

PROPOSED CHANGE RTS 124 TO
THE DUANE ARNOLD ENERGY CENTER
TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached, new pages. A list of the affected pages is included.

The current DAEC Technical Specifications do not allow plant operation beyond 24 hours if an idle recirculation loop cannot be returned to service. The ability to operate at reduced power with a single loop is highly desirable from availability/outage planning standpoint in the event that maintenance or component inavailability renders one loop inoperable. Such events have occurred three times during the current cycle and have caused the licensee to apply for temporary amendments, sometimes on an emergency basis. Therefore, the holders of this license propose that the Technical Specifications be revised as indicated in the attached pages to allow single-loop operation. Supporting analysis is given in NEDO-24272, which is reference 11 on P. 3.12-11 and a copy is enclosed.

The other change consists of renumbering pages to delete blank pages which were created in previous changes to the Technical Specifications.

Affected Pages

1.1-1	1.1-24*
1.1-2	1.1-25*
1.1-3	1.1-26*
1.1-5	1.1-27*
1.1-6	1.1-28*
1.1-7	3.2-16
1.1-8	3.6-7
1.1-9	3.6-29
1.1-10	3.12-1
1.1-11	3.12-3
1.1-12	3.12-4
1.1-13	3.12-5a*
1.1-14	3.12-6
1.1-15	3.12-7
1.1-16	3.12-8
1.1-17	3.12-9
1.1-18	3.12-11
1.1-19*	3.12-13
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1.1-23*	3.12-17
	3.12-5

* These pages have been deleted.

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 (1.08 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 Fig. 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

16.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)

LIMITING SAFETY SYSTEM SETTING

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:

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

$S \leq (0.66W + 50.7) \frac{FRP}{MFLPD}$ for one recirculation loop operation.

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR \leq 18.5 KW/ft (7x7 array) or 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:

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

The definitions used above for the APRM scram trip apply.

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

$$S \leq (0.66W + 42) \frac{FRP}{MFLPD} \text{ for two recirculation loop operation, and}$$

$$S \leq (0.66W + 38.7) \frac{FRP}{MFLPD} \text{ for one recirculation loop operation.}$$

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 513.5 inches above vessel zero (+12" 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 ≥ 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 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 0.01 for single-loop operation as discussed in Reference 2.

B. Core Thermal Power Limit (Reactor Pressure ≤ 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 28×10^3 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 28×10^3 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 following 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. "Generic Reload Fuel Application," NEDE-24011-P-A and NEDO-24011-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. 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 880 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 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 an 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. The 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 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. To protect the main condenser against over-pressure, 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

