50-245

PROPOSED TECHNICAL SPECIFICATION CHANGES FOR MILLSTONE UNIT NO. 1, RELOAD 8

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1. Reload 8/Cycle 9 Work and Extended Load Line Limit Evaluation

Description of Changes and Safety Evaluation Summary

The Reload 8/Cycle 9 work is very similar to previous reloads except for the extension in cycle length from 6580 MWD/ST to 8022 MWD/ST and the use of the recently approved transient code ODYN versus the traditional REDY code. The extended load line work was performed to justify a new rod block line and to justify normal operation in the region bounded by the rod block line.

The Reload 8/Cycle 9 work was performed in accordance with "General Electric Standard Application for Reactor Fuel" NEDE-24011-P-A-4, January 1982, which is an NRC approved methodology that applies to all General Electric reloads. The Extended Load Line Analysis was performed for Millstone Unit 1 Cycle 8, reverified for Cycle 9 by plant unique analyses and by building on generic work for various BWR product lines.

MAJOR CHANGES - Reload 8/Cycle 9

The extension of the planned cycle length for Cycle 9 required the utilization of 192 8 x 8 prepressurized retrofit fuel assemblies. Seventy-two of the bundles are 2.82 weight percent enriched with 7 gadolinia rods at 3 percent, and 120 fuel bundles are 2.83 weight percent enriched with 7 gadolinia rods at 4 percent. The additional fuel assemblies, heavier uranium loadings and heavier gadolina loadings are necessary to achieve the longer cycles desired at Millstone Unit No. 1 (8022 MWD/ST versus 6580 MWD/ST). This longer cycle will result in an operating cycle of 535 days assuming an 85 percent capacity factor. The major impact of the heavier loadings are decreased shutdown margin and reactivity swings during the cycle. The gadolina loadings provide adequate shutdown margin and hold the reactivity swings within the Technical Specification limits and design limits. Additional beginning of life shutdown margin is assured by the fact that Millstone Unit No. 1 operated beyond its end of life during Cycle 8 (7300 MWD/ST versus 6580 MWD/ST).

The End of Life prediction problem experienced in Cycle 8 has no effect on Cycle 9 limits since the amount of energy extracted carried the unit well into the licensing window.

The change from the REDY code to ODYN for pressurization transients is a major change for Millstone Unit No. 1. ODYN, a transient code that reflects reactor [illegible] dynamics, predicts more conservative values for peak pressure and peak heat fluxes. The Millstone Reload 8/Cycle 9 work shows higher pressures in the MSIV Event and higher heat fluxes in the Load Reject Event. These increases although typical of the changes expected from the switch to ODYN, cannot be explicitly attributed to ODYN due to other code changes.

The MSIV event for Reload 8/Cycle 9 shows a peak pressure of 1285 psig versus a Reload 7/Cycle 8 value of 1276 psig. 1285 psig is well within the acceptance criteria for pressure which is determined by the ASME Pressure Vessel Code. The code value for Millstone Unit No. 1 is 1375 psig, which is the basis of the Technical Specifications.

The limiting MCPR event for Millstone Unit No. 1 is the Load Reject Event which for Reload 8/Cycle 9 gives a MCPR of 1.48 versus a Reload 7/Cycle 8 value of 1.39. Although the plant will be operating with less freedom between the operating MCPR and the limiting MCPR, adequate margin should exist to prevent the MCPR from becoming limiting. An additional credit can be achieved if desirable by a systematic rod scram timing program which would have to be conducted every 120 days on 15 or more control rods. This margin improvement would bring the limit from 1.48 to 1.43.

For Reload 8/Cycle 9 the Rod Withdrawal Event (RWE) and the Rod Drop Accident (RDA) were eliminated and are now covered in the "General Electric Standard Application for Reload Fuel". The elimination of the RWE was approved by the NRC in the January 1982 version of the Standard Application. In February 1982, GE presented justification for not performing a plant specific RDA to the NRC. The justification included a statistical treatment of approximately 60 reloads with the rod worths being consistently lower than the limit of 1.27% ΔK for approximately 280 calories per gram. The peak fuel enthalpy in the study was 157 calories per gram. Millstone Unit No. 1 Reload 7/Cycle 8 demonstrated a much lower maximum rod worth of 0.86%ΔK. Thus, we are well within the envelope.

The specific Technical Specifications relating to the Reload are:

a. Figure 3.11.1 - This figure represents the exposure dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for the different fuel types at Millstone Unit No. 1. Each fuel type has its own curve which is based on the LOCA analysis performed for Millstone Unit No. 1. In Reload 8/Cycle 9, we have deleted one fuel type (8D274L) and added one fuel type (P8DRB283). Figure 3.11.1 was also revised to reflect the removal of the high burnup MAPLHGR limits previously imposed by the NRC due to fission gas concerns. The removal of these limits was approved on a generic basis (Generic Letter 82-03) on March 31, 1982 in a letter from Darrell G. Eisenhut to All Operating BWR's.

b. Section 3.11.b - Deletion of the power spiking penalty is justified as the spiking penalty is included in the design evaluation for the fuel (GESTARII, Section 2.4.2.1). With the new methods, the penalty is inherent in the design. Thus, a Technical Specification penalty would result in double accounting of the penalty.

c. Section 3.11.c - Minimum Critical Power Ratio - The revised Technical Specification contains dual operating limits for Cycle 9. This dual limit is due to the use of ODYN which has an option for additional credit for a less conservative scram insertion curve (Option B). If the plant were to exercise Option B, they would have to implement a specific rod timing program that would require surveillance on greater than 15 control rods every 120 days.

MAJOR CHANGES - Extended Load Line

The utilization of the Extended Load Line concept for Millstone is a major change due to the extensive redefinition of the normal operating envelope. The Extended Load Line Analysis (ELLA) report and Cycle 9 reverification justifies the expansion of the operating region and gives the underlying technical analyses. Previous analyses of this type, the Load Line Limit Analysis (LLLA), were more restrictive in scope, and operation at rated power and less than rated flow was not analyzed. However, ELLA for BWR/4's routinely included analyses at rated power and minimum flows of 91 to 94% of rated.

In early 1981, an ELLA was performed for a typical BWR/3 to support operation at rated power with flow as low as 87%. This work draws on the previous analyses to develop a set of restricted generic conclusions regarding applicability of the license basis safety analyses to operation within this expanded domain. It was further shown that Cycle 8 meets the conditions of validity of the generic conclusions, and the consequences of events initiated from within the extended domain are bounded by the consequences of the same events initiated from the license basis condition.

The present operating region for Millstone Unit No. 1 is refined by Figure 3.3.1 of the Technical Specifications and is valid for an area bounded by the recirculation pump minimum flow line, the present rod block LCO line up to 80 percent flow and 92.4 percent power and a Rod Intercept Line to the 100 percent flow/100 percent power.

New Figure 3.3.1 extends the normal operating region to 100 percent power and 87 percent flow with the boundaries being the Rod Block LCO Line and an intersecting line at 100 percent power. In defining a new region, we have also redefined the slope of the Rod Block and Scram Lines from 0.65W to 0.58W. This line is somewhat less conservative than the existing line but is shown to be bounded by the 100 percent flow/100 percent power intersect.

The anlaysis was done generically, verified for Reload 7/Cycle 8 and reverified for Reload 8/Cycle 9. In all cases, the existing reload analyses are more limiting than the reduced flow case. The core parameter that dominates this behavior is increased scram activity. At reduced flow, the power shape tends to skew toward the bottom of the core which increases the effectiveness of the control rods in terminating the transient. The effect of the LOCA analysis for low flow domains was generically reviewed and approved by the NRC in 1978. The Millstone LOCA work is covered in this analysis.

The Technical Specification affected by the ELLA report are:

a. Section 2.1.1.A - APRM Flux Settings

The plant is limited by and analyzed to the Full Flow/Full Power Cases. The trips that utilize a Flux Scram only credit the 120 percent scram. Thus, the flow bias is an anticipatory trip and not limiting.

b. Figure 2.1.2 - New APRM and Rod Block Curves

These new curves are within the analyzed region and are not limited with respect to the safety analyses. The same reasoning applies for (b) and for (a).

c. Page 2-3, Section 2.1.2.A

The scope redefinition remains the same as items (a) and (b). The other change on this page is the deletion of the definition of total recirculation flow as 29.7×10^6 lbm/Hr. The term 29.7×10^6 lb/Hr. is incorrect in that the recirculation flow necessary to achieve 100 percent core flow changes with each reload. A more appropriate definition is given on page 2-3, which recognizes the variable requirement and not to exceed number.

d. Pages 2-5 and 2-6 - Repeats of a, b, and c.

e. Table 3.2.3 - Section a, b and c.

f. Figure 3.3.1 - Millstone Unit No. 1 Power Flow Map

This change explicitly defines the new operating region as discussed in the ELLA submittal and in this evaluation.

g. Figure 3.11.2 - Multiplier for Utilizing Various Recirculation Flow Maximums

The ELLA report only justifies having a maximum scoop tube setpoint of 102.5 percent.

A technical review of these changes has found them to be acceptable. A safety evaluation has also been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions. The Millstone Unit No. 1 Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

2.1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is greater than 800 psia and the core flow is greater than 10% of rated design, a minimum critical power ratio (MCPR) less than 1.07 shall constitute a violation of the fuel cladding integrity safety limit.

B. When the reactor pressure is less than or equal to 800 psia or reactor flow is less than 10% of design, the reactor thermal power transferred to the coolant shall not exceed 25% of rated.

