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The Northeast Utilities System

MAR 2 2001

Docket No. 50-423
B18311

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 3
Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval

Pursuant to 10 CFR 50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Millstone Unit No. 3 Technical Specifications. NNECO is proposing to change Technical Specification 3.3.2, "Instrumentation - Engineered Safety Features Actuation System Instrumentation." The Bases of the affected Technical Specification will also be modified to reflect this change.

The proposed changes will extend the required surveillance interval for Potter & Brumfield MDR Series slave relays, which are installed in the Millstone Unit No. 3 Engineered Safety Features Actuation System (ESFAS), from a quarterly surveillance interval to an 18 month frequency surveillance interval for those relays which meet the reliability assessment criteria established in Westinghouse Electric Corporation WCAP-13878-P-A[1] and WCAP-13900.[2] By letter dated May 31, 1996,[3] the Nuclear Regulatory Commission (NRC) approved WCAP-13878 and WCAP-13900 as acceptable justification for extending the surveillance interval for testing Potter & Brumfield MDR Series slave relays.

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(1) Westinghouse Topical Report WCAP-13878-P-A, "Reliability Assessment of Potter & Brumfield MDR Series Relays," Revision 2, August 2000.

(2) Westinghouse Topical Report WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," Revision 0, April 1994.

(3) NRC Letter from Bruce A. Boger to T. Green, "Review of Westinghouse Electric Corporation Technical Reports WCAP-13878 Revision 1, WCAP-14117 Revision 1, WCAP-13900 Revision 0, ESFAS Subgroup Test Interval Extension," dated May 31, 1996.

A001

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 and associated Bases. Attachment 5 provides plant specific information relating to the applicability of WCAP-13878 and WCAP-13900 to the Millstone Unit No. 3 ESFAS.

Environmental Considerations

NNECO has reviewed the proposed License Amendment Request against the criteria of 10 CFR 51.22 for environmental considerations. These changes will not significantly increase the type and amounts of effluents that may be released offsite. In addition, this amendment request will not significantly increase individual or cumulative occupational radiation exposures. Therefore, NNECO has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes were evaluated and we have concluded that they are safe. The proposed changes do not involve a significant 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).

Plant Operations Review Committee and Nuclear Safety Assessment Board

The Plant 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. 3 prior to December 1, 2001 with the amendment to be implemented within 60 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 in this letter.

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

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Sworn to and subscribed before me

this 2ND day of March, 2001

Notary Public

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

Attachments (5)

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

Director
Bureau of Air Management
Monitoring and Radiation Division
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79 Elm Street
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Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Discussion of Proposed Changes

Technical Specifications Change Request 3-1-01
Slave Relay Testing Surveillance Interval
Discussion of Proposed Changes

Introduction

Technical Specification 3.3.2 for Millstone Unit No. 3 requires quarterly testing of slave relays installed in the Engineered Safety Features Actuation System (ESFAS). WCAP-13878 and WCAP-13900 evaluated the acceptability of extending the surveillance interval for both Westinghouse AR series relays and Potter & Brumfield (P&B) MDR series relays which meet certain reliability assessment criteria.

The Millstone Unit No. 3 ESFAS does not utilize Westinghouse AR series relays. Therefore, the proposed change will extend the surveillance interval for only P&B MDR series relays which meet the reliability assessment criteria specified within WCAP-13878 and WCAP-13900 from a "Q" (at least once every 92 days) surveillance interval to an "R" (at least once every 18 months) surveillance interval.

Background

The Standard Technical Specifications for Westinghouse Pressurized Water Reactor (PWR) plants, both NUREG-0452, Rev. 4 and 5, and NUREG-1431, the new improved standard, require quarterly testing of slave relays in the Engineered Safety Features Actuation System (ESFAS). This requirement involves testing the relays at power, with the attendant risk of inadvertent actuation of the Engineered Safety Features equipment. In addition, the on-line testing of slave relays requires significant plant manipulation, abnormal configurations, and removes from service various equipment making it unavailable to perform its intended safety function.

