December 23, 1992

Docket No. 50-336

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Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M84774)

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your application dated October 28, 1992, as supplemented November 20, 1992, and December 4, 1992.

The amendment incorporates into the Technical Specifications changes in the area of Tables 3.3-3, 3.3-4, 3.3-5 and 4.3-2, of Section 4.8.1.1.2 and of the Bases Section 3/4.3, to add the high containment pressure signal as an input to main steam isolation (MSI) and to reduce the feed isolation portion of MSI from 60 seconds to 14 seconds.

We note your commitment to request a technical specification change to provide a surveillance requirement to perform an emergency diesel generator start on safety injection actuation signal.

A copy of the related Safety Evaluation is enclosed. Also enclosed is the notice of issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by
D. Jaffe for

Guy S. Vissing, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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Enclosures:
1. Amendment No. 167 to DPR-65
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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DATE	12/9/92	12/9/92	12/10/92	12/15/92	/ /

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Mr. John F. Opeka
Northeast Nuclear Energy Company

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Unit 2

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UNITED STATES
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WASHINGTON, D. C. 20555

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-65

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 October 28, 1992, as supplemented November 20, 1992, and December 4, 1992, 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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 167, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: December 23, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following page of the Appendix A Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
3/4 3-13	3/4 3-13
3/4 3-18	3/4 3-18
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 8-2	3/4 8-2
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

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

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	4.75 psig	≤ 5.20 psig
c. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1592.5 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 9.48 psig	≤ 10.11 psig
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 4.75 psig	≤ 5.20 psig
d. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1592.5 psia
4. MAIN STEAM LINE ISOLATION		
a. Containment Pressure - High	≤ 4.75 psig	≤ 5.20 psig
b. Steam Generator Pressure - Low	≥ 500 psia	≥ 492.5 psia

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation	Not Applicable
Enclosure Building Filtration System	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. SRAS Containment Sump Recirculation	Not Applicable
e. EBFAS	
Enclosure Building Filtration System	Not Applicable
f. Auxiliary Feedwater Initiation	Not Applicable
g. Main Steam Isolation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	
1) High Pressure Safety Injection	$\leq$ 25.0*/5.0**
2) Low Pressure Safety Injection	$\leq$ 45.0*/5.0**
3) Charging Pumps	$\leq$ 35.0*/35.0**
4) Containment Air Recirculation System	$\leq$ 26.0*/15.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 45.0*/45.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Containment Pressure - High	
a. Safety Injection (ECCS)	
1) High Pressure Safety Injection	≤ 25.0*/5.0**
2) Low Pressure Safety Injection	≤ 45.0*/5.0**
3) Charging Pumps	≤ 35.0*/35.0**
4) Containment Air Recirculation System	≤ 26.0*/15.0**
b. Containment Isolation	≤ 7.5
c. Enclosure Building Filtration System	≤ 45.0*/45.0**
d. Main Steam Isolation	≤ 6.9
e. Feedwater Isolation	≤ 14
4. Containment Pressure--High-High	
a. Containment Spray	≤ 35.6*(1)/16.0**(1)
5. Containment Radiation-High	
a. Containment Purge Valves Isolation	≤ Counting period plus 7.5
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 6.9
b. Feedwater Isolation	≤ 14
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 120
8. Steam Generator Level-Low	
a. Auxiliary Feedwater System	≤ 240*/240**(2)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included.

** Diesel generator starting and sequence loading delays not included. Offsite power available.

(1) Header fill time not included.

(2) Includes 3-minute time delay.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

ACTION (Continued)

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.2 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 4 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the switchyard shall be determined OPERABLE at least once per 24 hours by verifying correct breaker alignments and indicated power availability.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the fuel oil supply tank,

2. Verifying the diesel starts from ambient condition and accelerates to $\geq$ 90% of rated speed and to $\geq$ 97% of rated voltage in $\leq$ 15 seconds.

3. Verifying the generator is synchronized, loaded to $\geq$ 1300 kw in $\leq$ 60 seconds, and operates for $\geq$ 60 minutes.

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

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

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

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 measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

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 times.

