UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Iowa Electric Light and Power Company, et al, dated August 17, 1984, January 11 and March 15, 1985, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 120, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DB Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 28, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by removing the current pages and inserting the revised pages listed below. The revised areas are identified by vertical lines.

LIST OF AFFECTED PAGES

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1.0-2	3.2-2a*	3.6-7a	3.12-12
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1.1-2	3.2-18	3.6-34	3.12-17
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1.1-5	3.2-42	3.12-2	3.12-19
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1.1-20	3.3-18	3.12-6	
3.1-1	3.3-19	3.12-7	

*new page

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

5. OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

6. OPERATING

Operating means that a system or component is performing its intended functions in its required manner.

7. IMMEDIATE

Immediate means that the required action will be initiated as soon as practical considering the safe operation of the unit and the importance of the required action.

8. REACTOR POWER OPERATION

Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

a) SINGLE LOOP OPERATION (SLO): REACTOR POWER OPERATION with only one of the two recirculation loops in operation.

9. HOT STANDBY CONDITION

Hot standby condition means operation with coolant temperature greater than 212°F, reactor vessel pressure less than 1055 psig, and the mode switch in the Startup/Hot Standby position.

10. COLD CONDITION

Reactor coolant temperature equal to or less than 212°F.

11. HOT SHUTDOWN

The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

12. COLD SHUTDOWN

The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.

19. ALTERATION OF THE REACTOR CORE (CORE ALTERATION)

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.

20. REACTOR VESSEL PRESSURE

Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

21. THERMAL PARAMETERS

a. Minimum Critical Power Ratio (MCPR) - The value of critical power ratio (CPR) for that fuel bundle having the lowest CPR.

b. Critical Power Ratio (CPR) - The ratio of that fuel bundle power which would produce boiling transition to the actual fuel bundle power.

c. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

d. Limiting Control Rod Pattern - A limiting control rod pattern for rod withdrawal error (RWE) exists when a) core thermal power is greater than or equal to 30% of rated and less than 90% of rated ($30\% \leq P < 90\%$) and the MCPR is less than 1.70, or b) core thermal power is greater than or equal to 90% of rated ($P \geq 90\%$) and the MCPR is less than 1.40.

e. Linear Heat Generation Rate - The heat output per unit length of fuel pin.

f. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 1658 MWth.

g. Total Peaking Factor (TPF) - The ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

h. Maximum Total Peaking Factor (MTPF) - The largest TPF which exists in the core for a given class of fuel for a given operating condition.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior.

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation [1.10 for SINGLE LOOP OPERATION (SLO)] shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit

(Reactor Pressure $\leq$ 785 psig

or Core Flow $\leq$ 10% of Rated

When the reactor pressure is $\leq$ 785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specifications:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trips

1. APRM High Flux Scram When In Run Mode.

The APRM scram trip setpoint shall be as shown on Figure 2.1-1 and shall be:

$S \leq (0.58W + 62)$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater for two recirculation loop operation and

$S \leq (0.58W + 58.5)$

for SLO.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

C. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.

D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

Where: S = Setting in percent of rated power (1,658 MWt)

W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^6 lb/hr core flow.

NOTE: This setting assumes operation within the basic thermal design criteria. These criteria are LHGR $\leq$ values given in Section 3.12.B and MCPR $\geq$ values as given in Section 3.12.C. If it is determined that either of these design criteria is being violated during operation, IMMEDIATE action must be taken to return to operation within these criteria.

2. APRM High Flux Scram

When in the REFUEL or STARTUP and HOT STANDBY MODE, the APRM scram shall be set at less than or equal to 15 percent of rated power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	3. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.
	B. Scram and Isolation on reactor low water level — $\geq$ 514.5 inches above vessel zero (+170" indicated level)
	C. Scram - turbine stop valve closure — $\leq$ 10 percent valve closure
	D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure > 785 psig and Core Flow > 10% of Rated

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 procedure 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 critical power ratio 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 generically determined in Reference 1, for two recirculation loop operation. This Safety Limit MCPR is increased by 0.03 for SLO.

For analyses of the thermal consequences of the transients the MCPRs stated in Section 3.12 as a limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

As discussed in Reference 2, the core-wide transient analyses for SLO is conservatively bounded by two-loop operation analyses and the flow-dependent rod block and scram setpoint equations are adjusted for SLO.

Steady-state operation without forced recirculation will not be permitted, except during special testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients have been analyzed to a power level of 102% of 1658 MWt.

ii. The licensed maximum power level is 1658 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

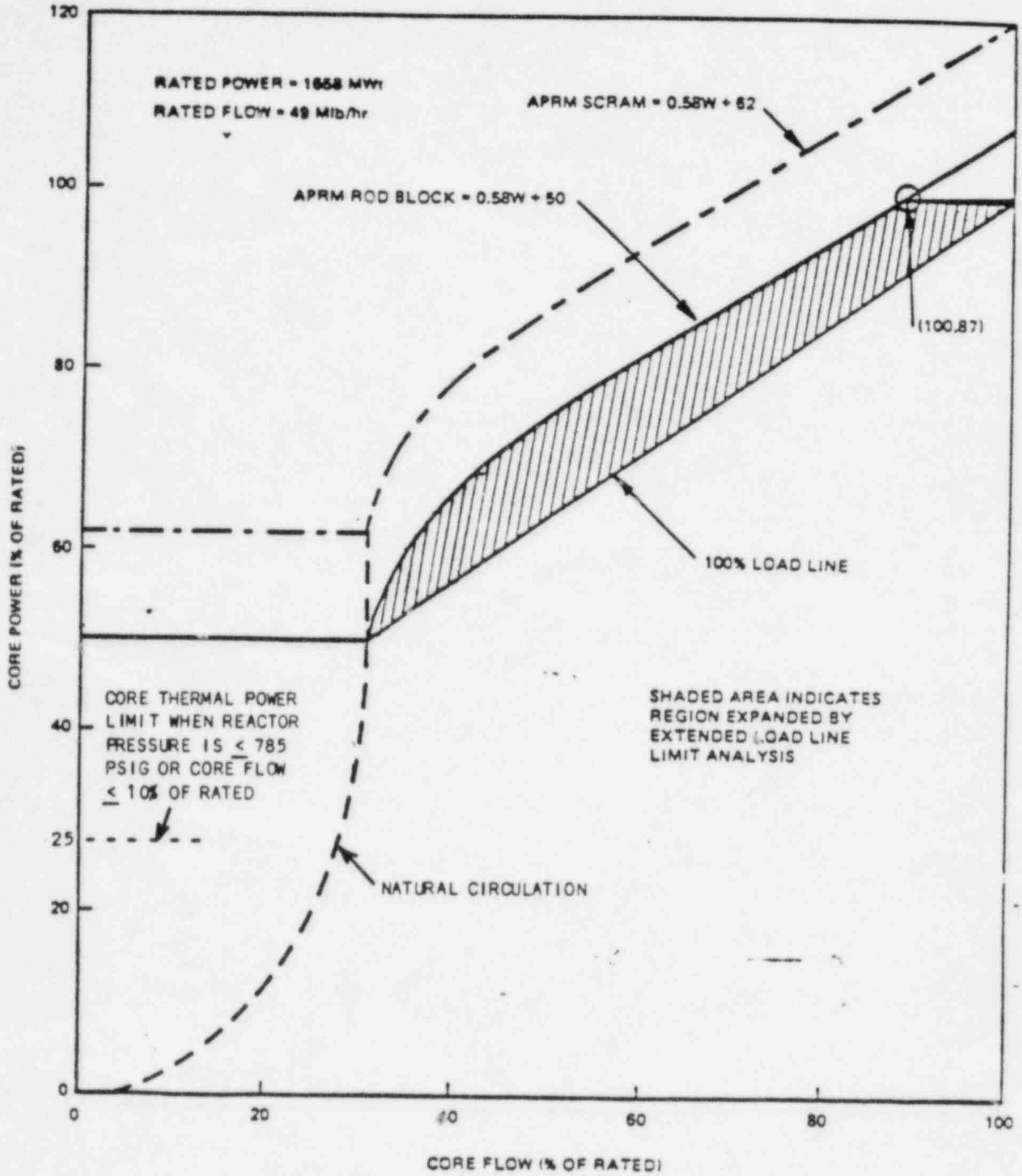
scram trip provides additional margin. An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that the heat flux is in equilibrium with the neutron flux, and the IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents has been analyzed. This analysis included starting the accident at various power levels. Ths most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that



