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Docket No. 50-336

SEP 26 1975

Amendment No. 4
Change No. 4
License No. DPR-65

Northeast Nuclear Energy Company
ATTN: Mr. Donald C. Switzer
President
P. O. Box 270
Hartford, Connecticut 06101

Gentlemen:

In response to your requests dated August 29, 1975, September 22, 1975, September 23, 1975, and September 25, 1975, the Commission has issued Amendment No. 4 to Facility Operating License No. DPR-65. This amendment (1) authorizes power operation of the Millstone Nuclear Power Station, Unit 2 and (2) changes certain of the Technical Specifications, Appendix A of License No. DPR-65. A signed copy of the amendment is enclosed. A copy of a related notice, which has been forwarded to the Office of the Federal Register for publication, is also enclosed.

Amendment No. 4 to License No. DPR-65 consists of (1) a completely revised operating license which permits you to operate Millstone Unit 2 at steady state reactor core power levels not in excess of 2560 megawatts thermal (rated power) in accordance with the amended license and the Technical Specifications, as revised through Change No. 4, and (2) Change No. 4 to the Technical Specifications. The specifications that have been changed are listed in Change No. 4. A copy of the staff's safety evaluation is also enclosed.

Item 2.C.1 of the amended license incorporates certain preoperational tests which must be completed in accordance with the sequence indicated. You provided information regarding these incomplete tests in your letters of July 22, 1975, July 28, 1975, and August 29, 1975. In addition, the staff discussed these tests with your representatives during meetings on July 25, 1975 and July 28, 1975. Based on its review, the staff has concluded that performance of the incomplete preoperational tests in the sequence indicated in the enclosure to amended license is acceptable.

Change No. 4 to the Technical Specifications (1) revises the specification concerning control element assembly position indicator channels to make the surveillance requirements consistent with the limiting condition for

PAR

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SURNAME ➤						
DATE ➤						

operation, (2) adds the containment high radiation level particulate monitoring system to the trip values for the engineered safety feature actuation system instrumentation, (3) includes a requirement (inadvertently omitted from Supplement 2 to the Safety Evaluation Report) that a bypass valve in the charging line be locked closed, and (4) corrects certain proofreading errors. The staff has concluded that Change No. 4 does not create a significant hazards consideration.

Sincerely,

Original Signed By
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 4 to DPR-65
2. Staff Safety Evaluation
3. Federal Register Notice

cc: See page 3

OFFICE ➤	RL:LWR 1-3	OELD	RL:LWR 1-3	I&E PDO for		
SURNAME ➤	PDO'Reilly:mt	J. S. COHEN	ODParr	F. Dreher per telecon 9/26/75		
DATE ➤	9/26/75	9/26/75	9/26/75	9/26/75		

cc: William H. Cuddy
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Counselors at Law
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Mr. Wallace Stickney
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OFFICE ➤						
SURNAME ➤						
DATE ➤						

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 2)

FACILITY OPERATING LICENSE

License No. DPR-65
Amendment No. 4

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

A. The application for license filed by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees) 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 and all required notifications to other agencies or bodies have been duly made;

B. Construction of the Millstone Nuclear Power Station, Unit 2, (facility) has been substantially completed in conformity with Construction Permit No. CPPR-76 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

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

D. There is reasonable assurance: (i) that the activities authorized by this amended operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

E. The licensees are technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

G. The issuance of this amended operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Amendment No. 4 to Facility Operating License No. DPR-65, subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this amended license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. Facility Operating License No. DPR-65, issued to The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company, is hereby amended in its entirety to read as follows:

A. This amended license applies to the Millstone Nuclear Power Station, Unit 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company. The facility is located on the north shore of Long Island Sound and on the east side of Niantic Bay in the Town of Waterford, Connecticut, about three miles from New London, Connecticut, and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 13 through 42, and the Environmental Report as amended (Amendments 1 through 5).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the north shore of Long Island Sound and on the east side of Niantic Bay, in the Town of Waterford, Connecticut, about three miles from New London, Connecticut, in accordance with the procedures and limitations set forth in this amended license.

