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. 119
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light & Power Company, et al, dated December 7, 1984, complies 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 amendme ' 's in accordance with 10 CFR Part 51 of the Commission's regu.ations 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. 119, 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

DBVassallo

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. 119

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

1.1-1	3.3-7
1.1-2	3.3-7a**
1.1-3	3.3-18
1.1-5	3.3-20**
1.1-6	3.6-7
1.1-7	3.6-7a**
1.1-8	3.6-7b**
1.1-9	3.6-31
1.1-10	3.6-34
1.1-11	3.6-35
1.1-12	3.12-1
1.1-13	3.12-2
1.1-14	3.12-3
1.1-15	3.12-4
1.1-16	3.12-7
1.1-17	3.12-8
1.1-18	3.12-9
1.1-19 (Figure 1.1-1)	3.12-9a*
1.1-20 (Figure 2.1-1)	3.12-10
1.1-21*	3.12-11
1.1-22*	3.12-13 (blank page)
1.1-23*	3.12-15
1.1-24*	3.12-16
3.1-3	3.12-17
3.2-16	3.12-18
	3.12-19*
	3.12-20*

*These pages have been deleted.

**These are new pages

SAFETY LIMIT

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) shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure ≤ 785 psig or Core Flow ≤ 10% of Rated

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

LIMITING SAFETY SYSTEM SETTING

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.

For operation with the fraction of rated power (FRP) greater than or equal to the maximum fraction of limiting power density (MFLPD), the APRM scram trip setpoint shall be as shown on Figure 2.1-1 and shall be:

$S \leq (0.66W + 54)$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

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.

For a MFLPD greater than FRP, the APRM scram setpoint shall be:

$$S \leq (0.66\ W + 54)\ \frac{FRP}{MFLPD}$$ for two

recirculation loop operation, and

$$S \leq (0.66\ W + 50.5)\ \frac{FRP}{MFLPD}$$

for one recirculation loop operation.

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 13.4 KW/ft (8x8 array) and MCPR $\geq$ values as indicated in Table 3.12-2 times K_f, where K_f is defined by Figure 3.12-1. Therefore, at full power, operation is not allowed with MFLPD greater than unity even if the scram setting is reduced.* If it is determined that either of these design criteria is being violated during operation, action must be taken immediately 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.

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

3. APRM Rod Block when in Run Mode.

   For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be:

   $S \leq (0.66\ W + 42)$

   The definitions used above for the APRM scram trip apply.

   For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be:

   $S \leq (0.66\ W + 42)\ \frac{FRP}{MFLPD}$ for two

   recirculation loop operation, and

   $S \leq (0.66\ W + 38.5)\ \frac{FRP}{MFLPD}$

   for one recirculation loop operation.

4. 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 single-loop operation.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig or Core Flow $\leq$10% of Rated)

At pressures below 785 psig, the core evaluation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux folowing close of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Duane Arnold has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc., occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored.

*Top of the active fuel zone is defined to be 344.5 inches above vessel zero (See Bases 3.2).

1.1 REFERENCES

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A* |

2. "Duane Arnold Energy Center Single-Loop Operation," NEDO-24272 July 1980. |

*Approved Revision at time reload analyses are performed.

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Duane Arnold Energy Center have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 102% of 1658 MWt in accordance with Regulatory Guide 1.49. The analyses were based upon plant operation in accordance with the operating map given in Figure 1.1-1 of the Technical Specifications. In addition, 1658 MWt is the licensed maximum power level of the Duane Arnold Energy Center, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analysis in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis mode. Conservatism incorporated into the transient analysis is documented in Reference 1.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more conservative results than would be obtained by using expected values of control parameters and analyzing at higher power levels.

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 one recirculation pump operation is conservatively bounded by two-loop operation analyses and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

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.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Trips

1. APRM High Flux Scram (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1658 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin to the threshold for fuel damage. Therefore, the use of flow referenced 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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum fraction of limiting power density is greater than the fraction of rated power. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow-referenced APRM High Flux Scram curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than or equal to the Safety Limit when the transient is initiated from MCPR $\geq$ values as indicated in Figures 3.12-2 and 3.12-3.

2. APRM High Flux Scram (Refuel or Startup & Hot Standby Mode)

For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide adequate thermal margin between the setpoint and the Safety Limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worths of individual rods are very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15 percent

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. APRM Rod Block (Run Mode)

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than the Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents excessive reactor power level increase resulting from control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

4. 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 the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the Safety Limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. Scram and Isolation on Reactor Low Water Level

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. Analyses show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than the Safety Limit in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 21 inches below the normal operating range and is thus adequate to avoid spurious scrams.

C. Scram - Turbine Stop Valve Closure

The turbine stop-valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above the Safety Limit even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when core thermal power is below 30 percent of rated, as measured by the turbine first stage pressure.

D. Turbine Control Valve Fast Closure (Loss of Control Oil Pressure Scram)

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection. It prevents MCPR from becoming less than the Safety Limit for this transient.

E. F. and J. Main Steam Line Isolation on Low Pressure, Low Condenser Vacuum, and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 850 psig has been provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase. To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the main steam isolation valves.