Generic Letter 93-05[1] resulted from the recommendations of a 1993 NRC task group formed to investigate problems with surveillance testing required by the Technical Specifications. The objective of the NRC task group was: (1) to review the basis for test frequencies; (2) to ensure that the tests promote safety and do not degrade equipment; and (3) to review surveillance tests for unnecessary burden on plant personnel. The studies found that while some testing at power is essential to verify equipment and system operability, safety can be improved, equipment degradation decreased, and unnecessary personnel burden relaxed by reducing the amount of testing performed at power. Slave relay test frequency relaxation is consistent with the objectives of the NRC task group and the recommendations of Generic Letter 93-05.[1]

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[1] Generic Letter 93-05, "Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation," dated September 1993.

The Westinghouse Owners Group (WOG) sponsored a reliability assessment of specific relay types to establish a slave relay surveillance test interval based on relay reliability. The study is documented in WCAP-13878 for P&B MDR series relays. A review of the slave relay testing surveillance interval against the six Science Applications International Corporation (SAIC) review criteria determined that the reliability of the P&B MDR series relays, as used in an ESFAS application, is so high that elimination of the routine testing of slave relays when the reactor is at power will have a positive impact on ESFAS availability and, therefore, plant safety. There is no decrease in relay reliability from the reduced testing but there is a significant reduction in operator burden, and the potential for challenges to the safety systems coupled with less time that the safety systems are unavailable. Furthermore, reduced plant testing could reduce occupational exposure, and potentially, the NRC regulatory burden as well. Therefore, there is no significant negative impact from the reduced testing of P&B MDR series relays.

Pacific Gas & Electric Diablo Canyon, acting as Lead Plant for this WOG program, referenced this reliability assessment in their plant license amendment request to extend the slave relay surveillance test interval to the refueling interval. The license amendment was approved by NRC Safety Evaluation Report (SER) dated August 19, 1996.[2] In addition, the proposed changes are similar to the surveillance test interval extensions previously approved by the NRC for the Callaway, Wolf Creek, South Texas and Vogtle plants.

Northeast Nuclear Energy Company (NNECO) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 3. NNECO is proposing to change Technical Specification 3.3.2, "Instrumentation - Engineered Safety Features Actuation System Instrumentation." The Bases of Technical Specification 3.3.2 will be modified to reflect this change. These changes are consistent with WCAPs 13878 and 13900, as well as Generic Letter 93-05.[1]

Technical Specification Changes

Technical Specification 3.3.2, Table 4.3-2, ESFAS Instrumentation Surveillance Requirements

Table Notation 4 will be added to Table 4.3-2. Table notation 4 will state, "For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP-13900, the surveillance test

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[2] NRC letter from Steven D. Bloom to Gregory M. Rueger, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit No. 1 (TAC No. M90971) and Unit No. 2 (TAC No. M90972)," dated August 19, 1996.

interval for slave relay testing is R." This wording differs slightly from the recommended wording provided in WCAP-13878 in that the description of the testing to be performed has been revised to apply to "slave relay testing" instead of "the slave relay." This change was made to ensure consistency and clarity with respect to how Table 4.3-2 of the Millstone Unit No. 3 Technical Specifications is worded.

Table Notation 4 revises the required surveillance interval for P&B MDR series slave relays, which meet the identified criteria, from "Q" (at least once per 92 days) to "R" (at least once per 18 months) for the following functional units:

- Functional Unit 1.b., Safety Injection - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 2.b., Containment Spray - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 3.a.(2), Containment Isolation, Phase "A" Isolation - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 3.b.(2), Containment Isolation, Phase "B" Isolation - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 4.b., Steam Line Isolation - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 5.a., Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays;
- Functional Unit 5.b, Turbine Trip and Feedwater Isolation - Steam Generator Water Level High-High Relays;
- Functional Unit 6.b., Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays; and
- Functional Unit 7.c., Control Building Isolation - Automatic Actuation Logic and Actuation Relays.

Technical Specification Bases

The Technical Specification Bases for Section 3.3.1 and 3.3.2, "Reactor Trip System Instrumentation and Engineered Safety Features Actuation System Instrumentation," will be revised to include a discussion of what slave relay testing can be performed on an "R" (at least once every 18 months) frequency.

Safety Summary

As part of the development of WCAP-13878, Westinghouse performed evaluations to determine the reliability of P&B MDR series relays used in the Solid State Protection System (SSPS) to actuate ESF components. These evaluations include a generic review of industry information on slave relay problems, a slave relay surveillance test failure study, a Failure Modes and Effects Analysis (FMEA), and an aging assessment.

