UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 9 1987

Docket No.: 50-423

Mr. E. J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: REVISION TO TECHNICAL SPECIFICATIONS - ENGINEERED SAFETY FEATURES RESPONSE TIME

Re: Millstone Nuclear Power Station, Unit 3

The Commission has issued the enclosed Amendment No. 3 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit 3. This amendment is in response to your application dated April 6, 1987, and supplemental letters dated April 7, 1987 and April 8, 1987.

The amendment would increase the engineered safety features (ESF) response time for Low Steamline Pressure in Technical Specification Table 3.3-5, Item 4.a by 15 seconds to 27 seconds with offsite power and 37 seconds without offsite power.

The staff has reviewed the circumstances associated with your request and has concluded that this change is needed to avoid a delay in startup. Therefore, in accordance with 10 CFR 50.91(a)(5), justification for the issuance of an emergency technical specification change existed.

A notice of Consideration of Issuance of Amendment to License will be included in the Commission's biweekly Federal Register notices. A copy of our related Safety Evaluation is enclosed.

Sincerely,

Elizabeth L. Doolittle

Elizabeth L. Doolittle, Project Manager
Project Directorate No. 5
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 3 to License No. NPF-49
2. Safety Evaluation

cp/ EJD.

E. J. Mroczka, Senior V.Pres.
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.*

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Northeast Nuclear Energy Company, et al., (the licensees) dated April 6, 1987 and supplemented by letters dated April 7, 1987 and April 8, 1987 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safty of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*Northeast Nuclear Energy Company is authorized to act as agent and representative for the following Owners: Central Maine Power Company, Central Vermont Public Service Corporation, Chicopee Municipal Lighting Plant, City of Burlington, Vermont, Connecticut Municipal Electric Light Company, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, The Village of Lyndonville Electric Department, Western Massachusetts Electric Company, and Vermont Electric Generaton and Transmission Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the technical specifications, as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The technical specifications contained in Appendix A revised through Amendment No. 3, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Northeast Nuclear Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses

Victor Nerses, Acting Director
PWR Project Directorate No. 5
Division of PWR Licensing-A

Attachments:
Changes to the Technical
Specifications

Date of Issuance: APR 9 1987

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Unit No. 3

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated April 6, 1987, and supplemented by letters dated April 7, and April 8, 1987 the Northeast Nuclear Energy Company (NNECO) submitted a request to change the Millstone Unit 3 technical specifications.

The amendment revises Technical Specification Table 3.3-5, Item 4.a by increasing the ESF response time for Low Steamline Pressure by 15 seconds to 27 seconds with offsite power and 37 seconds without offsite power.

The staff's evaluation of the licensee's request for an emergency license amendment is provided below.

2.0 BACKGROUND

During the performance of a safety evaluation to delete certain valves from ESF surveillance procedures, NNECO examined the interlock between the Volume Control Tank (VCT) outlet valves (3CHS*LCV 112 B&C) and the Refueling Water Storage Tank (RWST) outlet valves (3CHS*LCV 112 D&E). It was determined that the closure response time of 3CHS*LCV 112 B&C is not critical for a Safety Injection (SI) in the case of an SI required to protect against a LOCA since the important function for the ESF response time was to deliver cooling water to the core. However, when the SI from low steamline pressure was examined, it was determined that the closure response time of 3CHS*LCV 112 B&C is critical to ensure that borated water can be delivered to the core in the time assumed in the FSAR.

The FSAR steamline break analysis which supports the current Technical Specifications (Table 3.3-5) assumes the following response times for delivery of borated water to the reactor coolant system (RCS):

1. SI signal generation (2 seconds)
2. Diesel start (including time to come up to speed [10 seconds])
3. Valve stroke times and pump to full speed (10 seconds)

However, this analysis assumes that the VCT and RWST isolation valves stroke simultaneously rather than sequentially. The valve interlock logic increases the delay time for the availability of borated water by 15 seconds to 27 seconds with offsite power and 37 seconds without offsite power. The only transient impacted by the increased time delay is the steamline break event.

On April 3, 1987 NNECO notified the NRC of a need for an emergency tech spec change to Table 3.3-5, Item 4.a. The NRC staff held several conference calls with NNECO following their request. Additionally, NNECO submitted a proposed revision to technical specifications on April 6, 1987.

Based on its review of this information the NRC staff verbally granted a temporary waiver of compliance from the subject technical specification to permit operation with 27 and 37 second times at the Millstone 3 plant on April 3, 1987. On April 6, 1987 the NRC issued a letter confirming the waiver of compliance to be in effect through April 7, 1987.

