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

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

DOCKET NO. 50-245

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 67
License No. DPR-21

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

   A. The application for amendment by Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees) dated March 19, 1980, as supplemented April 16, 1980, April 29, 1980 and May 2, 1980, and as supported by letter dated February 24, 1980, 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.

8005270289 P

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:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 67, 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 as of the date of its 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: May 8, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 67

PROVISIONAL OPERATING LICENSE NO. DPR-21

DOCKET NO. 50-245

Replace the following pages of the 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 2-3	3/4 2-3
3/4 2-6	3/4 2-6
3/4 11-1	3/4 11-1
B3/4 2-1	B3/4 2-1
--	B3/4 2-2a
B3/4 2-4	B3/4 2-4
B3/4 5-4	B3/4 5-4
B3/4 5-5	*B3/4 5-5
B3/4 11-1	B3/4 11-1

*There are no changes to the provisions contained on this page. The Technical Specifications have merely been repositioned.

TABLE 3.2.2

INSTRUMENTATION THAT INITIATES AND CONTROLS THE EMERGENCY CORE COOLING SYSTEMS

Minimum Number of Operable Inst. Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	79 (+4-0) inches above top of active fuel	1 - In conjunction with low reactor pressure initiates core spray and LPCI. 2 - In conjunction with high dry well pressure, 120 sec. time delay, and LP core cooling pump interlock initiates auto blowdown. 3 - Initiates FWCI and Isolation Condenser. 4 - Initiates starting of diesel generator and gas turbine generator.
2	High Drywell Pressure	≤ 2 Psig	1 - Initiates core spray, LPCI, and FWCI, and SBGTS. 2 - In conjunction with low low water level, 120 sec. time delay, and LP core cooling pump interlock initiates auto blowdown. 3 - Initiates starting of diesel and gas turbine generator.
1	Reactor Low Pressure Permissive	300 Psig ≤ P ≤ 350 Psig	1 - Permissive for opening core spray and LPCI admission valves. 2 - In conjunction with low low reactor water level initiates core spray and LPCI.
1	High Reactor Pressure	≤ 1085 Psig	1 - In conjunction with 15 second time delay, initiates Isolation Condenser.
1	Timer, Isolation Condenser	≤ 15 seconds	1 - In conjunction with high reactor pressure, initiates Isolation Condenser.

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)
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)	Refeuling Outage	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day

Amendment No. ~~24~~, 67

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.11 REACTOR FUEL ASSEMBLY	4.11 REACTOR FUEL ASSEMBLY
Applicability	Applicability
The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	The Surveillance Requirements apply to the parameters which monitor the fuel rod operation conditions.
Objective	Objective
The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	The Objective of Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications	Specifications
A. Average Planar Linear Heat Generation Rate (APLHGR)	A. Average Planar Linear Heat Generation Rate (APLHGR)
1. During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed 91% of the limiting value shown in Figure 3.11.1. This limiting value of APLHGR has been provided to insure fuel cladding integrity under postulated small break LOCA conditions with a gas turbine failure and LPCI injection into the broken loop.	The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.
2. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR specified in Section 3.11.A.1 is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	

3.2 Bases:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations and to prescribe the trip settings required to assure adequate performance.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guideline values of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127 inches above the top of the active fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. Ref. Section VII-4.4 FSAR. For a trip setting of 127 inches above the top of the active fuel and a 60-second valve closure time the valves will be closed before core uncovery occurs even for the maximum break in the line; and therefore, the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 79 inches above the top of the active fuel. This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section VII-7.2.2 FSAR and also activates the ECC subsystems and starts the emergency diesel generator and the gas turbine generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that post accident cooling can be effectively accomplished and the guideline values of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Ref. Section VI-2.7 and XIV-2.4 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Ref. Section VI-2.7 FSAR. The Isolation Condenser system has been added to the ECC systems to insure that cladding integrity is maintained for postulated small break LOCA conditions in the recirc. discharge piping with a gas turbine failure and LPCI injection into the damaged loop.

High pressure actuation of the Isolation Condenser (IC) will be a backup to direct activation on Low-Low level; similar to other ECCS systems. Activation is based on the high pressure signal (10[illegible]5 PSIG for 15 seconds) which occurs after MSIV closure on Low-Low water level and subsequent depressurization. The activation of the IC requires only the opening of normally closed valve IC-3 in the condensate return line. This valve is powered by the safety-grade DC battery. All valves in the system are powered by safety-grade AC or DC power and are also used for containment isolation. All are normally in the open position (other than IC-3). The IC system is safety Class 2 and is seismically qualified. The shell side water volume is sufficient for approximately 30 minutes of operation at rated conditions without makeup. Two sources of makeup are available. For small break mitigation, less than 10 minutes of operation is required, and generally at less than rated conditions.

