UPDATED

ATTACHMENT TO WNP-2 CYCLE 5 RELOAD SUMMARY REPORT

TECHNICAL SPECIFICATION CHANGES

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INDEX

LIST OF FIGURES

THERMAL POWER, Low Pressure or Low Flow (Continued)

at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 585 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors[a] which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the ANF nuclear critical heat fluxenthalpy XN-3 correlation. The XN-3 correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1. ~~and the nominal values of the core parameters listed in Bases Table B2.1.2-2.~~

The bases for the uncertainties in the core parameters are given in XN-NF-524(A), Rev. 1[a] and the basis for the uncertainty in the XN-3 correlation is given in XN-NF-512(A), Rev. 1[b]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a. Exxon Nuclear Critical Power Methodology for Boiling Water Reactors, XN-NF-524(A), Rev. 1.

b. Exxon Nuclear Company XN-3 Critical Power Correlation, XN-NF-512(A), Rev. 1.

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time of all operable control rods from the fully withdrawn position, for the four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	~~0.455~~ .430
39	~~0.920~~ .868
25	~~2.052~~ 1.936
5	~~3.708~~ 3.497

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

and 3.2.1-6

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 fuel and average bundle exposure for ANF fuel shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, ~~and~~ 3.2.1-3, when in two loop operation, and Figures 3.2.1-3 3.2.1-4, 3.2.1-5, and 3.2.1-6 when in single loop operation.

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

ACTION:

3.2.1-3,

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, ~~or~~ 3.2.1-3 or 3.2.1-6 in two loop operation or Figure 3.2.1-4, 3.2.1-5, or 3.2.1-6 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.

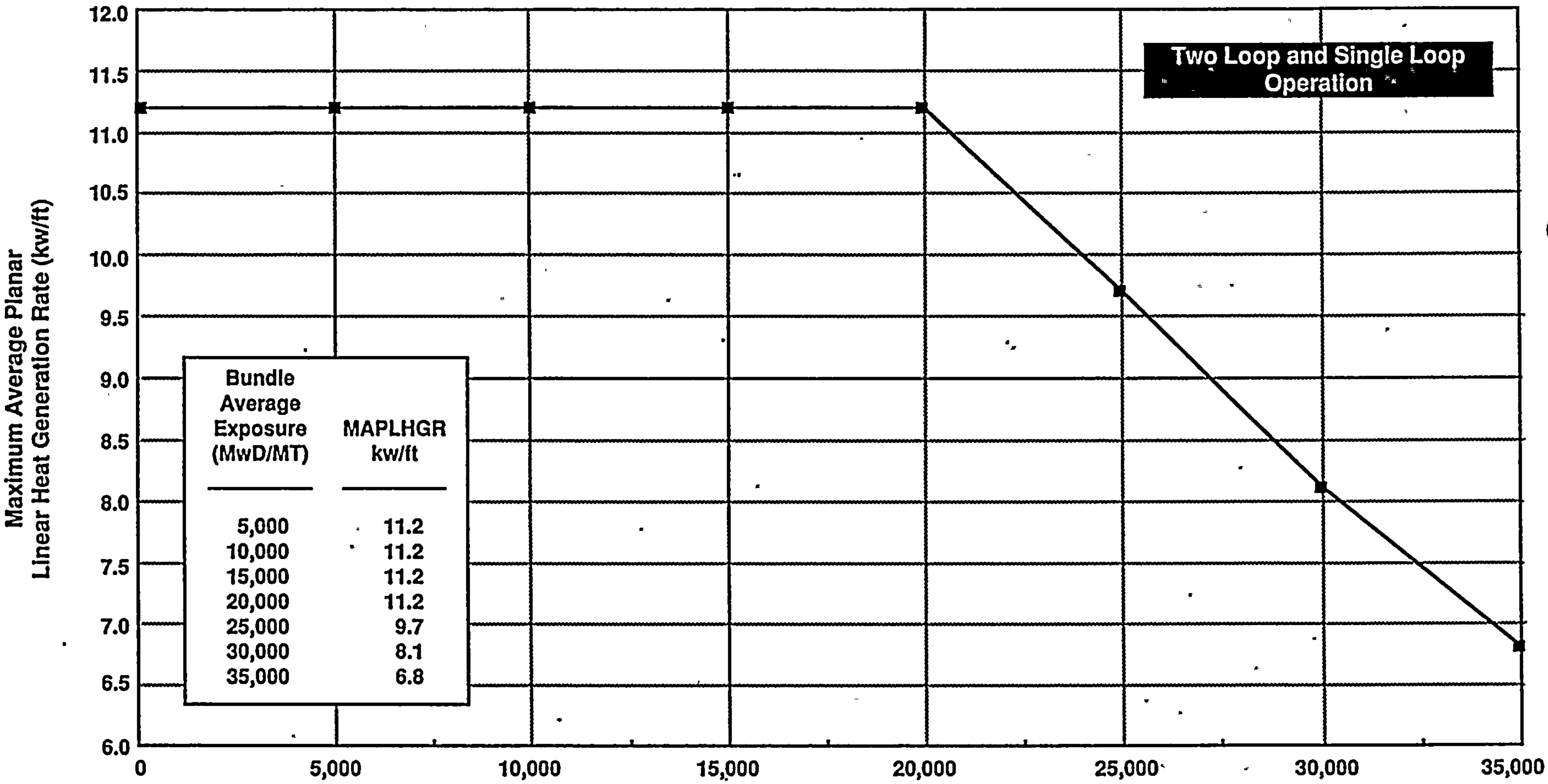
SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from 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.

