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. 98
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, Western Massachusetts Electric Company and Northeast Nuclear Energy Company (the licensees) dated April 9, 1984 as supplemented May 15, 1984 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 the 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B as revised through Amendment No. 98, 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

for J. E. Lyons

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 14, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 98

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 vertical lines indicates the areas of change.

PAGES

iii

2-7

B2-8

3/4 1-1, 1-1a, 1-2, 1-3 and 1-4

3/4 5-5, 5-6, 5-7 and 5-8

3/4 6-5, 6-6

3/4 10-1, 10-2, 10-3

3/4 11-2, 11-3. 11-4, 11-5, 11-6, 11-7 and 11-7a

3/4 11-10

3/4 14-1

B 3/4 1-3 and 1-4

B 3/4 5-1

B 3/4 5-4

B 3/4 6-4

B 3/4 10-1 and 10-2

B. 2. The APRM rod block trip setting for the refuel and startup/hot standby mode, shall be less than or equal to 12% rated thermal power.

C. The reactor Low Water Level Scram trip setting shall be greater than or equal to 127 inches above the top of the active fuel.

D. The Reactor Low Low Water Level ECCS Initiation trip point shall not be greater than 83 inches nor less than 79 inches.

E. The turbine Stop Valve Scram trip setting shall be less than or equal to ten percent valve closure from full open.

F. The Turbine Control Valve Fast Closure Scram shall trip upon actuation of the acceleration relay in conjunction with failure of selected bypass valves to start opening within 280 milliseconds.

The maximum setting of the time delay relays which bypass this scram shall be 280 milliseconds.

G. The Main Steam Isolation Valve Closure Scram trip settings shall be less than or equal to ten percent valve closure from full open.

H. The Main Steam Line Low Pressure trip which initiates main steam line isolation valve closure shall be greater than or equal to 825 psig.

The design of the ECCS components to meet the above criteria was dependent on three previousl/ set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS would prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram

The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting $\leq$ 10% of valve closure the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is < 45% of rated, as measured by the turbine first stage pressure.

F. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCPR from becoming less than 1.06 for this transient. For the load rejection from 100% power, the heat flux increases to only 106.5% of its rated power value which results in only a small decrease in MCPR. This trip is bypassed below a generator output of 307 MWe because, below this power level, the MCPR is greater than 1.07 throughout the transient without the scram.

In order to accommodate the full load rejection capability, this scram trip must be bypassed because it would be actuated and would scram the reactor during load rejections. This trip is automatically bypassed for a maximum of 280 millisec following initiation of load rejection. After 280 millisec, the trip is bypassed providing the bypass valves have opened. If the bypass valves have not opened after 280 millisec, the bypass is removed and the trip is returned to the active condition. This bypass does not adversely affect plant safety because the primary system pressure is within limits during the worst transient even if this trip fails. There are many other trip functions which protect the system during such transients. Reference Response D-3 of Amendment 16.

LIMITING CONDITION FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram and provide automatic isolation of the Reactor Protection System buses from their power supplies.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

B. Response Time

The time from initiation of any channel trip to the de-energization of the scram solenoid relay shall not exceed 50 milliseconds.

C. Reactor Protection System Power Monitoring

Two RPS electric power monitoring channels for each inservice RPS MG set or alternate supply shall be operable at all times except as follows:

1. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENT

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram and provide automatic isolation of the reactor protection system buses from their power supplies.

Objective:

To specify the type and frequency of surveillance to be applied to the reactor protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Daily during reactor power operation, the peak heat flux shall be checked and the APRM scram and rod block settings given by the equations in Specifications 2.1.2A and 2.1.2B shall be determined.

C. The RPS electrical protection assemblies shall be determined operable as follows:

1. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST, and

2. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   a. Over-voltage $\leq$ (132) VAC,

   b. Under-voltage $\geq$ (108) VAC,

   c. Under-frequency $\geq$ (57) Hz, and

   d. Time-delay $\leq$ (4.0) seconds.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	D. When the reactor mode switch is in REFUEL or SHUTDOWN and fuel is in the reactor vessel, no trip functions are required to be operable provided that all control rods are fully inserted, valved out and electrically disarmed. Thereafter, daily surveillance shall be performed to verify that all control rods remain valved out and electrically disarmed.

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: Refuel/Shutdown (8,11)	Startup/Hot Standby	Run	Action*
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	≤ 1005 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 centerline of the lower end cap to SDIV pipe weld	X (2)	X	X	A

Amendment No. 11, 98

TABLE 3.1.1 (Continued)

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: Refuel/ Shutdown (8,11)	Startup/Hot Standby	Run	Action*
2	Turbine Condenser Low Vacuum	≥ 23 in.Hg Vacuum	X (3)	X (3)	X	A or C
2	Main Steamline Radiation	≤ 7 x Normal Full Power Background	X	X	X	A or C
4 (6)	Main Steamline Isolation Valve Closure	≤ 10% Valve Closure	X (3)	X (3)	X	A or C
2	Turbine Control Valve Fast Closure	See Section 2.1.2 F	X (4)	X (4)	X (4)	A or C
2	Turbine Stop Valve	≤ 10% Valve Closure	X (4)	X (4)	X (4)	A or C

Notes:

1. There shall be two operable or tripped trip systems for each function.
2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.
3. Bypassed when reactor pressure is < 600 psig.
4. Bypassed when first stage turbine pressure is less than that which corresponds to 45% rated steam flow (generator output approximately 307 MWe).

