PROPOSED CHANGE RTS-74 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specifications 2.1.A, 3.6.F and 3.12.A specify certain limits on reactor operation concerning APRM High Flux Scram, APRM Rod Block, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and recirculation loop operability.

II. Proposed Change in Technical Specifications

The licensees of DPR-49 propose the following change in the Technical Specifications set forth in I above:

Delete portions of the Technical Specifications and replace with the attached sheets as appropriate.

III. Justification for Proposed Change

This change is proposed so that reactor operation can be continued for greater than 24 hours after one recirculation loop is made or found to be inoperable. The analysis and justification for single loop operation is contained in the attached document NEDO-21226, August 1976, "Duane Arnold Energy Center License Amendment Submittal for Single-Loop Operation with the Bypass Flow Holes Plugged".

IV. Review Procedures

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

TSTECH

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 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.a. APRM High Flux Scram When In Run Mode.

For operation with a peaking factor less than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), 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	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.

* 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

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.

MTPF = Actual Maximum Total peaking factor.

For a peaking factor greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram setpoint shall be:

$$S \leq (0.66\, W + 54) \frac{(\ *\ )}{MTPF}$$

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR ≤ 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array) and MCPR ≥ 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 total peaking factor greater than * 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.

b. APRM High Flux Scram When In Run Mode and Single Loop Operation.

For single loop operation with a peaking factor less than or equal to 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram trip setpoint shall be:

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

$S \leq (0.66\ W + 50.7)$

with a maximum setpoint of 116.7 rated power at 100% rated recirculation flow or greater.

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.

MTPF = Actual Maximum Total peaking factor.

For a peaking factor greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram setpoint shall be:

$$S \leq (0.66\ W + 50.7) \frac{(\ *\ )}{MTPF}$$

NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 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, operation is not allowed with total peaking factor greater than * 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.

* 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	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. 3.a. APRM Rod Block When in Run Mode For operation with a peaking factor less than or equal to 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 42)$ The definitions used above for the APRM scram trip apply. For a peaking factor greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 42) \frac{(\ *\ )}{MTPF}$ 3.b. APRM Rod Block When in Run Mode and Single Loop Operation For single loop operation with a peaking factor less than or equal to 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 38.7)$ The definitions used above for the APRM scram trip apply. For a peaking factor greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 38.7) \frac{(\ *\ )}{MTPF}$

* 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING		
	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	≥ 514.5 inches above vessel zero (+12" on level instruments)
	C.	Scram - turbine stop valve closure	≤ 10 percent valve closure
	D.	Turbine control valve fast closure scram shall occur within 30 milliseconds of the start of turbine control valve fast closure.	
	E.	Scram - main steam line isolation valve	≤ 10 percent valve closure
	F.	Main steam isolation valve closure nuclear system low pressure	≥ 880 psig
	G.	Core spray and LPCI actuation-reactor low water level	≥ 363 inches above vessel zero (-139.5 inches indicated level)
	H.	HPCI and RCIC actuation - reactor low water level	≥ 464 inches above vessel zero (-38.5 inches indicated level)
	I.	Main steam isolation valve closure-reactor low water level	≥ 464 inches above vessel zero (-38.5 inches indicated level)
	J.	Main steam isolation valve closure-loss of main condenser vacuum	≤ 10 inches Hg vacuum

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 MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1, when the maximum total peaking factor is greater than the design value. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope 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 $\geq$ 1.07 when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

An evaluation of operation with one recirculation loop out of service demonstrates that the scram trip setpoint must be modified to assure Safety Limits are not violated. This evaluation is presented in reference (2).

2. APRM High Flux Scram (Refuel or Startup and 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 1.07. 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 total peaking factor exceeds the design value, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

An evaluation of operation with one recirculation loop out of service demonstrates that the APRM Rod Block trip setpoint must be modified to assure that a MCPR less than the Limiting Condition for Operation does not occur. This evaluation is presented in reference (2).

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 1, the scram setting would be 120 divisions for that range; likewise, 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.

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. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

2. Duane Arnold Energy Center License Amendment Submittal for Single-Loop Operation with the Bypass Flow Holes Plugged, NEDO-21226, August, 1976.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
2	IRM High Flux	≤ 120/125 of Full Scale	X	X	(5)	6 Instrument Channels	A
2	IRM Inoperative		X	X	(5)	6 Instrument Channels	A
2	APRM High Flux	See Specification 2.1.A.1.a or 2.1.A.1.b			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

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

10. To be considered operable, APRM's A, B, C and D must have at least 9 LPRM inputs while APRM's E and F must have at least 13 LPRM inputs. Additionally each APRM must have at least 2 LPRM inputs per level.

11. Deleted

12. Deleted

13. The design permits closure of any two lines without a scram being initiated.

14. The trip setting and alarm setting for the Main Steam Line High Radiation Monitor shall be $\leq$ 6 X and $\leq$ 3 X, respectively, Normal Rated Power Background during the period prior to achieving 50 per cent rated power for the first time.

NOTES FOR TABLE 3.2-B

1. Whenever any CSCS subsystem is required by Subsection 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument setpoint corresponds to 18.5" above the top of active fuel.

5. HPCI has only one trip system for these sensors.

6. The relay drop-out voltage will be measured once per operating cycle and the data examined for evidence of relay deterioration. 15

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)	See Specification 2.1.A.3.a or 2.1.A.3.b	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)	$\leq (0.66W + 41) (\frac{*}{TPF})$ (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 (7)	Rod Block Monitor (Flow Biased) (Single Loop Operation)	$\leq (0.66W + 37.7) \frac{(*)}{(TPF)}$ (2)	2 Inst. Channels	(1)

* 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

DAEC-1

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	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. <u>Jet Pump Flow Mismatch</u>	F. <u>Jet Pump Flow Mismatch</u>
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. From and after the date that one recirculation loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 8 hours unless the modifications to APRM High Flux Scram, APRM Rod Block Monitor and MAPLHGR's as specified in Specifications 2.1.A.1.b, 3.1.A.3.b and 3.12.A are made.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
If these requirements cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. 3. Following 1-pump operation, the discharge valve of the lower speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed and the requirements of Specification 3.6.A.4 and 3.6.A.5 are met. G. Structural Integrity The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standard throughout the life of the plant.	G. Structural Integrity 1.a. Nuclear Class I Components - Components within the reactor coolant pressure boundary (as defined in Article IS-120 of the ASME Boiler and Pressure Vessel Code) will be pressure tested prior to startup following each reactor refueling outage. During the pressure test, components will be inspected for leakage without removal of insulation. The test pressure and temperature will be maintained for at least four hours prior to the final leakage inspection. The test pressure will not be less than the system nominal operating pressure at 100% rated reactor power. The pressure test will be conducted at a vessel temperature above the nil ductility temperature of the vessel. Near the end of each inspection interval, one system pressure test will be upgraded to a system hydrostatic pressure test. The hydrostatic test will be identical with the pressure test, except that the minimum test pressure will be higher and the test will be witnessed by an authorized inspector. The test pressure will not be less than 1.08 times the system nominal operating pressure as required by Subsection IS-522 of the Winter 1972 Addenda to Section XI of the ASME Boiler and Pressure Vessel Code. b. Nuclear Class II Components - Near the end of each inspection interval, the following systems (as defined in Subsection ISC-261 of the Winter 1972 Addendum to Section XI of the ASME Boiler and Pressure

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 of ECCS performance and transient analyses has been made for single loop operation (Ref. 2). This evaluation shows that with modifications to APRM High Flux Scram, APRM Rod Block and MAPLHGR's, continuous operation may be allowed. The short period of time allowed to operate without setpoint changes permits appropriate corrective action to 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 CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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. 3.12-2, 3.12-3, 3.12-4 and 3.12-5. During periods of single loop operation the limiting value shall be no more than 0.86 times the values shown in Figures 3.12-2, 3.12-3, 3.12-4 and 3.12-5. 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 LIMITS

Applicability:

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

Objective

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

Specifications

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

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

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 the 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 duel 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. The limiting values for MAPLHGR's are shown in Figures 3.12-2, 3.12-3, 3.12-4 and 3.12-5.

