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March 1, 1979

Docket No.: 50-336

Mr. W. G. Counsil, Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated August 15, 1978.

The Technical Specifications being changed are for the engineered safety feature actuation system and radiation monitoring instrumentation. The changes concern the alarm/trip setpoints, the minimum channels operable and the action statements for the following instruments:

- containment airborne radioactivity monitor,
- spent fuel storage area monitors, and
- spent fuel storage area radioactivity monitor.

Some portions of your proposed Technical Specifications have been modified to meet your own or our requirements. These modifications have been discussed with and agreed to by your staff.

We understand, from your January 3, 1979 response to our November 29, 1978 letter, that it is your intention to provide justification for unlimited containment purging by May 15, 1979. We further acknowledge your commitment to limit purging to an absolute minimum, not to exceed 90 hours per year, pending completion of our review of your justification for unlimited containment purging.

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CP 1
GD

OFFICE➤						
SURNAME➤						
DATE➤						

Northeast Nuclear Energy
Company

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 49
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

STSG Brinkman 1/31/79

Read + understood

Concurred in Federal Register Notice only

3/1/79

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DATE	1/29/79	1/29/79	1/31/79	2/21/79	3/1/79	2/27/79

Northeast Nuclear Energy Company

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cc w/enclosure(s) and incoming
dtd.: 8/15/78

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ATTN: Assistant Director, Research
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY,
WESTERN MASSACHUSETTS ELECTRIC COMPANY, AND
NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 49
License No. DPR-65

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

   A. The application for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company and Northeast Nuclear Energy Company (the licensees) dated August 15, 1978, 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.

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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 Appendices A and B, as revised through Amendment No. 49, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 49

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

IX
3/4 3-11
3/4 3-19
3/4 3-27
3/4 3-29
B 3/4 3-1
B 3/4 3-2
B 3/4 3-3
B 3/4 3-4 (added)

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BASES

BASES

SECTION | PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

SURVEILLANCE REQUIREMENTS (Continued)

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

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure - High	4	2	3	1, 2, 3	7
c. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	7
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	7
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	6
c. Containment Pressure - High	4	2	3	1, 2, 3	7
d. Pressurizer Pressure - Low	4	2	3	1, 2(e), 3(a)	7

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. ENCLOSURE BUILDING FILTRATION (EBFAS)		
a. Manual EBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 5 psig	≤ 5 psig
d. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1600 psia
6. CONTAINMENT SUMP RECIRCULATION (SRAS)		
a. Manual SRAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	48 ± 18 inches above tank bottom	48 ± 18 inches above tank bottom
7. CONTAINMENT PURGE VALVES ISOLATION		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Automatic CIAS Actuation Logic	Not Applicable	Not Applicable
d. Containment Radiation - High		
Gaseous Activity	≤ the value determined in accordance with Specification 4.3.2.1.4	≤ the value determined in accordance with Specification 4.3.2.1.4.
Particulate Activity (Half Lives greater than 8 days)	≤ the value determined in accordance with Specification 4.3.2.1.4.	≤ the value determined in accordance with Specification 4.3.2.1.4.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
8. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level one	$\geq$ 2912 volts	$\geq$ 2912 volts
b. 4.16 kv Emergency Bus Undervoltage (Undervoltage relays) - level two	$\geq$ 3700 volts with an 8.0 $\pm$ 2.0 second time delay	$\geq$ 3700 volts with an 8.0 $\pm$ 2.0 second time delay

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Storage Criticality Monitor and Ventilation System Isolation	2	*	$\leq$ 100 mR/hr	10^{-1} - 10^{+4} mR/hr	13 and 15
2. PROCESS MONITORS					
a. Containment Atmosphere-Particulate	1	ALL MODES	$\leq$ the value determined in accordance with specification 4.3.2.1.4.	10 - 10^{+6} cpm	14 and (a)
b. Containment Atmosphere-Gaseous	1	ALL MODES	$\leq$ the value determined in accordance with Specification 4.3.2.1.4.	10 - 10^{+6} cpm	14 and (a)

*With fuel in storage building.

TABLE 3.3-6 (Continued)

TABLE NOTATION

(a) - During MODE 6, also comply with the ACTION requirements of Specification 3.9.9, as applicable.

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

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

ACTION 15 - With the number of area monitors OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.13.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Storage				
Criticality Monitor and Ventilation System Isolation	S	R	M	*
2. PROCESS MONITORS				
a. Containment Atmosphere-Particulate	S	R	M	ALL MODES
b. Containment Atmosphere-Gaseous	S	R	M	ALL MODES

*With fuel in storage building

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with at least one OPERABLE detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the AZIMUTHAL POWER TILT:

At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient OPERABLE detector segments in these detector groups to compute at least two AZIMUTHAL POWER TILT values at each of the four axial elevations containing incore detectors.

b. For recalibration of the excore neutron flux detection system:

1. At least 75% of all incore detector segments,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the UNRODDED PLANAR RADIAL PEAKING FACTOR, the UNRODDED INTEGRATED RADIAL PEAKING FACTOR, or the linear heat rate:

1. At least 75% of all incore detector locations,

2. A minimum of 9 OPERABLE incore detector segments at each detector segment level, and

3. A minimum of 2 OPERABLE detector segments in the inner 109 fuel assemblies and 2 OPERABLE segments in the outer 108 fuel assemblies at each segment level.

