December 28, 1995

Mr. Robert E. Busch
President - Energy Resources Group
Northeast Utilities Service Company
c/o Mr. Richard M. Kacich
P.O. Box 128
Waterford, CT 06385

SUBJECT: ISSUANCE OF AMENDMENT FOR 24-MONTH FUEL CYCLE (TAC NO. M92203)

Dear Mr. Busch:

The Commission has issued the enclosed Amendment No. 122 to Facility Operating License No. NPF-49 for the Millstone Nuclear Power Station, Unit No. 3, in response to your application dated May 1, 1995.

The amendment revises the Technical Specifications to extend the interval for performance of selected surveillances to accommodate a 24-month fuel cycle. The amendment responds to the first of your group of submittals supporting a change from an 18-month to a 24-month refueling interval. Specifically, the amendment changes the definition for a refueling interval, changes the BASES for surveillances that are performed at least once each fuel cycle and changes the surveillance frequencies for:

1) the flow path tests of the boron injection system,
2) the operability tests of the digital rod position indicators,
3) the drop time of the full-length shutdown and control rods,
4) the channel calibration of the loose-part detection system,
5) the channel calibration of the seismic monitoring instrumentation,
6) the activation of the pumps and the flow path tests of the valves in the containment quench and recirculation spray systems and
7) the tests of the intended actuation positions of the containment isolation valves.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by Phillip F. McKee
for: Vernon L. Rooney, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-423
Enclosures: 1. Amendment No. 122 to NPF-49
2. Safety Evaluation
cc w/encls: See next page

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DOCUMENT NAME: G:\ROONEY\M92203.AMD

OFFICE	LA:PDI-3	PM:DPRE	PM:PDI-3	D:PDI-3	OGC
NAME	SNorris	RClark:bf	VRooney	PMcKee	RBachmann
DATE	11/29/95	11/29/95	12/15/95	11/15/95	11/30/95

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OFFICE	LA:PDI-3	PM:DPRE	PM:PDI-3	D:PDI-3	OGC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 28, 1995

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President - Energy Resources Group
Northeast Utilities Service Company
c/o Mr. Richard M. Kacich
P.O. Box 128
Waterford, CT 06385

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Vernon L. Rooney, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 122 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 122
License No. NPF-49

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated May 1, 1995, 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 safety 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.

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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 122 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee 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, to be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: December 28, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 122

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
1-8	1-8
3/4 1-14	3/4 1-14
3/4 1-24	3/4 1-24
3/4 1-25	3/4 1-25
3/4 3-68	3/4 3-68
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
3/4 6-15	3/4 6-15
B3/4 0-4	B3/4 0-4

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
REFUELING INTERVAL, R	At least once per 24 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two* of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid storage system via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the limits as shown in Figure 3.1-4 at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the Boric Acid Transfer Pump Room temperature and the boric acid storage tank solution temperature are greater than or equal to 67°F when it is a required water source;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position,

c. At least once each REFUELING INTERVAL by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

d. At least once each REFUELING INTERVAL by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 33 gpm to the RCS.

*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 350°F.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ±12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel at least once each REFUELING INTERVAL.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 65% of RATED THERMAL POWER with the reactor coolant stop valves in the nonoperating loop closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once each REFUELING INTERVAL.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once each REFUELING INTERVAL.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Quench Spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Quench Spray subsystem inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Quench Spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days:

   1) Verifying that each valve (manual, power operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, is in its correct position; and

   2) Verifying the temperature of the borated water in the refueling water storage tank is between 40°F and 50°F.

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 114 psid when tested pursuant to Specification 4.0.5;

c. At least once each REFUELING INTERVAL, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal, and

   2) Verifying that each spray pump starts automatically on a CDA test signal.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

RECIRCULATION SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent Recirculation Spray Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Recirculation Spray System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Recirculation Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each Recirculation Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 130 psid when tested pursuant to Specification 4.0.5;

c. At least once each REFUELING INTERVAL by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 ±20 second delay;

d. At least once each REFUELING INTERVAL, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal; and

e. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each isolation valve shall be demonstrated OPERABLE* prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once each REFUELING INTERVAL by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position, and

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

---

*The provisions of Specification 4.0.4 are not applicable for main steam line isolation valves entry into MODE 3 and MODE 4.