The NRC SER for the approval of WCAP-13878, Revision 1, and WCAP-13900, Revision 0, identifies four issues that must be addressed by any licensee pursuing surveillance test interval extension. The four issues and discussion of the Millstone Unit No. 3 evaluation of each issue is provided in Attachment 5.

The changes being proposed involve the extension of the surveillance interval for the P&B slave relays in the ESFAS from the current "Q" (at least once every 92 days) surveillance interval to an "R" (at least once every 18 months) surveillance interval. The NRC approved WCAP-13878 provides the technical justification for relaxing the slave relay surveillance test interval. The basis for relaxing the surveillance interval is the reliability assessment, which established that for normally de-energized relays of these two types, the reliability is invariant with time and there are no significant factors that will cause the relays to age or wear out within the plant lifetime.

The slave relay reliability studies demonstrate that the slave relays are highly reliable, principally due to the very low cycle demand and the extended periods during which no demand is expected, and is supported by the results of the aging assessments. The aging assessments concluded that the degradation of both the normally de-energized slave relays and the normally energized slave relays is sufficiently slow (i.e., the time to failure due to degradation is sufficiently long) that an 18-month surveillance test interval will adequately identify slave relay failures. A 92-day slave relay surveillance test interval is no more likely to detect significant changes in the SSPS slave relays than an 18-month surveillance test interval. The aging assessments demonstrate that the normally de-energized slave relays, and the normally energized slave relays with a low duty cycle will not experience temperature-induced, age-related degradation sufficient to result in failure in the 40 year plant life.

Aging will, however, affect the reliability of these relays in normally energized applications within the plant lifetime. So, to assure the original relay reliability a replacement interval for normally energized relays was specified in WCAP-13878. However, the Millstone Unit No. 3 ESFAS design does not utilize P&B MDR series slave relays which are normally energized. Therefore, a formal replacement interval has not been developed at this time for P&B MDR series relays.

Based on the results of these evaluations, the P&B MDR series slave relays have a design and cycle life capability greatly in excess of that required for the SSPS slave relay application. The maximum temperature experienced by the slave relays in the

SSPS cabinets is far less than the manufacturer's recommended temperature for reliable slave relay operation.

The failure probability of standby components, as in the case of Millstone Unit No. 3 SSPS slave relays (which are normally de-energized) consists of two parts, namely, a shock failure and a standby failure. The shock failure probability is independent of changes to the surveillance testing interval. Additionally, the standby failure probability would not increase linearly with increasing the testing interval. The Millstone Unit No. 3 Probabilistic Risk Analysis (PRA) model assumes that the fault exposure factor for standby components is unity unless the components have never been tested. With respect to the slave relays, this PRA assumption is also consistent with WCAP-13878 study which shows that the reliability of normally de-energized P&B MDR series slave relays is invariant with time and there are no significant factors that would cause the relays to age or wear out within the plant lifetime. Accordingly, this proposed TS change will not result in increasing the plant's core damage frequency. To the contrary, there are safety benefits of extending the surveillance testing interval of P&B MDR series slave relays from quarterly to once per 18 months such as avoiding the potential for inadvertent actuation of ESF components, less components cycling and manipulations, and less ESFAS unavailability during at-power operations. It should also be noted that the continuity of the P&B MDR series slave relay coils will continue to be verified during SSPS master relay monthly testing.

Therefore, NNECO concludes that the reliability and operating history of the slave relays are such that the extension of the surveillance interval will not adversely affect the ability of the ESFAS to perform its safety function. Implementing the proposed changes will also reduce system degradation due to less component cycling and manipulation and will reduce unnecessary burden on plant personnel due to less frequent testing.

The proposed changes to the slave relay testing surveillance interval for P&B MDR series slave relays installed in the ESFAS which meet the criteria specified in WCAP-13878 and WCAP-13900 will not adversely affect the availability of the equipment used to mitigate design basis accidents. The plant response to the design basis accidents will not change. Therefore, there will be no adverse impact on public health and safety. Thus, the proposed changes are safe.

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Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Significant Hazards Consideration

Technical Specifications Change Request 3-1-01
Slave Relay Surveillance Test Interval
Significant Hazards Consideration

Description of License Amendment Request

The proposed changes will revise Technical Specification 3.3.2, "Instrumentation - Engineered Safety Features Actuation System Instrumentation," by extending the slave relay surveillance test interval for P&B MDR series relays which meet the criteria of WCAP-13878 and WCAP-13900 from a "Q" (at least once every 92 days) surveillance interval to an "R" (at least once every 18 months) surveillance interval.