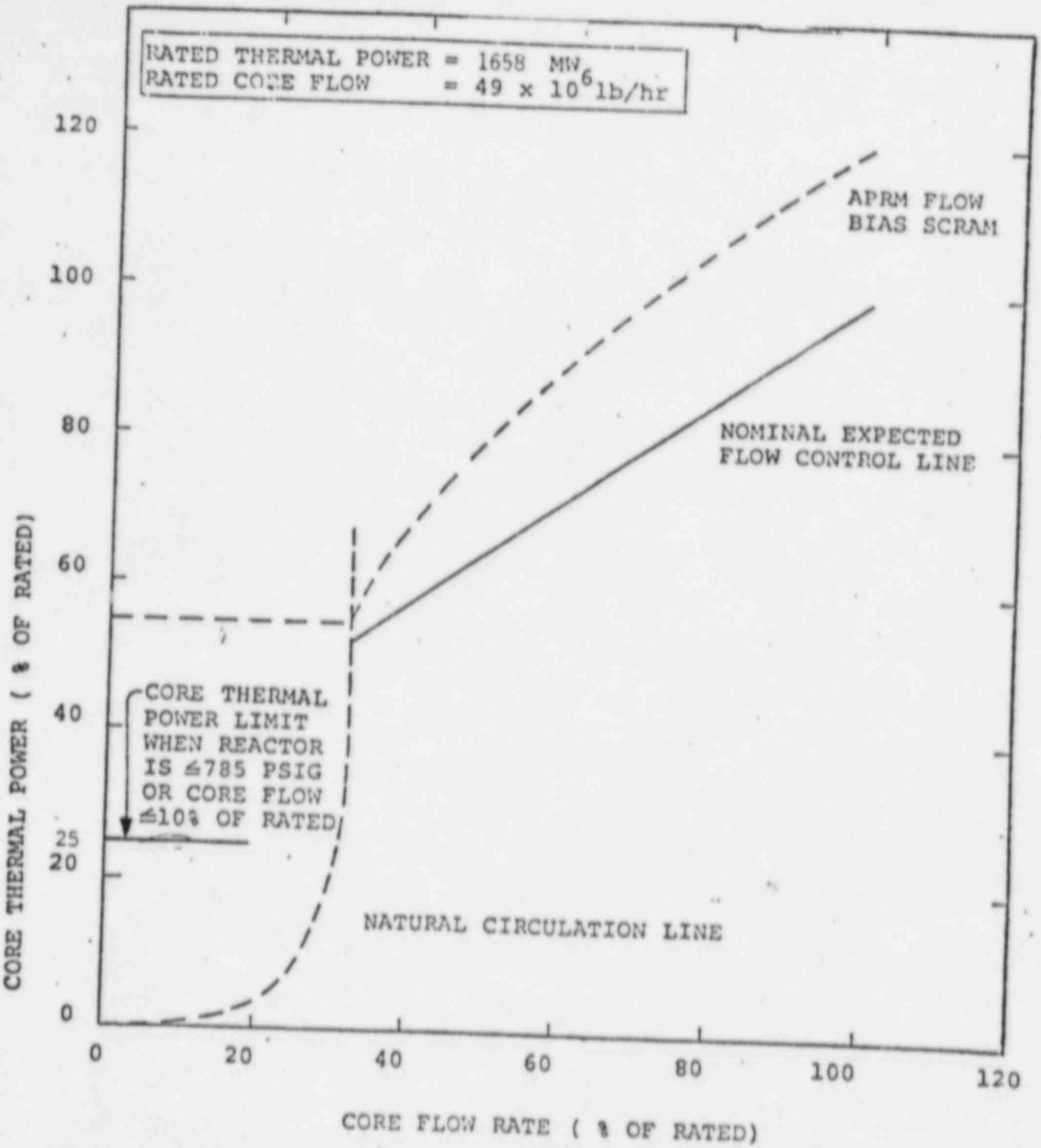
G. H. and I. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC, Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

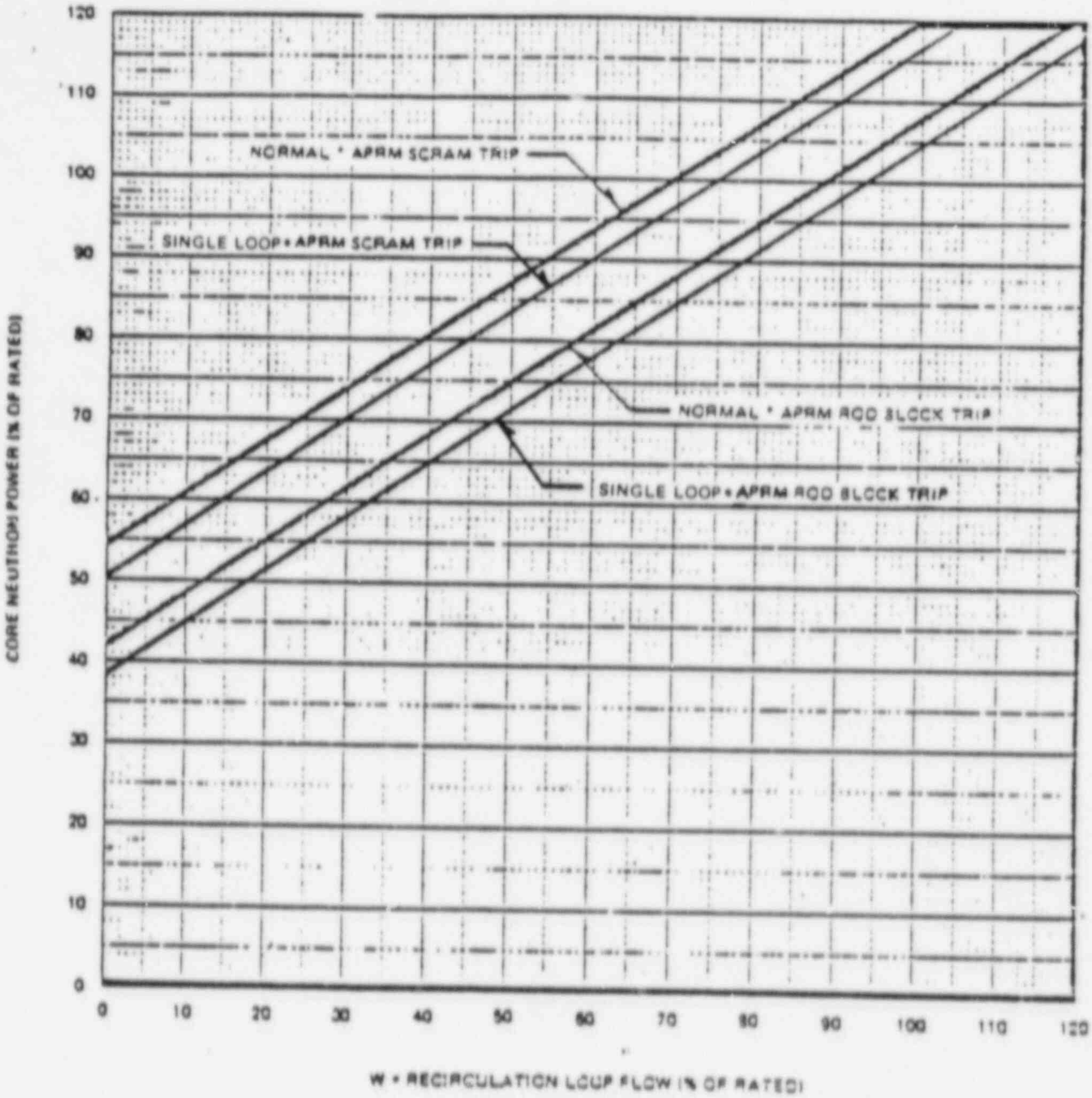
2.1 REFERENCES

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A*
2. "Duane Arnold Energy Center Single-Loop Operation," NEDO-24272, July 1980.

