March 27, 1985

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications in response to your application dated August 17, 1984.

The amendment revises the Technical Specifications to permit an increase in the DAEC rated power from current 1593 megawatt thermal (MWt) to a maximum steady state core power level of 1658 MWt. The review of the Extended Load Line Limit analysis presented in your application will be performed as a separate action.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Mohan C. Thadani, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No.115 to
   License No. DPR-49
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Lee Liu
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CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light & Power Company, et al, dated August 17, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 115, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

/s/ Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 27, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 115

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by removing the current pages and inserting the revised pages listed below. The revised areas are identified by vertical lines.

LIST OF AFFECTED PAGES

vii	1.1-17	3.1-21	3.6-28	3.7-44
1.0-2	1.1-18	3.2-16	3.6-41	3.7-48a
1.0-3	1.1-21	3.2-17	3.7-1	3.7-49
1.0-5	1.1-23	3.2-23	3.7-3	3.12-1
		3.5-6	3.7-4	3.12-11
1.1-2	1.2-1	3.5-8	3.7-6	3.12-16*
1.1-3	1.2-4	3.5-9	3.7-24	3.12-17
1.1-14	3.1-3	3.5-19	3.7-31	3.12-18
1.1-15	3.1-4	3.5-22	3.7-36	3.12-19
1.1-16	3.1-6	3.5-26	3.7-37	3.12-20

*These pages have been deleted.

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

5. OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

6. OPERATING

Operating means that a system or component is performing its intended functions in its required manner.

7. IMMEDIATE

Immediate means that the required action will be initiated as soon as practical considering the safe operation of the unit and the importance of the required action.

8. REACTOR POWER OPERATION

Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

9. HOT STANDBY CONDITION

Hot standby condition means operation with coolant temperature greater than 212°F, reactor vessel pressure less than 1055 psig, and the mode switch in the Startup/Hot Standby position.

10. COLD CONDITION

Reactor coolant temperature equal to or less than 212°F.

11. HOT SHUTDOWN

The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

12. COLD SHUTDOWN

The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.

13. MODE OF OPERATION

A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

a. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by main steam line isolation valve closure, are bypassed. The reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. The lower pressure MSIV closure 850 psig trip is also bypassed. This is intended to imply the Startup/Hot Standby position of the mode switch.

b. Run Mode - In this mode the reactor vessel pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

c. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 seconds), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram is bypassed.

d. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicates at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram is bypassed. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

14. RATED POWER

Rated power (100% power) refers to operation at a reactor power of 1658 Mwt.

19. ALTERATION OF THE REACTOR CORE (CORE ALTERATION)

The addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

20. REACTOR VESSEL PRESSURE

Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

21. THERMAL PARAMETERS

a. Minimum Critical Power Ratio (MCPR) - The value of critical power ratio (CPR) for that fuel bundle having the lowest CPR.

b. Critical Power Ratio (CPR) - The ratio of that fuel bundle power which would produce boiling transition to the actual fuel bundle power.

c. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

d. Deleted

e. Linear Heat Generation Rate - the heat output per unit length of fuel pin.

f. Fraction of Limiting Power Density (FLPD) - The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

g. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

h. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 1658 MWth.

i. Total Peaking Factor (TPF) - The ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

j. Maximum Total Peaking Factor (MTPF) - The largest TPF which exists in the core for a given class of fuel for a given operating condition.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

C. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.

D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone. Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

Where: S = Setting in percent of rated power (1,658 MWt)

W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^6 lb/hr core flow.

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

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

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

2. APRM High Flux Scram

When in the REFUEL or STARTUP and HOT STANDBY MODE, the APRM scram shall be set at less than or equal to 15 percent of rated power.

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

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 Figure 2.1-1 and shall be:

   $$S \leq (0.58\ W + 50)$$

   The definitions used above for the APRM scram trip apply.

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

   $$S \leq (0.66\ W + 54)\ \frac{FRP^*}{MFLPD}$$

4. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.

B.	Scram and Isolation on reactor low water level	$\geq$ 514.5 inches above vessel zero (+170" indicated level)
C.	Scram - turbine stop valve closure	$\leq$ 10 percent valve closure

D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

*see footnote to 2.1.A.1

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

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

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

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

For analyses of the thermal consequences of the transients the MCPRs stated in Section 3.12 as a limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

Steady-state operation without forced recirculation will not be permitted, except during special testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients have been analyzed to a power level of 102% of 1658 MWt.

ii. The licensed maximum power level is 1658 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

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

Trip Settings

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

A. Neutron Flux Trips

1. APRM High Flux Scram (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1658 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin 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. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow-referenced APRM High Flux Scram curve by the reciprocal of the APRM gain change.