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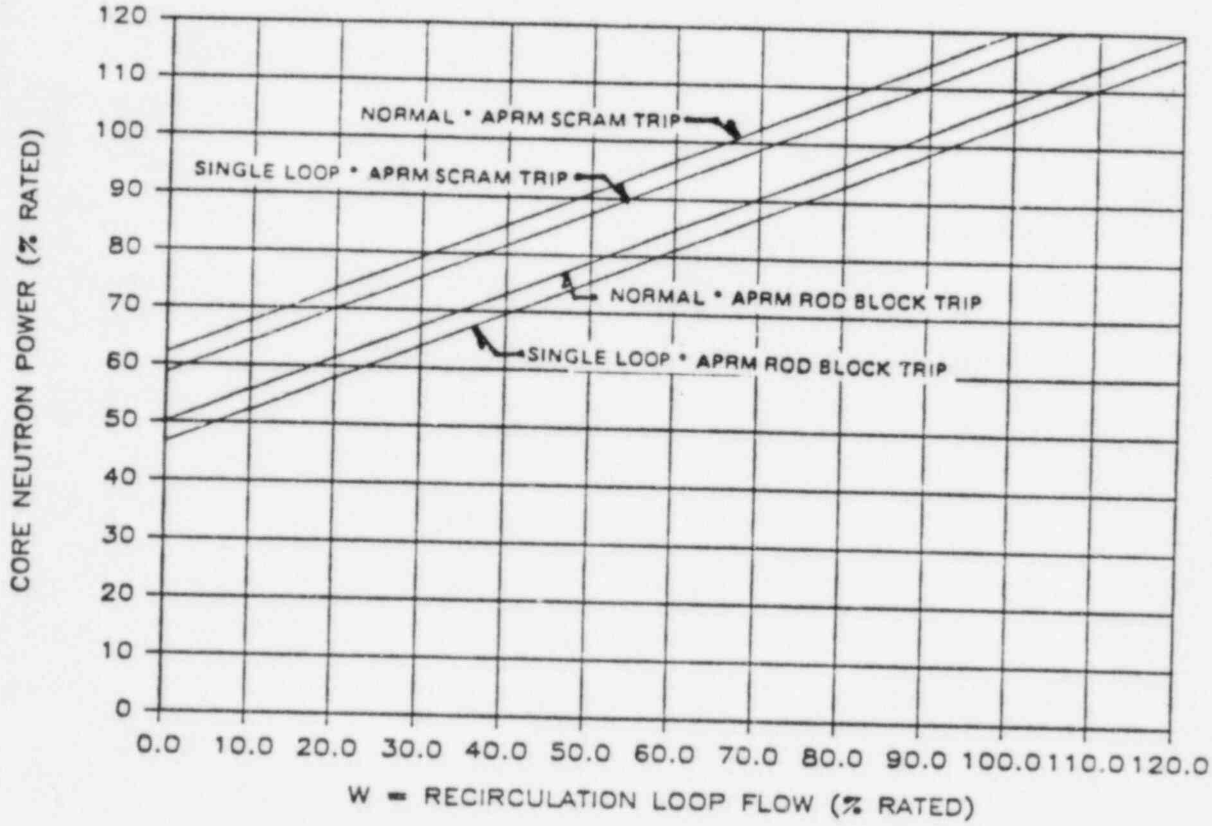
APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS FIGURE 1.1-1



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY TECHNICAL SPECIFICATIONS
Core Power Vs Recirc Loop Flow FIGURE 2.1-1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1-1. The designed system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.

As a minimum, the reactor protection system instrumentation channels of Table 3.1-1 shall be operable with response times as shown in Table 3.1-2.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

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

Specification:

A.1 Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

.2 Response time measurements (from actuation of sensor contacts or trip point to de-energization of scram solenoid relay) are not part of the normal instrument calibration. The reactor trip system response time of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

.3 When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
2	IRM High Flux	≤ 120/125 of Full Scale	X	X	(5)	6 Instrument Channels	A
2	IRM Inoperative		X	X	(5)	6 Instrument Channels	A
2	APRM High Flux	For two recirc loop operation: (.58W+62) (11) (12) For SLO: (.58W+58.5) (11) (12)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤ 1055 psig	X(8)	X	X	4 Instrument Channels	A

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Control Rod Block Actuation	C. Control Rod Block Actuation
1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks The Limiting Conditions of Operation for the instrumentation that initiates these control rod block are given in Table 3.2-C.	1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-C. System logic shall be functionally tested as indicated in Table 4.2-C.
2. Rod Block Monitor (RBM) (a) Both RBM channels shall be demonstrated to be Operable prior to control rod withdrawal when a Limiting Control Rod Pattern exists; otherwise, control rod withdrawal may take place with the RBM bypassed. A Limiting Control Rod Pattern exists when: i) core thermal power is greater than or equal to 30% of rated and less than 90% of rated ($30\% \leq P < 90\%$) and the Minimum Critical Power Ratio (MCPR) is less than 1.70, or ii) core thermal power is greater than or equal to 90% of rated ($P \geq 90\%$) and the MCPR is less than 1.40. When a Limiting Control Rod Pattern exists: With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, unless Operability is restored within this time period. With both RBM channels inoperable, control rod withdrawal shall be blocked until Operability of at least one channel is restored.	2. When a Limiting Control Rod Pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(b) The RBM control rod block setpoints are given in Table 3.2-C. The upscale High Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 85% of rated ($P \geq 85\%$). The upscale Intermediate Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 65% of rated and less than 85% of rated ($65\% \leq P < 85\%$). The upscale Low Power Trip Setpoint shall be applied when the core thermal power is greater than or equal to 30% of rated and less than 65% of rated ($30\% \leq P < 65\%$). The RBM can be bypassed when core thermal power is less than 30% of rated. The RBM bypass time delay (t_{d2}) shall be less than or equal to 2.0 seconds.	
D. <u>Radiation Monitoring Systems - Isolation & Initiation Functions</u>	D. <u>Radiation Monitoring Systems - Isolation & Initiation Functions</u>
1. <u>Steam Air Ejector Offgas System</u>	1. <u>Steam Air Ejector Offgas System</u>
(a) Except as specified in (b) below, both post treatment steam air ejector offgas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set at a value not to exceed the equivalent of the stack release limit specified in the Environmental Technical Specifications. The Steam Air Ejector isolation valves close immediately if the Steam Air Ejector Offgas Radiation Monitor output exceeds the trip setting.	Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2-D.
(b) From and after the date that one of the two steam air ejector radiation monitors is made or found to	

TABLE 3.2-C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	For two recirc loop operation: $\leq$ (0.58 W + 50) (2) For SLO: $\leq$ (0.58W + 46.5) (2)	6 inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor a) Upscale (Power Referenced)		2 Inst. Channels	(1)
	1) Low Power Trip Setpoint	$\leq$ 115/125 of full scale		
	2) Intermediate Power Trip Setpoint	$\leq$ 109/125 of full scale		
	3) High Power Trip Setpoint	$\leq$ 105/125 of full scale		
	b) Downscale	$\geq$ 94/125 of full scale		
	c) RBM Bypass Time Delay (t_{d2}) (10)	$\leq$ 2.0 seconds		
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5)(6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1	Scram Discharge Volume Water Level-High	$\leq$ 24 gallons	1 Inst. Channel	(9)

NOTES FOR TABLE 3.2-C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM [except for APRM Upscale (Not in Run Mode)] and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (1658 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is > 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function is bypassed when in the IRM range switches are on a range 8 or above.

7. There are three upscale trip levels. Only one is applied over a specified range of core thermal power. All RBM trips are bypassed below 30% of rated power. An RBM channel will be considered to be inoperable if less than one-half of the required number of LPRM inputs are available.

8. This function is bypassed when the mode switch is placed in Run.

9. If the number of operable channels is less than required by the minimum number of operable instrument channels per trip system requirement, place the inoperable channel in the tripped condition within one hour.

10. RBM bypass time delay (t_{d2}) is set low enough to assure minimum rod movement while upscale trips are bypassed.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MCPR does not decrease below the Safety Limit. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, six IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criterion is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than safety limit.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a Limiting Control Rod Pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 5 indicated on scale for APRM's and 5/125 full scale for IRM's.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when in SLO.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
e. If Specification 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 30% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified. 2) The RWM computer on line diagnostic test shall be successfully performed.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified. 4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point. c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with Limiting Control Rod Patterns, either: a. Both RBM channels shall be operable, or b. With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, unless Operability is restored within this time period, or c. With both RBM channels inoperable, control rod withdrawal shall be blocked until Operability of at least one channel is restored.	5. When a Limiting Control Rod Pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
levels are less than or equal to three times their established baseline levels, continue to determine the noise levels at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated core thermal power while operating in this region of the power/flow map, or	
b) if the APRM and/or LPRM* neutron flux noise levels are greater than three times their established baseline levels, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow, and/or by initiating an orderly reduction of core thermal power by inserting control rods.	
See Specifications 3.6.F.2 for SLO.	
A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.	
F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN condition within 24 hours.	

*Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.

drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 30% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

d. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

e. The RBM provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a Limiting Control Rod Pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR of 1.20 prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. When a Limiting Control Rod Pattern exists, with one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked (Reference 2). Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

A Limiting Control Rod Pattern for rod withdrawal error (RWE) exists when (a) core thermal power is greater than or equal to 30% of rated and less than 90% of rated ($30\% \leq P < 90\%$) and the MCPR is less than 1.70, or (b) core thermal power is greater than or equal to 90% of rated ($P \geq 90\%$) and the MCPR is less than 1.40.

During the use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal dces not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Plant Superintendent, Nuclear.

3. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit.

After initial fuel loading and subsequent refuelings when operating above 950 psig, all control rods shall be scram tested within the constraints imposed by the Technical Specifications and before the 40% power level is reached. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

4. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poision in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1%ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3.3 and 4.3 REFERENCES

1) NEDO 24087-3, 78NED265, Class 1, June 1978 "General Electric Boiling Water Reactor Reload 3 (Cycle 4) Licensing Amendment for Duane Arnold Energy Center, Supplement 3: Application of Measured Scram Times".

