The 48 hour completion time is also consistent with the existing technical specification requirements for restoration of an inoperable manual trip channel for the associated ESFAS functional unit. Additionally, the proposed action statement is consistent with NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants."

b. For the Containment Spray functional unit, Table 3.3-3 Item 2, the AAL will be added as Item 2.c, with the total number of channels as "2," the channels to trip as "1," the minimum channels operable as "2," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

c. For the Containment Isolation functional unit, Table 3.3-3 Item 3, the AAL will be added as Item 3.e, with the total number of channels as "2," the channels to trip as "1," the minimum channels operable as "2," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

d. For the Main Steam Isolation functional unit, Table 3.3-3 Item 4, the AAL will be added as Item 4.d, with the total number of channels as "2/steam generator," the channels to trip as "1/steam generator," the minimum

channels operable as "2/steam generator," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

e. For the Enclosure Building Filtration functional unit, Table 3.3-3 Item 5, the AAL will be added as Item 5.e, with the total number of channels as "2," the channels to trip as "1," the minimum channels operable as "2," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

f. For the Containment Sump Recirculation functional unit, Table 3.3-3 Item 6, the AAL will be added as Item 6.c, with the total number of channels as "2," the channels to trip as "1," the minimum channels operable as "2," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

g. For the Loss of Power functional unit, Table 3.3-3 Item 8, the existing 4.16 kv Emergency Bus Undervoltage - Level One OPERABILITY requirements (item a), will be renumbered as item "a.1" to support the inclusion of the AAL. The AAL will be added as Item "a.2," with the total number of channels as "4/bus," the channels to trip as "2/bus," the minimum channels operable as "3/bus," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "2," an existing action within Millstone Unit No. 2 Technical Specification 3.3.1.1, Table 3.3-3 which is the action currently specified for the existing OPERABILITY requirements for the 4.16 kv Emergency Bus Undervoltage - Level One functional unit.

   For the 4.16 kv Emergency Bus Undervoltage - Level Two functional unit, the existing OPERABILITY requirements (item b), will be renumbered as item "b.1," to support inclusion of the AAL. The AAL will be added as Item "b.2," with the total number of channels as "4/bus," the channels to trip as "2/bus," the minimum channels operable as "3/bus," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "2," which is the action currently specified for the existing OPERABILITY requirements for the 4.16 kv Emergency Bus Undervoltage - Level Two functional unit.

h. For the Auxiliary Feedwater functional unit, Table 3.3-3 Item 9, the AAL will be added as Item 9.c, with the total number of channels as "2/steam generator," the channels to trip as "1/steam generator," the minimum channels operable as "2/steam generator," the applicable modes as "1, 2, 3" and the appropriate action for failure to comply with the LCO OPERABILITY requirements as Action "5."

14. The action statement, action 1, for Technical Specification 3.3.2.1, Table 3.3-3, Item 9, "Auxiliary Feedwater" requires the unit to be brought to COLD SHUTDOWN if the manual actuation function (Item 9.a) is inoperable and cannot be restored to service within the identified timeframe. However, the Auxiliary

Feedwater manual actuation function is only required to be OPERABLE in Modes 1, 2, and 3. It is not necessary to bring the unit to COLD SHUTDOWN (Mode 5) if an Auxiliary Feedwater manual actuation functional unit is inoperable as this technical specification does not apply to the Auxiliary Feedwater manual actuation function in Mode 4. Therefore, a new action, Action 6, will be created which is identical to the existing action (Action 1) for the Auxiliary Feedwater manual actuation functional unit except that the unit will only be required to be brought to HOT SHUTDOWN.

Action 6 will state, "With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours."

15. Technical Specification 3.3.2.1, Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," will be revised for completeness to address the AAL trip values (or identify the lack thereof) associated with the Safety Injection, Containment Spray, Containment Isolation, Main Steam Isolation, Enclosure Building Filtration, Containment Sump Recirculation, and Auxiliary Feedwater functional units as identified in Table 4.3-2, "Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements."

    Additionally, Technical Specification 3.3.2.1, Table 3.3-4, will be revised to include trip values (or identify the lack thereof) associated with the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One and Level Two functional units AAL, which is not explicitly addressed by NUREG-0212, Revision 2.

    a. For the Safety Injection functional unit, Table 3.3-4 Item 1, the AAL will be added as Item 1.d, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual reactor trip functional unit, Table 3.3-4, Item 1.a.

    b. For the Containment Spray functional unit, Table 3.3-4 Item 2, the AAL will be added as Item 2.c, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual trip buttons, Table 3.3-4, Item 2.a.

    c. For the Containment Isolation functional unit, Table 3.3-4 Item 3, the AAL will be added as Item 3.e, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual trip buttons, Table 3.3-4, Item 3.a.

    d. For the Main Steam Isolation functional unit, Table 3.3-4 Item 4, the AAL will be added as Item 4.d, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the trip buttons, Table 3.3-4, Item 4.a.

e. For the Enclosure Building Filtration functional unit, Table 3.3-4 Item 5, the AAL will be added as Item 5.e, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual trip buttons, Table 3.3-4, Item 5.a.

f. For the Containment Sump Recirculation functional unit, Table 3.3-4 Item 6, the AAL will be added as Item 6.c, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual trip buttons, Table 3.3-4, Item 6.a.

g. For the Loss of Power functional unit, Table 3.3-4 Item 8, the existing OPERABILITY requirements for the 4.16 kv Emergency Bus Undervoltage - Level One and 4.16 kv Emergency Bus Undervoltage - Level Two functional units will be renumbered as items a.1 and b.1, respectively, to support inclusion of the AAL. The AAL for the level one undervoltage will be added as Item a.2, with the trip setpoint and allowable values both "Not Applicable." The AAL for the level two undervoltage will be added as Item b.2, with the trip setpoint and allowable values both "Not Applicable," since the AAL for both undervoltage functional units does not have a specific trip or allowable value.

h. For the Auxiliary Feedwater functional unit, Table 3.3-4 Item 9, the AAL will be added as Item 9.c, with the trip setpoint and allowable values both "Not Applicable," since the AAL for this functional unit does not have a specific trip or allowable value, similar to the manual trip buttons, Table 3.3-4, Item 9.a.

16. Technical Specification 3.3.2.1, Table 4.3-2, "Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements," will be revised to address surveillance requirements associated with the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One and Level Two functional units AAL, which is not explicitly addressed by NUREG-0212, Revision 2. These proposed changes are consistent with current plant practices for verifying OPERABILITY of the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One and Level Two AAL, as well as consistent with the existing surveillance requirements of Technical Specification 3.3.2.1 for the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One and Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level Two functional units.

For the Loss of Power functional unit, Table 4.3-2 Item 8, the existing surveillance requirements for the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level One functional unit will be renumbered as items a.1 to support the inclusion of the AAL. The AAL for the level one undervoltage will be added as Item a.2, with the channel check as "N.A.," the channel calibration as "N.A.," the channel functional test as "M" and the applicable modes as "1, 2, 3."

For the Loss of Power functional unit, Table 4.3-2 Item 8, the existing surveillance requirements for the Loss of Power - 4.16 kv Emergency Bus Undervoltage - Level Two functional unit will be renumbered as items b.1 to support the inclusion of the AAL. The AAL for the level two undervoltage will be added as Item b.2, with the channel check as "N.A.," the channel calibration as "N.A.," the channel functional test as "M" and the applicable modes as "1, 2, 3."

17. Technical Specification 3.3.3.1 will be revised to include a new surveillance requirement, Surveillance Requirement 4.3.3.1.3. This surveillance requirement will verify that the response time for the control room isolation function, Table 3.3-6, Item 1.b, is consistent with Millstone Unit No. 2 accident analysis assumptions for the Steam Generator Tube Rupture Event and for a Millstone Unit No. 3 Loss of Coolant Accident (see Millstone Unit No. 2 FSAR Sections 14.6.3 and 14.6.5). Each of these accidents assumes that the control room isolation functions occurs upon receipt of a high radiation signal from radiation monitors (one per train) in the operating Control Room Ventilation System inlet duct.

    The new surveillance requirement will state " Verify the response time of one control room isolation channel at least once per 18 months. Each test shall include at least one control room isolation channel such that both channels are tested at least once every 36 months."

18. Technical Specification 3.3.3.1, Table 3.3-6, Item 1.d, the "Noble Gas Effluent Monitor (high range) (Unit 2 stack)" will be relocated within Table 3.3-6 as Item 2.c. The "Noble Gas Effluent Monitor (high range) (Unit 2 stack)" is a process monitor, as opposed to an area monitor, and therefore, belongs within item 2. This is an editorial change.

19. Technical Specification 3.3.3.1, Table 4.3-3, Item 1.d, the "Noble Gas Effluent Monitor (high range) (Unit 2 stack)" will be relocated within Table 4.3-3 as Item 2.c. The "Noble Gas Effluent Monitor (high range) (Unit 2 stack)" is a process monitor, as opposed to an area monitor, and therefore, belongs within item 2. This is an editorial change.

20. The action statement for Technical Specification 3.3.3.5 will be revised by including the phrase "less than the minimum required channels of" at the beginning of the action. This change revises the action statement consistent with standard practices for Technical Specification format such that the action statement will not be entered unless the minimum channels required OPERABLE, as defined in Table 3.3-9 of this specification, are not maintained.

21. Action a. of Technical Specification 3.3.3.5 will be removed and Action b. and Action c. will be renumbered to reflect the deletion of Action a. Surveillance Requirement 4.3.3.5 provides the necessary activities to ensure that OPERABILITY of redundant channels is maintained. Immediate demonstration of the OPERABILITY of a redundant channel does not provide any significant benefit in ensuring OPERABILITY of the remote shutdown instrumentation.

22. Action b. of Technical Specification 3.3.3.5 will be revised by replacing "inoperable channel" with "minimum required channels." This change will focus the action statement on ensuring that the minimum number of channels required by this Technical Specification are restored to service within the required timeframe.

23. Action b. of Technical Specification 3.3.3.5 will also be revised by extending the restoration period for an inoperable channel from 7 days to 31 days. The existing surveillance requirements for this specification, as defined in Table 4.3-6, require channel checks every 31 days and channel calibrations on an 18 month interval, as applicable. Industry experience has shown that 31 days is a reasonable and acceptable duration for verifying the OPERABILITY of Remote Shutdown Instrumentation, as shown in Technical Specification 3.3.12, "Remote Shutdown System (Analog)" of NUREG-1432, Revision 2, "Standard Technical Specifications, Combustion Engineering Plants."