The calculational procedure used to establish the MAPLHGR's shown on Figures 3.12-2, 3.12-3, 3.12-4 and 3.12-5 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis as compared to previous analyses performed using the methods in Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figures 3.12-2, 3.12-3, 3.12-4 and 3.12-5; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations. The procedures used to determine the MAPLHGR's for use during single loop operation are described in Reference (9).

(35)

Iowa Electric Light and Power Company

June 24, 1983
NG-83-0671

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Subject: Single Recirculation Loop Operation

Dear Mr. Denton:

In accordance with the requirements of 10 CFR 50.59 and 50.90, we transmitted our proposed technical specification change regarding single recirculation loop operation on October 17, 1980, which was subsequently amended in our December 18, 1981 transmittal. We hereby amend that application and amendment with the enclosed technical specification page changes, which are intended to supersede those previously submitted.

This amendment has been reviewed by the Duane Arnold Energy Center Operations Committee and the Safety Committee. A check for $4,000 was submitted with our original application and, therefore, no further fee is required.

Three signed and 37 additional copies of this application are transmitted herewith. Pursuant to the requirements of 10 CFR 50.91, a copy of this application and analysis of no significant hazards considerations is being sent to our appointed state official. This application, consisting of the foregoing letter and enclosures, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Richard W McGaughy
Richard W. McGaughy
Manager, Nuclear Division

~~8306270420~~

Subscribed and sworn to Before Me on this 24th day of June 1983.

Kathleen M. Herlier
Notary Public in and for the State of Iowa

RWM/RAB/dmh*
Attachment
cc: R. Browning
L. Liu
S. Tuthill
F. Apicella
T. Houvenagle
NRC Resident Office

Evaluation of Change
With Respect to 10 CFR 50.92

The enclosed application is judged to involve no significant hazards based upon the following information:

i) The enclosed application is for permanent licensing of Single Loop Operation (SLO), i.e. operation with one recirculation loop out of service. The NRC, by granting Iowa Electric temporary license amendments in the past for SLO, has determined that such operation does not involve a significant hazards consideration. As the present application follows the guidelines previously established by the NRC for SLO, the enclosed amendment is therefore judged to involve no significant hazards as well.

ii) The enclosed amendment request is judged to involve no significant hazards based upon the precedent set by the NRC is granting similar requests for other Operating Reactors.

iii) In the April 6th 1983 Federal Register the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example number four of that list states:

> "A relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in the satisfactory way that the criteria have been met."

As the NRC has previously established the criteria for SLO and the enclosed application follows those criteria, the amendment request is judged to fall within the scope of the above example.

Revision 2 to Proposed Change RTS-124
to the
Duane Arnold Energy Center
Technical Specifications

The purpose of this change to the previous revision, RTS-124A submitted December 18, 1981, is to update the submittal for permanent licensing of Single Loop Operation (SLO) to be consistent with the Reload Licensing amendment for Cycle 7 (Amendment 88, issued April 25, 1983). The major changes deal with the increased Minimum Critical Power Ratio (MCPR) operating limits for 8 X 8 and P8 X 8R fuel and new Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) curves for the fresh fuel for Cycle 7. Also this revision reflects the new NRC position which allows isolation of the out-of-service recirc. loop by electrically isolating the recirc. pump instead of closing the suction valve, as proposed in the previous submittal. This change allows the temperature in the idle loop to be maintained, reducing the thermal stress on the pipe. Several administrative changes are also included. dealing with updating references and deletion of blank pages. For clarity and conciseness, a list of the affected pages is included which supersedes those previously submitted.

The changes being made are as follows:

1) Change the List of Tables on page vi to show that Table 3.12-2 is moved to page 3.12-10.

2) Change 1.1.A to include the safety limit MCPR for SLO.

3) Add the APRM Flow-Biased Flux Scram and Rod Block Equations for SLO to Sections 2.1.A.1 and 2.1.A.3 respectively as well as Figure 2.1-1 and to Tables 3.1-1 and 3.2-C (pages 1.1-19, 3.1-3 and 3.2-16). Also add the SLO-modified Rod Block Monitor (RBM) equation to Table 3.2-C.

4) Add Section 2.1.A.4 detailing APRM Flux noise surveillance requirements for SLO, and add supporting Bases to page 1.1-3.

5) Update the bases for Section 1.1.A (page 1.1-6) and 2.1 (page 1.1-9) for SLO.

6) Update the References for Section 1.1, 2.1 and 3.12 to include the SLO analysis for DAEC and to show the new title of the GE Reload Fuel Licensing Topical Report.

7) Paragraph 4.6.E.b has been moved from page 3.6-7 to page 3.6-6.

8) Change Sections 3.3.E and 3.6.F.2 to allow SLO for more than 24 hours and add the necessary Limiting Condition for Operation (LCO).

9) Add the core plate ΔP surveillance requirement to Section 4.6.F.2 and supporting bases to page 3.6-34.

10) Change the bases for Sections 3.6 and 4.6 to allow SLO and to reference the appropriate supporting analysis. Also, clarify the bases on jet pumps (section 3.6.E) and out-of-service loop isolation (section 3.6.F) for SLO.

11) Add the core thermal limits LCO's for SLO to section 3.12.A and 3.12.C.

12) Update the bases for Section 3.12 to support SLO and to reference the SLO analysis.

13) Consolidate the text on page 3.12-5a onto 3.12-6.

14) Update the MCPR limits given in Table 3.12-2 to reflect the increased operating limits for Cycle 7 and to show the comparable limits for SLO.

15) Add required reference for SLO to footnote on Figures 3.12-5 through 3.12-9.

16) The following pages are renumbered as a result of deleting blank pages:

1.1-6	1.1-18
1.1-7	1.1-19
1.1-8	3.12-6
1.1-9	3.12-8
1.1-10	3.12-9
1.1-11	3.12-10
1.1-12	3.12-12
1.1-13	3.12-13
1.1-14	3.12-14
1.1-15	3.12-15
1.1-16	3.12-16
1.1-17	

Affected Pages

vi	3.2-16
vii	3.3-7
1.1-1	3.6-6
1.1-2	3.6-7
1.1-3	3.6-31
1.1-5	3.6-34
1.1-6	3.12-1
1.1-7	3.12-3
1.1-8	3.12-4
1.1-9	3.12-5a*
1.1-10	3.12-6
1.1-11	3.12-7
1.1-12	3.12-8
1.1-13	3.12-9
1.1-14	3.12-9a*
1.1-15	3.12-10
1.1-16	3.12-11
1.1-17	3.12-12
1.1-18	3.12-13
1.1-19	3.12-14
1.1-20*	3.12-15
1.1-21*	3.12-16
1.1-22*	3.12-17*
1.1-23*	3.12-18*
1.1-24*	3.12-19*
1.2-7	3.12-20*
3.1-3	

*These pages have been deleted.

