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. 50-245

MILLSTONE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 86
License No. DPR-21

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 Electric Company (the licensees) dated October 16, 1980, as supplemented December 9, 1981, March 9, October 15, and November 2, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 3.B of Provisional Operating License No. DPR-21 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B as revised through Amendment No. 86 , are hereby incorporated in the license. Northeast Nuclear Energy Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 12, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 86

PROVISIONAL OPERATING LICENSE NO. DPR-21

DOCKET NO. 50-245

Replace the following pages of Appendix A Technical Specifications with the enclosed pages. The revised pages contain the captioned amendment number and vertical lines indicating the area of change.

REMOVE	INSERT
3/4 1-2	3/4 1-2
3/4 2-5	3/4 2-5
--	3/4 2-5a*
3/4 2-6	3/4 2-6
3/4 3-4	3/4 3-4
--	3/4 3-4a
3/4 3-5	3/4 3-5
--	3/4 3-5a*
3/4 3-6	3/4 3-6**
B3/4 1-2	B3/4 1-2

* Included for pagination purposes only.
** Overleaf page 3/4 3-6 is included for maintaining document completeness.

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in which Function Must Be Operable			Action*
			Refuel (8)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM:					
3	High Flux	≤ 120/125 of full scale	X	X	(5)	A
3	Inoperative	A. HI Voltage < 80 volt DC B. IRM Module Unplugged C. Selector Switch not in Operate Position	X	X	X (10)	A
	APRM:					
2	Flow Biased High Flux	See Section 2.1.2A	X	X	X	A or B
2	Reduced High Flux	See Section 2.1.2A	X	X	X	A or B
2	Inoperable	A: > 50% LPRM Inputs** B. Circuit Board Removed C. Selector Switch not in Operate Position	X	X	X	A or B
2	High Reactor Pressure	≤ 1085 psig	X	X	X	A
2	High Drywell Pressure	≤ 2 psig	X (9)	X (7)	X (7)	A
2	Reactor Low Water Level	≥ 1.0 inch***	X	X	X	A
2	Scram Discharge Vol. High Level	≤ 26 inches above the center-line of the lower end cap to SDIV pipe weld	X (2)	X	X	A

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	See Specification 2.1.2B
2	APRM Downscale	≥ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	≤ .65 w + 42 (2)
1 (6)	Rod Block Monitor Downscale	≥ 3/125 Full Scale
3	IRM Downscale (3)	≥ 3/125 Full Scale
3	IRM Upscale	≤ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	≤ 10^5 counts/sec.
1	Scram Discharge Volume - Water Level High	≤ 14 inches above lower cap to SDIV pipe weld
1	Scram Discharge Volume - Scram Trip Bypassed	N/A

(1) For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks; IRM downscale are not operable in the RUN position and APRM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

(2) W is the total core flow in percent of design (69×10^6 #/hr.). Trip level setting is in percent of full power.

(3) IRM downscale may be bypassed when it is on its lowest range.

(4) This function may be bypassed when the count rate is ≥ 100 cps or when all IRM range switches are above Position 2.

(5) One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

Table 3.2.3 Continued
Instrumentation That Initiates Rod Block

(6) The trip may be bypassed when the reactor power is $\leq$ 30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE COOLING INSTRUMENTATION ROD BLOCKS AND ISOLATIONS

Instrument Channel	Instrument Functional Test(2)	Calibration(2)	Instrument Check(2)
ECCS Instrumentation			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	--
2. Drywell High Pressure	(1)	Once/3 Months	--
3. Reactor Low Pressure (Pump Start)	(1)	Once/3 Months	--
4. Reactor Low Pressure (Valve Permissive)	(1)	Once/3 Months	--
5. APR LP Core Cooling Pump Interlock	(1)	Once/3 Months	--
6. Containment Spray Interlock	(1)	Once/3 Months	--
7. Loss of Normal Power Relays	Refueling Outage	None	--
8. Power Available Relays	(1) (5)	None	--
9. Reactor High Pressure		Once/3 Months	--
Rod Blocks			
1. APRM Downscale	(1) (3)	Once/3 Months	(1)
2. APRM Flow Variable	(1) (3)	Once/3 Months	(1)
3. IRM Upscale	(6)	(6)	(6)
4. IRM Downscale	(6)	(6)	(6)
5. RBM Upscale	(1) (3)	Once/3 Months	(1)
6. RBM Downscale	(1) (3)	Once/3 Months	(1)
7. SRM Upscale	(6)	(6)	(6)
8. SRM Detector not in Startup Position	(6)	(6)	(6)
9. Scram Discharge Volume - Water Level High	Quarterly	Refueling Outage	--
10. Scram Discharge Volume - Scram Trip Bypassed	Quarterly	None	--
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	--
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1) (3)	Refueling Outage	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months(4)	Once/Day

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

5. During operation with limiting control rod patterns, as determined by the reactor engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (Sec.)
5	0.375
20	0.900
50	2.000
90	3.500

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. During each operating cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

2. The scram discharge volume drain and vent valves shall be verified open at least once per month.

3. The following conditions of operability of the scram discharge volume drain and vent valves shall be verified at least once per operating cycle in accordance with Section 3.13, Inserview Inspection:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	a. Closing time after signal for control rods to scram and b. Verification of opening when scram signal is reset and when the scram discharge volume trip is bypassed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec.)
5	0.398
20	0.954
50	2.120
90	3.800

3. a. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

   b. The scram discharge volume drain and vent valves will close in less than 30 seconds after receipt of a signal for control rods to scram.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.
2. Directional control valve electrically disarmed while in a non-fully inserted position.
3. Scram insertion greater than maximum permission insertion time.

D. Control Rod Accumulators

Once a shift, check the status in the control room of the pressure and level alarms for each accumulator.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
E. Reactivity Anomalies The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% ΔK. If this limit is exceeded, the reactor will be shutdown until the cause has been determined and corrective actions have been taken if such actions are appropriate. In accordance with Specification 6.6, the NRC shall be notified of this abnormal occurrence within 24 hours. F. If Specification 3.3 A through D above are not met, a normal orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours. G. Allowable combinations of thermal power and total core flow shall be restricted to Curve 1 shown in Figure 3.3.1.	E. Reactivity Anomalies During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

APRM's #4, #5 and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water, generator load rejection, and turbine stop valve closure are discussed in Section 2 of these specifications.

Instrumentation (pressure switches) in the drywell is provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumenatation which is discussed in Specification 3.2. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent the reactor from going critical following the accident.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume which is the low point in the piping. No credit was taken for the volume contained in the piping below a point which is 26 inches above the lower cap to the SDIV pipe weld when calculating the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrumented volume which alarm and scram the reactor while there is still greater than 3.34 gallons per drive available to accept scram water. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. Ref. Section 4.4.3 FSAR. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum and bypass closure at 7" Hg vacuum.