C. 1. To assure that the Limiting Safety System Settings established in Specifications 2.1.2A and 2.1.2B are not exceeded, each required scram shall be initiated by its primary source signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.2 and shall be:

$$S \leq 0.58\ W + 62$$

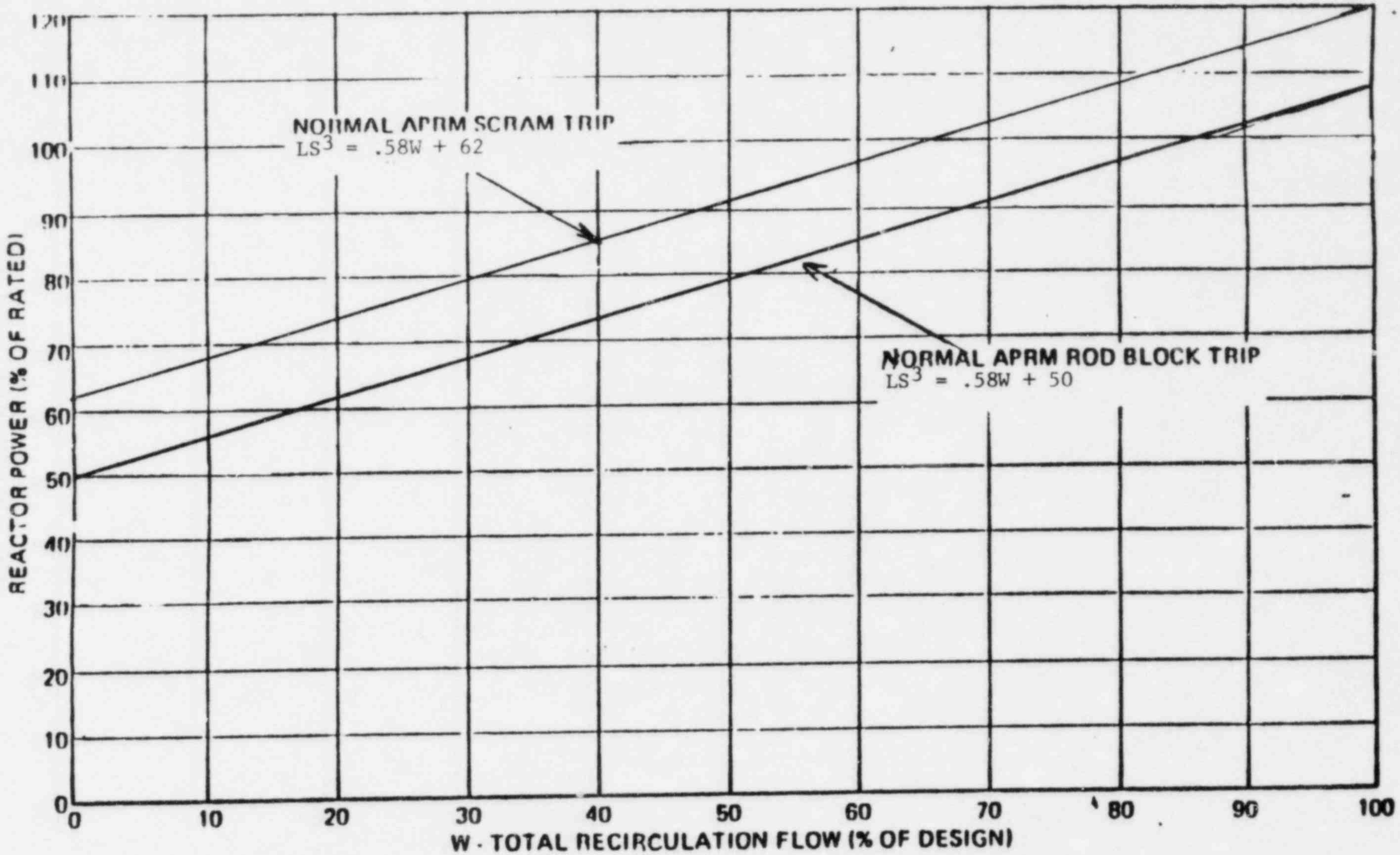


Figure 2.1.2 MILLSTONE APRM SCRAM AND ROD BLOCK TRIP LIMITING SAFETY SYSTEM SETTINGS

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.2A and a control rod scram does not occur.	where:
	S = Setting in percent of rated thermal power (2011 MWt)
	W = Total recirculation flow in percent of design See Note (1)
D. Whenver the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored.	The trip setting shall not exceed 90 percent of rated power during generator load rejections from an initial generator power greater than 307 MWe. The APRM scram setdown shall be 90% of rated within 30 seconds after initiation of full load rejection.
	b. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:
	$S \leq (0.58\,W + 62)\left[\frac{FRP}{MFLPD}\right]$
	where,
	FRP = fraction of rated thermal power (2011 MWt)

Note (1) Design flow to be defined as the recirculation flow (not to exceed 33.48×10^6 lbs/hr.) needed to achieve 100% core flow.

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

2. <u>APRM Reduced Flux Trip Setting (Refuel or Startup/Hot Standby Mode)</u>

When the mode switch is in the refuel or Start Up/Hot Standby position, the APRM scram shall be setdown to less than or equal to 15% of rated thermal power. The IRM scram trip setting shall not exceed 120/125 of full scale.

B. 1. **APRM Rod Block Trip Setting**

a. The APRM rod block trip setting shall be as shown in Figure 2.1.2 and shall be: (Run Mode)
$S_{RB} \leq 0.58W + 50$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2011 MWt).

W = Total recirculation flow in percent of design (Note 1, Page 2-3).

b. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$S_{RB} \leq (0.58W + 50)\ [\frac{FRP}{MFLPD}]$

where:

FRP = fraction of rated thermal power (2011 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8, 8x8R and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

c. During power ascensions with power levels less than or equal to 90%, APRM Rod Block Trip Setting adjustments may be made as described below, provided that the change in scram setting adjustment is less than 10% and a notice of the adjustment is posted on the reactor control panel:

The APRM meter indication is adjusted by:

$ARPM = (\frac{MFLPD}{FRP})\ P$

where:

APRM = APRM Meter Indication

P = % Core Thermal Power

Amendment No. 10, 66, 76

2.1.2 Bases:

The transients expected during operation of the Millstone Unit 1 have been analyzed up to the thermal power condition of 2011 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.3.1 (2). In addition, 2011 MWt is the licensed maximum steady-state power level of Millstone Unit 1. This maximum steady-state power will never be knowingly exceeded.

Conservatism was incorporated by conservatively estimating the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for the evaluation of reactor dynamics performance. Comparisons have been made showing results obtained from a General Electric boiling water reactor and the predictions made by the model. The comparisons and results are summarized in Reference 1.

The void reactivity coefficient utilized in the analysis is conservatively estimated to be about 25% larger than the most negative value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 80% of the control rods. The scram delay time and rate of rod insertion are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity strongly turns the transient and the stated 5% and 20% insertion times conservatively accomplished this desired initial effect. The time for 50% and 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients, MCPRs specified in Section 3.11.C are conservatively assumed to exist prior to initiation of the transients.

(1) Linford, R. B. "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802.

(2) "Extended Load Line Limit Analysis, Millstone Point Nuclear Power Station, Unit 1" NEDO-24366 and NEDO 24366-1.

TABLE 3.2.3

INSTRUMENTATION THAT INTIATES ROD BLOCK

Minimum Number of Operable Inst. Channels per Trip System (1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	See Specification 2.1.2B
2	APRM Downscale	$\geq$ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	$\leq$ .58W + 50 (2)
1 (6)	Rod Block Monitor Downscale	$\geq$ 3/125 Full Scale
3	IRM Downscale (3)	$\geq$ 3/125 Full Scale
3	IRM Upscale	$\leq$ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	$\leq 10^5$ counts/sec.

(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 recirculation flow required to achieve rated core flow expressed in percent.

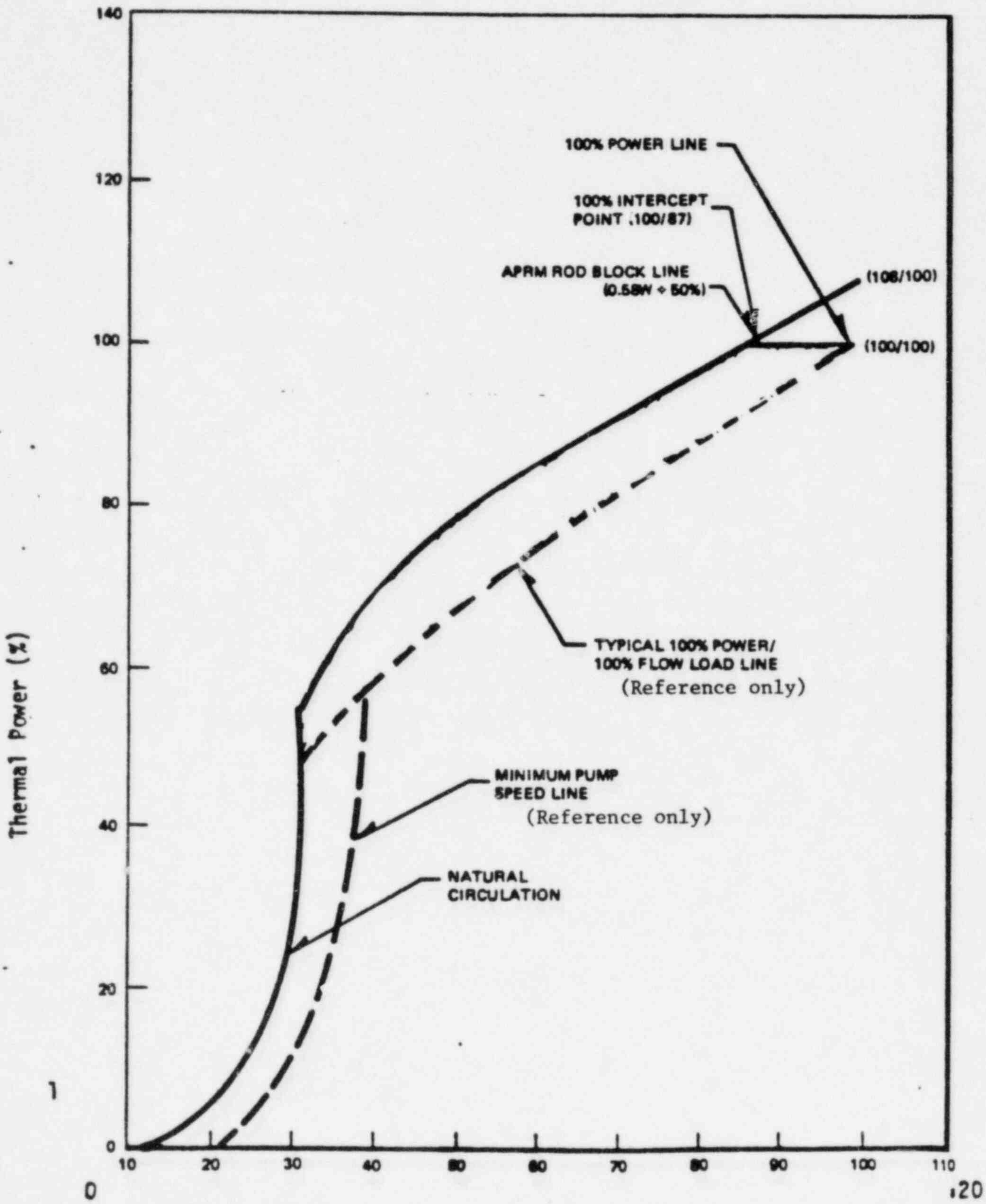
(3) IRM downscale may be bypassed when it is on its lowest range.