*Approved revision number at time analyses are performed.



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

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: Refuel (6)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
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 ≤(.66W+54)(FRP/MFLPD) (11) (12) for one recirc loop operation ≤(.66W+50.5)(FRP/MFLPD) (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

TABLE 3.2-C
Instrumentation That Initiates Control Rod Blocks

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	for 2 recirc loop operation $\leq(0.66\ W + 42)\ (\frac{FRP}{MFLPD})^{(2)}$ for 1 recirc loop operation $\leq(0.66\ W + 38.5)\ (\frac{FRP}{MFLPD})^{(2)}$	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 (Flow Biased)	for 2 recirc loop operation $\leq(0.66\ W + 39)(\frac{FRP}{MFLPD})^{(2)}$ for 1 recirc loop operation $\leq(0.66\ W + 35.5)\ (\frac{FRP}{MFLPD})^{(2)}$	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
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)

LIMITING CONDITIONS FOR OPERATION

3.3.D Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. Recirculation Pumps

When the reactor mode switch is in startup or run position, the reactor shall not be operated in the natural circulation flow mode.

With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3-1 and total core flow less than 45% of rated, the APRM and LPRM* neutron flux noise levels shall be determined within 2 hours, and:

a) if the APRM and LPRM* neutron flux noise

*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.

SURVEILLANCE REQUIREMENT

4.3.D Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

E. Recirculation Pumps

With two recirculation pumps in operation and with core thermal power greater than the limit specified in Figure 3.3-1 and total core flow less that 45% of rated, establish baseline APRM and LPRM* neutron flux noise levels within 2 hours, provided that baseline values have not been previously established since the last core refueling.

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 operation with one recirculation loop not in operation. 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.	

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.

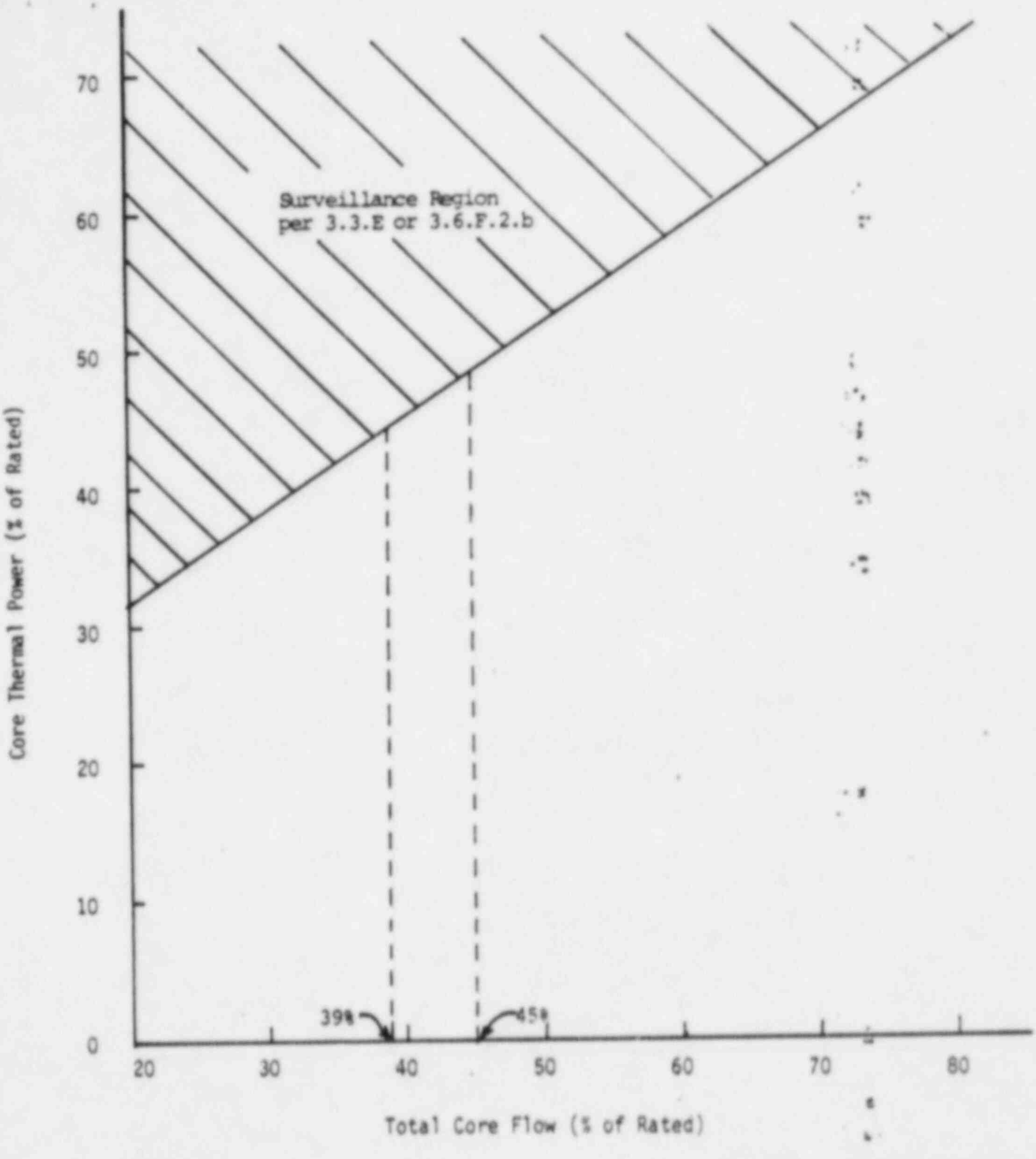
4. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison 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.

5. Recirculation Pumps

APRM and/or LPRM oscillations in excess of those specified in section 3.3.E could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. These specifications are based upon the guidance of GE SIL #380, Rev. 1, 2/10/84.



DAEC Iowa Electric Light & Power Company Technical Specifications
Thermal Power vs Core Flow Limits for Thermal Hydraulic Stability Surveillance Figure 3.3-1

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	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.	1. Recirculation pump speeds shall be checked and logged at least once per day.
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 with one recirculation loop not in operation provided that:	2. a. Prior to operation with one recirculation pump not in operation 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. With one recirculation pump not in operation 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 with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.3-1. b. Prior to operation with one recirculation pump not in operation 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 with one recirculation pump not in operation 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. With one recirculation pump not in operation 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 single recirculation loop operation, 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 reactor operation with one recirculation loop not in operation (Sec. 3.12, Ref. 11). Therefore, continuous operation under such conditions is appropriate. The reactor may also be operated up to 24 hours with one recirculation loop not in operation 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 operation with one recirculation loop not in operation.