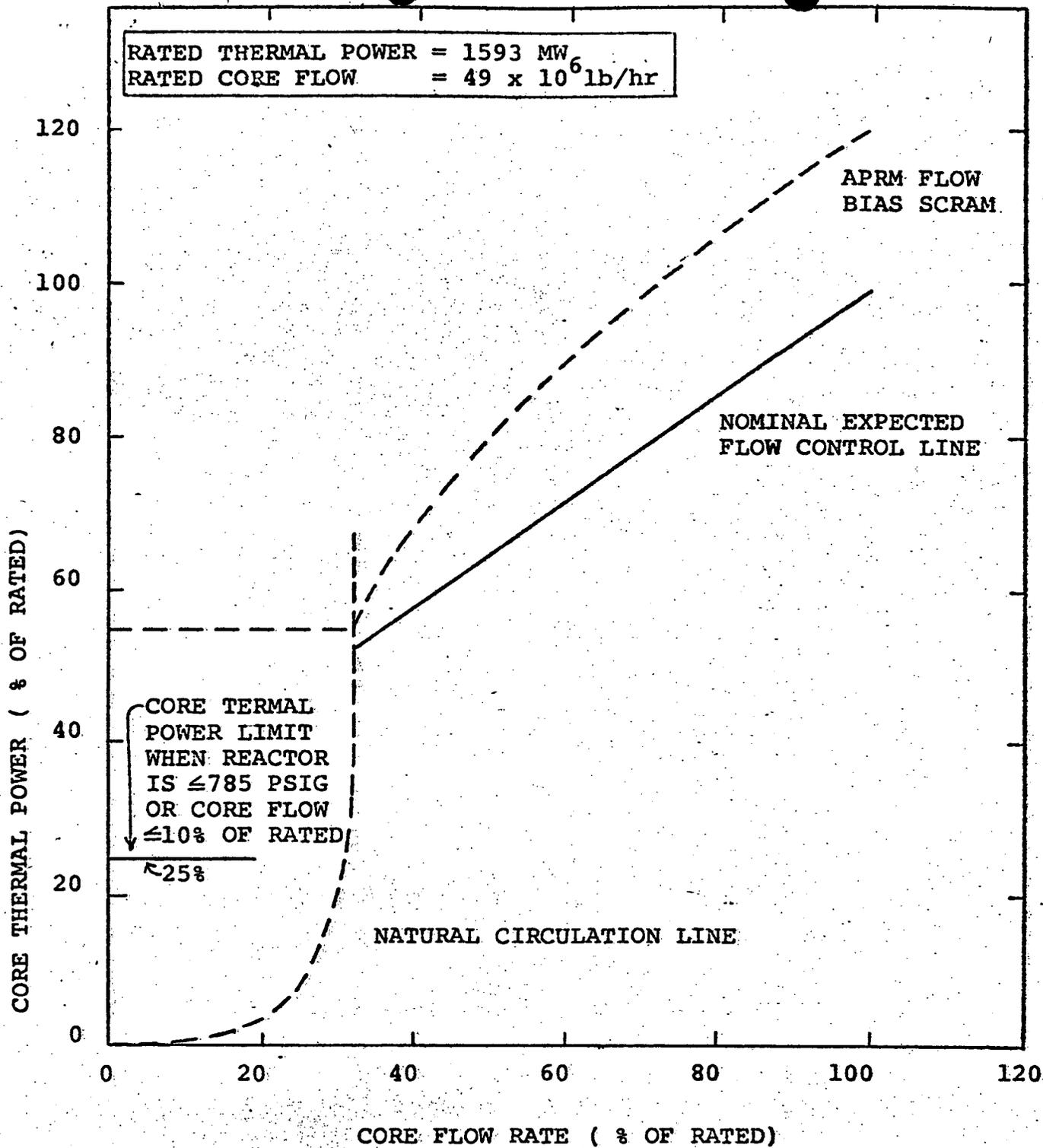
These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is

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. "Generic Reload Fuel Application," NEDE-24011-P-A* or NEDO-24011-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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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 $\leq \frac{(0.66W + 42)FRP}{MFLPD}$ (2)	6 Inst. Channels	(1)
		for 1 recirc loop operation $\leq \frac{(0.66W + 38.7)FRP}{MFLPD}$ (2)		
2	APRM Upscale (Not in Run Mode)	≤ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	≥ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	for 2 recirc loop operation $\leq \frac{(0.66W + 39)FRP}{MFLPD}$ (2)	2 Inst. Channels	(1)
		for 1 recirc loop operation $\leq \frac{(0.66W + 35.7)FRP}{MFLPD}$ (2)		
1 (7)	Rod Block Monitor Downscale	≥ 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)

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

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 with one recirculation loop out of service provided that:
 - a. MAPLHGR multipliers as indicated in section 3.12A are applied.
 - b. The power level is limited to maximum of 82% of licensed power.
 - c. The idle loop is isolated prior to startup, or if disabled during reactor operation, within 24 hours (suction valve closed and electrically disconnected).

- b. The indicated value of core flow rate varies from the value derived from 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 with 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.

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. 4). 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.

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 CONDITION FOR OPERATIONSURVEILLANCE REQUIREMENT3.12 CORE THERMAL LIMITSApplicability

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. 3.12-2, 3, 4, 5, 6, and 7. For single-loop operation, the values in these curves are reduced by multiplying by 0.86, 0.87 and 0.87 for 7x7, 8x8 and 8x8R fuel, respectively. If at any time during reactor power operation 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 two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

4.12 CORE THERMAL LIMITSApplicability

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

Objective

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

Specifications

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

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

LIMITING CONDITIONS FOR OPERATIONC. Minimum Critical Power Ratio (MCPR)

During reactor power operations, MCPR for two recirculation loop operation shall be \geq values as indicated in Table 3.12-2 at rated power and flow. If at any time during reactor power operation 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, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be \geq values as indicated in Table 3.12-2 times K_f , where K_f is shown in Figure 3.12-1.