(4) This function may be bypassed when the count rate is $\geq$ 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.

(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.

FIGURE 3.3.1



Amendment No. 32
(corrected)

Ref. NEDO 24366 Sept. 1981

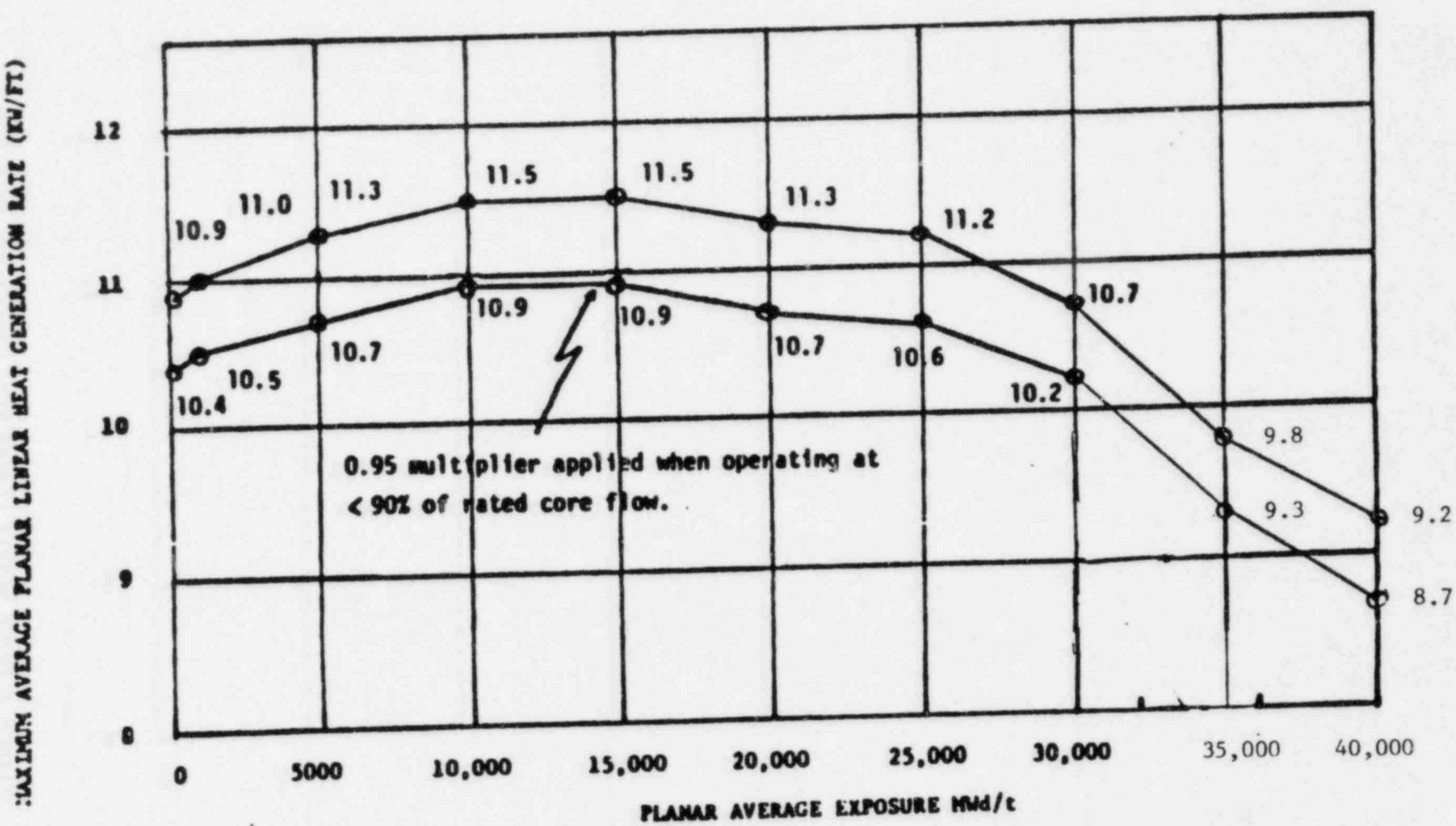


Figure 3.11.1a - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DB262.

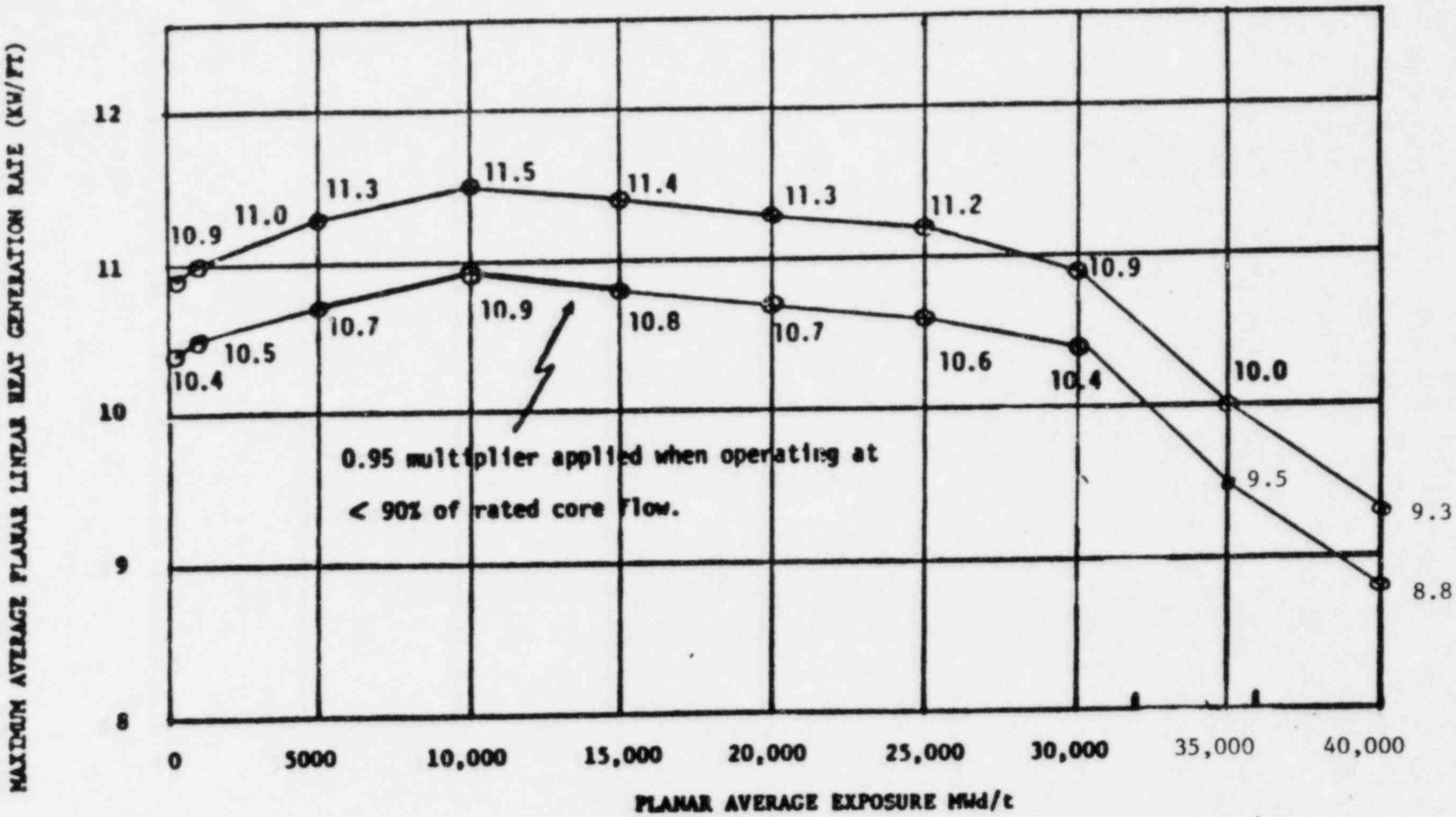


Figure 3.11.1b - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DB274H.