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensees are authorized to operate the facility at steady state reactor core power levels not in excess of 2560 megawatts thermal provided that the preoperational test items identified in Enclosure 1 to this amendment have been completed in sequence. Enclosure 1 is an integral part of Amendment No. 4 to DPR-65.

(2) Technical Specifications

The Technical Specifications contained in Appendices A & B issued on August 1, 1975, as revised, are hereby incorporated in this amended license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised through Change No. 4. A copy of Change No. 4 is attached as Enclosure 2 to this amendment.

D. The licensees shall submit, within thirty (30) days of issuance of this amended license, proposed permanent modifications to the low pressure safety injection pump mini-flow bypass line to eliminate the potential for adverse consequences of single failures.

E. The licensees shall submit, within thirty (30) days after completion of the shielding measurements performed as part of the ascent-to-power test program, a report of the results of the shielding measurements, and a proposed design and installation schedule for any required shielding modifications.

F. The licensees shall submit within six (6) months from the date of this amended license a report analyzing the pre-operational reactor coolant pump flow test data and justifying the continued applicability of the pre-operational vibration test results for the Millstone Nuclear Power Station, Unit 2.

G. This amended license is effective as of its date of issuance and shall expire at midnight December 11, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Enclosures:
1. Incomplete Preoperational Test Items Which Must Be Completed
2. Change No. 4 to Technical Specifications Contained in Appendix A to DPR-65

Date of Issuance: SEP 26 1975

SEP 26 1975

ENCLOSURE 1 TO AMENDMENT NO. 4
TO LICENSE NO. DPR-65

Incomplete Preoperational Test Items
Which Must Completed

A. The following items must be completed to the satisfaction of the Commission prior to initial criticality:

1. Preoperational Tests (Numbered in accordance with Millstone Plant Operating Procedures).

   a. T2311P - Sampling System
   b. T2335P - Clean Liquid Radwaste
   c. T2404BPI03 - Radwaste Area Monitor
   d. T2404API04 - Process Rad. Monitor - Letdown
   e. T-INT-5001 - Post Core Load Hot Functional Test

   The above tests are to be completed, evaluated and accepted by the licensee.

B. The following items must be completed to the satisfaction of the Commission prior to commencing the Power Ascension Test Program:

1. Preoperational Test

   a. T2414P - Internals Vibration Monitor

2. Incore Analysis Computer Program Acceptance Test

The above items are to be completed, evaluated, and accepted by the licensee.

C. The following items must be completed to the satisfaction of the Commission during the Power Ascension Test Program.

1. Preoperational Test

a. T2343P003 - 4.16 KV Loading
b. T2344P002 - 480 V LC/MCC Loading
c. T2320P - Feedwater Htr. Dr. and Vent
d. Dynamic testing of the following piping systems during the Power Ascension Test Program:

(1) Main Feedwater

The above tests are to be completed, evaluated, and accepted by the licensee.

D. The following items must be completed to the satisfaction of the Commission during the first year of commercial operation.

1. Preoperational Tests

a. T2340BP - Primary Makeup Water and Storage

The above test is to be completed, evaluated, and accepted by the licensee within one month of installation of system modifications.

ENCLOSURE 2 TO AMENDMENT NO. 4
TO LICENSE NO. DPR-65

SEP 26 1975

CHANGE NO. 4
TO
TECHNICAL SPECIFICATIONS
CONTAINED IN APPENDIX A TO
LICENSE DPR-65

1. Make the following changes in Table 3.3-4:

   Under the heading, FUNCTIONAL UNIT, change Item 7.d to read as follows,

   "Containment Radiation-High
   Gaseous Monitor
   Particulate Monitor"

   Under the heading, TRIP VALUE, insert "9100 cpm" as the value for the gaseous monitor, and insert "1.0 x 10^6 cpm/hr" as the value for the particulate monitor. Under the heading, ALLOWABLE VALUES, insert "9100 cpm" as the value for the gaseous monitor and "1.0 x 10^6 cpm/hr" as the value for the particulate monitor.