An OPERABLE incore detector segment shall consist of an OPERABLE rhodium detector constituting one of the segments in a fixed detector string.

An OPERABLE incore detection location shall consist of a string in which at least three of the four incore detector segments are OPERABLE.

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 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.

BASES

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

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

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

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

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

The spent fuel storage area monitors are provided to serve two functions. First, the monitors are required by 10 CFR Part 70 to detect accidental criticality and to provide an alarm warning to personnel. A setpoint of 100 mR/hr meets the sensitivity requirements of 10 CFR Part 70 Second, the monitors provide a signal to direct the ventilation exhaust from the spent fuel storage area through a filter train when the dose rate exceeds the setpoint. The filter train is provided to reduce the particulate and iodine radioactivity released to the atmosphere. Should an accident involving spent fuel occur, the 100 mR/hr actuation setpoint would be sufficient to limit any consequences at the exclusion area boundary to those evaluated in the NRC Safety Evaluation, Section 15 (May 1974).

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs."

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

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

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

The operability of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically isolate the control room and initiate its operation in the recirculation mode of operation to provide the required protection. The chlorine detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2
DOCKET NO. 50-336

Introduction

By letter dated August 15, 1978, Northeast Nuclear Energy Company (NNECO or the licensee) requested an amendment to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit No. 2. The proposed amendment would change the Technical Specifications for engineered safety feature actuation system and radiation monitoring instrumentation (Sections 3.3.2.1 and 3.3.3.1) concerning the alarm/trip setpoints, the minimum channels operable and the action statements for the following instruments:

- containment airborne radioactivity monitor
- spent fuel storage area monitors
- spent fuel storage airborne radioactivity monitor

We have modified the proposed changes somewhat and the licensee has agreed with these modifications.

Discussion

The licensee proposed increasing the allowable value of the trip setpoint on the containment airborne radioactivity monitor in specification 3.3.2.1 to permit containment purging when the levels of airborne radioactivity inside containment are above the current containment purge valve isolation setpoint. He has experienced difficulty with the current setpoint when the airborne radioactivity levels are high because the purge valves will not stay open. The valves and the monitor were designed to close the purge valves to prevent the release of excessive radioactivity from the containment in the event of an accident. However, in the case of Millstone 2, setpoints specified are presently lower than necessary to perform that function. The present allowable setpoint values are less than or equal to 9100 counts per minute for gaseous activity and 1.0×10^6 counts per minute per hour for particulate activity. A major purpose of purging at Millstone 2 is to reduce the levels of airborne radioactivity inside the containment during normal operation. The licensee has found that the levels of airborne radioactivity inside containment have been higher than the monitor setpoints at the time of the attempted purges. Therefore, with the current setpoints, the valves would close, and the licensee could not purge.

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The licensee proposed increasing the allowable value of alarm/trip setpoint of the spent fuel storage area monitors in specification 3.3.3.1 to help prevent spurious alarms and auxiliary exhaust system starts. The value of the setpoint is presently specified to be less than or equal to two times background. These monitors are designed 1) to provide an alarm if a criticality occurs in the new fuel storage area and 2) to switch the ventilation exhaust from the spent fuel pool area to an auxiliary exhaust system if an accident causes radioactivity to be released from the stored spent fuel. However, for low background levels of radiation, less than five mR/hr, a setpoint of two times background can cause inadvertent alarms and trips from handling small radiation sources in the area and it is impractical to constantly change the setpoint as the radiation background in the area changes. Also, for simplicity and to be consistent with ANSI N16.2-1969, the licensee proposed to consolidate the specifications for the two monitor functions: 1) criticality alarm and 2) auxiliary exhaust system actuation.

The licensee proposed to change specification 3.3.3.1 such that the spent fuel storage airborne radioactivity monitor alarm setpoint would have units consistent with the other proposed change for the containment airborne radioactivity monitor.

Evaluation

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 inside containment.

The maximum allowable trip value for these monitors will correspond to the calculated concentrations of airborne radioactivity at the site boundary which would not exceed the concentrations listed in 10 CFR Part 20, Appendix B, Table II. Exposure for a year to the concentrations in 10 CFR Part 20, Appendix B, Table II corresponds to a thyroid dose to an individual of 1.5 rem or a total body dose of 0.5 rem; these doses are a small fraction of the guidelines of 10 CFR Part 100 for an individual at any point on the exclusion area boundary for two hours, i.e., 300 rem thyroid and 25 rem total body. This analysis bounds the dose consequences of a situation in which air containing a level of radioactivity just below the trip setpoint would be purged for a long period of time. Realistically, plant operators, based on indications from the effluent monitors, would terminate the release long before such doses could be approached.