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 This specification establishes the limit for which the specified time interval for surveillance requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each REFUELING INTERVAL. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each REFUELING INTERVAL. Likewise, it is not the intent that REFUELING INTERVAL surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the surveillance requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when the Surveillance Requirements have not

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 122

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated May 1, 1995, the Northeast Nuclear Energy Company (the licensee), submitted a request for changes to the Millstone Nuclear Power Station, Unit No. 3 Technical Specifications (TS). The requested changes would revise the TSs to extend the interval for performance of selected surveillances to coincide with a 24-month operating cycle. Specifically, TS that specify an 18-month surveillance will be changed to state that these surveillances are to be performed at least once each refueling interval (i.e., 24-months). Guidance on the proposed TS changes was provided by NRC Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle" dated April 2, 1991.

2.0 EVALUATION

Millstone Unit 3 shutdown for the fifth refueling outage on April 14, 1995, and started in Cycle 6 on June 7, 1995. During the outage, the core was reloaded with fuel designed for a nominal 24 months of operation. To permit operation with this longer fuel cycle, the licensee has or will be proposing to modify the frequency for those surveillance requirements that are normally performed once per fuel cycle. The current Millstone 3 TS specify that these surveillances be performed "at least once per 18 months." The licensee will be proposing that all such surveillance frequencies be changed to "at least once each refueling interval."

The subject application is the first of a group of submittals which the licensee has submitted. This submittal addresses the definition for a refueling interval, the BASES for extending surveillance intervals and the eight sections of the TS listed below.

Section 1.0 of the TS defines the terms used throughout the document. Table 1.1 "FREQUENCY NOTATION" lists the frequency for surveillances identified by various symbols. For those surveillances designated by the letter "R", the frequency is currently specified as "at least once per 18 months." The licensee proposed to further define "R" as "REFUELING INTERVAL" and specify the frequency as "At least once per 24 months." This is identical to the suggested wording in the second paragraph of GL 91-04 and is acceptable.

Section 4.0.2 of the TS states that "Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. This requirement is not being changed. As discussed in GL 91-04, the provision to extend surveillances by 25% of the specified interval would extend the time limit for completing the "R" surveillances from the existing limit of 22.5 months to a maximum of 30 months. The licensee is proposing to revise the BASES for Specification 4.0.2 to delete reference to an 18-month surveillance interval and specify that the surveillances be performed at least once each refueling interval. The proposed change to the BASES is acceptable.

In accordance with the guidance in GL 91-04, for each of the proposed changes in surveillance intervals listed below, the licensee has reviewed the historical plant maintenance and surveillance results to support their conclusion that extending the surveillance intervals has a small effect on safety. In this application, the licensee is proposing to change one or more of the surveillance requirements associated with the following eight sections of the TS:

Section 3.1.2.2 - Reactivity Control Systems - Flow Path Operating

Section 3.1.3.3 - Reactivity Control Systems - Position Indication System - Shutdown

Section 3.1.3.4 - Reactivity Control Systems - Rod Drop Time

Section 3.3.3.3 - Instrumentation - Seismic Instrumentation

Section 3.3.3.8 - Instrumentation - Loose-Part Detection System

Section 3.6.2.1 - Containment Systems - Depressurization and Cooling Systems - Containment Quench Spray System

Section 3.6.2.2 - Containment Systems - Recirculation Spray System

Section 3.6.3 - Containment Systems - Containment Isolation Valves

2.1 Reactivity Control Systems - Flow Path Operating

2.1.1 Design

Among other functions, the Chemical and Volume Control System provides safety grade backup systems for emergency boration of the primary coolant to bring the plant to cold shutdown. Boric acid (3.6 weight percent) can be supplied from the boric acid tanks to the suction of the charging pumps via several different flow paths. It is delivered to the reactor coolant system through the normal charging line and the reactor coolant pump seal injection lines. (The ECCS high head safety injection headers provide a backup path for boration.) There are two boric acid tanks, each of which have 24,000 gallons of useable capacity and which hold 3.6 to 4.1 weight percent boric acid solution. The boric acid normally would be pumped to the suction of the charging pumps by one of the two boric acid transfer pumps, each of which is rated for 75 gpm at 235 feet of head. If both pumps were not available, the solution can flow by gravity from the boric acid tanks to the suction of the charging pumps. Borated water can also be supplied to the charging pumps from the reactor water storage tank.

2.1.2 Technical Specification Change

Limiting Condition for Operation 3.1.2.2 requires that at least two of the above boron injection flow paths shall be operable, which could be the flow path from the boric acid storage system via a boric acid transfer pump and a charging pump or the two flow paths from the refueling water storage tank and the charging pumps. Surveillance Requirements 4.1.2.2.c and 4.1.2.2.d currently state that at least two of the required flow paths shall be demonstrated OPERABLE:

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 33 gpm to the RCS.