2. In Specification 4.1.3.3, change "agree within 3 steps" to "agree within 6 steps".

3. In Specification 4.5.2.8, under the heading, VALVE NUMBER, insert "CH-434".

   Under the heading, VALVE FUNCTION, insert "Thermal Bypass". Under the heading, VALVE POSITION, insert "Closed".

4. Change the following specifications as indicated to correct proof-reading errors.

   In Table 2.2-1, Item 3, under the headings TRIP SETPOINT and ALLOWABLE VALUES, change the inequality sign in front of "95.0%" to read "greater than or equal to".

   In Section 2.2.1 of the Bases under the heading, POWER LEVEL-HIGH, change the "20%" in the sixth line of the second paragraph to "15%".

SEP 26 1975

SAFETY EVALUATION BY OFFICE OF
NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 2 TO
LICENSE NO. DPR-65
CHANGE NO. 4 TO TECHNICAL SPECIFICATIONS
THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
THE NORTHEAST NUCLEAR ENERGY COMPANY
MILLSTONE NUCLEAR POWER STATION
UNIT 2
DOCKET NO. 50-336

INTRODUCTION

By letters dated August 29, 1975, September 22, 1975, September 23, 1975 and September 25, 1975, the Licensee requested changes to the Technical Specifications for the Millstone Nuclear Power Station, Unit 2. The changes, which apply to Appendix A to the license would (1) revise the specification concerning control element assembly (CEA) position indicator channels to make the surveillance requirements consistent with the limiting condition for operation, (2) add the trip values for the containment high radiation level particulate monitoring system to the engineered safety feature actuation system instrumentation trip values, and (3) correct certain proofreading errors.

The staff's evaluation of the changes is discussed below. The staff's conclusions follow the discussion.

DISCUSSION

The Licensee has proposed a change in Specification 4.1.3.3 governing the surveillance requirements for the CEA position indicator channels to make these requirements consistent with the corresponding limiting conditions for operation (Specification 3.1.3.3). Specification 3.1.3.3 requires that all CEA reed switch position indicator channels and all CEA pulse counting position indicator channels be operable and capable of determining the absolute CEA positions within $\pm$ 3 steps. This tolerance considers the following sources of error:

(1) Performance variation in reed switches and CEA extension shaft magnets,

(2) The method by which the plant computer determines that a CEA is latched prior to counting pulses,

(3) Reed switch resistor tolerance,

(4) Temperature effects,

(5) The accuracy in positioning of reed switch assemblies during installation.

Section 4.1.3.3 presently requires that the operability of each position indicator channel be determined by periodically verifying that the pulse counting CEA position indicator channels and the reed switch CEA position indicator channels agree within 3 steps. This requirement that the two types of position indication agree within 3 steps is not consistent with the deviation of $\pm$ 3 steps allowed by Specification 3.1.3.3 for each type of position indication when compared to the absolute CEA position. The proposed variation of six steps between the pulse counting CEA position indicator channels and the reed switch CEA position indicator channels would make the surveillance requirements of Specification 4.1.3.3 consistent with the limiting condition for operation of Specification 3.1.3.3.

Section 3/4.1.3 of the Bases defines a small misalignment as less than a 20-step variation among the CEA's in a group. Specification 3.1.3.1 requires that each CEA of a given group be positioned within 10 steps (indicated position) of all other CEA's in its group. The proposed allowable variation between the pulse counter and reed switch CEA position indicators in conjunction with the requirement of Specification 3.1.3.1, remains consistent with the small misalignment of less than 20 steps defined in the Bases. Therefore, we conclude that the requested change to Specification 4.1.3.3 should be authorized.

The Licensee has proposed a change in Table 3.3-4 of Specification 3.3.2.1, governing the engineered safety feature actuation system instrumentation trip values to include trip values for the containment high radiation level particulate monitoring system.

As described in Section 7.5.6 of the FSAR, two redundant off-line particulate monitoring and gaseous monitoring and halogen sampling systems are used to continuously monitor the containment atmosphere by extracting representative samples from the containment auxiliary recirculation system. High radiation alarm signals from any one of the four monitored channels will cause the engineered safety features actuation system to initiate the containment purge isolation.