    Additionally, consistency between the surveillance requirement and the action statement time is desirable to eliminate the possibility of an inadvertent Technical Specification violation. For example, if a channel of Remote Shutdown Monitoring Instrumentation were to become inoperable 7 days after the applicable surveillance was performed due to a component failure, the equipment failure might not be discovered until the next surveillance interval, resulting in a period of inoperability of approximately 24 days for that channel of Remote Shutdown Monitoring Instrumentation. Since the action statement requires action to be taken within 7 days of the component failure, a violation of the Technical Specification would occur before operator action could be taken to restore operability of the component or comply with the action statement requirements. An example of this type of Technical Specification violation occurred in August 2001 for Millstone Unit No. 3 as reported in Licensee Event Report 2001-004-00.[3]

24. For Technical Specification 3.3, "Instrumentation," blank pages 3-6, 3-21, 3-22, 3-22a, 3-30, 3-31, 3-32, 3-33, 3-34, 3-35, 3-36, 3-37, 3-38, 3-42, 3-43, 3-44, 3-45, 3-45a, 3-50, 3-51, 3-52, 3-53, 3-54, 3-55, 3-56, 3-57, 3-58, 3-59, 3-60 will be deleted.

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[3] C. J. Schwarz to U.S. Nuclear Regulatory Commission, "Millstone Power Station, Unit No. 3, Licensee Event Report 2001-004-00, "Failure of Neutron Accident Monitoring Instrumentation," dated October 11, 2001.

The follow pages will be renumbered consistent with the deletion of these blank pages:

Old Page Number	New Page Number	Old Page Number	New Page Number
3-7	3-6	3-25	3-22
3-8	3-7	3-25a	3-23
3-9	3-8	3-26	3-24
3-10	3-9	3-27	3-25
3-11	3-10	3-28	3-26
3-12	3-11	3-29	3-27
3-13	3-12	3-39	3-28
3-14	3-13	3-40	3-29
3-15	3-14	3-41	3-30
3-16	3-15	3-46	3-31
3-17	3-16	3-47	3-32
3-18	3-17	3-48	3-33
3-19	3-18	3-48a	3-34
3-20	3-19	3-49	3-35
3-23	3-20	3-61	3-36
3-24	3-21	3-62	3-37

The Index will also be revised consistent with these changes. These changes are editorial changes.

Technical Specification Bases

The Technical Specification Bases for Section 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," will be revised to reflect the deletion of the Underspeed - Reactor Coolant Pumps trip function and the inclusion of the four new functional units. The Technical Specification Bases for Section 3.3.1.1 will be revised to include a discussion of the RTCBs. The Technical Specification Bases for Section 3.3.2.1 will be revised to include a discussion of the ESFAS AAL. The Technical Specification Bases for Section 3.3.3.1 will be revised to include a discussion of the response time surveillance requirement for the control room isolation function.

Safety Summary

The proposed changes to Millstone Unit No. 2 Technical Specifications 2.2.1, 3.3.1.1, 3.3.2.1, 3.3.3.1, and 3.3.3.5 do not pose a condition adverse to safety and do not create any adverse safety consequences. The rationale for this conclusion is provided below.

The proposed changes to Technical Specifications 2.2.1 and 3.3.1.1 for the Reactor Coolant Pump Speed - Low (or Underspeed - Reactor Coolant Pumps) functional unit reflect the assumptions of the Millstone Unit No. 2 accident analysis in that the Reactor Coolant Pump Speed - Low functional unit is not credited with any safety-related functions. The Reactor Coolant Pump Speed - Low functional unit provides a backup to the Reactor Coolant Flow - Low functional unit, thereby providing an additional level of protection beyond what is credited in the Millstone Unit No. 2 accident analysis. The

accident analysis. The Reactor Coolant Pump Speed - Low functional unit trip was originally credited in the Millstone Unit No. 2 accident analysis for the four reactor coolant pump loss of electrical power event. Upon the change of fuel vendors for Millstone Unit No. 2 to Advanced Nuclear Fuels (now Siemens Power Generation) in Cycle 10, the Reactor Coolant Flow - Low functional unit was credited in the Millstone Unit No. 2 accident analysis for mitigation of the reactor coolant pump loss of electrical power event. Removal of the Reactor Coolant Pump Speed - Low functional unit from the Millstone Unit No. 2 Technical Specifications will not have an adverse effect on plant operation, and the plant response to the design basis accidents will not change. Therefore, there will be no adverse impact on public health and safety and the proposed changes are safe.

The proposed changes to Technical Specifications 2.2.1 and 3.3.1.1 for the RPS Logic (Logic Matrices and Logic Matrix Relays) will not result in any changes as to how the plant is operated, nor will the proposed changes adversely affect operator response due to a failure in the RPS. As the Millstone Unit No. 2 Technical Specifications are currently written, no guidance is given as to the required response if an RPS Logic functional unit is determined inoperable or fails a surveillance requirement. The proposed change will provide guidance to the operator as to the minimum requirements for ensuring the RPS Logic can perform its safety-related functions, as well as the required actions if the minimum OPERABILITY requirements for the RPS Logic are not met. Therefore, the proposed changes to Technical Specifications 2.2.1 and 3.3.1.1 for the RPS Logic are safe.

The proposed change to remove the "and *" applicability from Technical Specification 3.3.1.1, Table 4.3-1, Item 11 for the Wide Range Logarithmic Neutron Flux Monitor is not a technical change. The proposed change is a correction which will result in the implementation of a change which has been previously reviewed and approved by the NRC and which reflects the current design basis of the plant. Therefore, the proposed change is safe.

The proposed changes to Technical Specifications 3.3.2.1 to add OPERABILITY requirements for the Engineered Safety Feature (ESF) Actuation Instrumentation AAL will not result in any changes as to how the plant is operated, nor will the proposed changes adversely affect operator response due to a failure in the RPS. As the Millstone Unit No. 2 Technical Specifications are currently written, no guidance is given as to the required response if the AAL for any ESF functional unit is determined inoperable or fails a surveillance requirement. The proposed change will provide guidance to the operator as to the minimum requirements for ensuring the AAL for a functional unit can perform its safety-related functions, as well as the required actions if the minimum OPERABILITY requirements for any ESF AAL functional unit are not met. Therefore, the proposed changes to Technical Specifications 3.3.2.1 for the ESF Actuation Instrumentation AAL are safe.

The proposed change to Technical Specification 3.3.2.1 for the required action if an Auxiliary Feedwater manual actuation unit is inoperable will not result in any changes as to how the plant is operated, nor will the proposed change affect operator response due to a failure of an Auxiliary Feedwater manual actuation unit. The proposed change

will still require that the unit be brought to a mode of operation where the Limiting Condition of Operation does not apply (an Auxiliary Feedwater manual actuation unit is not necessary or required OPERABLE in Mode 4). The proposed required shutdown time of 12 hours is consistent with Section 3.0 of the Technical Specifications. Therefore, the proposed change is safe.

The proposed change to Technical Specification 3.3.3.1 to create a new surveillance requirement, Surveillance Requirement 4.3.3.1.3, will verify that the response time of radiation monitoring channels for the control room isolation function, Table 3.3-6, Item 1.b, are consistent with Millstone Unit No. 2 accident analysis assumptions for the Steam Generator Tube Rupture Accident and for a Millstone Unit No. 3 Loss of Coolant Accident. Each of these accidents assumes that the control room isolation functions occurs upon receipt of a high radiation signal from radiation monitors (one per train) in the operating Control Room Ventilation System inlet duct. The response time of the control room isolation function is currently verified through administrative controls consistent with the requirements of the proposed surveillance, including surveillance frequency. This change will not result in changes as to how the plant is operated, nor as to how the control room isolation function is tested. Therefore, the proposed change is safe.

The proposed change to Technical Specification 3.3.3.1 to relocate the Noble Gas Effluent Monitor within the existing Technical Specification to reflect its current (and original) plant design function is not a technical change. The proposed change will not create, change, or delete any actions related to compliance with the associated specification. Therefore, the proposed change is safe.

The proposed changes to Millstone Unit No. 2 Technical Specifications 3.3.3.5, which modify the action statement for this specification such that entry into the action statement is only required when less than the minimal channels required OPERABLE (as defined in Table 3.3-9) will not have any adverse impact on plant operation, including operator response to a failed remote shutdown instrumentation channel since the minimum channels required OPERABLE, and the actions required if less than the required number of channels of a radiation monitoring instrumentation channel is OPERABLE, will not change. As the action statement for Technical Specification 3.3.3.5 is currently written, entry into the action statement is required for the loss of certain instrumentation channels even if the minimum channels required OPERABLE by this specification (as defined in Table 3.3-9) are still satisfied. The proposed changes will only eliminate the requirement for entry into the action statement if a channel of remote shutdown instrumentation fails and the minimum required channels is still OPERABLE. For this condition, the existing Technical Specification action statement allows for indefinite operation of the plant with a failed channel of remote shutdown instrumentation provided a redundant channel of remote shutdown instrumentation is shown to be OPERABLE. Therefore, the proposed changes to the action statement of Technical Specification 3.3.3.5 are safe.

The proposed changes to Millstone Unit No. 2 Technical Specifications 3.3.3.5 to remove verification of OPERABILITY for a redundant channel of remote shutdown instrumentation if a channel of remote shutdown instrumentation has failed and the

minimum number of instrumentation channels are still OPERABLE will not have an adverse affect on plant operation, including operator action in response to a failed channel of remote shutdown instrumentation. Surveillance Requirement 4.3.3.5 provides the necessary activities to ensure that OPERABILITY of redundant channels is maintained. Immediate demonstration of the OPERABILITY of a redundant channel does not provide any significant benefit in ensuring OPERABILITY of the remote shutdown instrumentation. Therefore, the proposed changes are safe.

The proposed changes to Millstone Unit No. 2 Technical Specifications 3.3.3.5 for extending the restoration period of an inoperable channel of remote shutdown instrumentation from 7 days to 31 days when less than the minimum number of channels are OPERABLE will not have an adverse affect on plant operation, including operator response to a failed remote shutdown instrumentation channel since the existing surveillance requirements for this specification, as defined in Table 4.3-6, require channel checks every 31 days and channel calibrations on an 18 month interval, as applicable. Revision of Action b. to specify 31 days for restoration of an inoperable channel provides consistency with the minimum surveillance frequency currently specified within the specification for the performance of a channel check, the methodology typically used to demonstrate OPERABILITY of a redundant channel. The potential risk increase associated with the proposed extension of this surveillance frequency from 7 days to 31 days is insignificant. Therefore, the proposed changes are safe.