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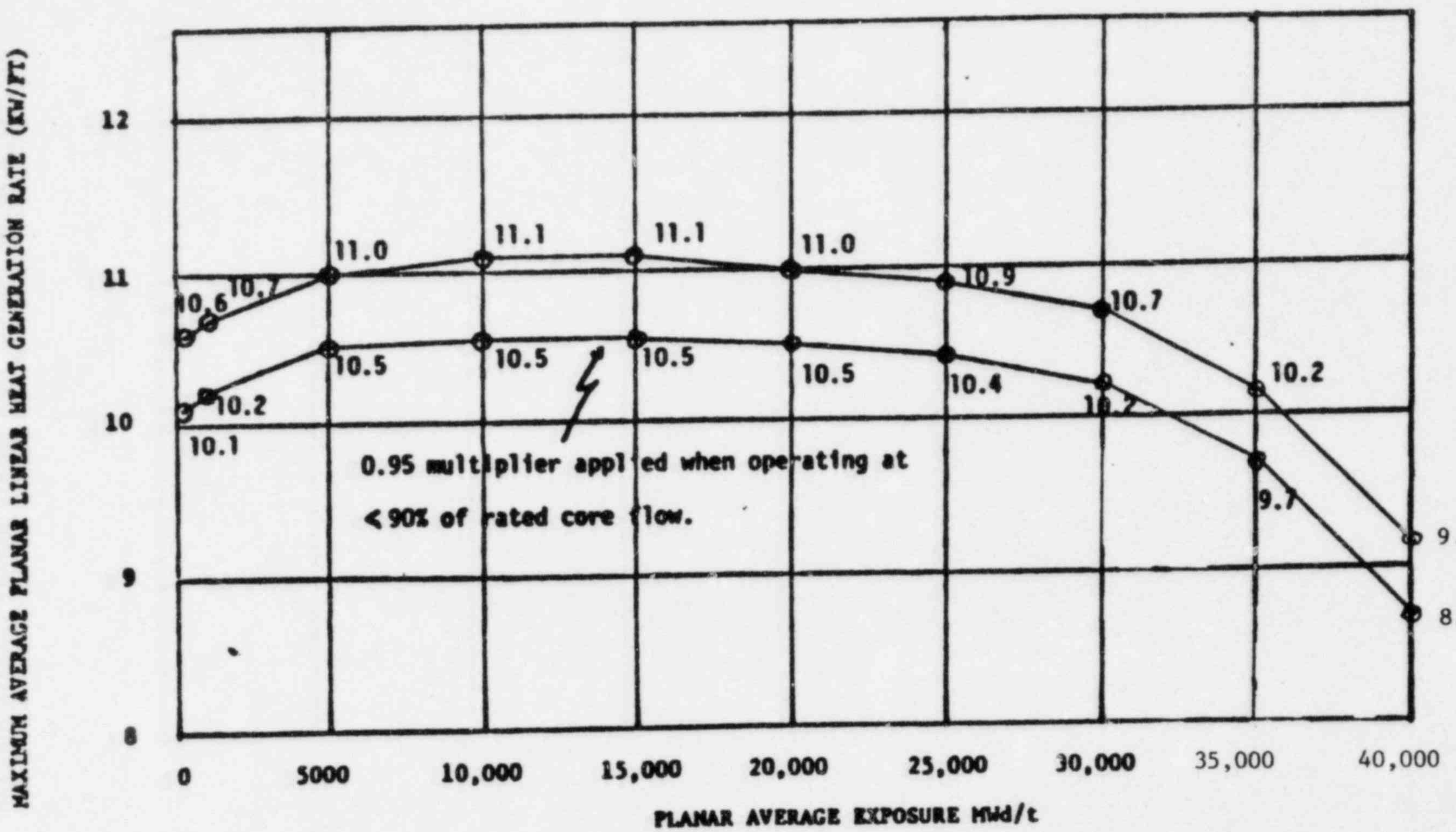


Figure 3.11.1c - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DRB265L.

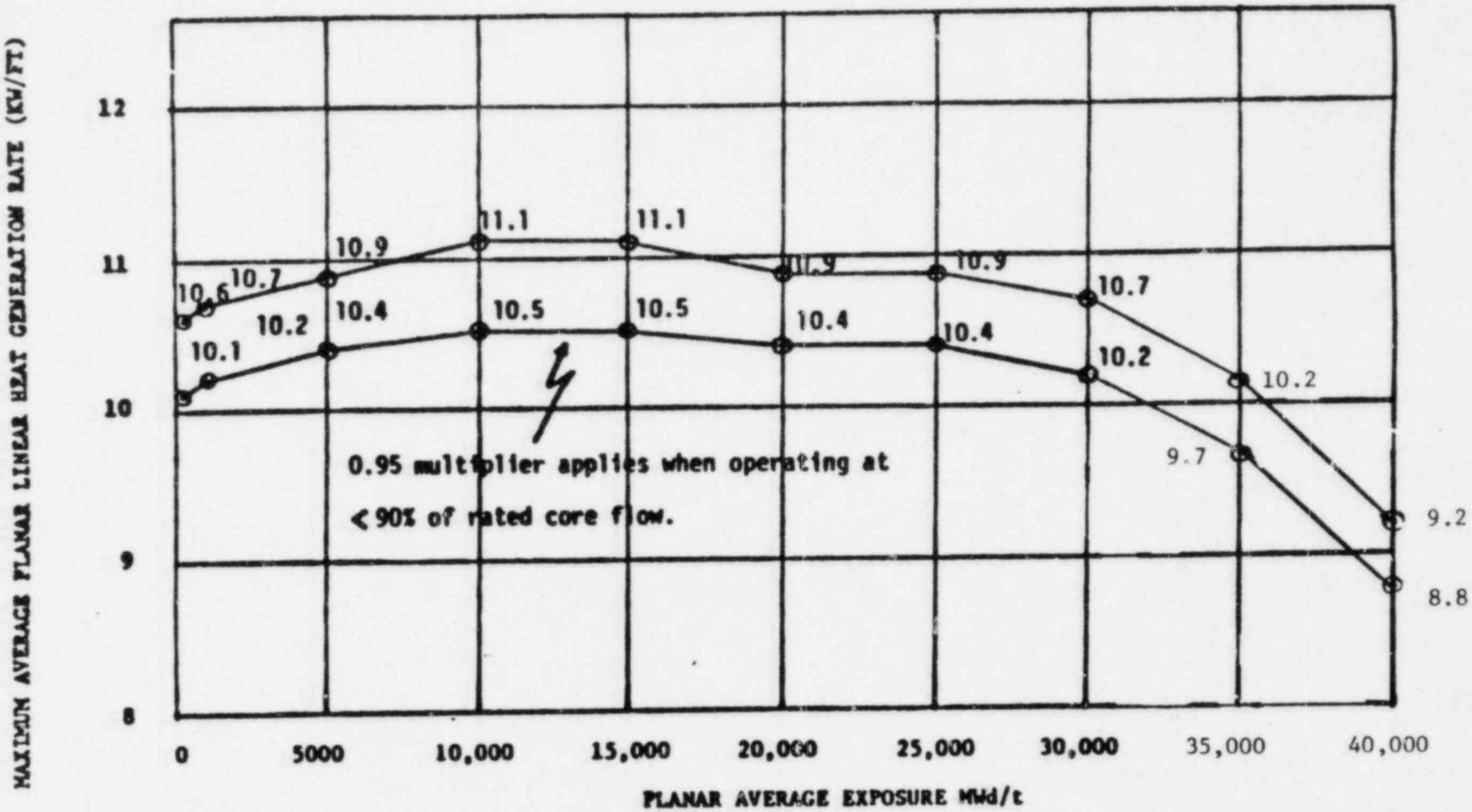


Figure 3.11.1d - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DRB265H.

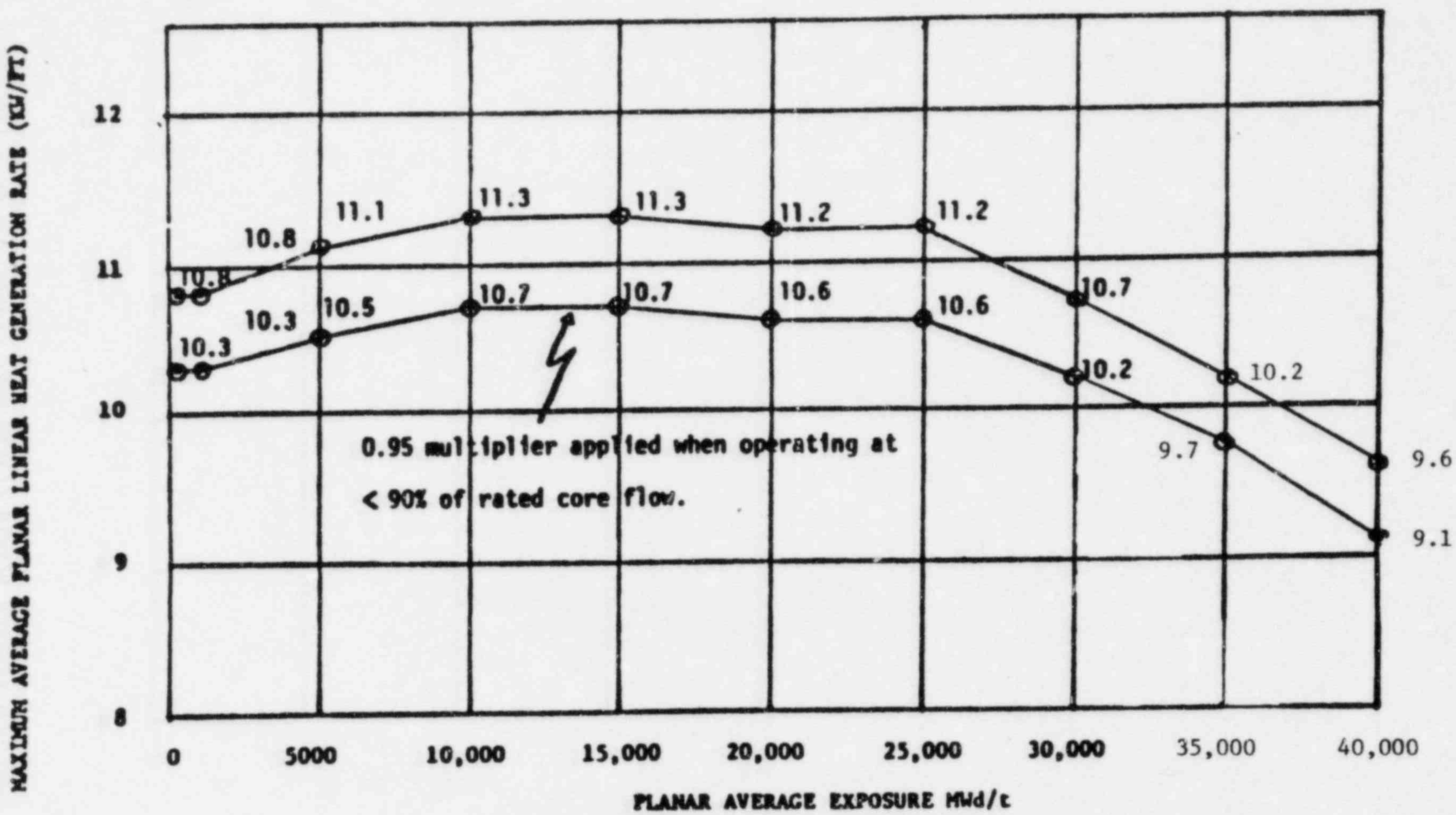


Figure 3.11.1e - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB265H.