The trip value currently specified in Item 7.d of Table 3.3-4 for containment surge valves isolation containment high radiation is the trip value for the containment gaseous monitoring system. The containment particulate monitoring system trip value was not included in the development of the original specification. This change would insert the containment particulate monitoring system trip value in Table 3.3-4 in a format consistent with Table 3.3-3 of Specification 3.3.2.1.

The trip value of 9000 cpm specified in Table 3.3-4 for the gaseous monitoring system was selected such that purging the containment atmosphere through the nonfiltered path which releases effluents through the Millstone Unit 2 stack will limit the radioactivity concentrations at the site boundary to one-tenth of the limits specified in 10 CFR Part 20, Appendix B, Table II. The Licensee has proposed a containment particulate monitoring system trip value of 1×10^6 cpm/hr. This value represents the upper end of the range of the instrument. The trip value for the containment particulate monitoring system which would correspond to the trip value specified in Table 3.3-4 for the gaseous monitoring system is 1.78×10^6 cpm/hr. Therefore the proposed set value for the particulate monitoring system has been conservatively chosen to assure that the radioactivity concentrations through the Millstone Unit 2 stack will be limited to less than one-tenth of the 10 CFR Part 20 limits. Therefore, we conclude that the requested change to Specification 3.3.2.1, Table 3.3-4 should be authorized.

In addition, the Licensee has proposed changes to correct a number of proofreading errors. These changes are identified in Change No. 4.

CONCLUSIONS

We have concluded, based on the considerations discussed above, that Change No. 4 can be made without endangering the health and safety of the public and that activities permitted by these changes will be conducted in compliance with the Commission's regulations and that the issuance of this change will not be inimical to the common defense and security and that Change No. 4 involves no significant hazards consideration.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY, AND
NORTHEAST NUCLEAR ENERGY COMPANY
(Millstone Nuclear Power Station, Unit 2)

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the Nuclear Regulatory Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. DPR-65 issued to The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (licensees). This amendment revises the license in its entirety and permits the licensees to operate the Millstone Nuclear Power Station, Unit 2, at steady state reactor core power levels not in excess of 2560 megawatts thermal (rated power) in accordance with the amended license and the August 1, 1975 Technical Specifications, as revised through Change No. 4. This amendment also revises the specification concerning control element assembly position indicator channels, adds the containment high radiation level particulate monitoring system, and corrects proofreading errors in the Technical Specifications. The correction has been designated as Change No. 4.

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

The amended license is effective as of its date of issuance and shall expire at midnight December 11, 2010.

For further details with respect to this action, see Amendment No. 4 to Facility Operating License No. DPR-65 and Supplements No. 2 and No. 3 to the Safety Evaluation Report. These items are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut. A copy of the amended license and Supplement No. 2 to the Safety Evaluation Report may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 26th day of September 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Docket File

CHANGE NO. 1, DATED AUGUST 25, 1975

TO

DPR-65

MILLSTONE NUCLEAR POWER STATION

UNIT 2

The enclosed pages of Change No. 1, dated August 25, 1975, should be substituted for the corresponding pages of Appendix "A" to DPR-65. The vertical lines in the right hand margin indicate the portion of the specification that has been changed.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION | PAGE

INDEX

DESIGN FEATURES

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	$\leq$ 30.0
Containment Isolation	$\leq$ 5.5
Enclosure Building Filtration System	$\leq$ 35.0
b. CSAS	
Containment Spray	$\leq$ 18.5
c. CIAS	
Containment Isolation	$\leq$ 5.5
d. SRAS	
Containment Sump Recirculation	$\leq$ 120
e. EBFAS	
Enclosure Building Filtration System	$\leq$ 35.0
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Containment Pressure--High-High	
a. Containment Spray	≤ 34.5*/20.5**
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	≤ 5.5
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 120

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

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

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent enclosure building filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one enclosure building filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each enclosure building filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsoerber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemcial release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying removal efficiency of $\geq$ 90% for radioactive methyl iodide at an air flow velocity of 0.67 ft/sec $\pm$ 20% with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$ 95% relative humidity, and $\geq$ 125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying once entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm $\pm$ 10% during system operation.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that each pump operates for at least 15 minutes.

d. Cycling each testable, remote operated valve through at least one complete cycle.

e. Verifying the correct position for each manual valve not locked, sealed or otherwise secured in position.

f. Verifying the correct position for each remote operated valve.