Conclusion

The proposed changes have no effect on how any of the associated systems or components function to mitigate the consequences of design basis accidents. In addition, the proposed changes will not result in any significant change in, or new approach to, plant operation. The proposed changes will not adversely affect public safety. Therefore, the proposed changes are safe.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-1-02
Limiting Safety System Settings and Instrumentation
Significant Hazards Consideration

License Basis Document Change Request (LBDCR) 2-1-02
Limiting Safety System Settings and Instrumentation
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to revise the Millstone Unit No. 2 Technical Specifications as described in this License Amendment Request. The proposed changes are associated with Technical Specification 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," 3.3.1.1, "Instrumentation - Reactor Protective Instrumentation," Technical Specification 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation," Technical Specification 3.3.2.2, "Instrumentation - Engineered Safety Feature Actuation System Sensor Cabinet Power Supply Drawers," Technical Specification 3.3.3.1, "Monitoring Instrumentation - Radiation Monitoring," and Technical Specification 3.3.3.5, "Instrumentation - Remote Shutdown Instrumentation." A brief summary of the proposed changes is provided below. Refer to Attachment 1 of this submittal for a detailed discussion of the proposed changes.

Technical Specification Changes

- The Reactor Coolant Pump Speed - Low functional unit, also known as the Underspeed - Reactor Coolant Pumps functional unit, which is not credited by the facility accident analysis, will be deleted.

- The mode applicability for the Wide Range Logarithmic Neutron Flux Monitor functional unit will be revised consistent with a previously approved license amendment (Millstone Unit No. 2 License Amendment Number 38, dated April 19, 1978).

- The Safety Limits And Limiting Safety System Settings Technical Specification will be revised for completeness and consistency with the Reactor Protection System Instrumentation Technical Specification to include those functional units which do not have specific trip or allowable values.

- The Reactor Protection System Instrumentation Technical Specification will be revised to include OPERABILITY requirements for the Reactor Protection System Logic Functional unit.

- The Reactor Protection System Instrumentation Technical Specification will be revised to include OPERABILITY requirements for the Reactor Trip Breakers Functional unit.

- The Engineered Safety Feature Actuation System Instrumentation Technical Specification will be revised to include operability, trip setpoint, and surveillance requirements for the Automatic Actuation Logic, as applicable, associated with the Safety Injection, Containment Spray, Containment Isolation, Main Steam Isolation, Enclosure Building Filtration, Containment Sump Recirculation, Loss of Power, and Auxiliary Feedwater functional units.

- The Engineered Safety Feature Actuation System Instrumentation Technical Specification action statement for the Auxiliary Feedwater manual actuation functional unit shall be revised such that the required actions are consistent with the applicability of the Technical Specification.

- The Engineered Safety Feature Actuation System Instrumentation table, which identifies Engineered Safety Features Trip Values, will be revised for completeness and consistency to include those functional units which do not have specific trip or allowable values.

- The Radiation Monitoring Instrumentation Technical Specification will be revised to include a new surveillance requirement which will verify that the response time for the control room isolation function is consistent with facility accident analysis assumptions.

- The Noble Gas Effluent Monitor (high range) (Unit 2 stack) functional unit will be relocated within the applicable Technical Specification as a process monitor, consistent with its current (and original) design function.

- The Remote Shutdown Instrumentation Technical Specification will be revised consistent with standard practices for Technical Specification format such that the action statement will not be entered unless the minimum channels of remote shutdown instrumentation required OPERABLE, as defined by this specification, are not maintained.

- The Remote Shutdown Instrumentation Technical Specification will be revised by extending the restoration period for an inoperable channel of remote shutdown instrumentation from 7 days to 31 days.

Basis for No Significant Hazards Consideration

In accordance with 10 CFR 50.92, DNC has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes associated with the operability requirements, surveillance requirements and allowed outage times will improve usability of the technical specifications. The proposed changes will clearly reflect the existing plant design for the Reactor Protection System, Engineered Safety Features Actuation System, and Radiation Monitoring System instrumentation. The proposed changes will also provide consistency within the individual technical specifications tables (e.g. Table 2.2-1, Table 3.3-1, and Table 4.3-1). In addition, there are no hardware changes associated with the proposed changes. Therefore, these systems will continue to perform within the bounds of the previously performed accident analyses.

The proposed changes to the operability requirements will not affect the instrumentation's ability to mitigate the design basis accidents. The proposed surveillance requirements are adequate to ensure proper operation of the associated instrumentation. The proposed allowed outage times (i.e. the required action times) are reasonable and consistent with industry guidelines to ensure the affected instrumentation will be restored in a timely manner. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report, and the consequences of these events will not be affected. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes (i.e. renumbering and reformatting) will not result in any technical changes to the current technical specification requirements. Therefore, these additional proposed changes will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The proposed changes will not reduce the margin of safety since they have no impact on any accident analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions. The effectiveness of Technical

Specifications will be maintained since the changes will not alter the operation of any component or system, nor will the proposed changes affect any safety limits or safety system settings which are credited in a facility accident analysis. Therefore, there is no reduction in a margin of safety.

Docket No. 50-336
B18638

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-1-02
Limiting Safety System Settings and Instrumentation
Marked Up Pages

License Basis Document Change Request 2-1-02
Limiting Safety System Settings and Instrumentation

List of Affected Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
Index	Index	V, Amendment No. 250
2.2.1	Limiting Safety System Settings - Reactor Trip Setpoints	2-4, Amendment No. 232 2-5, Amendment No. 226 B 2-7, Amendment No. 232 B 2-8, Amendment No. 139
3.3.1.1	Instrumentation - Reactor Protective Instrumentation	3-3, Amendment No. 52 3-5, Amendment No. 225 3-6, Amendment No.198 3-7, Amendment No. 225 3-8, Amendment No. 225 3-9, Amendment No. 225 B 3-1, Amendment No. 225 B 3-1a, Amendment No. 225
3.3.2.1	Instrumentation - Engineered Safety Feature Actuation System Instrumentation	3-10, Amendment No. 198 3-11, Amendment No. 245 3-12, Amendment No. 227 3-13, Amendment No. 179 3-14, Amendment No. 245 3-15, Amendment No. 226 3-16, Amendment No. 227 3-17, Amendment No. 245 3-18, Amendment No. 226 3-19, Amendment No. 245 3-20, Amendment No. 245 3-21, Amendment No. 198 3-22, Amendment No. 198 3-22a, Amendment No. 198 3-23, Amendment No. 179 3-24, Amendment No. 245 3-25, Amendment No. 230
3.3.2.2	Instrumentation - Engineered Safety Feature Actuation System Sensor Cabinet Power Supply Drawers	3-25a, Amendment No. 179
3.3.3.1	Monitoring Instrumentation - Radiation Monitoring	3-26, Amendment No. 245 3-27, Amendment No. 245 3-28, Amendment No. 245 3-29, Amendment No. 157 B 3-2a, Amendment 245

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
NA	NA	3-30, Amendment No. 237 3-31, Amendment No. 164 3-32, Amendment No. 237 3-33, Amendment No. 237 3-34, Amendment No. 237 3-35, Amendment No. 237 3-36, Amendment No. 237 3-37, Amendment No. 237 3-38, Amendment No. 237
3.3.3.5	Instrumentation - Remote Shutdown Instrumentation	3-39, Original Issue 3-40, Amendment No. 183 3-41, Amendment No. 225
NA	NA	3-42, Amendment No. 127 3-43, Amendment No. 191 3-44, Amendment No. 191 3-45, Amendment No. 191 3-45a, Amendment No. 191
3.3.3.8	Installation, Accident Monitoring	3-46, Amendment No. 151 3-47, Amendment No. 140 3-48, Amendment No. 120 3-48a, Amendment No. 140 3-49, Amendment No. 140
NA	NA	3-50, Amendment No. 250 3-51, Amendment No. 250 3-52, Amendment No. 250 3-53, Amendment No. 250 3-54, Amendment No. 250 3-55, Amendment No. 250 3-56, Amendment No. 250 3-57, Amendment No. 250 3-58, Amendment No. 250 3-59, Amendment No. 250 3-60, Amendment No. 250
3.3.4	Instrumentation - Containment Purge Valve Isolation Signal	3-61, Amendment No. 245 3-62, Amendment No. 245

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.2.3	TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T	3/4 2-9
3/4.2.4	AZIMUTHAL POWER TILT	3/4 2-10
3/4.2.5	Deleted	
3/4.2.6	DNB MARGIN	3/4 2-13
3/4.3 INSTRUMENTATION		
3/4.3.1	REACTOR PROTECTIVE INSTRUMENTATION	3/4 3-1
3/4.3.2	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION	3/4 3-~~10~~ 9
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3/4.4.1	COOLANT LOOPS AND COOLANT CIRCULATION	3/4 4-1
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For Information Only

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Level-High		
	Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3.	Reactor Coolant Flow - Low (1) DELETED	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.9% of reactor coolant flow with 4 pumps operating.
4.	Reactor Coolant Pump Speed - Low (1)	≥ 830 rpm	≥ 823 rpm
5.	Pressurizer Pressure - High	≤ 2397 psia	≤ 2407 psia
6.	Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
7.	Steam Generator Pressure - Low (2) (5)	≥ 691 psia	≥ 677 psia
8.	Steam Generator Water Level - Low (5)	≥ 48.5% Water Level - each steam generator	≥ 47.5% Water Level - each steam generator
9.	Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per Specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).
11. Loss of Turbine--Hydraulic Fluid (3) Pressure - Low	≥ 500 psig	≥ 500 psig

Insert A

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is ≥ 800 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurements, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

Insert A to Page 2-5

12. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	Not Applicable	Not Applicable
13. Reactor Protection System Logic Matrices	Not Applicable	Not Applicable
14. Reactor Protection System Logic Matrix Relays	Not Applicable	Not Applicable
15. Reactor Trip Breakers	Not Applicable	Not Applicable

For Information Only

June 10, 1996

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

For Information Only

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level - High	4	2(f)	3	1, 2, 3(d)	2
3. Reactor Coolant Flow - Low	4	2(a)	3	1, 2 (e)	2
4. Pressurizer Pressure - High	4	2	3	1, 2	2
5. Containment Pressure - High	4	2	3	1, 2	2
6. Steam Generator Pressure - Low	4	2(b)	3	1, 2	2
7. Steam Generator Water Level - Low	4	2	3	1, 2	2
8. Local Power Density - High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1,2(e)	2
10. Loss of Turbine - Hyraulic Fluid Pressure - Low	4	2(c)	3	1	2

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. Underspeed - Reactor Coolant Pumps	4	2(a)	3	1, 2(e)	2

Insert B

~~May 12, 1979~~

Insert B to Page 3/4 3-3

13. Reactor Protection System Logic Matrices	6	1	6	1, 2, and *	5
14. Reactor Protection System Logic Matrix Relays	4/Matrix	3/Matrix	4/Matrix	1, 2, and *	6
15. Reactor Trip Breakers	4	3	4	1, 2, and *	6

For Information Only

February 8, 1999

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is ≥ 800 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(d) Trip does not need to be operable if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) ΔT Power input to trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, within 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 3 - NOT USED

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, immediately verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and at least once per 4 hours thereafter.

