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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE BUNDLE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE is equal to the sum of the axially averaged exposure of all the fuel rods in the specified bundle divided by the number of fuel rods in the bundle.

AVERAGE PLANAR EXPOSURE

1.3 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.4 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.7 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

CORE ALTERATION

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.8A The CORE OPERATING LIMITS REPORT is the WNP-2 specific document that provides CORE OPERATING LIMITS for the current operating reload cycle. These cycle-specific CORE OPERATING LIMITS shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these Operating Limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be that power in the assembly which is calculated by application of the appropriate critical power correlation to cause some point in the assembly to experience boiling transition divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel ~~as a function of AVERAGE PLANAR EXPOSURE for GE initial core fuel and average bundle exposure for ANF, SVEA-96 and GE11 LFA fuel~~ shall not exceed the limits ~~shown~~ specified in ~~Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-6, 3.2.1-7 and 3.2.1-8 when in two loop operation, and Figures 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8 when in single loop operation.~~ the Core Operating Limits Report.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits ~~of~~ specified in the Core Operating Limits Report, ~~Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-6, 3.2.1-7 or 3.2.1-8 in two loop operation or Figure 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 or 3.2.1-8 in single loop operation,~~ initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

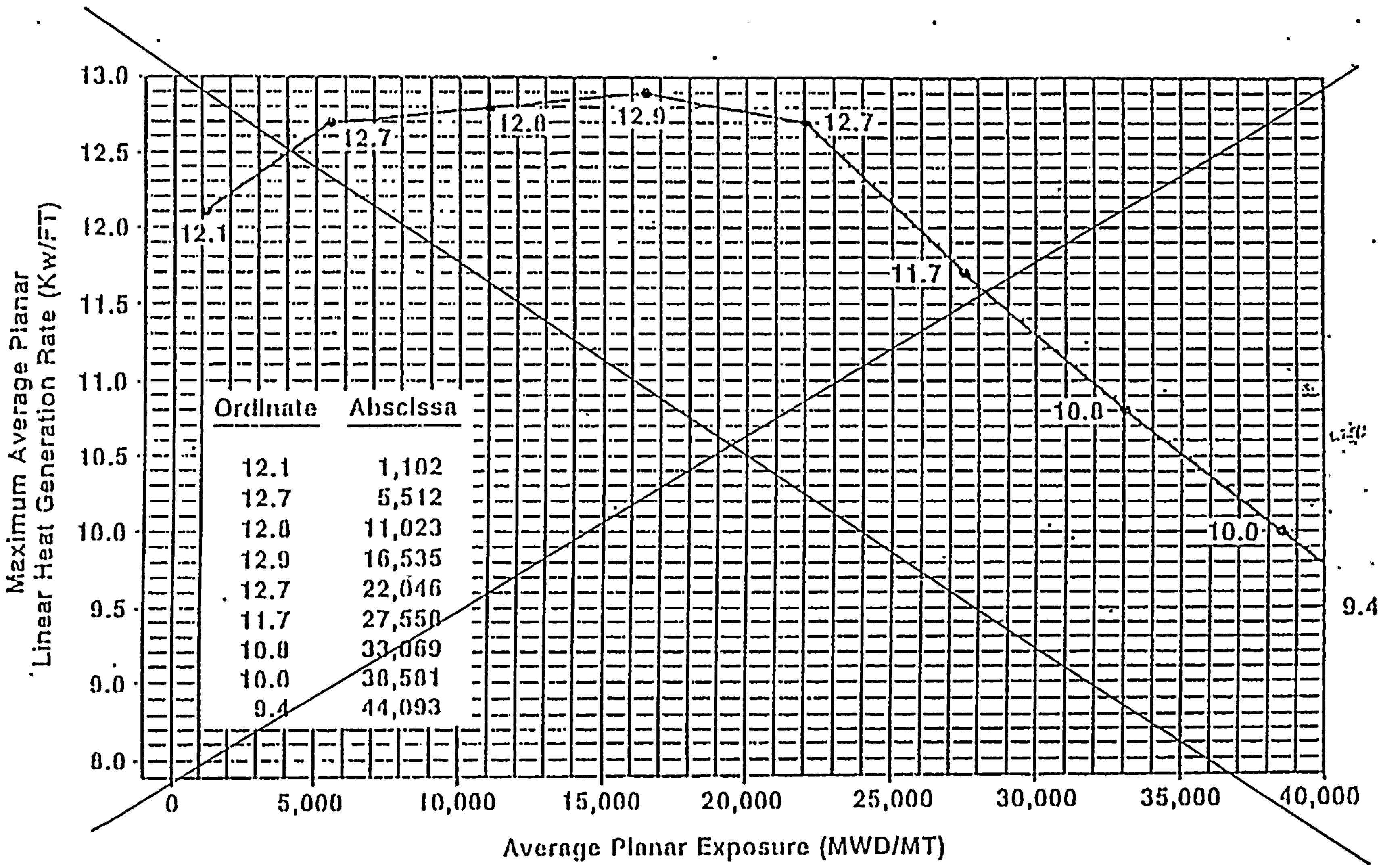
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits ~~determined from~~ specified in ~~Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8.~~ the Core Operating Limits Report.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



Ordinate	Abscissa
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Average Planar Exposure
Initial Core Fuel Type 8CR183
Figure 3.2.1-1

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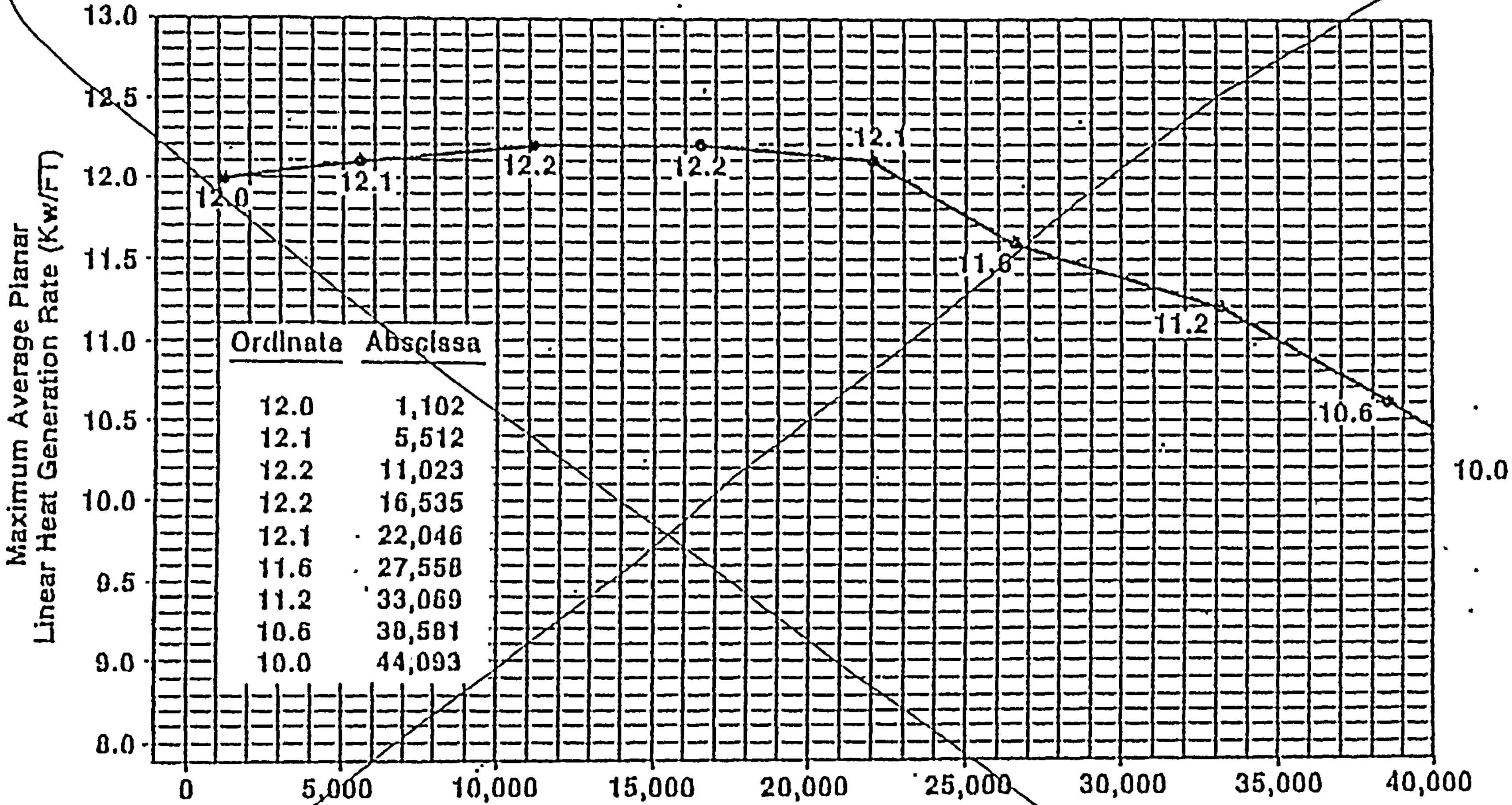

Ordinate	Abscissa
12.0	1,102
12.1	5,512
12.2	11,023
12.2	16,535
12.1	22,046
11.6	27,558
11.2	33,069
10.6	38,581
10.0	44,093

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Initial Core Fuel Type 8CR233