During periods of Single Loop Operation (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 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.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is below 50% of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G & 4.6.G BASES:

REACTOR COOLANT SYSTEM
Structural Integrity

A pre-service inspection of Nuclear Class I Components was conducted to assure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the reactor coolant system as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no LOCA would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the reactor coolant system, portions of the ECCS, and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II Components because it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

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)

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. 3.12-5, -6, -7, -8 and -9. For single-loop operation, the values in these curves are reduced by multiplying by 0.87. If at any time during reactor power operation (one or two loop) at ≥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 ≤ 25% of rated thermal power, or to such a power level that the limits are again being met, within the next 4 hours.

If the reactor is being operated with one recirculation loop not in operation 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.

For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed limits are again being met.

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 surveillancento 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 ≥ 25% rated thermal power and any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for specification 3.3.2. During operation with a limiting control rod pattern, the MAPLHGR (LAPLHGR) shall be determined at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
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 P8X8R, BP8X8R 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. If at any time during reactor power operation at ≥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 ≤ 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.	The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern the LHGR shall be determined at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

During reactor power operation, MCPR for one or two recirculation loop operation shall be $\geq$ values as indicated in Figures 3.12-2 and -3. These values are multiplied by K_f which is shown in Figure 3.12-1. Note that for one recirculation loop operation the MCPR limits at rated flow are 0.03 higher than the comparable two-loop values. 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 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.

If the reactor is being operated with one recirculation loop not in operation, 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.

For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed limits are again being met.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern, the MCPR shall be determined at least once per 12 hours.

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.

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 one recirculation loop operation were derived in Reference 13.

2. MCPR Limits for Core ows Other than Rated Flow

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of values as indicated in Figures 3.12-2 and 3.12-3 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.12-1 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow, as described in Reference 2.

The K_f factors shown in Figure 3.12-1 are conservative for Duane Arnold operation because the operating limit MCPR of values as indicated in Figures 3.12-2 and 3.12-3 is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

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.

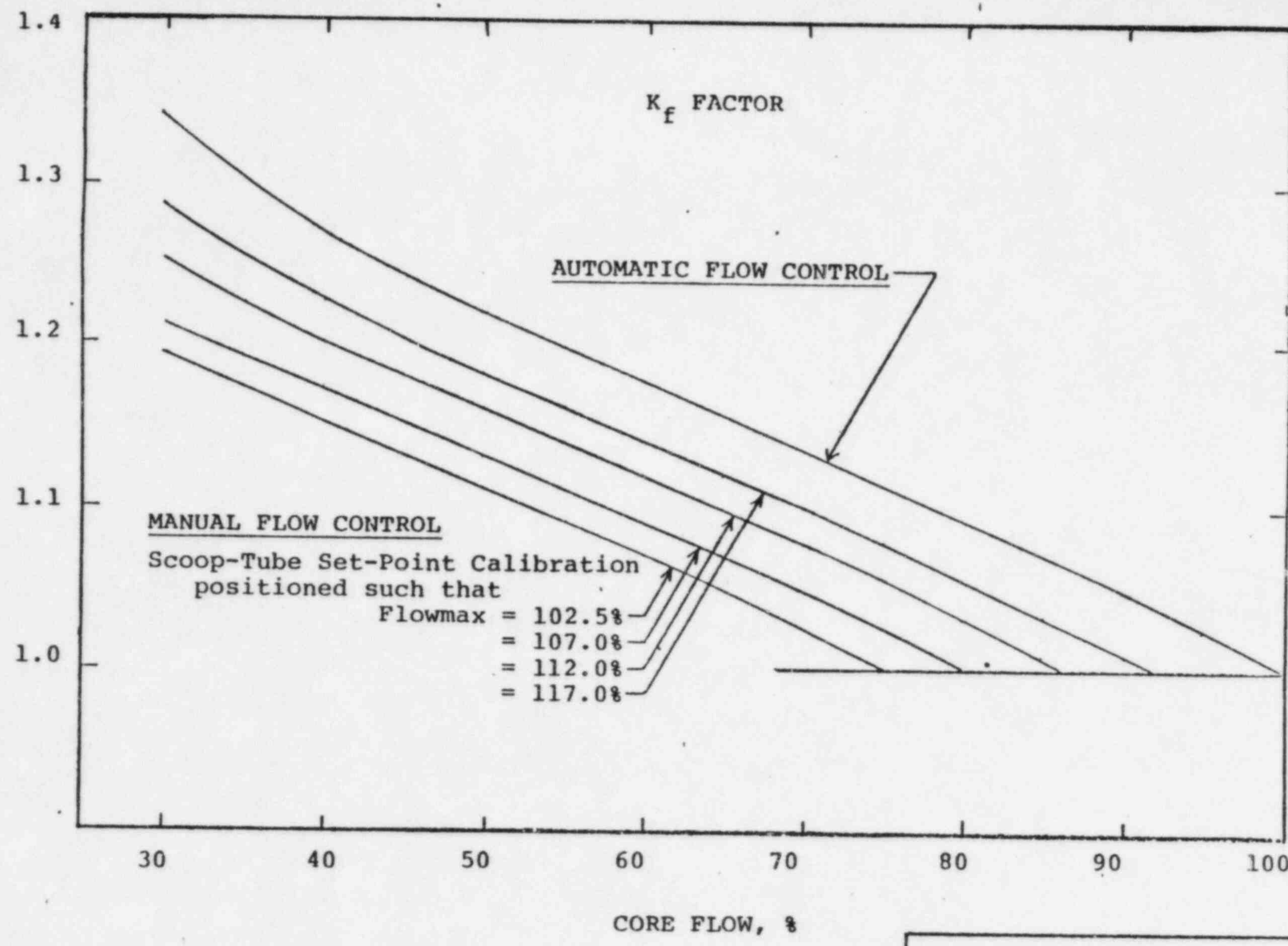
10. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 5: Revised Operating Limits for Loss of Feedwater Heating, NEDO-24987-5.

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.