STEAM GENERATOR WATER ADDITION

LIMITING CONDITION FOR OPERATION

3.7.1.6 Steam generator water addition shall be made by the use of the auxiliary feedwater pumps and limited to a recovery rate of $\leq$ 168 GPM.

APPLICABILITY: Whenever the secondary water level in a steam generator is below the elevation of the feedwater sparger.*

ACTION:

With the rate of water addition from the auxiliary feedwater pumps in excess of 168 GPM and the steam generator water level below the feedwater sparger, immediately throtte the feedwater flow to within its limit.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Steam generator water addition shall be verified to be within its limit by continuously monitoring the rate of secondary water increase whenever the steam generator water level is below the elevation of the feedwater sparger.

*See Special Test Exception 3.10.6.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 150,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With less than 150,000 gallons of water in the condensate storage tank, within 4 hours either:

a. Restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the fire water system as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank water volume to within its limits within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.5 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, immediately:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 2 hours.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.5.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.3.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

SPECIAL TEST EXCEPTIONS

STEAM GENERATOR WATER ADDITION

LIMITING CONDITION FOR OPERATION

3.10.6 The requirement of Specification 3.7.1.6 relating to a recovery rate limit may be suspended during the performance of tests to determine additional data to verify the adequacy of the feedwater sparger modifications provided no sustained water hammer in the feedwater system piping is allowed to exist.

APPLICABILITY: Whenever the secondary water level in a steam generator is below the elevation of the feedwater sparger.

ACTION:

If water hammer begins, immediately terminate feedwater flow to the steam generator.

SURVEILLANCE REQUIREMENTS

4.10.6 The absence of sustained water hammer shall be verified by the following whenever the steam generator water level is below the elevation of the sparger:

a. Continuous aural monitoring in the vicinity of the containment side feedwater piping to the affected steam generator.

b. Continuous monitoring of the output of temporary instrumentation installed for monitoring of the feedwater piping. Water hammer is indicated by oscillations in the piping.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT AND INSERTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

This special test exception permits the reactor to be critical at less than 5% of RATED THERMAL POWER during low temperature PHYSICS TESTING required to measure such parameters as CEA worth and SHUTDOWN MARGIN.

3/4.10.4 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at $\leq$ 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

3/4.10.5 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.6 STEAM GENERATOR WATER ADDITION

This special test exception permits the water level recovery rate in the steam generator to exceed 168 gpm. This special test exception is required to demonstrate the effectiveness of feedwater system modifications which have been accomplished to minimize the effects of "feedwater hammer".

Docket File

CHANGE NO. 2, DATED SEPTEMBER 3, 1975

TO

DPR-65

MILLSTONE NUCLEAR POWER STATION

UNIT 2

The enclosed page of Change No. 2, dated September 3, 1975, should be substituted for the corresponding page of Appendix "A" to DPR-65. The vertical line in the right hand margin indicates the portion of the specification that has been changed.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	$\leq$ 30.0
Containment Isolation	$\leq$ 5.5
Enclosure Building Filtration System	$\leq$ 35.0
b. CSAS	
Containment Spray	$\leq$ 18.5
c. CIAS	
Containment Isolation	$\leq$ 5.5
d. SRAS	
Containment Sump Recirculation	$\leq$ 120
e. EBFAS	
Enclosure Building Filtration System	$\leq$ 35.0
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 30.0*/30.0**
b. Containment Isolation	$\leq$ 7.5
c. Enclosure Building Filtration System	$\leq$ 35.0*/35.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Containment Pressure--High-High	
a. Containment Spray	≤ 34.5*/20.5**
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	≤ 60
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 120

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

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