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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 ≥ 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:

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

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

   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. 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 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. <u>Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC, Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps</u>

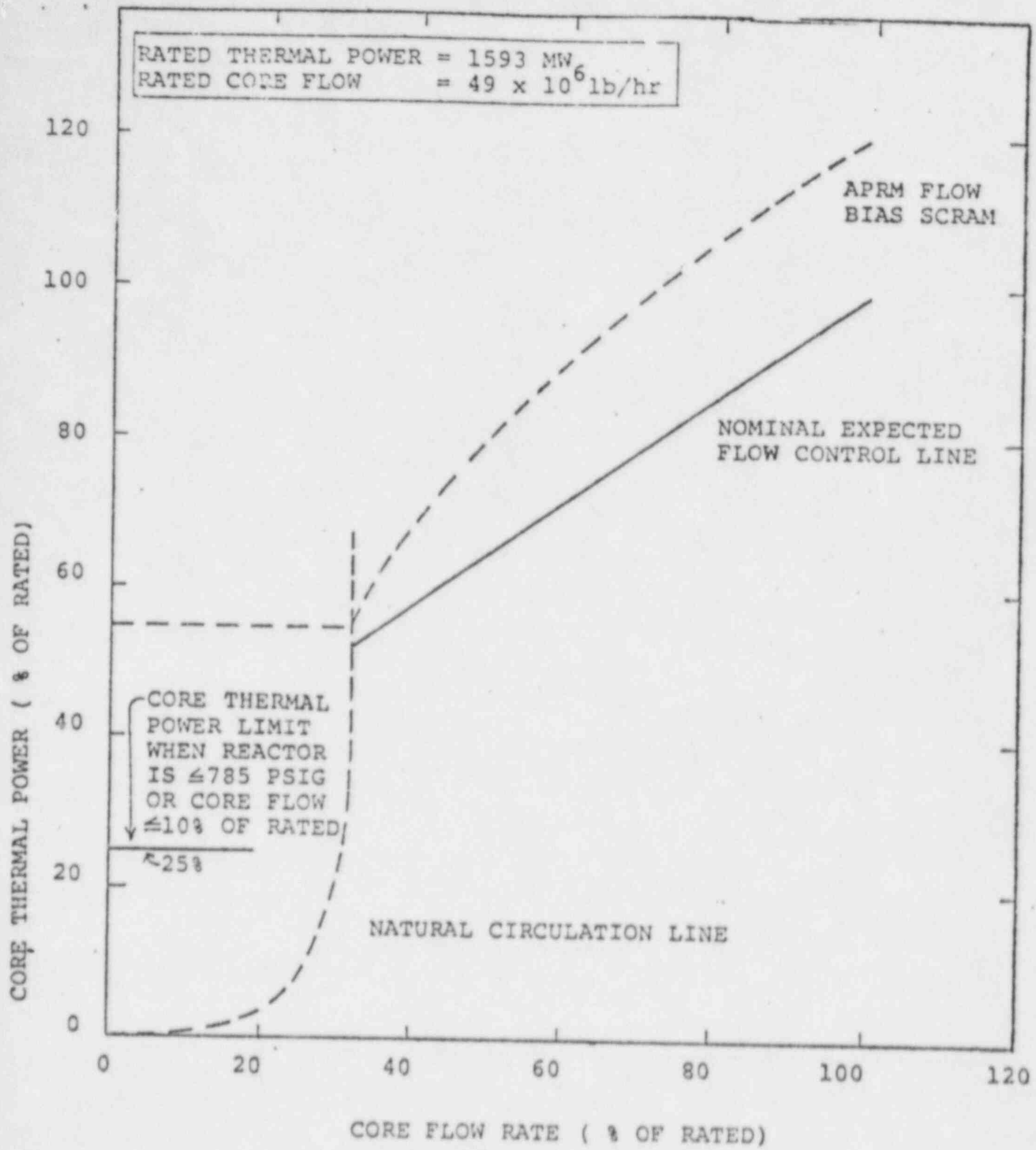
These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses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

2.1 REFERENCES

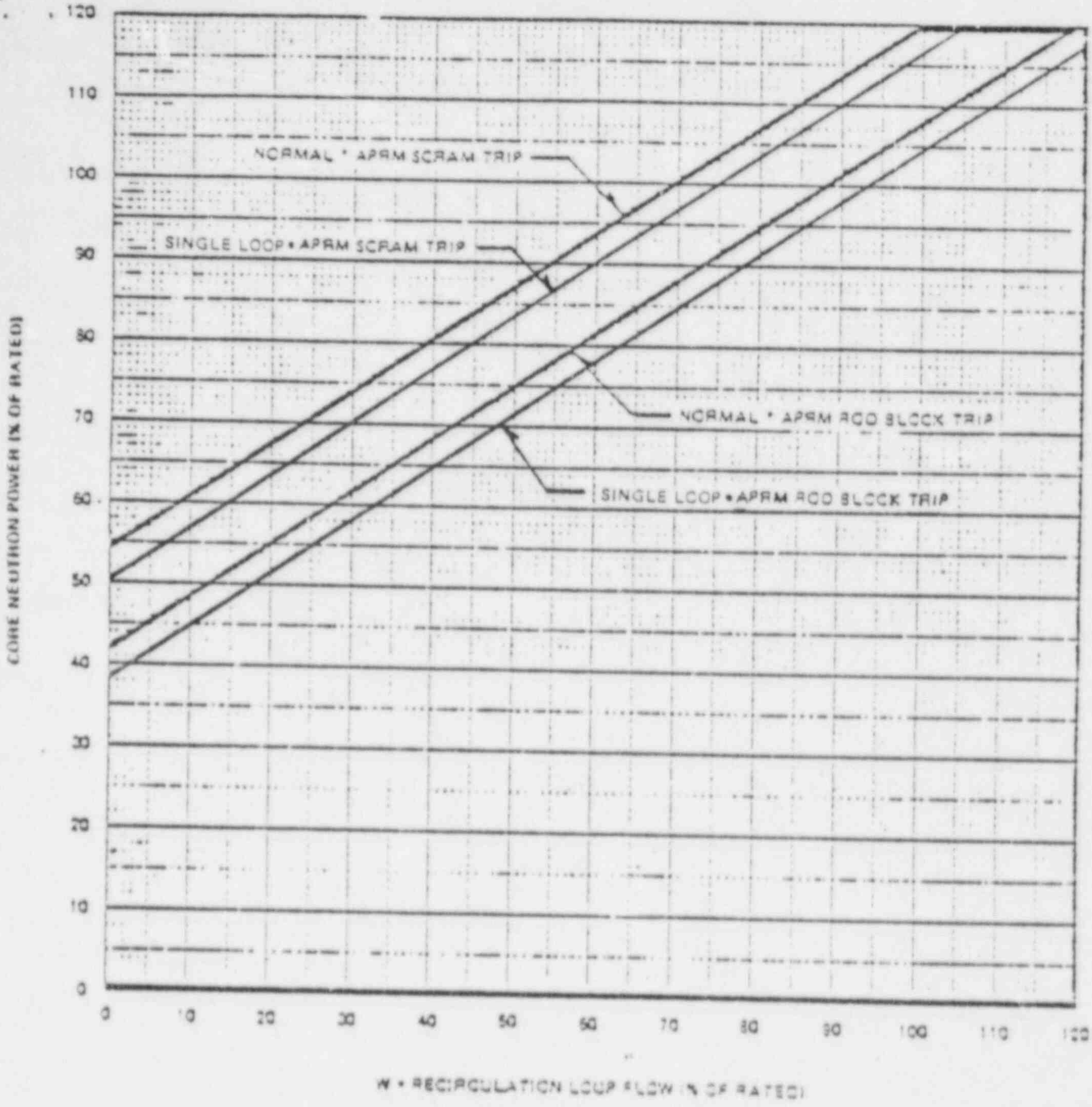
1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A*

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
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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			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
2	IRM High Flux	$\leq$ 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 $\leq$(.66W+54)(FRP/MFLPD) (11) (12) for one recirc loop operation $\leq$(.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	$\geq$ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	$\leq$ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	$\leq$ 1035 psig	X(8)	X	X	4 Instrument Channels	A

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 $\leq (0.66\ W + 42)\ (\frac{FRP}{MFLPD})^{(2)}$ 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	LIMITING SAFETY SYSTEM SETTING

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.

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. Jet Pumps	E. Jet Pumps
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	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. 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.

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 $\leq$ 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 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 $\geq$ 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)	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.	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.

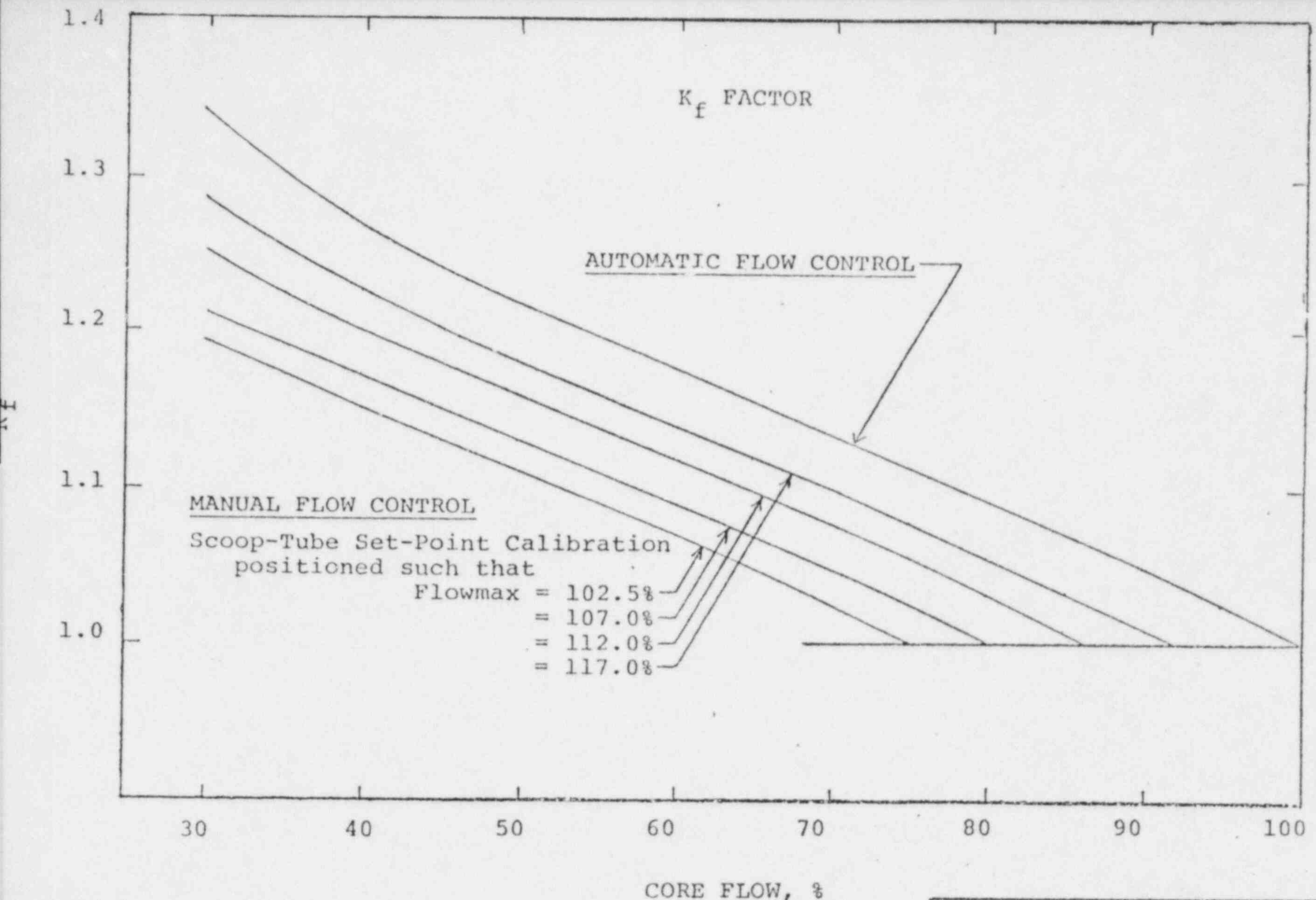
11. Duane Arnold Energy Center Single Loop Operation, NEDO-24272, July 1980.