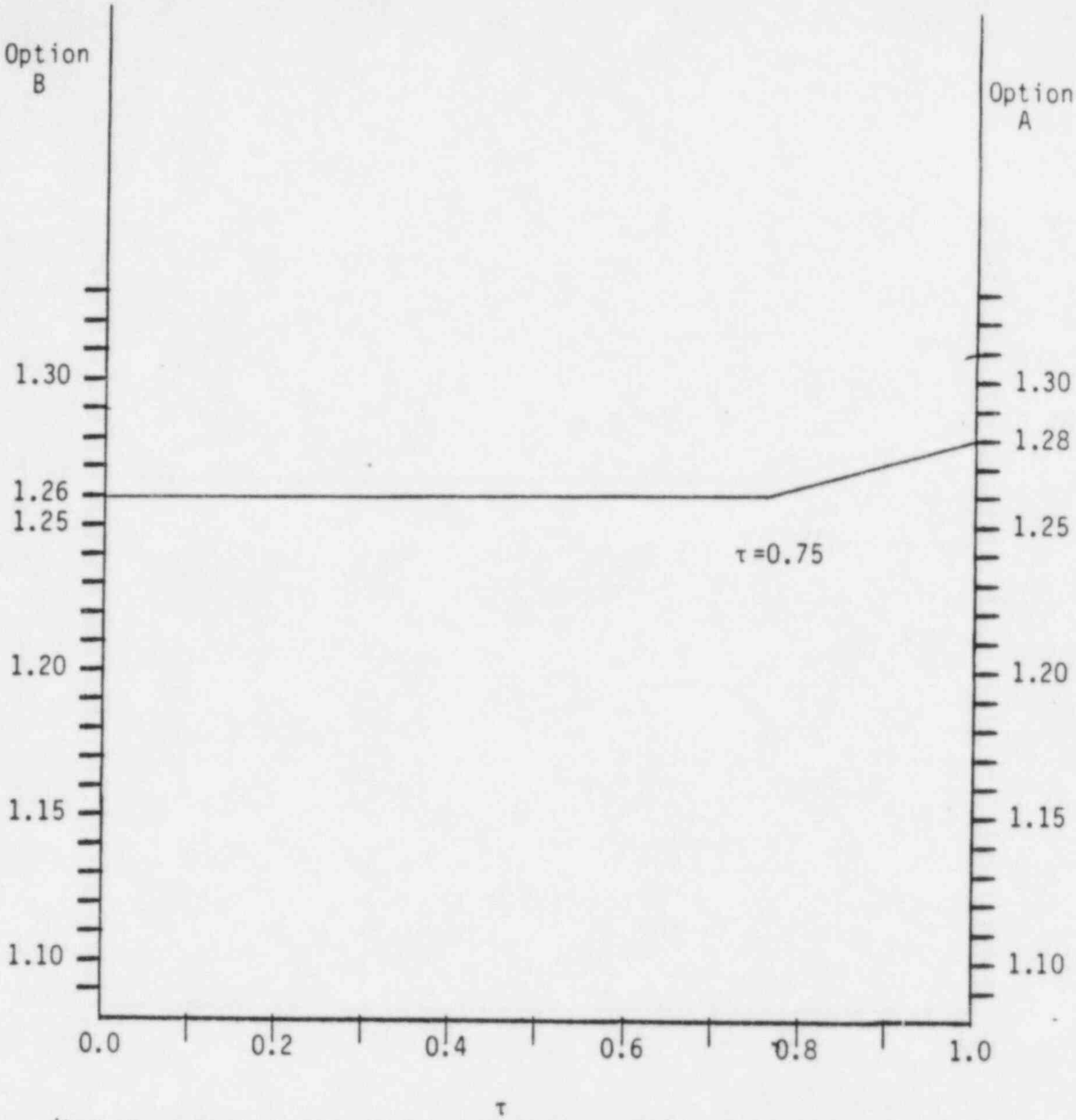
**Approved revision number at time reload fuel analyses are performed.



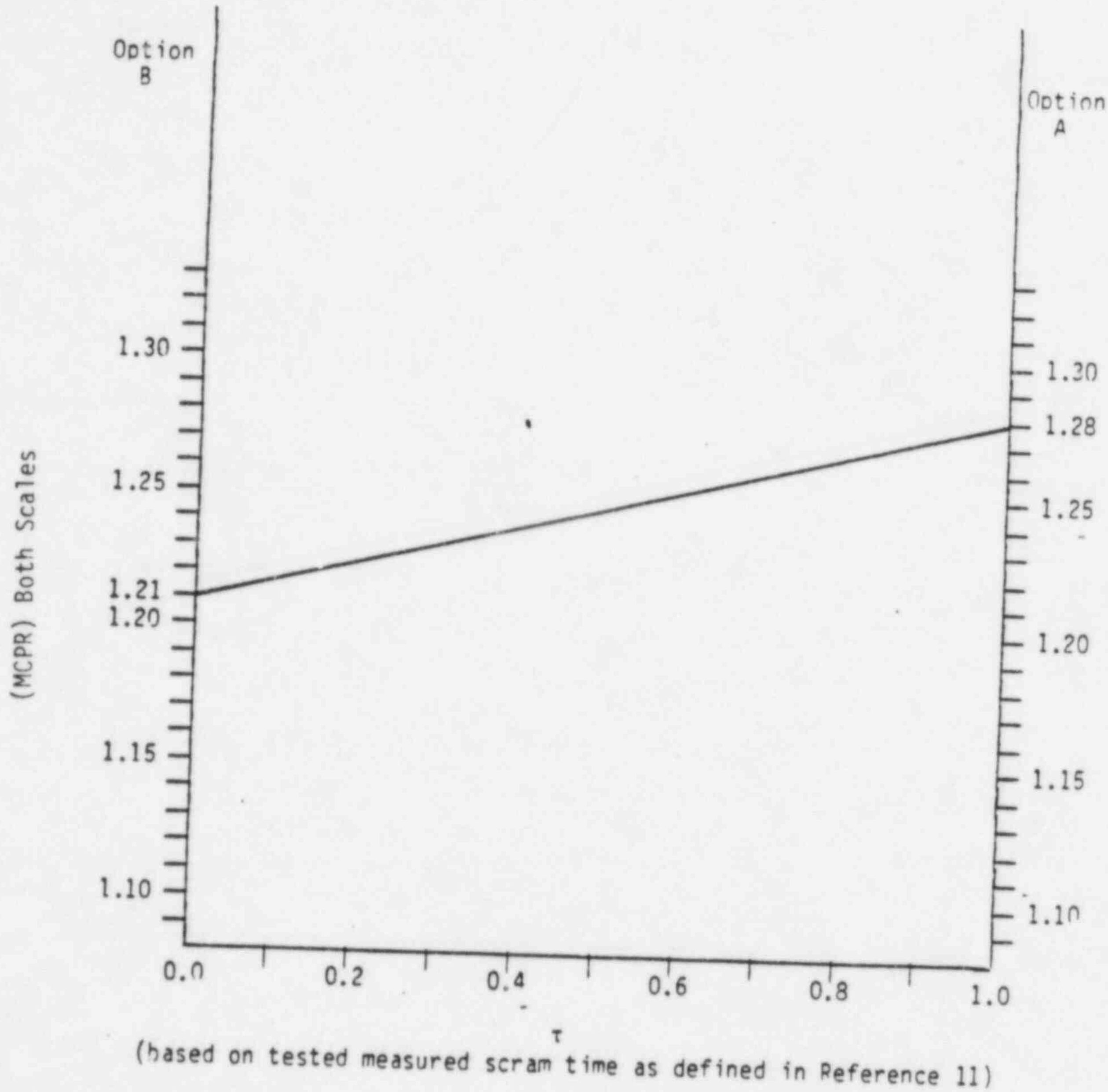
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
TECHNICAL SPECIFICATIONS