5. IRM's are bypassed when mode switch is placed in run. The detector for each operable IRM channel shall be fully inserted until the associated APRM channel is operable and indicating at least 3/125 full scale.****

6. The design permits closure of any one valve without a scram being initiated.

7. May be bypassed when necessary by closing the manual instrument isolation valve for scram of PS-1621 A through D during purging for containment inerting or deinerting.

8. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode Switch in Shutdown
   b. Manual Scram
   c. High Flux IRM
   d. Scram Discharge Volume High Level
   e. APRM Reduced High Flux

9. Not required to be operable when primary containment integrity is not required.

10. With the mode switch in "run position" an inoperative trip function also requires an associated APRM "downscale alarm."

11. Trip functions are not required to be operable if all control rods are fully inserted, valved out and electrically disarmed.

*Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within eight hours.
C. Reduce turbine load and close main steam line isolation valves within eight hours.

** An APRM will be considered inoperable if there are less than two LPRM inputs per level or there are less than 50% of the normal compliment of LPRM's to an APRM.

*** One inch on the water level instrumentation is 127 above the top of the active fuel.

**** Per errata sheet dated 10-7-70.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

5. From and after the date that one containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding four days provided that all active components of the other containment cooling subsystem, both core spray subsystems and both emergency power sources for operation of such components if no external source of power were available, shall be operable.

6. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

C. FWCI Subsystem

1. Except as specified in 3.5.C.3 below, the FWCI subsystem shall be operable whenever the reactor coolant temperature is greater than 330°F and irradiated fuel is in the reactor vessel.

2. There shall be a minimum of 225,000 gallons of water in the condensate storage tank for operation of the FWCI.

C. Surveillance of FWCI Subsystem shall be performed as follows:

1.

Item	Frequency
a. Pump and valve operability	Per Surveillance Requirement 4.13
b. Simulated Automatic Actuation Test	Every refueling outage

2. Once a week the quantity of water in the condensate storage tank shall be logged.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3. From and after the date that the FWCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Pressure Relief Subsystem, the core spray subsystems, LPCI subsystem, and isolation condenser system are operable.

4. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

D. Automatic Pressure Relief (APR) Sybsystems

1. Except as specified in 3.5.D.2 below, the APR subsystem shall be operable whenever the reactor coolant temperature is greater than 330°F and irradiated fuel is in the reactor vessel.

D. Surveillance of the Autonic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle, the following shall be performed:

   a. A simulated automatic initiation of the system throughout its operating sequence but excludes actual valve opening, and

   b. With the reactor at low pressure, each relief valve shall be manually opened until valve operability has been verified by torus water level instrumentation, or by an audible discharge detected by an individual located outside the torus in the vicinity of each relief line.

LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the four relief/safety valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor coolant temperature is above 330°F with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are completed and the subsystem made fully operable and provided that during such time the remaining automatic pressure relief valves, FWCI subsystem, and gas turbine generator are operable.

3. If the requirements of 3.5.D cannot be met, an orderly reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. Isolation Condenser System

1. Whenever the reactor coolant temperature is greater than 330°F and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.E.2 and the shell side water level shall be greater than 66 inches.

SURVEILLANCE REQUIREMENT

2. When it is determined that one safety/relief valve of the automatic pressure relief subsystem is inoperable the actuation logic of the remaining APR valves and FWCI subsystem shall be demonstrated to be operable immediately and daily thereafter.

E. Surveillance of the Isolation Condenser System shall be performed as follows:

1. Isolation Condenser System Testing:

a. The shell side water level and temperature shall be checked daily.

LIMITING CONDITION FOR OPERATION

2. From and after the time that the Isolation Condenser is made or found to be inoperable, for any reason, power operation shall be restricted to a maximum of 40% of full power, i.e., (804 MW_{th}) within 24 hours until such time the Isolation Condenser is returned to service provided that all active components of the core spray subsystems and LPCI subsystems are operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

F. Minimum Core and Containment Cooling System Availability

1. Except as specified in 3.5.F.2, 3.5.F.3, 3.5.F.7 and 3.5.F.8 below, both emergency power sources shall be operable whenever irradiated fuel is in the reactor.

SURVEILLANCE REQUIREMENT

b. Simulated automatic actuation and functional system testing shall be performed during each refueling outage or whenever major repairs are completed on the system.

c. The system heat removal capability shall be determined once every five years.

d. Calibrate vent line radiation monitors quarterly.

e. Motor operated valves shall be tested per surveillance requirement 4.13.