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

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

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

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is not more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. APRM Rod Block (Run Mode)

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than 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

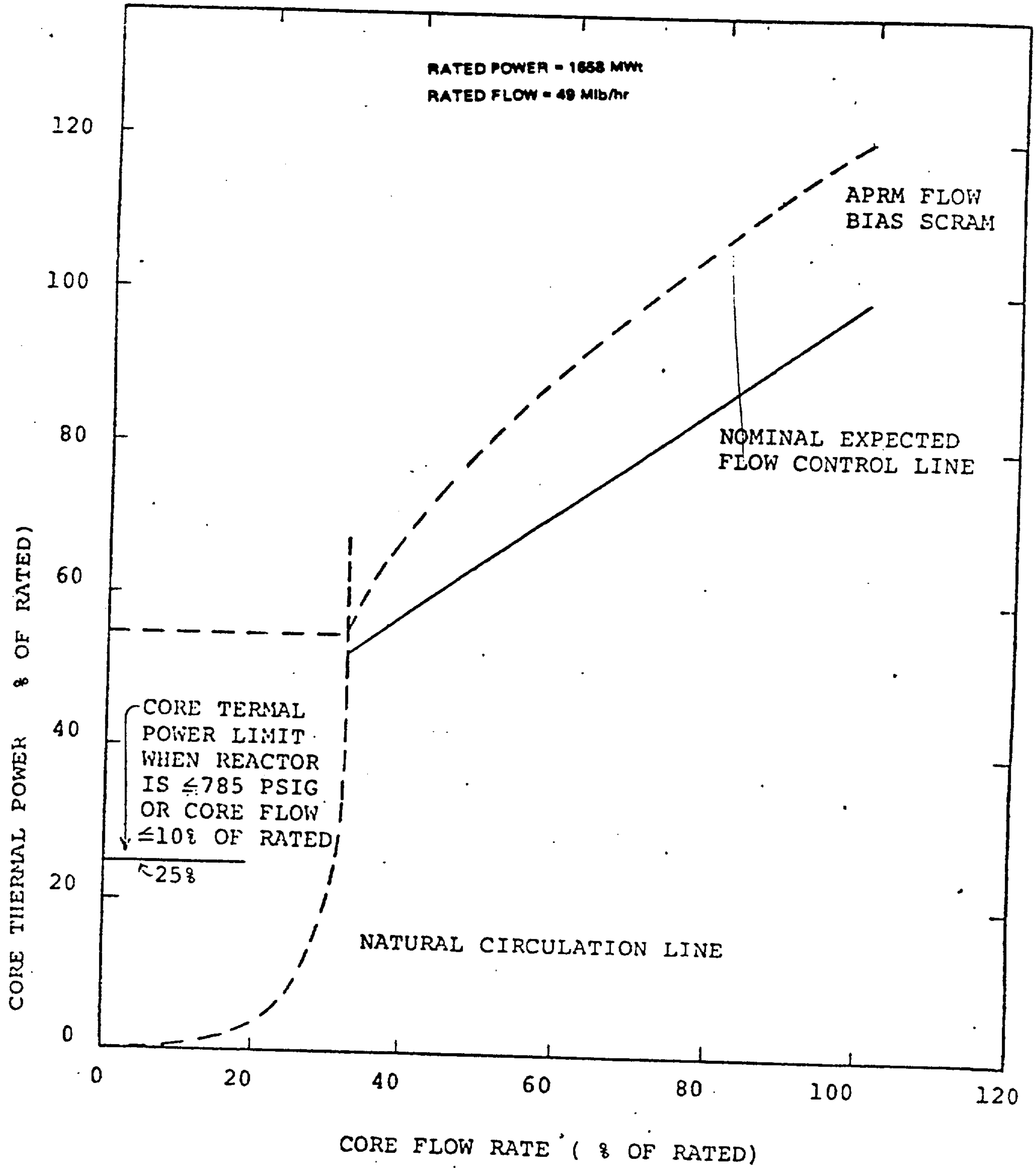
With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when turbine steam flow is below approximately 30 percent 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 850 psig has been provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY TECHNICAL SPECIFICATIONS
APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS **FIGURE 1.1-1**

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

1. The reactor vessel dome pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM 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 pressure safety limit from being exceeded.

Specification:

1. The limiting safety system setting shall be as specified below:

Protective Action/Limiting Safety System Setting

A. Scram on Reactor Vessel high pressure

1055 psig

B. Relief valve settings
1110 psig ± 11 psi (1 valve)

1120 psig ± 11 psi (1 valve)

1130 psig ± 11 psi (2 valves)

1140 psig ± 11 psi (2 valves)

design pressure (120% x 1150 = 1380 psig; 120% x 1325 = 1590 psig).

The analysis of the worst overpressure transient, a 3 second closure of all main steam isolation valves with a direct valve position scram failure (i.e., scram is assumed to occur on high neutron flux), shows that the peak vessel pressure experienced is much less than the code allowable overpressure limit of 1375 psig (Reference 1). Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected overpressure transients.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. At this time it is included in the reactor coolant system.

1.2 References

1. "General Electric Boiling Water Reactor Supplemental Reload Licensing Submittal for Duane Arnold Energey Center", 23A1739.*

*Refer to analyses for the current operating cycle.