K_f AS A FUNCTION OF CORE FLOW

FIGURE 3.12-1

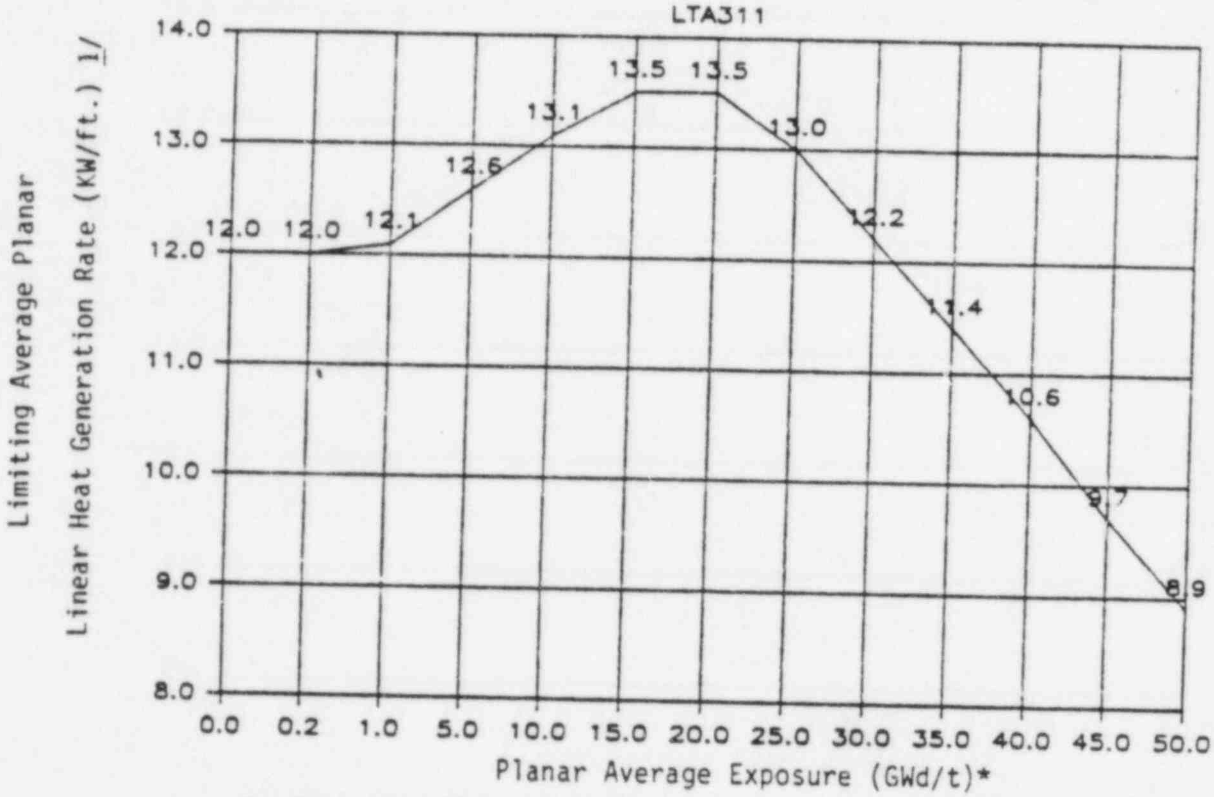


DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ FUEL TYPES: BP/P8X8R and ELTA FIGURE 3.12-2



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ FUEL TYPE: LTA 311 FIGURE 3.12-3