Insert C

Insert C to Page 3/4 3-5

Action 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

Action 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

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TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3),Q(5)	M	1, 2, 3*
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	R	S/U(1)	N.A.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Wide Range Logarithmic Neutron Flux Monitor	S	R(5)	S/U(1)	3, 4, 5 and *
12. ~~Underspeed - Reactor Coolant Pumps~~ Deleted	S	R	M	1, 2
13. Reactor Protection System Logic Replace with Insert D	N.A.	N.A.	M and S/U(1)	1, 2
~~14.~~ 15. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

Insert D to Page 3/4 3-8

13. Reactor Protection System Logic Matrices	N.A.	N.A.	M and S/U(1)	1,2 and *
14. Reactor Protection System Logic Matrix Relays	N.A.	N.A.	M and S/U(1)	1,2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometers to make nuclear power signals agree with calorimetric calculation. During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Above 15% of RATED THERMAL POWER, adjust "ΔT Pwr Calibrate" potentiometers to null "Nuclear Pwr - ΔT Pwr". During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(5) Neutron detectors are excluded from the CHANNEL CALIBRATION.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either adjust the trip setpoint to be consistent with the value specified in the Trip Setpoint column of Table 3.3-4 within 2 hours or declare the channel inoperable and take the ACTION shown in Table 3.3-3.

b. With an engineered safety feature actuation system instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each engineered safety feature acutation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affects by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)(d)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure - High	4	2	3	1, 2, 3	2
c. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2
d. Automatic Actuation Logic	2	1	2	1,2,3	5
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-- High - High	4	2(b)	3	1, 2, 3	2
c. Automatic Actuation Logic	2	1	2	1,2,3	5
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure - High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2
e. Automatic Actuation Logic	2	1	2	1,2,3	5

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual MSI (Trip Buttons) High	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-High	4	2	3	1, 2, 3	2
c. Steam Generator Pressure - Low	4	2	3	1, 2, 3(c)	2
d. Automatic Actuation Logic	2/steam Generator	1/steam Generator	2/steam Generator	1,2,3	5
5. ENCLOSURE BUILDING FILTRATION (EBFAS)					
a. Manual EBFAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure-High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3(a)	2
e. Automatic Actuation Logic	2	1	2	1,2,3	5
6. CONTAINMENT SUMP RECIRCULATION (SRAS)					
a. Manual SRAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	4
c. Automatic Actuation Logic	2	1	2	1,2,3	5

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. DELETED					
8. LOSS OF POWER					
a.1 ~~a.~~ 4.16 kv Emergency Bus Undervoltage - level one	4/bus	2/Bus	3/bus	1, 2, 3	2
a.2 Automatic Actuation Logic	4/Bus	2/Bus	3/Bus	1,2,3	2
b.1 ~~b.~~ 4.16 kv Emergency Bus Undervoltage - level two	4/Bus	2/Bus	3/Bus	1, 2, 3	2
b.2 Automatic Actuation Logic	4/Bus	2/Bus	3/Bus	1,2,3	2

~~April 28, 2000~~

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. AUXILIARY FEEDWATER					
a. Manual	1/pump	1/pump	1/pump	1, 2, 3	16
b. Steam Generator Level - Low	4	2	3	1, 2, 3	2
c. Automatic Actuation Logic	2/Steam Generator	1/Steam Generator	2/Steam Generator	1, 2, 3	5
10. STEAM GENERATOR BLOWDOWN					
a. Steam Generator Level - Low	4	2	3	1, 2, 3	2

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed when pressurizer pressure is < 1850 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1850 psia. Amend 226

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed when steam generator pressure is < 700 psia; bypass shall be automatically removed when steam generator pressure is ≥ 700 psia. Amend 226

(d) In MODE 4 the HPSI pumps are not required to start automatically on a SIAS. Amend 227

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, with 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 3 - DELETED

ACTION 4 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1850 psia: immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing the pressurizer pressure above 1850 psia.

b. ≥ 1850 psia, operation may continue with the inoperable channel in the bypassed condition, provided the following condition is satisfied:

1. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided BOTH of the inoperable channels are placed in the bypassed condition.

Insert E

Insert E to Page 3/4 3-17

Action 5 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

Action 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 9.48 psig	≤ 10.11 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Steam Generator Pressure - Low	≥ 572 psia	≥ 558 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank -Low	46 ± 3 inches above tank bottom	46 ± 6 inches above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
7. DELETED		

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. LOSS OF POWER		
a.1 ~~a.~~ 4.16 kv Emergency Bus Undervoltage - level one	≥ 2912 volts with a 2.0 ± 0.1 second time delay	≥ 2877 volts with a 2.0 ± 0.1 second time delay
a.2 Automatic Actuation Logic	Not Applicable	Not Applicable
b.1 ~~b.~~ 4.16 kv Emergency Bus Undervoltage - level two	≥ 3700 volts with an 8.0 ± 2.0 second time delay	≥ 3663 volts with an 8.0 ± 2.0 second time delay
b.2 Automatic Actuation Logic	Not Applicable	Not Applicable
9. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%
c. Automatic Actuation Logic	Not Applicable	Not Applicable
10. STEAM GENERATOR BLOWDOWN		
a. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%

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TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N. A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure-- High - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Steam Generator Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
5. ENCLOSURE BUILDING FILTRATION (EBFAS)				
a. Manual EBFAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
7. DELETED				
8. LOSS OF POWER				
a.1 4.16 kv Emergency Bus Undervoltage - level one	S	R	M	1, 2, 3
a.2 Automatic Actuation Logic	N.A.	N.A.	M	1,2,3
b.1 4.16 kv Emergency Bus Undervoltage - level two	S	R	M	1, 2, 3
b.2 Automatic Actuation Logic	N.A.	N.A.	M	1,2,3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	R	N.A.
b. Steam Generator Level - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M	1, 2, 3
10. STEAM GENERATOR BLOWDOWN				
a. Steam Generator Level - Low	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The coincident logic circuits shall be tested automatically or manually at least once per 31 days. The automatic test feature shall be verified OPERABLE at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or other specified conditions for surveillance testing of the following:

a. Pressurizer Pressure Safety Injection Automatic Actuation Logic; and

b. Pressurizer Pressure Containment Isolation Automatic Actuation Logic; and

c. Steam Generator Pressure Main Steam Line Isolation Automatic Actuation Logic; and

d. Pressurizer Pressure Enclosure Building Filtration Automatic Actuation Logic.

Testing of the automatic actuation logic for Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, and Pressurizer Pressure Enclosure Building Filtration shall be performed within 12 hours after exceeding a pressurizer pressure of 1850 psia in MODE 3. Testing of the automatic actuation logic for Steam Generator Pressure Main Steam Line Isolation shall be performed within 12 hours after exceeding a steam generator pressure of 700 psia in MODE 3.

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INSTRUMENTATION

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SENSOR CABINET POWER SUPPLY DRAWERS

LIMITING CONDITION FOR OPERATION

3.3.2.2 The engineered safety feature actuation system Sensor Cabinets (RC02A1, RC02B2, RC02C3 & RC02D4) Power Supply Drawers shall be OPERABLE and energized from the normal power source with the backup power source available. The normal and backup power sources for each sensor cabinet is detailed in Table 3.3-5a:

CABINET	NORMAL POWER	BACKUP POWER
RC02A1	VA-10	VA-40
RC02B2	VA-20	VA-30
RC02C3	VA-30	VA-20
RC02D4	VA-40	VA-10

Table 3.3-5a

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With any of the Sensor Cabinet Power Supply Drawers inoperable, or either the normal or backup power source not available as delineated in Table 3.3-5a, restore the inoperable Sensor Cabinet Power Supply Drawer to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 The engineered safety feature actuation system Sensor Cabinet Power Supply Drawers shall be determined OPERABLE once per shift by visual inspection of the power supply drawer indicating lamps.

4.3.2.2.2 Verify the operability of the Sensor Cabinet Power Supply auctioneering circuit at least one per 18 months.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $x/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and particulate (Half Lives greater than 8 days) radioactivity shall be assumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5×10^5 cpm.

Insert F

Insert F to Page 3/4 3-26

4.3.3.1.3 Verify the response time of one control room isolation channel at least once per 18 months. Each test shall include at least one control room isolation channel such that both channels are tested at least once every 36 months.

TA 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Storage and Ventilation System Isolation	2	*	100 mR/hr	10^{-1} - 10^{+4} mR/hr	13
b. Control Room Isolation	1	ALL MODES	2 mR/hr	10^{-1} - 10^{4} mR/hr	16
c. Containment High Range	1	1,2,3,&4	100 R/hr	10^{0} - 10^{8} R/hr	17
~~d.~~ c. Noble Gas Effluent Monitor (high range) (Unit 2 stack)	1	1,2,3,&4	2×10^{-1} uci/cc	10^{-3} - 10^{5} uci/cc	17
2. PROCESS MONITORS					
a. Containment Atmosphere-Particulate	1	ALL MODES**	the value determined in accordance with specification 4.3.3.1.2	10 - 10^{+6} cpm	14
b. Containment Atmosphere-Gaseous	1	ALL MODES**	the value determined in accordance with Specification 4.3.3.1.2	10 - 10^{+6} cpm	14

* With fuel in storage building.
**These radiation monitors are not required to be operable during Type "A" Integrated Leak Rate testing.

TABLE 3.3-6 (Continued)

TABLE NOTATION

(a) DELETED

ACTION 13 - With the number of area monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of process monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement either (a) obtain and analyze grab samples of the monitored parameter at least once per 24 hours, or (b) use a Constant Air Monitor to monitor the parameter.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 17 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the discovery or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following discovery outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage Ventilation System Isolation	S	R	M	*
b. Control Room Isolation	S	R	M	ALL MODES
c. Containment High Range	S	R**	M	1, 2, 3, & 4
d. Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	S	R	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES

*With fuel in storage building
**Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

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INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than the minimum required channels of a remote shutdown monitoring channel inoperable, either:

a. Immediately demonstrate the OPERABILITY of a redundant channel, or

a. b. Restore the ~~inoperable channel~~ minimum required channels to OPERABLE status within ~~7~~ 31 days, or

b. c. Be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Wide Range Logarithmic Neutron Flux Monitor	Hot Shutdown Panel (C-21)	10^{-8}% - 100%	1
2. Reactor Trip Breaker Indication	Reactor Trip Switchgear (Q03)	OPEN-CLOSE	1/trip breaker
3. Reactor Cold Leg Temperature	Hot Shutdown Panel (C-21)	0-600°F	1
4. Pressurizer Pressure	Hot Shutdown Panel (C-21)		
a. Low Range		0-1600 psia	1
b. High Range		1500-2500 psia	1
5. Pressurizer Level	Hot Shutdown Panel (C-21)	0-100%	1
6. Steam Generator Pressure	Hot Shutdown Panel (C-21)	0-1200 psia	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel (C-21)	0-100%	1/steam generator

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wide Range Logarithmic Neutron Flux	M	R*
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Cold Leg Temperature	M	R
4. Pressurizer Pressure		
a. Low Range	M	R
b. High Range	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R

*Neutron detectors are excluded from the CHANNEL CALIBRATION.

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TALLATION

ACCIDENT MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. Actions per Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

Instrument	Total No. of Channels	Minimum Channels Operable	Action
1. Pressurizer Water Level	2	1	1
2. Auxiliary Feedwater Flow Rate	2/S.G.	1/S.G.	1
3. RCS Subcooled/Superheat Monitor	2	1	2
4. PORV Position Indicator Acoustic Monitor	1/valve	1/valve	3
5. PORV Block Valve Position Indicator	1/valve	1/valve	3
6. Safety Valve Position Indicator Acoustic Monitor	1/valve	1/valve	3
7. Containment Pressure (Wide Range)	2	1	4
8. Containment Water Level (Narrow Range)	1	1	7##
9. Containment Water Level (Wide Range)	2	1	4
10. Core Exit Thermocouples	4 CETs/core quadrant	2 CETs in any of 2 core quadrants	5
11. Main Steam Line Radiation Monitor	3	3	6
12. Reactor Vessel Coolant Level	2*	1*	8

*A channel is eight (8) sensors in a probe. A channel is operable if four (4) or more sensors, two (2) or more in the upper four and two (2) or more in the lower four, are operable.

##Refer to ACTION statement in Technical Specification 3.4.6.1.

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With the number of channels OPERABLE less than the MINIMUM CHANNELS OPERABLE, determine the subcooling margin once per 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information, and monitor discharge pipe temperature once per shift to determine valve position. This action is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.3.a or 3.4.3.b.

ACTION 4 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the total number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

ACTION 5 - With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or begin at least HOT SHUTDOWN within the next 12 hours.

ACTION 6 - With any channel of radiation monitoring instrumentation inoperable, portable hand-held radiation detection equipment will be used to assess radiation releases from the atmospheric dump valves and steam generator safeties subsequent to a steam generator tube rupture.

ACTION 7 - Restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours. (See the ACTION statement in Technical Specification 3.4.6.1.)

ACTION 8 - With the number of OPERABLE Channels one less than the MINIMUM CHANNELS OPERABLE in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

2. Restore the system to OPERABLE status at the next scheduled refueling; and

3. Initiate an alternate method of monitoring the Reactor Vessel inventory.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	M	R
2.	Auxiliary Feedwater Flow Rate	M	R
3.	Reactor Coolant System Subcooled/Superheat Monitor	M	R
4.	PORV Position Indicator (Acoustic Monitor)	M	R
5.	PORV Block Valve Position Indicator	N.A.	R
6.	Safety Valve Position Indicator (Acoustic Monitor)	M	R
7.	Containment Pressure	M	R
8.	Containment Water Level (Narrow Range)	M	R
9.	Containment Water Level (Wide Range)	M	R
10.	Core Exit Thermocouples	M	R*
11.	Main Steam Line Radiation Monitor	M	R
12.	Reactor Vessel Coolant Level	M	R*

*Electronic calibration from the ICC cabinets only.

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INSTRUMENTATION

CONTAINMENT PURGE VALVE ISOLATION SIGNAL

LIMITING CONDITION FOR OPERATION

3.3.4 One Containment Purge Valve Isolation Signal containment gaseous radiation monitor channel, one Containment Purge Valve Isolation Signal containment particulate radiation monitor channel, and one Containment Purge Valve Isolation Signal automation logic train shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS with the containment purge valves open.

During the movement of irradiated fuel assemblies inside containment with the containment purge valves open.

ACTION:

a. With no OPERABLE Containment Purge Valve Isolation Signal containment gaseous radiation monitor channel, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

b. With no OPERABLE Containment Purge Valve Isolation Signal containment particulate radiation monitor channel, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

c. With no OPERABLE Containment Purge Valve Isolation Signal automatic actuation logic train, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

SURVEILLANCE REQUIREMENTS

4.3.4.1 Perform a CHANNEL CHECK on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 12 hours.

4.3.4.2 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 31 days.

SURVEILLANCE REQUIREMENTS

4.3.4.3 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal automatic actuation logic train at least once per 31 days. This actuation logic shall include verification of the proper operation of the actuation relay.

4.3.4.4 Perform a CHANNEL CALIBRATION on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 18 months.

4.3.4.5 Verify Containment Purge Valve Isolation Signal response time at least once per 18 months. Each test shall include at least one containment gaseous and one containment particulate radiation monitor channel such that all channels are tested at least once every N times 18 months where N is the total number of containment gaseous or total number of containment particulate radiation monitor channels.

For Information Only

2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.6% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 106.6% of RATED THERMAL POWER and a minimum setpoint of 14.6% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 111.6% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow.

For Information Only

Reactor Coolant Flow-Low (Continued)

The low-flow trip setpoint and Allowable Value have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above the 95/95 limit for the DNB correlation under normal operation and expected transients.

Pressurizer Pressure-High

The pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is approximately 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpont for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The trip setting is sufficiently below the full-load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow.

For Information Only

Steam Generator Water Level - Low

The Steam Generator Water Level-Low Trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is below the 95/95 limit for the DNB correlation.

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits. A safety margin is provided which includes allowances for equipment response times, core power, RCS temperature, and pressurizer pressure measurement uncertainties, processing errors, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

Loss of Turbine

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

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The Wide Range Logarithmic Neutron Flux Monitor - Shutdown, Reactor Protection System Logic Matrices, Reactor Protection System Logic Matrix Relays, and Reactor Trip Breakers functional units are components of the Reactor Protective System for which OPERABILITY requirements are provided within the Technical Specifications (see Technical Specification 3.3.1.1, "Reactor Protective Instrumentation"). These functional units do not have specific trip setpoints or allowable values, similar to the manual reactor trip functional unit. However, these functional units are provided here for completeness and consistency with the RPS Instrumentation identified in Technical Specification 3.3.1.1.

Underspeed - Reactor Coolant Pumps

The Underspeed - Reactor Coolant Pumps trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant pump speed (with resulting decrease in flow) on all four reactor coolant pumps. The trip setpoint ensures that a reactor trip will be generated, considering instrument errors and response times, in sufficient time to allow the DNBR to be maintained above the 95/95 limit for the DNB correlation following a 4 pump loss of flow event.

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

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

Action Statement 2 of Tables 3.3-1 and 3.3-3 requires an inoperable Reactor Protection System (RPS) or Engineered Safety Feature Actuation System (ESFAS) channel to be placed in the bypassed or tripped condition within 1 hour. The inoperable channel may remain in the bypassed condition for a maximum of 48 hours. While in the bypassed condition, the affected functional unit trip coincidence will be 2 out of 3. After 48 hours, the channel must either be declared OPERABLE, or placed in the tripped condition. If the channel is placed in the tripped condition, the affected functional unit trip coincidence will become 1 out of 3. One additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

Plant operation with an inoperable pressurizer high pressure reactor protection channel in the tripped condition is restricted because of the potential inadvertent opening of both pressurizer power operated relief valves (PORVs) if a second pressurizer high pressure reactor protection channel failed while the first channel was in the tripped condition. This plant operating restriction is contained in the Technical Requirements Manual.

Insert H

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The surveillance testing verifies OPERABILITY of the RPS by overlap testing of the four interconnected modules: measurement channels, bistable trip units, RPS logic, and reactor trip circuit breakers. When testing the measurement channels or bistable trip units that provide an automatic reactor trip function, the associated RPS channel will be removed from service, declared inoperable, and Action Statement 2 of Technical Specification 3.3.1.1 entered. When testing the RPS logic (matrix testing), the individual RPS channels will not be affected. Each parameter within each RPS channel supplies three contacts to make up the 6 different logic ladders/ matrices (AB, AC, AD, BC, BD, and CD). During matrix testing, only one logic matrix is tested at a time. Since each RPS channel supplies 3 different logic

Insert H to Page B 3/4 3-1

The reactor trip switchgear consists of eight reactor trip circuit breakers, which are operated in four sets of two breakers (four channels). Each of the four trip legs consists of two reactor trip circuit breakers in series. The two reactor trip circuit breakers within a trip leg are actuated by separate initiation circuits. For example, if a breaker receives an open signal in trip leg A, an identical breaker in trip leg B will also receive an open signal. This arrangement ensures that power is interrupted to both Control Element Drive Mechanism buses, thus preventing a trip of only half of the control element assemblies (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

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

ladders, testing one ladder matrix at a time will not remove an RPS channel from the overall logic matrix. Therefore, matrix testing will not remove an RPS channel from service or make the RPS channel inoperable. It is not necessary to enter an action statement while performing matrix testing. This also applies when testing the reactor trip circuit breakers since this test will not remove an RPS channel from service or make the RPS channel inoperable.