2) "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program fcr the Duane Arnold Energy Center," NEDC-30813-P, December, 1984.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	b. The indicated value of core flow rate varies from the value derived form loop flow measurements by more than 10%. c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%. 2. Whenever there is recirculation flow from the reactor in the Startup or Run mode, and one recirculation pump is operating, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.
F. Jet Pump Flow Mismatch 1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power. 2. If Specification 3.6.F.1 cannot be met, one recirculation pump shall be tripped. The reactor may be started and operated, or operation may continue in SLO provided that:	F. Jet Pump Flow Mismatch 1. Recirculation pump speeds shall be checked and logged at least once per day. 2. a. Prior to SLO and core thermal power greater than the limit specified in Figure 3.3-1, establish baseline APRM and LPRM* neutron flux noise levels, provided that baseline values have not *Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
a. MAPLHGR multipliers as indicated in section 3.12.A are applied. b. During SLO and core thermal power greater than the limit specified in Figure 3.3-1, core flow must be greater than or equal to 39% of rated, and (i) the Surveillance Requirements of 4.6.F.2.a have not been satisfied, immediately initiate action to reduce core thermal power to less than or equal to the limit specified in Figure 3.3-1 within 4 hours, or (ii) the Surveillance Requirements of 4.6.F.2.a have been satified, continue to determine the APRM and LPRM neutron flux levels at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated core thermal power while operating in this region of the power/flow map. If the APRM and/or LPRM* neutron flux noise levels are greater than three times their established baseline values, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by *Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.	been previously established since the last core refueling. Baseline values shall be established during SLO and core thermal power less than or equal to the limit specified in Figure 3.3-1. b. Prior to SLO and core flow greater than 45% of rated, establish baseline core plate ΔP noise levels with core flow less than or equal to 45% of rated, provided that baseline values have not been previously established during SLO since the last core refueling.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
increasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods. c. During SLO and core flow greater than 45% of rated, and (i) the Surveillance Requirements of 4.6.F.2.b have not been satisfied, immediately initiate action to reduce core flow to less than or equal to 45% of rated within 4 hours, or (ii) the Surveillance Requirements of 4.6.F.2.b have been satisfied, continue to determine core plate ΔP noise at least once per 8 hours and also within 30 minutes after the completion of a core thermal power increase of at least 5% of rated thermal power. If the core plate ΔP noise level is greater than 1.0 psi and 2 times its established baseline value, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by decreasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods. d. The idle loop is isolated electrically by disconnecting the breaker to the recirculation pump motor generator (M/G) set drive motor prior to startup, or if disabled during reactor operation, within 24 hours. Refer to Specification 3.6.A for startup of the idle recirculation loop. e. The recirculation system controls will be placed in the manual flow control mode.	

c. The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.E.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the sixteen individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow in the case of a failed jet pump. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing jet pump.* Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true.

*Note: In the case of SLO, when the recirculation pump is tripped, the flow through the inactive jet pumps is subtracted from the total jet pump flow, yielding the correct value for the total core flow.

80% power cases, respectively. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

An evaluation has been provided for ECCS performance during SLO (Sec. 3.12, Ref. 11). Therefore, continuous operation under such conditions is appropriate. The reactor may also be operated in SLO up to 24 hours without electrically isolating the idle loop. This short period of time permits corrective action to be taken to re-activate the idle loop or to implement the changes for continuous SLO.

During periods of SLO, the idle recirculation loop is isolated by electrically disarming the recirculation pump. This is done to prevent a cold water injection transient caused by an inadvertent pump start-up. It is permissible to leave the suction and discharge valves open during SLO to allow flow through the loop in order to maintain the temperature. However, if for some reason the discharge valve is inoperable it should be closed and electrically disarmed. This is done to prevent degradation of LPCI flow during a LOCA. With the discharge valve disarmed, the temperature in the loop can be maintained by opening the bypass valve, as the loop selection logic will close the bypass valve, isolating the loop, prior to opening the LPCI injection valve.

APRM and/or LPRM oscillations in excess of those specified in Section 3.6.F.2 could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. By restricting core flow to greater than or equal to 39% of rated, which corresponds to the core flow at the 80% rodline with 2 recirculation pumps running at minimum speed, the region of the power/flow map where these oscillations are most likely to occur is avoided. Individual APRM and/or LPRM channels exhibiting excessive flux noise may be discounted upon verification that a true condition of thermal hydraulic instability does not exist by observation of the remaining available APRM and/or LPRM channels. These specifications are based upon the guidance of GE SIL #380, Rev. 1, 2/10/84.

Above 45% of rated core flow in SLO there is the potential to set up high flow-induced noise in the core. Thus, surveillance of core plate ΔP noise is required in this region of the power/flow map to alert the operators to take appropriate remedial action if such a condition exists.

LIMITING CONDITIONS FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

1. During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-6, -7, -8 and -9 multiplied by the smaller of the two MAPFAC factors determined from Figs. 3.12-10 and 3.12-11.

   For the central Lead Test Assembly (LTA 311), the actual MAPLHGR, as a function of average planar exposure, shall not exceed the limiting value shown in Fig. 3.12-5 multiplied by the smaller of the two MAPFAC factors from Figs. 3.12-11 and 3.12-12.

2. During SLO, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-5, -6, -7, -8 and -9 multiplied by the smaller of the two MAPFAC factors determined from Figs. 3.12-11 and 3.12-13.

SURVEILLANCE REQUIREMENTS

4.12 CORE THERMAL LIMITS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3. If at any time during reactor power operation (one or two loop) at $\geq$25% rated thermal power, it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 2 hours, reduce reactor power to $\leq$ 25% of rated thermal power, or to such a power level that the limits are again being met, within the next 4 hours.	
4. If the reactor is being operated in SLO and cannot be returned to within prescribed limits within this 4 hour period, the reactor shall be brought to the COLD SHUTDOWN condition within 36 hours.	
5. For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed action is met.	
B. Linear Heat Generation Rate (LHGR)	B. Linear Heat Generation Rate (LHGR)
1. During reactor power operation the linear heat generation rate (LHGR) of any rod in any BP/P8X8R or ELTA fuel assembly shall not exceed 13.4 KW/ft, while the LHGR of any rod in an LTA 311 fuel assembly shall not exceed 14.4 KW/ft.	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
2. If at any time during reactor power operation at $\geq$25% rated thermal power it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours, reduce reactor power to $\leq$ 25% of Rated Thermal Power, or to such a power level that the limits are again being met, within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
1. During reactor power operation, the MCPR shall be equal to or greater than the Operating Limit MCPR, which is a function of core thermal power, core flow, fuel type and scram time (τ). For core thermal power greater than or equal to 25% of rated and less than 30% of rated (25% $\leq$ P < 30%), the Operating Limit MCPR is given by Fig. 3.12-4. For core thermal power greater than or equal to 30% of rated (P $\geq$ 30%), the Operating Limit MCPR is the greater of either: a) The applicable flow-dependent MCPR ($MCPR_F$) determined from Figure 3.12-1, or b) The appropriate rated power MCPR from Figures 3.12-2 and 3.12-3 [MCPR(100)] multiplied by the applicable power-dependent MCPR multiplier	MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a Limiting Control Rod Pattern as defined in Section 3.2.C.2(a). During operation with a Limiting Control Rod Pattern, the MCPR shall be determined at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

(K_p) determined from Figure 3.12-4.

2. During SLO with core thermal power greater than or equal to 25% of rated, the Operating Limit MCPR is increased by adding 0.03 to the above determined Operating Limit MCPR.

3. If at any time during reactor power operation (one or two recirc. loop) at $\geq$25% rated thermal power, it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within two hours, reduce reactor power to $\leq$ 25% of rated thermal power, or to such a power level that the limits are again being met, within the next 4 hours.

4. If the reactor is being operated in SLO, and cannot be returned to within prescribed limits within this 4 hour period, the reactor shall be brought to a COLD SHUTDOWN condition within 36 hours.

5. For either the one or two recirc. loop operating condition surveillance and corresponding action shall continue until the prescribed action is met.

3.12 BASES: CORE THERMAL LIMITS

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident (LOCA) will not exceed the limit specified in 10CFR50.46. LOCA analyses are performed using General Electric calculational models which conform to the requirements of 10 CFR Part 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50.46 limit.

The flow-dependent correction factor (Figure 3.12-10) applied to the MAPLHGR limits at rated conditions assures that (1) the 10CFR50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal-mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions (Reference 14).

The power-dependent correction factor (Figure 3.12-11) applied to the MAPLHGR limits at rated conditions assures that the fuel thermal-mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions (Reference 14).

For two recirculation loop operation, the calculational procedures used to establish the MAPLHGR's shown on Figures 3.12-5 thru 3.12-9 are documented in Reference 7. The reduction factors for SLO were derived in Reference 13.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat eneration rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 3 and in References 4 and 5, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the Maximum Total Peaking Factor (MTPF) would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients (2). 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 settings given in Specification 2.1.

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, which determines the required steady state MCPR limit, is the transient which yields the largest ΔCPR. The minimum Operating Limit MCPR of Specification 3.12.C bounds the sum of the Safety Limit MCPR and the largest ΔCPR.

The required MCPRs at rated power [MCPR(100)] are determined by the methods described in References 11 and 12. These limits were derived by using the GE 67B scram times, given in Section 3.3.C, which are based upon extensive operating plant data, as well as GE test data. The ODYN Option B scram insertion times were statistically derived from the 67B data to ensure that the resulting Operating Limit from the transient analysis would, with 95% probability at the 95% confidence level, result in the Safety Limit MCPR not being exceeded. The scram time parameter (τ), as calculated by the following formula, is a measure of the conformance of the actual plant control rod drive performance to that used in the ODYN Option-B licensing basis:

$$\tau = \frac{\tau_{ave} - \tau_b}{\tau_a - \tau_b}$$

where: τave = average scram insertion time to Notch 38, as measured by surveillance testing

τ_b = scram insertion time to Notch 38 used in the ODYN Option-B Licensing Basis.

τ_a = 67B scram insertion time to Notch 38

As the average scram time measured by surveillance testing (τave), exceeds the ODYN Option B scram time (τ_b), the MCPRs at rated power [MCPR(100)] must be adjusted using Figures 3.12-2 and -3.

2. MCPR Limits for Other Than Rated Power and/or Rated Flow Conditions

At less than 100% of rated power and/or flow the required Operating Limit MCPR is the larger value of the flow-dependent MCPR ($MCPR_F$) or the power-dependent multiplier (K_p) times the rated power MCPR [MCPR(100)] at the existing core power/flow state. The required Operating Limit MCPR is a function of flow in order to protect the fuel from inadvertent core flow increases such that the Safety Limit MCPR requirement can be assured.