Basis for No Significant Hazards Consideration

In accordance with 10 CFR 50.92, Northeast Nuclear Energy Company (NNECO) 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 would not:

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

   This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same ESFAS instrumentation is being used and the same ESFAS system reliability is expected. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter assumptions previously made in evaluating the radiological consequences of an accident described in the SAR. Therefore, the proposed amendment does not result in any increase in 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.

   This change does not alter the performance of the ESFAS mitigation systems assumed in the plant safety analysis. Changing the surveillance interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios. Implementation of the proposed amendment does 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.

This change does not affect the total ESFAS system response assumed in the safety analysis. The periodic slave relay functional verification is relaxed because of the demonstrated high reliability of the relay and its insensitivity to any short term wear or aging effects. It is thus concluded that the proposed license amendment request does not result in a reduction in margin with respect to plant safety.

Conclusion

Based on the preceding evaluation, it is concluded that relaxing the slave relay surveillance test interval from a "Q" (at least once every 92 days) interval, to an "R" (at least once every 18 months) interval, is acceptable for the identified P&B MDR series relays, and the proposed license amendment does not involve a Significant Hazards Consideration Finding as defined in 10 CFR 50.92.

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Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Marked Up Pages

List of Affected Pages

Technical Specification Section Number	Title of Section	Page and Amendment Numbers
3.3.2	Instrumentation - ESFAS Instrumentation	3/4 3-36 Am. 100 3/4 3-37 Am. 129 3/4 3-38 Am. 100 3/4 3-39 Am. 100 3/4 3-41 Am. 129 B3/4 3-2a

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

Amendment No. 55, 79, 79, 109

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Purge Isolation	S	R	Q	N.A.	N.A.	N.A.	N.A.	5, 6#
4. Steam Line Isolation								
a. Manual Initiation								
1) Individual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) System	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

Amendment No. 66, 70, 79, 199, 169

June 27, 1996

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2
b. Steam Generator Water Level-High-High	S	R	Q	N.A.	M(1)	M(1)	Q(4)	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

Amendment No. 55, 79, 79, 109

January 3, 1995

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	See Item 8. below for all Loss of Power Surveillance.							
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q (4)	1, 2, 3, 4
d. Containment Pressure--High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

Amendment No. 45, 79, 79, 100
January 3, 1995

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Control Building Isolation (Continued)								
e. Control Building Inlet Ventilation Radiation	S	R	Q	N.A.	N.A.	N.A.	N.A.	All
8. Loss of Power								
a. 4 kV Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Bus Undervoltage (Grid Degraded Voltage)	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low-Low T_{avg}, P-12	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
10. Emergency Generator Load Sequencer	N.A.	N.A.	N.A.	N.A.	Q(1, 2)	N.A.	N.A.	1, 2, 3, 4

For Information Only

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated operable by the performance of an ACTUATION LOGIC TEST at least once per 92 days.

(3) On a monthly basis, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the channel calibration required once ~~per 18 months~~.

(4) ~~During core~~ (Insert A) alterations or movement of irradiated fuel within the containment. The provisions of Specification 3.0.3 are not applicable.

Insert A

For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP-13900, the surveillance interval for slave relay testing is R.

3/4.3 INSTRUMENTATION

BASES

For Information Only

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches 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 from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, 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 safety analyses. 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 Engineered Safety Features Actuation System Nominal Trip Setpoints specified in Table 3.3-4 are the nominal values of which the bistables are set for each functional unit. The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) are considered the Limiting Safety System Settings as identified in 10CFR50.36 and have been selected to mitigate the consequences of accidents. A Setpoint is considered to be consistent with the nominal value when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, the Nominal Trip Setpoints may be adjusted in the conservative direction provided the calibration tolerance remains unchanged.

Measurement and Test Equipment accuracy is administratively controlled by plant procedures and is included in the plant uncertainty calculations as defined in WCAP-10991. Operability determinations are based on the use of Measurement and Test Equipment that conforms with the accuracy used in the plant uncertainty calculation.