The containment spray response time with a loss of normal power assumes that the LNP occurs simultaneously with the CSAS. Therefore, the valve stroke time is bounded by the time required for signal generation, diesel start, sequencer, and time for the spray pumps to reach operating speed.

The containment spray response time without a loss of power is composed of signal generation and valve stroke time.

CAR fan response time is determined for the idle fan and conservatively applied to all four. For the case with a loss of power, signal generation, diesel start, sequencer and the time for the fans to reach operating speed bounds valve stroke time.

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

CAR fan response time for the case without a loss of power is composed of signal generation and valves stroke time.

Feedwater isolation response time ensures a rapid isolation of feed flow to the steam generators via the feedwater regulating valves, feedwater bypass valves and, as backup, feed pump discharge valves. The response time includes signal generation time and valve stroke. Feed line block valves also receive a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Since the block valves are not credited with isolation, they are not required to operate as fast as the isolation valves although equal response times for all valves are specified.

The containment airborne radioactivity monitors (gaseous and particulate) are provided to initiate closure of the containment purge valves upon detection of high radioactivity levels in the containment. Closure of these valves prevents excessive amounts of radioactivity from being released to the environs in the event of an accident.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated October 28, 1992, as supplemented November 20, 1992, and December 4, 1992, Northeast Nuclear Energy Company (the licensee) submitted an application for amendment of the operating license for Unit 2 at the Millstone Nuclear Power Station (Ref. 1). The proposed amendment would change the technical specifications to permit the licensee to modify instrumentation and adjust setpoints to provide additional assurance that a main steam line break accident would not cause pressure and temperature in containment to exceed values for containment design and equipment qualification.

2.0 BACKGROUND

In 1979, the licensee analyzed the main steam line break (MSLB) accident to demonstrate that the containment would withstand an MSLB accident. The analysis assumed that the reactor was at hot zero power, that the break was double-ended and was located between the steam generator and the inboard main steam isolation valve (MSIV), and that an emergency diesel generator failed to operate resulting in failure of one of two containment spray pumps and two of four containment air recirculation fans. The analysis demonstrated that the pressure and temperature in containment would be less than the design pressure and temperature for the containment.

On February 8, 1980, the staff issued Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition." The bulletin requested that licensees review their analyses of the MSLB to determine whether continuation of main feedwater, auxiliary feedwater, or condensate flow would adversely affect pressure in containment. The staff, with assistance from Franklin Research Center, reviewed the licensee's response and concluded that there is no potential for overpressurizing containment because the reactor would be at hot zero-power, the main feedwater would be isolated from the steam generators, and the initiation of auxiliary feedwater flow to the affected steam generator would be delayed. The staff concluded that the licensee's response was acceptable (Ref. 2).

On October 18, 1991, while the unit was operating at full power, and during planning for replacement of the steam generators, the licensee determined that the existing MSLB accident analysis was not conservative with respect to containment response. The licensee notified the NRC and reduced reactor power to 3% (Ref. 3). These actions were based on an analysis by the licensee that assumed that an MSLB occurs while the reactor is operating at full power, that one of two main feedwater regulating valves fails to close automatically, and that an operator closes the valve after 10 minutes (Ref. 4). The licensee stated that the analysis indicated that pressure and temperature in the containment would reach 92 psig and 427°F because of the flow of feedwater to the containment through the failed main steam line. The design pressure and temperature for the containment are 54 psig and 289°F (Ref. 3).

The licensee prepared a justification for continued operation (Ref. 4) and the unit was returned to power. Initially, the licensee stationed a dedicated operator at the controls for the main feedwater block valves with instructions to close them if the reactor shut down automatically. Later, the licensee modified the main feedwater block valves to close automatically on a containment isolation signal and thus precluded the need for a dedicated operator.

On August 4, 1992, the licensee found two more conditions which would cause pressure and temperature to exceed the design values for the containment during an MSLB accident. One of the conditions was failure of the feedwater regulating bypass valve to stop the flow of feedwater to a steam generator. The other was failure of the vital buses to fast transfer to the reserve station services transformer. If this occurred, power would not be available to close the feedwater regulating valves and start the containment pressure control systems until the emergency diesel generators started and loaded (Ref. 4).