F. Surveillance of Core and Containment Cooling System

1. The surveillance requirements for normal operation are in Section 4.9.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor in the cold shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During power operation or whenever the reactor coolant pressure is greater than 90 psig with irradiated fuel in the reactor vessel, the safety valve function of the six relief/safety valves shall be operable, except as specified in 3.6.E.5 below. (The solenoid activated relief function of the relief/safety valves shall be operable as required by Specification 3.5.D.)

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have the reactor coolant pressure below 90 psig within 24 hours.

3. When the safety/relief valves are required to be operable per Specification 3.6.E.1, the Valve Position Indication shall be operable. Two of the six channels may be out of service provided backup indication for the affected valves is provided by the Valve Discharge Temperature Monitor.

4. If Specification 3.6.E.3 is not met, reactor operation is permissible only for the succeeding 30 days unless the Valve Position Indication System is made operable sooner.

E. Safety and Relief Valves

1. Three of the relief/safety valves top works shall be bench checked or replaced with a bench checked top works each refueling outage. All six valves top works shall be checked or replaced every two refueling outages. The set pressure shall be adjusted to correspond with a steam set pressure of:

No. of Valves	Set Point (psig)
1	1095 ± 1%
1	1110 ± 1%
4	1125 ± 1%

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. During each operating cycle with the reactor at low pressure, each safety valve shall be manually opened until operability has been verified by torus water level instrumentation, or by the Acoustic Valve Position Indication System, or by an audible discharge detected by an individual located outside the torus in the vicinity of each discharge.

4. The Valve Position Indication System shall be functionally tested once every three months and calibrated once per operating cycle.

5. The valve discharge temperature monitor shall be calibrated at least once every 18 months.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
5. During reactor vessel hydrostatic testing with all control rods inserted, the safety valve function of at least two of the six safety/relief valves shall be operable. F. <u>Structural Integrity</u> The structural integrity of the primary boundary shall be maintained as specified in Technical Specification 3.13.	F. <u>Structural Integrity</u> Inservice Inspection and Testing of primary system boundary components shall be performed as specified in Surveillance Requirement 4.13.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.10 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to fuel handling, core reactivity limitations, and spent fuel handling.

Objective:

To assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks.

Specification:

A. Refueling Interlocks

During core alterations, the reactor mode switch shall be locked in the REFUEL position and the refueling interlocks shall be operable, or all control rods shall be fully inserted, valved out and electrically disarmed.

4.10 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and spent fuel handling.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and spent fuel handling.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

If the refueling interlocks are not operable, all control rods shall be verified to be fully inserted, valved out and electrically disarmed in accordance with specification 4.1.C.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Core Monitoring During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant, except as specified in Paragraphs 3 and 4 below. For an SRM to be considered operable, the following conditions shall be satisfied: 1. The SRM shall be inserted to the normal operating level. (Use of special moveable dunking type detectors during fuel loading or major core alterations in place of normal detectors are permissible as long as the detector is connected into the normal SRM circuit.) 2. The SRM shall have a minimum neutron induced count rate of three per second with all rods fully inserted in the core. 3. Prior to unloading, the SRM's shall be proven operable as stated above, however, during spiral unloading, the count rate may drop below 3 cps. 4. Special movable dunking type detectors will be inserted into the core, prior to reloading fuel assemblies into the central core region (with all control rods inserted). Before the ninth fuel assembly is loaded into the core in the close proximity of the movable dunking chambers or the SRM's Paragraph 3.10.B.1 and 2 apply.	B. Core Monitoring 1. Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response when core alterations are being made. 2. Prior to spiral unloading or reloading, the SRM's shall be functionally tested. Prior to spiral unloading, the SRM's should also be checked for neutron response.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Fuel Storage Pool Water Level Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level greater than or equal to 33 feet.	C. Fuel Storage Pool Water Level Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.
D. Crane Operability The 110-ton redundant crane shall be operable when the crane is used for handling of a spent fuel cask.	D. Crane Operability Within 4 days prior to Spent Fuel Cask handling operations, a visual inspection of crane cables, sheaves, hook, yoke, and cask lifting trunnions will be made. Following these inspections, no-load mechanical and electrical tests will be conducted to verify proper operation of crane controls, brakes and lifting speeds. A load test will then be conducted by lifting the empty cask out of the pivot cradle. The above inspections and pre-lifting procedure shall meet the requirements of ANSI Standard B30.2, 1967.
E. Crane Travel With a Spent Fuel Cask Spent fuel casks shall be prohibited from travel over irradiated fuel assemblies. When handling a spent fuel cask, the crane mode switch shall be in the "Mode 2" position and the mode switch key removed.	E. Crane Interlocks and Switches Crane interlocks and limit switches which prevent crane travel over irradiated fuel assemblies shall be demonstrated OPERABLE within seven days prior to handling of all spent fuel casks and every seven days thereafter during spent fuel cask handling.