The license proposes to change the surveillance interval from 18 months to each refueling interval and to delete the words "during shutdown."

2.1.3 Justification for the Change

In accordance with GL 91-04, the licensee evaluated the equipment performance over the last four operating cycles, including a review of surveillance results, preventative maintenance records and the frequency and types of corrective maintenance.

The review indicated that the automatic valves in the 'A' and 'B' trains actuated as required in response to the safety injection test signal in each case, except two. These two failures were attributed to "procedural deficiencies." The results of the retests, after correction of the procedural deficiencies, were deemed satisfactory. A review of past surveillances indicated that, in each of the cases, the pumps delivered at least 33 gpm of flow to the reactor. Corrective maintenance work performed on the valves during the last four cycles involved minor packing leaks, actuation coil overheating/aging, actuator overthrust, and relay failure. In each of the cases, repairs were able to be performed with no adverse impact on plant operation. In addition, the types of failures that were observed and the number of occurrences were not indicative of a recurring problem. Corrective maintenance work performed on the pumps during the last four cycles involved low oil levels, oil leaks, and breaker linkage bent. In each of the cases, repairs were able to be performed with no adverse impact on plant operation. In addition, the types of failures that were observed and the number of occurrences were not indicative of a recurring problem.

Based on past performance and the maintenance history of the components in the boron injection system, there is reasonable confidence that extending the surveillance frequency from 18 to 24 months will not degrade the ability of this system to perform the intended function. The proposed changes to surveillance requirements 4.1.2.2.c and 4.1.2.2.d are acceptable.

As noted above, the licensee proposes to delete the words "during shutdown" in 2.1.2.c. Generic Letter 91-04 stated that licensees may omit the TS qualification that surveillances be performed "during shutdown." Because the terms "Hot" and "Cold" shutdown are defined in the TSs as operating modes or conditions, the restriction to perform certain surveillances during shutdown could be misinterpreted. The generic letter noted that if the performance of a refueling interval surveillance during plant operation would adversely affect safety, the licensee should postpone the surveillance until the plant is shutdown for refueling or in a condition or mode consistent with safe conduct of that surveillance. In the application, the licensee stated that they agreed with this position. Deletion of the term "during shutdown" is in accordance with the recommendation in GL 91-04 and is acceptable.

2.2 Reactivity Control Systems-Position Indication System - Shutdown

2.2.1 Design

The existing Millstone Unit 3 Digital Rod Position Indication (DRPI) system measures rod position by use of two trains of coils mounted at discrete axial intervals around the control rod drive housing. As a rod transits the coil region, a perturbation is created in the electromagnetic flux generated by each coil and causes a change in the applied voltage to the coil. The voltage signal is converted to rod position which is indicated at the main control board DRPI panel and is also transmitted to the plant computer. Panel indication for each rod consists of one display card with 39 light emitting diodes (LEDs) arranged vertically. The 39 LEDs represent six-step intervals from rod at bottom (0) step to rod full out at six step intervals from rod at bottom to rod full out (228) steps.

2.2.2 Technical Specification Change

The Reactivity Control System - Position Indication System - Shutdown, Section 4.1.3.3 Surveillance Requirements of the Millstone Unit 3 TS states "Each of the above required digital rod position indicator(s) shall be determined to be

OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over their full-range of rod travel at least once per 18 months." The licensee proposes to change the surveillance interval from 18 months to each Refueling Interval.

2.2.3 Justification for the Change

The licensee evaluated equipment performance over four operating cycles that included a review of surveillance results, preventive maintenance records, and frequency and type of corrective maintenance and found that the DRPI system performance was within expected bounds. No major corrective or preventive maintenance activities were performed on the DRPI system. A random failure identified in a rod deviation card during the last Millstone Unit 3 operating cycle (June 19, 1994) did not indicate a recurring problem and did not adversely impact the performance assumptions used to support the proposed refueling extension. The staff reviewed the above failure as reported in Licensee Event Report (LER) 94-009 issued on August 2, 1994, and a similar failure at Millstone Unit 3 that occurred on February 3, 1988, reported in LER 88-007 issued on March 4, 1988, and agrees with the licensee's conclusion that these failures are rare and random, and do not adversely impact on the proposed refueling interval surveillance extension. The staff's review of the NRC's records did not reveal any other DRPI system failures at Millstone Unit 3. Based on the above, the staff finds the proposed change in TS surveillance frequency from 18 to 24 months to be acceptable.