For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.12.A, B or C are exceeded, a Reportable Occurrence report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

SURVEILLANCE REQUIREMENTSC. 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.

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 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 $\pm 20^{\circ}\text{F}$ 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 10 CFR Part 50, Appendix K limit.

For two recirculation loop operation the calculational procedure used to establish the MAPLHGR's shown on Figures 3.12-2 to 3.12-6, are documented in Reference 2.

Reduction factors for one recirculation loop operation were derived in Reference 4.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at \geq 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are

derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients⁽¹⁾. 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).

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 .

D. Reporting Requirements

The Limiting Conditions for Operation associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR and MCPR. It is a requirement, as stated in Specifications 3.12.A, B and C that if at any time during reactor power operation, it is determined that the limiting values for MAPLHGR, LHGR or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable.

DAEC - 1

TABLE 3.12-2

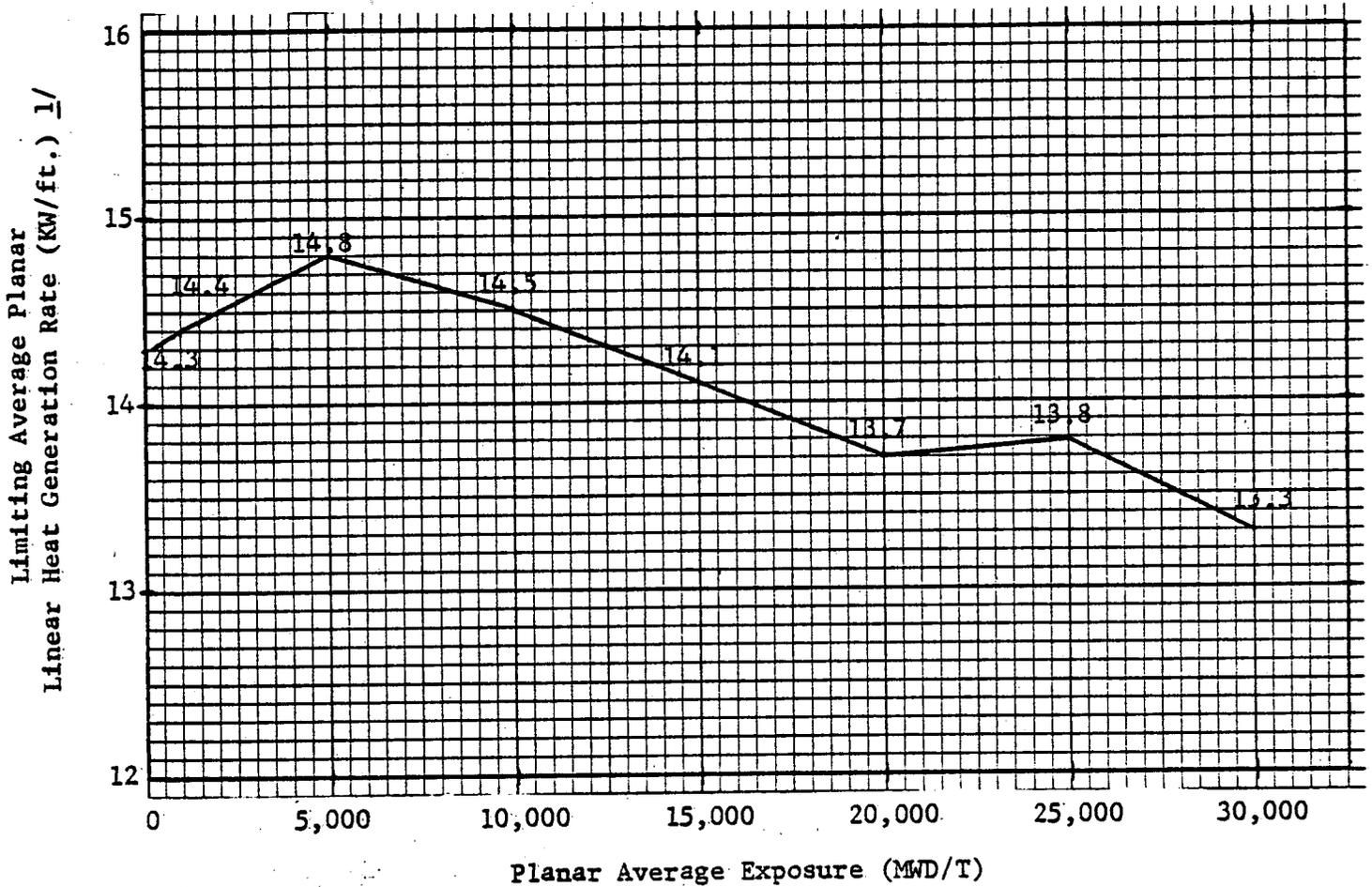
MCPR LIMITS

Fuel Type

7 x 7	1.25
8 x 8	1.24
8 x 8R	1.26

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.
2. "Generic Reload Fuel Application," NEDE-24011-P-A^{**}.
3. Current Reload Submittal for Duane Arnold Energy Center.
4. "Duane Arnold Energy Center Single Loop Operation", NEDO-24272 July 1980.



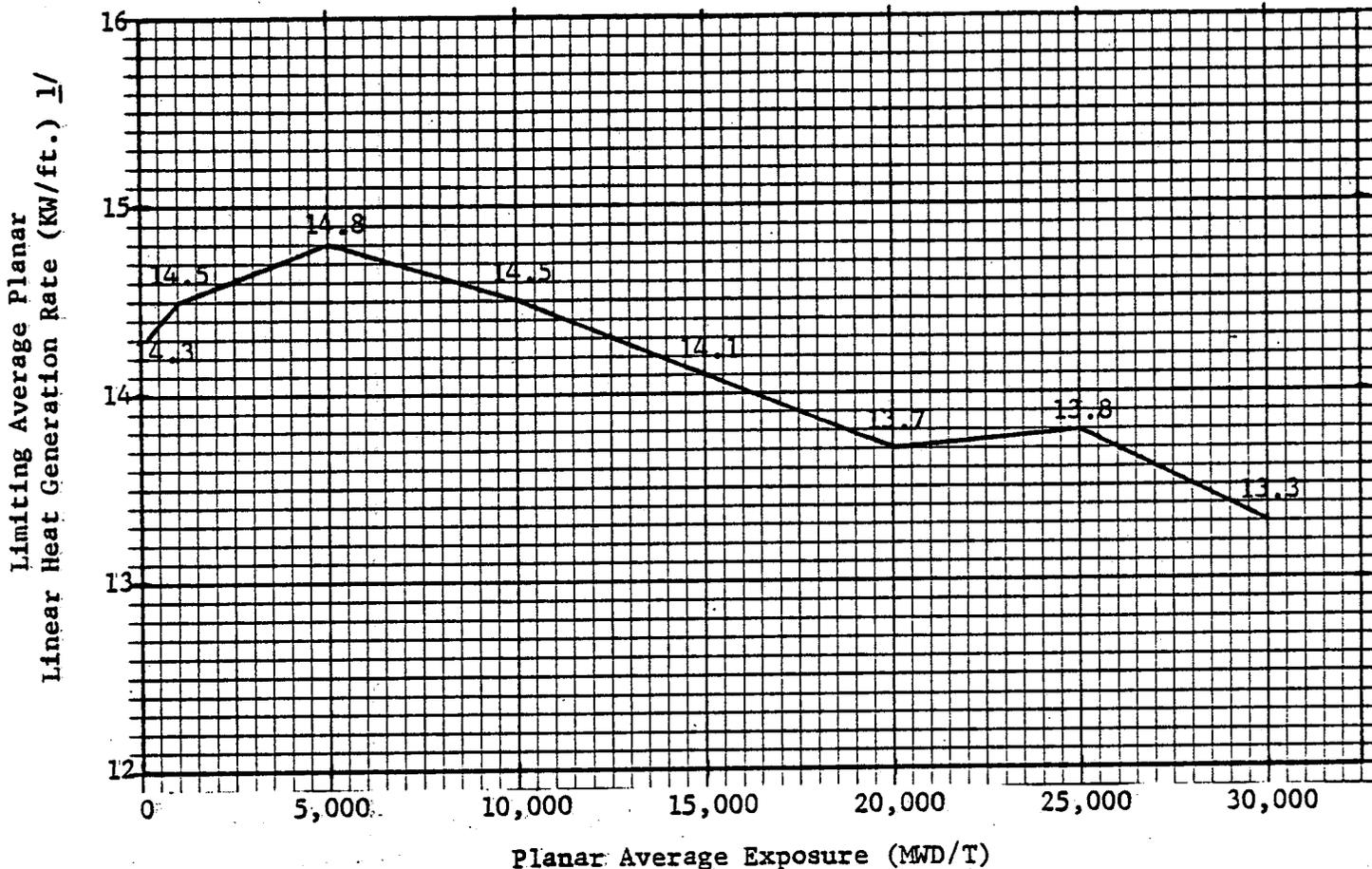
1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