The $MCPR_F$s were calculated such that, for the maximum core flow rate and core thermal power along a conservative load line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit MCPR. Using this relative bundle power, the MCPRs were calculated at different points along this conservative load line corresponding to different core flows. The resulting $MCPR_F$s are given in Figure 3.12-1.

For operation above 30% of rated thermal power, the core power-dependent MCPR operating limit is the rated power MCPR [MCPR(100)], multiplied by the factor given in Figure 3.12-4, i.e., K_p. For operation below 30% of rated thermal power, where the direct scrams on turbine control valve fast closure and turbine stop valve closures are bypassed, absolute MCPR limits are established. This limit is taken directly from Figure 3.12-4. This limit protects the fuel from abnormal operating transients, including localized events, such as a rod withdrawal error, other than those resulting from inadvertent core flow increases, which are covered by the flow-dependent MCPR limits. This power-dependent MCPR limit was developed based upon bounding analyses for the most limiting transient at the given core power level. Further information on the MCPR operating limits for off-rated conditions is presented in Reference 14.

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of the requirements by considerable margin. Therefore, monitoring of MCPRs below this power level is unnecessary. The daily monitoring of MCPRs above 25% of rated thermal power is sufficient, since power distribution shifts are very slow, provided that no significant changes in core flow or control rod pattern have taken place.

During SLO, the Operating Limit MCPR must be increased by 0.03 to account for the increased uncertainty in the core flow and Transversing In-core Probe (TIP) readings used in the statistical analyses to derive the Safety Limit MCPR (see Reference 13).

4.12 BASES: CORE THERMAL LIMITS

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative state relative to MCPR. During initial start up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached assures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-03, June 1984.

2. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A**.

3. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-19735, August 1973.

4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

5. Communication: V.A. Moore to I.S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

6. R.B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

7. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, NEDE-20566, August 1974.

8. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977.

9. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

10. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 5: Revised Operating Limits for Loss of Feedwater Heating, NEDO-24987-5.

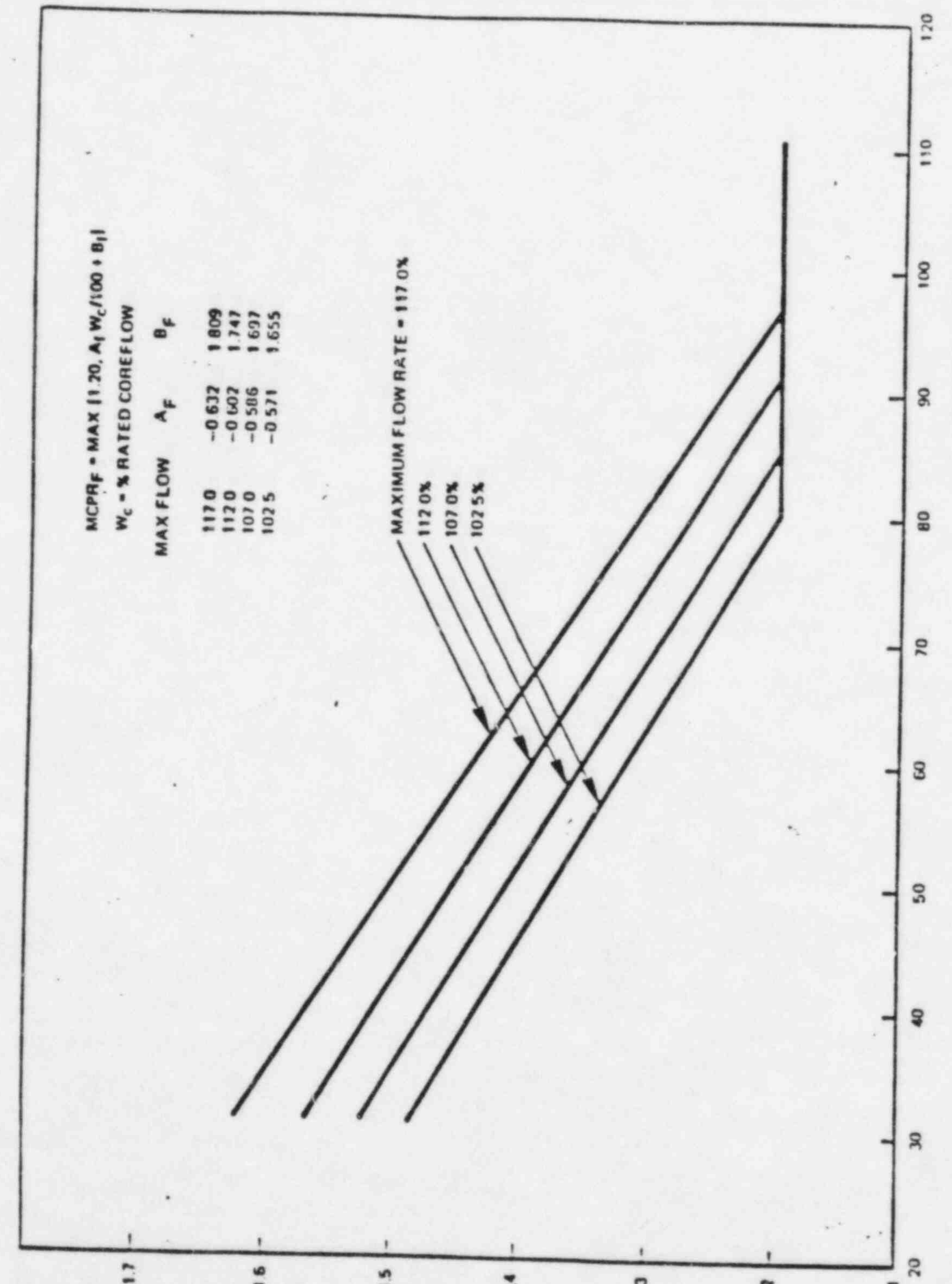
11. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

12. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits," January 19, 1981.

13. Duane Arnold Energy Center Single Loop Operation, NEDO-24272, July 1980.

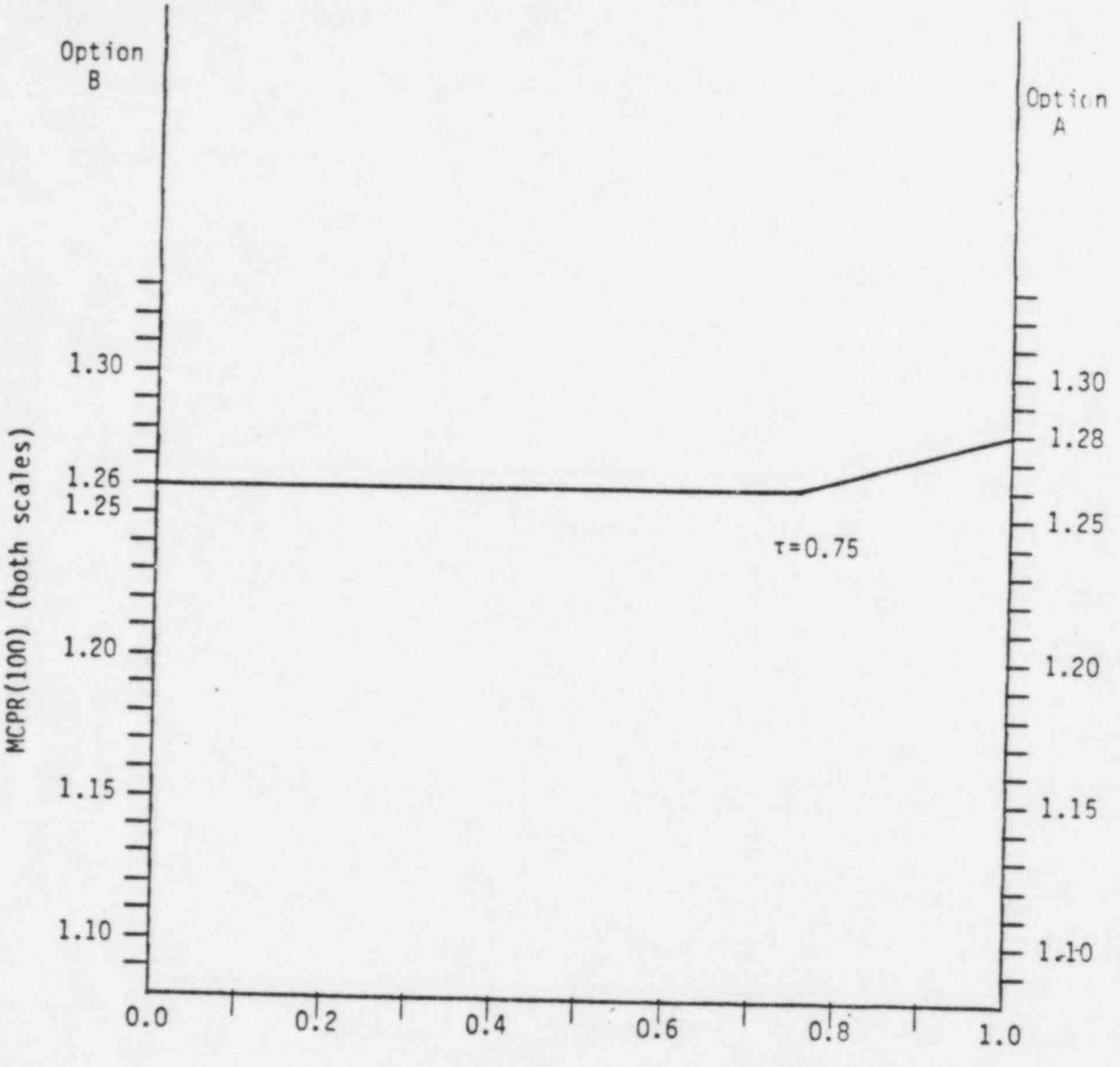
14. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center," NEDC-30813-P, December, 1984.