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 Fuel Scale	X	X	(5)	6 Instrument Channels	A
2	IRM Inoperative		X	X	(5)	6 Instrument Channels	A
2	APRM High Flux	**(.66W+54)** (FRP/MFLPD) (11) (12)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤ 1055 psig	X(8)	X	X	4 Instrument Channels	A

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Number of Instrument Channels Provided By Design	Action (1)
2	High Drywell Pressure	$\leq$ 2.0 psig	X(7)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	$\geq$ +170" Indicated Level (15)	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	$\leq$ 60 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	$\leq$ 3 x Normal Rated Power Background*	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	$\leq$ 10% Valve Closure	X (3)(13)	X (3)(13)	X(13)	8 Instrument Channels	A or C
2	Turbine Control Valve Fast Closure (Loss of Control Oil Pressure)	Within 30 milliseconds of the Start of Control Valve Fast Closure			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	$\leq$10% Valve Closure			X(4)	8 Instrument Channels	A or D
2	First Stage	Bypass below 165 psig	X	X	X	4 Instrument Channels	A or D

*Alarm setting $\leq$1.5 X Normal Rated Power Background

3. A main steam line isolation valve closure trip bypass is effective when the reactor mode switch is in the shutdown, refuel or startup positions.

4. Bypassed when turbine first stage pressure is less than 165 psig (corresponding to 30% of rated core power). This value of first stage pressure assumes that the second stage reheaters are not in service below 30% of rated core power.

5. IRM's are bypassed when APRM's are on-scale and the reactor mode switch is in the run position.

6. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode switch in shutdown

   b. Manual scram

   c. High flux IRM

   d. Scram discharge volume high level - may be bypassed in the refuel and shutdown modes for the purpose of resetting the scram.

   e. APRM 15% flux

to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

Turbine stop valve closure trip occurs at approximately 10% of valve closure. Below 165 psig turbine first stage pressure (corresponding to 30% of rated core power), the scram signal due to turbine stop valve closure is by-passed because the flux and pressure scrams are adequate to protect the reactor below 30% of rated core power.

Turbine Control valve fast closure scram trip shall initiate within 30 milliseconds of the start of control valve fast closure. The trip level setting is verified by measuring the time interval from energizing the fast acting solenoid (from valve test switch) to pressure switch response; the measured result is compared to base line data taken during each refueling outage. Turbine control valve fast closure is sensed by measuring disc dump electro-hydraulic oil line pressure (Relay Emergency Trip Supply) which decreases rapidly upon generator load rejection. This scram is only effective when turbine first stage pressure is above 165 psig (corresponding to 30% of rated core power).

The requirement that the IRM's be inserted in the core when the APRM's read 5 as indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

TABLE 3.2-C
INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq(0.66\ W + 42)\left(\frac{FRP}{MFLPD}\right)^{(2)}$	6 inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq(0.66\ W + 39)\left(\frac{FRP}{MFLPD}\right)^{(2)}$	6 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)

NOTES FOR TABLE 3.2-C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM [except for APRM Upscale (Not in Run Mode)] and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W is the recirculation loop flow in percent of rated. Trip level setting is in percent of rated power (1658 MWt). A ratio of FRP/MFLPD $<$ 1.0 is permitted at reduced power. See Subsection 2.1.A.1.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is > 100 cps.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action
1	(ATWS) Reactor High Pressure	$\leq$ 1140 psig	4	(2)
1	(ATWS) Reactor Low-Low Water Level	$\geq$ +119.5 in. Indicated level(5)	4	(2)
1	(EOC) RPT Logic	N/A	2	(3)
1	(EOC) RPT System (Response Time)	$\leq$ 140 msec (4)	2	(3)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for operating recirculation pump. If this cannot be met, the indicated action shall be taken.
2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.
3. Two EOC RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 consecutive hours, an orderly power reduction shall be initiated and the reactor power shall be less than 85% within four hours.
4. This response time is from initiation of Turbine control valve fast closure or Turbine stop valve closure to actuation of the breaker secondary (auxiliary) contact.
5. Zero referenced to top of active fuel.**

**Top of active fuel zone is defined to be 344.5" above vessel zero (see bases 3.2).

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
D. HPCI Subsystem	D. HPCI Subsystem
1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 and 3.5.D.3 below.	1. HPCI Subsystem testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	At rated reactor pressure demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1040 psig.	Once/3 months
e.	At reactor pressure of 150 $\pm$ 10 psig demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection.	Once/operating cycle

The HPCI pump shall deliver at least 3000 gpm for a system head corresponding to a reactor pressure of 1040 to 150 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

	Item	Frequency
	than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1040 psig.	
e.	At reactor pressure of 150 + 10 psig demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection.	Once/operating cycle

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to 1040 to 150 psig.

f.	Verify that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.	Once/operating cycle

2. When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

Limiting Condition for Operation:

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 150 psig within 24 hours.

LIMITING CONDITIONS FOR OPERATION

F. Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 100 psig and prior to a startup from a Cold Condition, except as specified in 3.5.F.2 below.

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days unless such valve is sooner made operable, provided that during such thirty days the HPCI subsystem is operable.

3. If the requirements of 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 100 psig within 24 hours.

G. Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided

SURVEILLANCE REQUIREMENT

F. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least daily thereafter.

G. Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In

does not result in rapid depressurization of the reactor vessel. The HPCIS permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray System operation maintains core cooling.

The capacity of the system is selected to provide this required core cooling. The HPCI pump is designed to pump 3000 gpm at reactor pressures between approximately 1135 and 150 psig. Two sources of water are available. Initially, demineralized water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor.

When the HPCI System begins operation, the reactor depressurizes more rapidly than would occur if HPCI was not initiated due to the condensation of steam by the cold fluid pumped into the reactor vessel by the HPCI System. As the reactor vessel pressure continues to decrease, the HPCI flow momentarily reaches equilibrium with the flow through the break. Continued depressurization causes the break flow to decrease below the HPCI

Because the Automatic Depressurization System does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the CSCS. Performance analysis of the Automatic Depressurization System is considered only with respect to its depressurizing effect in conjunction with LPCI and Core Spray and is based on 3 valves. There are four valves in the ADS and each has a capacity of approximately 810,000 lb/hr at a set pressure of 1125 psig.

The allowable out-of-service time for one ADS valve is determined as thirty days because of the redundancy and because the HPCIS is demonstrated to be operable during this period. Therefore, redundant protection for the core with a small break in the nuclear system is still available.

The ADS test circuit permits continued surveillance on the operable relief valves to assure that they will be available if required.

3.5 REFERENCES

1. Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Company, APED, April 1968 (APED 5736).

2. General Electric Company, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, NEDO-20566, 1974, and letter MFN-255-77 from Darrell G. Eisenhut, NRC, to E.D. Fuller, GE, Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-lead Plants, dated June 30, 1977.

3. General Electric, Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant), NEDO-21082-03, June 1984.

the direct scram (valve position scram) results in a peak vessel pressure less than the Code allowable overpressure limit of 1375 psig if a flux scram is assumed.

The relief valve setpoints given in Section 2.2.1.B have been optimized to maximize the simmer margin, i.e., the difference between the normal operating pressure and the lowest relief valve setpoint. The Reference 2 analysis shows that the six relief valves assure margin below the setting of the safety valves such that the safety valves would not be expected to open during any normal operating transient.* This analysis verifies that the peak system pressure during such an event is limited to greater than the 60 psi design margin to the lowest spring safety valve setpoint.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their setpoints are within the ± 1 percent tolerance. Additionally, once per operating cycle, each relief valve is tested manually with reactor pressure above 100 psig and with turbine bypass flow to the main condenser to demonstrate its ability to pass steam. By observation of the change in position of the turbine bypass valve, the relief valve operation is verified.

*A normal operating transient is defined as an event whose probability of occurrence is greater than once per 40 years, e.g., Turbine Trip with Bypass, MSIV closure with direct scram.

The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.6 and 4.6 References

1) General Electric Company, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center, NEDE-30021-P, January, 1983.

2) "General Electric Boiling Water Reactor Increased Safety/Relief Valve Simmer Margin Analysis for Duane Arnold Energy Center," NEDC-30606, May, 1984.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.7 PLANT CONTAINMENT SYSTEMS	4.7 PLANT CONTAINMENT SYSTEMS
Applicability:	Applicability:
Applies to the operating status of the primary and secondary containment systems.	Applies to the primary and secondary containment system integrity.
Objective:	Objective:
To assure the integrity of the primary and secondary containment systems.	To verify the integrity of the primary and secondary containments.
Specification:	Specification:
A. Primary Containment	A. Primary Containment
1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained with the following limits.	1.a. The pressure suppression pool water level and temperature shall be checked once per day.
a. Maximum water volume - 61,500 cubic feet	b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.
b. Minimum water volume - 58,900 cubic feet	c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 200°F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.
c. Maximum water temperature (1) During normal power operation - 95F. (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.	d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage. 2. The primary containment integrity shall be demonstrated as follows:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	2) Closure of containment isolation valves for the Type A test shall be accomplished by normal mode of actuation and without any preliminary exercising or adjustments.
	3) The containment test pressure shall be allowed to stabilize for a period of about 4 hours prior to the start of a leakage rate test.
	4) The reactor coolant pressure boundary shall be vented to the containment atmosphere prior to the test and remain open during the test.
	5) Test methods are to comply with ANSI N45.4-1972.
	6) The accuracy of the Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4-1972.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

7) Periodic Leakage Rate Tests

Periodic leakage rate tests shall be performed at or above the peak pressure (Pa) of 43 psig.

8) Acceptance Criteria

Reduced pressure tests. (Pt, reduced pressure) The leakage rate Ltm shall be less than 0.75 Lt.

Peak pressure test. (Pp, peak pressure) The leakage rate Lpm shall be less than 0.75 (La).

9) Additional Requirements

If any periodic Type A test fails to meet the applicable acceptance criteria the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

If two consecutive periodic Type A tests fail to meet the acceptance criteria of 4.7.A.2.(a)(9) a Type A test shall be performed at each plant shutdown for major refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the subject acceptance criteria after which time the retest schedule of 4.7.A.2.(d) may be resumed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

d. Periodic Retest Schedule

1) Type A Test

After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. (These intervals may be extended up to eight months if necessary to coincide with refueling outages.) The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under administrative control and in accordance with the plant safety procedures.

2) Type B Tests

a) A continuous leakage monitoring system is provided to measure changes in containment leakage during service. Accordingly, penetrations and seals of this type (except air locks) shall be leak tested at greater than or equal to 43 psig (Pa) every other reactor shutdown for major fuel reloading.

b) The personnel airlock shall be pressurized to greater than or equal to 43 psig (Pa) and leak tested at an interval no longer than one operating cycle. The airlock will be monitored for leakage with the continuous leakage monitoring system during plant operation. A report

NOTES TO TABLE 3.7-2

[1]Test volume is filled with demineralized water then pressurized to greater than or equal to 43 psig with air or nitrogen for test.

For all other penetrations (except Main Steam Lines) test volumes are pressurized to greater than or equal to 43 psig with air or nitrogen for test.

[2]MO-4441, MO-4442 will be remote manually closed.

[3]Subject isolation valves to be installed at earliest practicable date per FSAR P. 6.4-10.C, dated 9/73.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-colant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum allowable pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 43 psig which is below the design pressure of 56 psig. The minimum volume of 58,900 ft^3 results in a submergence of approximately 3 feet. Based on Humboldt Bay, Bodega Bay, and Marviken test facility data as utilized in General Electric Company document number NEDE-21885-P and data presented in Nutech document, Iowa Electric document number 7884-M325-002, the following technical assessment results were arrived at:

1. Condensation effectiveness of the suppression pool can be maintained for both short and long term phases of the Design Basis Accident (DBA), Intermediate Break Accident (IBA), and Small Break Accident (SBA) cases with three feet submergence.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 43 psig which would rapidly reduce to 27 psig within 30 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure shortly thereafter and then rapidly decays with the drywell pressure decay, (Reference 1).*

The design pressure of the drywell and suppression chamber is 56 psig, (Reference 2). The design basis accident leakage rate is 2.0%/day at a pressure of 43 psig. As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated by the AEC staff incorporating the primary containment design basis accident leak rate of 2.0%/day, (Ref. 3). The analysis showed that

*NOTE: The initial leak rate testing performed during plant startup was conducted at a pressure of 54 psig in accordance with the original FSAR analysis of peak containment pressure (Pa).

with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 90% for particulate iodine, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 2 rem and the maximum thyroid dose is about 32 rem at the site boundary over an exposure duration of two hours. The resultant thyroid dose that would occur over the course of the accident is 98 rem at the boundary of the low population zone (LPZ). Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate is conservative and provides additional margin between expected offsite doses and 10 CFR 100 guidelines.

The design basis accident leak rate at the peak accident pressure of 43 psig (P_p) is 2.0 weight percent per day (L_a). To allow a margin for possible leakage deterioration during the interval between Type A tests allowable containment operational leak rate (L_{to}), is 0.75 L_{to}. In addition to these the accidents analyzed, as the FSAR analysis shows compliance with 10 CFR 100 guidelines with an assumed efficiency of 99% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 11 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. (The design of the SGTS system allows the removal of charcoal samples from the bed directly through the use of a grain thief.) Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings, (see Bases Section 3.7.A.1).

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

3.7.A & 4.7.A REFERENCES

1. "Duane Arnold Energy Center Power Uprate", NEDC-30603-P, May, 1984 and Attachment 1 to letter L. Lucas to R.E. Lessly, "Power Uprate BOP Study Report," June 18, 1984.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR 50.54, Appendix J, Reactor Containment Testing Requirements, Federal Register, August 27, 1971.

5. DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-065, August 1976.

6. Supplement to DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-071, October 1976.

LIMITING CONDITION FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability

The Limited 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. 3.12-6, -7, -8 and -9. If at 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 2 hours, reduce reactor power to ≤ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENT

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 ≥ 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 MAPLHGR shall be determined at least once per 12 hours hours.

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-03, June 1984.

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

3. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-19735, August 1973.

4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

5. Communication: V.A. Moore to I.S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

6. R.B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

7. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, NEDE-20566, August 1974.