On April 7, 1987 NNECO submitted supplemental information on the emergency nature of the requested change and stated that it intended to submit additional clarification to support the proposed amendment. Also since Millstone Unit No. 3 went from Mode 2 to Mode 3 due to a main steam isolation solenoid problem, the emergency authorization would not be required until April 9, 1987. In a letter dated April 8, 1987 NNECO submitted additional information which should be added to Table 3.3-5 Items 2.a, 3.a and 4.a noting the sequential transfer of charging pump suction from the VCT to the RWST. Also NNECO stated its intention to revise the Technical Specification Bases Section to reflect the valve interlock in the safety injection system between the VCT isolation valves and the RWST isolation valves.

3.0 EVALUATION

In the original steamline break analysis (SLB) for Millstone 3, a 22 second delay for safety injection (SI) was assumed. The licensee's evaluation indicates that previous sensitivity studies have shown that increased SI actuation delay times, specifically 27 seconds when offsite power is available and 37 seconds with loss of offsite power, result in only small changes in analysis results. The licensee concludes that for the SLB analysis the DNB design basis is still met and that the conclusions presented in the FSAR remain valid. For the LOCA case, RCS boration is only required for the long term. The staff concludes that the licensee's conclusions are reasonable and that the requested tech spec changes should not result in unacceptable fuel design limits or adversely affect the health and safety of the public.

4.0 EMERGENCY CIRCUMSTANCES

On March 29, 1987 while in Mode 5, the licensee was performing emergency response time testing and noticed that several response times did not meet ESF response times required by Technical Specification Table 3.3-5. On March 29, 1987 NNECO filed a four hour report with NRC in accordance with station administrative procedures for reporting events required by 10 CFR 50.73. On March 30, 1987 the licensee subsequently decided that the valves which did not pass the surveillance had been added to the ESF response time surveillance since the original performance of the surveillance and the valves should be removed from the surveillance procedure. In order to approve this procedure change, the Millstone 3 Plant Operations Review Committee (PORC) requested that Millstone 3

Engineering perform a safety evaluation to delete the valves from the surveillance procedure. During performance of the safety evaluation, the interlock between the VCT outlet control valves and the RWST outlet valves was examined. At this time the licensee realized that the FSAR steamline break analysis which supports the current Technical Specifications, Table 3.3-5, assumes that the VCT and RWST isolation valves stroke simultaneously rather than sequentially, thereby requiring an additional 15 seconds for delivery of borated water to the RCS. An additional evaluation was performed to see if the RWST head was enough to overcome the head of the VCT when the RWST outlet valves go open. It was determined that the static head of the VCT was too great. On April 2, 1987 the licensee requested Westinghouse to perform a safety evaluation to support the technical specification change. Following completion of this work on the morning of April 3, 1987 NNECO filed a report with the NRC pursuant to 10 CFR 50.72(b)(2)(iii)(D) while Millstone 3 was in Mode 5.

Thus upon confirmation of the need for this change and completion of the supporting work, the licensee promptly notified NRC and pursued expeditious resolution.

The staff has reviewed the licensee's assessment and concludes that the licensee acted in a timely fashion to pursue resolution of this issue.

Based upon the information in the licensee's letter dated April 6, 1987, and supplemented by letters dated April 7, 1987 and April 8, 1987, the staff has determined that the above circumstances constitute an emergency situation since, if no action were taken, delay of plant startup would be required.

4.1 No Significant Hazards Consideration Determination

In accordance with 10 CFR 50.92, the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) involve a significant reduction in a margin of safety.

The information in this section provides the staff's evaluation of this license amendment against these criteria:

A. <u>Involve a significant increase in the probability or consequences of an accident previously evaluated</u>.

An increase in the acceptance criterion for the ESF response time is acceptable since the evaluation of the impact of the increased delay on the steamline break event demonstrated that the DNB design basis is still met. The conclusions in the FSAR remain valid.

B. <u>Create the possibility of a new or different kind of accident from any accident previously evaluated</u>.

There are no new failure modes associated with this proposed change as no design changes have been made. No new accident is created because the same equipment is assumed to perform in the same manner as before. Therefore, an increase in the ESF response time for low steamline pressure does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

C. Involve a significant reduction in margin of safety.

The proposed change is intended to bring the technical specification surveillance in line with the basis. There is no impact on the consequences or protective boundaries and all acceptance criteria in the FSAR will still be met. Therefore, safety limits will still be met.

4.2 State Consultation

Mr. K. McCarthy, Director, Radiation Control Unit, Department of Environmental Protection, State of Connecticut, was contacted conerning the proposed emergency license amendment on April 7, 1987. After a discussion on the subject amendment, Mr. McCarthy indicated that all his comments have been resolved.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. Therefore the amendment will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) increase the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therefore, the amendment does not involve significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and the security or to the health and safety of the public.