**Approved revision number at time reload fuel analyses are performed.



DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT AND POWER COMPANY
TECHNICAL SPECIFICATIONS

FLOW-DEPENDENT MINIMUM CRITICAL POWER RATIO (MCPR$_F$)
FIGURE 3.12-1



(based on tested measured scram time as defined in Reference 11)

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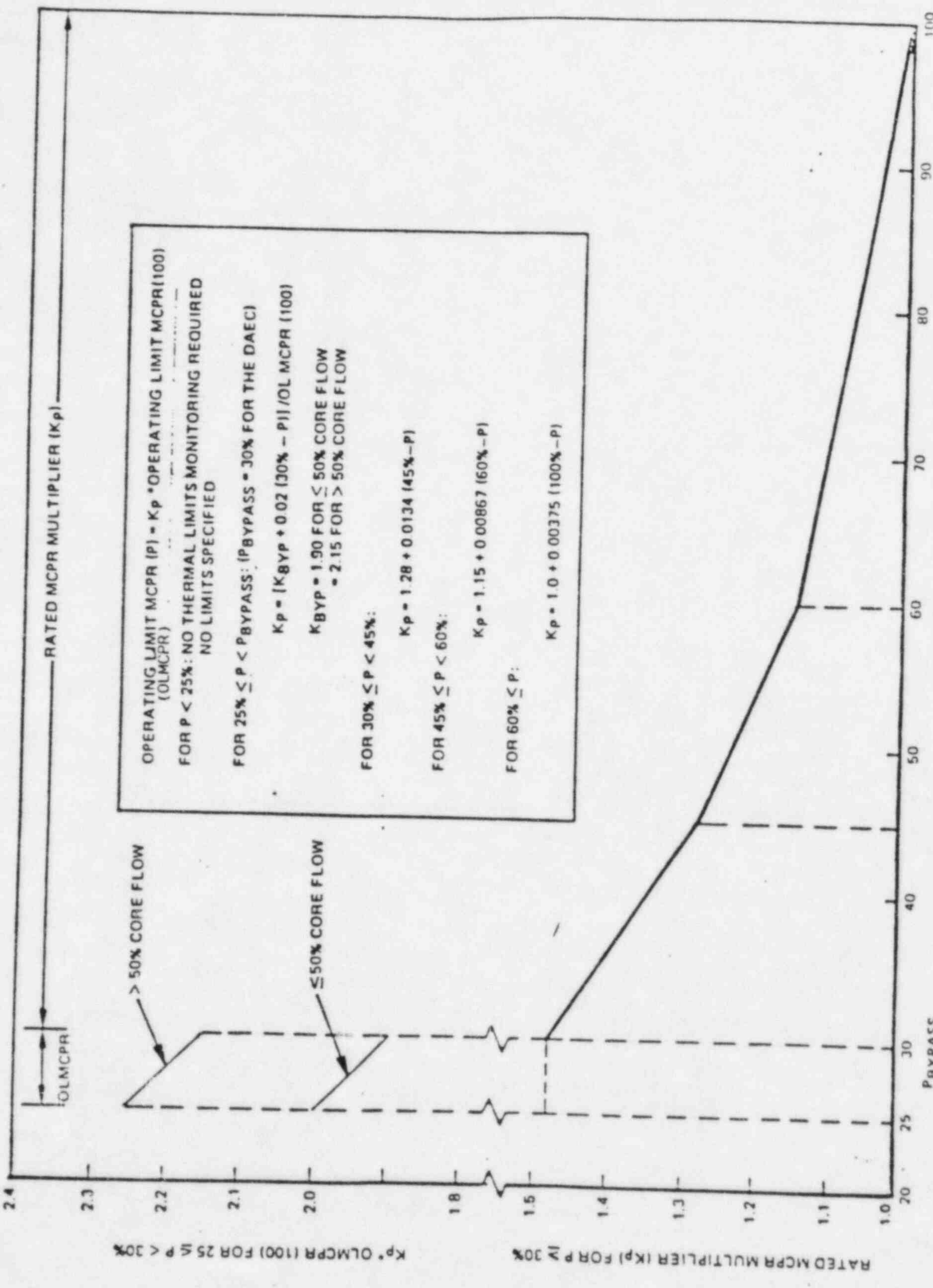
MINIMUM CRITICAL POWER RATIO AT RATED POWER [MCPR(100)] VERSUS τ FUEL TYPE: BP/P8X8R AND ELTA FIGURE 3.12-2



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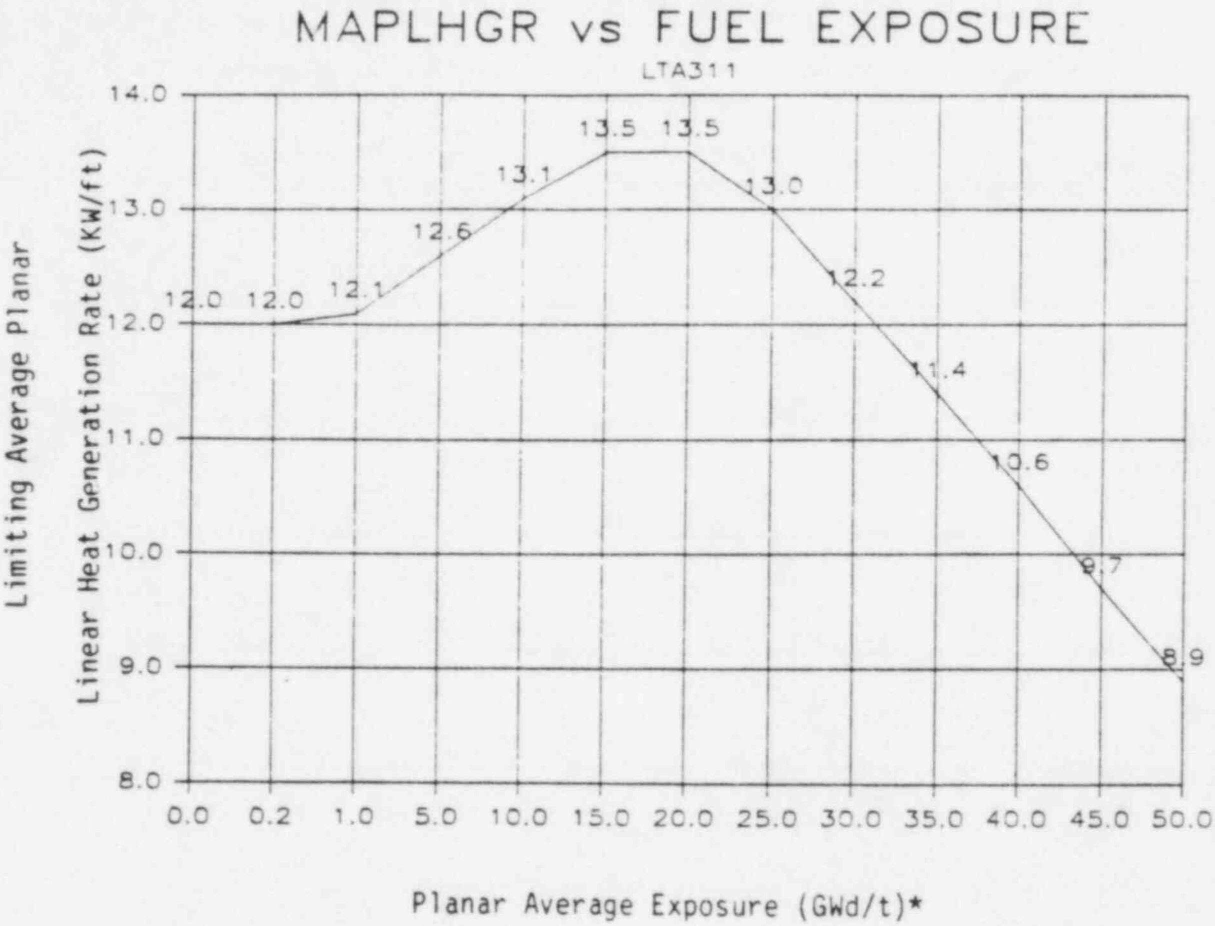
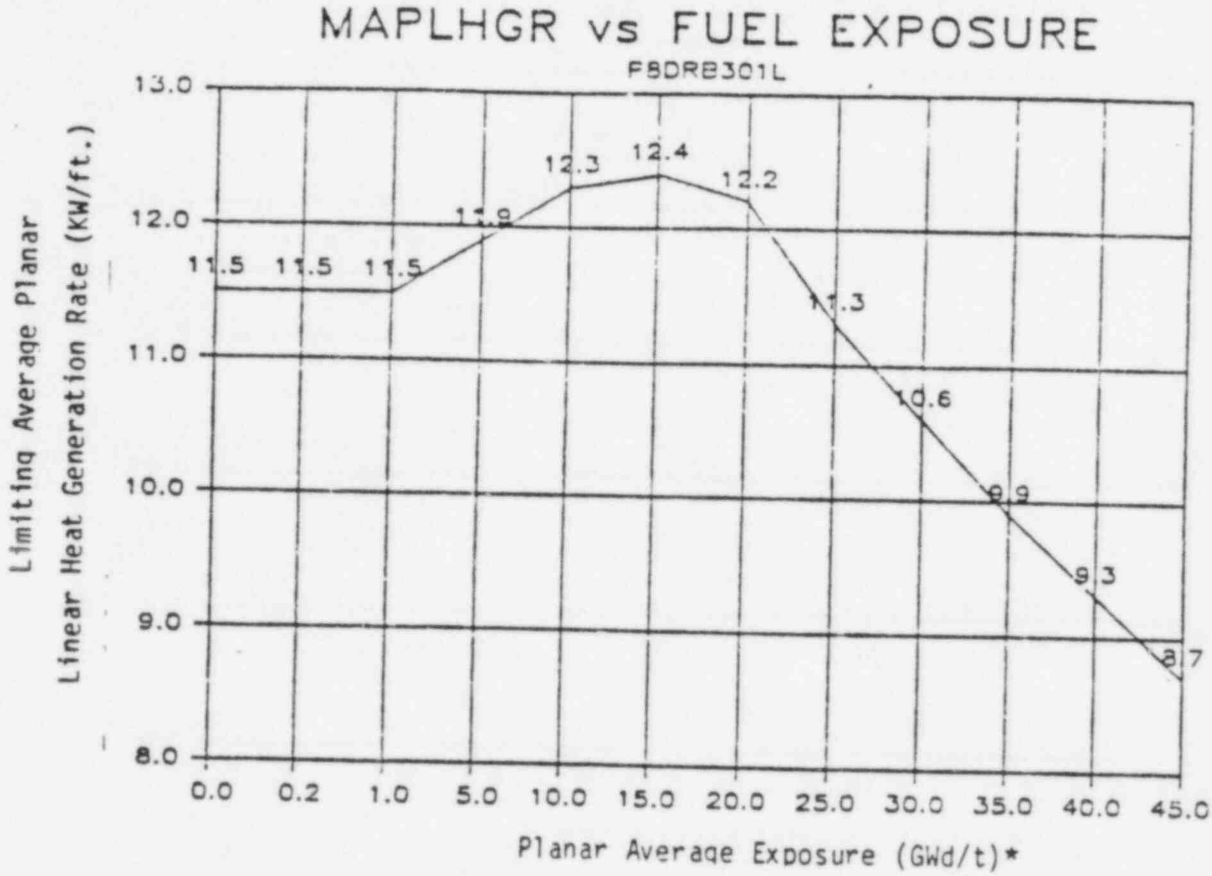
POWER-DEPENDENT MINIMUM CRITICAL POWER RATIO MULTIPLIER (K_p)
FIGURE 3.12-4