2.3 Reactivity Control - Rod Drop Time

2.3.1 Design

The measure of control rod drop time is made by connecting the existing Millstone Unit 3 Automatic Rod Drop Test Cart (ARDTC) to the DRPI system and Control Rod Drive System in accordance with approved plant procedures. The ARDTC is a microprocessor-based system which is used to unlatch a preselected group of rods and measure the rod drop times.

2.3.2 Technical Specification Change

Millstone Unit 3 TS Surveillance Requirement 4.1.3.4.c states

"The rod drop time for the full-length control rods be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following maintenance on or modification to the Control Rod Drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months."

The licensee is proposing to extend the frequency of Surveillance Requirement 4.1.3.4.c from at least once every 18 months to at least once each refueling interval (i.e., 24 months).

2.3.3 Justification for the Change

The licensee has evaluated the control rod drive system equipment performance over four operating cycles that included review of surveillance results, preventive maintenance records, and frequency and type of corrective maintenance and found that the change to the frequency of surveillance for rod drop time required by Surveillance Requirement 4.1.3.4.c will not degrade the ability of the control rods to perform their safety function. The surveillance results indicated that for each test conducted to verify that rod drop time was in compliance with Surveillance Requirement 4.1.3.4.c, the results have been within the acceptance criterion of less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry. The staff's review of NRC records did not identify any Millstone Unit 3 events concerning unacceptable control rod drop times. Based on the above, the staff finds the proposed change in the frequency of TS Surveillance Requirement 4.1.3.4.c from 18 to 24 months to be acceptable.

2.4 Seismic Monitoring Instrumentation

2.4.1 Design

The existing Millstone Unit 3 seismic monitoring instrumentation system is nonsafety related and uses both mechanical and electronic equipment to detect and record the amplitude (acceleration) and frequency of a seismic event. It performs no automatic safety functions. The installed systems comply with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes (and ANSI/ANS-2.2-1978 Earthquake Instrument Criteria For Nuclear Power Plants)." The seismic information is measured and recorded and can be compared to the design basis requirements of structures, systems and components of Millstone Unit 3 to determine whether the design basis has been exceeded. The system functions automatically upon the detection of a seismic event and is used by plant operators to determine conditions which could be limiting with regard to continued plant operations following a seismic event and/or to restart following a seismic event. The following are the seismic monitoring instruments and their location at the Millstone Unit 3 facility:

a. Triaxial Time-History Accelerographs

- Containment Mat
- Containment Wall
- Emergency Diesel Generator Mat in Diesel Fuel Oil Vault

- Aux. Building Wall near the charging pump cooling surge tank

b. Triaxial Peak Accelerographs

- Containment Safety Injection Accumulator Tank
- Safety Injection Accumulator Discharge Line
- Aux Building Charging Pumps Cooling Surge Tank

c. Triaxial Seismic Trigger

- Control Room - Horizonal
  .01g * and .09g **
- Control Room - Vertical
  .006g * and .06g **

d. Triaxial Response-Spectrum Recorders

- Control Room - Spectrum Analyzer *
- Steam Generator Support - Self-Contained Recorder

* Unit activated by signal from Triaxial Accelerograph located at the Containment mat.

** Unit activated by signal from Triaxial Accelerograph located at the Containment mat and is connected to reactor control room annunciator.

The triaxial peak recording accelerographs (b) are used to provide qualitative seismic motion data to compare against analog seismic instrumentation and are considered to be the lowest order with respect to the level of data reliability when compared to the time history accelerographs (a) and the response spectrum recorders (d).

2.4.2 <u>Technical Specification Change</u>

Seismic Instrumentation Surveillance Requirement 4.3.3.3.1 of Millstone Unit 3 Technical Specifications states that "Each of the above required seismic monitoring instruments [shown in Table 3.3-7] shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATION TEST at the frequencies shown in Table 4.3-4." Table 4.3-4 lists the instrument channel calibration as R which the licensee proposes to change from "At least once per 18 months" to "At least once per 24 months".