The Allowable Value specified in Table 3.3-4 defines the limit beyond which a channel is inoperable. If the process rack bistable setting is measured within the "as left" calibration tolerance, which specifies the difference between the Allowable Value and Nominal Trip Setpoint, then the channel is considered to be operable.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The methodology, as defined in WCAP-10991 to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

The above Bases do not apply to the two radiation monitors in the ESF Table (Item 3C and Item 7E). For these radiation monitors the allowable values are essentially nominal values. Due to the uncertainties involved in radiological parameters, the methodologies of WCAP-10991 were not applied. Actual trip setpoints will be reestablished below the allowable value based on calibration accuracies and good practices.

The measurement for response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. The RTS and ESF response times are included in the "Technical Requirements Manual." Any changes to the RTS and ESF response times shall be in accordance with Section 50.59 of 10CFR50 and approved by the Plant Operations Review Committee. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time. Detector response times may be measured by the in situ on line noise analysis-response time degradation method described in the Westinghouse Topical Report, "The Use of Process Noise Measurements To Determine Response Characteristics of Protection Sensors in U.S. Plants," August 1983.

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

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

REACTOR TRIP BREAKERS ← (Insert B)

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

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

Insert B

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

Docket No. 50-423
B18311

Attachment 4

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Retyped Pages

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Building Isolation (Manual Initiation Only), Start Diesel Generators, and Service Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4 |
c. Containment Pressure-High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4 |
c. Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

Amendment No. 46, 70, 79, 100,

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4 |
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4 |
3) Containment Pressure-High-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Purge Isolation	S	R	Q	N.A.	N.A.	N.A.	N.A.	5, 6#
4. Steam Line Isolation								
a. Manual Initiation								
1) Individual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) System	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4

Amendment No. 46, 70, 79, 100, 129,

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
c. Containment Pressure-High-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Steam Line Pressure-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	Q	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2
b. Steam Generator Water Level-High-High	S	R	Q	N.A.	M(1)	M(1)	Q(4)	1, 2, 3
c. Safety Injection Actuation Logic	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
d. T_{ave} Low Coincident with Reactor Trip (P-4)	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	See Item 8. below for all Loss of Power Surveillance.							
f. Containment Depressurization Actuation (CDA)	See Item 2. above for all CDA Surveillance Requirements.							
7. Control Building Isolation								
a. Manual Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Manual Safety Injection Actuation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(4)	1, 2, 3, 4
d. Containment Pressure--High-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

Amendment No. 49, 70, 79, 100,

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) This surveillance may be performed continuously by the emergency generator load sequencer auto test system as long as the EGLS auto test system is demonstrated operable by the performance of an ACTUATION LOGIC TEST at least once per 92 days.

(3) On a monthly basis, a loss of voltage condition will be initiated at each undervoltage monitoring relay to verify individual relay operation. Setpoint verification and actuation of the associated logic and alarm relays will be performed as part of the channel calibration required once per 18 months.

(4) For Engineered Safety Features Actuation System functional units with only Potter & Brumfield MDR series relays used in a clean, environmentally controlled cabinet, as discussed in Westinghouse Owners Group Report WCAP-13900, the surveillance interval for slave relay testing is R.

During core alterations or movement of irradiated fuel within the containment. The provisions of Specification 3.0.3 are not applicable.

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

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

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

REACTOR TRIP BREAKER

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

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

Docket No. 50-423
B18311

Attachment 5

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Plant Specific Information on the applicability of
WCAP-13878 and WCAP-13900

Technical Specifications Change Request 3-1-01
Instrumentation - Engineered Safety Features Actuation System
Instrumentation Slave Relay Testing Surveillance Interval
Plant Specific Information on the applicability of
WCAP-13878 and WCAP-13900

The Nuclear Regulatory Commission (NRC) SER for the approval of WCAP-13878, Revision 1, and WCAP-13900, Revision 0, stated that licensees referencing these WCAPs in plant-specific license amendment requests for Solid State Protection System (SSPS) slave relay surveillance test interval extensions should provide certain additional information. The NRC SER identifies four issues that must be addressed by any licensee pursuing surveillance test interval extension. The four issues and discussion of the Millstone Unit No. 3 evaluation of each issue follows:

1. Confirm the applicability of the WCAP-13878, Revision 1, analyses to the facility design.

   WCAP-13878 provides a Failure Mode and Effects Analysis (FMEA) and an aging assessment of Potter & Brumfield (P&B) MDR series relays. The aging assessment is predicated on the relays being located in a clean, environmentally controlled environment. Millstone Unit No. 3 MDR series slave relays are located in a clean, environmentally controlled environment. Specifically, MDR series slave relays are located in the SSPS cabinets in the Instrument Rack Room (IRR). The IRR is adjacent to the facility control room and is a clean, environmentally controlled environment as defined in the Millstone Unit No. 3 Electrical Equipment Qualification (EEQ) Program.