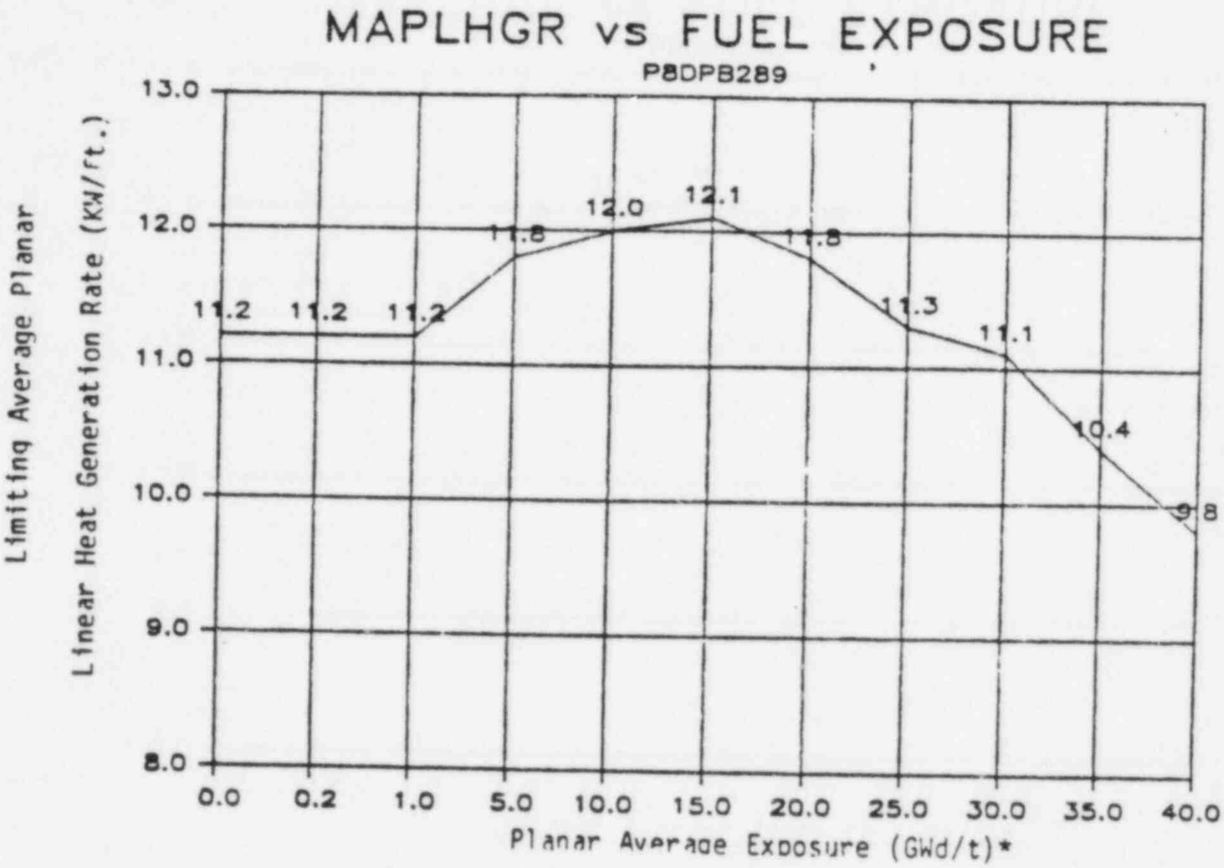
* 1 GWd/t = 1000 MWd/t

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: LTA-311 FIGURE 3.12-5



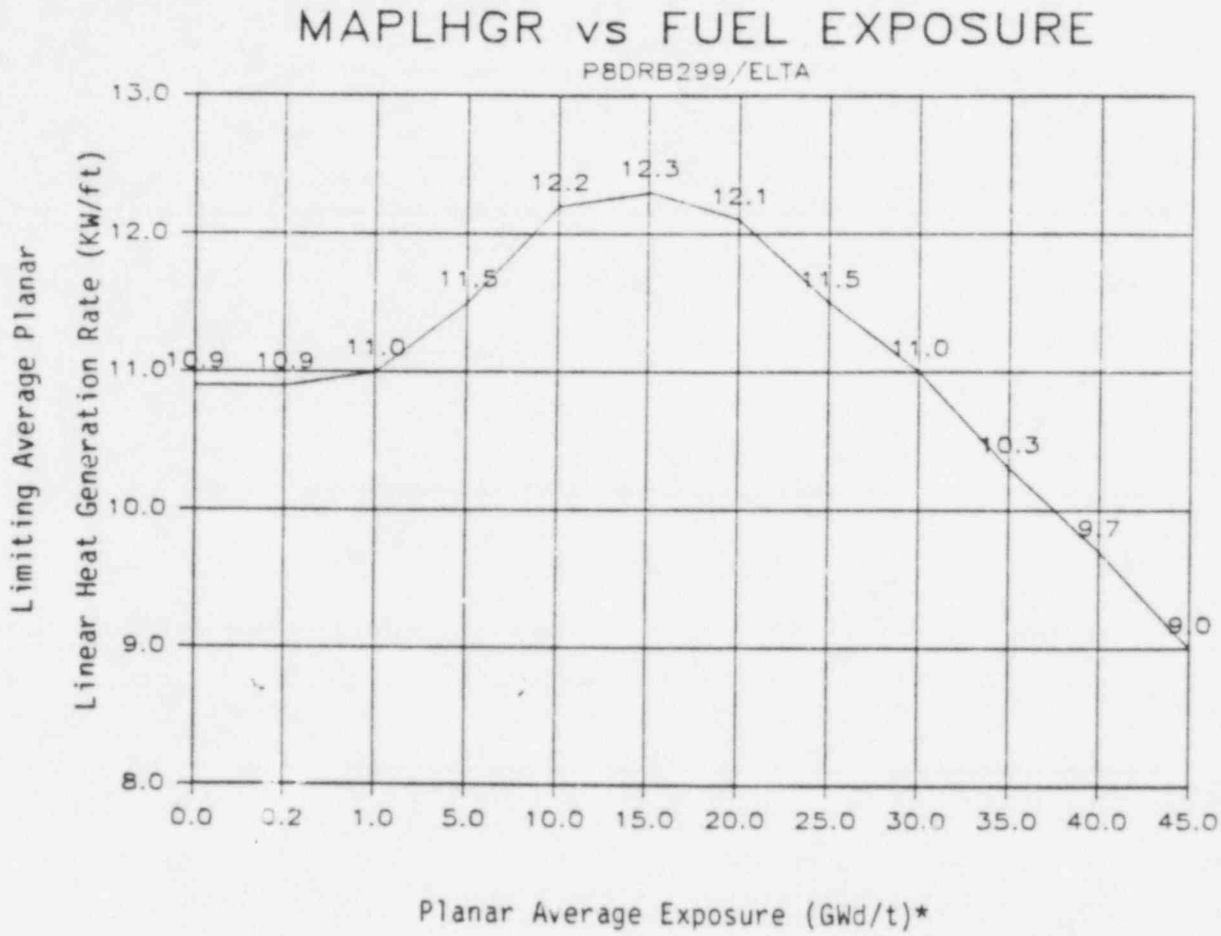
* 1 GWd/t = 1000 MWd/t

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: BP/P8DRB301L FIGURE 3.12-6