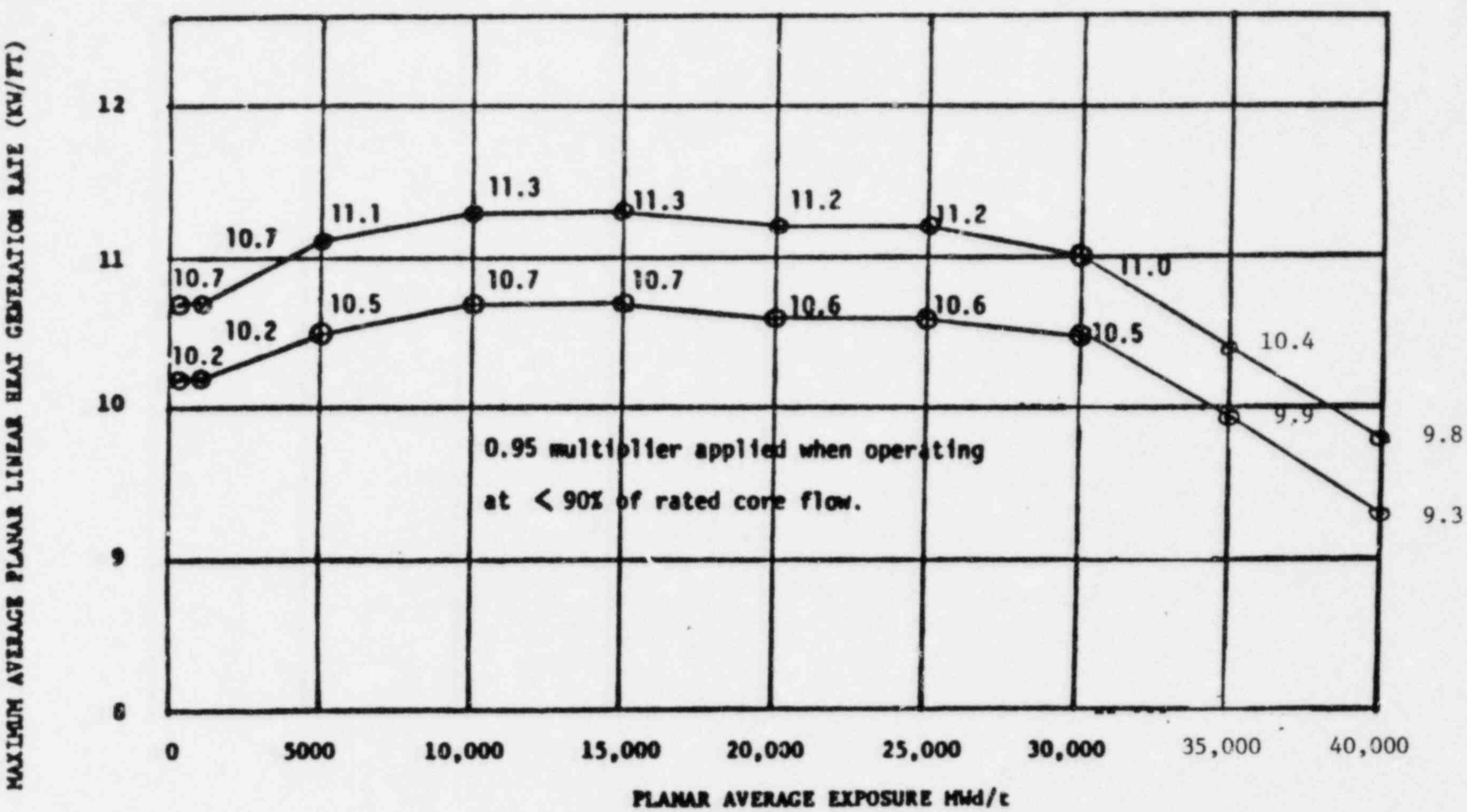


Figure 3.11.1f - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB282.

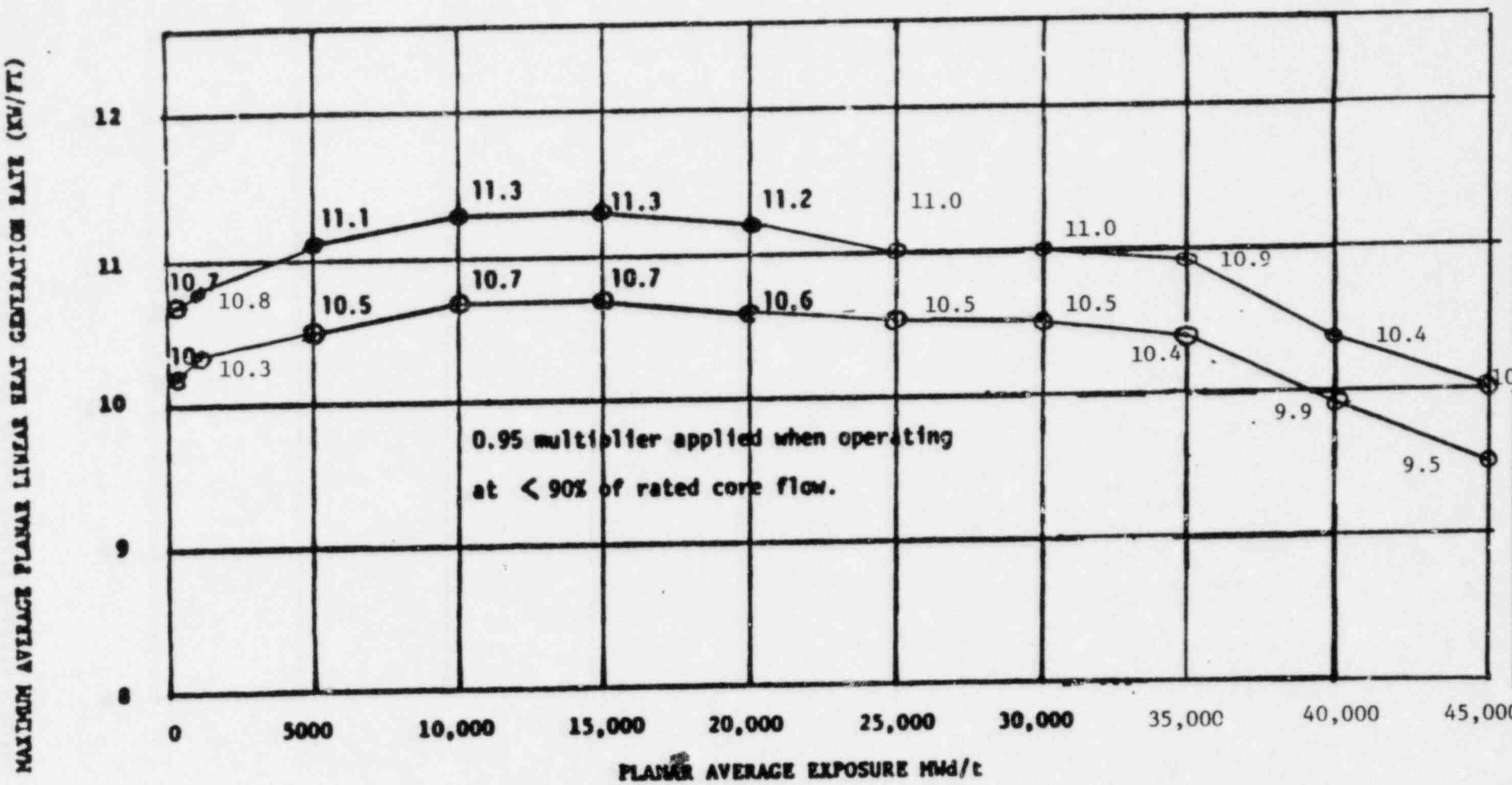


Figure 3.11.1g - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB283.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Linear Heat Generation Rate (LHGR) During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR of 13.4 KW/ft for 8 x 8 fuel bundles. **During power operation, the LHGR shall not exceed the limiting value. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.**	B. Linear Heat Generation Rate (LHGR) The LHGR **shal** be checked daily during reactor **operation** at $\geq$ 25% rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Minimum Critical Power Ratio (MCPR) During power operation, MCPR shall be as shown in Table 3.11.1. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR 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. For core flows other than rated the MCPR's in Table 3.11.1 shall be multiplied by K_f, where K_f is as shown in figure 3.11.2. D. If any of the limiting values identified in Specifications 3.11.A, B, or C. are exceeded, even if corrective action is taken, as prescribed, a Reportable Occurrence report shall be submitted.	C. Minimum Critical Power Ratio (MCPR) 1. MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for specification 3.3.B.5. 2. Utilization of Option B Operating limit MCPR values requires the scram time testing of 15 or more control rods on a rotating basis every 120 operating days.

TABLE 3.11.1

OPERATING LIMIT MCPR'S FOR CYCLE 9

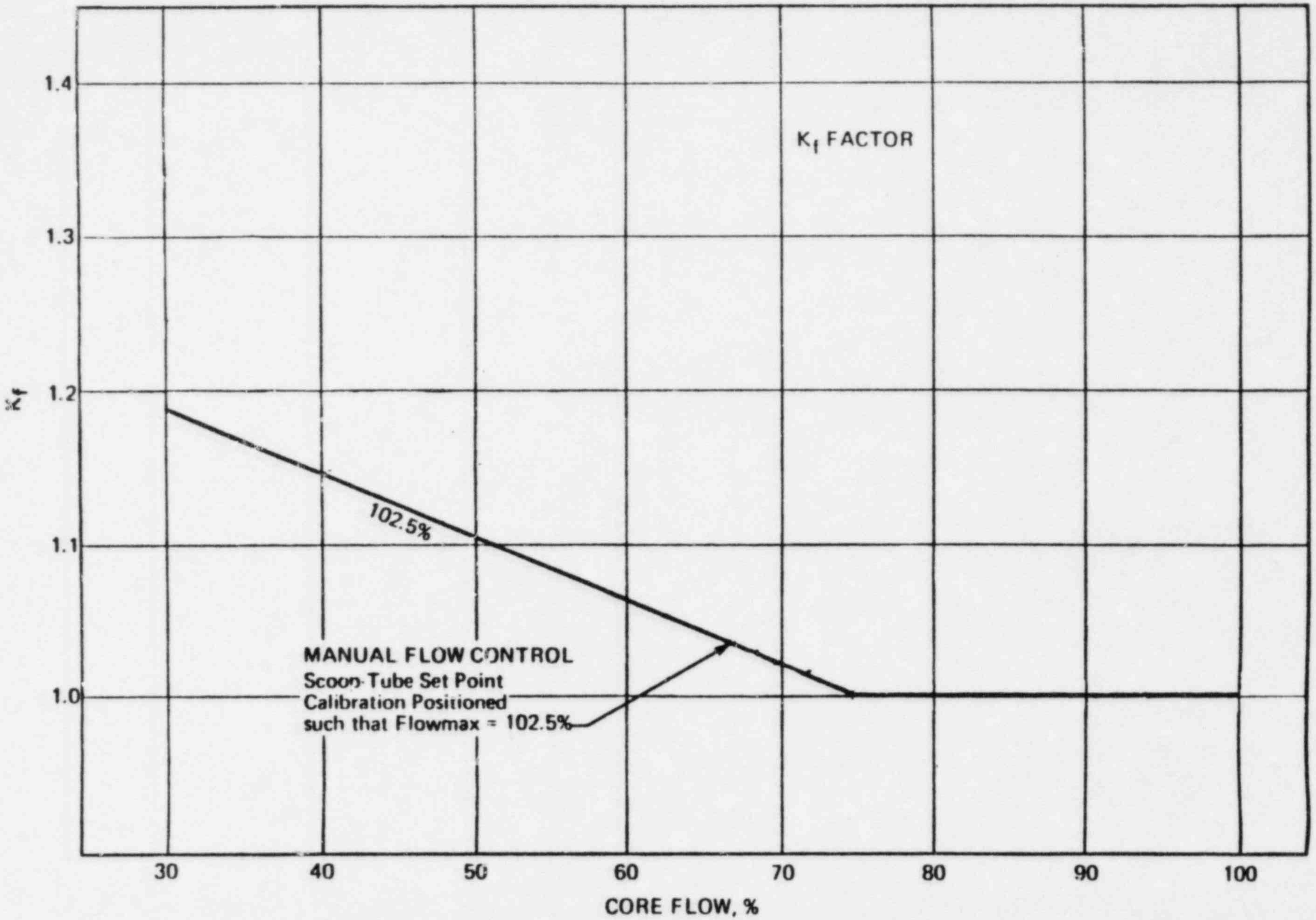
(OPTION B)

BOC9 TO EOC 9	EOC9 TO 70% COASTDOWN	FUEL TYPE
1.40	1.40	8 x 8
1.40	1.40	8 x 8R
1.43	1.43	P8 x 8R

OPERATING LIMIT MCPR'S FOR CYCLE 9

(OPTION A)

BOC9 TO EOC 9	EOC9 TO 70% COASTDOWN	FUEL TYPE
1.45	1.45	8 x 8
1.45	1.45	8 x 8R
1.48	1.48	P8 x 8R



K_f FACTOR
K_f
1.4
1.3
1.2
1.1
1.0
102.5%
MANUAL FLOW CONTROL
Scoop-Tube Set Point
Calibration Positioned
such that Flowmax = 102.5%
30
40
50
60
70
80
90
100
CORE FLOW, %

Figure 3.11.2

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The requirement of at least 3 counts per second assures that adequate monitoring capability is available. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's less than 1.07. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is that, resulting from a generator load rejection coincident with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.06. Amendment 21 shows the control rod scram reactivity insertion data used in analyzing the transients. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the reactivity insertion rate effects of rods of long scram times are minimized. Grouping of long scram time rods is prevented by not allowing more than one control rod in any group of four control rods to have long insertion times. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no more than 10% of the operable rods to have long scram times. In the analytical treatment of the transient, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Specification 4.1.

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 Figure 3.11.1.

Conservative LOCA calculations predict that nucleate boiling will be maintained for several seconds following a design basis LOCA. This results in early removal of significant amounts of stored energy which, if present later in the transient, when heat transfer coefficients are considerably lower, would result in higher peak cladding temperature. As core flow is reduced below about 90%, the time of onset of boiling transition makes a sudden change from greater than about 5 seconds to less than 1 second. The approved ECCS evaluation model requires that at the first onset of local boiling transition, the severely reduced heat transfer coefficients must be applied to the affected planar area of the bundle, and thus exaggerates the calculated peak clad temperature. The effect is to significantly reduce the energy calculated to be removed from the fuel during blowdown. This results in an increase in calculated peak clad temperature of about 100°F which can be offset by a 5% reduction in MAPLHGR. For flows less than 90% of rated, a 5% reduction in the MAPLHGR limits in Figure 3.11.1, derived for 100% flow will assure that the plant is operated in compliance to 10 CFR 50.46 at those lower flows.

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. The LHGR shall be checked daily during reactor operation at ≥ 25% power to determine if fuel

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burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. The steady state value for MCPR was selected to provide a margin to accomodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. This value ensures that;

1. For the initial conditions of the LOCA analysis a MCPR of 1.18 is satisfied. For the low flow ECCS analysis, an initial MCPR of 1.24 is assumed, and

2. For any of the special transients or disturbances caused by single operator error or single equipment malfunction the value of MCPR is conservatively assumed to exist prior to the initiation of the transient or disturbance.

At core thermal power levels $\leq$ 25%, the reactor will be operating at minimum recirculation pump speed, and moderator void content will be very small. For all designated control rod patterns which may be employed at this power, thermal hydraulic analysis indicates that the resultant MCPR value is in excess of requirements. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculation of MCPR at greater than 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

The use of the Option B operating limit MCPR requires additional SCRAM time testing and verification in accordance with GE letter A. D. Vaughn to P. A. Blasioli, July 9, 1982 Proposed Technical Specification Changes for Millstone Unit 1.

D. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times or corrected to within the limiting values of MAPLHGR, LHGR, and MCPR within 2 hours of the time the plant is determined to be exceeding them. It is a requirement, as stated in Specifications 3.11.A, B, and C that if at any time during power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is to be initiated within 15 minutes if normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be logged and a reportable occurrence issued. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

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2. Protective Instrumentation

Description of Changes and Safety Evaluation Summary

This change reduces the turbine low pressure setpoint which initiates MSIV closure from 880 psig to 825 psig. This change is consistent with GE recommendations in SIL 130 and also is part of our long-term fix to preclude water hammer in the isolation condenser. (See W. G. Council letter to D. M. Crutchfield dated April 27, 1982).

Since no credit is taken for this trip for steam line breaks, the only analysis potentially impacted is the rapid depressurization transient resulting from pressure regulator malfunction with maximum demand. It was determined that the vessel cooldown rate would not be affected by the setpoint change and the original analysis is still valid.

A technical review of these changes has found them to be acceptable. Also, a safety evaluation has been performed in accordance with 10CFR50.59 and has conlcuded that these changes do not constitute any unreviewed safety questions. The Millstone Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTINGS
	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 260 milliseconds. The maximum setting of the time delay relays which bypass this scram shall be 260 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.

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum Number of Operable Instrument Channels Per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	≥ 127 inches above top of active fuel	A
2	Reactor Low Low Water	79 (+4-0) inches above top of active fuel	A
2 (4)	High Drywell Pressure	≤ 2 psig	A
2 (2) (5)	High Flow Main Steamline	≤ 120% of rated steam flow	B
2 of 4 in each of 2 subchannels	High Temperature Main Steamline Tunnel	≤ 200°F	B
2	High Radiation Main Steamline Tunnel	≤ 7 times normal rated power background	B
2	Low Pressure Main Steamlines	≥ 825 psig	B
2	High Flow Isolation Condenser Line	164 inches ≥ trip setting (water differential on steam line) ≥ 150 inches. 44 inches ≥ trip setting (water differential on water side) ≥ 35 inches.	C

(1) Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the Run position.

(2) Per each steamline.

(3) 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 an orderly shutdown and have reactor in cold shutdown condition in 24 hours.
B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.
C. Close isolation valves in isolation condenser system.

(4) May be bypassed when necessary by closing the manual instrument isolation valve for PS-1621, A through D, during purging for containment inerting or deinerting.

(5) Minimum number of operable instrument channels per trip system requirement does not have to be met for a steamline if both containment isolation valves in the line are closed.

G. Main Steam Line Isolation Valve Closure Scram

The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure, there is no increase in neutron flux during valve closure.

H. Main Steam Line Low Pressure Initiates Main Steam Isolation Valve Closure

The low pressure isolation at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal-hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge and sump isolation valves, and reactor building ventilation isolation valves. Group 2 actuation also initiates the SBGTS. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring the steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow in conjunction with the flow limiters and main steamline closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity to the environs is well below 10 CFR 100 guideline values. The main steamline high flow break detection is a one out of two twice logic for each individual steamline, four detectors per line for a total of 16 detectors. When a steamline is isolated by closing both main steam isolation valves the operable instrument channels per trip system requirements are not required to be met because the protection afforded by the remaining operable logic in the in-service steamlines provides complete recognition of the steam flow measurements required for correct protective action.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close to prevent further release to the environment. With the established setting of seven times normal background, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guideline values are not exceeded for the most rapid failure mechanism postulated (control rod drop accident).

Pressure instrumentation is provided which trips when main steamline pressure at the turbine drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel," "Shutdown," and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there is no release of fission products other than those in the reactor water.

Amendment No. ~~A~~, ~~78~~

3. Instrumentation that Initiates Rod Blocks

Description of Changes and Safety Evaluation Summary

This change reduces the minimum number of operable channels for APRM upscale and downscale rod block per trip system from two (2) to one (1). However, a minimum of four (4) operable or operating APRM changes must still exist. Because of the present channel assignments of the APRMs for the RPS (Channel A uses APRM 1, 2, and 3 and Channel B uses APRM 4, 5, and 6) and for the rod block function (Channel A uses APRM 1, 3, and 5 and Channel B uses APRM 2, 4, and 6), it is possible to bypass one RPS Channel A APRM and one RPS Channel B APRM and violate the minimum operable channels requirement for the rod block function.

This technical specification change will permit any two of the six APRM channels which make up APRM rod block channels A and B to be placed in bypass. Since one out of three APRM channels upscale or downscale in either APRM rod block channel will provide the block function, there is no single failure which will disable the block with two APRM channels in bypass (even when both are in the same APRM rod block channel). Only one APRM channel per RPS channel may be in bypass; this has not changed. The proposed change does not alter the original design bases for rod block or trip function but will permit two APRM channels in the same APRM rod block channel to be in bypass.

A technical review of these changes has found them to be acceptable. Also, a safety evaluation has been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions. The Millstone Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

TABLE 3.2.3

INSTRUMENTATION THAT INTIATES ROD BLOCK

Minimum Number of Operable Inst. Channels per Trip System (1)	Instrument	Trip Level Setting
1 (7)	APRM Upscale (Flow Biased)	See Specification 2.1.2B
1 (7)	APRM Downscale	$\geq$ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	$\leq$ .65 w + 42 (2)
1 (6)	Rod Block Monitor Downscale	$\geq$ 3/125 Full Scale
3	IRM Downscale (3)	$\geq$ 3/125 Full Scale
3	IRM Upscale	$\leq$ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	$\leq 10^5$ counts/sec.

(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 $\geq$ 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.

(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.

(7) There must be a total of at least four (4) operable or operating APRM channels.

4. Special Report

Description of Changes and Safety Evaluation Summary

This change deletes the requirement to submit a special report regarding the results of the secondary containment leak rate test, which is still required to be performed in accordance with Specification 4.7.C. The NRC will still be notified if the test is not satisfactorily performed via a LER. Deletion of the requirement to submit the results of this test does not in any way adversely affect safety.

A technical review of these changes has found them to be acceptable. Also, a safety evaluation has been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions. The Millstone Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

THIRTY-DAY WRITTEN REPORTS (Continued)

completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety features instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in 6.9.1 B.c, above, designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. In-service Inspection Results, Specification 4.6.F.

b. Primary Containment Leak Rate Test Results, Specification 4.7.A.2.

c. (Deleted).

d. Materials Radiation Surveillance Specimen Examination and Results, Specification 4.6.B.3.

e. Fire detection instrumentation, Specification (3.12.E.2).

f. Fire suppression systems, Specifications (3.12.A.2, 3.12.B.2, 3.12.C.2 and 3.12.C.4).

5. Containment Systems

Description of Changes and Safety Evaluation Summary

This change simply corrects a typographical error and is therefore only administrative in nature.

A technical review of these changes has found them to be acceptable. Also a safety evaluation has been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions. The Millstone Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. Primary containment integrity as defined in Section 1.0 shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).	3. The primary containment integrity shall be demonstrated as follows: a. Integrated Primary Containment Leak Test (IPCLT). (1) Integrated leak rate tests shall be performed prior to initial unit operation at the test pressure of 43 psig, P_t (43), to obtain measured leak rate L_m (43). (2) Subsequent leak rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test at an initial pressure of approximately 43 psig.

THIRTY-DAY WRITTEN REPORTS (Continued)

completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety features instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in 6.9.1 8.c, above, designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. In-service Inspection Results, Specification 4.6.F.

b. Primary Containment Leak Rate Test Results, Specification 4.7.A.3.

c. Secondary Containment Leak Rate Test Results, Specification 4.7.C.

d. Materials Radiation Surveillance Specimen Examination and Results, Specification 4.6.B.3.

e. Fire detection instrumentation, Specification (3.12.E.2).

f. Fire suppression systems, Specifications (3.12.A.2, 3.12.B.2, 3.12.C.2 and 3.12.C.4).

6. Primary Containment Isolation

Description of Changes and Safety Evaluation Summary

In the Millstone Unit No. 1 design, the reactor water clean-up (RWCU) system is isolated on a low reactor water level setpoint of 482 1/16 inches above the vessel bottom. Following a reactor trip, the reactor level initially shrinks (due to collapsing voids), to approximately 480 inches above the vessel bottom, thus resulting in RWCU system isolation.

In W. G. Counsil's letter dated April 27, 1982, we proposed lowering the RWCU system isolation setpoint below 480 inches, but no lower than 434 1/16 inches above the vessel bottom. This will allow the RWCU system to be available when the reactor vessel begins to refill. The operator will thus have a letdown path available to prevent reactor vessel overfill during the low feedwater demand conditions shortly after shutdown. Therefore, the proposed changes in part would no longer require the RWCU system to isolate on low water level, but on low-low water level.

The proposed changes regarding the addition of Core Spray and LPCI isolation valves are necessitated by the addition of test connections to permit periodic leakage testing of valves in the Core Spray and LPCI systems in accordance with 10CFR50, Appendix J.

Primary containment isolation valves AC-40 and AC-41 have been added due to the installation of a drywell compressor system during this refueling outage. This system is being installed to reduce oxygen in-leakage to the containment. (See NNECO's Combustible Gas Control Evaluation, dated August 6, 1982).

Lastly, the Control Rod Hydraulic Return Check Valves were deleted since the portion of the control rod hydraulic return line penetrating containment was cut on both sides of the drywell wall and a cap installed on the outboard side during the 1980-1981 refueling outage. The penetration was sealed using a pipe cap which was full penetration butt welded in accordance with ASME Code, Class MC. A leak test was accomplished and the installation was found to be acceptable. The containment penetration now resembles other spare piping penetrations. This modification improves containment integrity since the seal mechanism is now permanent as opposed to the mechanical protection provided by the check valves.

TABLE 3.7.1
PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification (Valve Number)	Number of Power Operated Valves: Inboard	Number of Power Operated Valves: Outboard	Maximum Operating Time (sec)	Position	Action on Initiating Signal
1	Main Steam Line Isolation (MS-1A, 2A, 1B, 2B, 1C, 2C, 1D, 2D)	4	4	$3 \leq T \leq 5$	0	GC
1	Main Steam Line Drain (MS-5)	1		35	C	SC
1	Main Steam Line Drain (MS-6)		1	35	C	SC
1	Recirculation Loop Sample Line (SM-1, 2)	1	1	5	C	SC
1	Isolation Condenser Vent to Main Steam Line (IC-6, 7)		2	5	0	GC
2	Drywell Floor Drain (SS-3, 4)		2	20	0	GC
2	Drywell Equipment Drain (SS-13, 14)		2	20	0	GC
2	Drywell Vent (AC-7)	1	10		C	SC
2	Drywell Vent Relief (AC-9)		1	15	C	SC
2	Drywell and Suppression Chamber Vent from Reactor Building (AC-8)		1	10	C	SC
2	Drywell Vent to Standby Gas Treatment System (AC-10)		1	10	C	SC
2	Suppression Chamber Vent (AC-11)		1	10	C	SC
2	Suppression Chamber Vent Relief (AC-12)		1	15	C	SC
2	Suppression Chamber Supply (AC-6)		1	10	C	SC
2	Drywell Supply (AC-5)	1	10		C	SC
2	Drywell and Suppression Chamber Supply (AC-4)		1	10	C	SC
2	Drywell Compressor System Suction Line (AC-40, 41)		2	10	0	GC
3	Shutdown Cooling System (SD-1)	1		48	C	SC
3	Shutdown Cooling System (SD-2A, 2B, 4A, 4B)		4	48	C	SC
3	Shutdown Cooling System (SD-5)		1	48	C	SC
3	Reactor Head Cooling Line (HS-4)		1	45	C	SC

TABLE 3.7.1
PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	(Valve Number)	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Position	Action on Initiating Signal
4	Isolation Condenser Steam Supply (1C-1)		1		24	0	GC
4	Isolation Condenser Steam Supply (1C-2)			1	24	0	GC
4	Isolation Condenser Condensate Return (1C-3)		1		19	C	SC
4	Isolation Condenser Condensate Return (1C-4)			1	19	0	GC
5	Cleanup Demineralizer System (CU-2)		1		18	0	GC
5	Cleanup Demineralizer System (CU-2A)		1		18	C	SC
5	Cleanup Demineralizer System (CU-3, 28)			2	18	0	GC
5	Cleanup Demineralizer System (CU-5)			1	18	C	SC
	Feedwater Check Valves (FW-9A, 10A, 9B, 10B)		2	2	NA	0	Process
	Reactor Head Cooling Check Valves (HS-5)		1		NA	C	Process
	Low Pressure Coolant injection (LP-10A, 10B)			2	18	C	0
	Core Spray (CS-5A, 5B)			2	10	C	0
	Standby Liquid Control Check Valves (SL-7, 8)		1	1	NA	C	Process

TABLE 3.7.1

Key: 0 = Open
C = Closed
SC = Stays Closed
GC = Goes Closed
NA = Not Applicable

Note: Isolation groupings are as follows:

GROUP 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor low-low water level (This signal also trips the reactor recirculation pumps.)
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line tunnel high temperature.
5. Main steam line low pressure.