7.0 REFERENCE

1. E. J. Mroczka to U.S. Nuclear Regulatory Commission, dated April 6, 1987, Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification Engineered Safety Features (ESF) Response Time.

2. E. J. Mroczka to U.S. Nuclear Regulatory Commission, dated April 7, 1987, Millstone Nuclear Power Station, Unit No. 3 Amendment to Facility Operating License No. NPF-49, Supplemental Information, ESF Response Time.

3. E. J. Mroczka to U.S. Nuclear Regulatory Commission, dated April 8, 1987, Millstone Nuclear Power Station, Unit No. 3 Amendment to Facility Operating License No. NPF-49, Supplemental Information, ESF Response Time.

Date of Issuance: APR 9 1987

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ATTACHMENT TO LICENSE AMENDMENT NO. 3

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the areas of change. Overleaf pages* have been provided to maintain document completeness.

REMOVE	INSERT
3/4 3-31*	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34*	3/4 3-34*
3/4 3-35	3/4 3-35
3/4 3-36*	3/4 3-36*
B 3/4 3-1*	B 3/4 3-1*
B 3/4 3-2	B 3/4 3-2
	B 3/4 3-2a

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is less than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Steam Line Isolation	N.A.
f. Feedwater Isolation	N.A.
g. Auxiliary Feedwater	N.A.
h. Service Water	N.A.
i. Control Building Isolation	N.A.
j. Reactor Trip	N.A.
k. Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	$\leq 27^{(9)}/12^{(10)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generator	≤ 12
b. Control Building Isolation	≤ 5
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(9)}/12^{(10)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generators	≤ 12

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(5)}/37^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 6.8^{(3)}$
3) Phase "A" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
4) Auxiliary Feedwater	≤ 60
5) Service Water	$\leq 90^{(1)}$
6) Start Diesel Generators	≤ 12
b. Steam Line Isolation	$\leq 6.8^{(3)}$
5. Containment Pressure--High-3	
a. Quench Spray	$\leq 32^{(2)}/42^{(1)}$
b. Phase "B" Isolation	$\leq 2^{(2)(6)}/12^{(1)(6)}$
c. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
d. Service Water	$\leq 90^{(1)}$
6. Containment Pressure--High-2	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
7. Steam Line Pressure - Negative Rate--High	
a. Steam Line Isolation	$\leq 6.8^{(3)}$
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	$\leq 6.8^{(3)}$
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pump	≤ 60

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. Loss of Power	
a. 4 kV Bus Undervoltage (Loss of Voltage)	$\leq$ 12
b. 4 kV Emergency Bus Undervoltage (Grid Degraded Voltage)	$\leq 18^{(7)}/310^{(8)}$
12. T_{ave} Low Coincident With Reactor Trip (P-4)	
a. Feedwater Isolation	$\leq 6.8^{(3)}$
13. Control Building Inlet Ventilation Radiation	
a. Control Building Isolation	$\leq$ 3.7
14. Outside Chlorine High	
a. Control Building Isolation	$\leq$ 7

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air-operated valves.

(4) Diesel generator starting and sequence loading delays included. Sequential transfer of Charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included. RHR pumps not included.

(5) Diesel generator starting and sequence loading delays not included. Sequential transfer of Charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included. RHR pumps not included.

(6) Time required to close valves as indicated in Table 3.6-2.

(7) With an ESF signal present.

(8) Without an ESF signal present.

(9) Diesel generator starting and sequence loading delays included. Sequential transfer of Charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assures only opening of RWST valves.

(10) Diesel generator starting and sequence loading delays not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. RHR pumps not included.

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	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	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	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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 Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of

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

the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of 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. 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.

ESF response time specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves are based on values assumed in the non-LOCA safety analyses. For these analyses, injection of borated water from the RWST is credited. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response time, the values specified are based on the LOCA analyses which credit injection flow regardless of the source. Exceptions to this rule are the response times with table notation 10. These response times do not include sequential operation of the RWST and VCT isolation valves but are derived from the non-LOCA analyses. Theses exceptions insure that safety injection pumps (except RHR) are started within an appropriate time when offsite power is present. Since SI functions are identical regardless of the actuation signal, the individual component verification will assure that the response times specified with and without sequential operation of the VCT and RWST valves are met for LOCA and non-LOCA accidents.

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

The Engineered Safety Features Acutation 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.

SUBJECT: ISSUANCE OF AMENDMENT NO. 3 TO NPF-49 -- MILLSTONE NUCLEAR POWER STATION, UNIT 3

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