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

TABLE 3.12-2

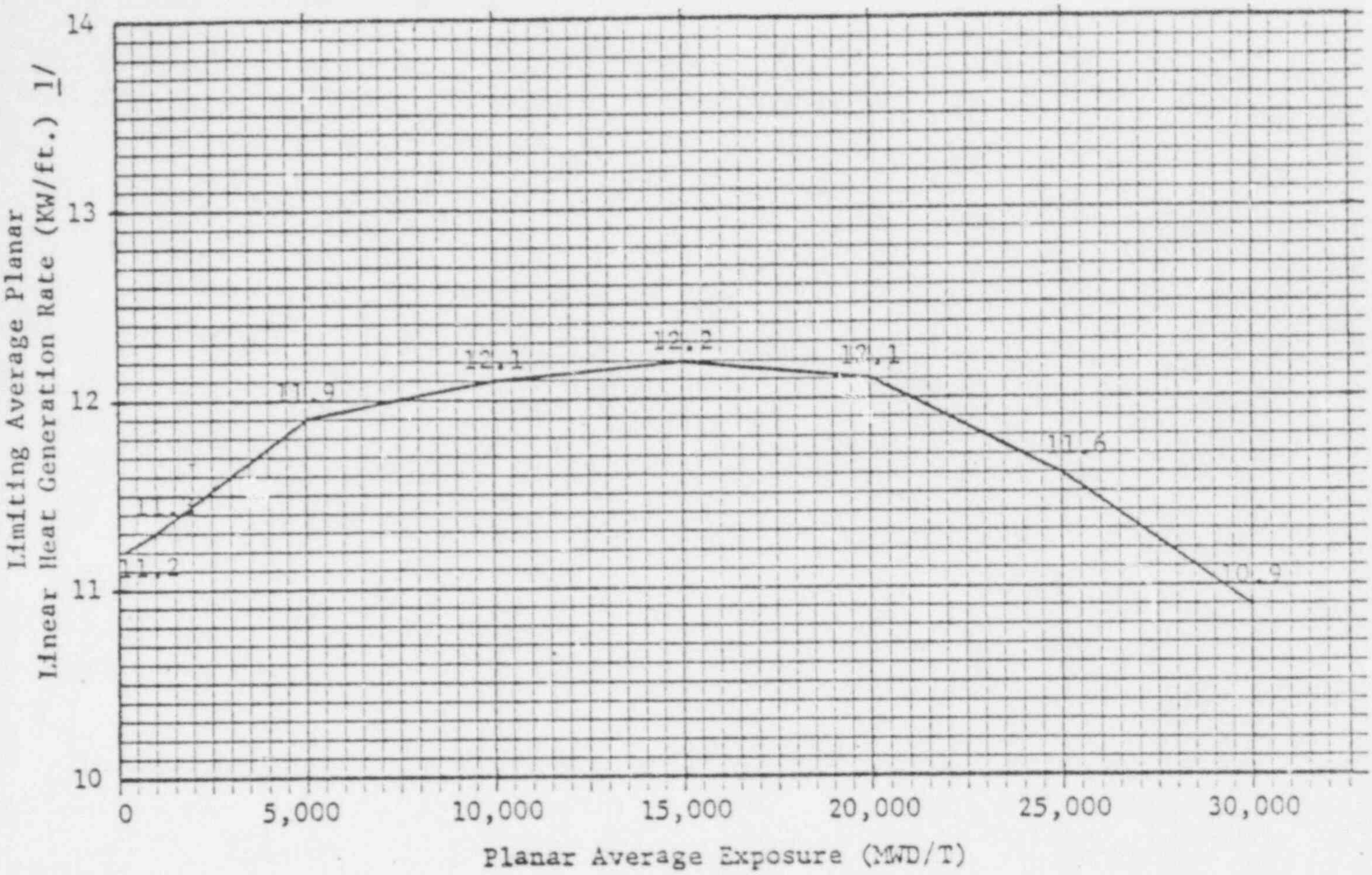
MCPR LIMITS

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



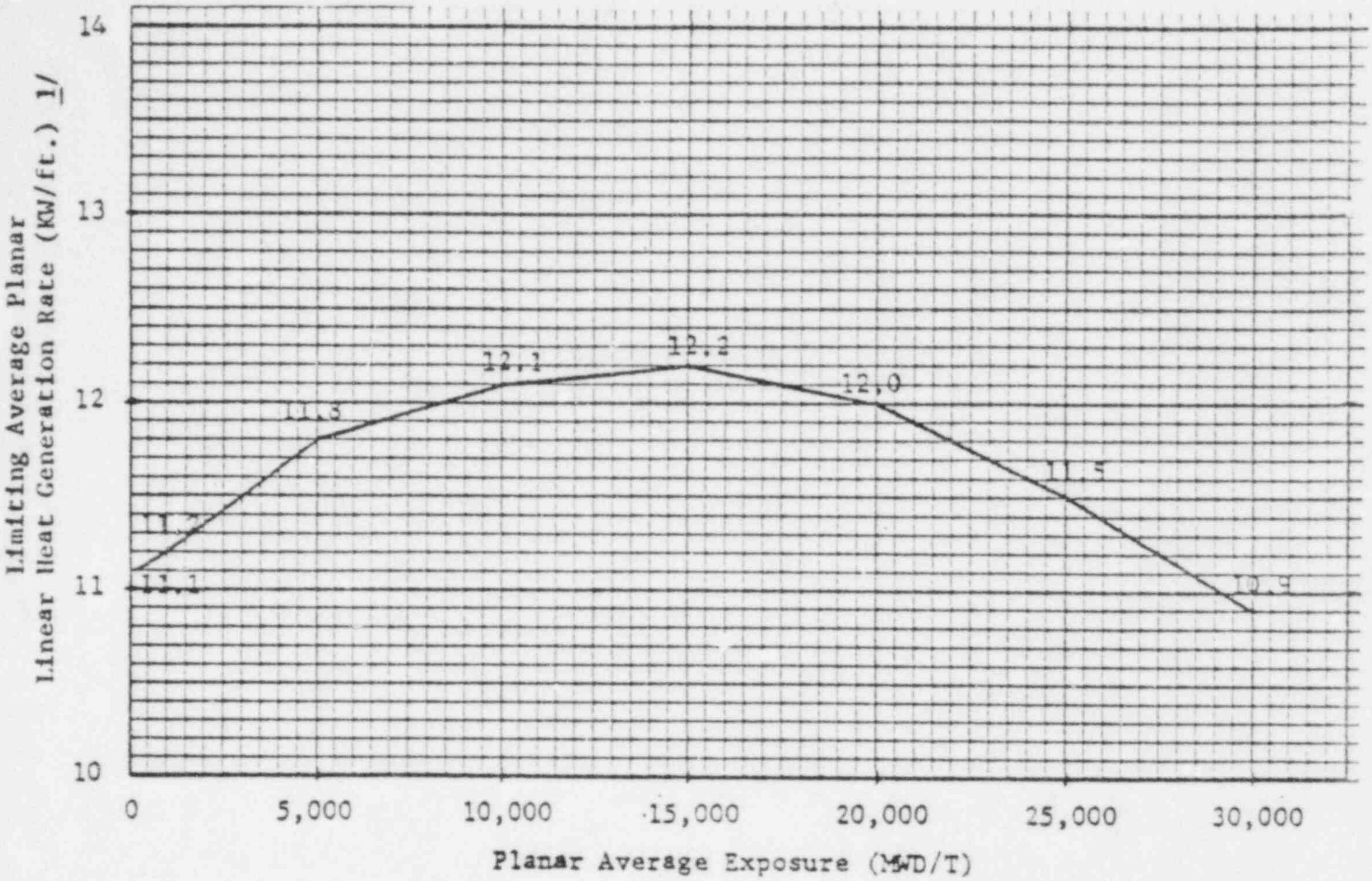
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
TECHNICAL SPECIFICATIONS

K_f AS A FUNCTION OF
CORE FLOW
FIGURE 3.12-1



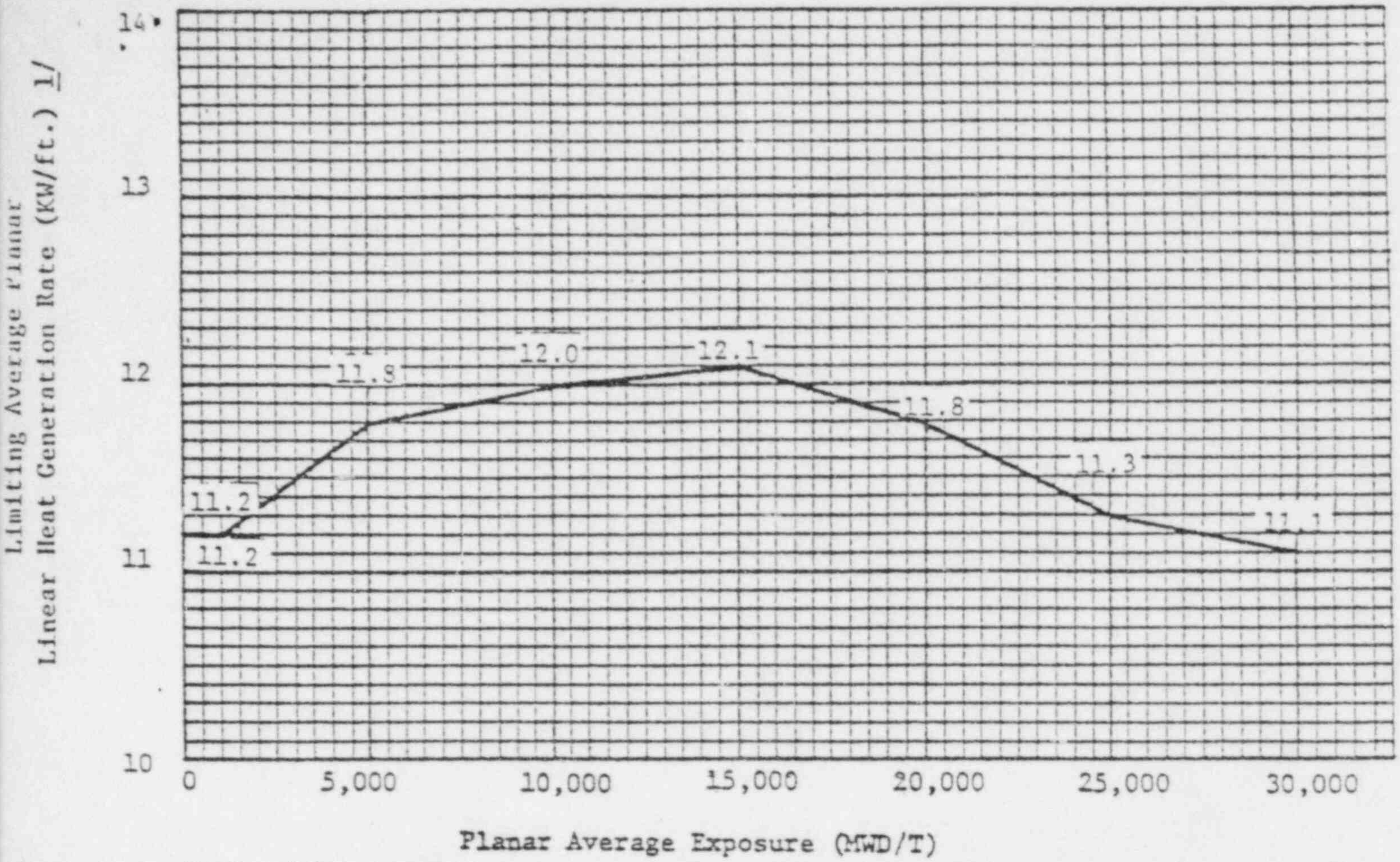
1/ When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown. Values shown are for two recirculation loops. Reduction factors for one recirculation loop operation are given in Section 3.12.A.