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: INITIAL CORE TYPE 2

FIGURE 3.12-2



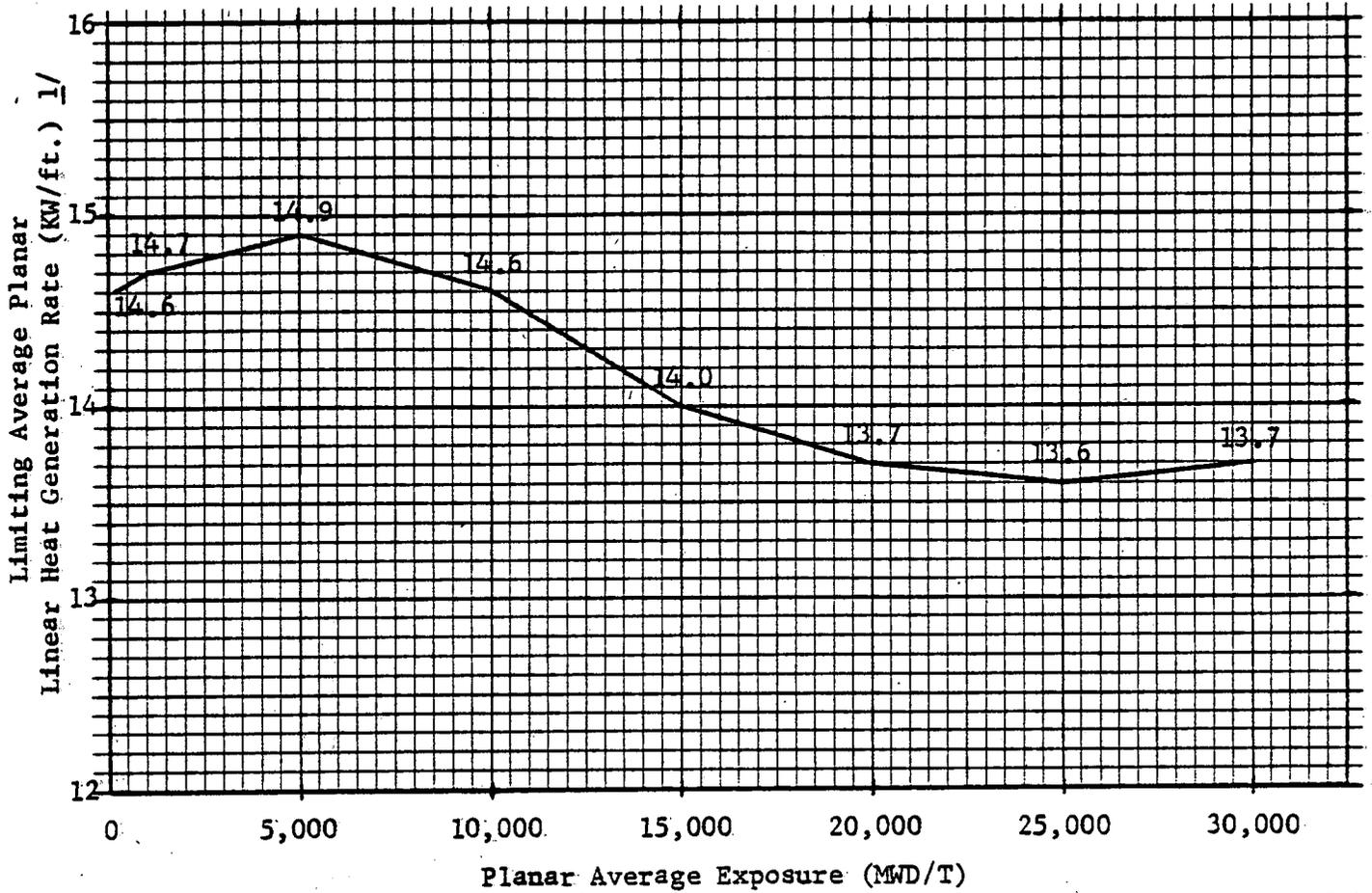
1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

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 AVERAGE EXPOSURE

FUEL TYPE: INITIAL CORE TYPE 3

FIGURE 3.12-3



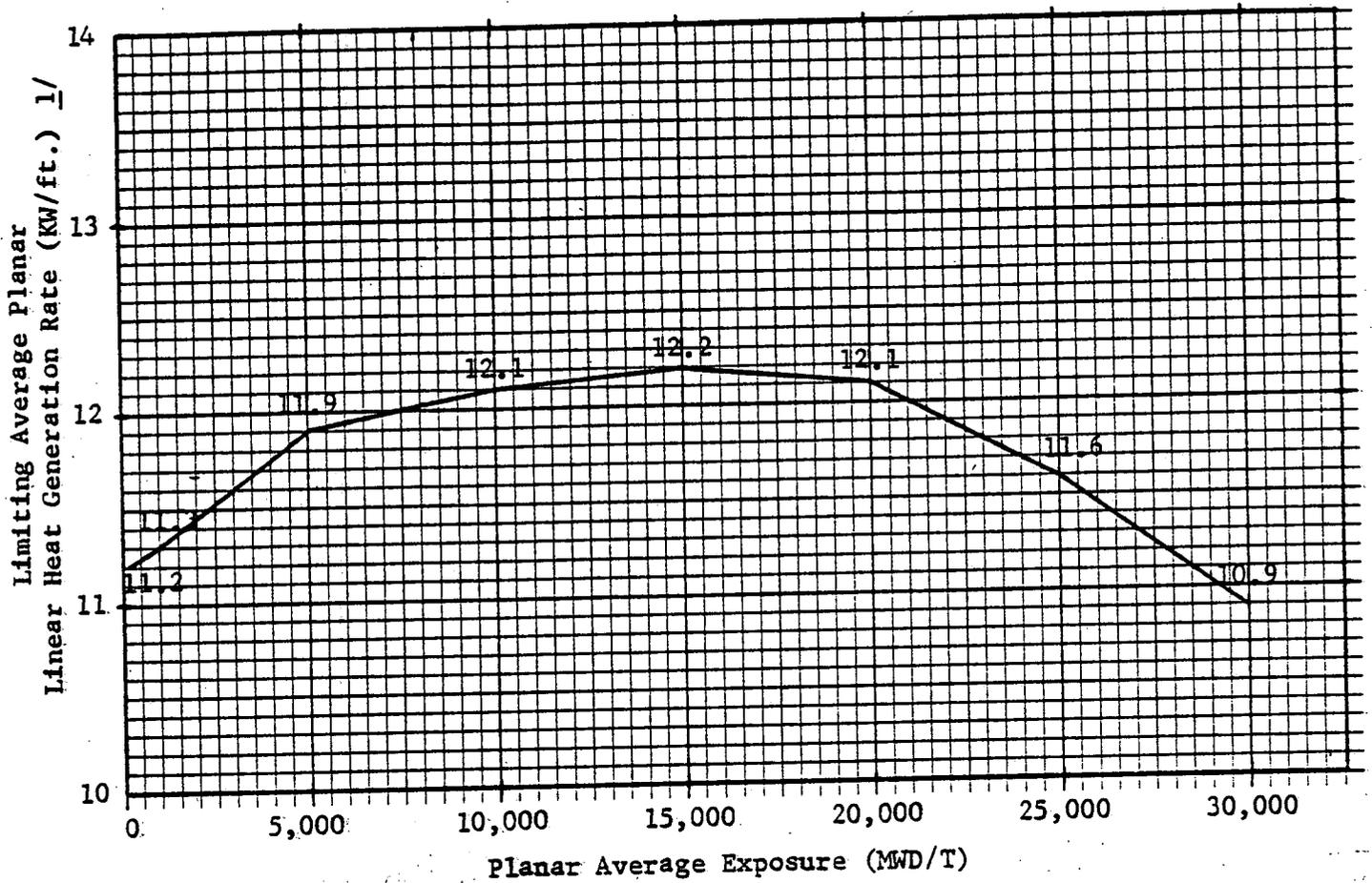
1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop were derived in Reference 4.

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FUEL TYPE: 7D230 TYPE 4

FIGURE 3.12-4



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 were derived in Reference 4.

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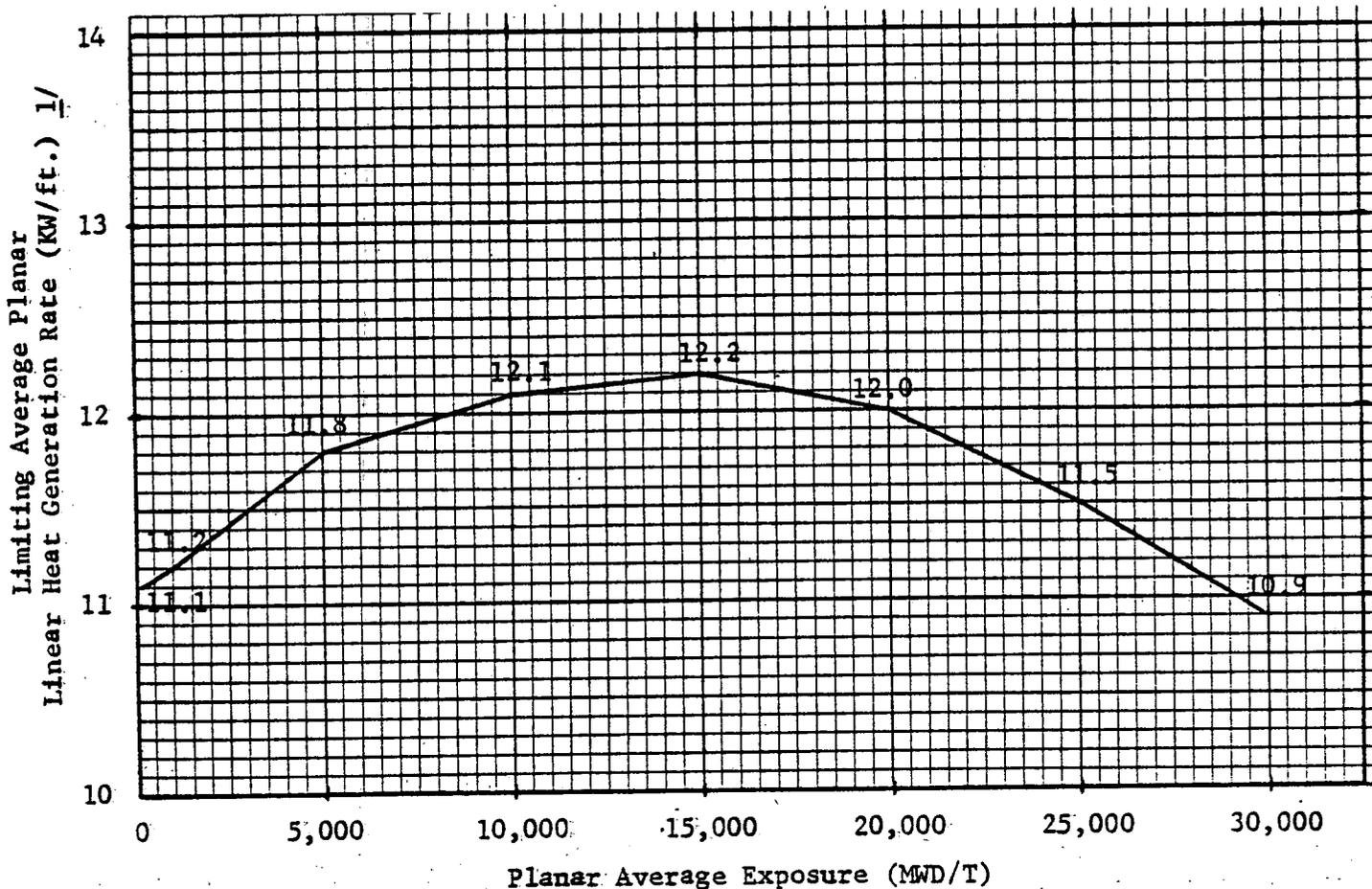
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TECHNICAL SPECIFICATIONS

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 were derived in Reference 4.

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LIMITING AVERAGE PLANAR LINEAR HEAT
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 AVERAGE EXPOSURE

FUEL TYPE: 8D274H

FIGURE 3.12-6