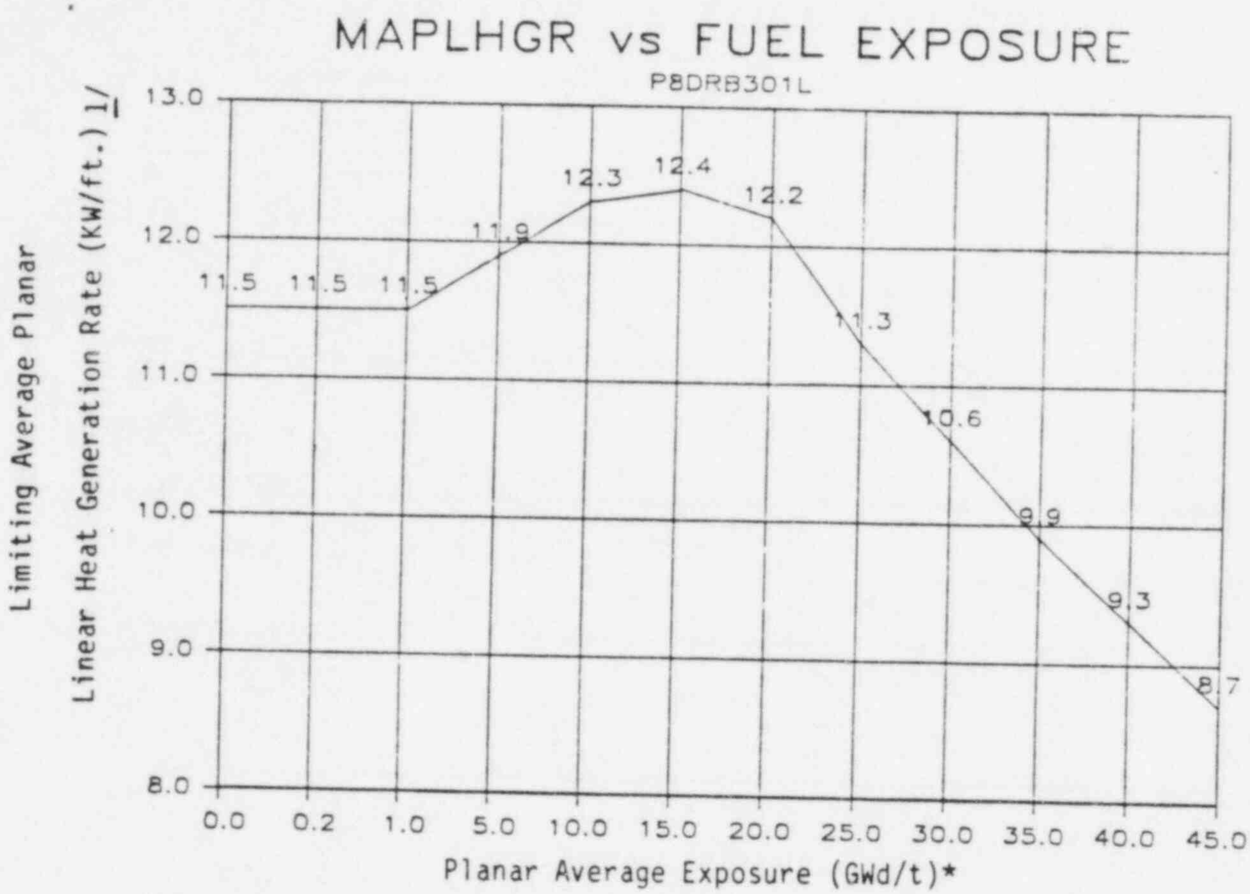
MAPLHGR vs FUEL EXPOSURE



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown.

* 1 GWd/t = 1000 MWd/t

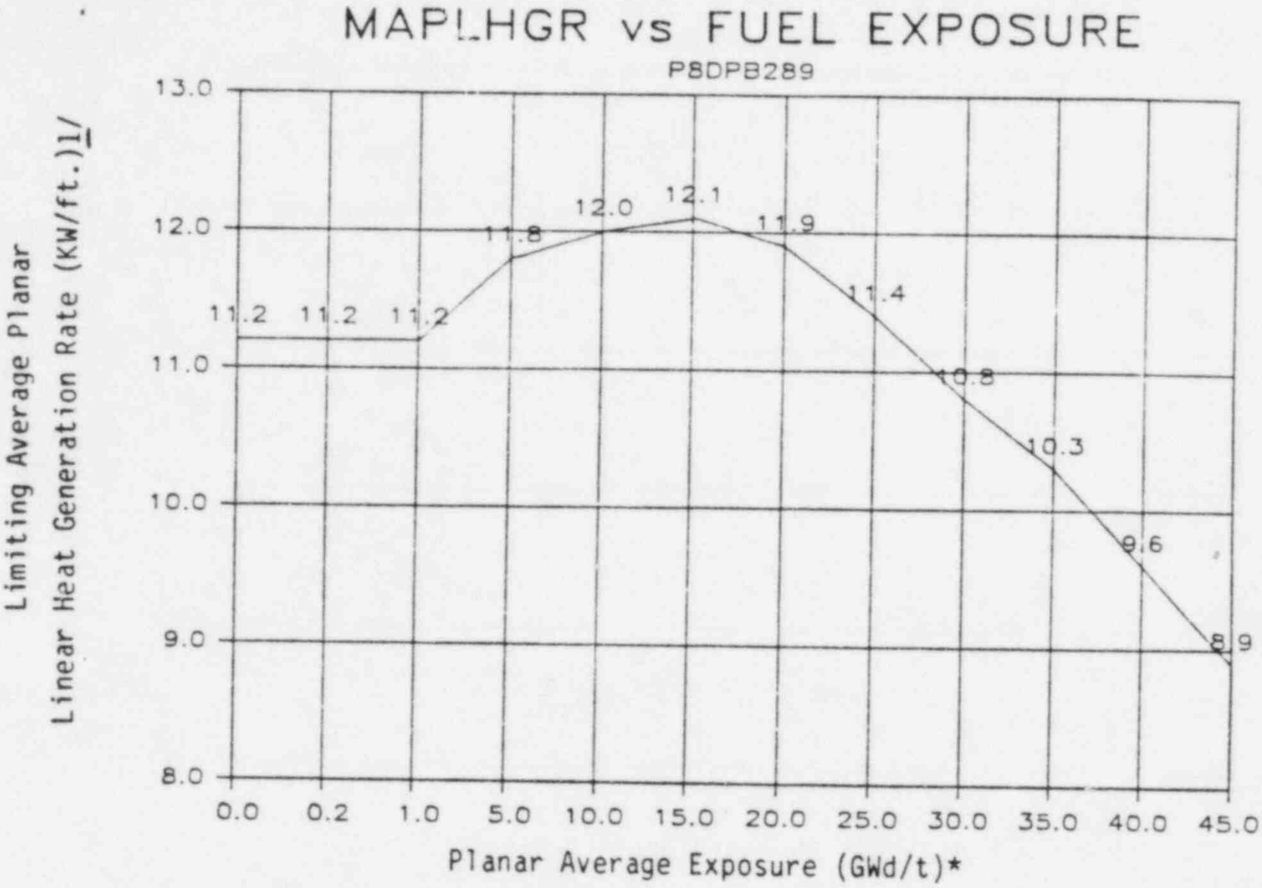
DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: LTA 311 FIGURE 3.12-5



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop operation are given in Section 3.12.A.