To provide permanent solutions to these problems, the licensee modified the facility as permitted by 10 CFR 50.59, proposed changes to the technical specifications as described in their application dated October 28, 1992 for amendment to the operating license, and provided additional information to support their application on November 20 and December 4, 1992 (Ref. 5 and 6).

3.0 <u>CHANGES TO THE FACILITY</u>

3.1 Engineered Safety Features Actuation System

Prior to modification of the engineered safety features actuation system, a main steam isolation signal was generated when the output of two of four steam generator pressure channels dropped to the low pressure setpoint. The system was modified to connect four containment pressure channels to the system. These channels generate a main steam isolation signal when the outputs of two of the channels exceed the high pressure setpoint.

3.2 Main Feedwater System

Prior to modification of the control circuitry for the main feedwater system, single main steam isolation signals were connected to the pumps, the block valves, the regulating valves, and the bypass regulating valves for the main feedwater system. Redundant main steam isolation signals were added to each of these components. Redundant main steam isolation signals were also added to the pump discharge valves which isolate main and main bypass feedwater on closure. In addition, the source of control power for the main feedwater regulating valves was changed from normal power to vital power with battery backup.

3.3 Emergency Diesel Generators (EDGs)

Prior to modification of the control circuitry for the EDGs, the diesel engines started automatically only on initiation of a loss of normal power signal. The control circuitry has been modified to add automatic starting on initiation of a safety injection actuation signal (SIAS).

4.0 PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

4.1 Isolation of the Main Steam System

An existing limiting condition for operation (LCO) requires that three of four steam generator pressure channels be operable when the reactor is in modes 1, 2, or 3 and that a main steam isolation signal be initiated when any two of the channel outputs reach the setpoint. The proposed change to the technical specifications would add an LCO setting forth similar requirements for the containment pressure channels. The proposed change would also include an LCO requiring that the setpoint for these channels be $\leq$ 4.75 psig and the allowable value be $\leq$ 5.20 psig. In addition, the proposed change to the technical specifications would require the same periodic surveillance for the containment pressure instrumentation which is now required for the steam generator pressure instrumentation.

4.2 Isolation of the Main Feedwater System

For main feedwater isolation from a steam generation low pressure signal, an existing LCO requires that the response time, including signal generation and valve closure, be $\leq$ 60 sec. The proposed change to the technical specifications would require that the response time be $\leq$ 14 sec.

4.3 Actuation of the Containment Spray System

Existing LCOs require that the containment spray system actuate automatically when containment pressure is ≤ 27 psig with an allowable value that is ≤ 27.45 psig and a response time that is ≤ 35.6 sec with or without normal power available. The proposed change to the technical specifications would reduce the actuation setpoint to ≤ 9.48 and the allowable value to ≤ 10.11 psig.

Further, the proposed changes to the technical specifications would reduce the containment spray response time from ≤ 35.6 sec to ≤ 16 sec with normal power available.

4.4 Actuation of the Containment Air Recirculation System

Existing LCOs for the response time of the containment air recirculation system to a high containment pressure signal or a low pressurizer pressure are ≤ 31 sec with or without normal power available. The proposed change to the technical specifications would change this value to ≤ 15 sec with normal power available and to ≤ 26 sec without normal power available.

4.5 Actuation of the Charging Pumps and the Safety Injection System

Existing LCOs for the response time of the charging pumps to a low pressurizer pressure signal or a high containment pressure signal is ≤ 40 sec with or without normal power available. The proposed changes to the technical specification would change the response time to ≤ 35 sec with or without normal power available.

Existing LCOs the response time of the high and low pressure safety injection systems to a low pressurizer pressure signal or a high containment pressure signal are ≤ 30 sec and ≤ 50 sec respectively without normal power available. The proposed changes to the technical specification would change the response times to ≤ 25 sec and ≤ 45 sec.

4.6 Actuation of the Enclosure Building Filtration System

Existing LCOs for the response time of the enclosure building filtration system to a high containment pressure signal or a low pressurizer pressure signal is ≤ 50 sec with or without normal power available. The proposed changes to the technical specification would change the response time to ≤ 45 sec with or without normal power available.