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.



Bundle Average Exposure (MwD/MT)	MAPLHGR kw/ft
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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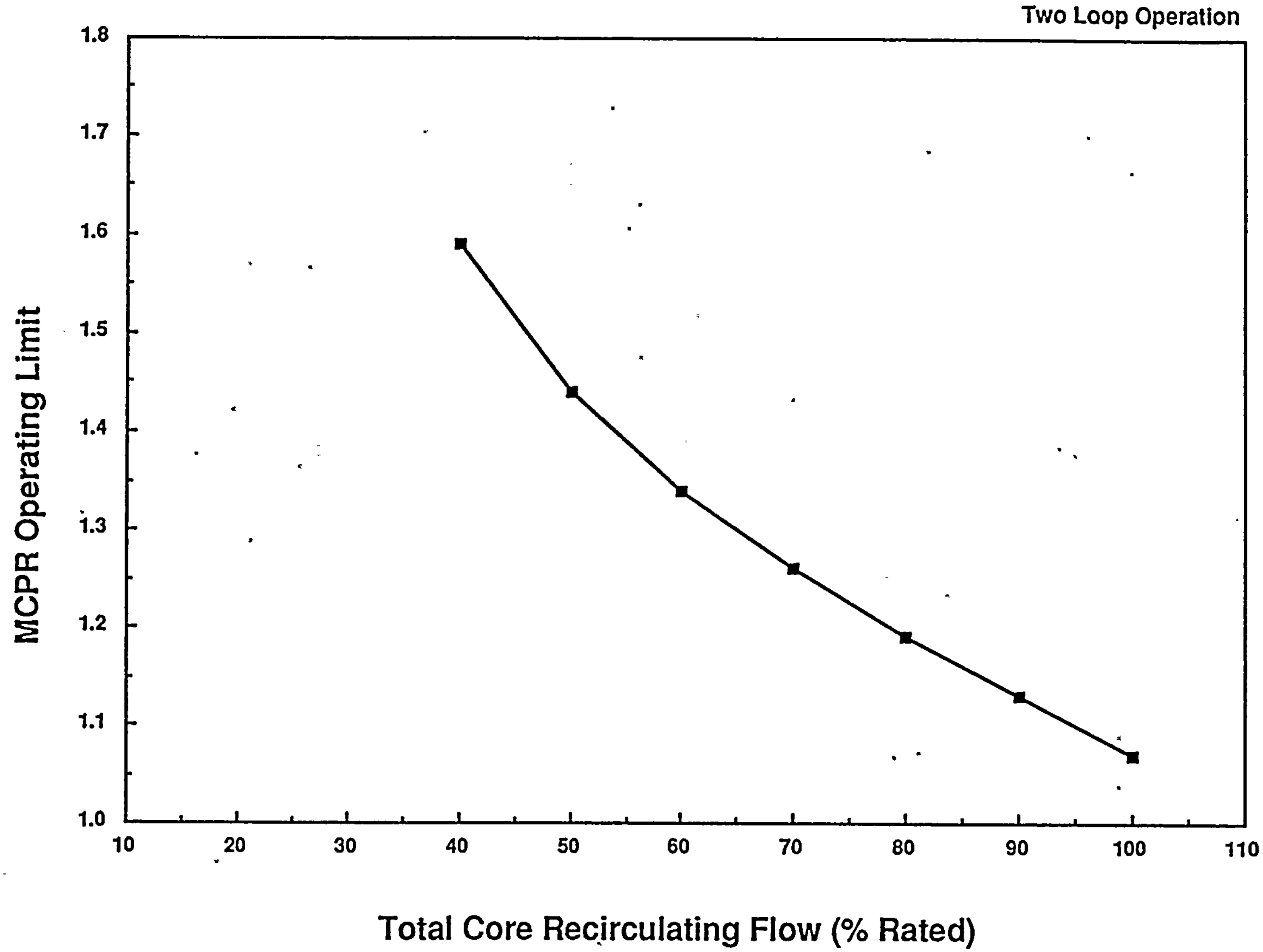
Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow: GE Fuel	MCPR Operating Limit Up to 106% Core Flow: ANF Fuel
1.	0 MWD/MTU - 3750 MWD/MTU	*	~~1.40~~ 1.24	~~1.28~~ 1.24
2.	3750 MWD/MTU - EOC MWD/MTU	Normal scram times**	~~1.40~~ 1.35	1.31
3.	3750 MWD/MTU - EOC MWD/MTU	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	~~1.50~~ 1.42	1.38
4.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times	~~1.50~~ 1.42	~~1.37~~ 1.38
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)	~~1.55~~ 1.48	~~1.43~~ 1.42
	0 MWD/MTU - EOC MWD/MTU	Single loop operation RPT operable Normal scram times**	1.40	1.37