GROUP 2: The actions in Group 2 are initiated by any one of the following conditions:

1. Reactor low water level.
2. High drywell pressure.

GROUP 3: Reactor low water level alone initiates the following:

1. Shutdown cooling system isolation.
2. Reactor head cooling isolation.

GROUP 4: Isolation valves associated with the isolation condenser are closed upon indication of either high isolation condenser steam or condensate flow.

GROUP 5: Reactor low-low water level alone initiates the following:

1. Cleanup demineralizer system isolation.

7. Scram Discharge System

Description of Changes and Safety Evaluation Summary

The proposed Technical Specification changes are necessitated by the modifications being made to the scram discharge system during this outage. These modifications are described in a W. G. Counsil letter to D. G. Eisenhut, dated March 20, 1981. The major modifications consist of (1) installing a second instrumented volume tank (IVT) for the south scram discharge volume (SDV), thereby having two separate SDV's and associated piping, (2) replacing the 2" piping connecting the SDV with the IVT with 6" piping, (3) installing redundant vent and drain valves, and (4) increasing the SDV to allow for 3.34 gallons per control rod drive. The proposed Technical Specification changes, along with plant modifications, will improve plant safety.

A technical review of these changes has found them to be acceptable. Also, a safety evaluation has been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions.

It is noted that page 3/4 2-5 is revised consistent with proposed Technical Specification changes contained in a W. G. Counsil letter to D. M. Crutchfield, dated October 16, 1980. It is our understanding that these changes have been approved by the NRC Staff.

In order to implement the above plant modifications, it is necessary to make the scam discharge system inoperable. During the early part of the refueling outage prior to completing defueling, all control rods were fully inserted, valved out, and electrically disarmed. Accordingly, the scram discharge system was declared inoperable since the scram function was no longer required. Daily surveillances were performed to assure that all control rods were in fact valved out and electrically disarmed.

With the scram function no longer necessary, disabling of portions of the RPS to allow for replacement of certain HFA relays and for scram discharge system modifications was determined to be acceptable. Technical Specification 3.1 dictates the limiting conditions for operation regarding RPS instrumentation. The respective Bases indicate that the function of the RPS is to initiate a reactor scram. Since the scram function is not required when the plant is in the above configuration, the Bases for Technical Specification 3.1 are not directly applicable. Furthermore, the Bases imply that the requirement for a scram function in the refuel mode is to ensure that shifting to the refuel mode during reactor power operation does not diminish the need for the RPS.

The requirements of Technical Specification 3.1 were reviewed in light of the reduced applicability of its Bases. The locking of all control rods fully inserted performs an equivalent, if not superior, function as the RPS. Therefore, we believed that we met the intent of the Technical Specifications and that our plans presented no safety concerns.

The above was discussed with the Millstone Unit No. 1 Project Manager and Resident Inspector on September 9, 1982. It was mutually agreed that the above interpretation is acceptable to the NRC and that NNECO would propose a Technical Specification change in the near future to more clearly state the intended purpose of the Technical Specification provisions. The purpose of this discussion is to document the understanding reached with the NRC Staff, and formalize our intention to improve the wording of the Technical Specification.

TABLE 3.1.1

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 (8)	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	$\leq$ 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	$\leq$ 1085 psig	X	X	X	A
2	High Drywell Pressure	$\leq$ 2 psig	X (9)	X (7)	X (7)	A
2	Reactor Low Water Level	$\geq$ 1.0 inch***	X	X	X	A
2	Scram Discharge Vol. High Level	$\leq$ 24 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	$\geq$ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	$\leq$ .65 w + 42 (2)
1 (6)	Rod Block Monitor Downscale	$\geq$ 3/125 Full Scale
3	IRM Downscale (3)	$\geq$ 3/125 Full Scale
3	IRM Upscale	$\leq$ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	$\leq 10^5$ counts/sec.
1	Scram Discharge Volume - Water Level High	$\leq$ 12 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 $\geq$ 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.

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 24 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.

8. Reactor Protection System

Description of Changes and Safety Evaluation Summary

The proposed Technical Specification changes are necessitated by the installation of fully redundant Class 1E qualified protective circuits for under and over voltage and under frequency at the interface between the Reactor Protection System (RPS) and the RPS power supplies in accordance with our letters dated November 24, 1980 and August 18, 1982.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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.

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

9. Fire Detection Instrumentation

Description of Changes and Safety Evaluation Summary

The proposed Technical Specification changes revise the surveillance requirements for fire detection instruments which are inaccessible during plant operation. The proposed changes also revise some of the numbers of fire detection instruments that were previously incorrectly listed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION

E. Fire Detection Instrumentation

1. The minimum required fire detection instrumentation for each fire detection zone shown in Table 3.12.2 shall be OPERABLE whenever equipment in that fire detection zone is required to be OPERABLE.

2. With less than the minimum required number of the fire detection instrument(s) shown in Table 3.12.2 OPERABLE:

   a. Within 1 hour establish a watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

   b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

SURVEILLANCE REQUIREMENTS

E. Fire Detection Instrumentation

1. The fire detection instruments listed in Table 3.12.2 shall be demonstrated OPERABLE at least once per 6 months be performance of an INSTRUMENT FUNCTIONAL TEST with the exception that the functional test may consist of injecting a simulated electrical signal into the measurement channel rather than the instrument. Due to the inaccessability of the fire detectors located in the condenser bay, a sample consisting of 1/3 of the detectors per channel will be tested during every refuel outage. The sample test cycle will be completed every third refueling outage.

2. The non-supervised circuits between the above required detection instruments and the control room shall be demonstrated OPERABLE at least once per 31 days, per approved procedures.

TABLE 3.12.2

FIRE DETECTION INSTRUMENTS

NOTE: No two (2) adjacent detectors inoperable.

INSTRUMENT LOCATION	HEAT TOTAL AVAILABLE	HEAT MINIMUM INSTRUMENTS REQUIRED	SMOKE TOTAL AVAILABLE	SMOKE MINIMUM INSTRUMENTS REQUIRED
1. Cable Vault	-	-	15	12
2. H_2 Seal Oil Unit	1	1	-	-
3. Condenser Bay	36	25	-	-
4. Diesel Generator Room	6	4	-	-
5. D/G Fuel Oil Day Tank Room	1	1	-	-
6. Gas Turbine Enclosure	3	2	-	-
7. RX Bldg (1-FDS-1)				
a. R-2a RX Bldg 14'6" N.W. Corner Including Crd Bank			7	5
b. R-2B RX Bldg 14'6" N.E. Corner			4	3
c. R-2C RX Bldg 14'6" S.E. Corner			3	2
d. R-2D RX Bldg 14'6" S.W. Corner including C.R.D. Bank			3	2
e. R-3 Tip Room	-	-	1	1
f. R-5 Shutdown Cooling Pump Room	-	-	1	1
8. RX Bldg (42' elev) 1-FDS-2				
a. R-17 Clean-up Pump Room	-	-	3	2
b. R-18 Shut Down Heat Exchanger Room			1	1
c. R-19 RX Bldg 42' N.W. Corner to SE Corner			9	7
9. RX Bldg (Elev. 65' and 82') 1-FDS-3)				
a. R-12 RX Bldg Elev. 65' except gated area along North Wall			9	7
b. R-13 RX Bldg (Elev.82") West Side			3	2
c. R-14 RX Bldg Elev. 82' East Side			2	1

TABLE 3.12.2

FIRE DETECTION INSTRUMENTS

NOTE: No two (2) adjacent detectors inoperable.

INSTRUMENT LOCATION	HEAT TOTAL AVAILABLE	HEAT MINIMUM INSTRUMENTS REQUIRED	SMOKE TOTAL AVAILABLE	SMOKE MINIMUM INSTRUMENTS REQUIRED
10. Turbine Bldg (14'-6") 1-FDS-4				
a. T-5A Condensate Pumps and Condensate Booster Pumps			6	4
b. Condensate Demin Panel, RX Feed Pumps	-	-	6	4
c. T-5C TBCCW to Stator Cooling Unit			15	12
d. T-6 Decon Room			1	1
e. T-12 Chem Lab			3	2
f. T-17 Mezzanine Above MCC D-2 (1A-2)			3	2
11. Rad. Waste 1-FDS-5				
a. RW-A Radwaste Storage Bldg.			3	2
b. RW-B Liquid Radwaste Bldg.			6	4
12. Turb. Bldg. (Elev. 36'-6". 7' 34'-6") 1-FDS-6)				
a. T-10A Make Up Demin Storage Tk, Make Up Demin Ultra Filter, RP MG-Sets, Vital AC MG Sets, 4KV Bus 14E & 480V Bus 12E			14	12
b. T-19B 4KV Bus 14F and 480V Bus 12F	-	-	1	1
c. T-19C 4KV Bus 14A, 4KV Bus 14C and 480V MCC's E5 & F5	-	-	3	2
d. T-19D 4KV Bus 14B & 4KV Bus 14D	-	-	3	2
e. T-19E 4KV Bus 14G, DC Buses 101A & 101B 480 V Buses 12C & 12D, Iso Phase Bus Duct	-	-	8	6
f. T-15A H &V Equipment Rm. Elev. 54'6'			2	1
13. Control Room T-21 (Elev. 36'-6")			9	7
14. Screen House SH	7	5		
14. Battery Rooms, Station No. 2				
a. Battery Rm. 1 (Sta. #2)	1	1		
b. Battery Rm. 1A (Sta. #2)	1	1		