4.7 Startup of the EDGs

Existing LCOs require that the EDGs achieve ≥ 97 percent of rated voltage in ≤ 20 sec. The proposed change to the technical specifications would reduce that value to ≤ 15 sec.

5.0 EVALUATION

5.1 Containment and Safety-related Equipment

If an MSLB accident were to occur, two main steam isolation signals would be initiated. Each signal would demand closure of the MSIVs, the main feedwater regulation and block valves, and the main feedwater bypass regulation valve in one train of the steam generation system. If the accident were to occur at full power and if a main steam isolation signal were not generated for the steam generator with the failed steam line, then feedwater would flow to the containment through the broken steam line until the main feedwater was isolated. If an operator were to fail to isolate main feedwater promptly, pressure and temperature in containment would exceed the design values. To assure that this does not occur, the licensee provided redundant main steam isolation signals to both main feedwater trains. The licensee has also provided redundant main steam isolation signals to the main feedwater pump discharge valves which previously did not receive a main steam isolation signal. Further, the licensee has modified the engineered safety features actuation system to generate main steam isolation signals on high containment pressure as well as low steam generator pressure. To reduce the challenge to containment, the licensee will assure that: (a) the feedwater isolation response time is reduced by a factor of 4.2, (b) containment gauge pressure at which containment spray is initiated is reduced by a factor of 2.7, and (c) the EDGs are started 5 sec earlier.

With these changes assumed to be in place, the licensee analyzed the MSLB accident with various single failures. For the cases analyzed, the maximum peak pressure in the containment was 53.7 psig and the maximum peak temporal and spatial temperature in the containment atmosphere was 425.4°F (Ref. 6). Because of the difference in masses and thermal capacities of the containment atmosphere and the components and materials contiguous with the containment atmosphere, it would cool rapidly after the discharge of feedwater and steam is stopped, and components and materials contiguous with the containment atmosphere would heat up slowly. With the containment atmosphere at saturation, the expected temperature and pressure are estimated to be 285°F (Ref. 3) and 38.5 psig. The safety-related equipment in containment is environmentally qualified for temperatures up to 289°F. Containment response and Equipment Qualification issues will be evaluated in response to a separate submittal.

The staff also looked at the potential for the changes to the facility and the proposed changes to the technical specifications to increase the probability of other accidents which were previously evaluated and the probability of a malfunction of equipment important to safety. The licensee indicated that the probability of occurrence of loss of load and an MSIV closure type of event could be affected (Ref. 1). Since the high containment pressure signal is two out of four logic, the impact on the probability of an inadvertent MSIV closure is negligible. Also, since the probability of an inadvertent main steam isolation or an SIAS signal is not significantly increased by the proposed plant changes, therefore, any resultant damage to or wear of equipment important to safety, actuated by these signals, would not be significantly affected.

The staff has evaluated the licensee's assessment and finds that there will be no significant change in the probability of an accident previously evaluated or the probability of occurrence of a malfunction of equipment affected by the proposed plant changes. The staff concludes that the changes to the facility and the proposed changes to the technical specifications will provide assurance that challenges to containment and qualification of equipment within containment will not exceed design values for pressure and temperature.

5.2 Core

The licensee has provided its assessment of the effect of the proposed changes described in Sections 3.0 and 4.0 of this evaluation on the core response to an MSLB or a loss-of-coolant accident (LOCA) (Ref. 6). For the MSLB, the following cases were analyzed: (a) hot zero power (HZP) with normal power available, (b) HZP without normal power available, (c) hot full power (HFP) with normal power available, and (d) HFP without normal power available. The licensee concluded that the proposed changes that could impact the MSLB cases are beneficial and, thus, the existing analysis in the docket remains bounding. For the LOCA analysis, the licensee has evaluated the effect of each of the proposed changes on the small break and large break LOCAs and has concluded that those changes are either bounded by the assumptions used in the existing analysis or cause only a negligible effect to the existing analysis. The staff has evaluated the licensee's assessment and finds that the LOCA analysis of record is not affected by the proposed plant changes and does not require reanalysis.