* 1 GWd/t = 1000 MWd/t

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: P8DPB289 FIGURE 3.12-7



* 1 GWd/t = 1000 MWd/t

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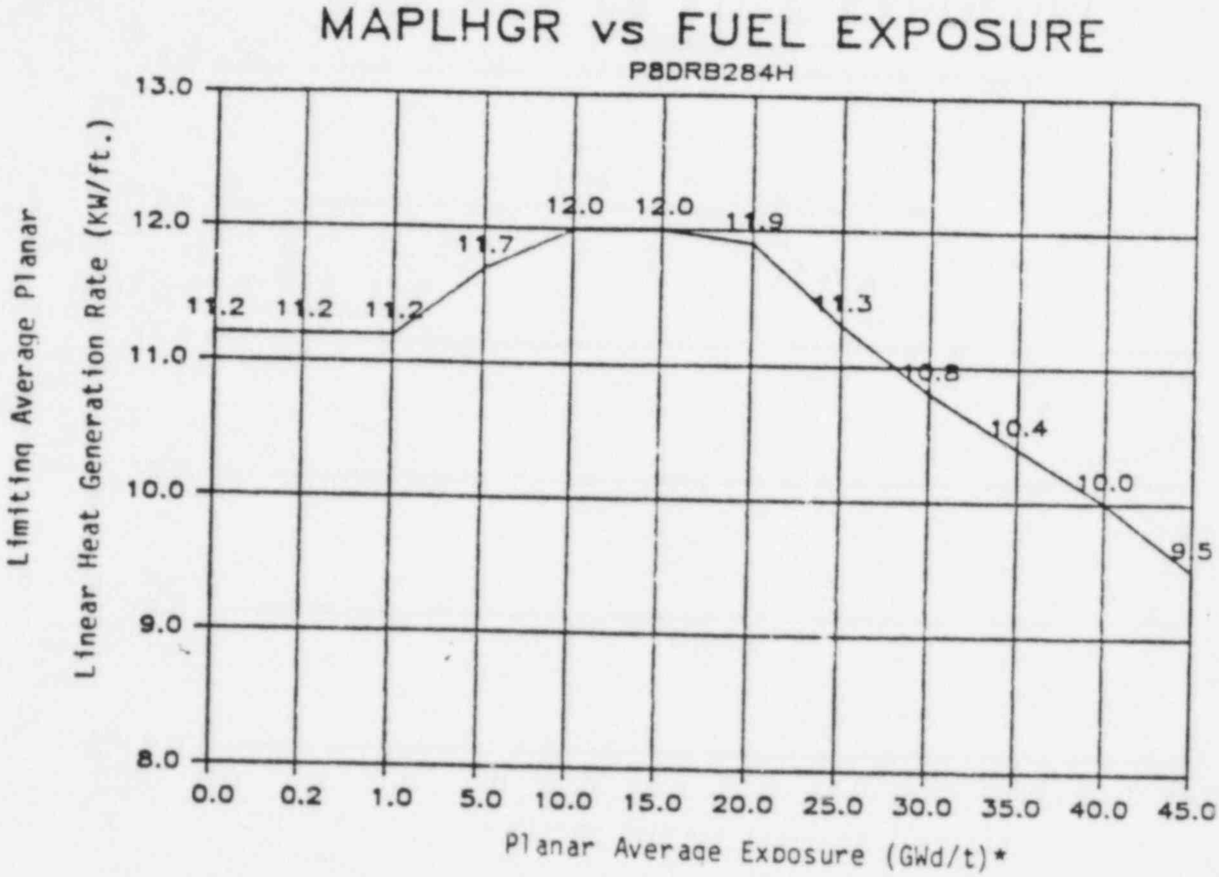
TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE

FUEL TYPE: BP/P8DRB299 and ELTA

FIGURE 3.12-8



* 1 GWd/t = 1000 MWd/t

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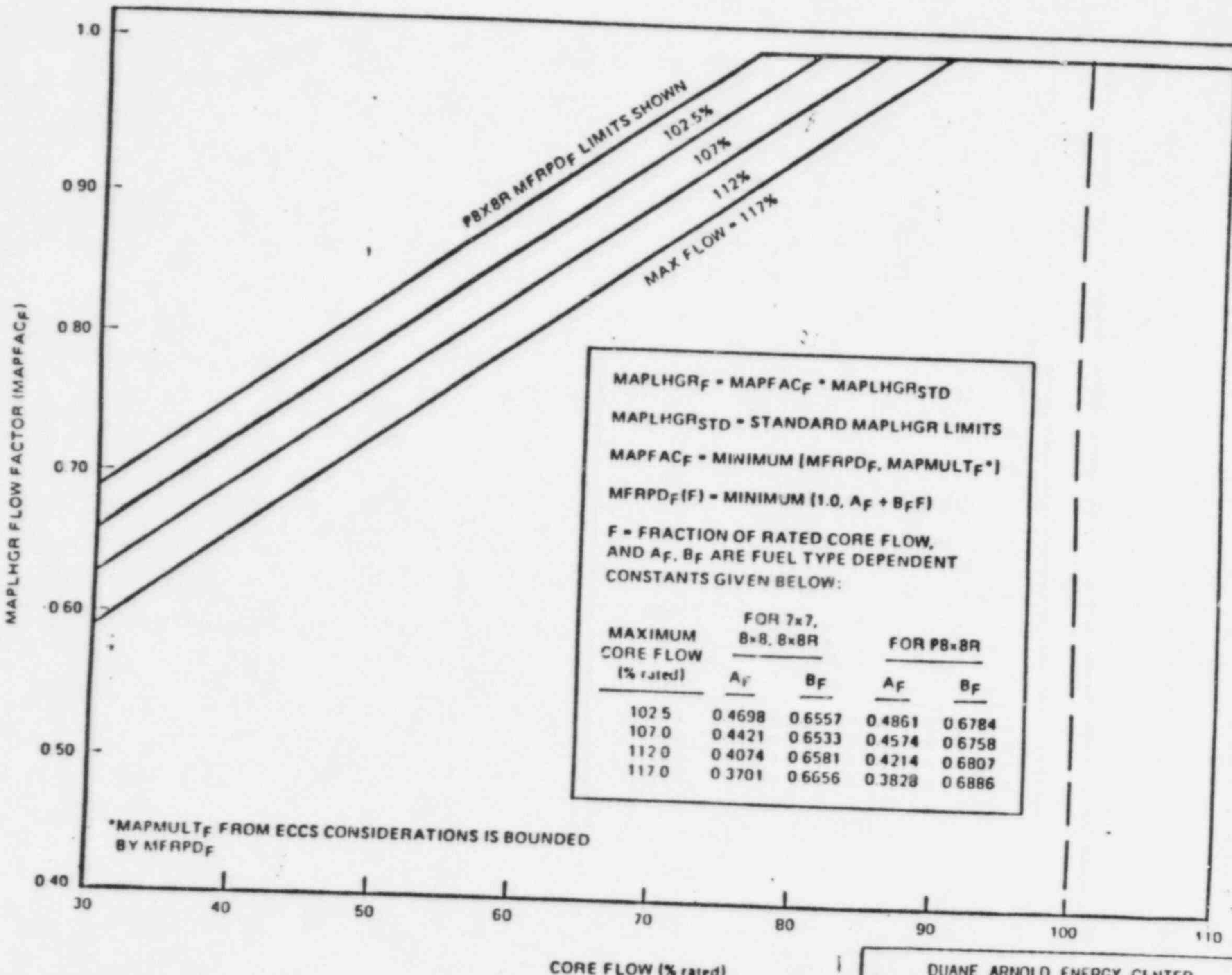
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: P8DRB284H FIGURE 3.12-9



$MAPLHGR_F = MAPFAC_F * MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$*)

$MFRPD_F(F)$ = MINIMUM (1.0, $A_F + B_F F$)

F = FRACTION OF RATED CORE FLOW, AND A_F, B_F ARE FUEL TYPE DEPENDENT CONSTANTS GIVEN BELOW:

MAXIMUM CORE FLOW (% rated)	FOR 7x7, 8x8, 8x8R		FOR P8x8R	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6557	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

*$MAPMULT_F$ FROM ECCS CONSIDERATIONS IS BOUNDED BY $MFRPD_F$

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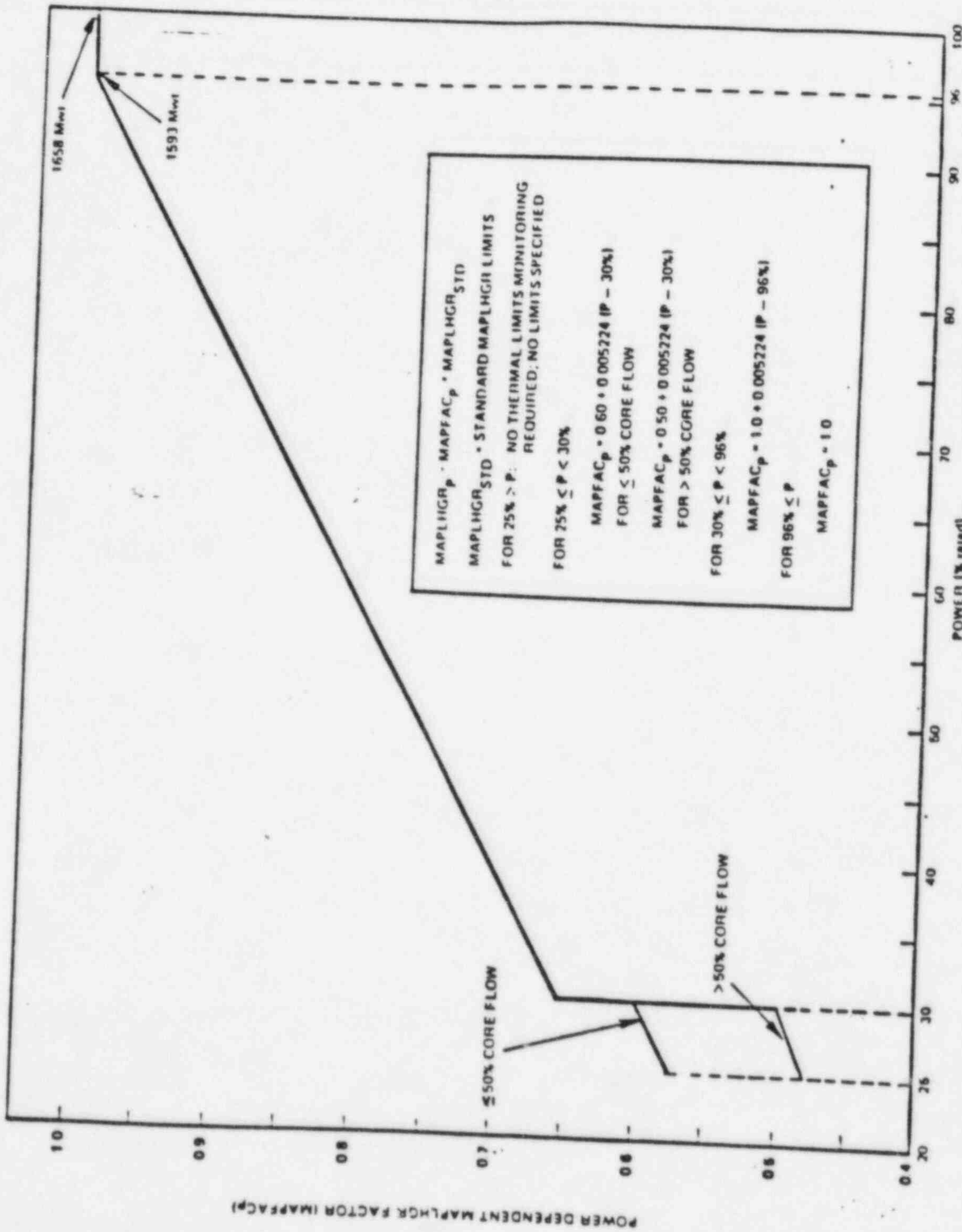
TECHNICAL SPECIFICATIONS

FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER ($MAPFAC_F$)
FIGURE 3.12-10



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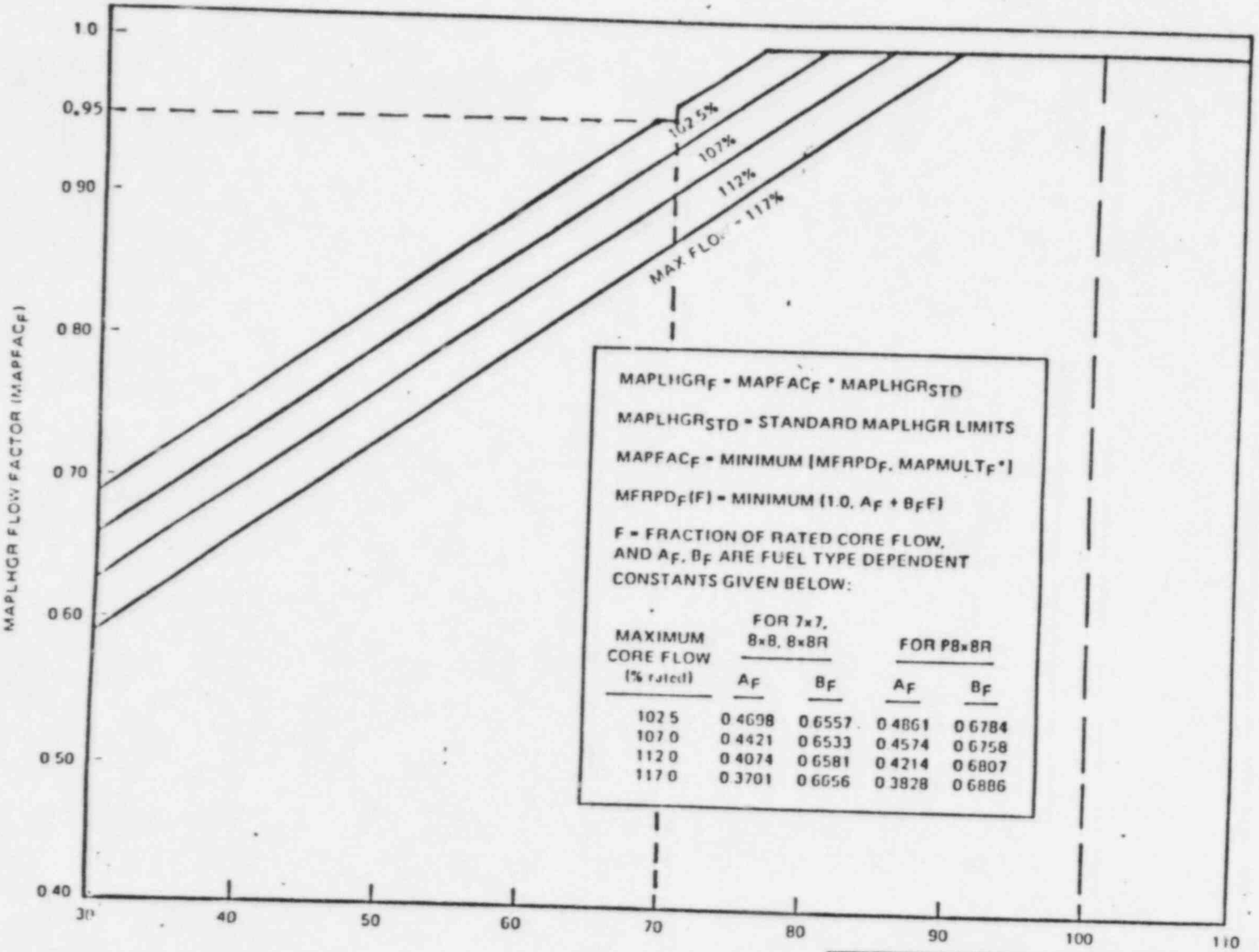
TECHNICAL SPECIFICATIONS

POWER-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER (MAPFAC$_p$)

FIGURE 3.12-11



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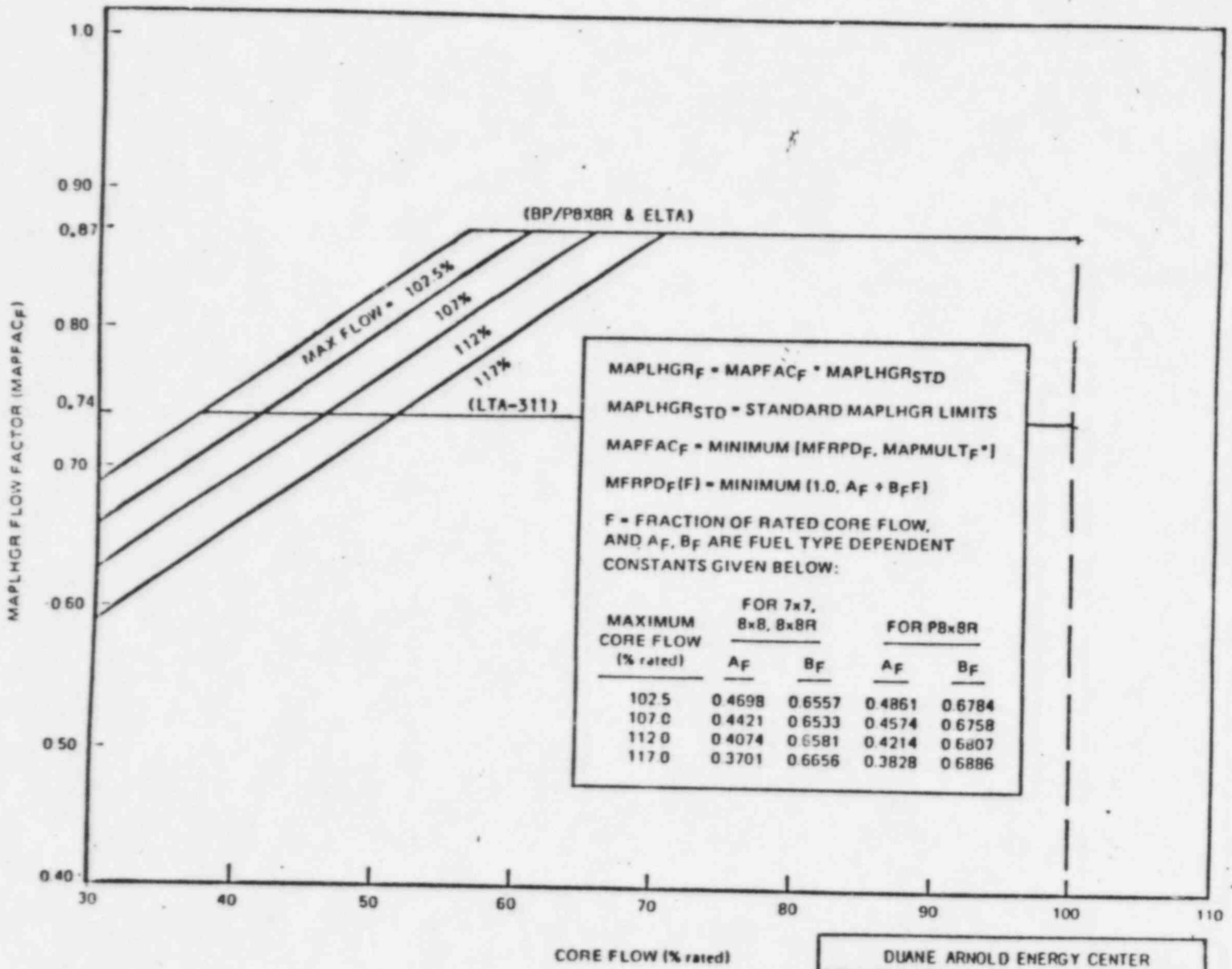
TECHNICAL SPECIFICATIONS

FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER ($MAPFAC_F$) FOR LTA-311
FIGURE 3.12-12



MAXIMUM CORE FLOW (% rated)	FOR 7x7, 8x8, 8x8R		FOR P8x8R	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6557	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

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TECHNICAL SPECIFICATIONS

FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER ($MAPFAC_F$) FOR SLO
FIGURE 3.12-13