Realistically, other signals such as the safety injection signal would be more likely to occur before the high radiation signal in the event of a Loss-of-Coolant-Accident (LOCA). However, high radiation is one of the signals which would isolate the containment including closing the purge valves. If a LOCA or any other accident occurs, the increase in the level of airborne radioactivity inside containment would be rapid. The rate of increase in radioactivity would be sufficiently rapid that the proposed setpoint would essentially be reached as quickly as the present setpoint. Therefore, the isolation signal and the closure of the purge valves would not be delayed nor would the amount of radioactivity released before purge valve closure increase significantly.

For the purposes of calculating the trip value setpoint, the licensee will be required to assume (1) the highest offsite annual average atmospheric dispersion factor of 5.8×10^{-6} seconds per cubic meter (site boundary, 0.48 miles in the NE sector) for purge releases through the building vent and (2) the highest offsite annual average atmospheric dispersion factor of 7.5×10^{-8} seconds per cubic meter (3 miles in the NNE sector) for purge releases through the Millstone Unit No. 1 stack. The licensee will also be required to assume that the gaseous and particulate radioactivity is xenon-133 and cesium-137, respectively.

The spent fuel storage area monitor serves two functions. First, the monitor detects any increase in radiation levels which may be caused by an accidental criticality in the new fuel storage area and provides an alarm to warn personnel of the criticality. The setpoint of 100 millirads/hour meets the sensitivity requirements of 10 CFR Part 70.24(a)(1). A radiation field of the magnitude specified in 10 CFR Part 70.24 (a)(1) caused by a criticality in the new fuel storage area would cause a radiation field of at least 100 millirads/hour at the spent fuel storage area monitors.

Second, the spent fuel storage area monitor provides a signal to direct the ventilation exhaust from the spent fuel storage area through an auxiliary exhaust system with HEPA and charcoal filters when the dose rate exceeds the monitor setpoint. The HEPA and charcoal filters reduce the particulate and iodine radioactivity released to the atmosphere. Specification 3.9.14 requires the use of the auxiliary exhaust system to exhaust the ventilation air from the pool area when fuel is being handled in the pool or loads are handled over the spent fuel pool. Specification 3.9.14 applies when spent fuel which has decayed less than 60 days is stored in the pool. We have performed an analysis of the potential consequences of a fuel handling accident in the spent fuel pool when specification 3.9.14 is not applicable. Our calculation assumes that the entire release occurs before the auxiliary system is fully actuated; therefore, no credit was taken for charcoal filtration or elevated release. Our other assumptions and the resulting calculated potential consequences of such an accident are given in Table 1. The calculated dose consequences are less than one percent of the guidelines of 10 CFR Part 100.11. Therefore, the dose consequences of the accident, with the monitor switching the exhaust to the auxiliary system at a setpoint of 100 mrad/hour, are also less than one percent of the guidelines of 10 CFR Part 100.11.

We have consolidated the two functions of the spent fuel storage area monitor in Specification 3.3.3.1 and established requirements which are more stringent than before with regard to the minimum number of operable channels. The requirement for the minimum number of operable channels has been increased from one to two for the criticality monitor function and the applicable mode requirement for the ventilation system isolation function has been changed from when irradiated fuel is being stored to when any fuel is in the storage building.

The spent fuel storage airborne activity monitor has no automatic safety feature actuation function. The monitor does provide an alarm, but that alarm is not required by regulation (as is the spent fuel storage area monitor). Therefore, specifications regarding the monitor are no longer necessary and have been deleted.

Based on our evaluation discussed above, we find the proposed changes to the Technical Specifications, as modified to meet our requirements, meet all regulatory requirements and are, therefore, acceptable.

TABLE 1

ASSUMPTIONS FOR AND POTENTIAL CONSEQUENCES AT THE EXCLUSION AREA BOUNDARY OF THE POSTULATED FUEL HANDLING ACCIDENT IN SPENT FUEL POOL FOR MILLSTONE UNIT 2

Assumptions:

Guidance in Regulatory Guide 1.25	
Power Level	2700 Mwt
Fuel Exposure Time	3 years
Peaking Factor	1.8
Equivalent Number of Assemblies Damaged	1
Number of Assemblies in Core	217
Charcoal Filters Available	None
Decay Time Before Moving Fuel	60 days
0 - 2 hours χ/Q Value, Exclusion Area Boundary (Ground Level Release)	6.6×10^{-4} sec/m^3

	Doses, Rem	
	Thyroid	Whole Body
Exclusion Area Boundary (EAB) Consequences from Accidents Inside Containment	1.1	<0.1

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or an increase in total amounts of effluents nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 1, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 49 to Facility Operating License No. DPR-65, issued to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company (the licensees), which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit No. 2 (the facility) located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications for engineered safety feature actuation system and radiation monitoring instrumentation. These changes concern the alarm/trip setpoints, the minimum channels operable and the action statements for the containment airborne radioactivity monitor, the spent fuel storage area monitors and the spent fuel storage airborne radioactivity monitor.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 15, 1978, (2) Amendment No. 49 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 1st day of March 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors