TECHNICAL SPECIFICATIONS

LIST OF FIGURES

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 $\leq$ 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 (1,593 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< 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.

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. For one recirculation loop operation APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise averaged over 1/2 hour exceeds 8% peak to peak, as measured on the APRM chart recorder.

5. 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" on level instruments)
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 $\geq$ 785 psig and Core Flow $\geq$ 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 uncertainity 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 10^3lbs/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 1658 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. 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.

Transient analyses performed each reload are given in Reference 1. Models and model conservatisms are also described in this reference. 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.

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 (1593 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 from fuel damage. Therefore, the use of flow referenced scram trip provides even 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.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR greater than or equal to safety limit when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

2. <u>APRM High Flux Scram (Refuel or Startup & Hot Standby Mode)</u>

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 880 psig.

3. <u>APRM Rod Block (Run Mode)</u>

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 establishd 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. APRM FLUX NOISE

APRM flux noise oscillations in excess of these specified in Section 2.1.A.4 could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken.

5. 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 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 safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when turbine steam flow 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 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 880 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 880 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

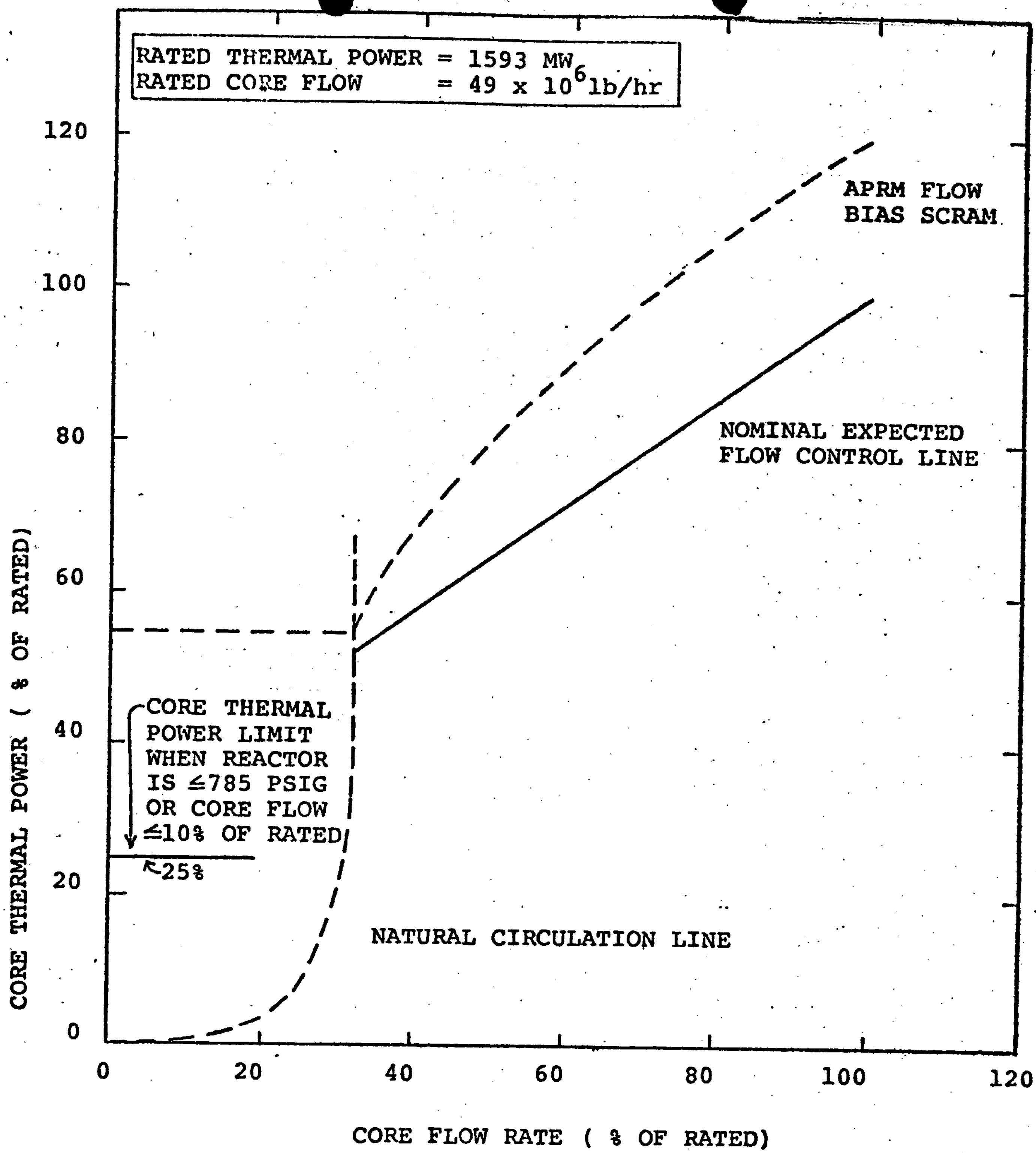
These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses 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.



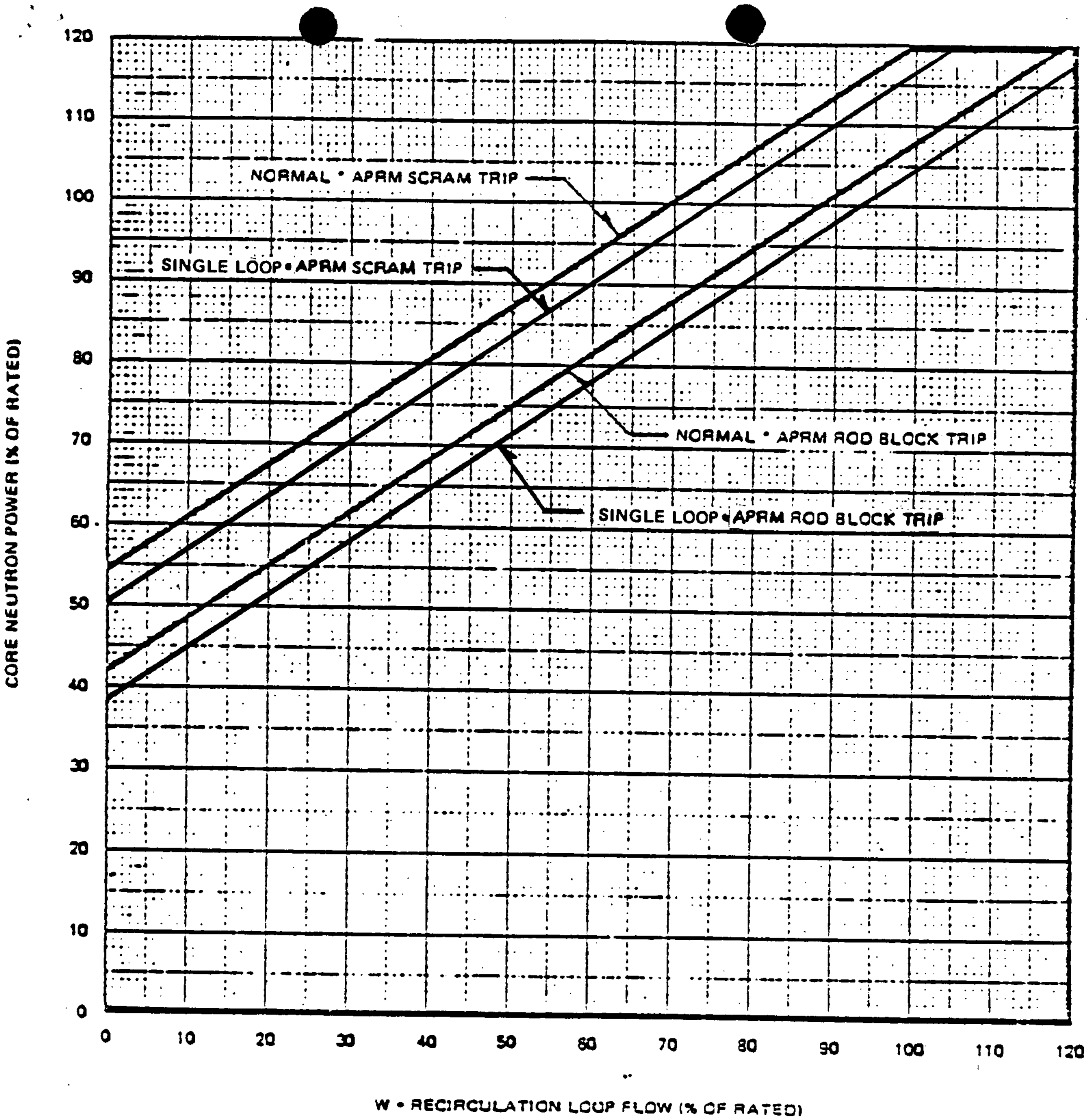
DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT & POWER COMPANY

TECHNICAL SPECIFICATIONS

APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

FIGURE 1.1-1



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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 Fuel 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	≤ 1035 psig	X(8)	X	X	4 Instrument Channels	A

Instrumentation That Initiates Control Rod Blocks

TABLE 3.2-C

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 $\leq(0.66\ W + 42)\ \left(\frac{FRP}{MFLPD}\right)$ (2) for 1 recirc loop operation $\leq(0.66\ W + 38.5)\ \left(\frac{FRP}{MFLPD}\right)$ (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 $\leq(0.66\ W + 39)\left(\frac{FRP}{MFLPD}\right)$ (2) for 1 recirc loop operation $\leq(0.66\ W + 35.5)\ \left(\frac{FRP}{MFLPD}\right)$ (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 diference 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.

See Specifications 3.6.F.2 for operation with one recirculation loop out of service.

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.

LIMITING SAFETY SYSTEM SETTING

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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.	2. At least one of the relief valves shall be disassembled and inspected each refueling outage.
a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.	
b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.	
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours.	3. With the reactor pressure $\geq$ 100 psig and turbine bypass flow to the main condenser, each relief valve shall be manually opened and verified open by turbine bypass valve position decrease and pressure switches and thermocouple readings downstream of the relief valve to indicate steam flow from the valve once per operating cycle.
E. <u>Jet Pumps</u>	E. <u>Jet Pumps</u>
1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:
	a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
	b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

LIMITING CONDITIONS FOR OPERATION

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 operated with one recirculation loop out of service provided that:

   a. MAPLHGR multipliers as indicated in section 3.12.A are applied.

   b. The power level is limited to maximum of 50% of rated power.

   c. The idle loop is isolated by electrically disconnecting the recirc. pump 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.

   d. The recirculation system controls will be placed in the manual flow control mode.

LIMITING SAFETY SYSTEM SETTING

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. Recirculation pump speeds shall be checked and logged at least once per day.

2. For one recirculation loop out of service the core plate delta p noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

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 thru 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 out of service (Sec. 3.12, Ref. 11). Therefore, continuous operation under such conditions is appropriate. The reactor may in any case be operated up to 24 hours with one recirculation loop out of service without 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 out of service.

During periods of Single Loop Operation (SLO), the out-of-service recirculation loop is isolated by electrically disarming the recirc. pump. This is done to prevent a cold water injection transient caused by an inadvertant pump start-up. It is permissible to leave the suction and discharge valves open during SLO to allow flow thru 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.

Core Plate ΔP oscillations in excess of these specified in Section 4.6.F.2 could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken.

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.

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 0.7. If at any time during reactor power operation (one or two loop) 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 out of service 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 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 Table 3.12-2. 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) 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 out of service, and cannot be returned to within prescribed limits within this 4 hour period the reactor shall be brought to 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 will not exceed the limit specified in 10CFR50.46.

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 and 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 11.

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 a safety limit MCPR and the largest ΔCPR.

2. MCPR Limits for Core Flows 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 Table 3.12-2 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 Table 3.12-2 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-21082-02-1A, Rev. 2, June 1982.

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.

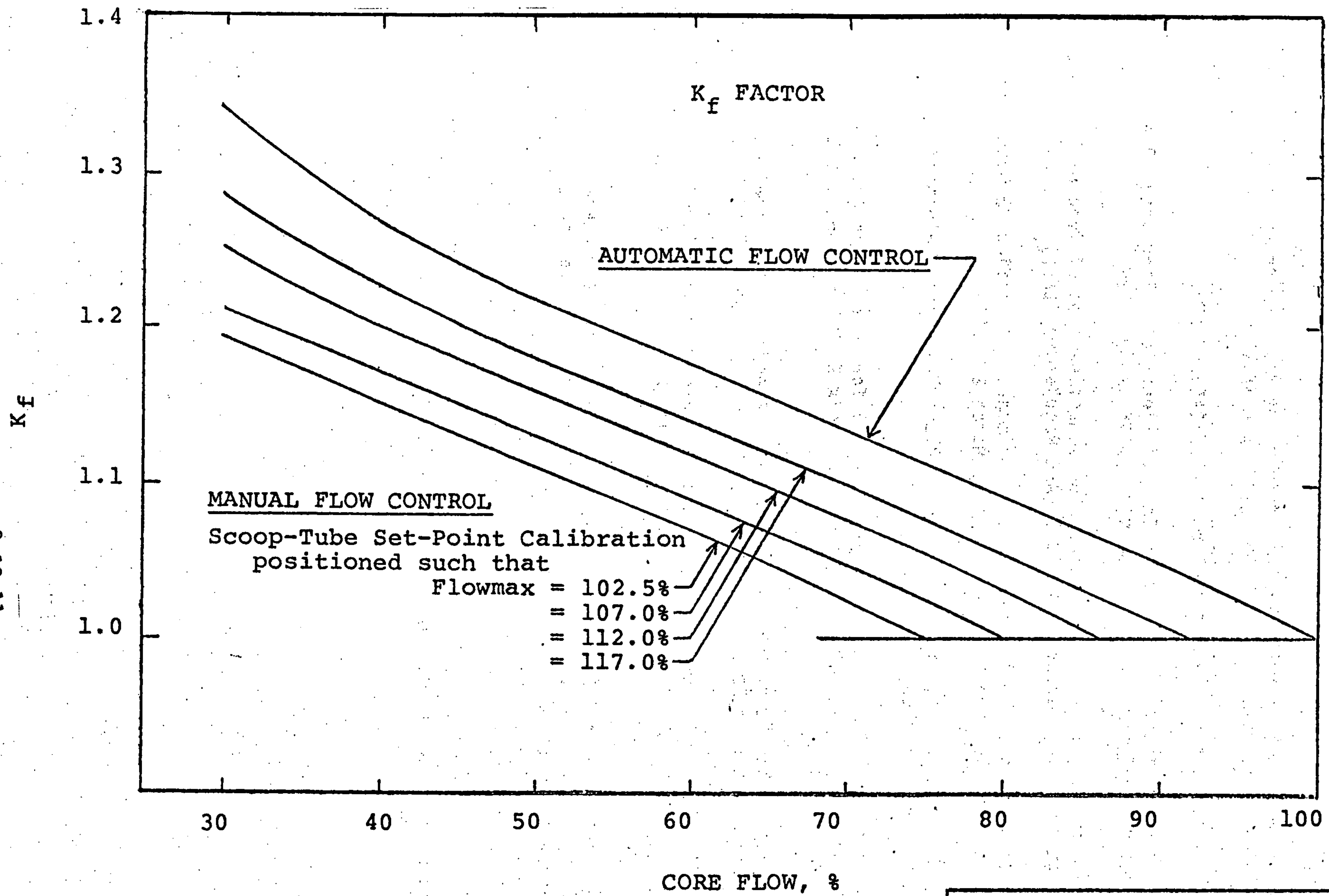
11. Duane Arnold Energy Center Single Loop Operation, NEDO-24272, July 1980.

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

TABLE 3.12-2

MCPR LIMITS

Fuel Type	For two recirculation loop operation	For one recirculation loop operation
8 x 8	1.25	1.28
8 x 8R/P8 x 8R	1.27	1.30



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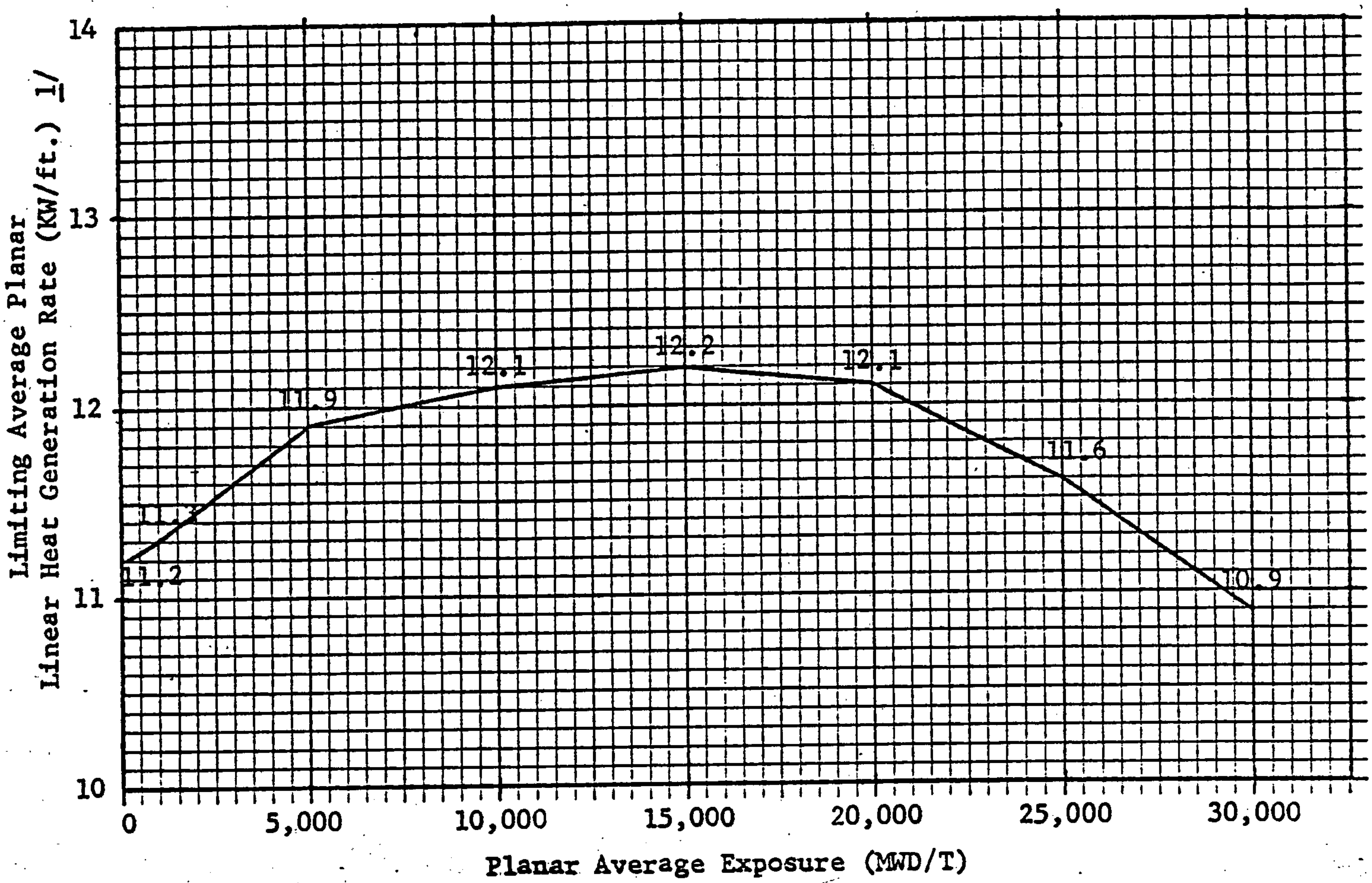
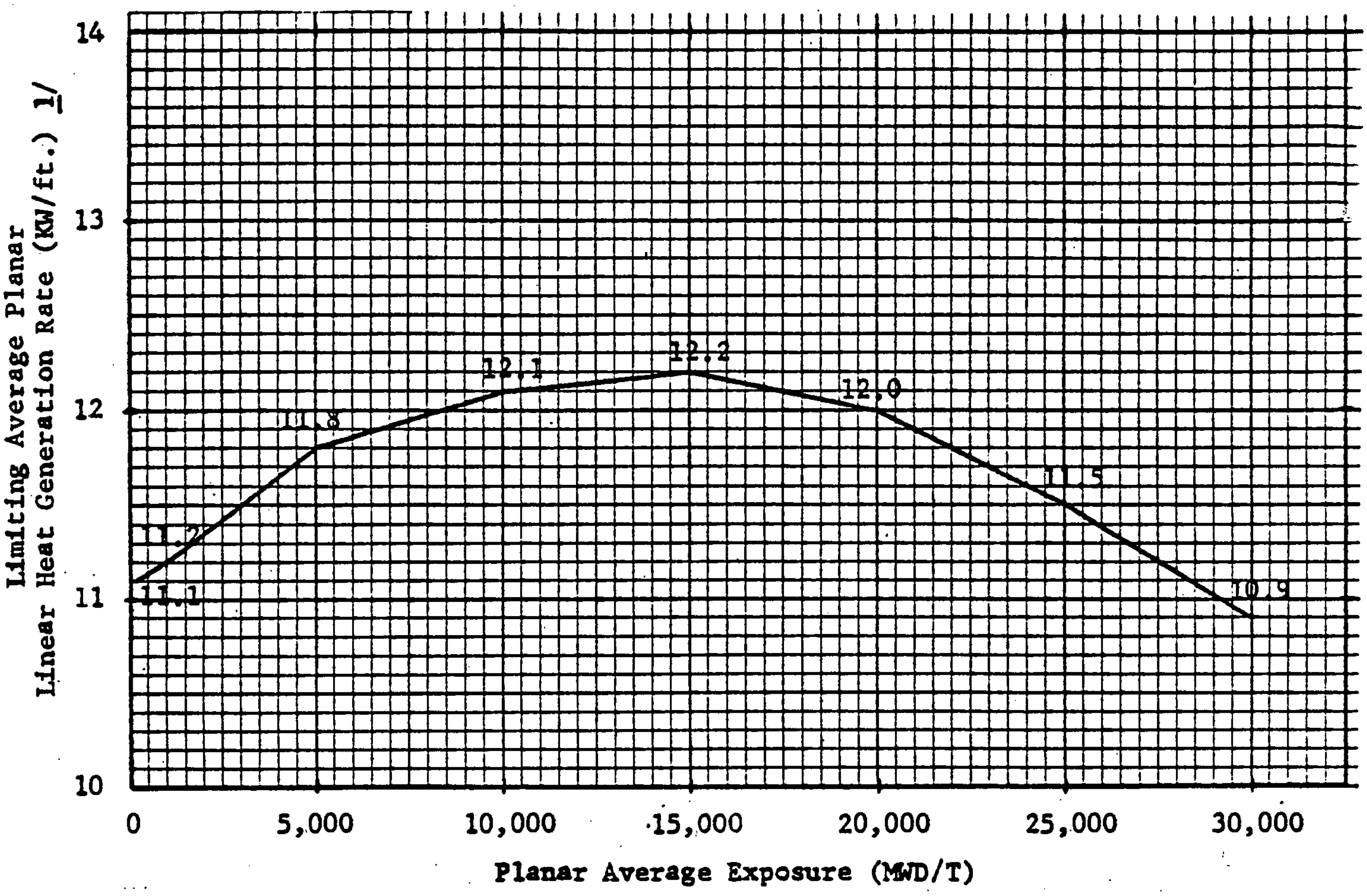
K_f AS A FUNCTION OF CORE FLOW

FIGURE 3.12-1



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.

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: 8D274L 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.

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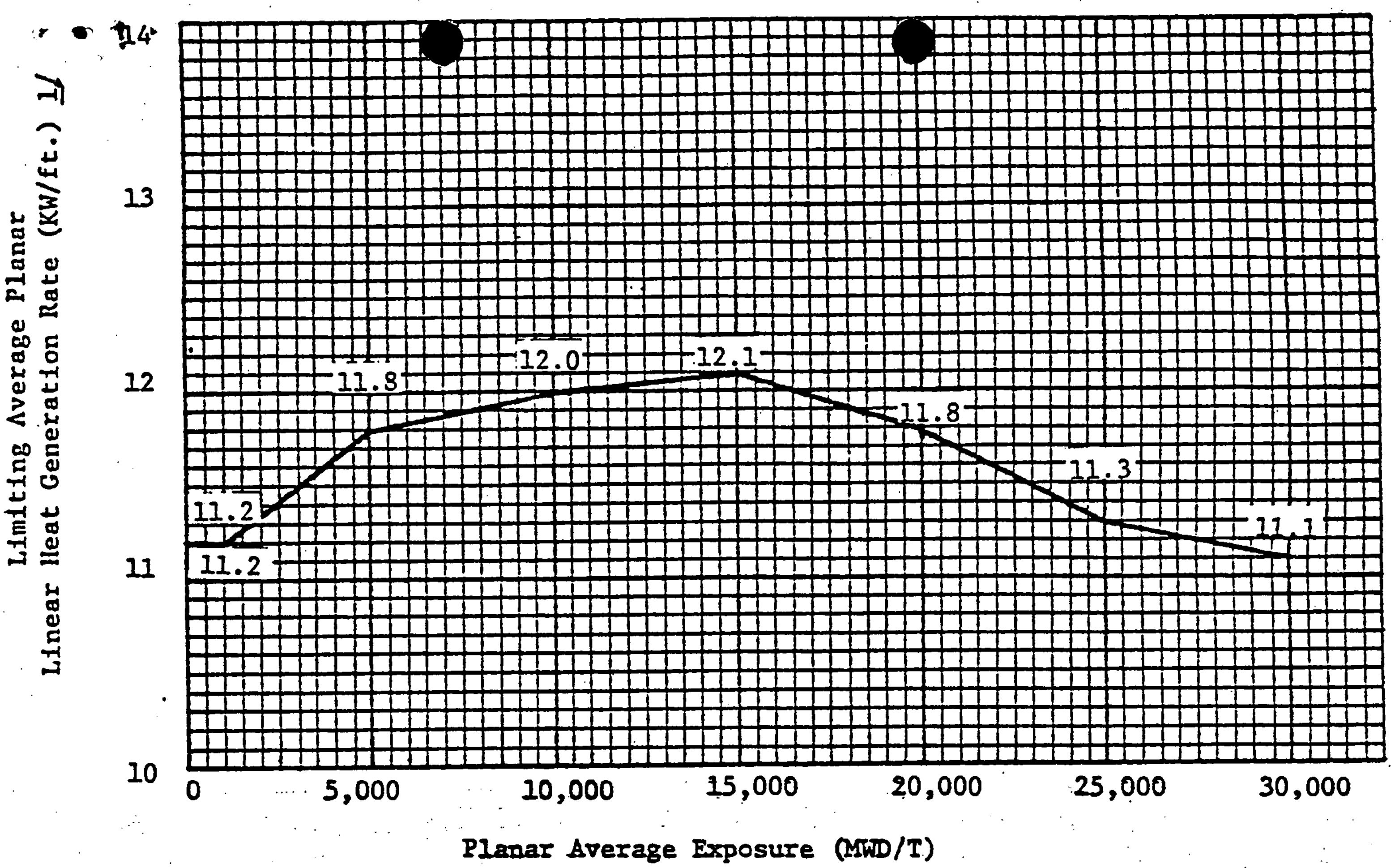
TECHNICAL SPECIFICATIONS

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

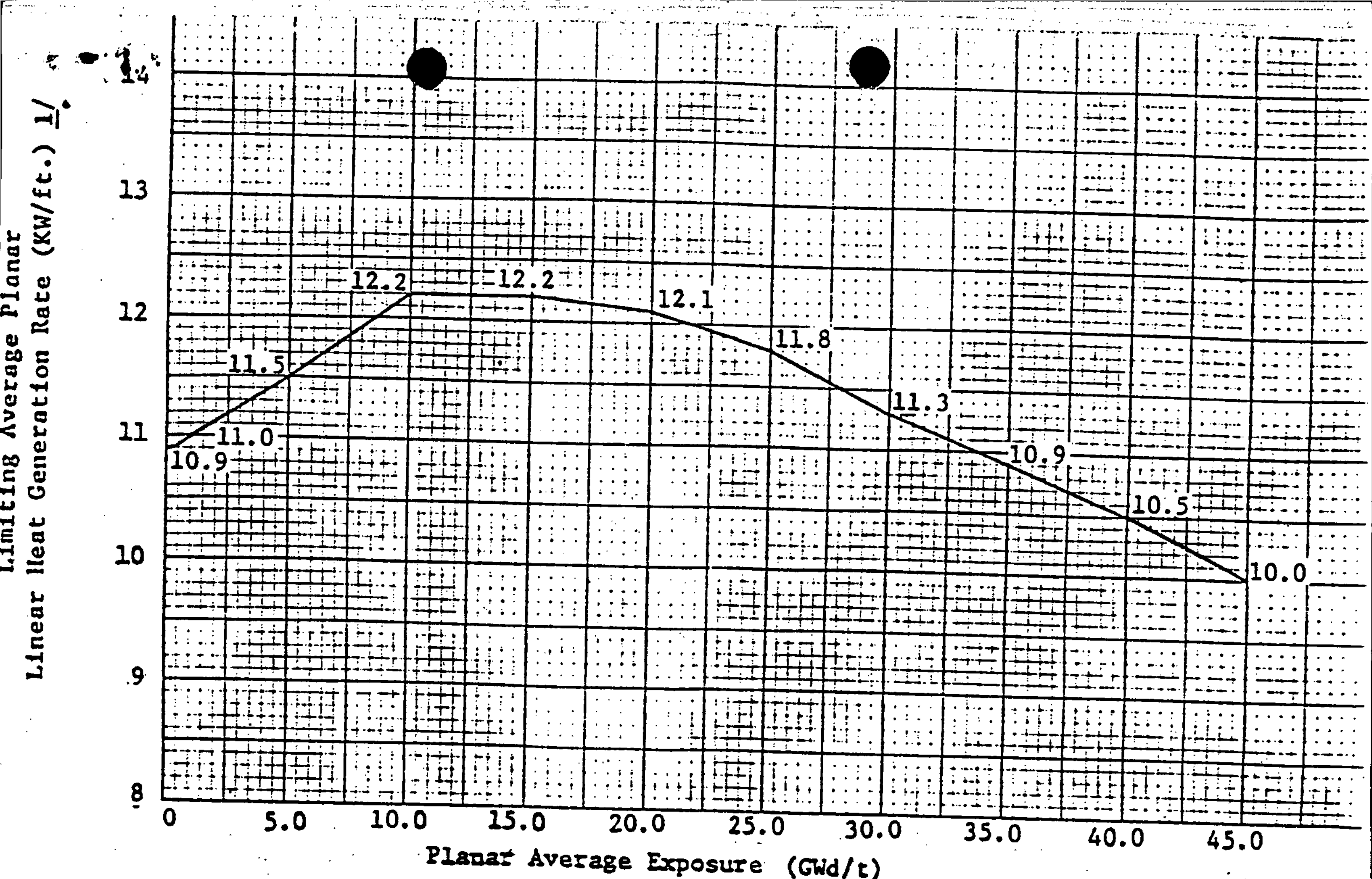
FUEL TYPE: 8D274H

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.

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: P8DP289 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.

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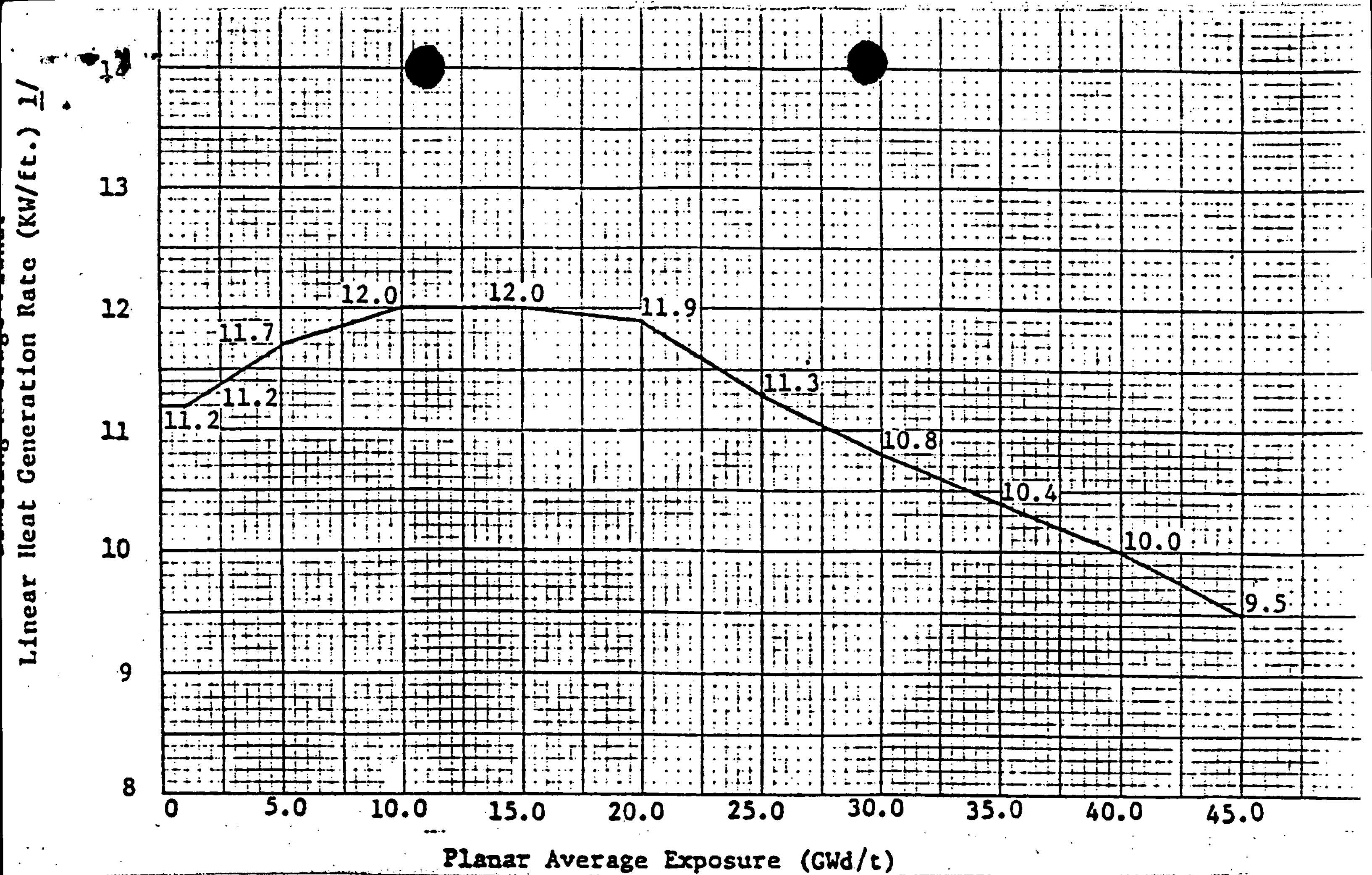
TECHNICAL SPECIFICATIONS

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

FUEL TYPE: P8DRB299

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.

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