5.3 EDGs

The original design for the EDGs included a requirement to start automatically on either an SIAS or loss of normal power (LNP) signal. The EDG start time was 20 seconds. On an SIAS signal the EDGs start automatically and run but would not load until an LNP signal was received. The licensee determined that if the EDGs operated unloaded, this could cause undesirable conditions and could render the EDGs inoperable and unavailable. As a result the licensee elected to modify EDG initiation logic to remove the starting of the EDGs on the SIAS signal. The starting of the EDG on a LNP signal was not changed.

After the MSLB was reanalyzed, the licensee determined that the EDG start time should be reduced by 5 seconds and that the EDGs should start on an SIAS signal allowing an earlier response time for the containment spray and containment air recirculation systems.

To meet the MSLB reanalysis, the licensee has proposed to start the EDGs on an SIAS signal per the original design. The licensee has reevaluated the EDG actuation on an SIAS signal and has concluded that operation of an EDG unloaded will not adversely affect the reliability or availability of the EDG. An EDG start on an SIAS signal without loss of normal power would require the

EDG to run unloaded for a short period of time and according to the manufacturer it would not cause any adverse or undesirable condition. The licensee has implemented procedures from the manufacturer that after the EDG is operated under no-load or light load conditions the diesel generator shall be run at $\geq$ 75 percent load (2076 KW) for at least 2 hours before shutdown. This will prevent fouling or other damage to the diesel generator. In addition, the licensee will test the SIAS signal prior to operation and has committed to propose an additional change to the technical specifications to include a surveillance requirement for the SIAS signal every 18 months.

Additionally, the licensee has also evaluated the capability of the EDG to start within 15 seconds and concluded that the EDG has the necessary capability. Therefore, the licensee proposed the change to the technical specifications to include the 15 second start time as part of the surveillance requirements.

The staff finds the licensee's proposed changes to start on an SIAS signal and to reduce the start time for the EDGs by 5 seconds to be acceptable.

6.0 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.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on December 23, 1992 (57 FR 61101). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

8.0 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.

9.0 REFERENCES

(1) Northeast Nuclear Energy Company, letter to NRC, Docket 50-336, "Proposed Revision to Technical Specifications, Main Steam Line Break Design Limits," dated October 28, 1992.

(2) NRC, letter to Northeast Nuclear Energy Company, Docket 50-336, "Resolution of Main Steam Line Break with Continued Feedwater Addition Event for Millstone Nuclear Power Station, Unit No. 2," dated October 7, 1982.

(3) Northeast Nuclear Energy Company, Docket 50-336, Licensee Event Report 91-010-02, dated September 2, 1992.

(4) Northeast Nuclear Energy Company, "Millstone Unit No. 2, Justification for Continued Operation #2-91-1, Main Steam Line Break Inside Containment," to be implemented 10/21/91. Docketed on November 24, 1992.

(5) Northeast Nuclear Energy Company, letter to NRC, Docket 50-336, "Proposed Revision to Technical Specifications, Main Steam Line Break Design Limits, Response to Request for Additional Information," dated November 20, 1992.

(6) Northeast Nuclear Energy Company, letter to NRC, Docket 50-336, "Proposed Revision to Technical Specifications, Main Steam Line Break Design Limits, Response to Request for Additional Information," dated December 4, 1992.

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Date: December 23, 1992

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 167 to Facility Operating License No. DPR-65 issued to Northeast Nuclear Energy Company (the licensee), which revised the Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 located in New London County, Connecticut. The amendment is effective as of the date of issuance.

The amendment incorporates into the Technical Specifications changes in the area of Tables 3.3-3, 3.3-4, 3.3-5 and 4.3-2, of Section 4.8.1.1.2 and of the Bases Section 3/4.3, to add the high containment pressure signal as an input to main steam isolation (MSI) and to reduce the feed isolation portion of MSI from 60 seconds to 14 seconds.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on November 10, 1992 (57 FR 53518). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (57 FR 61101).

For further details with respect to the action see (1) the application for amendment dated October 28, 1992, as supplemented November 20, 1992 and December 4, 1992, (2) Amendment No. 167 to License No. DPR-65, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555 and at the local public document room located at the Learning Resources Center, Thames Valley State Technical College, 574 New London Turnpike, Norwich, Connecticut 06360. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland this 23rd day of December 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

David H. Jaffe

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