Figure 3.2.1-2

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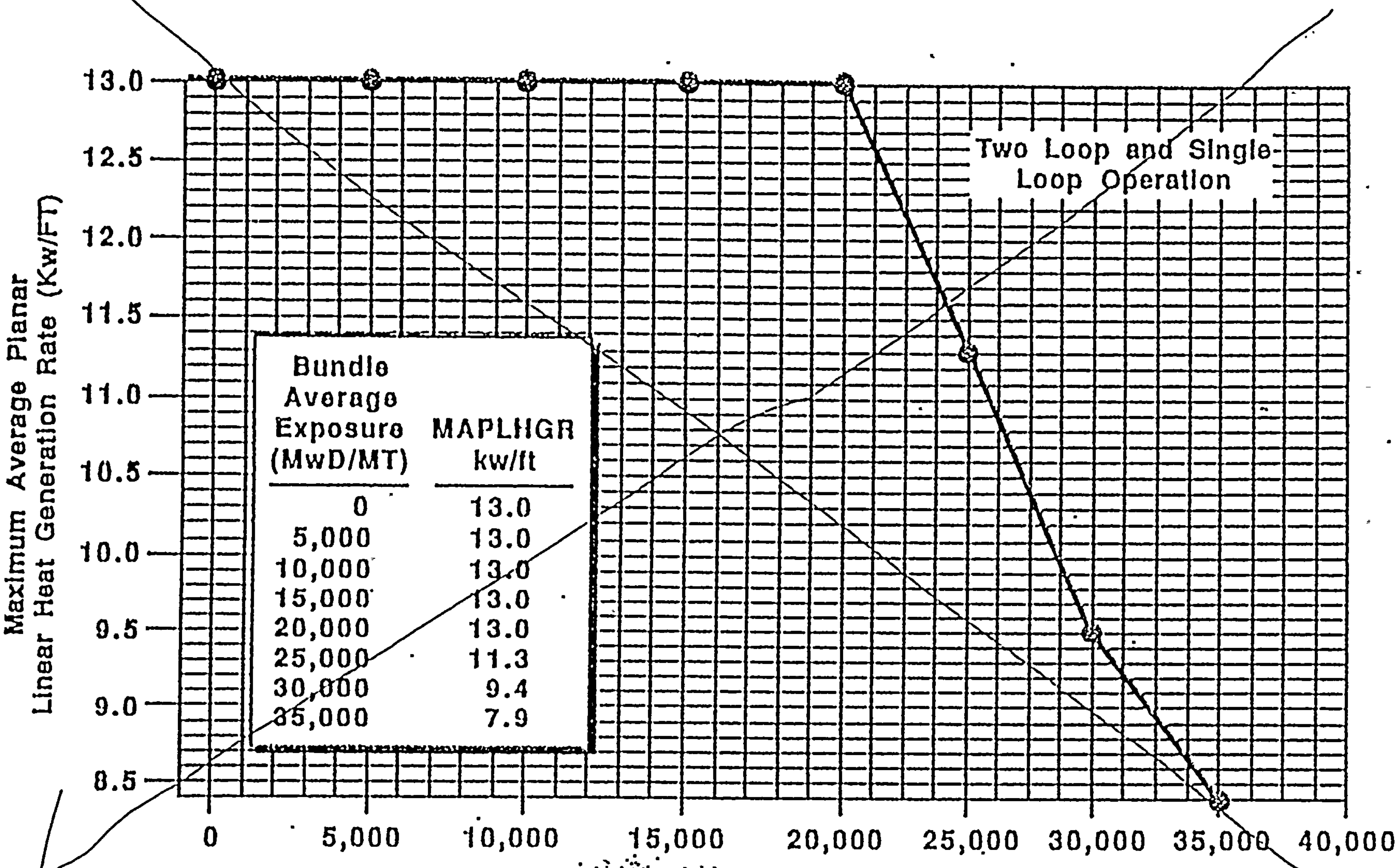
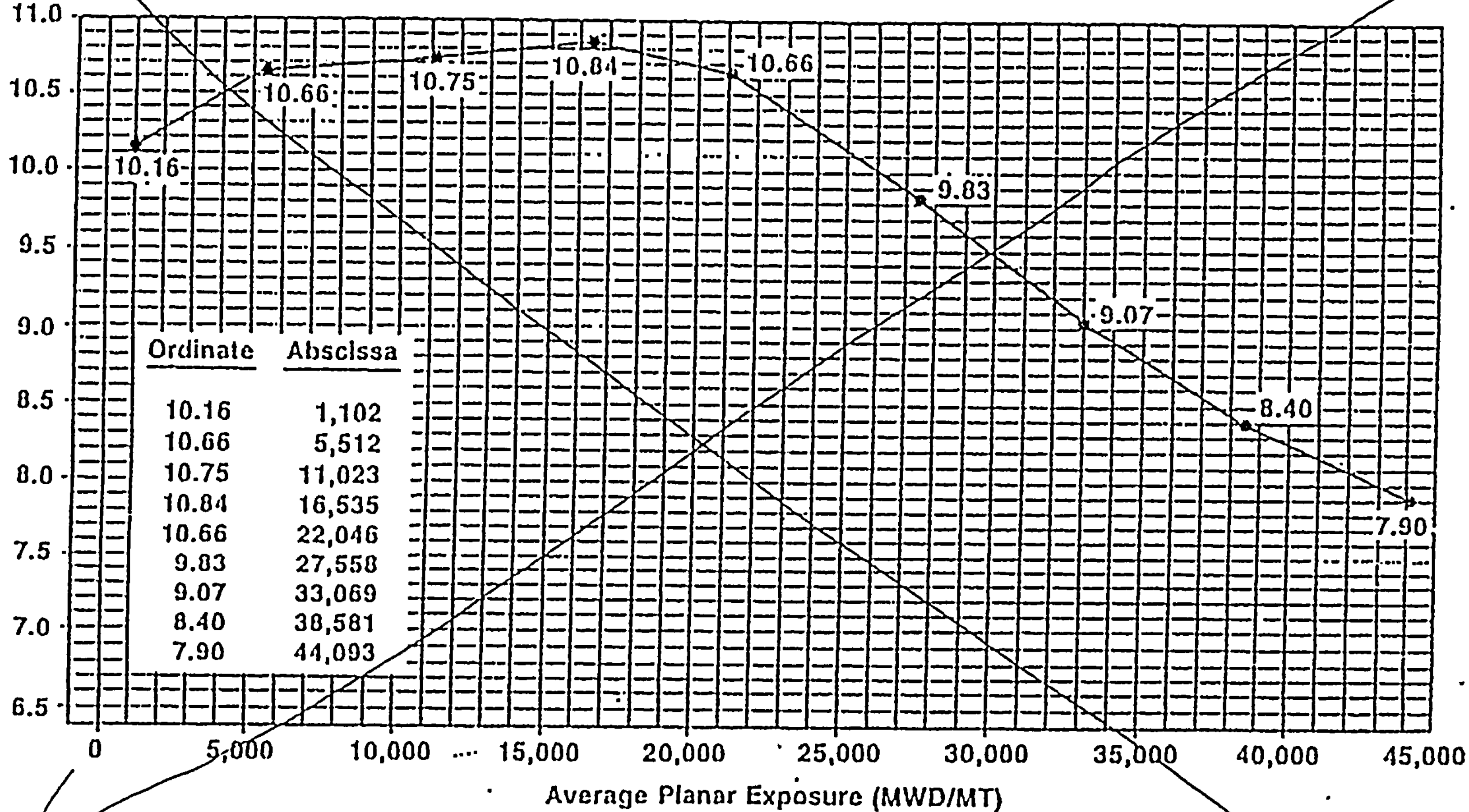


Bundle Average Exposure (MwD/MT)	MAPLHGR kw/ft
0	13.0
5,000	13.0
10,000	13.0
15,000	13.0
20,000	13.0
25,000	11.3
30,000	9.4
35,000	7.9

Bundle Average Exposure (MWD/MT)
ANF 8x8 Reload Fuel
Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure

Figure 3.2.1-3

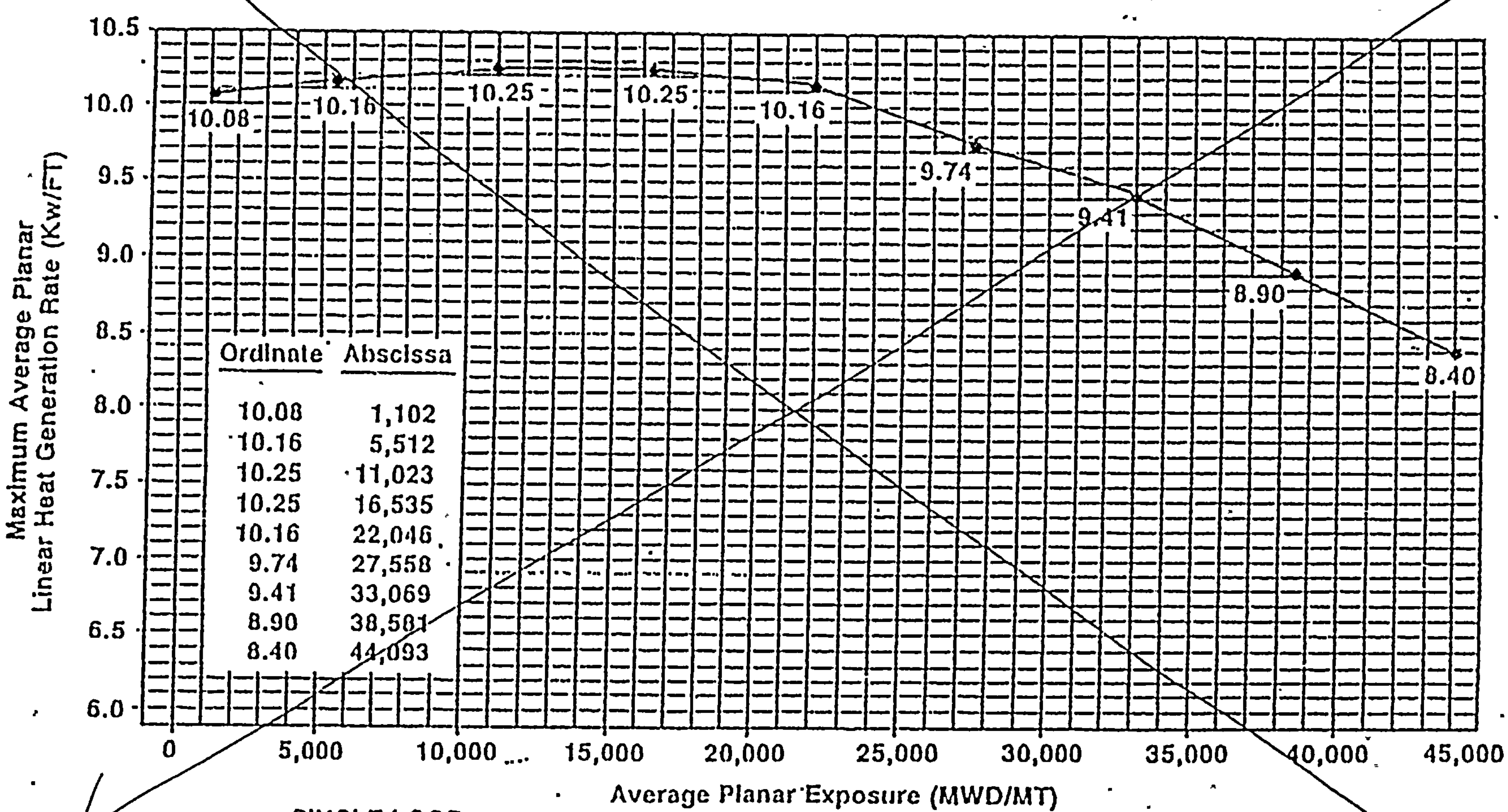


Ordinate	Abscissa
10.16	1,102
10.66	5,512
10.75	11,023
10.84	16,535
10.66	22,046
9.83	27,558
9.07	33,069
8.40	38,581
7.90	44,093

SINGLE LOOP

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Average Planar Exposure
Initial Core Fuel Type 8CR183
Figure 3.2.1-4

860590.5A

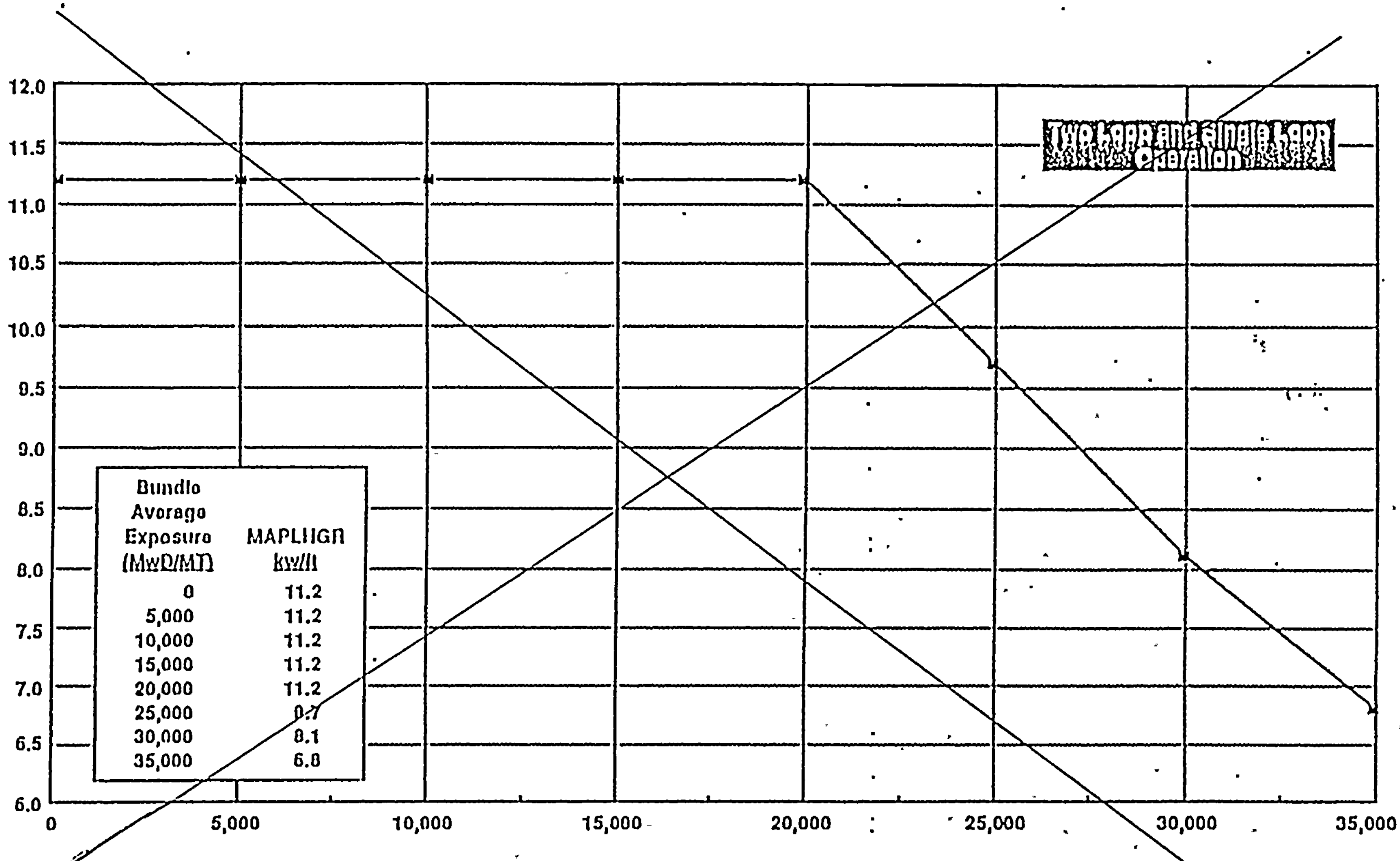


Ordinate	Abscissa
10.08	1,102
10.16	5,512
10.25	11,023
10.25	16,535
10.16	22,046
9.74	27,558
9.41	33,069
8.90	38,581
8.40	44,093

SINGLE LOOP

Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Average Planar Exposure
Initial Core Fuel Type 8CR233
Figure 3.2.1-5

660599 5A



Bundle Average Exposure (MwD/MT)	MAPLHGR kw/ft
0	11.2
5,000	11.2
10,000	11.2
15,000	11.2
20,000	11.2
25,000	9.7
30,000	8.1
35,000	6.8

Bundle Average Exposure (MWD/MT)
ANF 9 X 9 - IX AND 9 X 9 - 9X Reload Fuel
Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure

Figure 3.2.1-6

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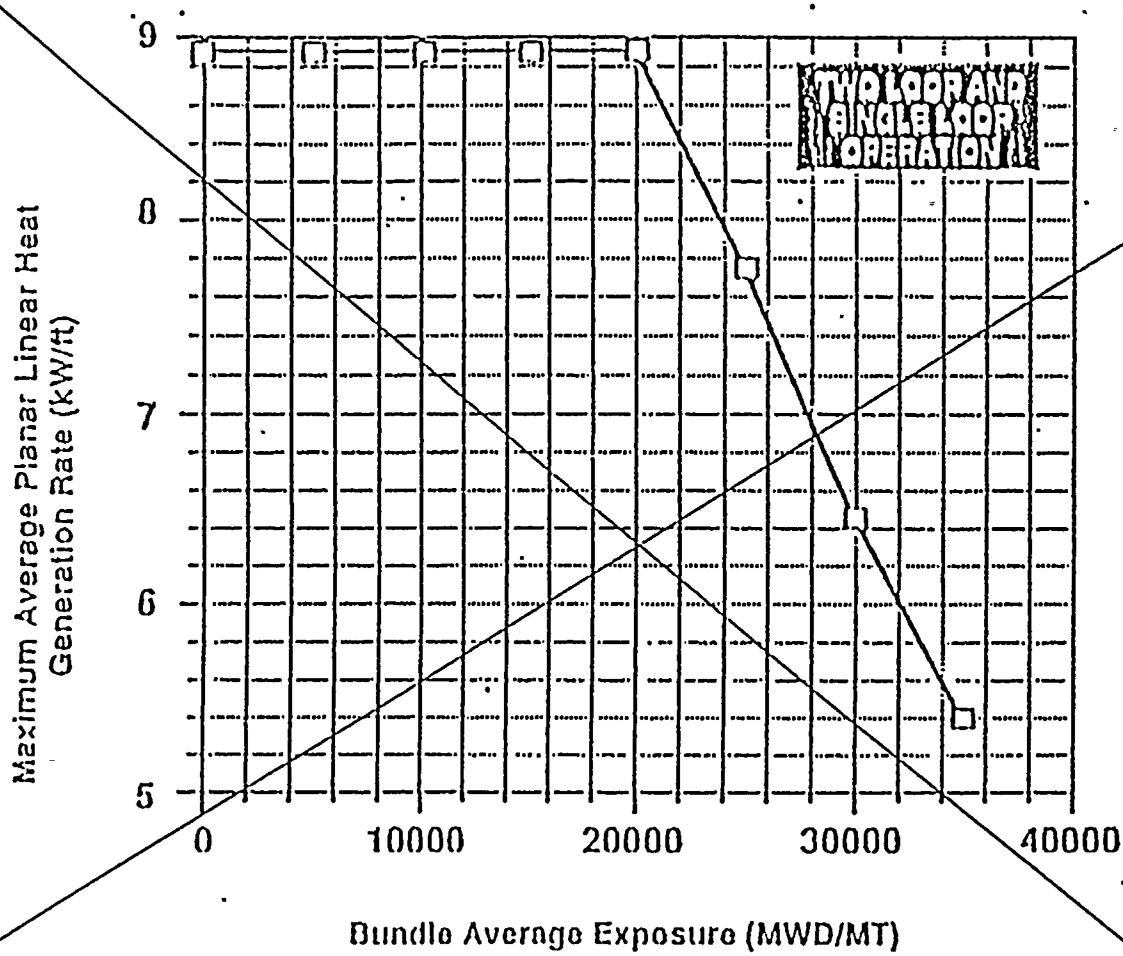