Insert I

The provisions of Specification 4.0.4 are not applicable for the CHANNEL FUNCTIONAL TEST of the Engineered Safety Feature Actuation System automatic actuation logic associated with Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, Steam Generator Pressure Main Steam Line Isolation, and Pressurizer Pressure Enclosure Building Filtration for entry into MODE 3 or other specified conditions. After entering MODE 3, pressurizer pressure and steam generator pressure will be increased and the blocks of the ESF actuations on low pressurizer pressure and low steam generator pressure will be automatically removed. After the blocks have been removed, the CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic can be performed. The CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic must be performed within 12 hours after establishing the appropriate plant conditions, and prior to entry into MODE 2.

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

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

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

For Information Only

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

SRAS Logic Modification

Action Statement 4 of Table 3.3-3, which applies only to the SRAS logic, specifies that during surveillance testing the second inoperable channel must also be placed in the bypassed condition. For the SRAS logic, placing the second inoperable channel in the tripped condition (as in Action Statement 2) could result in the false generation of a SRAS signal due to an additional failure which causes a trip signal in either of the remaining channels at the onset of a LOCA. The false generation of the SRAS signal leads to unacceptable consequences for LOCA mitigation.

With Action Statement 4, during the two-hour period when two channels are bypassed, no additional failure can result in the false generation of the SRAS signal. However, an additional failure that prevents a trip of either of the two remaining channels may prevent the generation of a true SRAS signal while in this Action Statement. If no SRAS is generated at the appropriate time, operating proceduresinstruct the operator to ensure that the SRAS actuation occurs when the refueling water storage tank level decreases. Due to the limited period of vulnerability, and the existence of operator requirements to manually initiate an SRAS if an automatic initiation does not occur, this risk is considered acceptable.

Steam Generator Blowdown isolation

Automatic isolation of steam generator blowdown will occur on low steam generator water level. An auxiliary feedwater actuation signal will also be generated at this steam generator water level. Isolation of steam generator blowdown will conserve steam generator water inventory following a loss of main feedwater.

Sensor Cabinet Power Supply Auctioneering

The auctioneering circuit of the ESFAS sensor cabinets ensures that two sensor cabinets do not de-energize upon loss of a D.C. bus, thereby resulting in the false generation of an SRAS. Power source VA-10 provides normal power to sensor cabinet A and backup power to sensor cabinet D. VA-40 provides normal power to sensor cabinet D and backup power to cabinet A. Power sources VA-20 and VA-30 and sensor cabinets B and C are similarly arranged.

If the normal or backup power source for an ESFAS Sensor Cabinet is lost, two sensor cabinets would be supplied from the same power source, but would still be operating with no subsequent trip signals present. However, any additional failure associated with this power source would result in the loss of the two sensor cabinets, consequently generating a false SRAS. The 48-hour Action Statement ensures that the probability of a Action Statement and an additional failure of the remaining power source, while in this Action Statement is sufficiently small.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Insert J

The spent fuel storage area monitors provide a signal to direct the ventilation exhaust from the spent fuel storage area through a filter train when the dose rate exceeds the setpoint. The filter train is provided to reduce the particulate and iodine radioactivity released to the atmosphere. However, neither the analysis of a fuel handling accident or the analysis of a spent fuel cask drop accident in the spent fuel storage area credit automatic diversion of the spent fuel storage area ventilation exhaust through an enclosure building filtration train for accident mitigation.

The spent fuel storage area radiation monitors will detect an increase in radiation levels due to a lowering of spent fuel pool water level. This will provide additional indication to the plant operators of an unexpected decrease in spent fuel pool water level.

The containment airborne radiation monitors (gaseous and particulate) provide early indication of leakage from the Reactor Coolant System as specified in Technical Specification 3.4.6.1. In addition, these radiation monitors will initiate automatic closure of the containment purge valves upon detection of high airborne radioactivity levels inside containment. The requirements for the automatic closure of the containment purge valves is addressed by Technical Specification 3.3.4

The maximum allowable trip value for these monitors corresponds to calculated concentrations at the site boundary which would not exceed the concentrations listed in 10 CFR Part 20, Appendix B, Table II. Exposure for a year to the concentrations in 10 CFR Part 20, Appendix B. Table corresponds to a total body dose to an individual of 500 mrem which is well below the guidelines of 10 CFR Part 100 for an individual at any point on the exclusion area boundary for two hours.

Determination of the monitor's trip value in counts per minute, which is the actual instrument response, involves several factors including: 1) the atmospheric dispersion (x/Q), 2) isotopic composition of the sample, 3) sample flow rate, 4) sample collection efficiency, 5) counting efficiency, and 6) the background radiation level at the detector. The x/Q of 5.8×10^{-6} sec/m^3 is the highest annual average x/Q estimated for the site boundary (0.48 miles in the NE sector) for vent releases from the containment and 7.5×10^{-8} sec/m^3 is the highest annual average x/Q estimated for an off-site location (3 miles in the NNE sector) for releases from the Unit I stack. This calculation also assumes that the isotopic composition is xenon-133 for gaseous radioactivity and cesium-137 for particulate radioactivity (Half Lives greater than 8 days). The upper limit of 5×10^5 cpm is approximately 90 percent of full instrument scale.

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The analysis for a Steam Generator Tube Rupture Event and for a Millstone Unit No. 3 Loss of Coolant Accident credits the control room ventilation inlet duct radiation monitors with closure of the Unit 2 control room isolation dampers. In the event of a single failure in either channel (1 per train), the control room isolation dampers automatically close. The response time test for the control room isolation dampers includes signal generation time and damper closure. The response time for the control isolation dampers is maintained within the applicable facility surveillance procedure.

For Information Only

3/4.3.3.2 - DELETED

3/4.3.3.3 - DELETED

3/4.3.3.4 - DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

Docket No. 50-336
B18638

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-1-02
Limiting Safety System Settings and Instrumentation
Retyped Pages

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Level-High		
	Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% Above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3.	Reactor Coolant Flow - Low (1)	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.9% of reactor coolant flow with 4 pumps operating.
4.	DELETED		
5.	Pressurizer Pressure - High	≤ 2397 psia	≤ 2407 psia
6.	Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
7.	Steam Generator Pressure - Low (2) (5)	≥ 691 psia	≥ 677 psia
8.	Steam Generator Water Level - Low (5)	≥ 48.5% Water Level - each steam generator	≥ 47.5% Water Level - each steam generator
9.	Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

*Design Reactor Coolant flow with 4 pumps operating is the lesser of either:
a. The reactor coolant flow rate measured per Specification 4.2.6.1, or
b. The minimum value specified in the CORE OPERATING LIMITS REPORT.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4 (4).
11. Loss of Turbine--Hydraulic Fluid (3) Pressure - Low	≥ 500 psig	≥ 500 psig
12. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	Not Applicable	Not Applicable
13. Reactor Protection System Logic Matrices	Not Applicable	Not Applicable
14. Reactor Protection System Logic Matrix Relays	Not Applicable	Not Applicable
15. Reactor Trip Breakers	Not Applicable	Not Applicable

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed when steam generator pressure is < 800 psia and all CEAs are fully inserted; bypass shall be automatically removed when steam generator pressure is ≥ 800 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(4) Calculations of the trip setpoint includes measurements, calculational and processor uncertainties, and dynamic allowances.

(5) Each of four channels actuate on the auctioneered output of two transmitters, one from each steam generator.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. DELETED					
13. Reactor Protection System Logic Matrices	6	1	6	1, 2, and *	5
14. Reactor Protection System Logic Matrix Relays	4/Matrix	3/Matrix	4/Matrix	1, 2, and *	6
15. Reactor Trip Breakers	4	3	4	1, 2, and *	6

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 3 - NOT USED

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, immediately verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and at least once per 4 hours thereafter.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3),Q(5)	M	1, 2, 3*
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	R	S/U(1)	N.A.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	S	R(5)	S/U(1)	3, 4, 5
12. DELETED				
13. Reactor Protection System Logic Matrices	N.A.	N.A.	M and S/U (1)	1, 2 and *
14. Reactor Protection System Logic Matrix Relays	N.A.	N.A.	M and S/U (1)	1, 2 and *
15. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometers to make nuclear power signals agree with calorimetric calculation. During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Above 15% of RATED THERMAL POWER, adjust "ΔT Pwr Calibrate" potentiometers to null "Nuclear Pwr - ΔT Pwr". During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(5) Neutron detectors are excluded from the CHANNEL CALIBRATION.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either adjust the trip setpoint to be consistent with the value specified in the Trip Setpoint column of Table 3.3-4 within 2 hours or declare the channel inoperable and take the ACTION shown in Table 3.3-3.

b. With an engineered safety feature actuation system instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each engineered safety feature acutation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affects by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)(d)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure - High	4	2	3	1, 2, 3	2
c. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2
d. Automatic Actuation Logic	2	1	2	1, 2, 3	5
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-- High - High	4	2(b)	3	1, 2, 3	2
c. Automatic Actuation Logic	2	1	2	1, 2, 3	5
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure - High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	2
e. Automatic Actuation Logic	2	1	2	1, 2, 3	5

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual MSI (Trip Buttons) High	2	1	2	1, 2, 3, 4	1
b. Containment Pressure-High	4	2	3	1, 2, 3	2
c. Steam Generator Pressure - Low	4	2	3	1, 2, 3(c)	2
d. Automatic Actuation Logic	2/Steam Generator	1/Steam Generator	2/Steam Generator	1, 2, 3	5
5. ENCLOSURE BUILDING FILTRATION (EBFAS)					
a. Manual EBFAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
c. Containment Pressure-High	4	2	3	1, 2, 3	2
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3(a)	2
e. Automatic Actuation Logic	2	1	2	1, 2, 3	5
6. CONTAINMENT SUMP RECIRCULATION (SRAS)					
a. Manual SRAS (Trip Buttons)	2	1	2	1, 2, 3, 4	1
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	4
c. Automatic Actuation Logic	2	1	2	1, 2, 3	5

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. DELETED					
8. LOSS OF POWER					
a.1 4.16 kv Emergency Bus Undervoltage - level one	4/bus	2/Bus	3/bus	1, 2, 3	2
a.2 Automatic Actuation Logic	4/bus	2/bus	3/bus	1, 2, 3	2
b.1 4.16 kv Emergency Bus Undervoltage - level two	4/Bus	2/Bus	3/Bus	1, 2, 3	2
b.2 Automatic Actuation Logic	4/bus	2/bus	3/bus	1, 2, 3	2

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. AUXILIARY FEEDWATER					
a. Manual	1/pump	1/pump	1/pump	1, 2, 3	6
b. Steam Generator Level - Low	4	2	3	1, 2, 3	2
c. Automatic Actuation Logic	2/Steam Generator	1/Steam Generator	2/Steam Generator	1, 2, 3	5
10. STEAM GENERATOR BLOWDOWN					
a. Steam Generator Level - Low	4	2	3	1, 2, 3	2

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed when pressurizer pressure is < 1850 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1850 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed when steam generator pressure is < 700 psia; bypass shall be automatically removed when steam generator pressure is ≥ 700 psia.