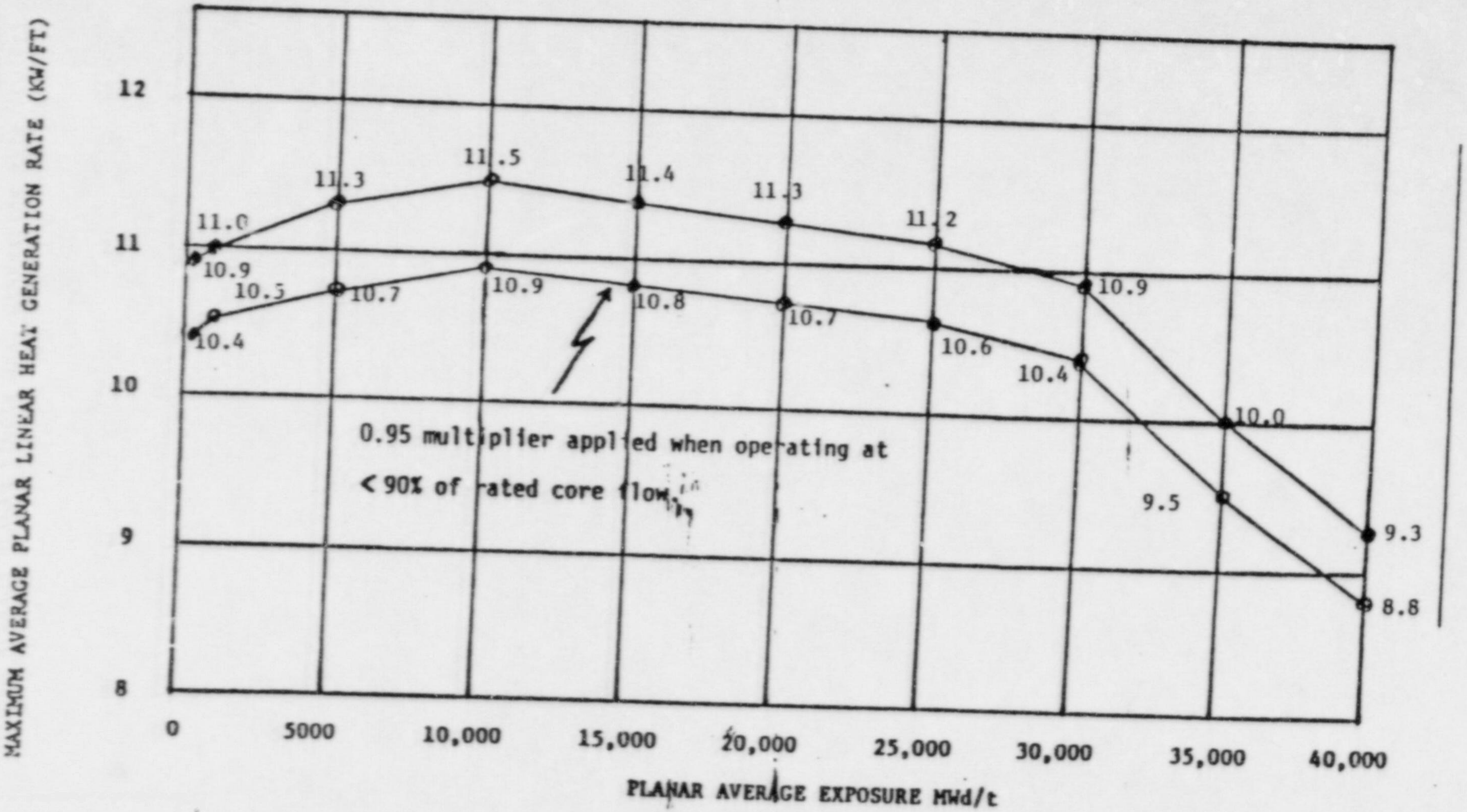

Figure 3.11.1a - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DB274H

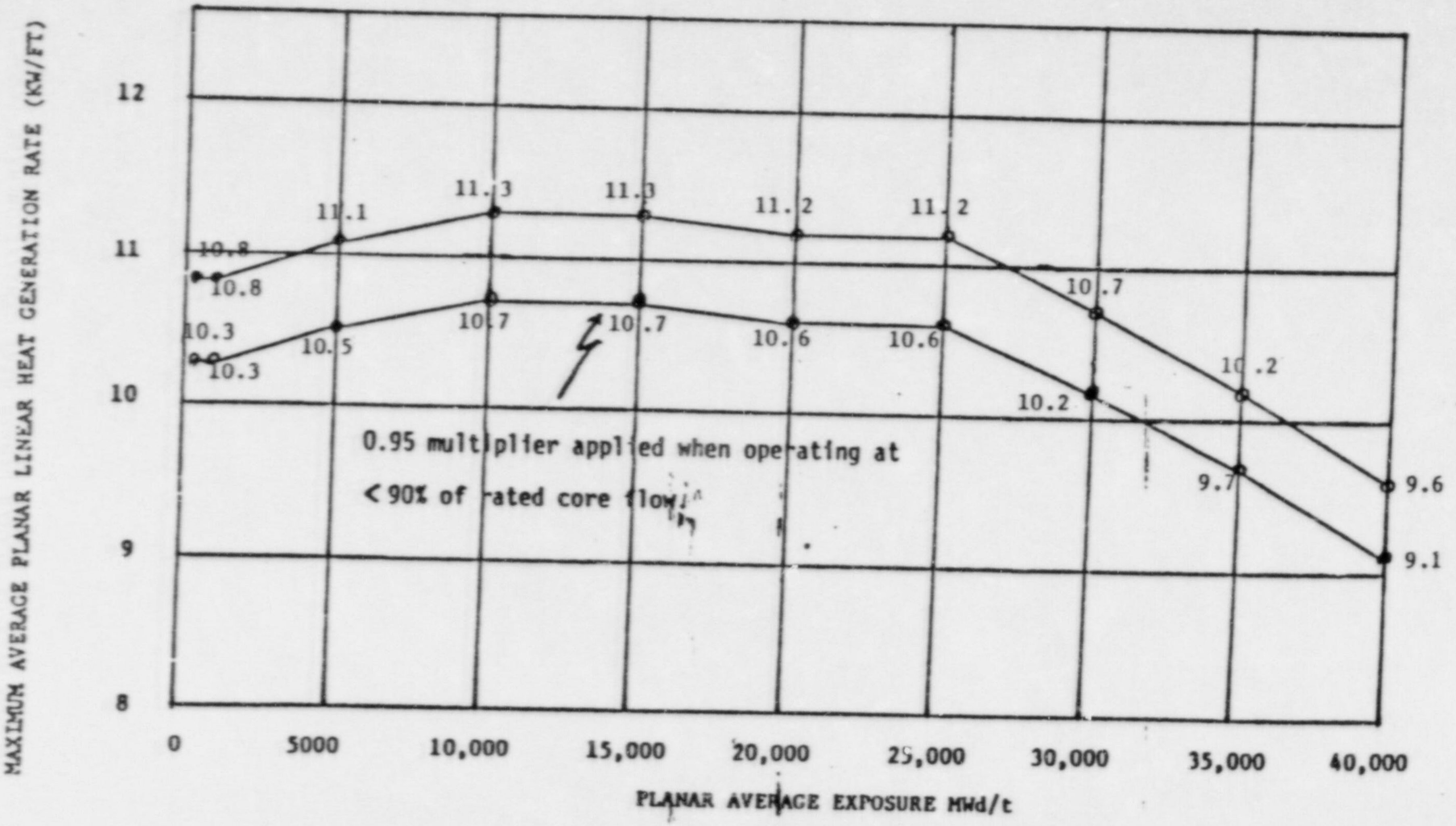

Figure 3.11.1b - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB265H

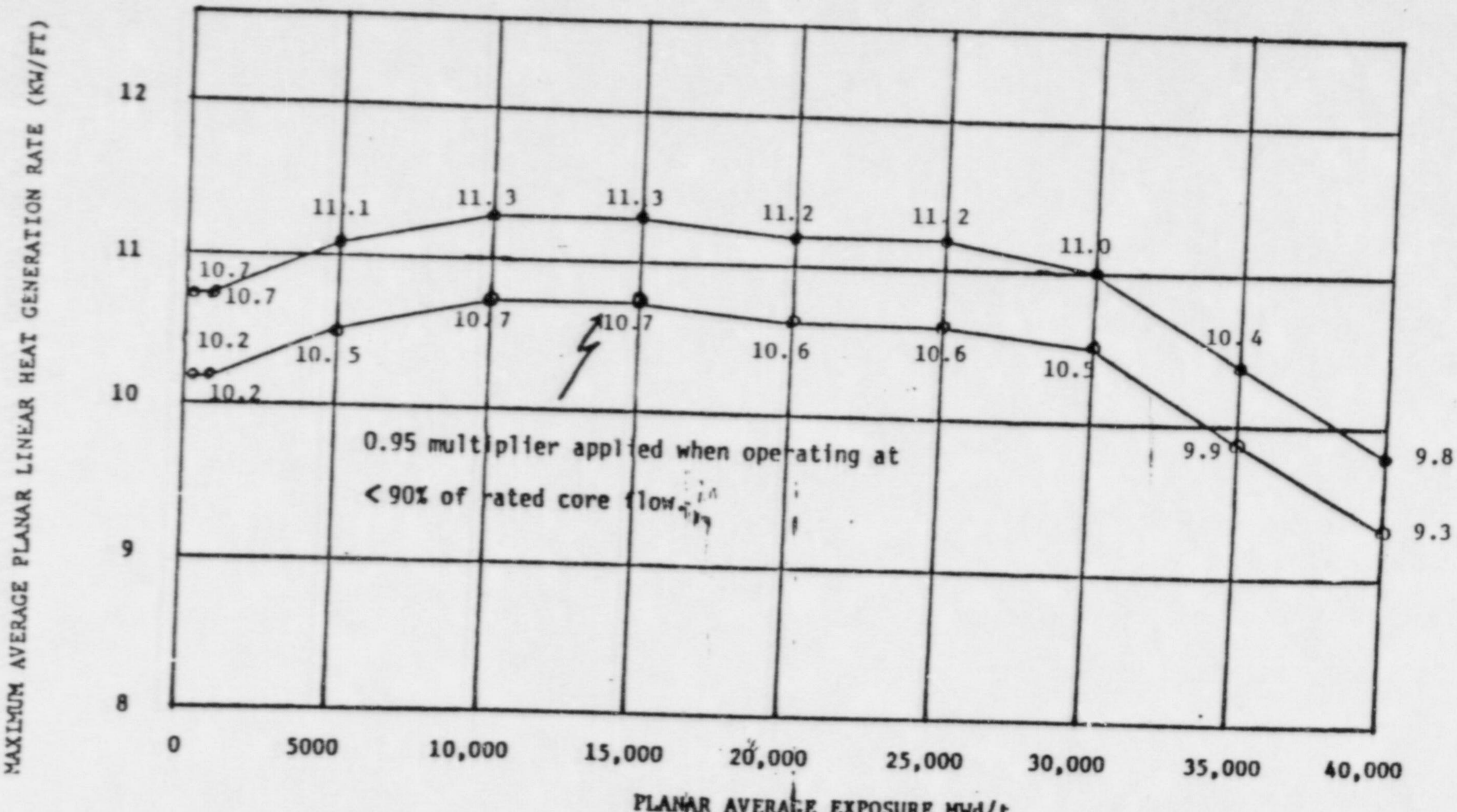