2.4.3 Justification for the Change

The licensee has reviewed the seismic equipment surveillance, preventative and corrective maintenance records over the last four fuel cycles. The review indicated that the seismic instruments responded as required when tested with approved procedures and no significant corrective maintenance activities were performed on this equipment. The licensee also stated that additional assurance of the operability of the seismic monitoring instrumentation system is provided by the monthly channel check and the semi-annual analog channel operational test during power operations for those instruments identified in Table 4.3-4 except for the Triaxial Peak Accelerographs and the Triaxial Self-Contained Recorder at the Steam Generator Support. The latter instruments are calibrated during a refueling outage and are only used to provide qualitative seismic motion data for comparison against analog seismic instrumentation.

The staff's review of NRC records identified one Millstone Unit 3 event concerning an incorrect range for the Triaxial Peak Accelerograph-Safety Injection Accumulator Discharge Line. The incorrect range was identified on September 5, 1991, during a calibration documentation review as part of the investigation into the reliability of the Triaxial Peak Accelerographs. This event is documented in LER 91-024 issued on October 7, 1991, and LER 91-024-01 issued on December 31, 1991. A Special Report, MP-91-756, dated September 25, 1991, was sent by the licensee to the NRC, entitled "Millstone Nuclear Power Station, Unit No. 3 Inoperable Seismic Monitoring Instrumentation." The instrument was installed as a replacement on March 18, 1987. The TS Table 3.3-7 listed a measurement range of ± 1g and the installed instrument had a range of ± 2g. An instrument with a range of ± 1g was installed on September 6, 1991, to comply with the TS. However, further review indicated that an instrument with a range of ± 2g is more suitable. Therefore, on November 7, 1991, an instrument with a range of ± 2g was installed and a TS change was submitted. This range is in accordance with the current TS Table 3.3-7. This occurrence has no effect on the proposed surveillance test frequency extension as the range of the instrument does not impact surveillance frequency. Based on the above, the staff finds the proposed change in the frequency of TS Surveillance Requirement 4.3.3.3.1 from 18 to 24 months to be acceptable.

2.5 Loose-Part Detection Instrumentation System

2.5.1 Design

The primary purpose of the existing Millstone Unit 3 loose-part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate damage to or malfunctions of safety-related primary system components. The loose-part detection (monitoring) system (LPMS) is an impact monitoring system which functions by detecting the acceleration (vibration) caused by the impact of foreign objects (failed or weakened components or an item inadvertently left in the primary system during refueling or maintenance) on the reactor vessel internal structure or on associated piping. Regulatory Guide 1.33 "Loose-Part

Detection Program" recommends a system capable of automatically detecting loose parts that weigh between 0.25 and 30 pounds and impact with an energy of 0.5 ft-lbs or more on the inside surface of the reactor coolant pressure boundary within 3 feet of a sensor. The LPMS is a nonsafety-related system and is not credited in any design basis accident because it performs no automatic safety functions.

2.5.2 Technical Specification Change

Surveillance Requirement 4.3.3.8 of the Millstone Unit No. 3 TS states "Each channel of the Loose-Part Detection System shall be demonstrated operable by performance of:

a. a channel check at least once per 24 hours,
b. an analog channel operation test at least once per 31 days, and
c. A channel calibration at least once per 18 months."

The licensee proposes to change 4.3.3.8.c to remove "per 18 months" and replace with "each refueling interval" (i.e., 24 months).

2.5.3 Justification for the Change

The licensee has reviewed the LPMS equipment surveillance, preventative and corrective maintenance records over the last four fuel cycles. The review indicated that some failures have occurred, but none of these failures were attributed to instrument drift associated with calibration frequency. The failures were random component malfunctions and cable/wire degradation.

The staff reviewed failures associated with the LPMS addressed in the following reports:

- LER 87-010-00 dated April 3, 1987
- LER 87-010-01 dated February 10, 1988.
- Special Report to the NRC dated August 25, 1989 when two of twelve channels failed on July 16, 1989, and were declared inoperable due to continuous alarming.
- Special Report to the NRC dated November 15, 1994 for failures that occurred on October 24, 1994.
- Special Report to the NRC dated April 11, 1995 for failures that occurred on March 15, 1995.

The staff agrees with the licensee that the above failures were random component failures and cable/wiring degradation and are not related to instrument drift associated with calibration frequency. In each case, a redundant channel was available to detect loose-parts during the time the failed channel was inoperable.

Further assurance of system operability is provided by the TS channel check, conducted once per 24 hours and the analog channel operational test conducted once per 31 days. The licensee has scheduled replacement of the LPMS during the next refueling outage with an upgraded impact monitoring system similar in operation to the current system. Based on the above, the staff finds the proposed change in the frequency of TS Surveillance Requirement 4.3.3.8.c from 18 to 24 months to be acceptable.

2.6 Containment Quench Spray System and Recirculation Spray System

At Millstone Unit 3, the systems provided for containment heat removal consist of: 1) the quench spray system (QSS) and 2) the containment recirculation system (CRS). These systems are described in chapter 6.2.2 of the Final Safety Analysis Report (FSAR). The containment heat removal systems are designed to reduce the containment pressure following a break in either the primary or secondary piping system inside the containment. Heat is transferred from the containment atmosphere to the QSS and the recirculation spray system (RSS), which is a spray subsystem of the CRS. The spray water goes to the containment sump, where the CRS transfers the heat to the service water system via its heat exchangers. Additionally, the QSS, currently in conjunction with the spray additive system, is responsible for the removal of iodine from the containment atmosphere following a design basis accident (DBA) in containment.

The QSS consists of two 360° spray headers inside the containment that are fed via two full capacity pumps and automatic valves. The suction source for the QSS pumps is the refueling water storage tank (RWST). The pumps and automatic valves in the QSS are activated by a containment depressurization actuation (CDA) signal on high containment pressure. The QSS is capable of performing its intended safety function even with a single failure of an active component.

The CRS is comprised of two redundant subsystems. Each of these subsystems possess two 50 percent capacity coolers, two 50 percent capacity pumps, automatic isolation valves, and share two 360° spray headers. The four pumps take suction from a common containment sump, and pump water through the coolers, up the risers, to the spray headers. The two pumps in each subsystem are connected to different spray headers, but share the same emergency bus. Failure of one emergency bus will not prevent the delivery of sufficient containment recirculation flow, because only one subsystem would be lost. Each spray header is fed by two risers which take suction from one of the coolers in each of the subsystems.

The QSS and CRS are not normally operated during reactor operation. During normal operation, the QSS and CRS are dry. The systems are isolated and the pumps are on standby.

In the event of a loss-of-coolant accident or high energy line break within the containment, a CDA signal causes the motor-operated isolation valves in the QSS and RSS to open automatically, the QSS pumps to start automatically, and the RSS pumps to start automatically after a time delay.

Complete tests of these systems cannot be performed while the plant is operating, because a safety injection signal would cause a reactor trip, feedwater isolation, and containment isolation. Therefore, a piecemeal approach is taken to demonstrate operability of the containment spray subsystems. Normally, the system tests are conducted during refueling outages, and select components (i.e., motor-operated valves and pumps) are tested during operation. Additionally, the actuation logic for the containment spray subsystems is checked periodically during reactor operation.

3.6.2 Technical Specification Change

Surveillance Requirement 4.6.2.1.c currently requires that each Containment Quench Spray subsystem be demonstrated OPERABLE at least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal, and

2) Verifying that each spray pump starts automatically on a CDA test signal.

Surveillance Requirements 4.6.2.2.c and 4.6.2.2.d require that each Recirculation Spray System shall be demonstrated OPERABLE:

c. At least once per 18 months by verifying that on a CDA test signal, each recirculation spray pump starts automatically after a 660 $\pm$20 second delay;

d: At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a CDA test signal;

The licensee is proposing to change the frequency of these surveillances to at least once each refueling interval (i.e., 24 months). In addition, the phrase "during shutdown" in Surveillance Requirements 4.6.2.1.c and 4.6.2.2.d is being deleted to be consistent with the recommendation in GL 91-04. (See discussion in 2.1.3. above).

3.6.3 Justification for the Change

The licensee evaluated equipment performance over the last four operating cycles to determine the impact of extending the frequency of Surveillance Requirements 4.6.2.1.c, 4.6.2.2.c, and 4.6.2.2.d. This evaluation included a review of surveillance results, preventive maintenance records, and the frequency and type of corrective maintenance.

The reviews determined that no significant equipment failures for the QSS have occurred in the last four cycles. The automatic valves for the QSS have actuated as required and the QSS pumps have started automatically in response to a CDA test signal.

There have been two failures of RSS motor-operated valves to actuate in response to a CDA test signal during the tests conducted for the last four cycles. 3RSS*MOV23A failed to actuate during the October 1993 test, due to an improper wiring connection. 3RSS*MOV23B failed to stroke completely closed during the March 1991 test. 3RSS*MOV23B was tested satisfactorily after limit switch adjustments were performed. Additionally, during the October 1993 test, 3RSS*MOV23C actuated but the limit switches gave an incorrect position indication.

The RSS pumps have started as required, except during the tests conducted in June 1987. During the June 1987 tests, the CDA signal was reset prior to the RSS pumps being sequenced to test.

The only preventive maintenance that is scheduled on an 18-month frequency for the QSS and RSS are lubrication of the motor-operated valves, breaker maintenance, and hypot testing of the motors and cables. Extending the frequency for lubrication of the motor-operated valves is acceptable based on the surveillance history, the low frequency of operation, and the moderate ambient environmental conditions. Extending the maintenance interval for the breakers is acceptable, because the extensions will not result in any additional wear since the breakers are normally in the open position. Extending the frequency for the hypot testing is acceptable, because experience has shown a very low failure rate in general when testing at 18-month intervals and no failures in the RSS system.

Corrective maintenance performed on the QSS motor-operated valves involved minor packing, gasket, and seat leakage, position indication adjustments, and adjustments to valve motor operator tripper fingers. Also, there have been repairs to rusty pins in the actuator linkage of motor-operated valves located outdoors. For the RSS motor-operated valves, corrective maintenance has involved seat leaks, flange leaks, and limit and torque switch adjustments.

Corrective maintenance performed on the QSS and RSS pumps during the last four cycles involved minor leaks and oil level adjustments. In each of these cases, the appropriate repair was made. Also, there was one incident of high vibration on the "B" train QSS pump in May 1989. This vibration was determined to be due to improper greasing of the motor inboard bearing. The problem was resolved and the pump was retested satisfactorily.

Based on the engineering review of equipment performance, preventive, and corrective maintenance history and the availability of quarterly inservice testing, there is reasonable assurance that extending the surveillance intervals will not reduce the availability or capability of these systems to perform their intended functions, if needed. The proposed changes are acceptable.

3.7 Containment Isolation Valves

3.7.1 Design

The containment isolation system is described in chapter 6.2.4 of the FSAR. The containment isolation system isolates piping lines which penetrate the containment boundary to minimize the release of radioactive materials to the environment from postulated design basis accidents (DBA) within the containment. The valve arrangements ensure containment integrity, assuming a single failure, by providing at least two barriers between the atmosphere outside the containment and the atmosphere within the containment, the reactor coolant system, or systems that would become connected to the containment atmosphere or the reactor coolant system as result of, or subsequent to, a DBA.

3.7.2 Technical Specification Change

Surveillance Requirement 4.6.3.2 currently requires that each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position, and

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

The licensee proposes to change the surveillance interval from at least once per 18 months to at least once each refueling interval. The components covered by these surveillances are shown on Table 6.2-65 of the FSAR.

3.7.3 Justification for the Change

Equipment performance over the last four operating cycles was evaluated to determine the impact of extending the frequency of Surveillance Requirement 4.6.3.2. This evaluation included a review of surveillance results, preventive maintenance records, and the frequency and type of corrective maintenance.

During the last four operating cycles, six surveillances have been performed on containment isolation valves that actuate in response to a Phase A isolation signal, and five surveillances have been performed on containment isolation valves that actuate to a Phase B isolation signal. In these tests, only three failures of the valves to actuate to their design position have occurred. Valve 3SSR*CTV32 (solenoid-operated, globe valve used to isolate a 3/4" safety injection accumulator sample line) failed during the test conducted in May 1988 and valves 3RSS*MOV23A and 3RSS*MOV23B (motor-operated, butterfly valves used to isolate a 12" containment recirculation pump suction line) failed during the test conducted in October 1993 and March 1991, respectively. Given the number of tests, the reliability of the containment isolation valves is considered high.

Valve 3SSR*CTV32 is a 3/4" valve in the reactor plant sampling (SSR) system. The valves in the SSR system are often affected by boron precipitation due to small clearances. When failure occurs these valves are replaced.

The failure of valve 3RSS*MOV23A to actuate was attributed to a blown fuse on the secondary side of the control power transformer. The valve was replaced during the fourth refueling outage due to excessive seat leakage. During valve installation, the wire in the limit switch was pinched and grounded. This resulted in a fuse blowing during the valve actuation test. 3RSS*MOV23B failed to stroke completely closed during the March 1991 test. The valve was tested satisfactorily after the limit switch adjustments were performed.

There are other TS requirements, such as the quarterly inservice testing of these valves and the monthly automatic actuation logic tests that also demonstrate the operability of containment isolation valves.