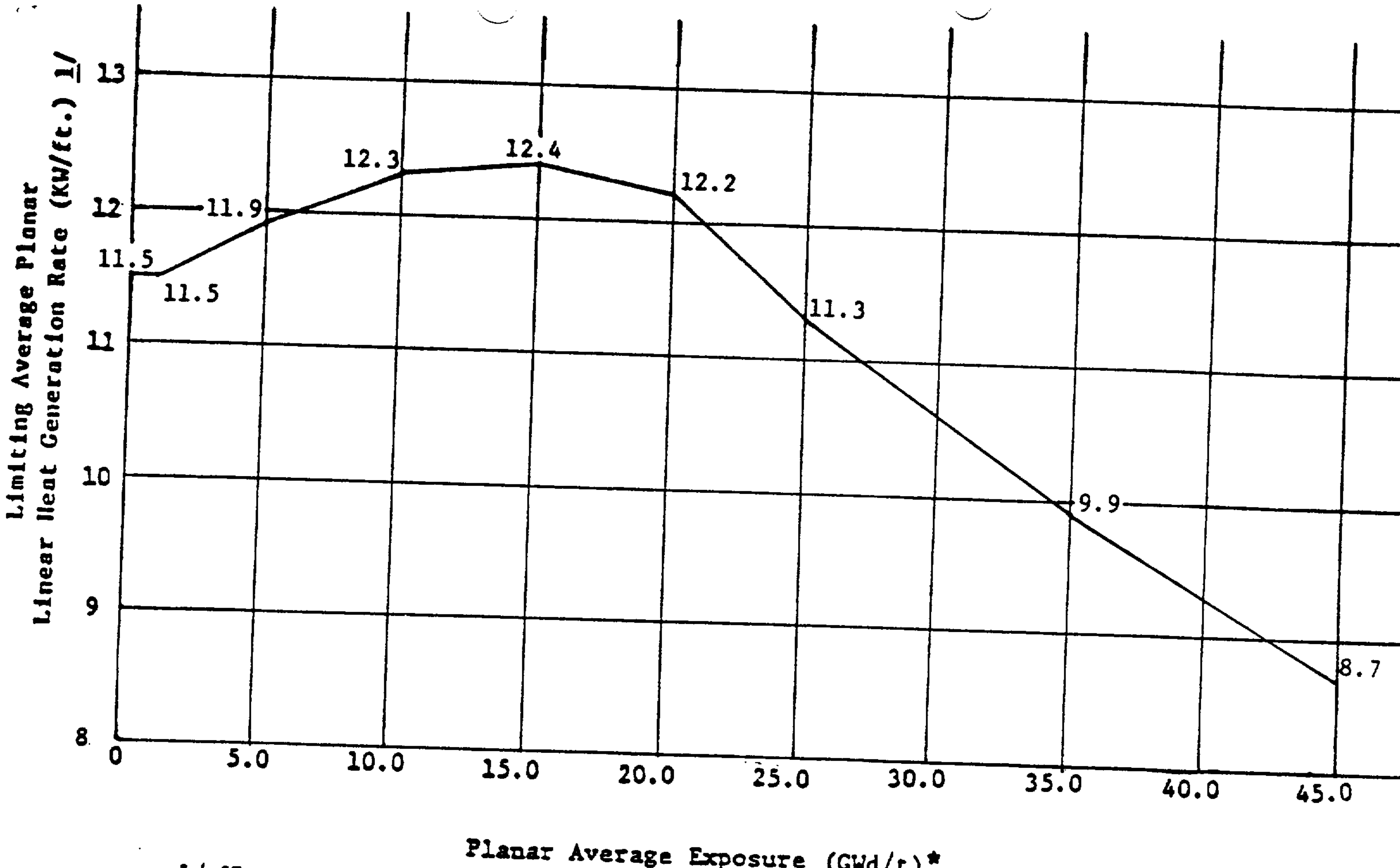
8. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977.

9. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

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

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

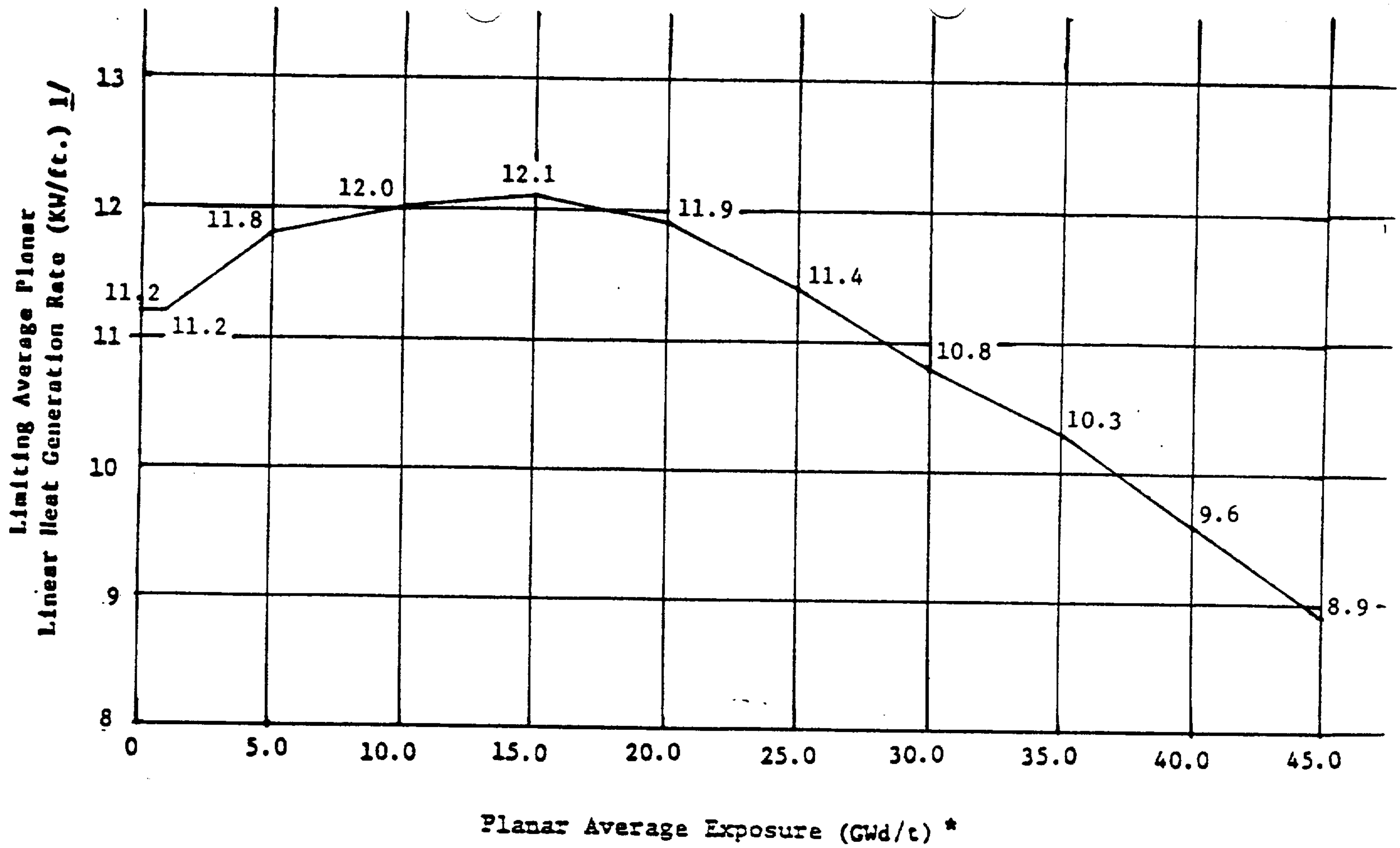
DELETED



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

* 1 GWd/t = 1000 MWd/t

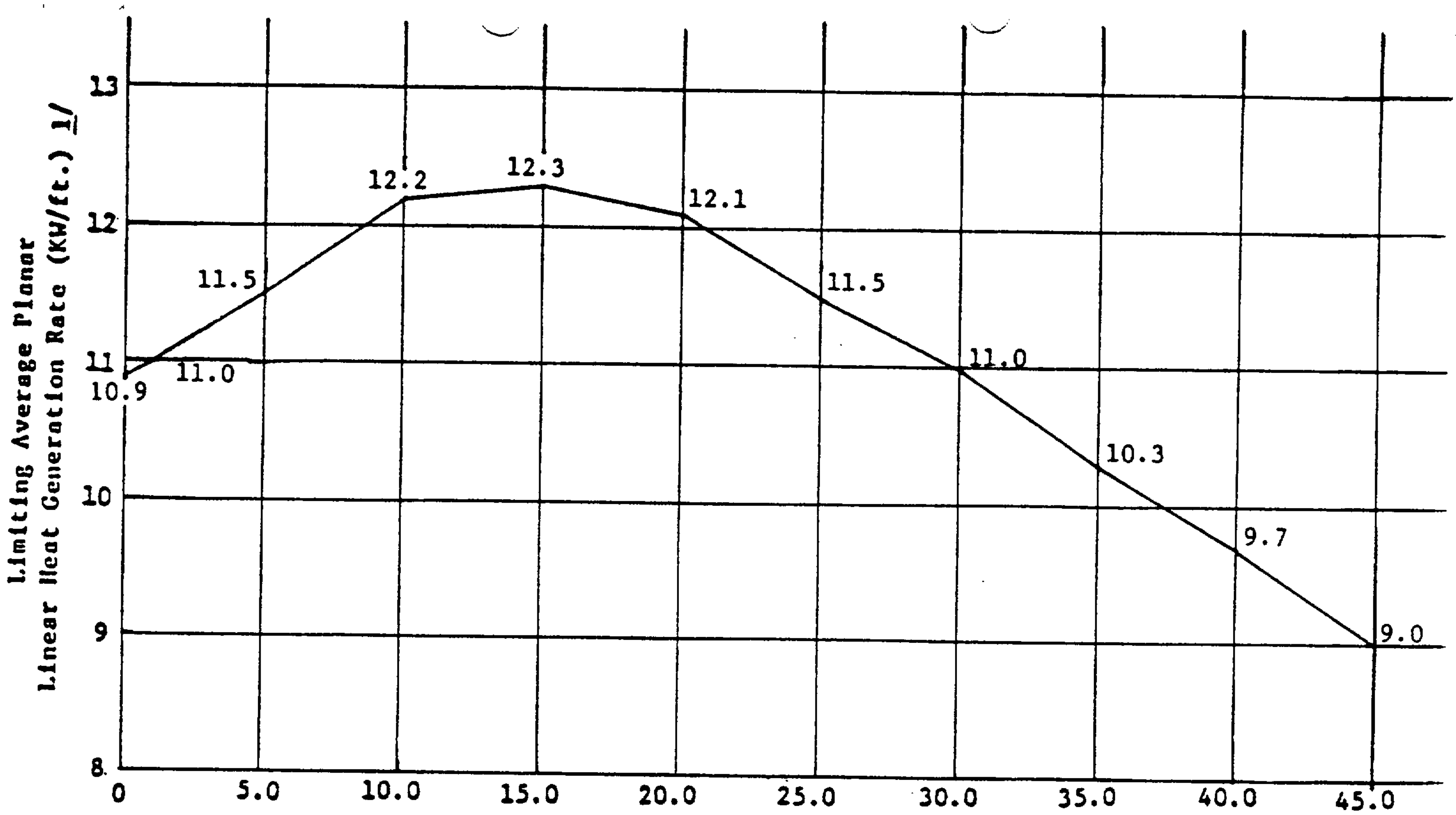
DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: BP/P8DRB301L FIGURE 3.12-6