*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



Reduced Flow MCPR Operating Limit
Figure 3.2.3-1

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3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kW/ft. The LHGR for ANF fuel shall not exceed the values shown in Figures 3.2.4-1, 3.2.4-2, and 3.2.4-3.

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.

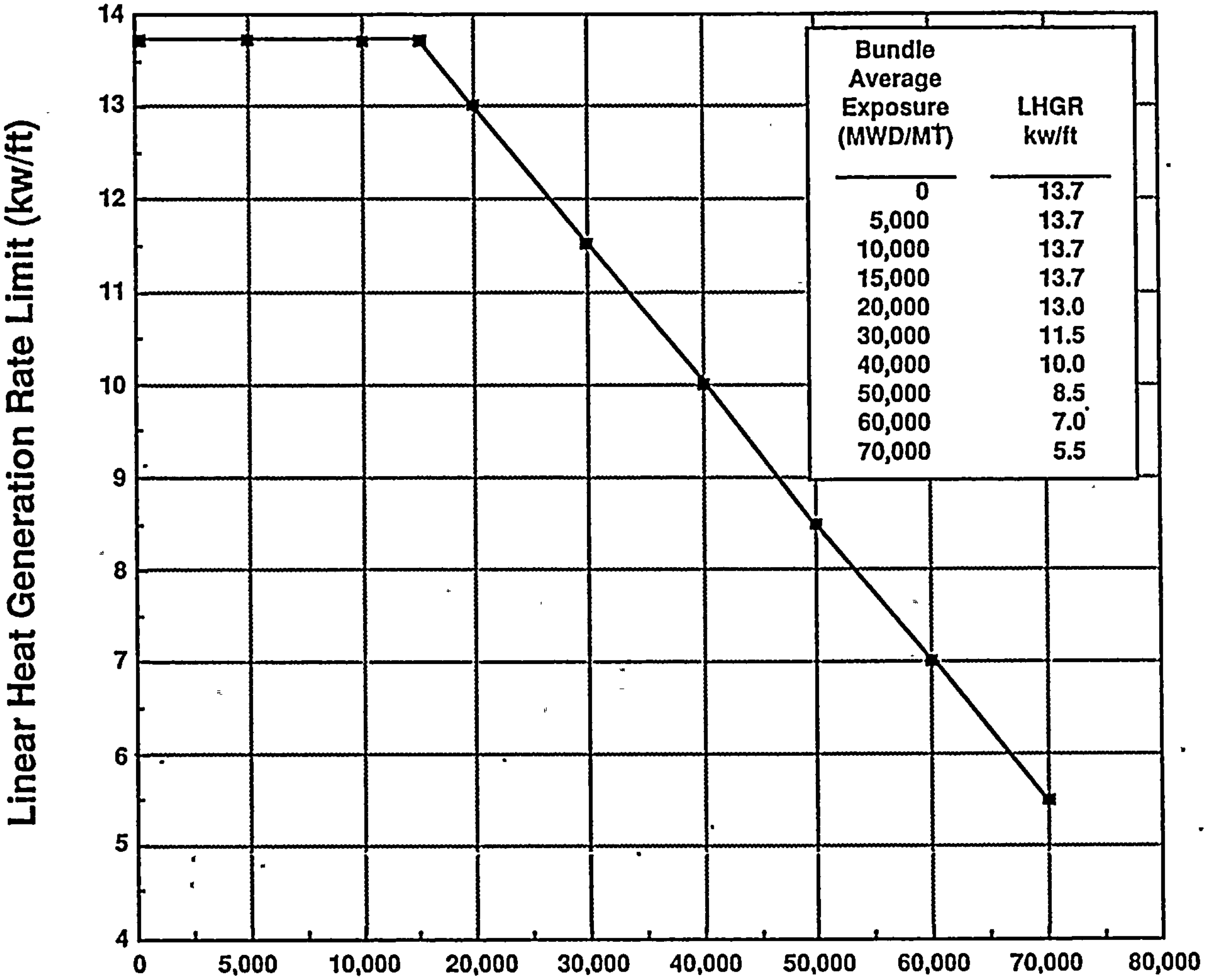
SURVEILLANCE REQUIREMENTS

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.