Figure 3.11.1c - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB282

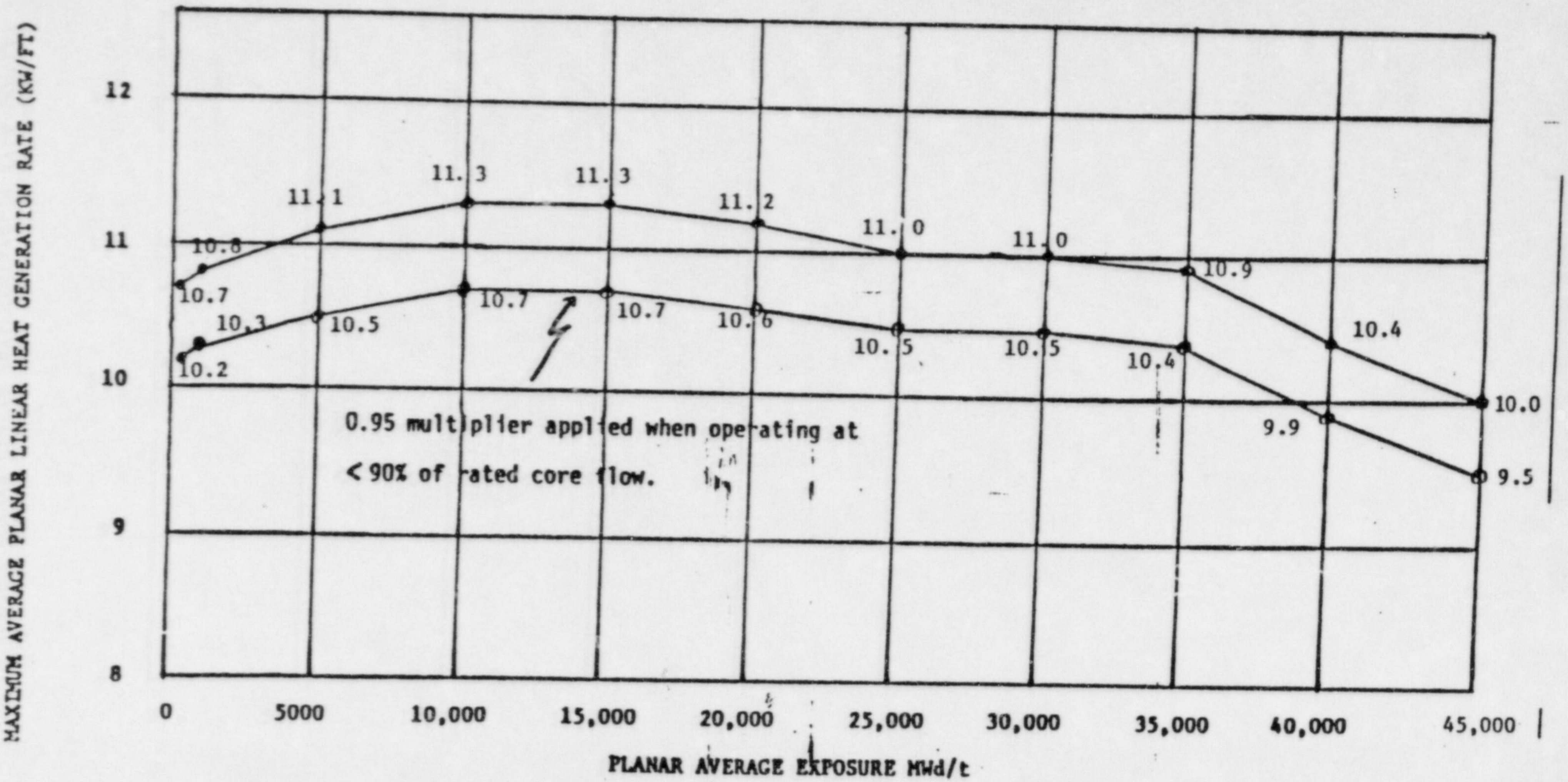

Figure 3.11.1d - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB283H

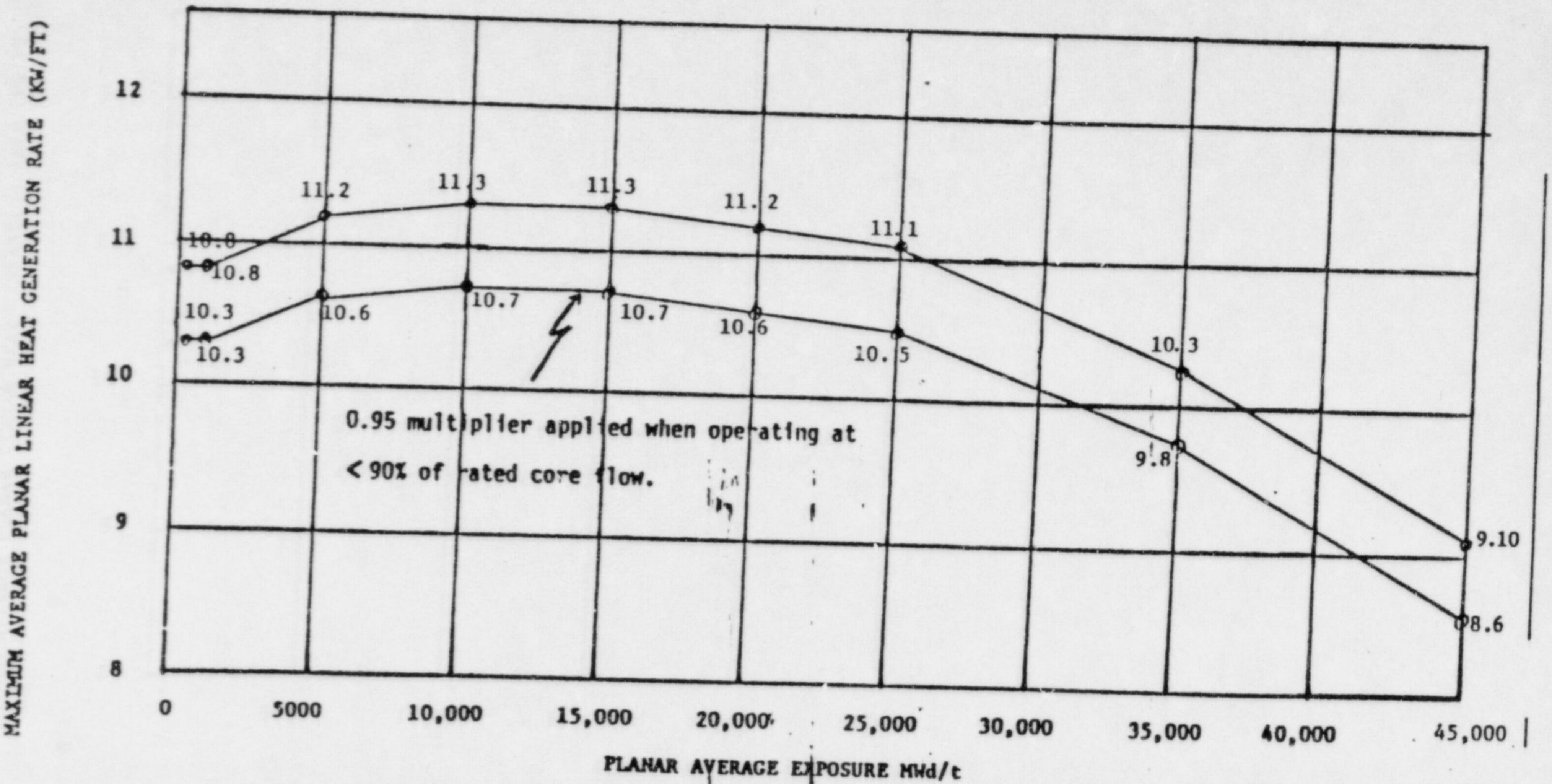

Figure 3.11.1e - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE BP8DRB300

LEFT BLANK INTENTIONALLY

LEFT BLANK INTENTIONALLY

TABLE 3.11.1

OPERATING LIMIT MCPR'S FOR CYCLE 10

(OPTION B)

BOC 10 TO EOC 10	EOC 10 TO 70% COASTDOWN	FUEL TYPE
1.40	1.40	8 x 8
1.43	1.43	P8 x 8R
1.43	1.43	BP8 x 8R

OPERATING LIMIT MCPR'S FOR CYCLE 10

(OPTION A)

BOC 10 TO EOC 10	EOC 10 TO 70% COASTDOWN	FUEL TYPE
1.45	1.45	8 x 8
1.48	1.48	P 8 x 8R
1.48	1.48	BP8 x 8R

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.14 PLANT SYSTEMS

Applicability:

Applies to the operational status of plant systems.