Bundle Average Exposure (MWD/MTU)	MAPLHGR (kW/ft)
0	8.90
5,000	8.90
10,000	8.90
15,000	8.90
20,000	8.90
25,000	7.74
30,000	6.44
35,000	5.41

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure SVEA-96 Lead Fuel Assemblies

Figure 3.2.1-7

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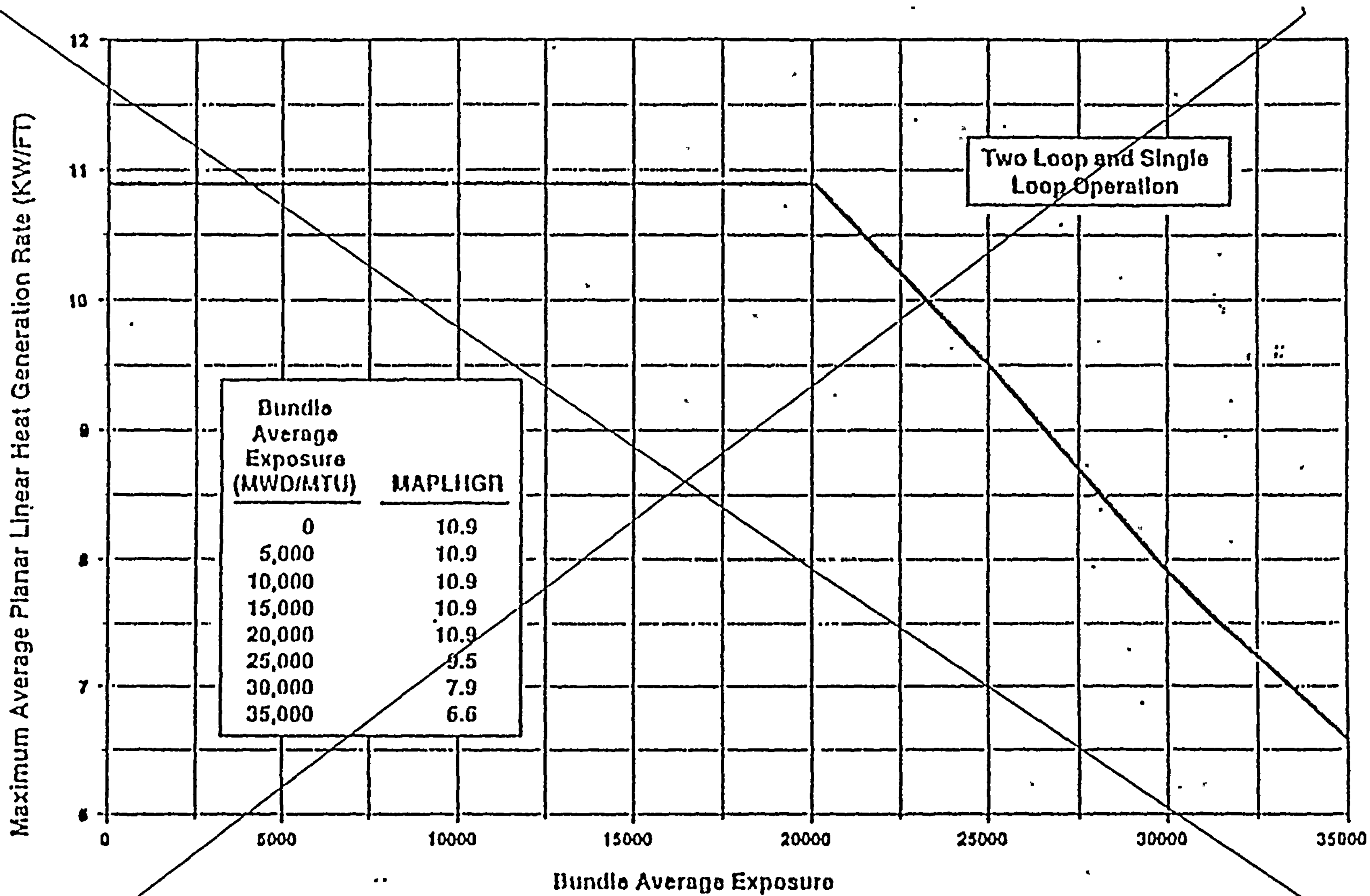

Bundle Average Exposure (MWD/MTU)	MAPLHGR
0	10.9
5,000	10.9
10,000	10.9
15,000	10.9
20,000	10.9
25,000	9.5
30,000	7.9
35,000	6.6

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure - GE 11 Lead Fuel Assemblies

Figure 3.2.1-8

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W + 51\%)T$	$S \leq (0.66W + 54\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 108.5 million lbs/h.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value(*) within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be:

~~a.~~ Greater than or equal to the applicable MCPR limit ~~determined from Table 3.2.3-1 during steady state operation at or above rated core flow in two loop operation, or when in single loop operation, or~~ specified in the Core Operating Limits Report.

~~b. Greater than or equal to the greater of the two values determined from Table 3.2.3-1 and Figure 3.2.3-1 during steady state operation at less than rated core flow when in two recirculation loop operation.~~

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER.

ACTION: With MCPR less than the applicable MCPR limit ~~determined from Table 3.2.3-1 and Figure 3.2.3-1~~ specified in the Core Operating Limits Report, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit ~~determined from Table 3.2.3-1 and Figure 3.2.3-1~~ specified in the Core Operating Limits Report.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15 percent of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

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Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow 8x8 ANF Fuel***	SVEA-96 LFA FUEL
1.	0 MWD/MTU - 3750 MWD/MTU	*	1.24	1.37
2.	3750 MWD/MTU - EOC MWD/MTU****	Normal scram times**	1.31	1.48
3.	3750 MWD/MTU - EOC MWD/MTU****	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.36	1.55
4.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times**	1.36	1.55
5.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.40	1.61
6.	0 MWD/MTU - EOC MWD/MTU	Single loop operation RPT operable Normal scram times**	1.35	1.54

*In this portion of the fuel cycle, operation with the given MCPR operating limits is allowed for both normal and Tech. Spec. scram times and for both RPT operable and inoperable.

**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times default to the values associated with Tech. Spec. scram times (3.1.3.4-p 3/4 1-8), and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest measured average control rod insertion times to specified notches for all operable control rods for each group of 4 control rods arranged in a a two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624

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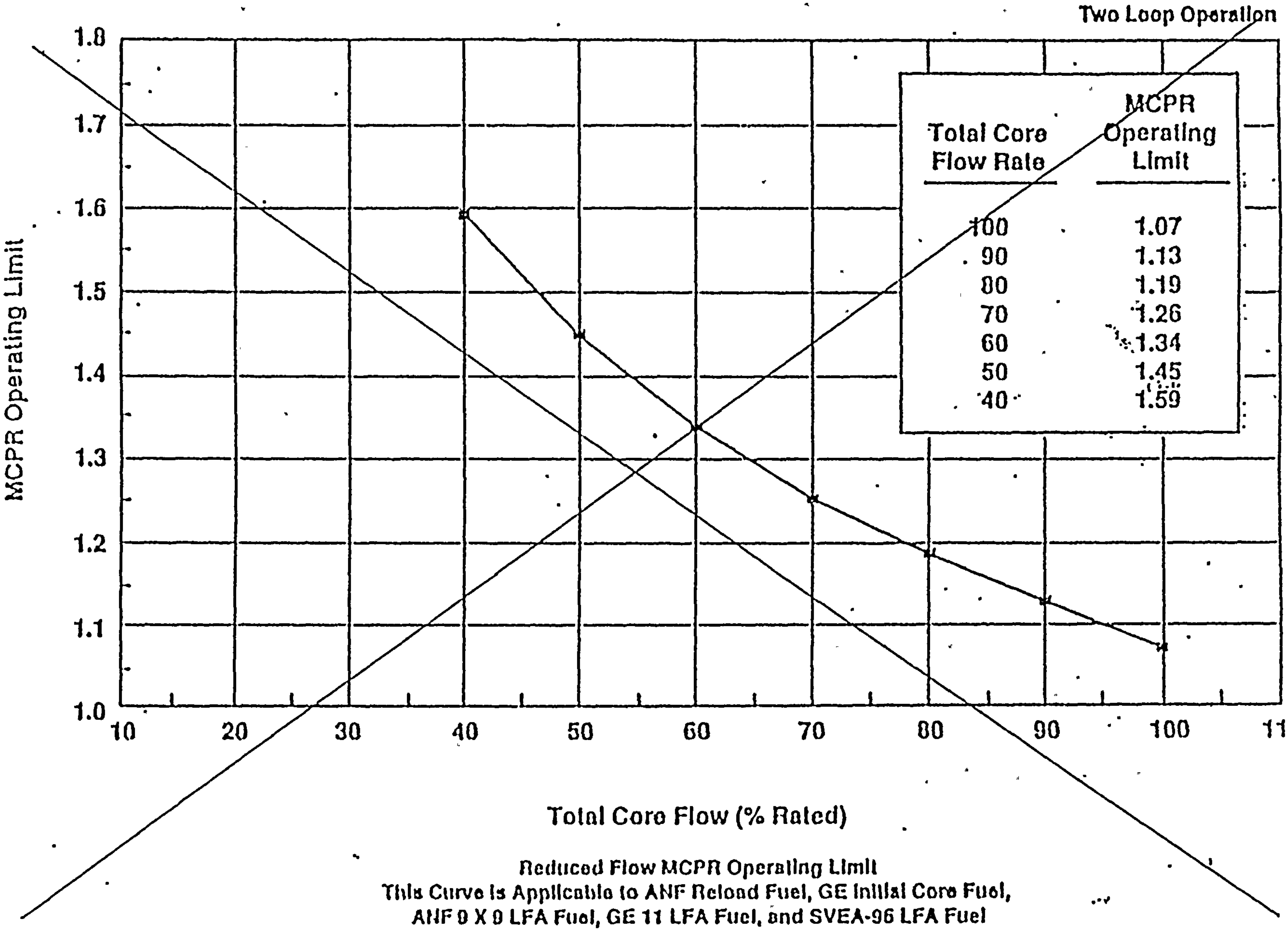
Table 3.2.3-1 (Continued)
MCPR OPERATING LIMITS

***The GE11 LFA fuel, the ANF LFA fuel and the GE initial core fuel are also monitored to the ANF 8x8 fuel MCPR Operating Limits (Reference: Power Distribution Limits, Bases, 3/4.2.3, Minimum Critical Power Ratio, p. B 3/4 2-3).

