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PLANT NAME: DUANE ARNOLD ENERGY CENTER

ENCLOSURE

APPLICATION FOR AMENDMENT CONSISTING OF PROPOSED TECH. SPEC. CHANGES AND NEDO-21226 DATED 8/76, DAEC LICENSE AMDT SUBMITTAL FOR SINGLE LOOP OPERATION WITH THE BYPASS FLOW HOLES PLUGGED NOT INVOLVING A SIGNIFICANT HAZARDS CONSIDERATION...,

(45 PAGES)
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IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
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January 12, 1977
IE-77-94

LEE LIU
VICE PRESIDENT - ENGINEERING

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Rusche:

Transmitted herewith in accordance with the requirements of 10CFR50.59 and 50.90 is an application for amendment for DPR-49 (Appendix A to License) for the Duane Arnold Energy Center.

This application consisting of proposed Technical Specification Change RTS-74 has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee. Also enclosed is NEDO-21226 dated August 1976, DAEC license Amendment Submittal for Single-Loop Operation with the Bypass Flow Holes Plugged. This application does not involve a significant hazards consideration.

Three signed and notarized originals and 37 additional copies of this application are transmitted herewith. This application, consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

By _____
Lee Liu
Vice President-Engineering

LL/OCS/D
Enc.
cc: K. Meyer
D. Arnold
R. Lowenstein
J. Shea (NRC)
J. Keppler (NRC)
L. Root
File A-117

Subscribed and Sworn to before me on this 13th day of January, 1977.

Marjorie E. McDonald
Notary Public in and for the State of Iowa.

834

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.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated 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.	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.
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.	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 $\leq$ 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, 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:

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

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. <u>APRM High Flux Scram (Refuel or Startup and Hot Standby Mode).</u> For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide adequate thermal margin between the setpoint and the Safety Limit,

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

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 880 psig.

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

Reactor power level may be varied by moving control rods or

by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than 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)	$\leq (0.66W + 41) \left(\frac{*}{TPF}\right)$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased) (Single Loop Operation)	$\leq (0.66W + 37.7) \left(\frac{*}{TPF}\right)$ (2)	2 Inst. Channels	(1)

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

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. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. When both recirculation pumps are in steady state operation, the speed of the faster pump may not exceed 122% of the speed of the slower pump when core power is 80% or more of rated power or 135% of the speed of the slower pump when core power is below 80% of rated power.	1. Recirculation pump speeds shall be checked and logged at least once per day.
2. 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.	

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	4.12 CORE THERMAL LIMITS
Applicability:	Applicability:
The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.
Objective	Objective
The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications	Specifications
A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)	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.	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).

3.12 REFERENCES

1. Duane Arnold Energy Center "Safety Analysis with Bypass Holes Plugged", June 9, 1975 and Supplement 1, June 16, 1975.

2. General Electric BWR Generic Reload Application for 8 x 8 Fuel, NEDO-20360, Revision 1, November 1975.

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 10 CFR Part 50, Appendix K, NEDE-20566 (Draft), August 1974.

8. Duane Arnold Energy Center Reload Number One Licensing Submittal, January 1976.

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