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

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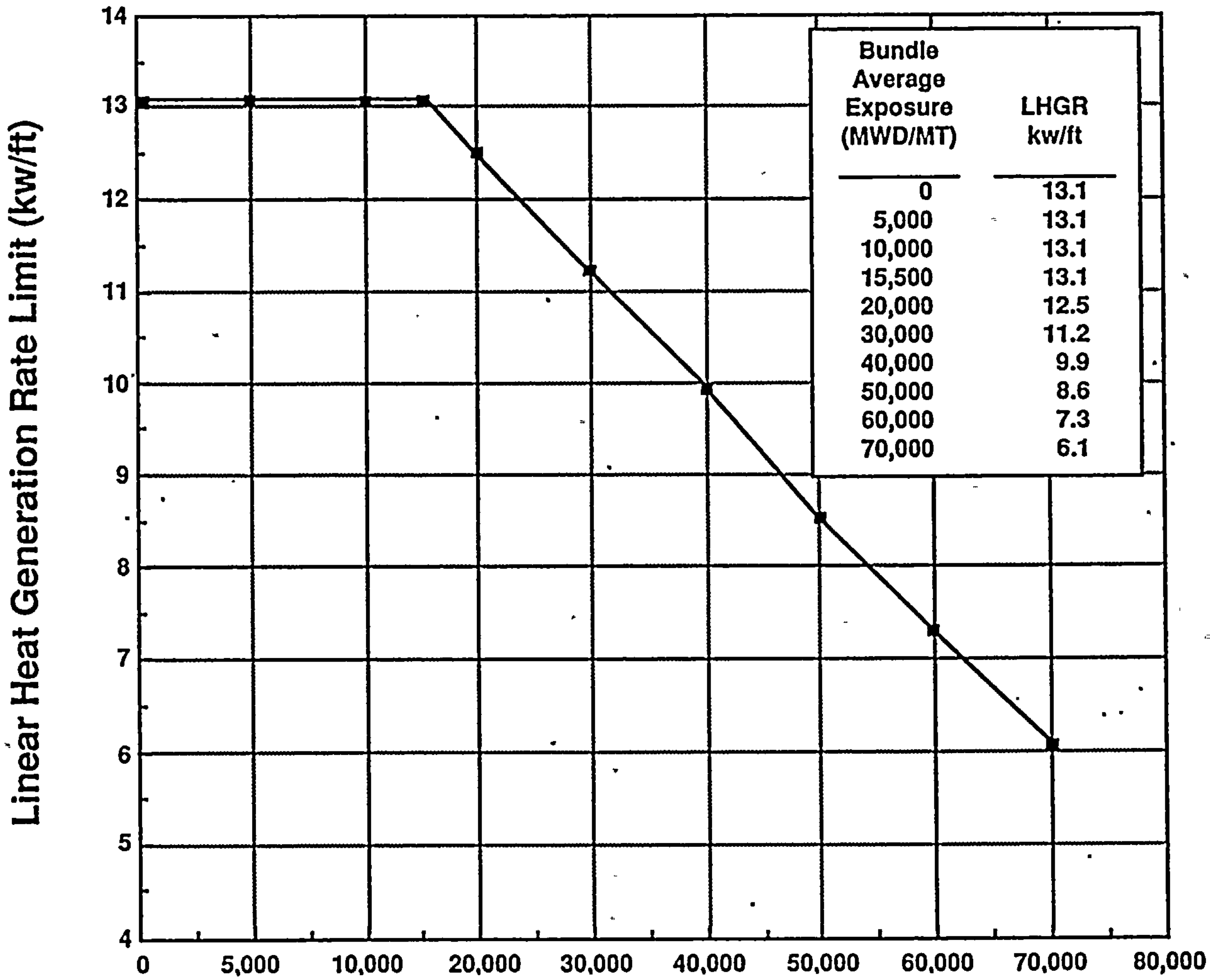

Bundle Average Exposure (MWD/MT)	LHGR kw/ft
0	13.1
5,000	13.1
10,000	13.1
15,500	13.1
20,000	12.5
30,000	11.2
40,000	9.9
50,000	8.6
60,000	7.3
70,000	6.1

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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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 appl~~ies~~ y to both single and two loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, ~~and~~ 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 in ~~Reference 1~~ NEDO-20566 P or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each INITIAL CORE fuel assembly containing 62 fuel rods and two water rods clad with Zircaloy-2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading, EXCEPT THAT THE RELOAD FUEL MAY EMPLOY A 9x9 ARRAY OF FUEL RODS.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable surveillance requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

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VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,539 cubic feet at a nominal steam dome saturation temperature of 545°F.

ATTACHMENT TO WNP-2 CYCLE 5 RELOAD SUMMARY REPORT

SUMMARY JUSTIFICATION FOR TECHNICAL SPECIFICATION CHANGES

Tech. Spec. No.	Justification
B2.1.2	Editorial change only to reflect change submitted and approved as Technical Specification Amendment No. 28.
3/4.1.3.4	This change corrects a previous oversight in the Tech. Specs. (see cover letter).
3/4.2.1	Addition of MAPLHGR curve for 9x9 fuel.
3/4.2.3	New MCPR values to reflect cycle specific transient analysis.
3/4.2.4	Addition of LHGR curves for 9x9-IX and 9x9-9X fuel.
B3/4.2.1	Editorial change to bases to reflect change to Tech. Spec. 3/4.2.1 discussed above.
5.3	Editorial change to Design Features to reflect the use of 9x9 lead test assembly fuel.