****For Final Feedwater Temperature Reduction rated conditions beyond all rods out point, add .02 to the MCPR for all fuel in the WNP-2 core except for the SVEA-96 LFA fuel. For the SVEA-96 LFA fuel, add .03 to the MCPR for Final Feedwater Temperature Reduction rated conditions beyond the all rods out point.



Total Core Flow Rate	MCPR Operating Limit
100	1.07
90	1.13
80	1.19
70	1.26
60	1.34
50	1.45
40	1.59

Reduced Flow MCPR Operating Limit
This Curve Is Applicable to ANF Reload Fuel, GE Initial Core Fuel,
ANF 9 X 9 LFA Fuel, GE 11 LFA Fuel, and SVEA-96 LFA Fuel
This curve is also applicable to FFTR operation
Figure 3.2.3-1

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POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION.

specified

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) ~~for GE initial core fuel shall not exceed 13.4 kW/ft. The LHGR for reload fuel~~ shall not exceed the values ~~shown~~ in ~~Figures 3.2.4-1, 3.2.4-2, 3.2.4-3, 3.2.4-4 and 3.2.4-5.~~ the Core Operating Limits Report

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

specified in the Core Operating Limits Report

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

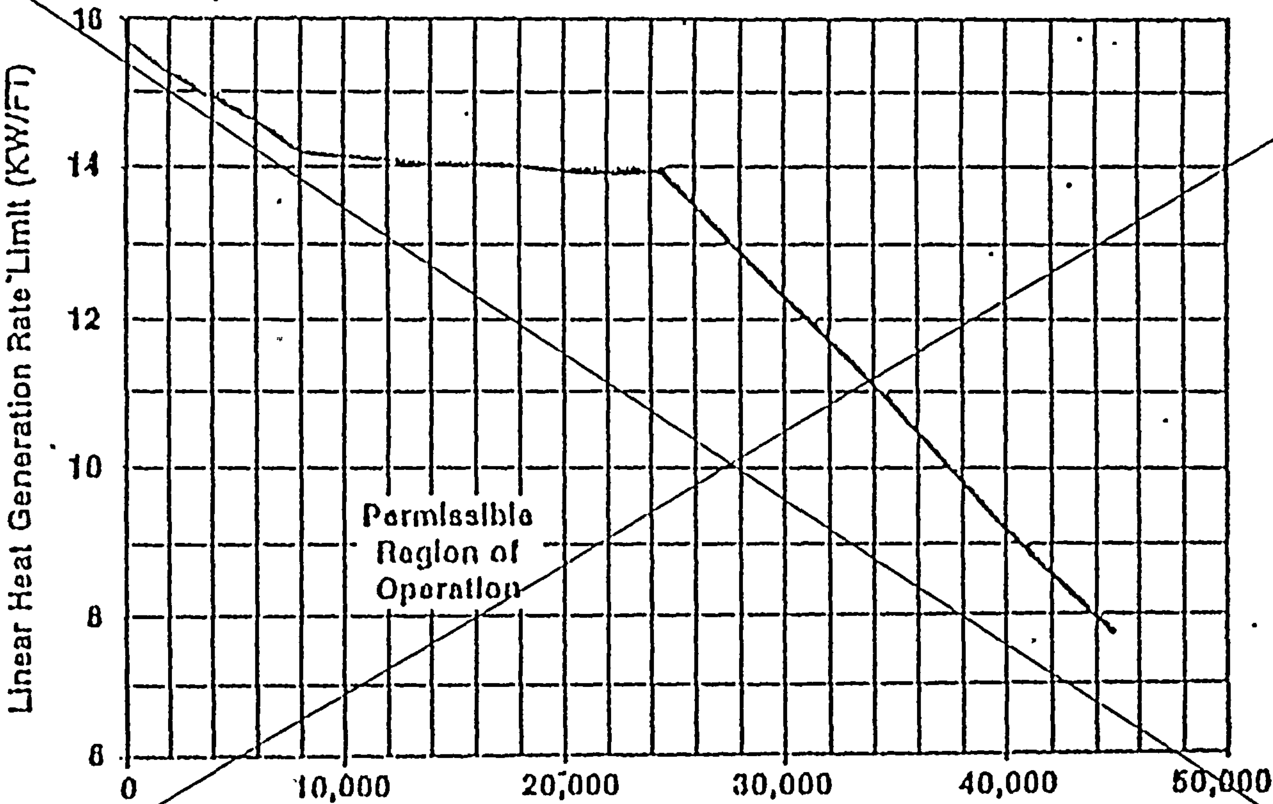
SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit: specified in the Core Operating Limits Report

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.



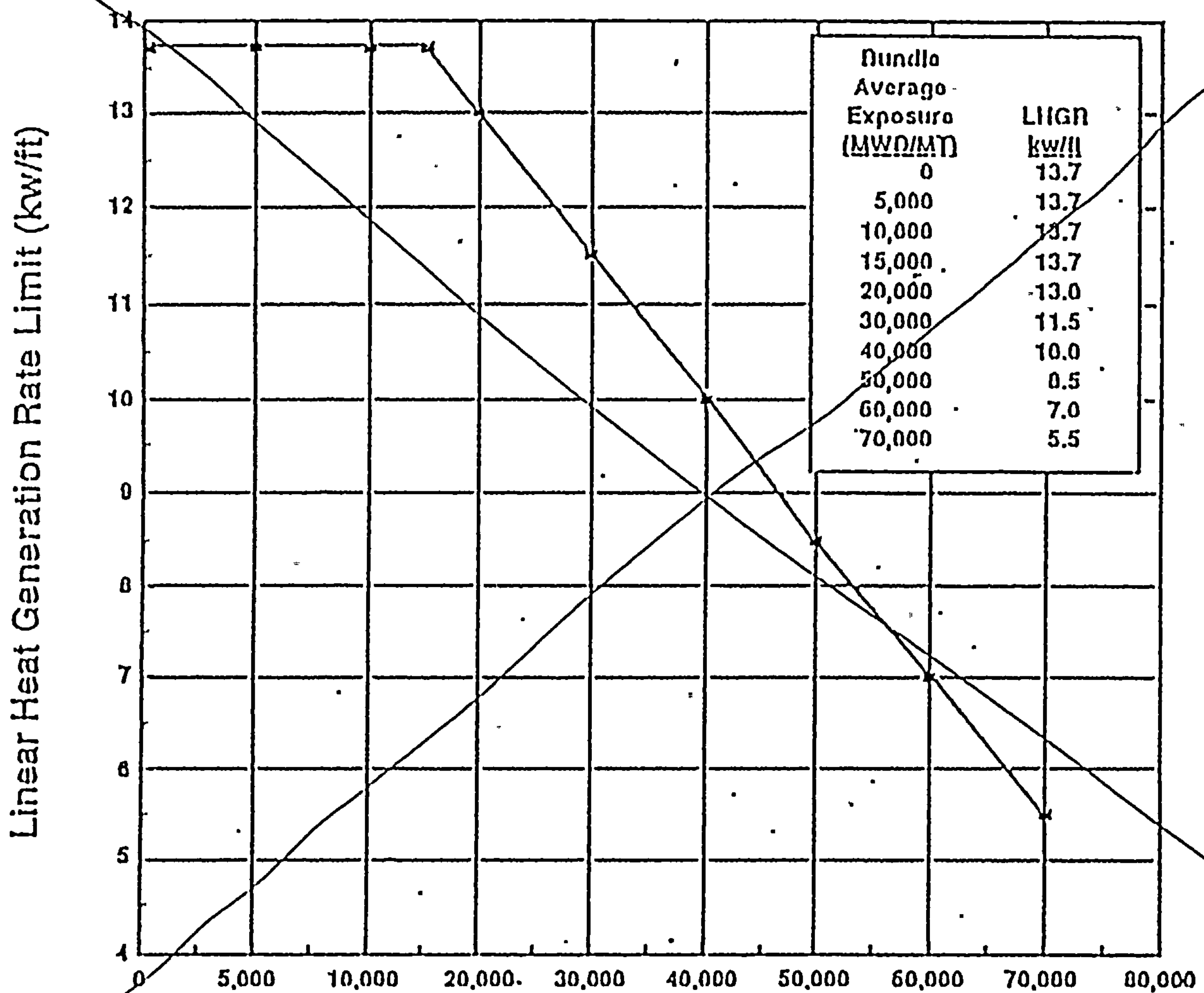
EXP	LHGR
0	15.62
510	15.621
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.08
15,990	14.06
18,708	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,060	10.47
38,900	9.55
41,830	8.65
44,760	7.77

Average Planar Exposure (MWD/MT)

ANF 8x8 Reload Fuel

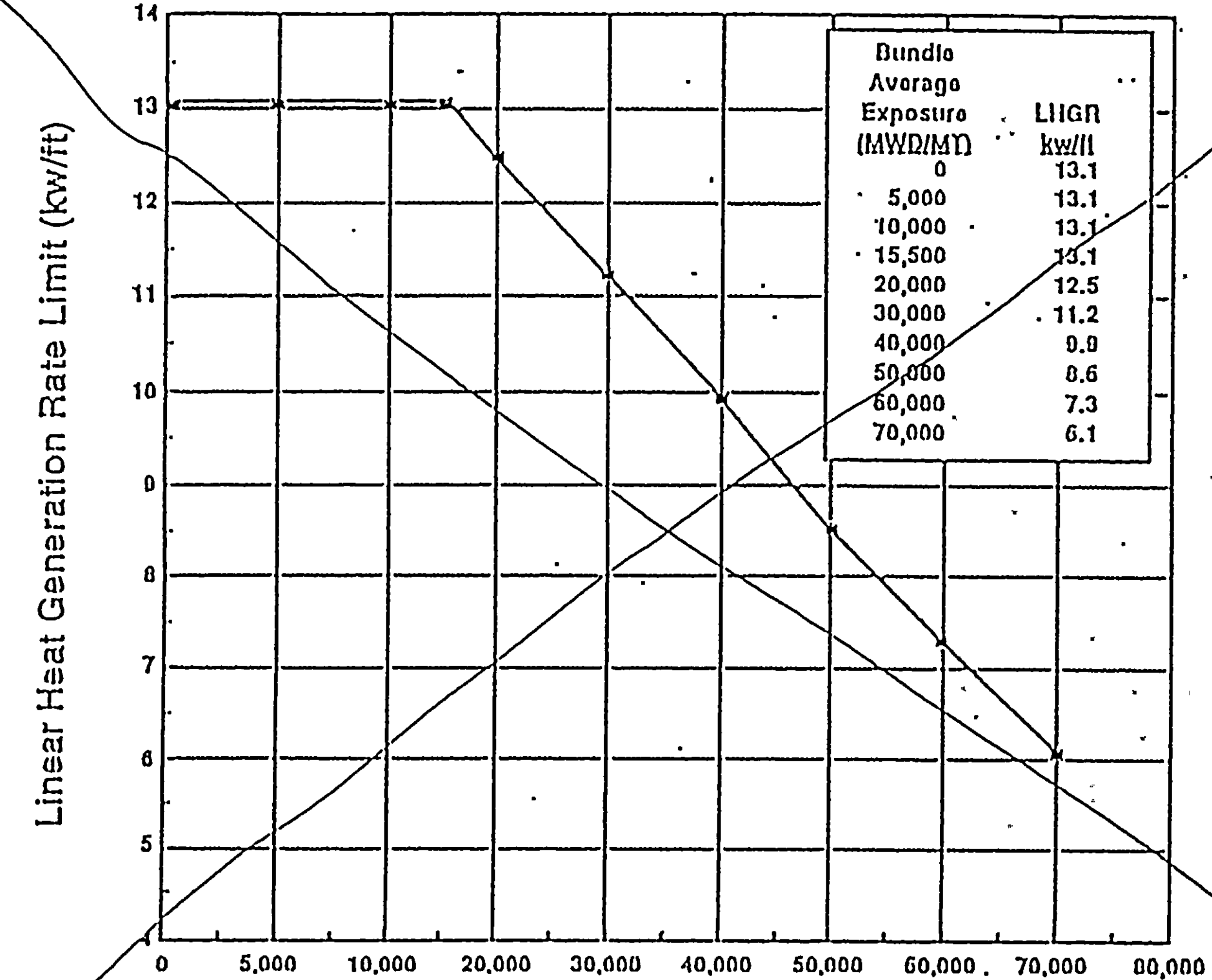
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-1



Average Planar Exposure (MWD/MT)
ANF 9 X 9 - IX Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-2



Average Planar Exposure (MWD/MT)
ANF 9 X 9 - 9X Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-3

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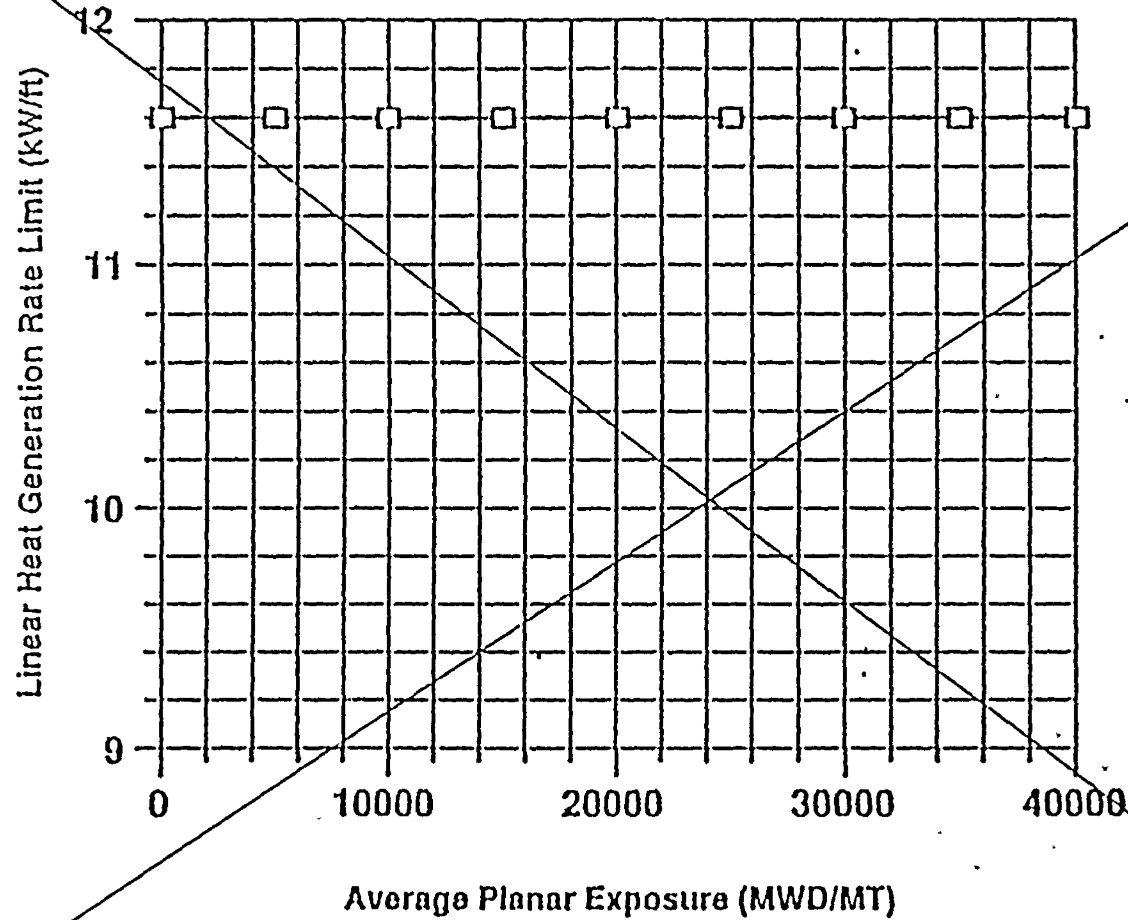

Exposure (MWD/MTU)	LHGR (kW/ft)
0 to 40,000	11.6

Linear Heat Generation Rate (LHGR) Limit Versus Average Planar Exposure
SVEA-96 Lead Fuel Assemblies

Figure 3.2.4-4

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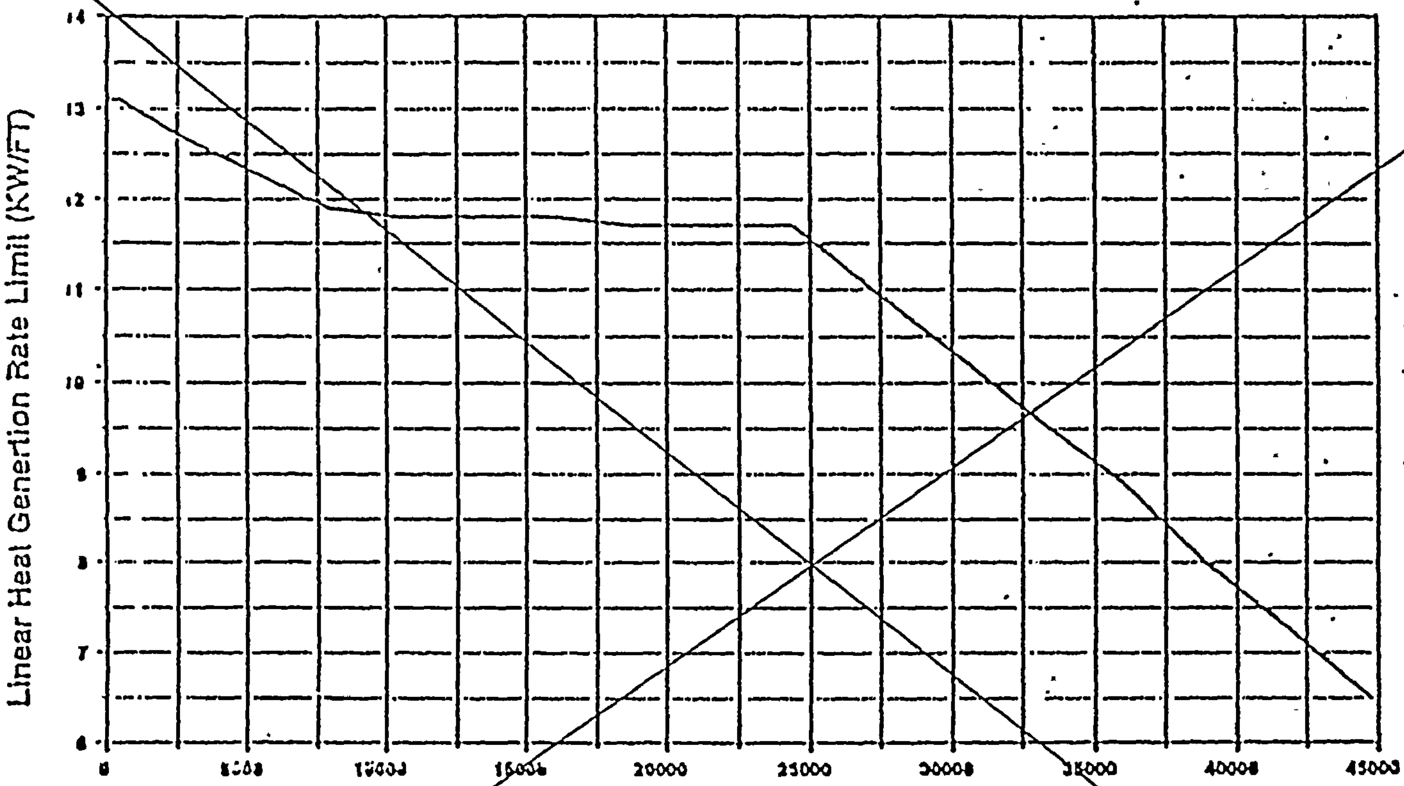

EXP	LHGR
0	13.1
510	13.1
2,500	12.7
5,230	12.3
7,040	11.9
10,470	11.8
13,220	11.8
15,990	11.8
18,708	11.7
21,590	11.7
24,420	11.7
27,200	11.0
30,150	10.3
33,050	9.8
35,060	9.0
38,000	8.0
41,830	7.3
44,760	6.5

Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
GE 11 Lead Fuel Assemblies

Figure 3.2.4-5

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POWER DISTRIBUTION LIMITS

3/4.2.6 POWER/FLOW INSTABILITY

LIMITING CONDITION FOR OPERATION

3.2.6 Operation with THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1 is prohibited.

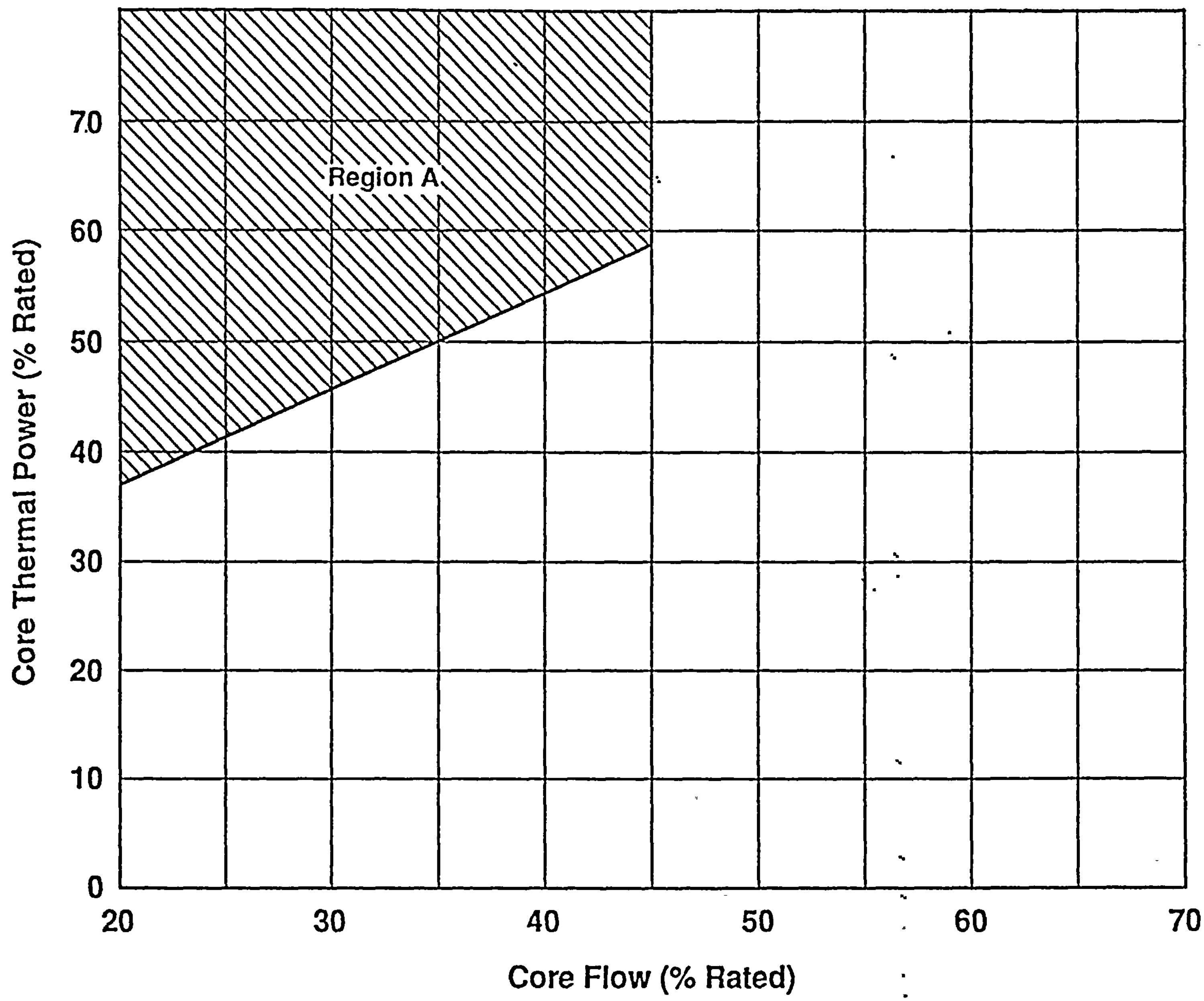
APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than 39% of RATED THERMAL POWER and core flow is less than or equal to 45% of rated core flow.

ACTION:

With THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1, then as soon as practical, but in all cases within 15 minutes, initiate a MANUAL SCRAM.

SURVEILLANCE REQUIREMENTS

4.2.6 The THERMAL POWER/core flow conditions shall be verified to lay outside Region A of Figure 3.2.6-1 once per 24 hours when operating in the region of APPLICABILITY.



Operating Region Limits of Specification 3.2.6
Figure 3.2.6-1

3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.2.7 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with two recirculation loops in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.7-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than .75 or with two (2) consecutive decay ratios on any single neutron signal greater than .75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of exiting the region of APPLICABILITY when the stability monitoring system is inoperable is specifically prohibited. Exit the region of APPLICABILITY within one (1) hour.

SURVEILLANCE REQUIREMENTS

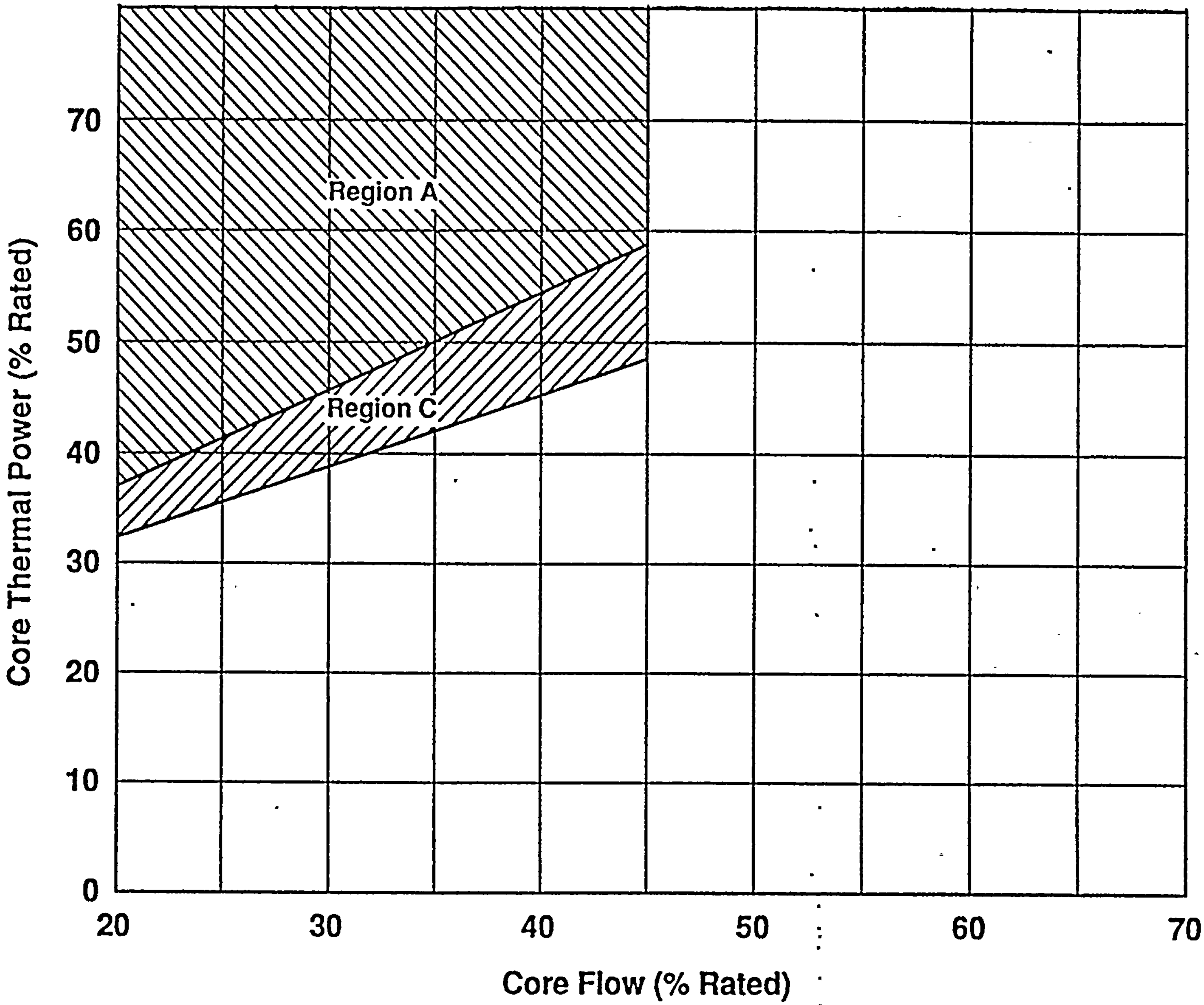
4.2.7.1 The provisions of Specification 4.0.4 are not applicable.

4.2.7.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.7.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

---

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.



Operating Region Limits of Specification 3.2.7
Figure 3.2.7-1

3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.2.8 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with one recirculation loop in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.8-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than .75 or with two (2) consecutive decay ratios on any single neutron signal greater than .75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by decreasing THERMAL POWER with control rod insertion. Exit the region of APPLICABILITY within one (1) hour.

SURVEILLANCE REQUIREMENTS

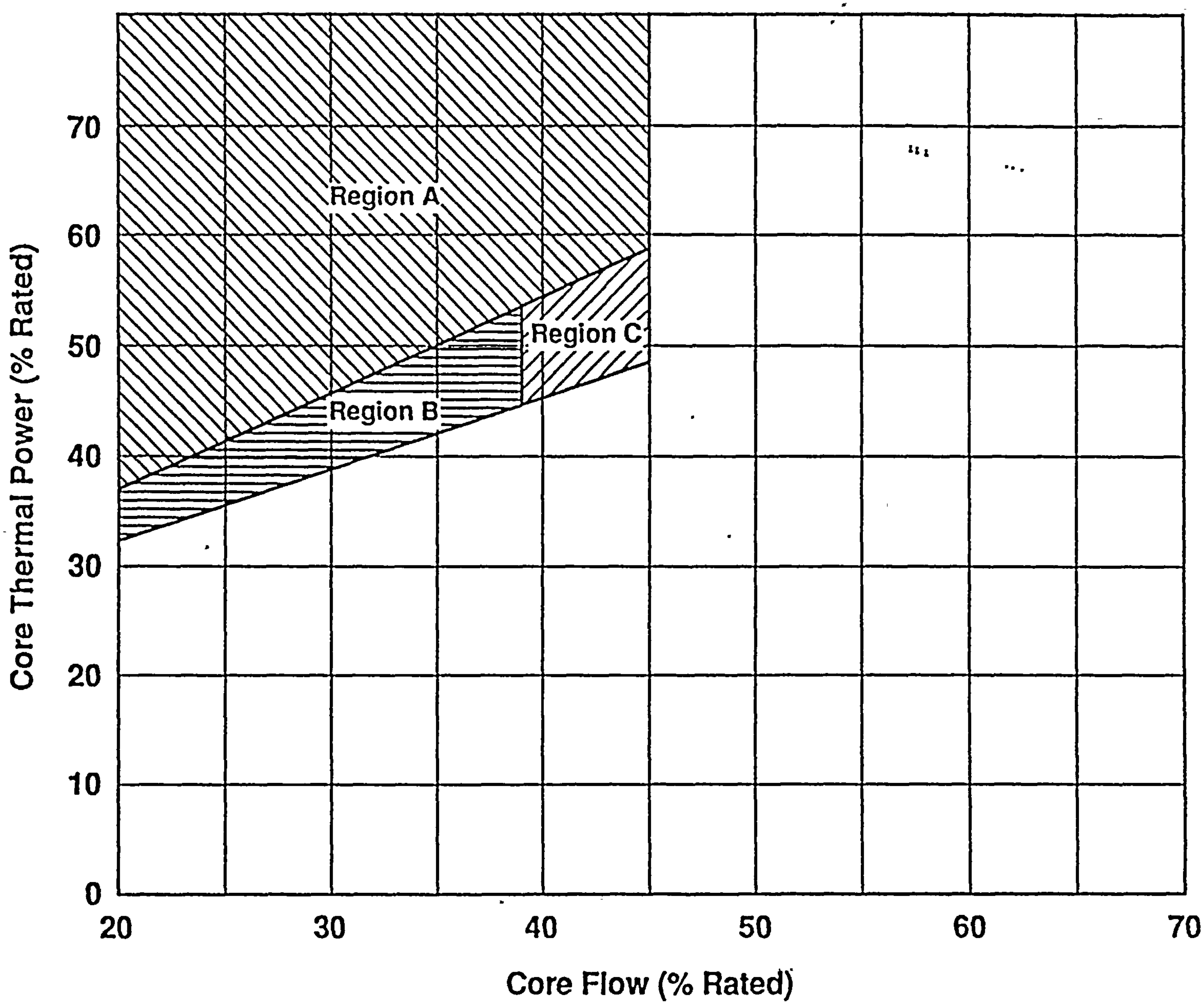
4.2.8.1 The provisions of Specification 4.0.4 are not applicable.

4.2.8.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.8.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

---

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.



Operating Region Limits of Specification 3.2.8
Figure 3.2.8-1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Verify that the requirements of LCO 3.2.6 and LCO 3.2.8 are met, or comply with the associated ACTION statements

2. Verify that THERMAL POWER/core flow conditions lay outside Region B of Figure 3.4.1.1-1.

   With THERMAL POWER/core flow conditions which lay in Region B of Figure 3.4.1.1-1, as soon as practical, but in all cases within 15 minutes, initiate action to exit Region B by either decreasing THERMAL POWER with control rod insertion or increasing core flow with flow control valve manipulation. Within 1 hour exit Region B. The starting or shifting of a recirculation pump for the purpose of exiting Region B is specifically prohibited.

3. Within 4 hours:

   a) Place the recirculation flow control system in the Local Manual (Position Control) mode, and

   b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.07 per Specification 2.1.2, and,

   c) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for General Electric fuel limit to ~~a value of 0.84 times~~ the SINGLE ~~two~~ recirculation loop operation limit ~~per Specification 3.2.1~~ SPECIFIED IN THE CORE OPERATING LIMITS REPORT, and,

   d) Reduce the volumetric flow rate of the operating recirculation loop to $\leq$ 41,725** gpm.

*See Special Test Exception 3.10.4.

**This value represents the actual volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. This value was determined during the Startup Test Program.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) 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 dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is ~~shown in Figures 3.2.1-1 and 3.2.1-2 for two recirculation loop operation and Figures 3.2.1-4 and 3.2.1-5 for single loop operation. Figures 3.2.1-3, and 3.2.1-6 apply to both single and two loop operation.~~ SPECIFIED IN THE CORE OPERATING LIMITS REPORT.

The calculational procedure used to establish the APLHGR SPECIFIED IN THE CORE OPERATING LIMITS REPORT ~~shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6~~ is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. These models are ~~described~~ referenced in ~~NEDO-20566P or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 2~~ SECTION 6.9.3 OF THE TECH. SPECS

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POWER DISTRIBUTION LIMITS

BASES

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is ~~obtained and presented in Table 3.2.3-1.~~ SPECIFIED IN THE CORE OPERATING LIMITS REPORT.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and nonpressurization events are ~~described~~ in ~~XN-NF-79-71(P) and XN-NF-84-105(A).~~ The principal result of this evaluation is the reduction in MCPR caused by the transient.
referenced SECTION 6.9.3 OF THE TECH. SPEC.

The purpose of the FLOW DEPENDENT $MCPR_f$ ~~of Figure 3.2.3-1~~ SPECIFIED IN THE CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required $MCPR_f$ is the maximum of the rated flow MCPR ~~determined from Table 3.2.3-1~~ and the reduced flow MCPR ~~determined from Figure 3.2.3-1~~ BOTH SPECIFIED IN THE CORE OPERATING LIMITS REPORT. $MCPR_f$ assures that the Safety Limit MCPR will not be violated. $MCPR_f$ is only calculated for the manual flow control mode. Automatic flow control operation is not permitted.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 DELETED

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.10 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.11 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall also include once a year an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

*In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

ALL CAPS Core Operating Limits Report

6.9.3.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specifications 3.2.1 and 3.4.1.

b. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

c. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.

and shall be documented in the CORE OPERATING LIMITS REPORT.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For WNP-2 the topical reports are:

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation", April 1990

2. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel", dated November 14, 1990

3. XN-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", November 1990

4. ANF-913(P)(A), Volume 1, Revision 1 and Volume 1, Supplements 2, 3 and 4, "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis", August 1990

5. ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option", January 1991.

6. XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4, "Exxon Nuclear Methodology for Boiling Water Reactors", November 1990

7. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads", June 1986

8. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description", January 1987

9. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactor Reload Fuel", September 1986

10. ANF-89-014(P), "Generic Mechanical Design for ANF 9x9-IX and 9x9-9X BWR Reload Fuel", May 1989

11. ANF-89-014(P), Supplement 1, "Generic Mechanical Design of ANF 9x9-IX and 9x9-9X BWR Reload Fuel", June 1990

12. (SER for 9x9 mechanical design)

13. XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology", November 1983

14. NEDE-24011-P-A-6, "General Electric Standard Application for Reactor Fuel", April 1983

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.