(d) In MODE 4 the HPSI pumps are not required to start automatically on a SIAS.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. The inoperable channel shall either be restored to OPERABLE status, or placed in the tripped condition, with 48 hours.

b. Within 1 hour, all functional units receiving an input from the inoperable channel are also declared inoperable, and the appropriate actions are taken for the affected functional units.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

ACTION 3 - DELETED

ACTION 4 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1850 psia: immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing the pressurizer pressure above 1850 psia.

b. ≥ 1850 psia, operation may continue with the inoperable channel in the bypassed condition, provided the following condition is satisfied:

1. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1.1 provided BOTH of the inoperable channels are placed in the bypassed condition.

ACTION 5 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 9.48 psig	≤ 10.11 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
d. Pressurizer Pressure - Low	≥ 1714 psia	≥ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 4.42 psig	≤ 5.07 psig
c. Steam Generator Pressure - Low	≥ 572 psia	≥ 558 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	$\leq$ 4.42 psig	$\leq$ 5.07 psig
d. Pressurizer Pressure - Low	$\geq$ 1714 psia	$\geq$ 1704 psia
e. Automatic Actuation Logic	Not Applicable	Not Applicable
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank -Low	46 $\pm$ 3 inches above tank bottom	46 $\pm$ 6 inches above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
7. DELETED		

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. LOSS OF POWER		
a.1 4.16 kv Emergency Bus Undervoltage - level one	≥ 2912 volts with a 2.0 ± 0.1 second time delay	≥ 2877 volts with a 2.0 ± 0.1 second time delay
a.2 Automatic Actuation Logic	Not Applicable	Not Applicable
b.1 4.16 kv Emergency Bus Undervoltage - level two	≥ 3700 volts with an 8.0 ± 2.0 second time delay	≥ 3663 volts with an 8.0 ± 2.0 second time delay
b.2 Automatic Actuation Logic	Not Applicable	Not Applicable
9. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%
c. Automatic Actuation Logic	Not Applicable	Not Applicable
10. STEAM GENERATOR BLOWDOWN		
a. Steam Generator Level - Low	≥ 26.8%	≥ 25.2%

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N. A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure-- High - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Steam Generator Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
5. ENCLOSURE BUILDING FILTRATION (EBFAS)				
a. Manual EBFAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Pressurizer Pressure - Low	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. CONTAINMENT SUMP RECIRCULATION (SRAS)				
a. Manual SRAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
7. DELETED				
8. LOSS OF POWER				
a.1 4.16 kv Emergency Bus Undervoltage - level one	S	R	M	1, 2, 3
a.2 Automatic Actuation Logic	N.A.	N.A	M	1, 2, 3
b.1 4.16 kv Emergency Bus Undervoltage - level two	S	R	M	1, 2, 3
b.2 Automatic Actuation Logic	N.A.	N.A.	M	1, 2, 3
9. AUXILIARY FEEDWATER				
a. Manual	N.A.	N.A.	R	N.A.
b. Steam Generator Level - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M	1, 2, 3
10. STEAM GENERATOR BLOWDOWN				
a. Steam Generator Level - Low	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The coincident logic circuits shall be tested automatically or manually at least once per 31 days. The automatic test feature shall be verified OPERABLE at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or other specified conditions for surveillance testing of the following:

a. Pressurizer Pressure Safety Injection Automatic Actuation Logic; and

b. Pressurizer Pressure Containment Isolation Automatic Actuation Logic; and

c. Steam Generator Pressure Main Steam Line Isolation Automatic Actuation Logic; and

d. Pressurizer Pressure Enclosure Building Filtration Automatic Actuation Logic.

Testing of the automatic actuation logic for Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, and Pressurizer Pressure Enclosure Building Filtration shall be performed within 12 hours after exceeding a pressurizer pressure of 1850 psia in MODE 3. Testing of the automatic actuation logic for Steam Generator Pressure Main Steam Line Isolation shall be performed within 12 hours after exceeding a steam generator pressure of 700 psia in MODE 3.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM SENSOR CABINET POWER SUPPLY DRAWERS

LIMITING CONDITION FOR OPERATION

3.3.2.2 The engineered safety feature actuation system Sensor Cabinets (RC02A1, RC02B2, RC02C3 & RC02D4) Power Supply Drawers shall be OPERABLE and energized from the normal power source with the backup power source available. The normal and backup power sources for each sensor cabinet is detailed in Table 3.3-5a:

CABINET	NORMAL POWER	BACKUP POWER
RC02A1	VA-10	VA-40
RC02B2	VA-20	VA-30
RC02C3	VA-30	VA-20
RC02D4	VA-40	VA-10

Table 3.3-5a

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With any of the Sensor Cabinet Power Supply Drawers inoperable, or either the normal or backup power source not available as delineated in Table 3.3-5a, restore the inoperable Sensor Cabinet Power Supply Drawer to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 The engineered safety feature actuation system Sensor Cabinet Power Supply Drawers shall be determined OPERABLE once per shift by visual inspection of the power supply drawer indicating lamps.

4.3.2.2.2 Verify the operability of the Sensor Cabinet Power Supply auctioneering circuit at least one per 18 months.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

4.3.3.1.2 The trip value shall be such that the containment purge effluent shall not result in calculated concentrations of radioactivity offsite in excess of 10 CFR Part 20, Appendix B, Table II. For the purposes of calculating this trip value, a $x/Q = 5.8 \times 10^{-6}$ sec/m^3 shall be used when the system is aligned to purge through the building vent and a $x/Q = 7.5 \times 10^{-8}$ sec/m^3 shall be used when the system is aligned to purge through the Unit 1 stack, the gaseous and particulate (Half Lives greater than 8 days) radioactivity shall be assumed to be Xe-133 and Cs-137, respectively. However, the setpoints shall be no greater than 5×10^5 cpm.

4.3.3.1.3 Verify the response time of one control room isolation channel at least once per 18 months. Each test shall include at least one control room isolation channel such that both channels are tested at least once every 36 months.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Storage and Ventilation System Isolation	2	*	100 mR/hr	10^{-1} - 10^{+4} mR/hr	13
b. Control Room Isolation	1	ALL MODES	2 mR/hr	10^{-1} - 10^{4} mR/hr	16
c. Containment High Range	1	1,2,3,&4	100 R/hr	10^{0} - 10^{8} R/hr	17
2. PROCESS MONITORS					
a. Containment Atmosphere-Particulate	1	ALL MODES**	the value determined in accordance with specification 4.3.3.1.2	10 - 10^{+6} cpm	14
b. Containment Atmosphere-Gaseous	1	ALL MODES**	the value determined in accordance with Specification 4.3.3.1.2	10 - 10^{+6} cpm	14
c. Noble Gas Effluent Monitor (high range) (Unit 2 stack)	1	1,2,3,&4	2×10^{-1} uci/cc	10^{-3} - 10^{5} uci/cc	17

* With fuel in storage building.
**These radiation monitors are not required to be operable during Type "A" Integrated Leak Rate testing.

TABLE 3.3-6 (Continued)

TABLE NOTATION

(a) DELETED

ACTION 13 - With the number of area monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of process monitors OPERABLE less than required by the MINIMUM CHANNELS OPERABLE requirement either (a) obtain and analyze grab samples of the monitored parameter at least once per 24 hours, or (b) use a Constant Air Monitor to monitor the parameter.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 17 - With the number of OPERABLE channels less than required by the MINIMUM CHANNELS OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the discovery or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following discovery outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage Ventilation System Isolation	S	R	M	*
b. Control Room Isolation	S	R	M	ALL MODES
c. Containment High Range	S	R**	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES
c. Noble Gas Effluent Monitor (high range) (Unit 2 Stack)	S	R	M	1, 2, 3, & 4

*With fuel in storage building
**Calibration of the sensor with a radioactive source need only be performed on the lowest range. Higher ranges may be calibrated electronically.

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than the minimum required channels of a remote shutdown monitoring channel inoperable, either:

a. Restore the minimum required channels to OPERABLE status within 31 days, or

b. Be in HOT SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Wide Range Logarithmic Neutron Flux Monitor	Hot Shutdown Panel (C-21)	10^{-8}% - 100%	1
2. Reactor Trip Breaker Indication	Reactor Trip Switchgear (Q03)	OPEN-CLOSE	1/trip breaker
3. Reactor Cold Leg Temperature	Hot Shutdown Panel (C-21)	0-600°F	1
4. Pressurizer Pressure	Hot Shutdown Panel (C-21)		
a. Low Range		0-1600 psia	1
b. High Range		1500-2500 psia	1
5. Pressurizer Level	Hot Shutdown Panel (C-21)	0-100%	1
6. Steam Generator Pressure	Hot Shutdown Panel (C-21)	0-1200 psia	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel (C-21)	0-100%	1/steam generator

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wide Range Logarithmic Neutron Flux	M	R*
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Cold Leg Temperature	M	R
4. Pressurizer Pressure		
a. Low Range	M	R
b. High Range	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R

*Neutron detectors are excluded from the CHANNEL CALIBRATION.

INSTALLATION

ACCIDENT MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentaiton channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. Actions per Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

Instrument	Total No. of Channels	Minimum Channels Operable	Action
1. Pressurizer Water Level	2	1	1
2. Auxiliary Feedwater Flow Rate	2/S.G.	1/S.G.	1
3. RCS Subcooled/Superheat Monitor	2	1	2
4. PORV Position Indicator Acoustic Monitor	1/valve	1/valve	3
5. PORV Block Valve Position Indicator	1/valve	1/valve	3
6. Safety Valve Position Indicator Acoustic Monitor	1/valve	1/valve	3
7. Containment Pressure (Wide Range)	2	1	4
8. Containment Water Level (Narrow Range)	1	1	7##
9. Containment Water Level (Wide Range)	2	1	4
10. Core Exit Thermocouples	4 CETs/core quadrant	2 CETs in any of 2 core quadrants	5
11. Main Steam Line Radiation Monitor	3	3	6
12. Reactor Vessel Coolant Level	2*	1*	8

* A channel is eight (8) sensors in a probe. A channel is operable if four (4) or more sensors, two (2) or more in the upper four and two (2) or more in the lower four, are operable.