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.

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: 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

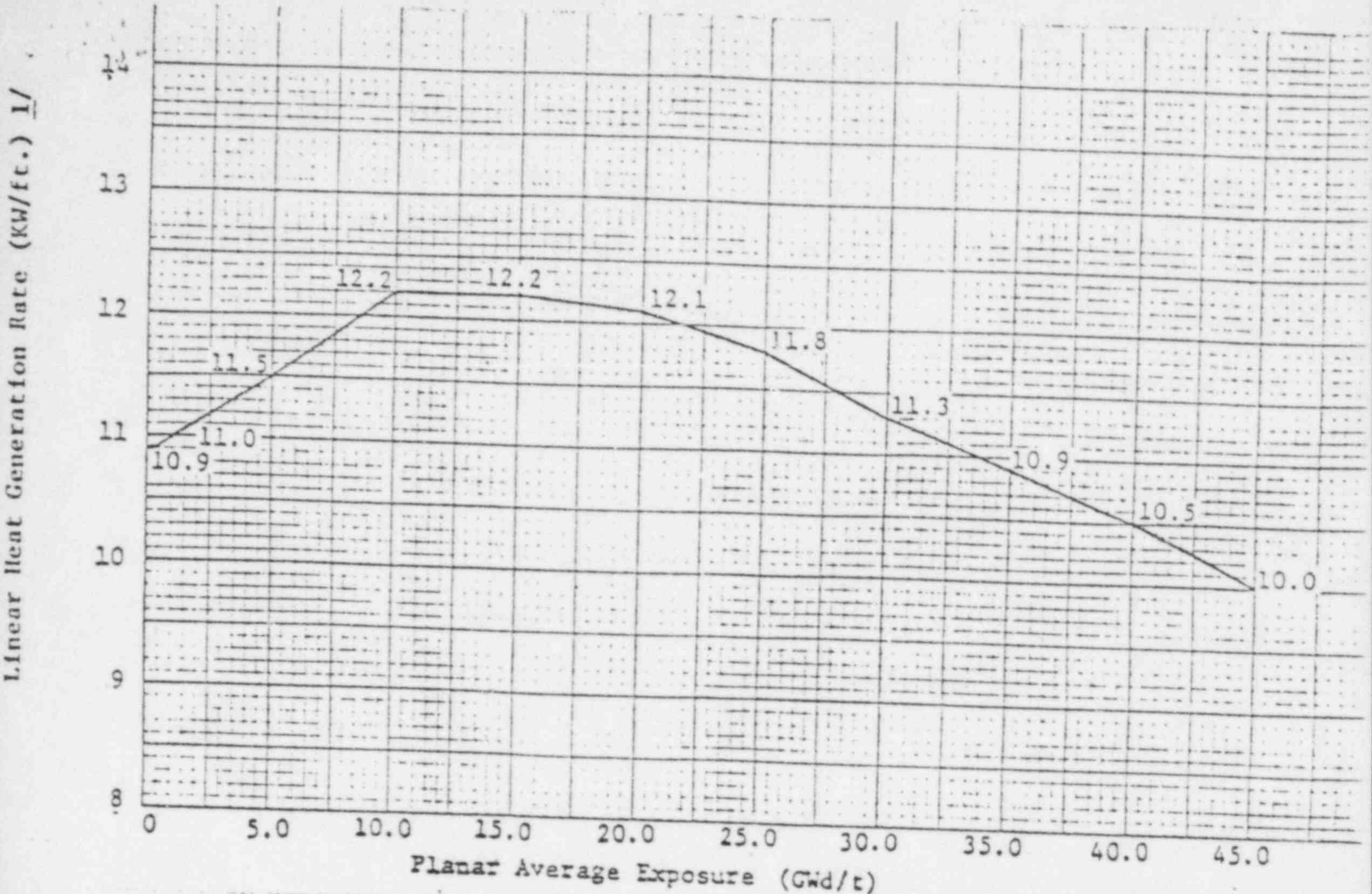
IOWA ELECTRIC LIGHT AND POWER COMPANY

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.

DUANE ARNOLD ENERGY CENTER

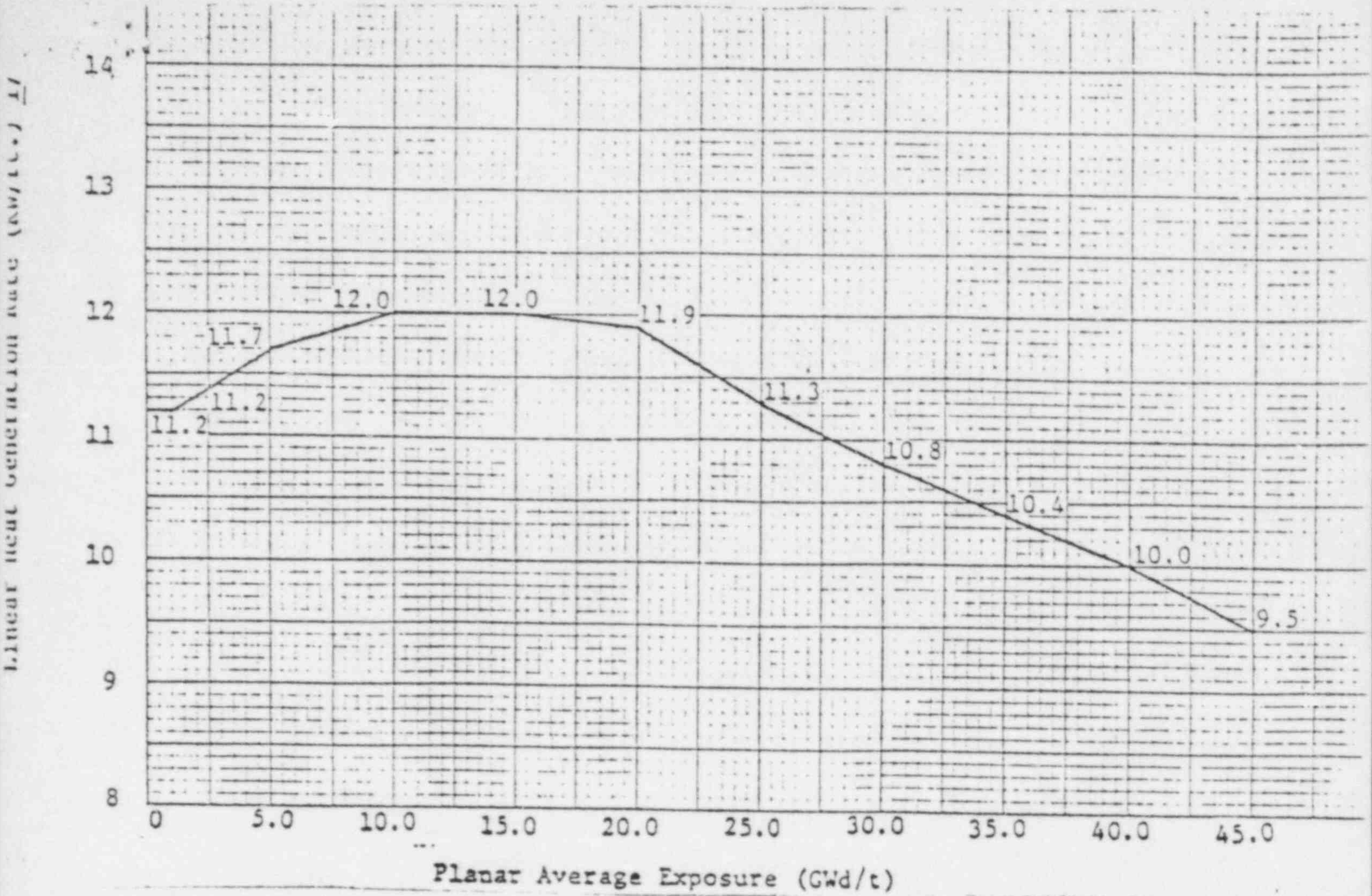
IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

FUEL TYPE: 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

(36)

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

September 23, 1981

Docket No. 50-331

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
P. O. Box 351
Cedar Rapids, Iowa 52406

RE: DUANE ARNOLD ENERGY CENTER

Dear Mr. Arnold:

My letter to you of August 24, 1981 informed you that we were proposing a meeting with licensees who have requested approval to operate on a single recirculation loop. The purpose of the meeting is to determine what may have caused variations in jet pump flow at Browns Ferry Unit No. 1 while operating on a single loop and what impact this should have on approval of other facilities to operate on one loop. Licensees and applicants who have not requested approval for single loop operation were also invited to the meeting.