* 1 GWd/t = 1000 MWd/t

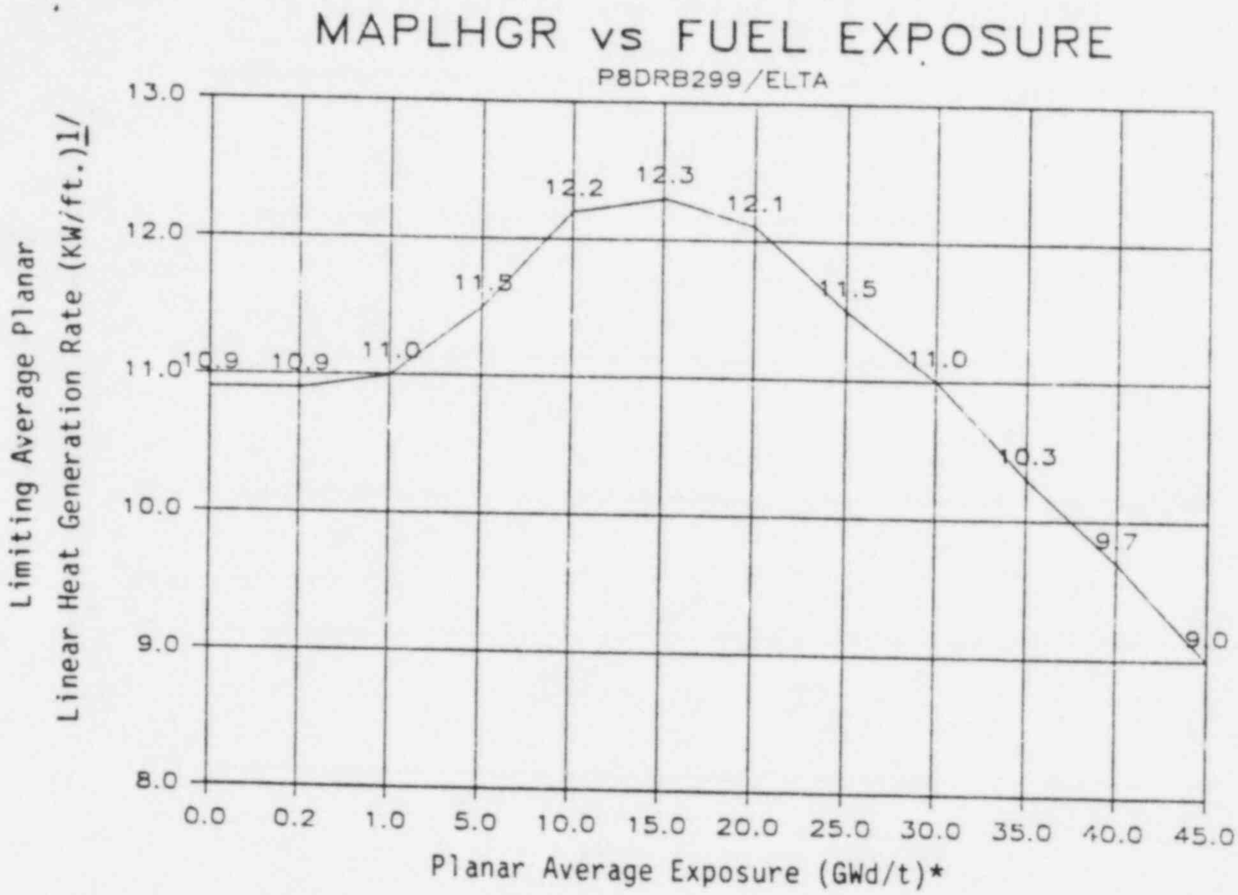
DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: BP/P8DRB301L FIGURE 3.12-6



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop operation are given in Section 3.12.A.

* 1 GWd/t = 1000 MWd/t

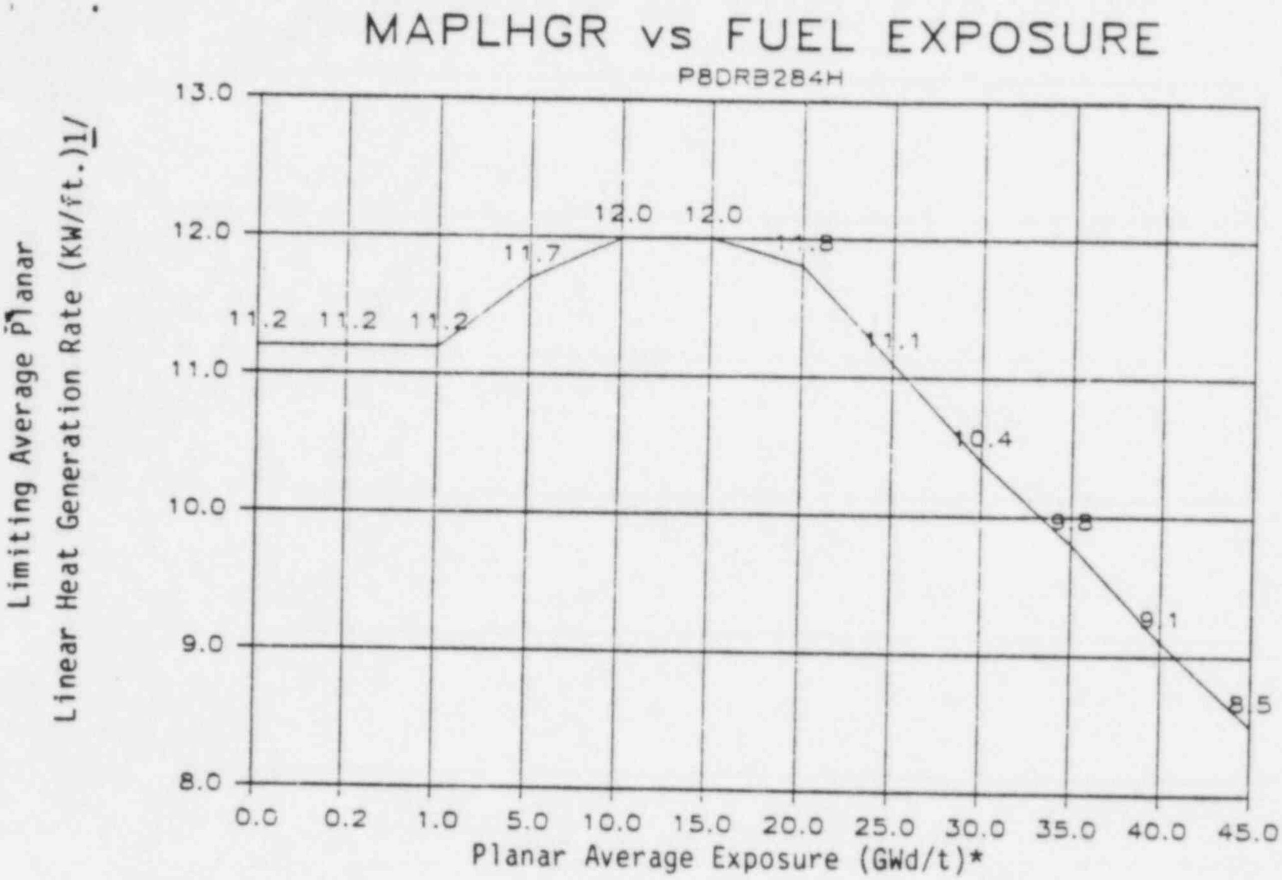
DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: P8DPB289 FIGURE 3.12-7



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop operation are given in Section 3.12.A., (except ELTA).

* 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/P8DRB299 and ELTA
FIGURE 3.12-8



1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop operation are given in Section 3.12.A.

* 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: P8DRB284H FIGURE 3.12-9