   MDR auxiliary relays that are used in slave relay circuits proposed for interval extension are also located in a clean, environmentally controlled environment. These MDR auxiliary relays are also located in the IRR in auxiliary relay cabinets.

   All Millstone Unit No. 3 slave relays proposed for surveillance test interval extension are either Westinghouse Replacement Components Services (WRCS) model number MDR-4121-1 (P&B model MDR-4076, 118 VAC latching relay) or WRCS model number MDR-4103-1 (P&B model MDR-134-1, 118 VAC non-latching relay). All Millstone Unit No. 3 MDR auxiliary relays used in slave relay circuits which are proposed for surveillance test interval extension are WRCS model number MDR-5076-1 (P&B model MDR-138-8, 125 VDC non-latching relay). Each of these three models of P&B MDR series relays were analyzed for reliability as discussed in WCAP-13878 and found to be acceptable for slave relay surveillance test interval extension to an "R" (at least once every 18 months) surveillance interval. WCAP-13878 also qualifies other models of P&B MDR series relays based on similarity, but these other models are currently not utilized in the Millstone Unit No. 3 ESFAS. Additionally, none of the P&B MDR series relays installed in the Millstone Unit No. 3 ESFAS are refurbished.

2. Confirm the adequacy of the facility procurement program for P&B MDR series relays.

   AEOD/S93-06,[1] NRC Information Notice 90-57,[2] NRC Information Notice 90-57 Supplement 1,[3] and a San Onofre Nuclear Generating Station (SONGS) 10 CFR 21 notification dated July 21, 1995 were referenced in the NRC SER for the approval of WCAP-13878, Revision 1, and WCAP-13900, Revision 0, as relevant documents which should be reviewed in determining the adequacy of the facility procurement program. These documents relate to the failure of P&B MDR series relays, as well as the delivery of P&B MDR series relays to SONGS with bent contact arms following rework by P&B. The SER requires that any licensee that proposes surveillance interval extensions based on WCAP-13878 and WCAP-13900 must ensure that their procurement program is adequate for detecting the type of failures described in these references.

   The Millstone Nuclear Power Station procurement programs are adequate for detecting potential failures as identified in the referenced documents. The Millstone Restricted Materials Suppliers List (RMSL) references each of these documents as a cross-reference to the manufacturer, P&B. Accordingly, inspections, dedications and testing of MDR relays performed as part of the procurement of all P&B MDR relays address the issues identified within these references.

3. Ensure that all pre-1992 MDR series relays normally energized or in a 20% duty cycle have been removed.

   AEOD Report S93-06 identified failure modes due to out-gassing of compounds in the varnish and grommet materials in pre-1992 P&B MDR ac relays. The NRC SER requires that all pre-1992 P&B MDR relays used in either normally energized or a 20% duty cycle be removed. All P&B MDR series slave and auxiliary relays currently installed in the Millstone Unit No. 3 ESFAS proposed for surveillance interval extension are normally de-energized (duty = 0%) and therefore, replacement of any Millstone Unit No. 3 P&B relays is not required.

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(1) Office for Analysis and Evaluation of Operation Data Special Study Report AEOD/S93-06, "Potter & Brumfield Model MDR Rotary Relay Failures," dated December 1993.

(2) NRC Information Notice 90-57, "Substandard, Refurbished Potter & Brumfield Relays Misrepresented as New," dated September 5, 1990.

(3) NRC Information Notice 90-57 Supplement 1, "Substandard, Refurbished Potter & Brumfield Relays Misrepresented as New," dated November 27, 1991.

4. Ensure/Perform a Contact Loading Analysis for all P&B MDR series relays which are candidates for surveillance extension.

AEOD Report S93-06 and NRC Information Notice 92-19[4] describe contact failures due to the misapplication of P&B MDR relays. Westinghouse issued Technical Bulletin, NSD-TB-92-02, Revision 0, dated January 24, 1992[5] in response to excess contact loading failures of MDR relays used as slave relays in SSPSs. Specifically, the concern was MDR relay contacts that are required to open to de-energize normally energized solenoid valves with DC coils.