As you were advised by your licensing project manager, the meeting scheduled for September 9, 1981 had to be postponed to allow more time for analysis of relevant operating data. We apologize for this inconvenience. The meeting will be held at 9:00 AM on Thursday, October 22, 1981 in Room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. It would be appreciated if you would inform your licensing project manager of the number of people who will be attending this meeting from your organization.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

cc: See Next Page

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Mr. Duane Arnold
Iowa Electric Light & Power Company - -

cc:

Mr. Robert Lowenstein, Esquire
Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

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428 Third Avenue, S. E.
Cedar Rapids, Iowa 52401

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Rural Route #1
Palo, Iowa 52324

(37)

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

July 2, 1981

Docket No. 50-331

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
P.O.Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

By letter dated October 17, 1980 (LDR-80-277) you submitted an application to permit Duane Arnold to be operated with a single recirculation loop in service rather than both loops. Three other facilities have requested similar authorization and we expect other BWRs will request approval for single loop operation in the near future.

Several BWRs have previously been authorized to operate for a short period of time with one recirculation loop and two BWRs are currently authorized to operate routinely on a single recirculation loop. In all but one case, power level has been been limited to 50 percent; the one exception was Browns Ferry Unit No. 1. On September 29, 1979, based on analyses performed for TVA by the General Electric Company (GE), we authorized TVA to operate Browns Ferry 1 for about two months at power levels up to 82 percent of full rated power. During power ascension with Browns Ferry 1 in single loop operation, jet pump flow variations were noted in the active loop above a pump speed of 65 percent of rated flow (about 59 percent of rated power). Whenever TVA tried to increase the power level above this point, they noted variations in jet pump flow, neutron flux, and related parameters. Accordingly, TVA administratively limited Browns Ferry Unit 1 operation to less than 60 percent for the approximately two months the unit operated on a single loop.

While analyses indicate that it should be safe to operate BWRs on a single loop in the range of 85 percent of rated power, the experience at Browns Ferry Unit 1 has raised concerns about authorizing single loop operation for BWRs above 50 percent rated power until there is a better understanding of what may have caused the variations in this facility.

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To try to develop a better understanding of what occurred at Browns Ferry 1 in the fall of 1979, we are proposing a meeting with you and the other licensees who have requested approval for single loop operations. We also propose to invite other BWR applicants and licensees since they may desire to have approval for single loop operation of their facilities in the future. The questions we wish to address in the proposed meeting are discussed in the enclosure to this letter. Since GE has provided the analysis to you to support your application, it appears highly desirable that representatives of GE be present in trying to determine what occurred at Browns Ferry Unit 1 and the implications, if any, to other BWRs.

To accommodate the appropriate personnel from your organization and other licensees, we have proposed a range of dates for the above meeting, specifically, the weeks of either July 27, 1981, August 10, 1981, August 17, 1981, August 24, 1981, or September 8, 1981. If you will advise your project manager of the date or dates most convenient to you, we will try to find a day that is most suitable to all parties and so advise you.

Sincerely yours,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Proposed Meeting Agenda

cc w/enclosure - See next page

Mr. Duane Arnold
Iowa Electric Light & Power Company

cc:

Mr. Robert Lowenstein, Esquire
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Mr. Ron E. Engel, Manager
Reload Fuel Licensing (MC 682)
General Electric Company
San Jose, California 95125

Proposed Meeting with BWR Applicants and
Licensees on Single Loop Operation

Purpose of Meeting:

1. To determine what may have caused the jet pump flow variations and other variations experienced by Browns Ferry Unit 1 during single loop operation and

2. Evaluate whether the Browns Ferry experience should result in power limits for other BWRs operating on a single loop.

Agenda:

1. Discussion of what may have caused the unexpected variations in operating parameters when Browns Ferry Unit 1 exceeded about 60 percent rated power while operating with only one recirculation loop.

2. Discussion of parameters affected at Browns Ferry 1 (i.e., jet pump flow, neutron flux, core flow, core pressure drop, etc.)

3. Discussion of whether the Browns Ferry 1 experience would be expected at other BWRs operating on one recirculation loop. If so, are safety limits likely to be violated or cause complications with respect to core stability, core flow symmetry, pump cavitation or damage to the jet pumps and reactor vessel internals.

4. Discussion of the benefits vs. potential problems and cost of testing single loop operation in another BWR that is instrumented to detect what parameters are affected.

5. Evaluation of whether single loop operation at power levels above 50 to 55 percent is a safe and prudent means of reactor operation.

(8) Eccleston

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

Docket No. 50-331

August 24, 1981

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
P. O. Box 351
Cedar Rapids, Iowa 52406

RE: DUANE ARNOLD ENERGY CENTER

Dear Mr. Arnold:

By letter dated July 2, 1981, we informed you that we were proposing to hold a meeting with you and other licensees who have requested approval to operate at power levels above 50% with only one recirculation loop in service in the event the other loop is inoperative. The announced purpose of this meeting is to obtain a better understanding of what might have caused variations in jet pump flow and related parameters at Browns Ferry Unit No. 1 during single loop operation and how this incident affects approval of your application. We had proposed a range of dates for the meeting to accomodate the people expected to attend. Based on expressed preferences, the meeting on single loop operating experience will be held at 9:00 A.M., Wednesday, September 9, 1981 in Room P-118, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland. The agenda, a copy of which is enclosed, is the same as that included with our previous letter.

You are requested to advise your NRC project manager of the personnel who will be attending from your organization.

Sincerely,

T.A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure:
Meeting Agenda

cc w/Enclosure
See next page

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Proposed Meeting with BWR Applicants and Licensees on Single Loop Operation

Purpose of Meeting:

1. To determine what may have caused the jet pump flow and other variations experienced by Browns Ferry Unit 1 during single loop operation and

2. Evaluate whether the Browns Ferry experience should result in power limits for other BWRs operating on a single loop.

Agenda

1. Discussion of what may have caused the unexpected variations in operating parameters when Browns Ferry Unit 1 exceeded about 60 percent rated power while operating with only one recirculation loop.

2. Discussion of parameters affected at Browns Ferry 1 (i.e., jet pump flow, neutron flux, core flow, core pressure drop, etc.)

3. Discussion of whether the Browns Ferry 1 experience would be expected at other BWRs operating on one recirculation loop. If ~~so,~~ are safety limits likely to be violated or cause complications with respect to core stability, core flow symmetry, pump cavitation or damage to the jet pumps and reactor vessel internals.

4. Discussion of the benefits vs. potential problems and cost of testing single loop operation in another BWR that is instrumented to detect what parameters are affected.

5. Evaluation of whether single loop operation at power levels about 50 to 55 percent is a safe and prudent means of reactor operation.

Mr. Duane Arnold
Iowa Electric Light & Power Company

cc:

Mr. Robert Lowenstein, Esquire
Harold F. Reis, Esquire
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Ron E. Engle, Manager
Reload Fuel Licensing (MC 682)
General Electric Company
San Jose, California 95125

K. Eccleston

(39)

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

April 7, 1981

Docket No. 50-331

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

Reference is made to your application of October 17, 1980 (LDR-80-277) requesting authorization for single recirculation loop operation of the Duane Arnold Energy Center. To complete our review, we need responses to the enclosed request for additional information within 60 days of receipt of this letter.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosure:
As stated

cc w/encl:
See attached page

~~8104170676~~

Mr. Duane Arnold
Iowa Electric Light & Power Company

cc:

Mr. Robert Lowenstein, Esquire
Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N. W.
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REQUEST FOR ADDITIONAL INFORMATION
DUANE ARNOLD ENERGY CENTER
SINGLE-LOOP OPERATION

1. Specify expected minimum and maximum operating core power/flow condition as percentage of Rated Core Power/Flow for Single-Loop Operation.

2. At the specified minimum and maximum operating Core Power/Flow Condition for Single-Loop Operation, provide the following:

   (1) Safety Limit MCPR values,

   (2) Fuel Loading Error MCPR analysis results,

   (3) Local Rod Withdrawal Error (with limiting instrument failure) Transient Summary, and

   (4) Core Wide Transients Analysis and Operating Limit MCPR results for all the fuel types in the core for the following transients per NEDE-24011-P-A-1: Flow decrease, Cold Water Injection, and Pressurization.

3. In Section 2.0, a 6% Core Flow Measurement Uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). Explain how the contribution to the total core flow measurement uncertainty value of 6% was calculated and justify that this value conservatively reflects the one standard deviation accuracy of the core flow measurement system.

4. Describe how the change from normal two recirculation cooling loop operation to one loop operation would be accomplished, with what physical and administrative controls, and while complying with branch technical position EICSB 12 (attached) regarding multiple setpoints and their control, and with IEEE STD. 279-4.15.

5. What provisions would be made in the technical specification for decreased flow stability in single loop operation?

6. Describe changes made to the flow computer to automatically account for magnitude and sense change for reverse flow in the idle loop jet pumps during single loop operation.

7. Is there a requirement for the recirculation flow equalizer valves to be closed and tagged prior to commencing single recirculation loop operation as stated in NEDO-24272 Page 1-1 and how is this requirement ensured in the technical specification change?

BRANCH TECHNICAL POSITION ICSB 12

PROTECTION SYSTEM TRIP POINT CHANGES FOR OPERATION WITH REACTOR COOLANT PUMPS OUT OF SERVICE

A. BACKGROUND

For the past several years, including a time prior to the development of IEEE Std 279, the staff has required automatic adjustment to more restrictive settings of trips affecting reactor safety by means of circuits satisfying the single failure criterion. The basis for this requirement is that the function can be accomplished more reliably by automatic circuitry than by a human operator. This design practice, which has also been adopted independently by the national laboratories and by much of industry, served as the basis for paragraph 4.15, "Multiple Set Points," of IEEE Std 279.

More recently, all applicants have stated that their protection systems were designed to meet IEEE Std 279. Paragraph 4.15 of IEEE Std 279 specified that where a mode of reactor operation requires a more restrictive set point, the means for ensuring use of the more restrictive set point shall be positive and must meet the other requirements of IEEE Std 279. A number of designs have been proposed and accepted which reliably and simply satisfy this requirement. During the review of some applications, however, certain design deficiencies have been found. The purpose of this position is to provide additional guidance on the application of Section 4.15 of IEEE Std 279.

B. BRANCH TECHNICAL POSITION

1. If more restrictive safety trip points are required for operation with a reactor coolant pump out of service, and if operation with a reactor coolant pump out of service is of sufficient likelihod to be a planned mode of operation, the change to the more restrictive trip points should be accomplished automatically.

2. Plants with designs not in accordance with the above should have included in the plant technical specifications a requirement that the reactor be shut down prior to changing the set points manually.

C. REFERENCES

1. Millstone-3 Safety Evaluation Report, September 24, 1973.

2. Beaver Valley-2 Safety Evaluation Report, October 10, 1973.

3. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

(40)

Iowa Electric Light and Power Company

March 28, 1983
NG-83-1039

LARRY D. ROOT
ASSISTANT VICE PRESIDENT
NUCLEAR GENERATION

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Single Loop Operation for the Duane Arnold Energy Center

Dear Mr. Denton:

The attachment to this letter provides the final responses to the request for additional information in Mr. Ippolito's letter to Mr. Arnold, dated April 7, 1981 on our submittal for a technical specification amendment on single recirculation loop operation.

Should you have further questions, please contact this office.

Very truly yours,

R. W. McGaughy

for Larry D. Root
Assistant Vice President
Nuclear Generation

LDR/RAB/dmh*
Attachment

A001

cc: R. Browning
D. Arnold
L. Liu
S. Tuthill
F. Apicella
NRC Resident Office

IOWA ELECTRIC RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON SINGLE LOOP OPERATION

The following is the Iowa Electric response to the Commission's request for additional information on single loop operation (SLO) at the Duane Arnold Energy Center (re: letter, Thomas Ippolito to Duane Arnold, "Request for Additional Information, Duane Arnold Energy Center, Single Loop Operation," Docket No. 50-331, April 7, 1981.)

NRC Question #4

Describe how the change from normal two recirculation cooling loop operation to one loop operation would be accomplished, with what physical and administrative controls, and while complying with branch technical position EICSB 12 (attached) regarding multiple setpoints and their control, and with IEEE STD. 279-4.15.

IE Response

In order to ensure conformance with the assumptions used in the GE analysis of SLO at DAEC, Iowa Electric proposes to operate under the following restrictions:

1) The suction valve will be closed and electrically isolated in the inoperable recirculation loop per proposed Technical Specification 3.6.F.2.C. This is to prevent degradation of LPCI flow during LOCA events.

2) DAEC does not have equalizer lines between the A and B loop jet pump risers, thus the requirement that the valves be verified to be closed is not applicable.

3) The recirculation system controls will be placed in the manual mode per proposed Technical Specification 3.6.F.2.d., thereby eliminating the need for a control systems evaluation.

4) The steady state thermal power level will not exceed 50% of the rated value, per proposed Technical Specification 3.6.F.2.b.

5) The settings for the Rod Block Monitor (RBM), APRM Rod Block and Scram flow-biased setpoints will be modified, per the proposed Technical Specifications 2.1.A.2, 2.1.A.3 and 3.2.C.1 and will be implemented by the appropriate adjustment in APRM gain factor, per Surveillance Test Procedure (STP) 42A001, Item 4.2.K.6. Setdown of Rod Block and Scram setpoints by amplifier gain adjustment is an accepted procedure per Amendment No. 30 to the DAEC Technical Specifications.

6) The fuel operating limits (MCPR and MAPLHGR) will be adjusted for SLO in the Technical Specifications, Sections 1.1.A, 3.12.A, 3.12.C. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased by 0.03 and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced by 30%.

7) Increased surveillance requirements on core plate ΔP and APRM flux noise will be implemented per proposed Technical Specifications 4.6.F.2 and 2.1.A.4, respectively.

NRC Question #6

Describe changes made to the flow computer to automatically account for magnitude and sense change for reverse flow in the idle loop jet pumps during single loop operation.

IE Response

No changes are necessary to account for reverse flow in the idle jet pumps as the existing circuitry will subtract this flow when the recirculation pump is deactivated. The coefficient of 0.95 applied to the reverse jet pump flow, to account for the difference in flow coefficient between forward and reverse flow, is accounted for in the calibration of the flow summer amplifier gains.