##Refer to ACTION statement in Technical Specification 3.4.6.1.

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With the number of channels OPERABLE less than the MINIMUM CHANNELS OPERABLE, determine the subcooling margin once per 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information, and monitor discharge pipe temperature once per shift to determine valve position. This action is not required if the PORV block valve is closed with power removed in accordance with Specification 3.4.3.a or 3.4.3.b.

ACTION 4 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the total number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or submit a special report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction, the plans for restoring the channel(s) to OPERABLE status, and any alternate methods in affect for estimating the applicable parameter during the interim.

ACTION 5 - With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours, or begin at least HOT SHUTDOWN within the next 12 hours.

ACTION 6 - With any channel of radiation monitoring instrumentation inoperable, portable hand-held radiation detection equipment will be used to assess radiation releases from the atmospheric dump valves and steam generator safeties subsequent to a steam generator tube rupture.

ACTION 7 - Restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours. (See the ACTION statement in Technical Specification 3.4.6.1.).

ACTION 8 - With the number of OPERABLE Channels one less than the MINIMUM CHANNELS OPERABLE in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

2. Restore the system to OPERABLE status at the next scheduled refueling; and

3. Initiate an alternate method of monitoring the Reactor Vessel inventory.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Pressurizer Water Level	M	R
2.	Auxiliary Feedwater Flow Rate	M	R
3.	Reactor Coolant System Subcooled/Superheat Monitor	M	R
4.	PORV Position Indicator (Acoustic Monitor)	M	R
5.	PORV Block Valve Position Indicator	N.A.	R
6.	Safety Valve Position Indicator (Acoustic Monitor)	M	R
7.	Containment Pressure	M	R
8.	Containment Water Level (Narrow Range)	M	R
9.	Containment Water Level (Wide Range)	M	R
10.	Core Exit Thermocouples	M	R*
11.	Main Steam Line Radiation Monitor	M	R
12.	Reactor Vessel Coolant Level	M	R*

*Electronic calibration from the ICC cabinets only.

CONTAINMENT PURGE VALVE ISOLATION SIGNAL

LIMITING CONDITION FOR OPERATION

3.3.4 One Containment Purge Valve Isolation Signal containment gaseous radiation monitor channel, one Containment Purge Valve Isolation Signal containment particulate radiation monitor channel, and one Containment Purge Valve Isolation Signal automation logic train shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS with the containment purge valves open.

During the movement of irradiated fuel assemblies inside containment with the containment purge valves open.

ACTION:

a. With no OPERABLE Containment Purge Valve Isolation Signal containment gaseous radiation monitor channel, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

b. With no OPERABLE Containment Purge Valve Isolation Signal containment particulate radiation monitor channel, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

c. With no OPERABLE Containment Purge Valve Isolation Signal automatic actuation logic train, immediately suspend CORE ALTERATIONS and the movement of irradiated fuel assemblies inside containment, or immediately place and maintain the containment purge valves in the closed position. Enter applicable conditions and required ACTIONS for the affected valves of Technical Specification 3.9.4, "Containment Penetrations."

SURVEILLANCE REQUIREMENTS

4.3.4.1 Perform a CHANNEL CHECK on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 12 hours.

4.3.4.2 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 31 days.

4.3.4.3 Perform a CHANNEL FUNCTIONAL TEST on each Containment Purge Valve Isolation Signal automatic actuation logic train at least once per 31 days. This actuation logic shall include verification of the proper operation of the actuation relay.

4.3.4.4 Perform a CHANNEL CALIBRATION on each Containment Purge Valve Isolation Signal containment gaseous and particulate radiation monitor channel at least once per 18 months.

4.3.4.5 Verify Containment Purge Valve Isolation Signal response time at least once per 18 months. Each test shall include at least one containment gaseous and one containment particulate radiation monitor channel such that all channels are tested at least once every N times 18 months where N is the total number of containment gaseous or total number of containment particulate radiation monitor channels.

Thermal Margin/Low Pressure (Continued)

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits. A safety margin is provided which includes allowances for equipment response times, core power, RCS temperature, and pressurizer pressure measurement uncertainties, processing errors, and a further allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

Loss of Turbine

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

The Wide Range Logarithmic Neutron Flux Monitor - Shutdown, Reactor Protection System Logic Matrices, Reactor Protection System Logic Matrix Relays, and Reactor Trip Breakers functional units are components of the Reactor Protective System for which OPERABILITY requirements are provided within the Technical Specifications (see Technical Specification 3.3.1.1, "Reactor Protective Instrumentation"). These functional units do not have specific trip setpoints or allowable values, similar to the manual reactor trip functional unit. However, these functional units are provided here for completeness and consistency with the RPS Instrumentation identified in Technical Specification 3.3.1.1.

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

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

Action Statement 2 of Tables 3.3-1 and 3.3-3 requires an inoperable Reactor Protection System (RPS) or Engineered Safety Feature Actuation System (ESFAS) channel to be placed in the bypassed or tripped condition within 1 hour. The inoperable channel may remain in the bypassed condition for a maximum of 48 hours. While in the bypassed condition, the affected functional unit trip coincidence will be 2 out of 3. After 48 hours, the channel must either be declared OPERABLE, or placed in the tripped condition. If the channel is placed in the tripped condition, the affected functional unit trip coincidence will become 1 out of 3. One additional channel may be removed from service for up to 48 hours, provided one of the inoperable channels is placed in the tripped condition.

Plant operation with an inoperable pressurizer high pressure reactor protection channel in the tripped condition is restricted because of the potential inadvertent opening of both pressurizer power operated relief valves (PORVs) if a second pressurizer high pressure reactor protection channel failed while the first channel was in the tripped condition. This plant operating restriction is contained in the Technical Requirements Manual.

The reactor trip switchgear consists of eight reactor trip circuit breakers, which are operated in four sets of two breakers (four channels). Each of the four trip legs consists of two reactor trip circuit breakers in series. The two reactor trip circuit breakers within a trip leg are actuated by separate initiation circuits. For example, if a breaker receives an open signal in trip leg A, an identical breaker in trip leg B will also receive an open signal. This arrangement ensures that power is interrupted to both Control Element Drive Mechanism buses, thus preventing a trip of only half of the control element assemblies (a half trip). Any one inoperable breaker in a channel will make the entire channel inoperable.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The surveillance testing verifies OPERABILITY of the RPS by overlap testing of the four interconnected modules: measurement channels, bistable trip units, RPS logic, and reactor trip circuit breakers. When testing the measurement channels or bistable trip units that provide an automtic reactor trip function, the associated RPS channel will be removed from service,

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

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

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

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

The provisions of Specification 4.0.4 are not applicable for the CHANNEL FUNCTIONAL TEST of the Engineered Safety Feature Actuation System automatic actuation logic associated with Pressurizer Pressure Safety Injection, Pressurizer Pressure Containment Isolation, Steam Generator Pressure Main Steam Line Isolation, and Pressurizer Pressure Enclosure Building Filtration for entry into MODE 3 or other specified conditions. After entering MODE 3, pressurizer pressure and steam generator pressure will be increased and the blocks of the ESF actuations on low pressurizer pressure and low steam generator pressure will be automatically removed. After the blocks have been removed, the CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic can be performed. The CHANNEL FUNCTIONAL TEST of the ESF automatic actuation logic must be performed within 12 hours after establishing the appropriate plant conditions, and prior to entry into MODE 2.

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

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The analysis for a Steam Generator Tube Rupture Event and for a Millstone Unit No. 3 Loss of Coolant Accident credits the control room ventilation inlet duct radiation monitors with closure of the Unit 2 control room isolation dampers. In the event of a single failure in either channel (1 per train), the control room isolation dampers automatically close. The response time test for the control room isolation dampers includes signal generation time and damper closure. The response time for the control isolation dampers is maintained within the applicable facility surveillance procedure.

The spent fuel storage area monitors provide a signal to direct the ventilation exhaust from the spent fuel storage area through a filter train when the dose rate exceeds the setpoint. The filter train is provided to reduce the particulate and iodine radioactivity released to the atmosphere. However, neither the analysis of a fuel handling accident or the analysis of a spent fuel cask drop accident in the spent fuel storage area credit automatic diversion of the spent fuel storage area ventilation exhaust through an enclosure building filtration train for accident mitigation.

The spent fuel storage area radiation monitors will detect an increase in radiation levels due to a lowering of spent fuel pool water level. This will provide additional indication to the plant operators of an unexpected decrease in spent fuel pool water level.

The containment airborne radiation monitors (gaseous and particulate) provide early indication of leakage from the Reactor Coolant System as specified in Technical Specification 3.4.6.1. In addition, these radiation monitors will initiate automatic closure of the containment purge valves upon detection of high airborne radioactivity levels inside containment. The requirements for the automatic closure of the containment purge valves is addressed by Technical Specification 3.3.4

The maximum allowable trip value for these monitors corresponds to calculated concentrations at the site boundary which would not exceed the concentrations listed in 10 CFR Part 20, Appendix B, Table II. Exposure for a year to the concentrations in 10 CFR Part 20, Appendix B. Table corresponds to a total body dose to an individual of 500 mrem which is well below the guidelines of 10 CFR Part 100 for an individual at any point on the exclusion area boundary for two hours.

Determination of the monitor's trip value in counts per minute, which is the actual instrument response, involves several factors including: 1) the atmospheric dispersion (x/Q), 2) isotopic composition of the sample, 3) sample flow rate, 4) sample collection efficiency, 5) counting efficiency, and 6) the background radiation level at the detector. The x/Q of 5.8×10^{-6} sec/m^3 is the highest annual average x/Q estimated for the site boundary (0.48 miles in the NE sector) for vent releases from the containment and 7.5×10^{-8} sec/m^3 is the highest annual average x/Q estimated for an off-site location (3 miles in the NNE sector) for releases from the Unit I stack. This calculation also assumes that the isotopic composition is xenon-133 for gaseous radioactivity and cesium-137 for particulate radioactivity (Half Lives greater than 8 days). The upper limit of 5×10^5 cpm is approximately 90 percent of full instrument scale.