A contact loading analysis was performed for Millstone Unit No. 3 P&B MDR series slave and auxiliary relays which are acceptable for surveillance test interval extension in accordance with WCAP-13878 and WCAP-13900. The contact loading study recorded the manufacturer, model, and device ratings of each actuation device (solenoid or relay) operated by each slave relay contact. In the contact loading study, all slave relay single contacts were evaluated for overload, continuous current and switching capabilities for both ac and dc contact applications of the slave relay contacts. The contact de-rating required for inductive loads was considered in determining the acceptability of the loading. Additionally, a review of the slave relay surveillance history found no intermittent contact failures indicative of contact erosion. The analysis demonstrated that all P&B MDR series slave and auxiliary relay contacts are used within their single contact ratings of 10A @ 115Vac or 0.3A @ 143Vdc, with one exception.

The total load for the K740(A&B) slave relays (one per ESFAS train, two total) was determined to be 6.0A, which exceeds both the 0.30A 125Vdc rating for breaking inductive, as well as the 0.8A 125Vdc resistive P&B rating for the MDR contact. For this application, the MDR contact closes to energize the trip coil for a 4160V GE breaker. The circuit is a typical 125Vdc breaker control circuit where a closed auxiliary contact off the breaker is wired in series with the trip coil and, therefore, the MDR contact. As the breaker opens, the breaker auxiliary contact also opens thereby removing power from the trip coil. The breaker/auxiliary contact opening time is less than 3 cycles, i.e., 50 ms. Therefore, the MDR contact closes and passes current for less than 50ms. Also, the MDR contact never opens to break the inductive load.

A review of various GE protective relays that are used to operate breaker trip coils shows that contact ratings are given for both continuous duty and for "One-Second"/"Momentary"/"Trip Contact" usage. The typical difference between the GE continuous ratings and the trip contact ratings is a factor of anywhere from 10 up to a factor of approximately 50. A factor of 10 relative to the 0.8A P&B

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(4) NRC Information Notice 92-19, "Misapplication of Potter & Brumfield MDR Rotary Relays," dated March 2, 1992.

(5) Westinghouse Technical Bulletin NSD-TB-92-02, Rev. 0, "Misapplied Relay Contacts," dated January 24, 1992.

rating would support the relay's 6A trip coil load. Additionally, the 6A dc trip coil load is less than the 10A 115Vac P&B contact rating. Thus, it can be concluded that a 3 cycle/50 ms duration dc load of 6A would not discharge more total energy to the contact than an analogous ac voltage load at 10A continuously. Finally, the application of the MDR contact in a fashion typical of GE protective relays, is supported by years of successful testing at Millstone Unit No. 3. Based on the above qualitative assessment, the surveillance test interval for the K740(A & B) slave relays can be extended.

The NRC SER for the approval of WCAP-13878, Revision 1, and WCAP-13900, Revision 0, also identified an additional action to be performed upon the implementation of extended surveillance intervals for the P&B MDR series relays. The SER states that the extended surveillance interval for MDR relays is acceptable based on the high reliability of the relays. However, if this high reliability is not demonstrated in the future, then an evaluation of the suitability of the surveillance interval should be performed. The SER concludes that if two or more P&B MDR Series ESFAS relays fail within a 12-month period, an evaluation which considers design, maintenance and testing of all P&B MDR ESFAS relays should be performed to verify the adequacy of the extended surveillance interval. The SER concludes that if the surveillance interval is inadequate for detecting a single relay failure, the surveillance interval should be decreased such that an ESFAS subgroup relay failure can be detected prior to the occurrence of a second failure.

A review requirement consistent with the NRC SER requirements has been incorporated into the Millstone Unit No. 3 Reactor Protection System Maintenance Rule Scoping Document (which addresses the ESFAS). This document implements the requirements of 10 CFR 50.65 and provides instructions for initiation, analysis, retrieval trending, and periodic reporting of data relative to performance indicators of plant systems and components. The program includes guidance for trending and reporting of repetitive preventable failures of functions. As such, any future P&B MDR Series slave or auxiliary relay functional failures will be evaluated to address the adequacy of the extended surveillance interval.