To prevent excessive fuel clad temperature for the small pipe break, the FWCI or Isolation Condenser systems must function since for these breaks reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function and Isolation Condenser system are provided as back-ups to the FWCI in the event the FWCI does not operate. The arrangements of the tripping contacts are such as to provide these functions when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criterion is met. Ref. Section VI-2.0 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen-minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Two sets of two radiation monitors are provided which initiate isolation of the reactor building ventilation and operation of the standby gas treatment system. One set of monitors is located in the reactor building ventilation exhaust duct, and the other set is located in the vicinity of the fuel pool. Any high level trip or two downscale trips will initiate the standby gas treatment system. Trip settings of 100 mr/hr, on the fuel pool monitor and 11 mr/hr, on the ventilation duct monitor are based on initiating normal ventilation isolation and standby gas treatment system operation.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1085 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and FWCI subsystem in a feed and bleed manner. The minimum shell side water volume in the isolation condenser is 15,500 gallons.

The function of the Isolation Condenser during a small break accident is to assist the automatic pressure relief system in depressurizing the reactor as a backup to the FWCI system. The two effects of isolation condenser depressurization are: (1) the minimization of reactor inventory loss which normally occurs during APR blowdown; this reduces the time of core uncovery prior to reflooding; and (2) earlier onset of low pressure core spray cooling.

Analysis performed by General Electric in March 1976, in support of extended operation of Millstone while the isolation condenser was being retubed indicated that from 40% rated power, over 30 minutes is available to initiate operator action to mitigate the consequences of a loss of all feedwater. This is based upon manual depressurization with APR and coolant supplied by the LPCI and core spray systems. The FWCI was assumed lost as part of the non-mechanistic assumption of loss of feedwater. The successful mitigation of this postulated event was no uncovering of the fuel. Operators are instructed regarding special procedures to be utilized during this mode of plant operation. Thus, reducing power to 40% when the isolation condenser is inoperable provides a limiting condition for operation that is sufficient to preclude the need for any additional limiting conditions for operation on other ECCS systems.

F. Emergency Cooling Availability

The purpose of Specification F is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the emergency power source which powered the opposite core spray were out of service, only two LPCI pumps would be available. Likewise, if two LPCI pumps were out of service and two emergency service water pumps on the opposite side were also out of service, no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that low pressure core cooling systems may be out of service depending on the activities being performed. Specificification F allows removal of one CRD mechanism or fuel removal and replacement while the torus is in a drained condition without compromising core cooling capability. The specification establishes the minimum operable low pressure core cooling system, water inventories, electrical power supplies and other additional requirements that must exist to allow such activities as CRD mechanism maintenance or fuel removal and replacement, to be performed in parallel with other major activities. The available core cooling capability for a potential draining of the reactor vessel while this work is performed is based on an estimated drain rate and the maintained minimum volume of water, 383,000 gallons, in the refueling cavity to be supplied to the reactor vessel. In addition, the available low pressure core cooling systems are lined up to the condensate storage tank which supplements the reactor cavity water with an additional 450,000 gallons of water. Thus, with the torus drained, a volume of approximately 800,000 gallons of water will be maintained available to be supplied to the reactor vessel.

A potential draining of the reactor vessel would allow this water to enter into the torus and after approximately 270,000 gallons accumulated (needed to meet minimum NPSH requirements for the LPCI and/or CS pumps) the torus would be able to serve as a common suction header. This would allow a closed loop operation of the LPCI or CS pumps after a re-lineup of these systems to the torus rather than the condensate storage tank is made.

During the time the vessel is open and CRD maintenance or fuel removal and replacement is underway the electrical sources of power will be: two onsite emergency power sources (gas turbine and a diesel generator) and one offsite power, or two sources of offsite power and one onsite emergency power source.

The single failure criterion is met by the utilization of two independent low pressure core cooling subsystems and the availability of the three independent power supplies at all times. With the worst single failure -- loss of transfer bus number 7 a redundant LPCI or CS subsystem would still be available for service.

3.11 and 4.11 Bases

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Limiting Condition for Operation 3.11.A.1.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of References 1 and in Reference 2, 3, and 4 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel