ATTACHMENT I

PROPOSED TECH. SPEC. CHANGES

CORE OPERATING LIMITS REPORT: CYCLE 6

CONTROLLED COPY

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS

CONTROLLED COPY

INDEX

ADMINISTRATIVE CONTROLS

CONTROLLED COPY

INDEX

LIST OF FIGURES

CONTROLLED COPY

INDEX

LIST OF FIGURES

CONTROLLED COPY

INDEX

LIST OF TABLES

DEFINITIONS

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.

INSERT A

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.

INSERT A

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.

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 ~~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,~~ the Core Operating Limits Report, 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 ~~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.

[illegible]

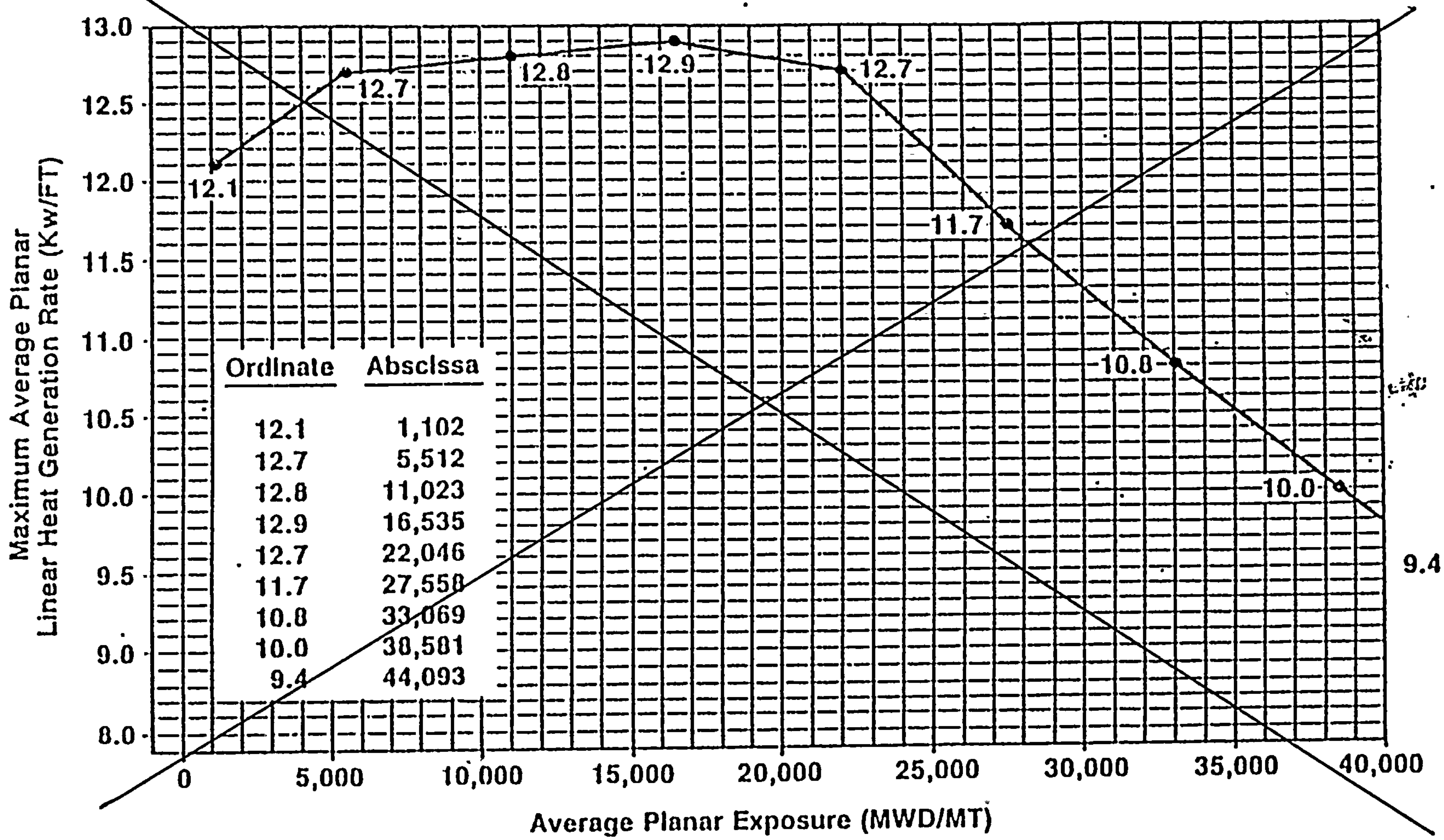
[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

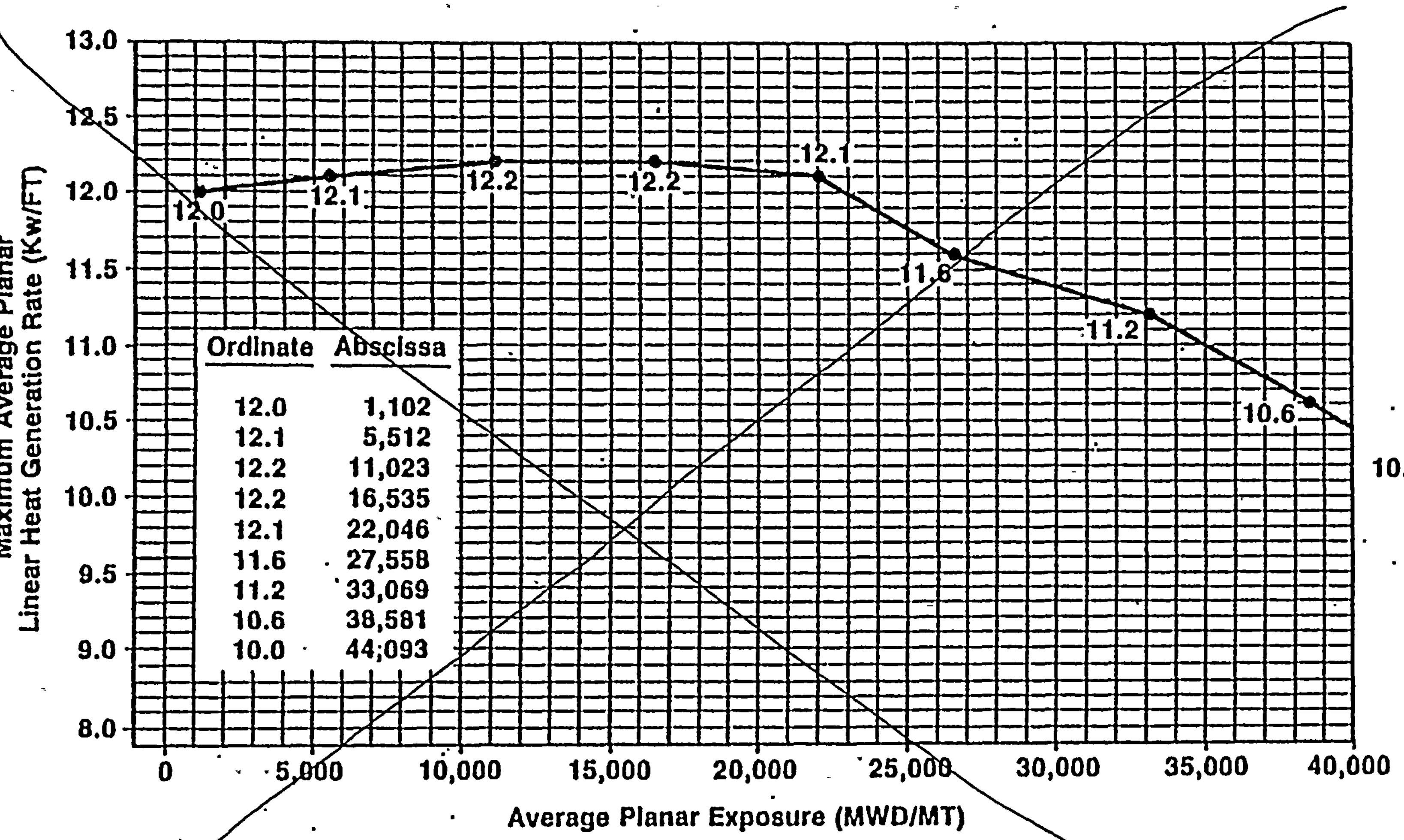


Ordinate	Abscissa
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

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

860599.4A

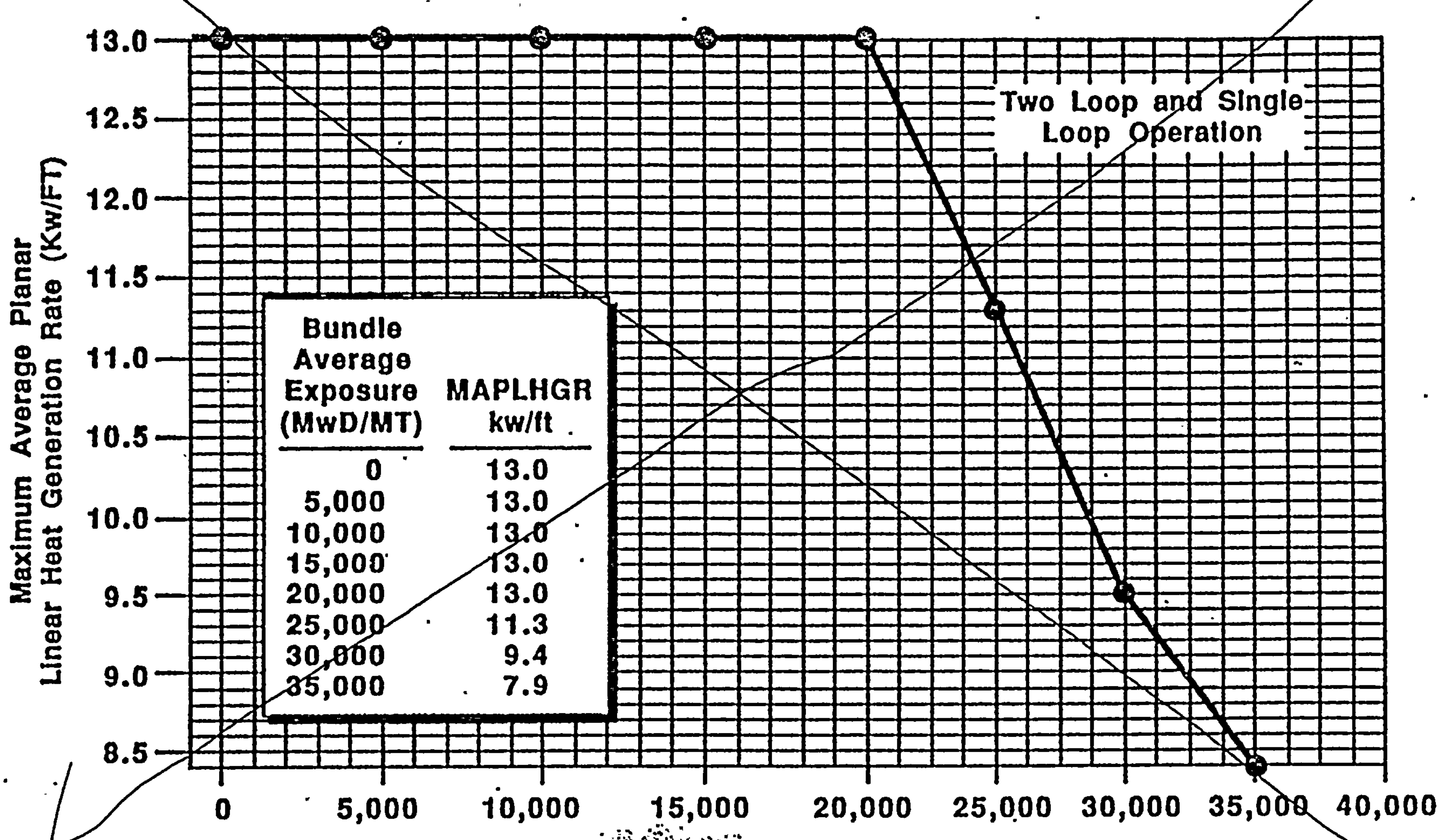
CONTROLLED COPY



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

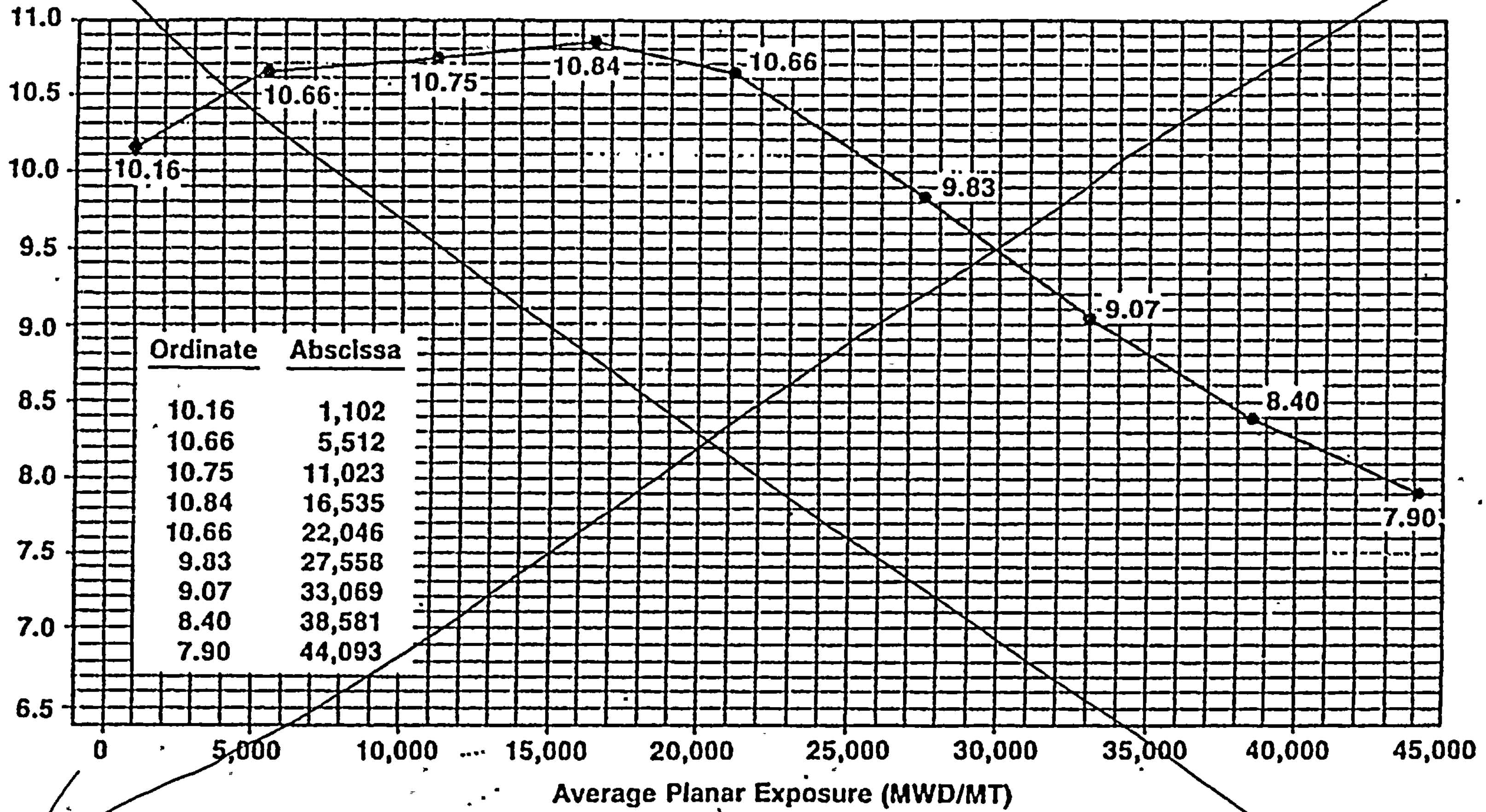
860599.5A



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



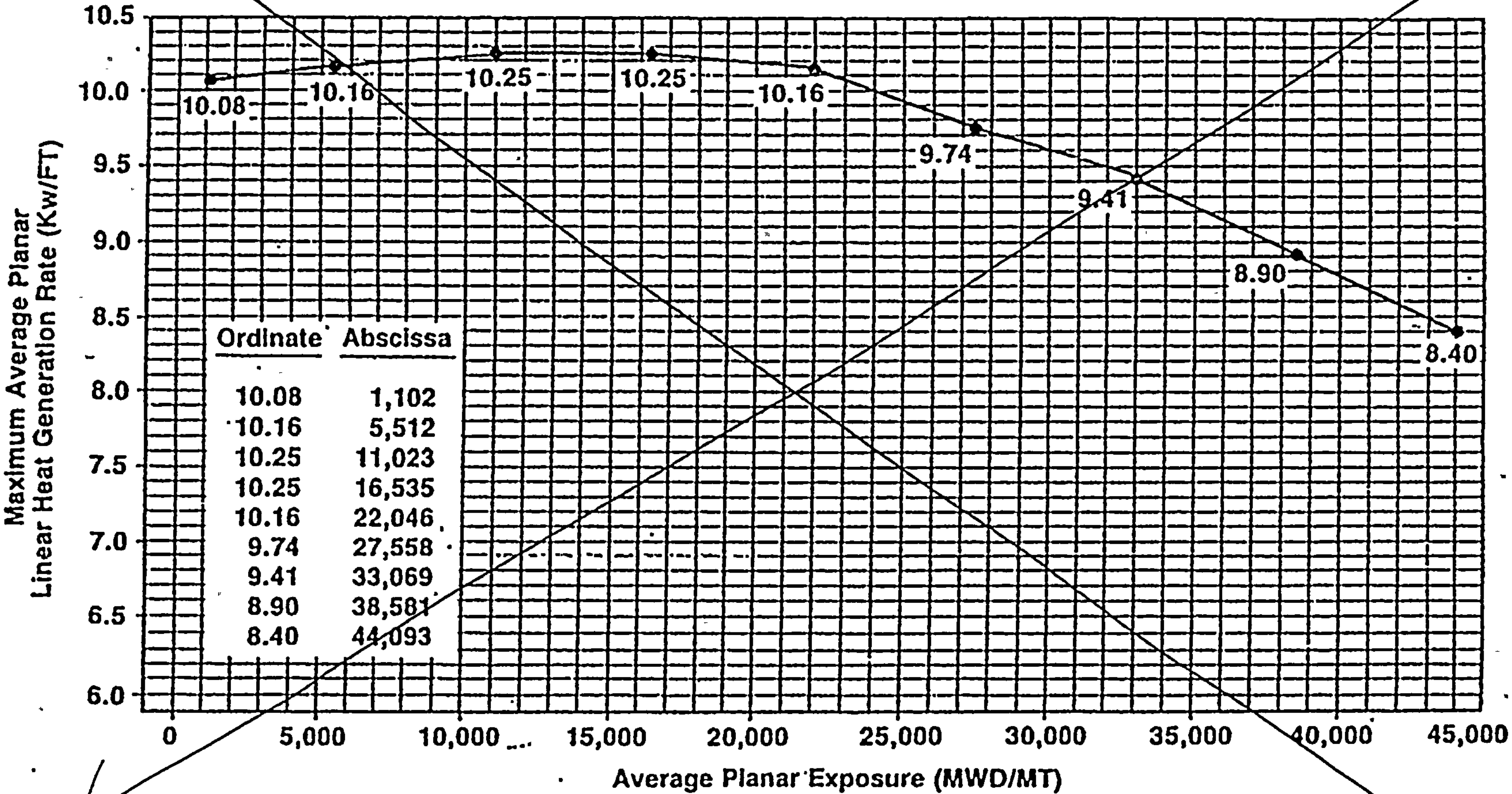
Ordinate	Abscissa
10.16	1,102
10.66	5,512
10.75	11,023
10.84	16,535
10.66	22,046
9.83	27,558
9.07	33,069
8.40	38,581
7.90	44,093

SINGLE LOOP

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

Figure 3.2.1-4

860599.5A



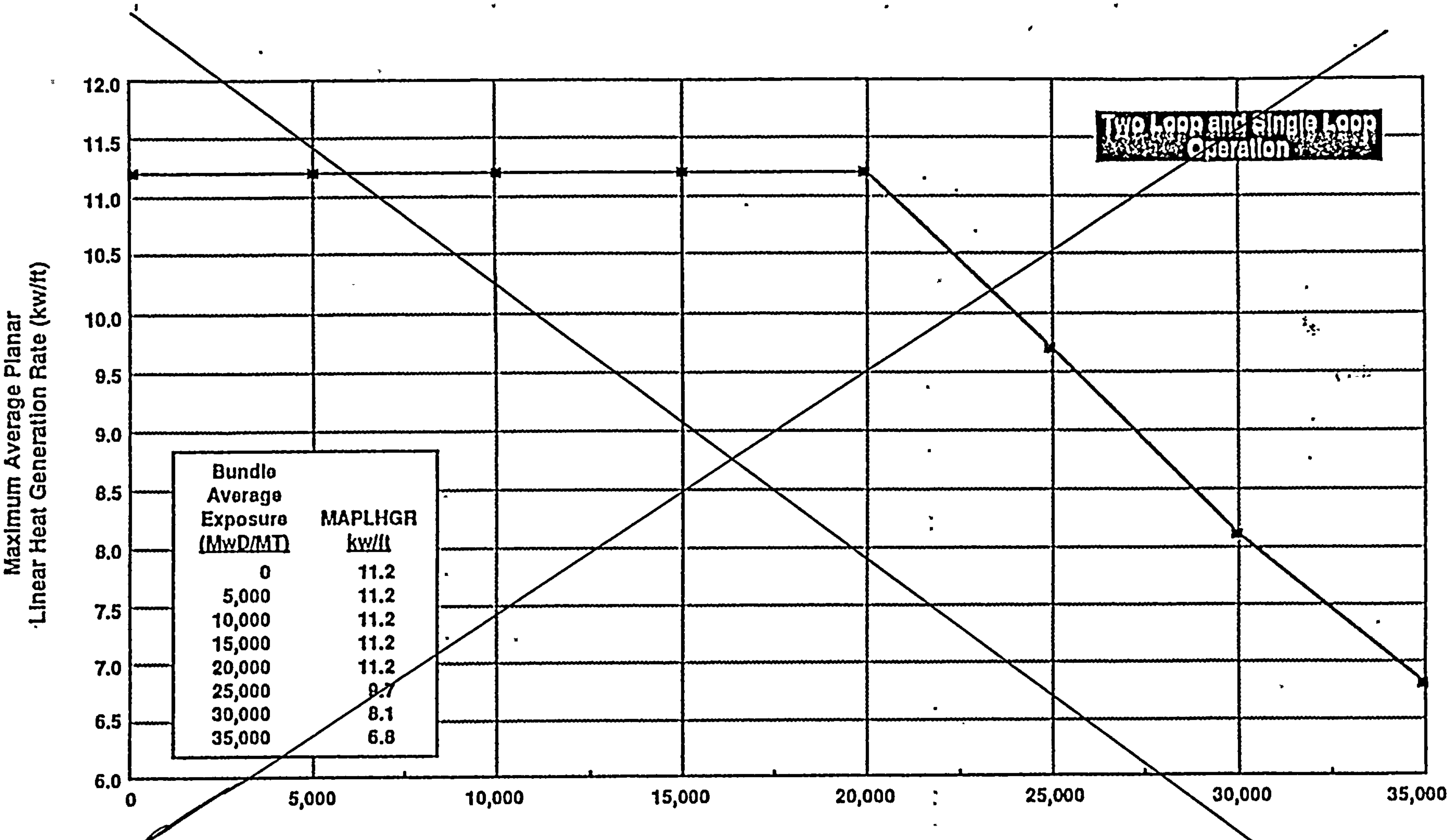
Ordinate	Abscissa
10.08	1,102
10.16	5,512
10.25	11,023
10.25	16,535
10.16	22,046
9.74	27,558
9.41	33,069
8.90	38,581
8.40	44,093

SINGLE LOOP

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

860599.5A

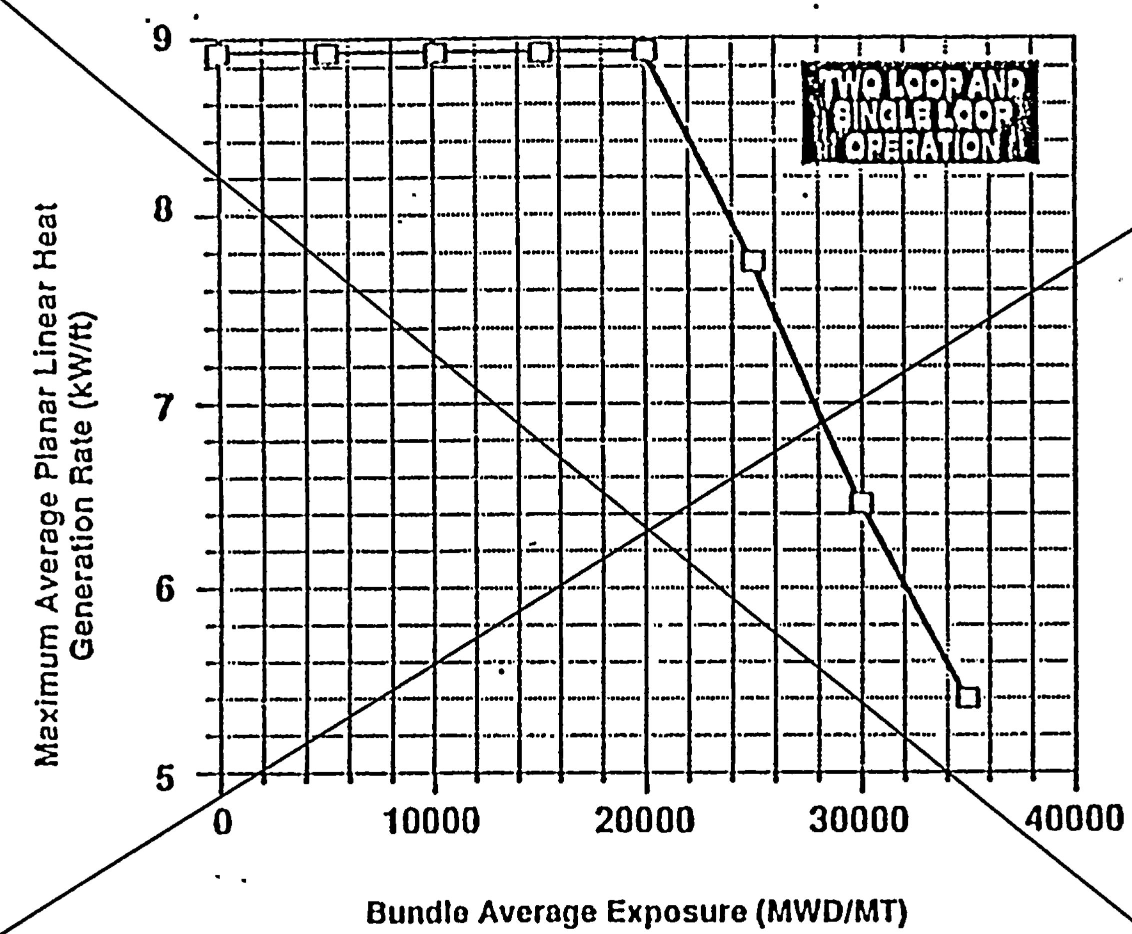
CONTROLLED COPY



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

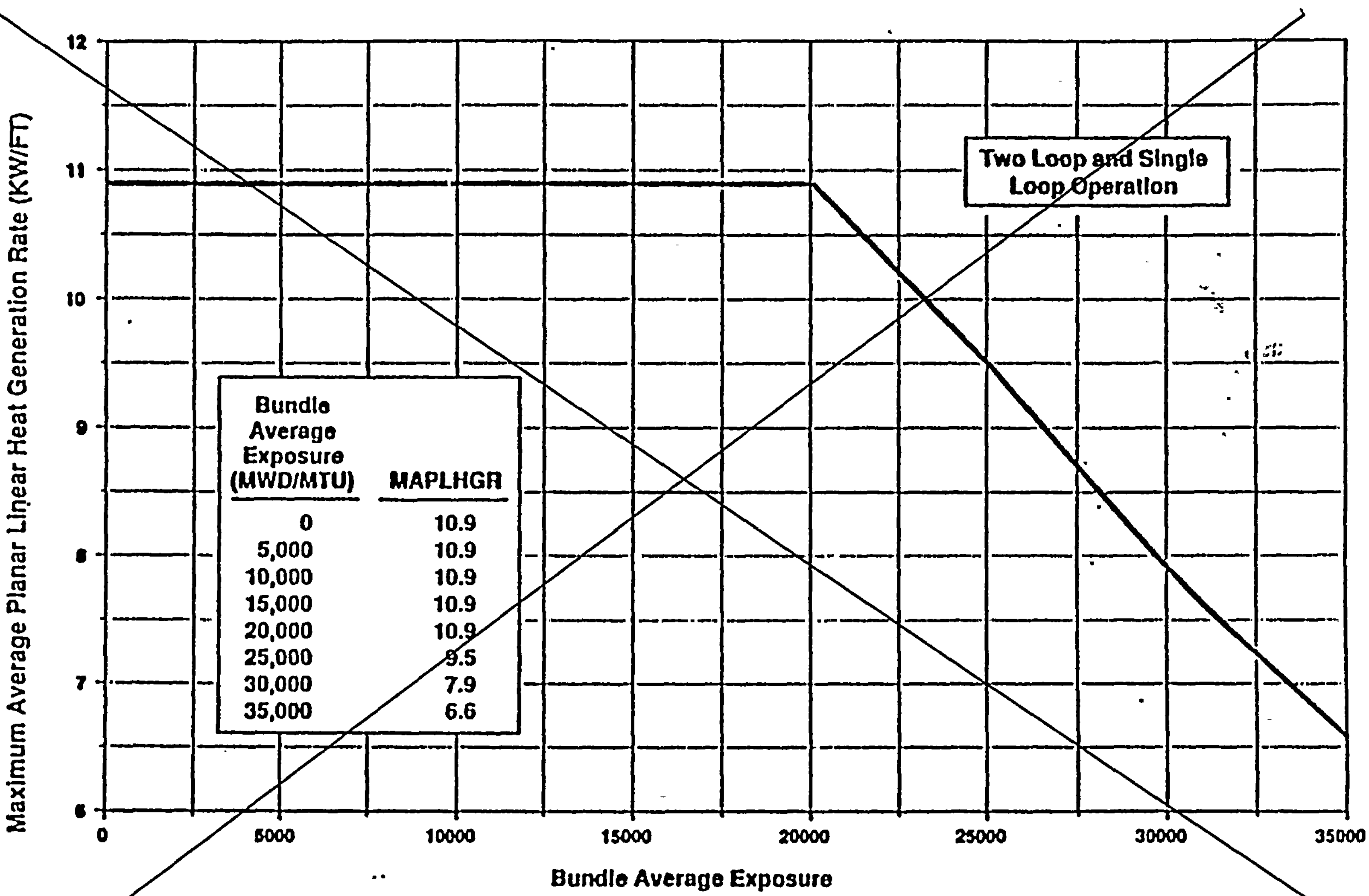
Figure 3.2.1-6



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



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

CONTROLLED COPY

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 the Core Operating Limits Report ~~Table 3.2.3-1 and Figure 3.2.3-1~~, 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.~~ from 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.

CONTROLLED COPY

Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow: 8x8 ANF Fuel***	MCPR Operating Limit Up to 106% Core Flow: 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.

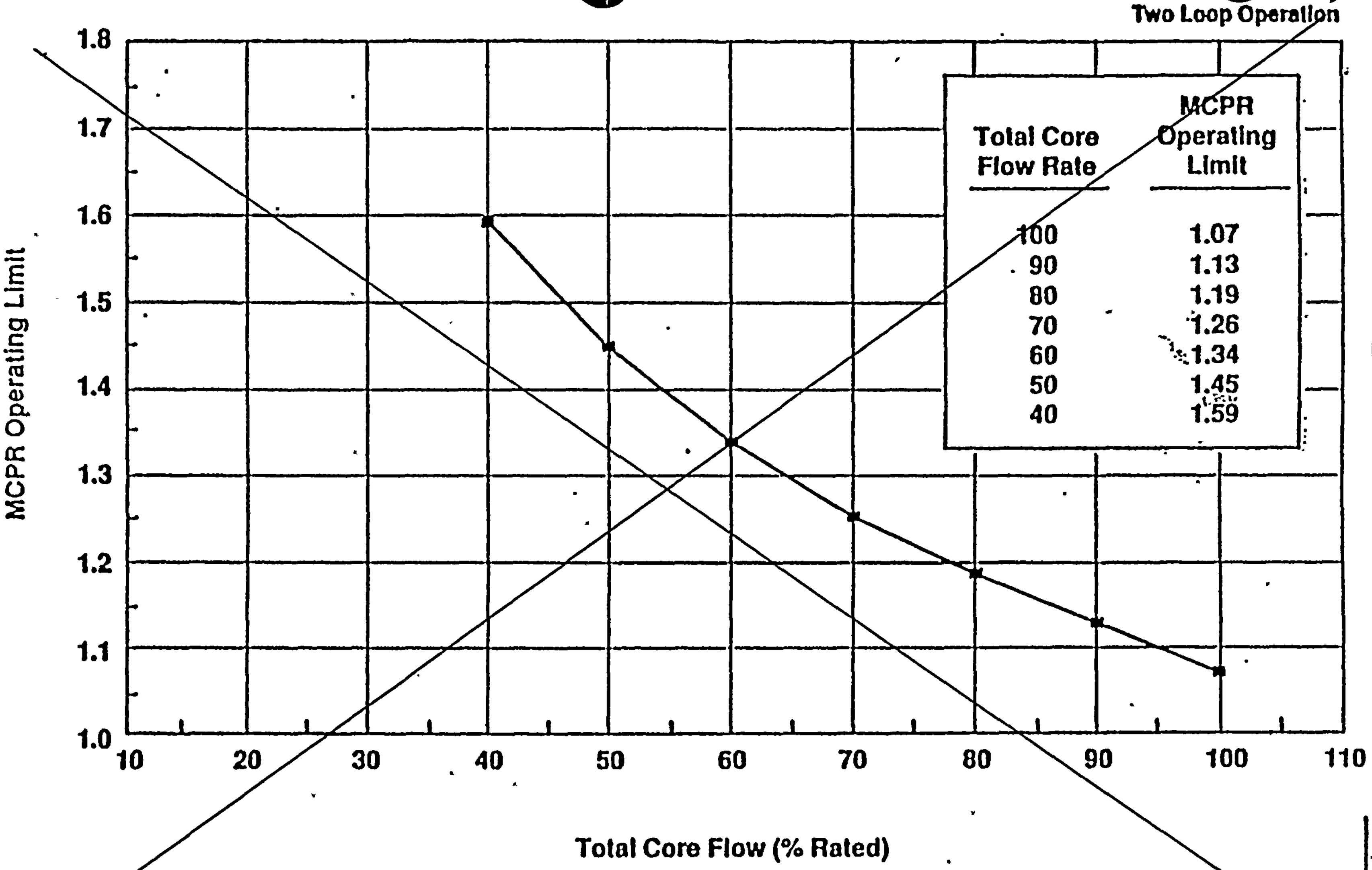
**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times 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

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

CONTROLLED COPY

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:

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.

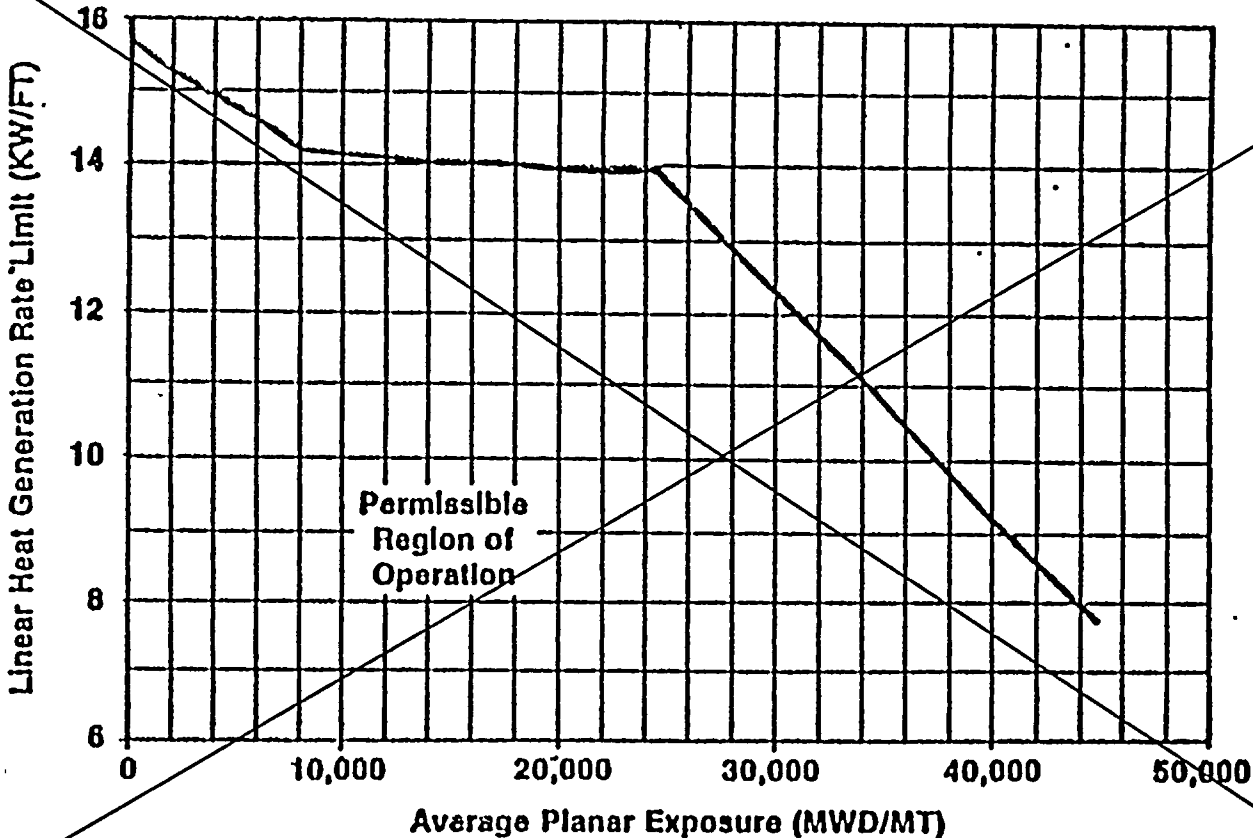
SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

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.

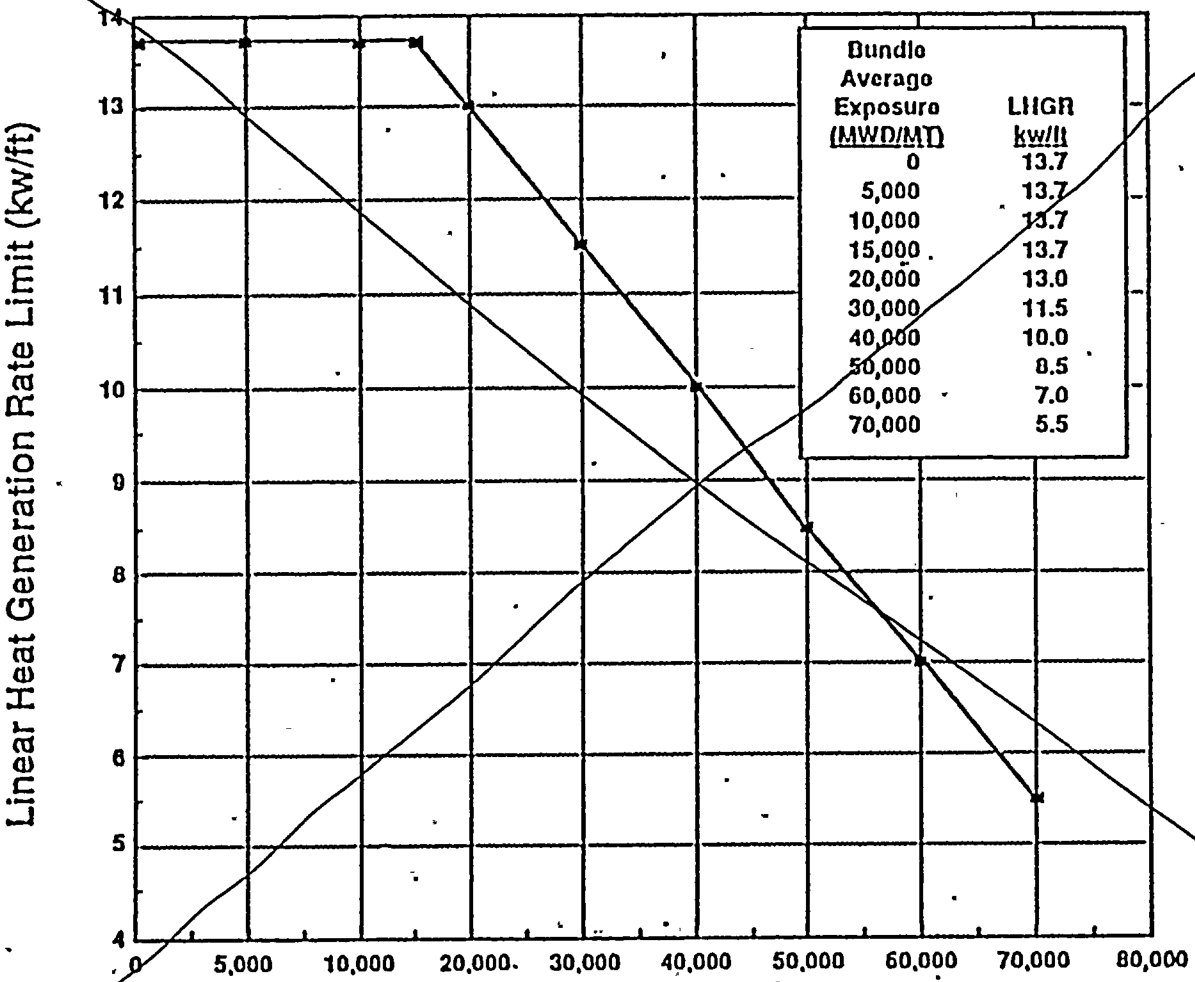


EXP	LHGR
0	15.62
510	15.621
2,580	15.10
5,230	14.71
7,940	14.19
10,470	14.13
13,220	14.06
15,990	14.06
18,708	14.00
21,590	13.93
24,420	13.93
27,280	13.08
30,150	12.24
33,050	11.40
35,960	10.47
38,900	9.55
41,830	8.65
44,760	7.77

ANF 8x8 Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

Figure 3.2.4-1

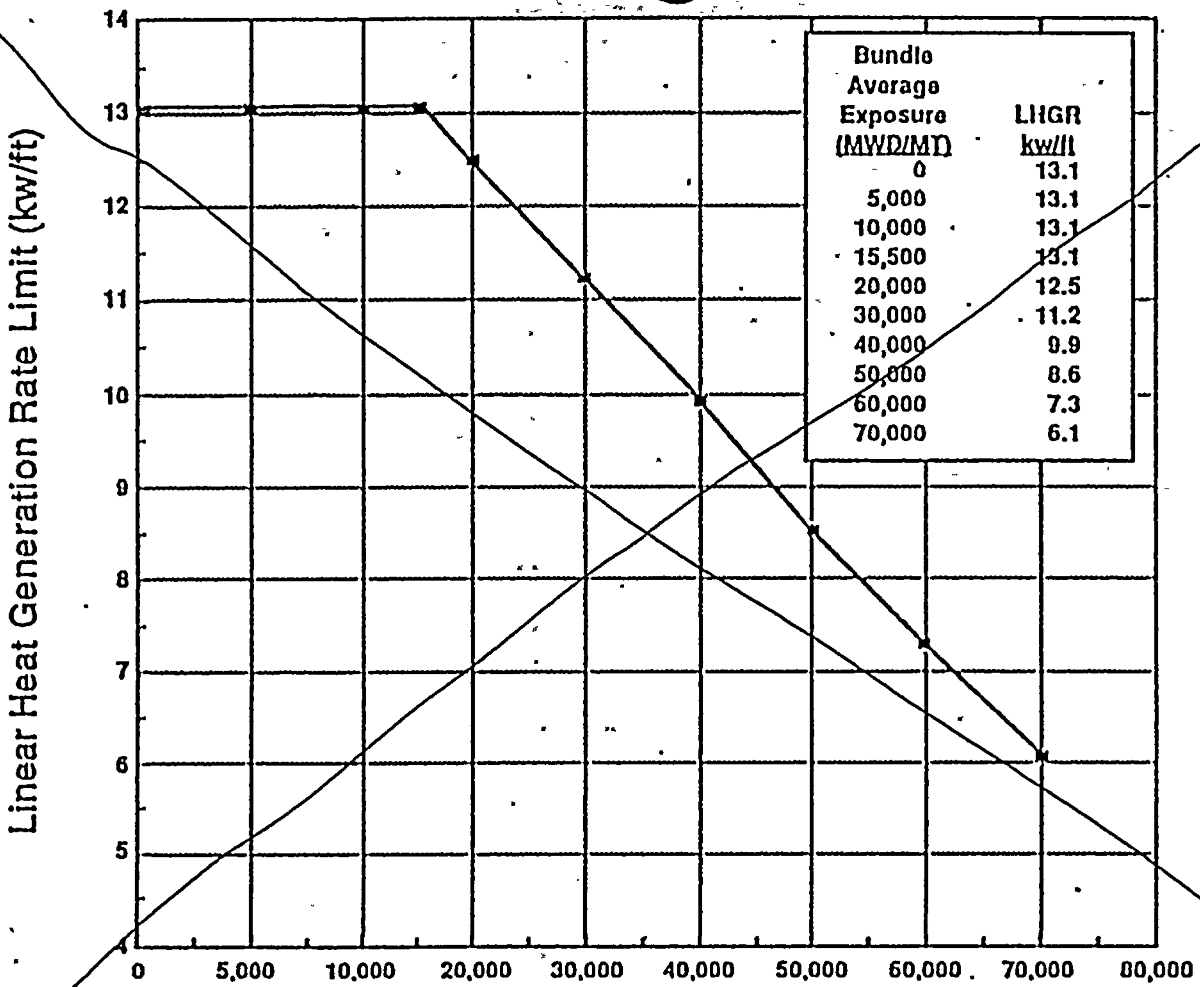
CONTROLLED COPY



Bundle Average Exposure (MWD/MT)	LHGR kw/ft
0	13.7
5,000	13.7
10,000	13.7
15,000	13.7
20,000	13.0
30,000	11.5
40,000	10.0
50,000	8.5
60,000	7.0
70,000	5.5

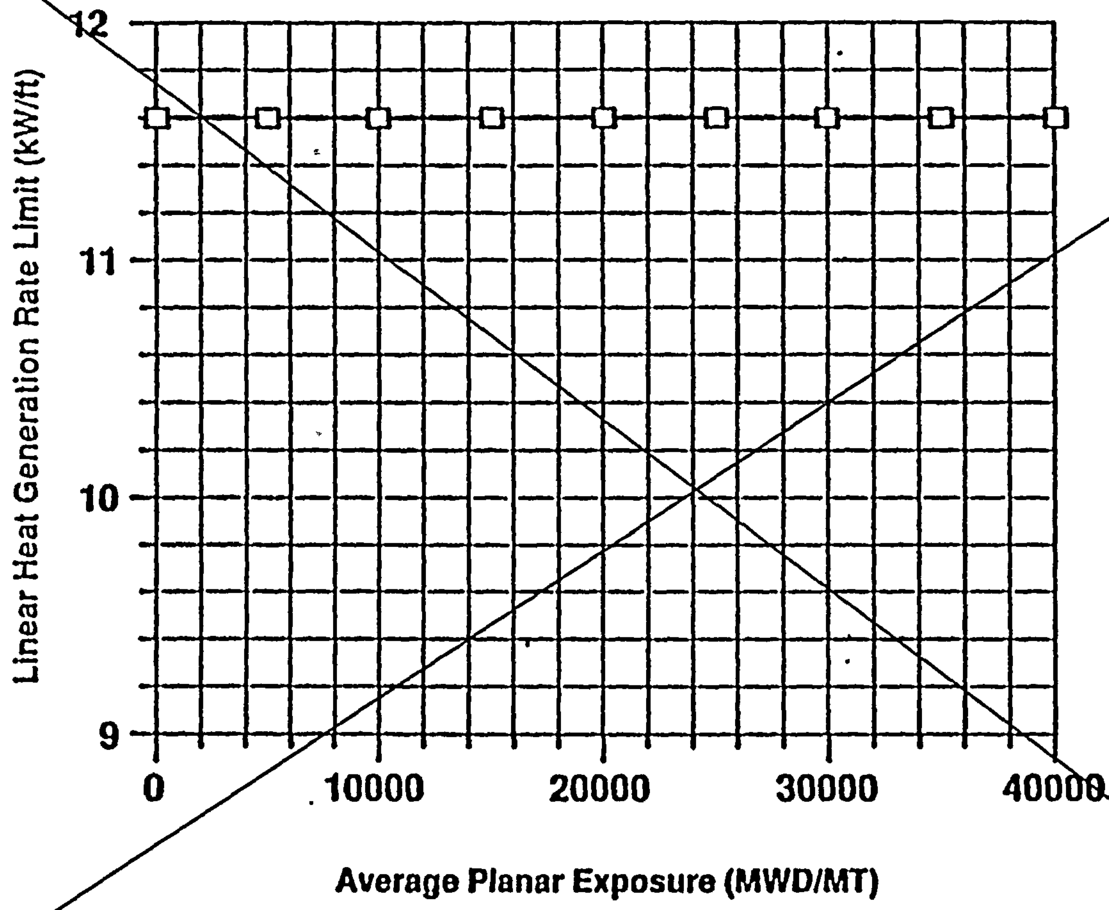
Average Planar Exposure (MWD/MT)
ANF 9 X 9 - IX Reload Fuel
Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure

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

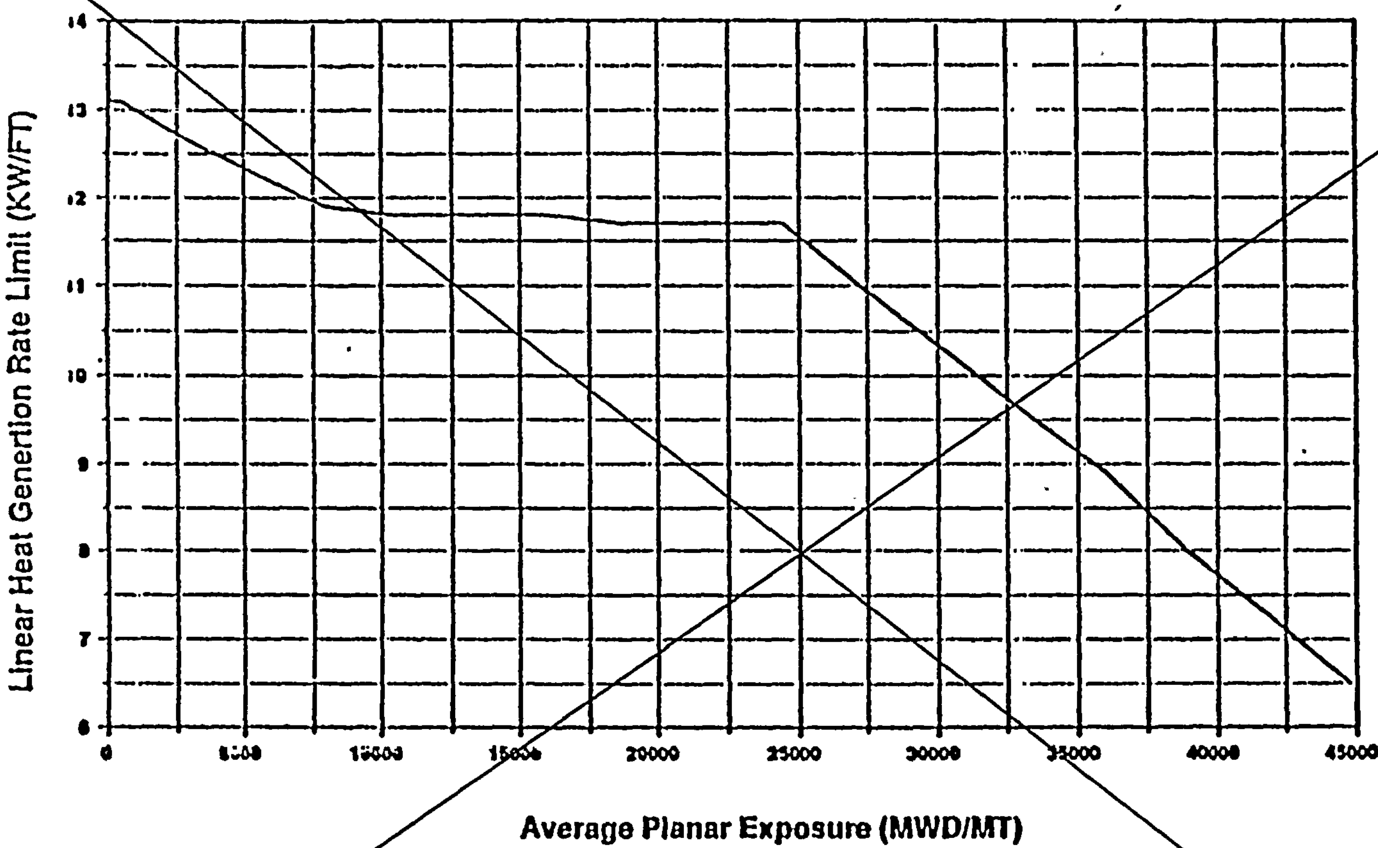
Figure 3.2.4-3



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



EXP	LHGR
0	13.1
510	13.1
2,580	12.7
5,230	12.3
7,940	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
27,280	11.0
30,150	10.3
33,050	9.6
35,960	8.9
38,900	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

CONTROLLED COPY

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 ~~two~~ SINGLE 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 ≤ 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.

[illegible]

[illegible]

[illegible]

[illegible]

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. 1~~ SECTION 6.9.3 OF THE TECH. SPECS.

CONTROLLED COPY

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~~ referenced in ~~XN-NF-79-71(P) and XN-NF-84-105(A).~~ SECTION 6.9.3 OF THE TECH. SPECS. The principal result of this evaluation is the reduction in MCPR caused by the transient.

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.

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.

INSERT B

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.

INSERT B

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.

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 reports previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For WNP-2, the topical reports are:

1. XN-NF-512(P)(A), "XN-3 Critical Power Correlation"

2. ANF-1125(P)(A), "ANFB Critical Power Correlation"

3. XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors"

4. XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors"

5. ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis"

6. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors"

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

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

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

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.

ATTACHMENT II
PROPOSED WNP-2 CYCLE 6
CORE OPERATING LIMITS REPORT

Controlled Copy No. __________

WNP-2

CYCLE 6

CORE OPERATING LIMITS REPORT

August, 1990

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WNP-2
CYCLE 6
CORE OPERATING LIMITS REPORT

LIST OF EFFECTIVE PAGES

PAGE	AMENDMENT
i	0
1	0
2	0
3	0
4	0
5	0
6	0
7	0
8	0
9	0
10	0
11	0
12	0
13	0
14	0
15	0
16	0
17	0
18	0
19	0
20	0

WNP-2 CYCLE 6

CORE OPERATING LIMITS REPORT

TABLE OF CONTENTS

1.0 INTRODUCTION AND SUMMARY

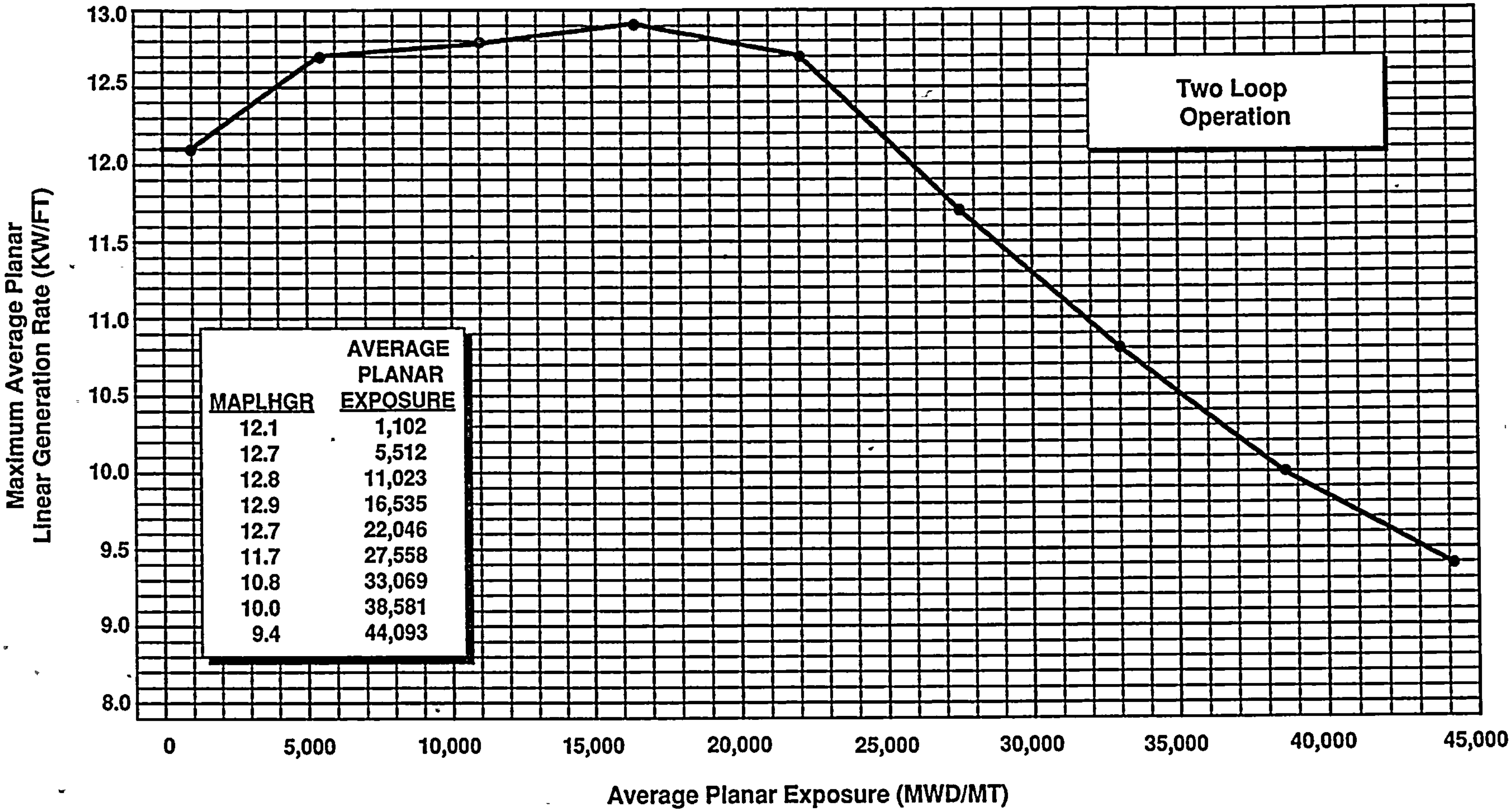
This report provides the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits, the MINIMUM CRITICAL POWER RATIO (MCPR) limits, and the LINEAR HEAT GENERATION RATE (LHGR) limits for WNP-2, Cycle 6 as required by Technical Specification 6.9.3.1. As required by Technical Specifications 6.9.3.2 and 6.9.3.3, these limits have been determined using NRC-approved methodology and are established such that all applicable limits of the plant safety analysis are met. The reload design is discussed in detail in the Cycle 6 Reload Summary Report (Reference 1.0). The thermal limits given here are developed in the Cycle 6 Transient Analysis Report (Reference 2.0), and the Cycle 6 Reload Analysis Report (Reference 3.0).

Included in the WNP-2 Cycle 6 reload are four General Electric (GE) lead fuel assemblies (LFA's) and four ABB Atom (ABB) LFA's. These LFA's have been designed to be compatible with the reload fuel assemblies that will constitute the remainder of the reload batch for Cycle 6. The Supply System will load the LFA's in core locations which have been analyzed to have sufficient margin such that the LFA's are not expected to be the limiting assemblies in the core on either a nodal or an assembly power basis. This approach is intended to prevent the possibility of the LFA's from ever being the limiting fuel assemblies. The GE11 LFA is described in the GE11 Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2 Reload 5, Cycle 6 (Reference 4.0). The SVEA-96 LFA is described in the Supplemental LFA Licensing Report - SVEA-96 LFA's for WNP-2 (Reference 5.0). The process for developing thermal limits for the SVEA-96 LFA fuel based upon the ANF Cycle 6 reload fuel thermal limits is described in this Reference.

Preparation, review and approval of this report were performed in accordance with applicable Supply System management, engineering and operating procedures. References 6.0 through 13.0 identify the specific topical report revisions and supplements which describe the methodology utilized in this cycle specific analysis.

2.0 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) LIMIT FOR USE IN TECHNICAL SPECIFICATION 3.2.1

All APLHGRs for use in Technical Specification 3.2.1 for each fuel type as a function of AVERAGE PLANAR EXPOSURE for General Electric (GE) initial fuel and AVERAGE BUNDLE EXPOSURE for Advanced Nuclear Fuels (ANF) fuel, GE11 LFA fuel and SVEA-96 LFA fuel, shall not exceed the limits shown in Figures 1, 2, 3, 6, 7 and 8 when in two-loop operation and Figures 3, 4, 5, 6, 7 and 8 when in single loop operation. For GE Fuel, with one recirculation loop not in operation, reduce the MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) to a value of 0.84 times the two loop limits found in Figures 1 and 2.

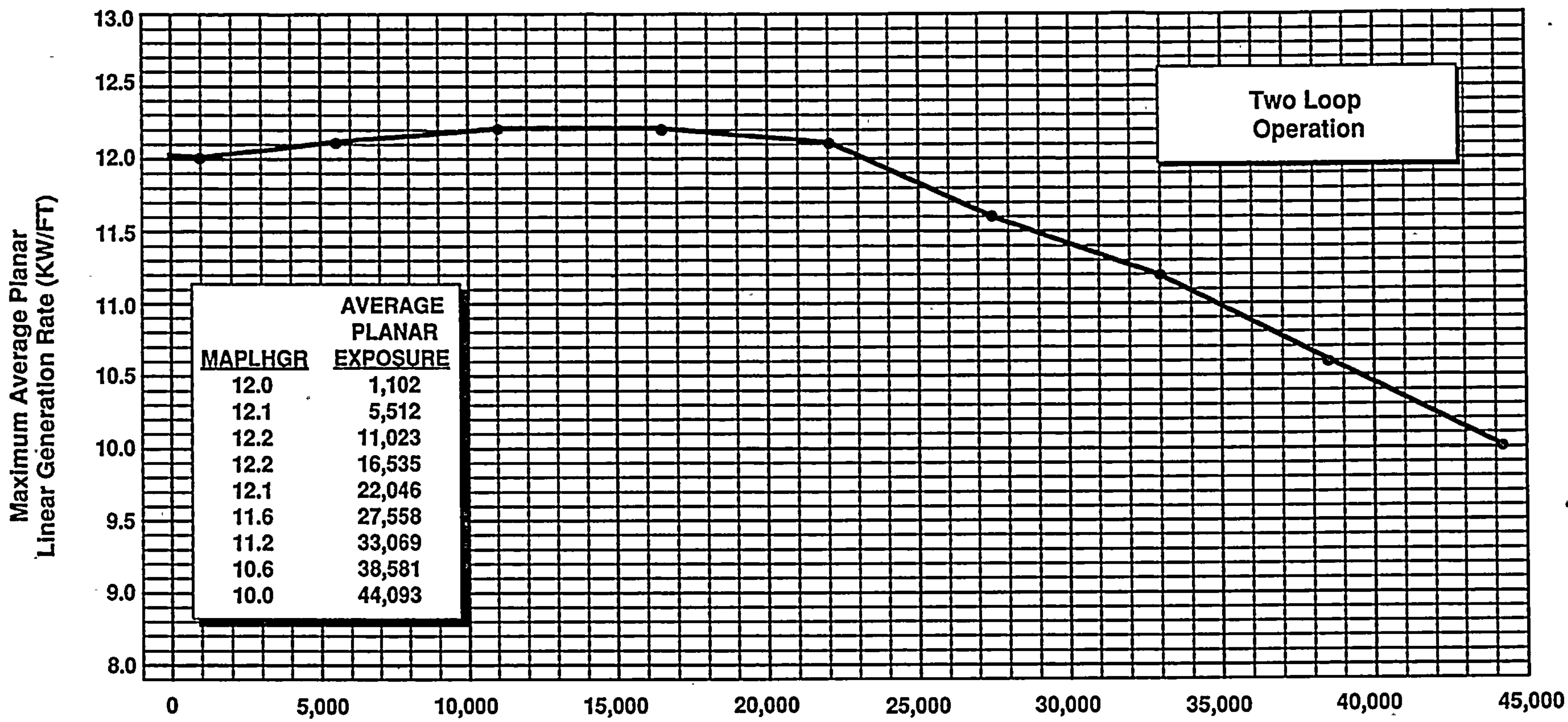


MAPLHGR	AVERAGE PLANAR EXPOSURE
12.1	1,102
12.7	5,512
12.8	11,023
12.9	16,535
12.7	22,046
11.7	27,558
10.8	33,069
10.0	38,581
9.4	44,093

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

Figure 1

900541.1
(6/90)



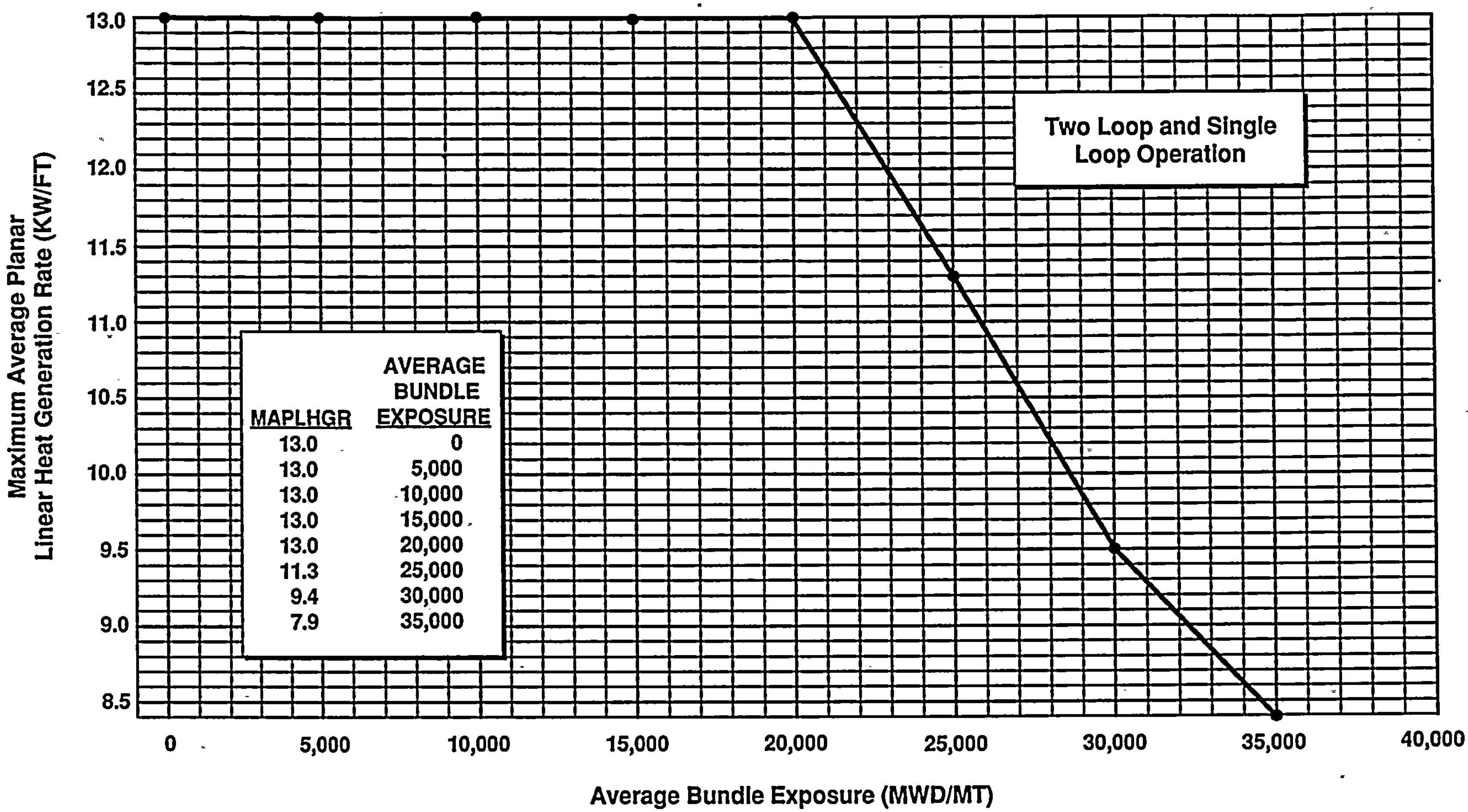
MAPLHGR	AVERAGE PLANAR EXPOSURE
12.0	1,102
12.1	5,512
12.2	11,023
12.2	16,535
12.1	22,046
11.6	27,558
11.2	33,069
10.6	38,581
10.0	44,093

Average Planar Exposure (MWD/MT)

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

Figure 2

900541.2
(6/90)

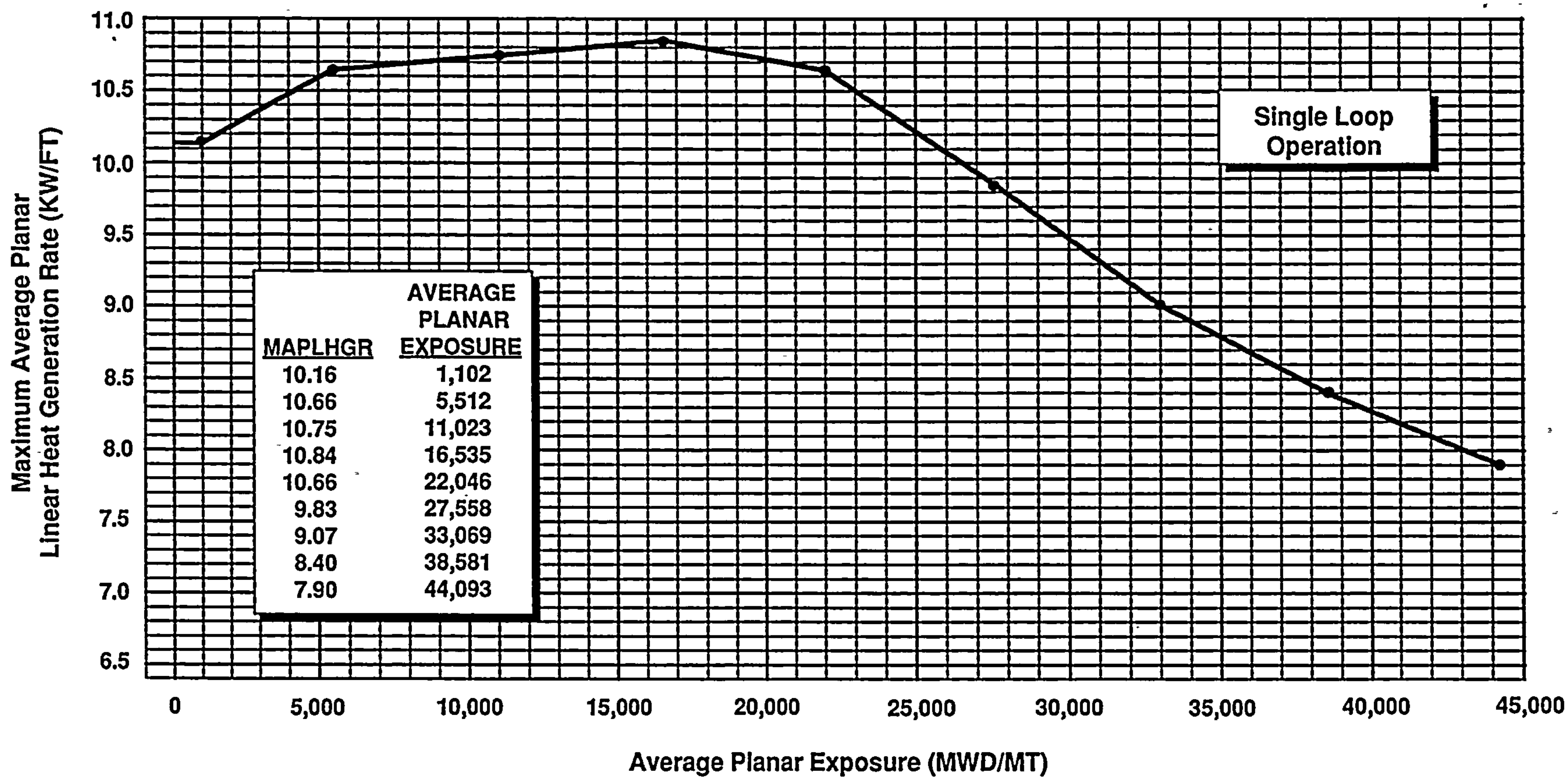


MAPLHGR	AVERAGE BUNDLE EXPOSURE
13.0	0
13.0	5,000
13.0	10,000
13.0	15,000
13.0	20,000
11.3	25,000
9.4	30,000
7.9	35,000

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

Figure 3

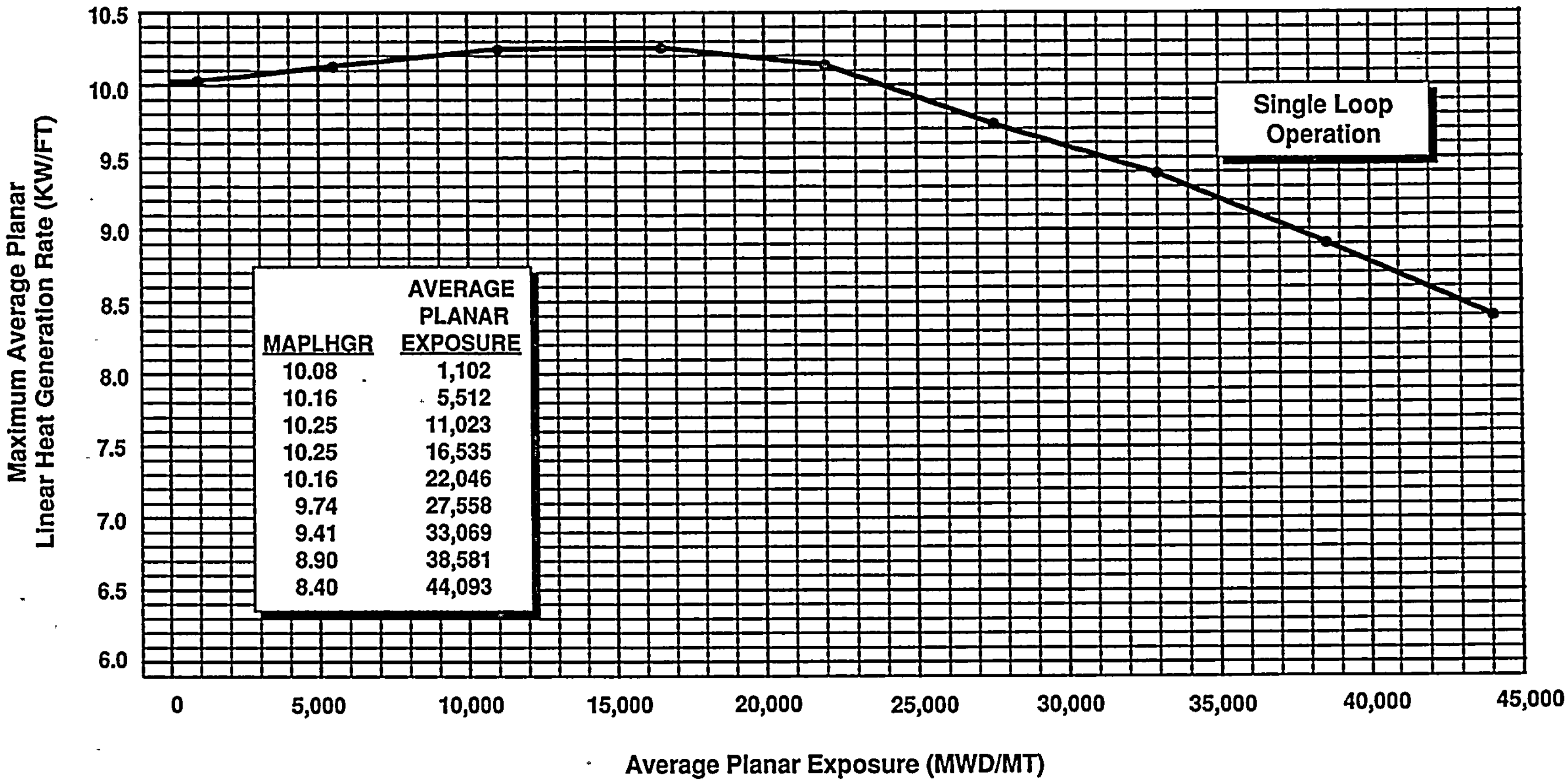
900541.3
(6/90)



MAPLHGR	AVERAGE PLANAR EXPOSURE
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

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

Figure 4

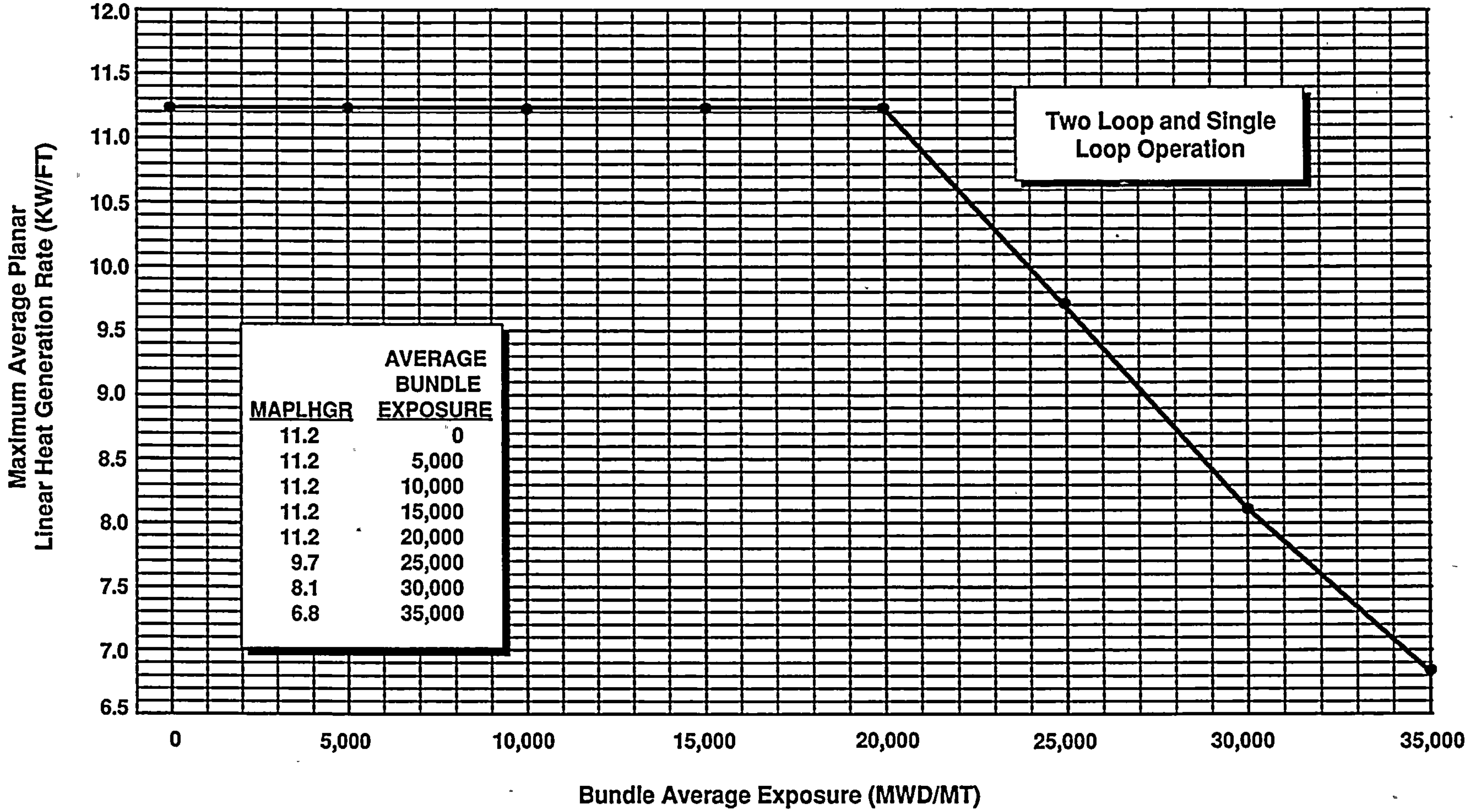


MAPLHGR	AVERAGE PLANAR EXPOSURE
10.08	1,102
10.16	5,512
10.25	11,023
10.25	16,535
10.16	22,046
9.74	27,558
9.41	33,069
8.90	38,581
8.40	44,093

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

Figure 5

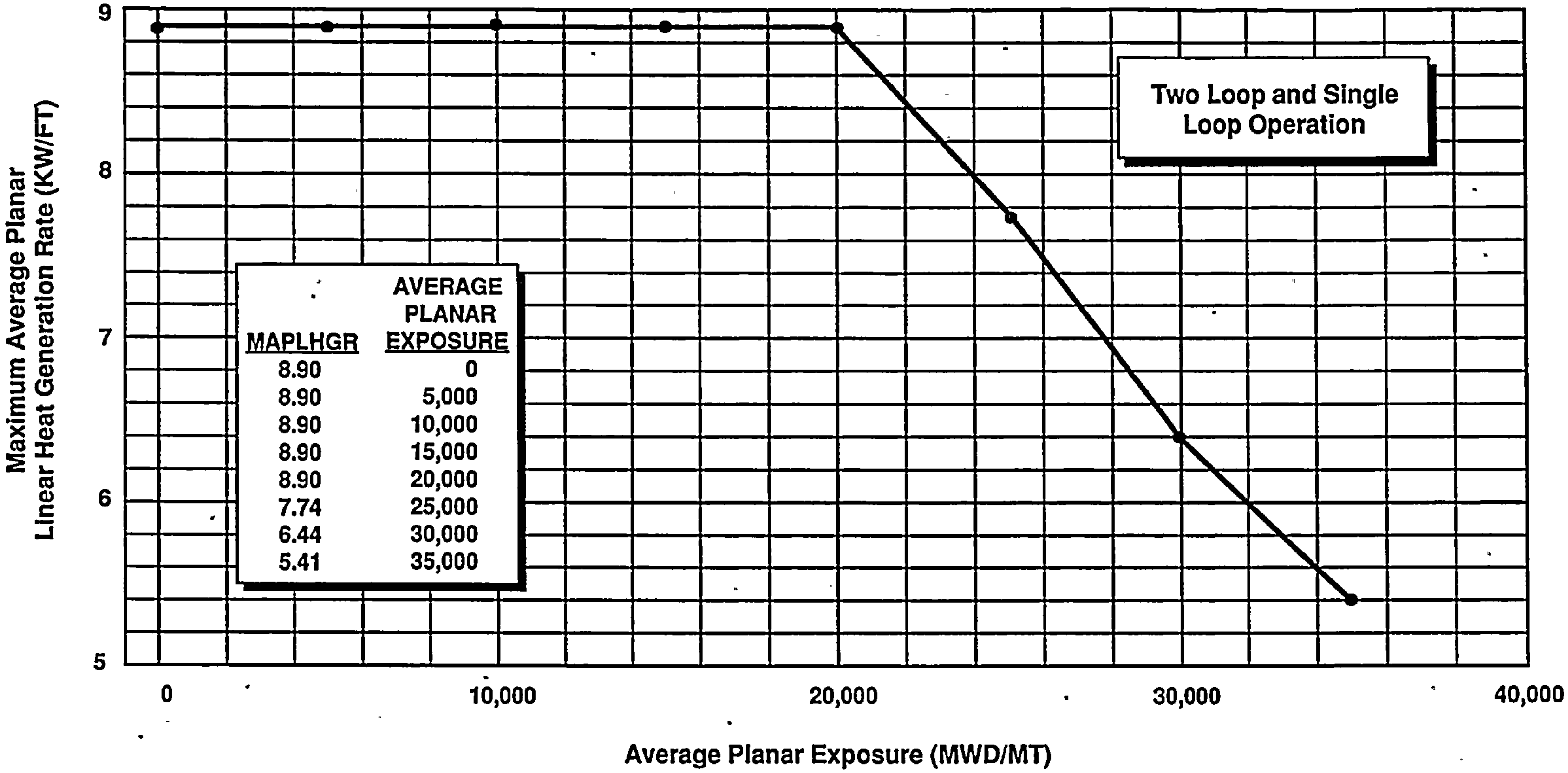
900541.5
(6/90)



Maximum Average Planar Linear Heat
Generation Rate (MAPLHGR) Versus
Bundle Average Exposure
ANF 9x9 - IX AND 9x9 - 9X Reload Fuel

Figure 6

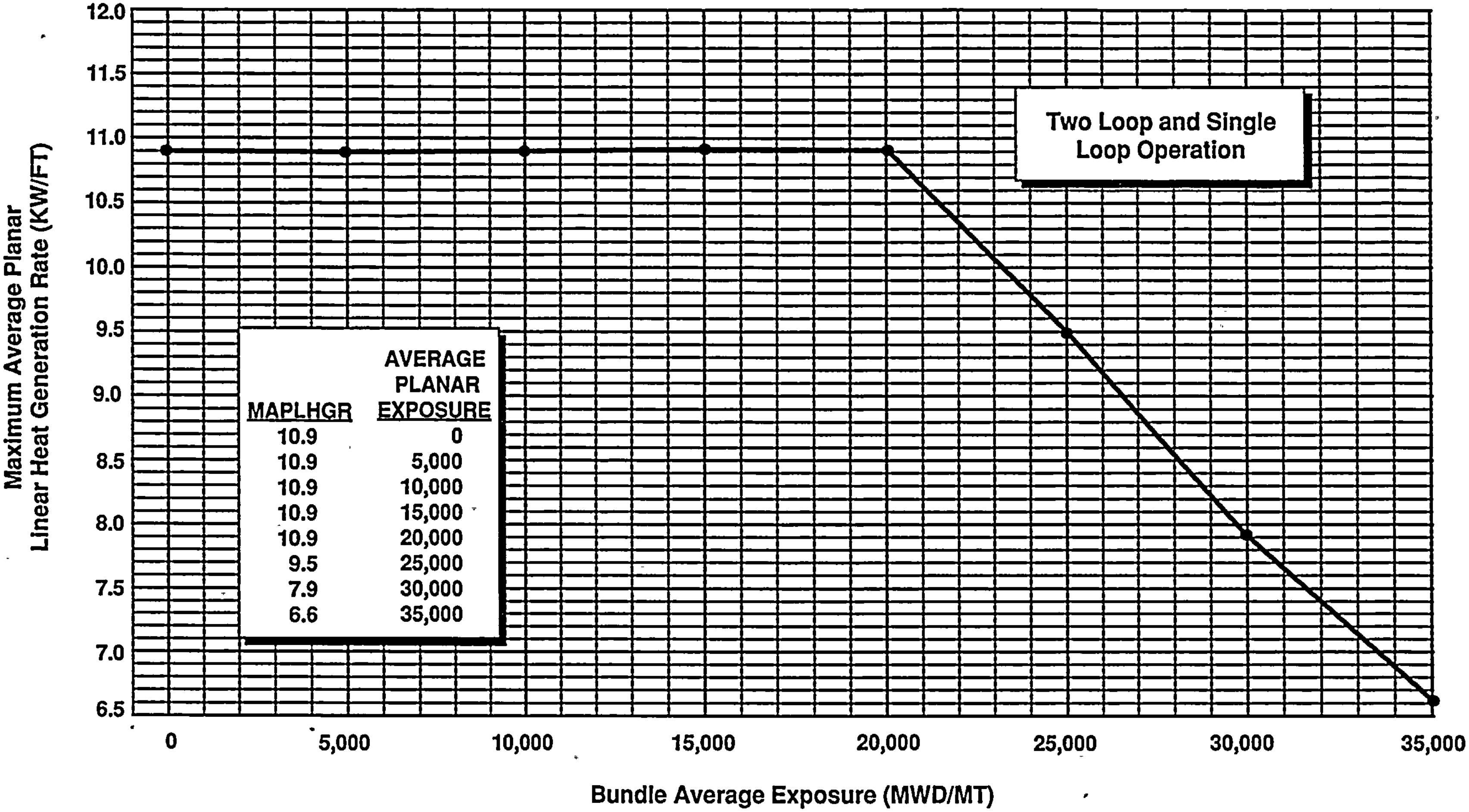
900541.6
(6/90)



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

Figure 7

900541.7
(6/90)



MAPLHGR	AVERAGE PLANAR EXPOSURE
10.9	0
10.9	5,000
10.9	10,000
10.9	15,000
10.9	20,000
9.5	25,000
7.9	30,000
6.6	35,000

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

Figure 8

900541.8
(6/90)

3.0 MINIMUM CRITICAL POWER RATION (MCPR) LIMIT FOR USE IN TECHNICAL SPECIFICATION 3.2.3

The MCPR limit for use in Technical Specification 3.2.3 shall be:

a. Greater than or equal to the applicable MCPR limit determined from Table 1 during steady state operation at or above rated core flow in two loop, or when in single loop operation; or;

b. Greater than or equal to the greater of the two limits determined from Table 1 and Figure 9 during steady state operation at less than rated core flow when in two recirculation loop operation.

TABLE 1

MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR OPERATING LIMIT UP TO 106% CORE FLOW ANF 8x8 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 inoperable 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.

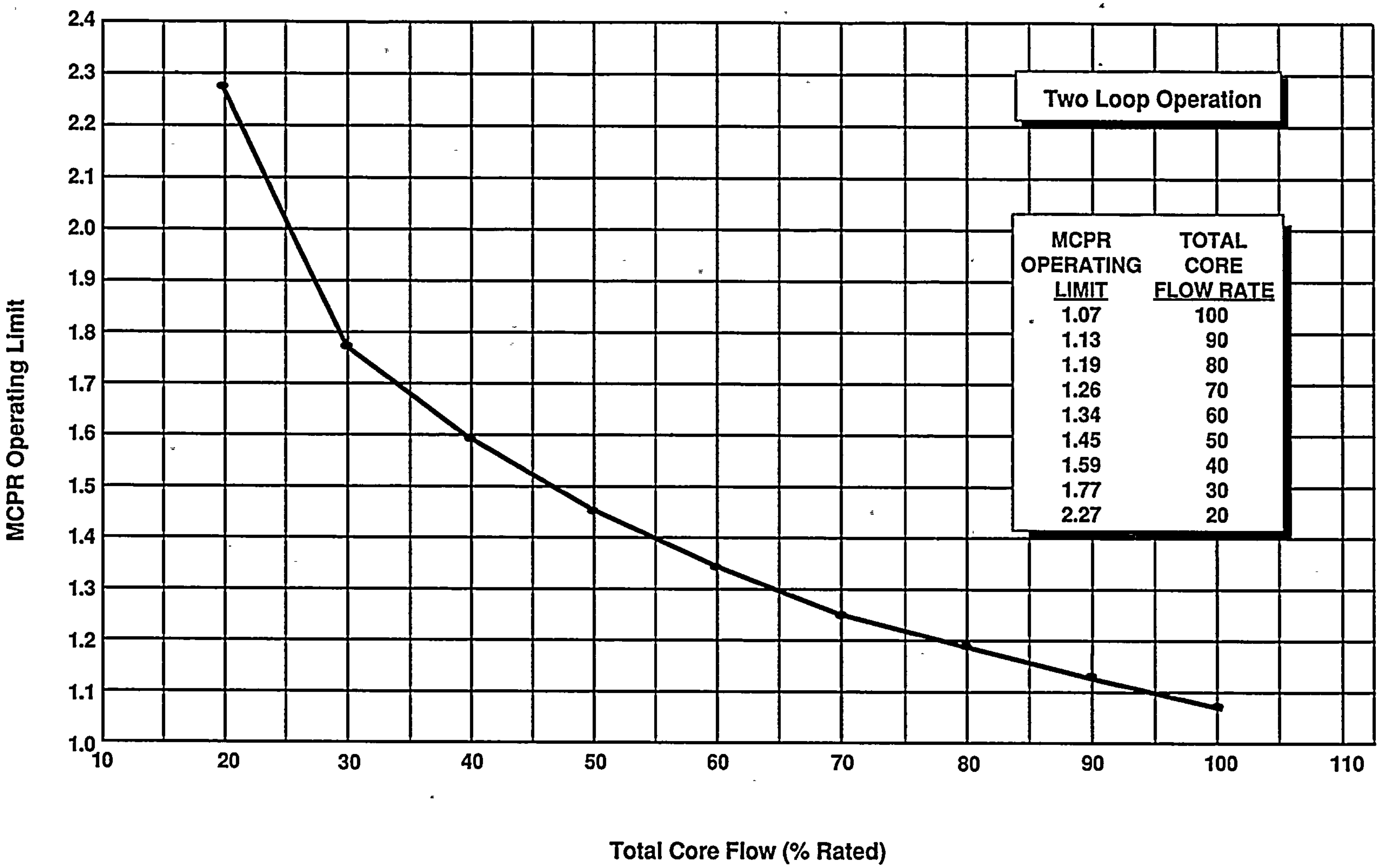
TABLE 1 (Continued)

MCPR OPERATING LIMITS

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 two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624

***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.

****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.



MCPR OPERATING LIMIT	TOTAL CORE FLOW RATE
1.07	100
1.13	90
1.19	80
1.26	70
1.34	60
1.45	50
1.59	40
1.77	30
2.27	20

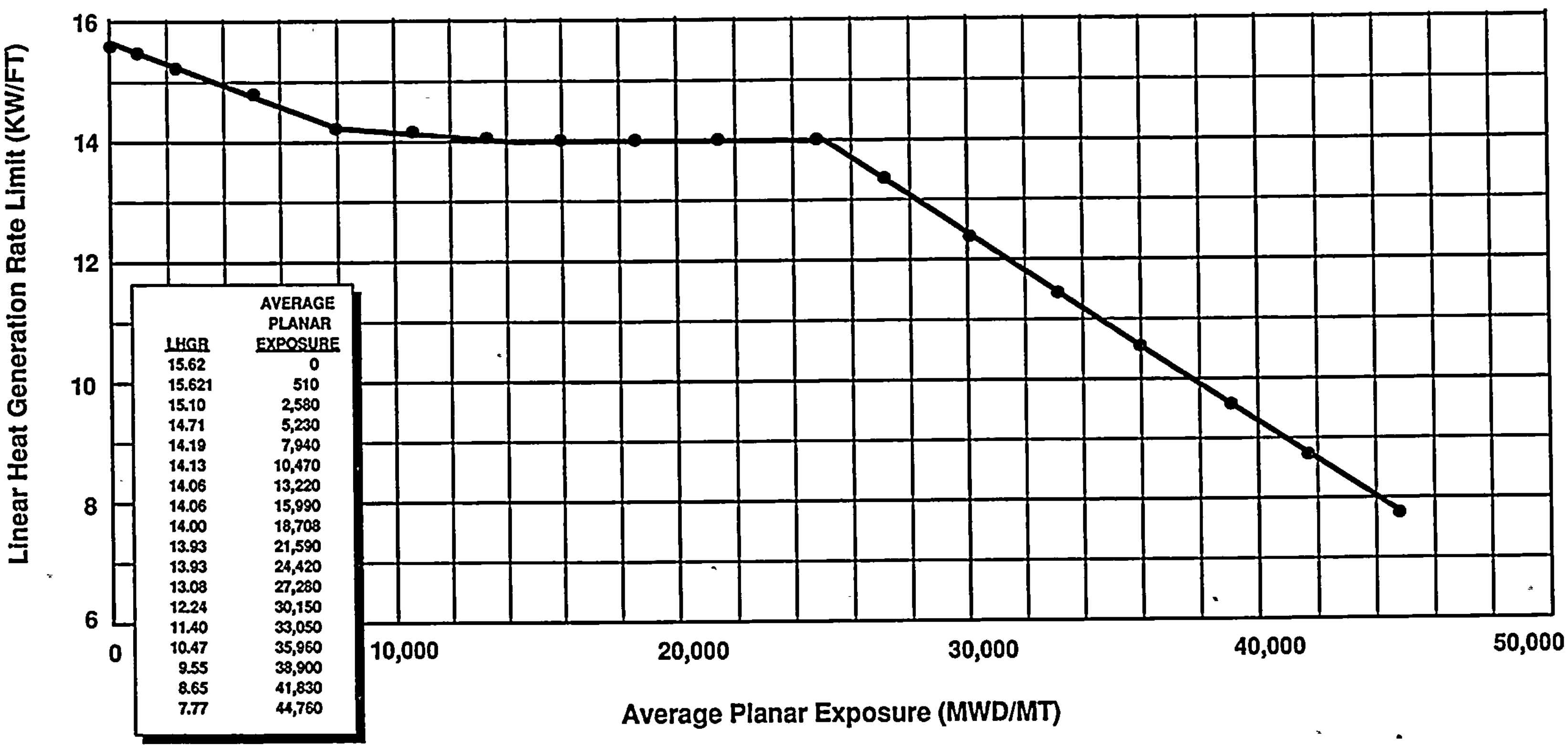
Reduced Flow MCPR Operating Limit
Versus Total Core Flow
All Fuel in WNP-2 Cycle 6
This curve is applicable to FFTR operation

Figure 9

900541.9
(6/90)

4.0 LINEAR HEAT GENERATION RATE (LHGR) LIMIT FOR USE IN TECHNICAL SPECIFICATION 3.2.4

The LHGR limit for use in Technical Specification 3.2.4 for GE initial core fuel shall not exceed 13.4 kw/ft. The LHGR limit for use in Technical Specification 3.2.4 for reload fuel shall not exceed the values shown in Figures 10, 11, 12, 13 and 14.

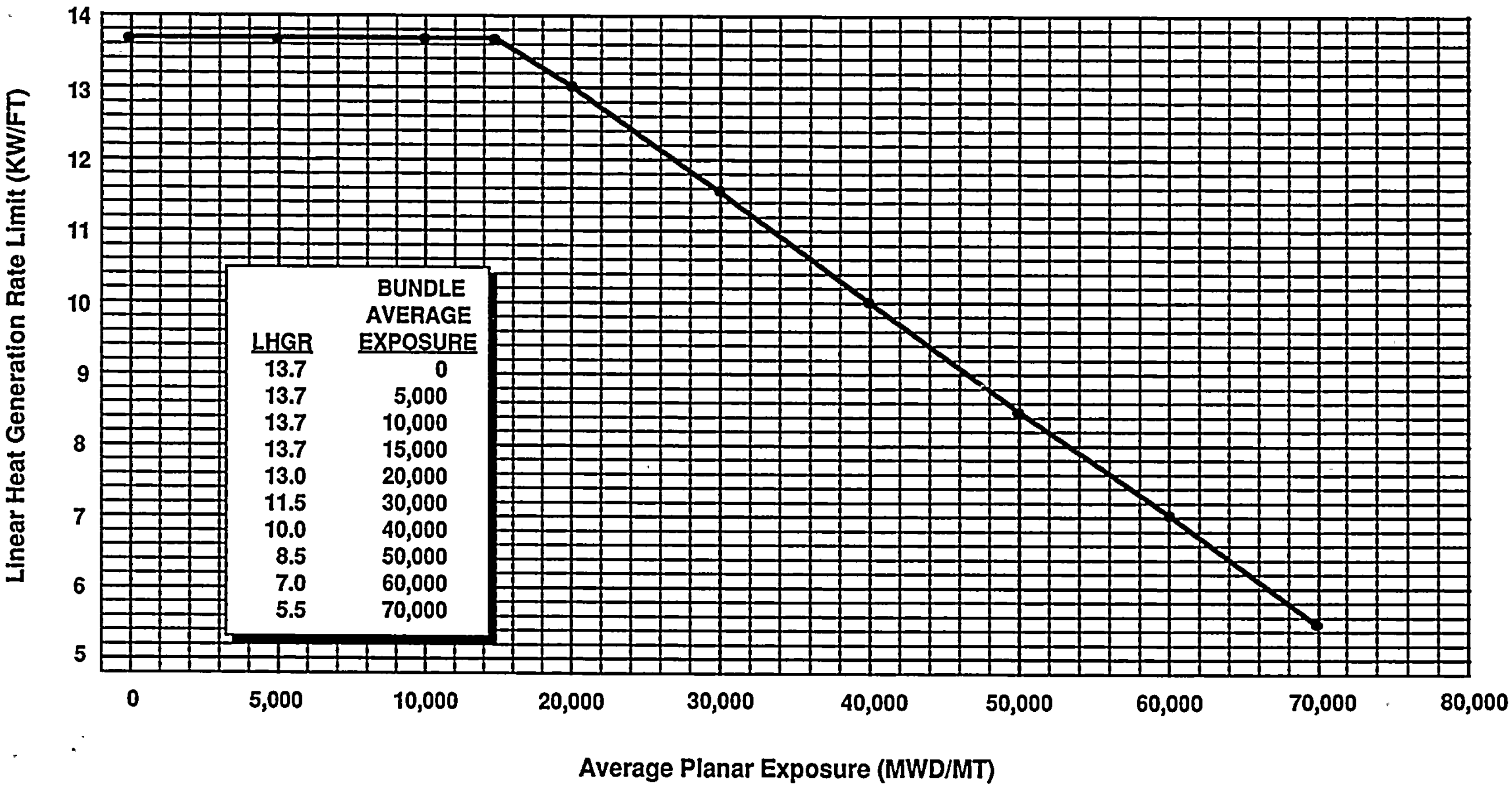


LHGR	AVERAGE PLANAR EXPOSURE
15.62	0
15.621	510
15.10	2,580
14.71	5,230
14.19	7,940
14.13	10,470
14.06	13,220
14.06	15,990
14.00	18,708
13.93	21,590
13.93	24,420
13.08	27,280
12.24	30,150
11.40	33,050
10.47	35,960
9.55	38,900
8.65	41,830
7.77	44,760

Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
ANF 8x8 Reload Fuel

Figure 10

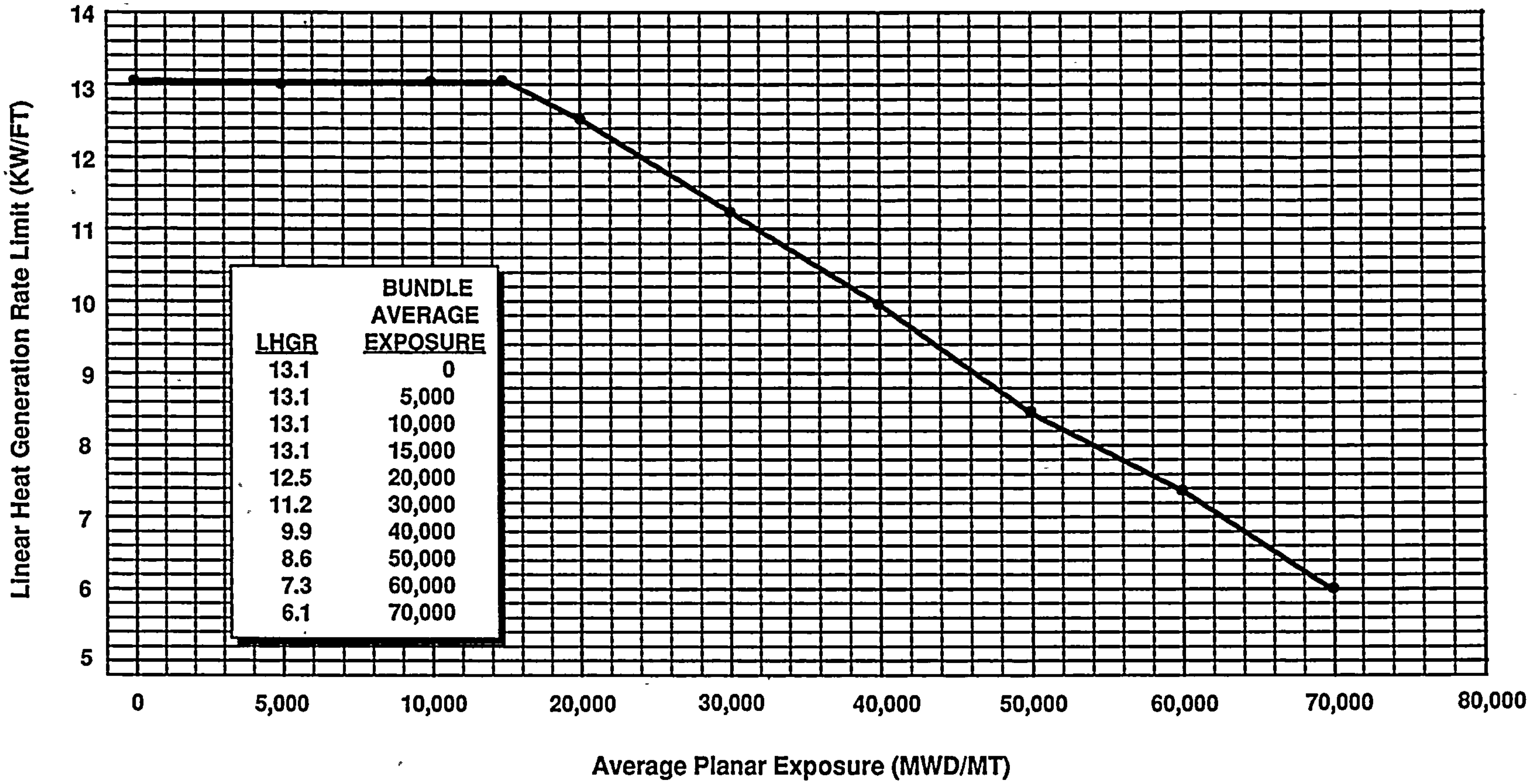
900541.10
(6/90)



Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
ANF 9x9 - IX Reload Fuel

Figure 11

900541.11
(6/90)

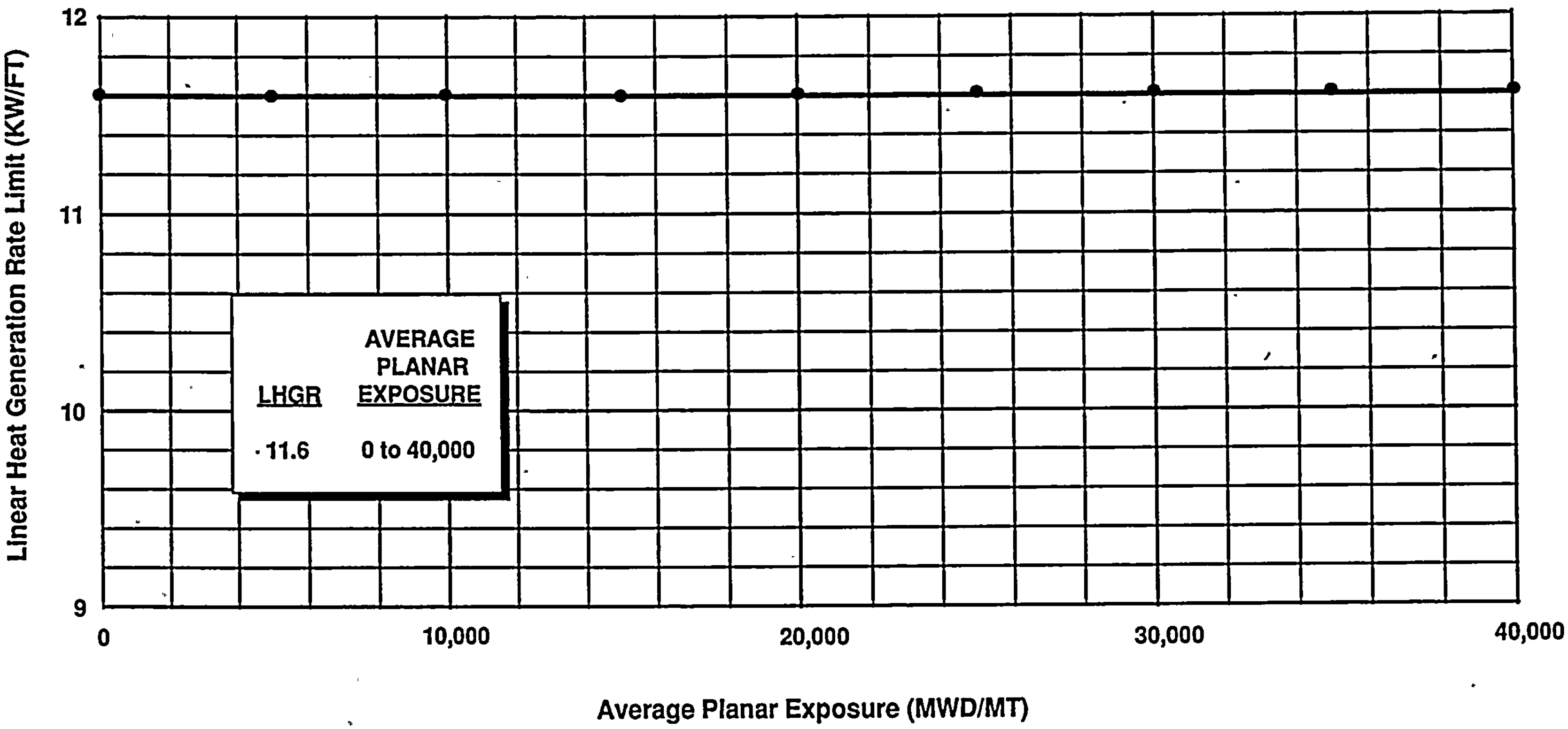


LHGR	BUNDLE AVERAGE EXPOSURE
13.1	0
13.1	5,000
13.1	10,000
13.1	15,000
12.5	20,000
11.2	30,000
9.9	40,000
8.6	50,000
7.3	60,000
6.1	70,000

Linear Heat Generation Rate (LHGR) Limit
Versus Average Planar Exposure
ANF 9x9 - 9X Reload Fuel

Figure 12

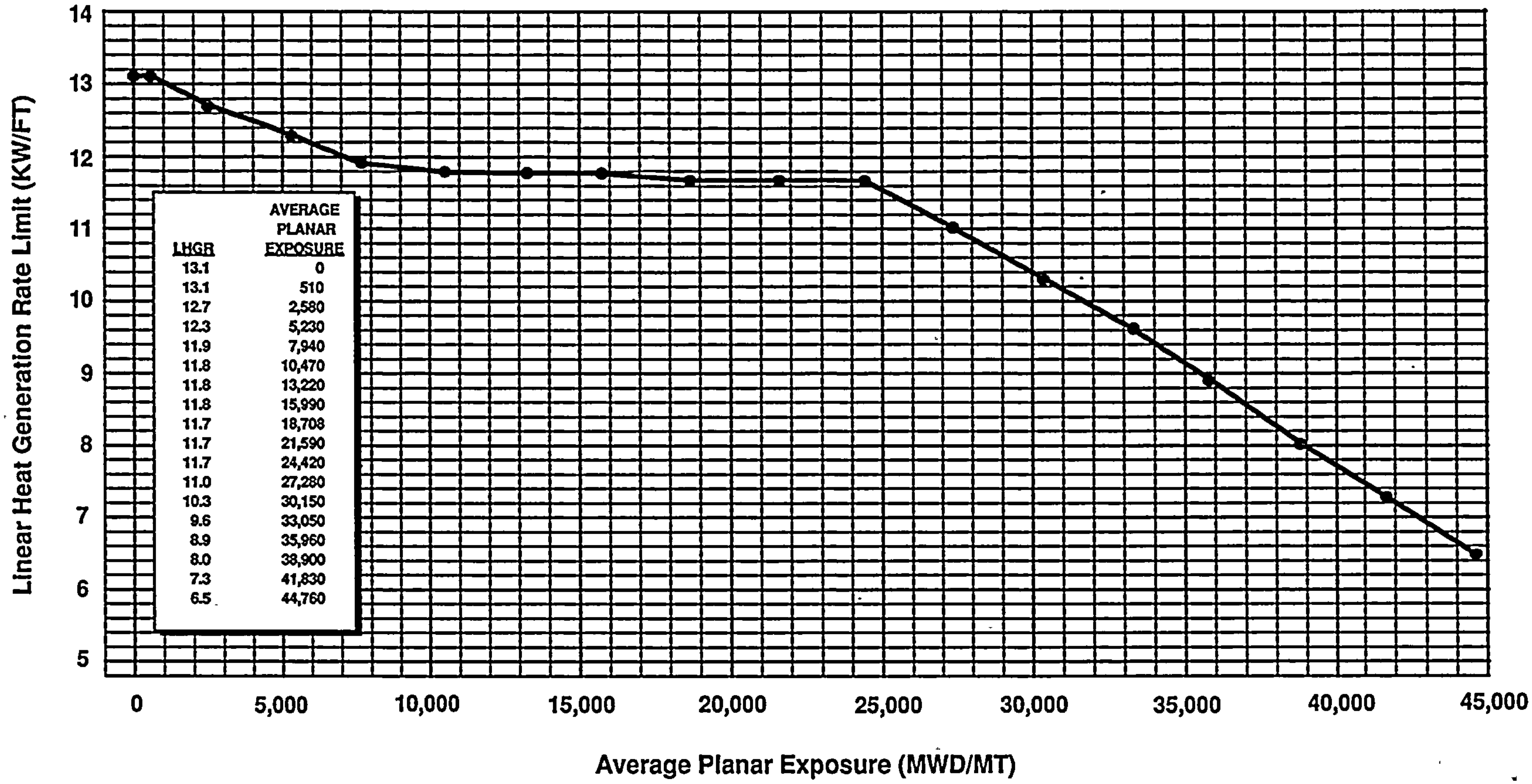
900541.12
(6/90)



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

Figure 13

900541.13
(6/90)



LHGR	AVERAGE PLANAR EXPOSURE
13.1	0
13.1	510
12.7	2,580
12.3	5,230
11.9	7,940
11.8	10,470
11.8	13,220
11.8	15,990
11.7	18,708
11.7	21,590
11.7	24,420
11.0	27,280
10.3	30,150
9.6	33,050
8.9	35,960
8.0	38,900
7.3	41,830
6.5	44,760

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

Figure 14

900541.14
(6/90)

5.0 REFERENCES

1.0 WPPSS-EANF-126, Rev. 1, "WNP-2 Cycle 6 Reload Summary Report", Washington Public Power Supply System, April 1990

2.0 ANF-90-01, "WNP-2 Cycle 6 Plant Transient Analysis Report", Advanced Nuclear Fuels Corporation, January 1990

3.0 ANF-90-02, "WNP-2 Cycle 6 Reload Analysis Report", Advanced Nuclear Fuels Corporation, January 1990

4.0 GE11, "Lead Fuel Assembly Report for Washington Public Power Supply System Nuclear Project No. 2 Reload 5 Cycle 6", General Electric Company, December 1989 (Proprietary)

5.0 UK 90-126, "Supplemental Lead Fuel Assembly Licensing Report - SVEA-96 LFA's for WNP-2", ABB Atom, January 1990 (Proprietary)

6.0 XN-NF-512(P)(A), "XN-3 Critical Power Correlation", Revision 1, Supplement 1, October 1982

7.0 XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", Revision 1, November 1983

8.0 XN-NF-79-71(P)(A), "Ex;xon Nuclear Plant Transient Methodology for Boiling Water Reactors", Revision 2, November 1981 and Revision 2, Supplements 1, 2 and 3, March 1986

9.0 ANF-913(P)(A), "A Computer Program for Boiling Water Reactor Transient Analysis", Volume 1, Supplmenets 1, 2 and 3, May 1988

10.0 XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors", Volume 1, May 1980, Volume 1, Supplements 1 and 2, March 1983, Volume 3, Revision 2, January 1987

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

12.0 ANF-89-014(P), "Generic Mechanical Design for ANF 9x9-IX and 9x9-9X BWR Reload Fuel", Revision 0, Supplement 1, June 1990

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

14.0 YUF:139:89, ANFWP-89-0106, YU Fresk, Advanced Nuclear Fuels, to Manager, Central Contracts, Supply System, "Justification for Cycle 5 Reduced Flow MCPR Curve", June 30, 1989