Objective:

To assure the operability of plant systems.

Specification:

A. Turbine Bypass System

The turbine bypass system shall be OPERABLE whenever the reactor is in the run mode. If the system is found to be inoperable, the reactor shall be taken out of the run mode within 12 hours.

4.14 PLANT SYSTEMS

Applicability:

Applies to the surveillance requirements of plant systems.

Objective:

To specify the type and frequency of surveillance to be applied to plant systems.

Specification:

A. Turbine Bypass System

The turbine bypass system shall be demonstrated OPERABLE at least once per cycle by performing a system functional test.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of the scram is to reduce the source of such radiation to the extent necessary to prevent excessive release of radioactive materials. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line provided the limit for a 15-minute period specified in specification 3.8 is not exceeded.**

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open in three out of four lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Ref. Section 11.3.7 FSAR.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.2 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM and APRM system provide protection against excessive power levels and short reactor periods in the refuel and Startup/Hot Standby modes. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Thus the IRM and APRM systems are required in the refuel and Startup/Hot Standby modes. In the power range the APRM provides the required protections; thus, the IRM system is not required in the Run mode.

The high reactor pressure, high drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have all scram functions except those listed in Note 7 of Table 3.1.1 operable in the Refuel and Shutdown mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system. As indicated in Note 11 of Table 3.1.1, no trip functions are required to be operable if all control rods are fully inserted, valved out and electrically disarmed, since this condition assures maximum negative reactivity insertion.

** Per errata sheet dated 10-7-70

The turbine condenser low vacuum scram is only required during power operation and must be bypassed to start up the unit. At low power conditions a turbine stop valve closure does not result in a transient which could not be handled safely by other scrams such as the APRM.

The requirement that the IRM's be inserted in the core when the APRM's read 3/125 or lower of full scale assures that there is proper overlap in the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation.

3.5 Bases

A. Core Spray and LPCI

This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analysis included in Section VI FSAR, either of the two core spray subsystems provides sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature (around 2000°F) to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%.

In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the coolant temperature has fallen to 330°F (90 psig).

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss of coolant accident. This system is completely independent of the core spray subsystem; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.2 square feet up to and including 5.8 square feet, the latter being the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (1). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified.

Although it is recognized that the information given in reference (1) provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

(1) APED 5736, Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards, April 1969, I. M. Jacobs and P. W. Marriott.

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 can be 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 all LPCI and core spray subsystems. 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.

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

For a crack size which gives a leakage rate of 2.5 gpm, the probability of rapid propagation is less than 10^{-5}. A leakage rate of 2.5 gpm is detectable and measurable.

The 25 gpm limit on total leakage to the containment was established by considering the removal capabilities of the pumps. The capacity of either of the drywell floor drain sump pumps is 50 gpm and the capacity of either of the drywell equipment drain sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the reactor coolant leak detection system will be evaluated during the first year of commercial operation and the conclusions of this evaluation will be reported to the AEC.

The main steam line tunnel leakage detection system is capable of detecting small leaks. The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the AEC.

E. Safety and Relief Valves

Present experience with the new safety/relief valves indicates that testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions: i.e., power relief or self-actuated by high pressure. The solenoid actuated function (automatic pressure relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to open. This function is discussed in Specification 3.5.D. In addition, the valves can be operated manually.

The safety function is performed by the same relief/safety valve with a pilot valve causing main valve operation.

It is understood that portions of the Valve Position Indication cannot be repaired or replaced during operation, therefore, the plant must be shutdown to accomplish such repairs. The 30-day period to do this allows the operator the flexibility to choose his time for shutdown; meanwhile, because of the redundancy provided by the valve discharge temperature monitor and the continued monitoring of the remaining valves by both methods, the ability to detect the opening of a safety/relief valve would not be compromised. The valve operability is not affected by failure of the Valve Position Indication System.

3.10 Bases

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with the fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position, only one control rod can be withdrawn.

With all control rods fully inserted, valved out and electrically disarmed, maximum negative reactivity insertion is assured, therefore, the refueling interlocks are not required to be operable.

For a new core, the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus, adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of three neutron induced count per second provides assurance that neutron flux is being monitored.

During unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality.

During the loading of an empty sourceless core, the special movable dunking type fission detectors will not detect a neutron count because of the lack of neutron sources. Therefore, fuel must be placed in the core to establish a neutron count rate. The restriction of eight fuel assemblies will minimize the probability of an inadvertent criticality prior to achieving 3 cps, while providing the flexibility to load fuel bundles around the portable detectors and SRM's.

C. Fuel Storage Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37' 9"), well above a level to assure adequate cooling (just above active fuel).

D. Crane Operability

The operability requirements of the crane used for handling of spent fuel casks ensures that the redundant features of the crane have been adequately inspected. The redundant hoist system ensures that a load will not be dropped for all postulated credible single-component failures.

E. Crane Travel

The restriction of movement of spent fuel casks over irradiated fuel ensures (in addition to the redundancy features) that a cask cannot be dropped on irradiated fuel assemblies.