Based on the maintenance and performance history, the containment isolation valves are highly reliable. There is reasonable assurance that extending the frequency of Surveillance Requirement 4.6.3.2 will not result in a deterioration in valve condition or performance. The proposed TS change is acceptable.

3.8 Bases

The Bases for Specification 4.0.2 discusses the extension of the time interval for surveillance requirements. The paragraph currently has a sentence which states that "it also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18 month surveillance interval." The licensee proposed to substitute the sentence that "it also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each refueling interval." The proposed change to Bases 4.0.2 is acceptable.

3.9 Probabilistic Risk Assessment (PRA)

As discussed above, the licensee performed a comprehensive safety assessment of the proposed changes to the TSs based on past performance and the maintenance history of the components. Using the same deterministic approach, the NRC staff has determined that the changes are supported by existing failure data and are acceptable.

Northeast Nuclear Energy Company (NNECO) also supported the proposed TS changes with a probabilistic safety assessment. In response to Generic Letter

(GL) 88-20, NNECO submitted an Individual Plant Examination (IPE) on August 31, 1990. The basis for the licensee's IPE was a 1983 full-scope Level 3 Probabilistic Safety Study (PSS) (which had been reviewed by the NRC) that had been periodically updated. The PSS contained a full range of both internal and external event probalistic safety assessment (PSA) models. The NRC's staff evaluation of the IPE was transmitted to NNECO by letter dated May 5, 1992. The licensee's estimated core damage frequency (CDF) from postulated internal events was 5.6E-5, which was about average at the time for Westinghouse 4-loop plants. There were no significant severe accident vulnerabilities identified.

Since the IPE submittal, the licensee has performed a major update to the PRA to reflect various plant modifications, improved procedures, revisions to the training provided to plant staff and increased use of plant specific data. For example, the addition of a third air cooled diesel significantly reduced the contribution from postulated loss of offsite power and station blackout scenarios. On the other hand, a reassessment of the loss of service water as an initiator indicated that the implications of this support system might be a more significant contributor than originally estimated. While the order of some of the dominant accident sequences has changed as the PRA has been updated with time, the significant insights have not been greatly affected.

NNECO PRA personnel interact with engineering and operations personnel to assess the potential impacts of significant design and/or operational changes on the PRA result. In the May 1, 1995, submittal the licensee discussed the possible effect of the proposed TS changes on the PRA models.

With respect to the proposed TS changes to the Quench Spray System (QSS) and the Recirculation Spray System (RSS), the licensee noted that the Millstone Unit 3 PRA models the QSS and RSS systems. The proposed changes to the surveillance frequency has no effect on the PRA availability models for the subject systems. The quarterly pump starts are credited in determining the pump failure to start probability. The quarterly valve tests are credited in determining the motor-operated valve failure to open or close probabilities. Thus, the system component failure probabilities are not affected by the proposed changes. The availability model of the engineered safety feature actuation system for containment depressurization actuation (CDA) component actuation is unaffected by the 24-month fuel cycle, since the constituent components (i.e, bistables, logic circuits, output relays) are tested more frequently.

The licensee also assessed what the proposed TS change to the Reactor Coolant System (RCS) boration flowpath test intervals (4.1.2.2.c and 4.1.2.2.d) would have on the Millstone Unit 3 PRA using rather pessimistic event trees.

To quantify the effect, the fault exposure factor (FEF) of numerous component basic events were revised from six to eight to reflect the change to a 24-month fuel cycle (this assumes component demand failures are linear with surveillance interval). Additionally, the fault factors of certain common cause basic events were revised.

The revisions have the following effect on the listed functions:

1. Charging pump unavailability for Safety Injection: 7 percent increase

2. Charging Pump unavailability for Sump Recirculation: 23 percent increase

3. RCS Emergency Boration unavailability for ATWS: 6 percent increase

The changes in the charging pump and the emergency boration unavailabilities are expected to result in a core melt frequency increase of approximately 1 percent. This change is considered insignificant.

The PRA groups evaluated the other proposed TS changes. The digital rod position indicators do not have an accident mitigation function and thus have a negligible effect on plant risk. Extending the frequency for demonstrating the rod drop time likewise has a negligible effect. The seismic and loose-part detection system are instrumentation non-safety related systems that do not play an active role in accident mitigation and thus changing the surveillance frequency would not be expected to have an effect on the CDF.

The probabilistic safety assessment of the proposed changes to the TSs fully supports and complements the deterministic assessment. The changes in surveillance frequencies from 18 to 24 months result in no significant reduction in the margin of safety and are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The NRC 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. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 58402). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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