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

* 1 GWd/t = 1000 MWd/t

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

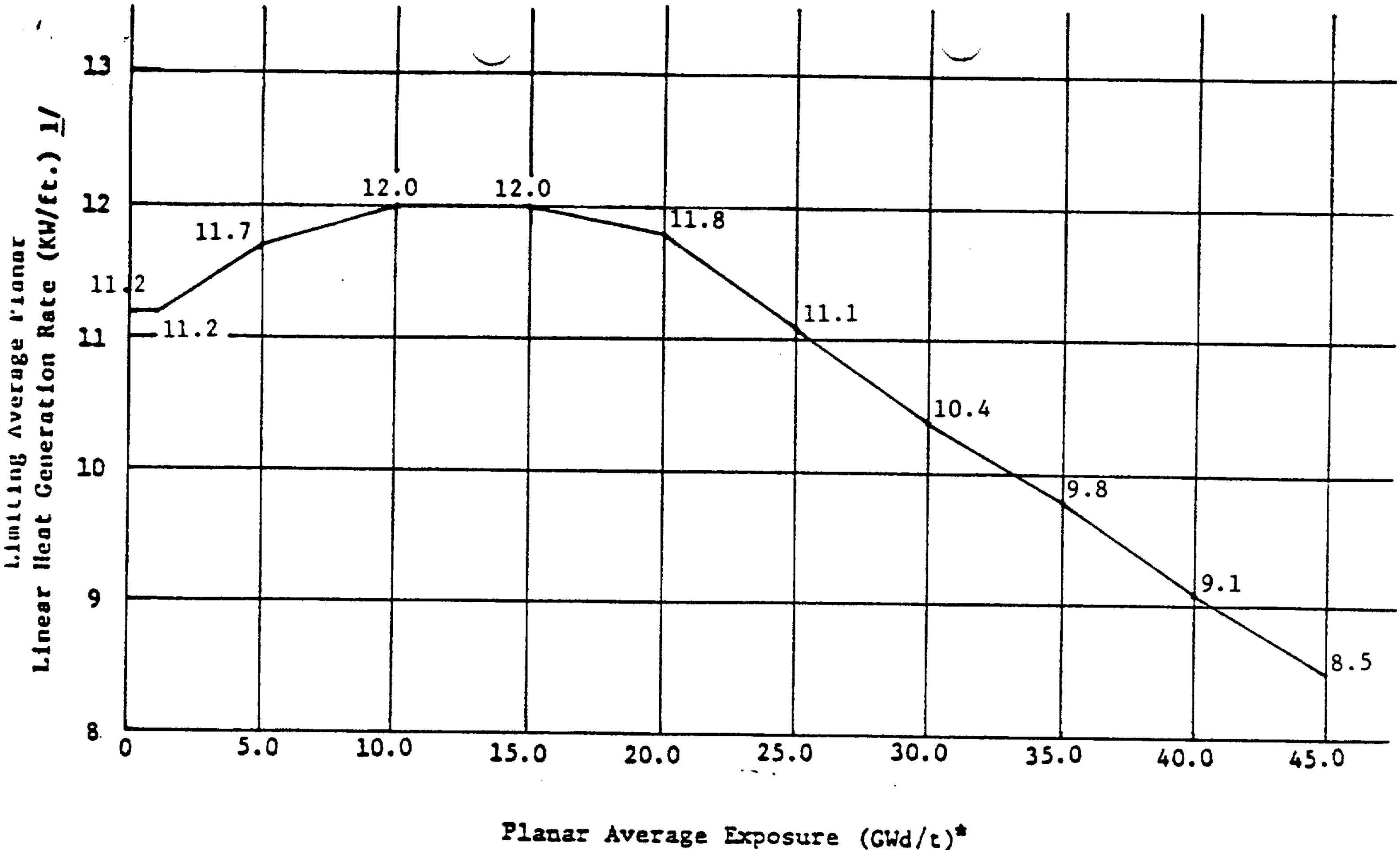


Planar Average Exposure (GWd/t)*

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

* 1 GWd/t = 1000 MWd/t

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

* 1 GWd/t = 1000 MWd/t

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

FUEL TYPE: P8DRB284H

FIGURE 3.12-9

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 115 TO LICENSE NO. DPR-49

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 Introduction

The Duane Arnold Energy Center (DAEC) was designed and constructed to operate at a steady state core power of 1658 MWt. The staff reviewed the Final Safety Analysis Report (FSAR) and the Environmental Report (ER) and issued a Safety Evaluation Report (SER) and a Final Environmental Statement (FES), addressing the operating power level of 1658 MWt. The staff stated in the SER that the DAEC power be restricted to 1593 MWt until the licensee satisfactorily resolved the power ascension program issues. Accordingly, the DAEC license was issued for a maximum power level of 1658 MWt but the rated power was restricted in the Technical Specifications to 1593 MWt. The Iowa Electric Light and Power Company (licensee), by letter dated August 17, 1984, proposed a revised power ascension program consistent with the staff recommendations in the DAEC SER, and requested an increase in their rated power from 1593 MWt to 1658 MWt. Since the issuance of the DAEC license in February 1974, several changes have occurred in the regulations and the regulatory bases. The Commission has since issued the Appendix K to 10 CFR 50, outlining the calculational models to be used to satisfy the Emergency Core Cooling System (ECCS) requirements. The Appendix K requires that the loss of coolant accident (LOCA) analyses be performed at 102% of the maximum authorized power level. Additionally, the staff's review philosophy outlined in the Standard Review Plan (SRP) was revised in 1980. Responding to these changes, the licensee has redone the LOCA analyses for a power level two percent above the proposed maximum power of 1658 MWt. The licensee has also evaluated the impact of all the changes on the reactor systems performance, core performance, engineered safety features, and the design basis accidents and their consequences. All evaluations have been done using the guidance contained in the revised Standard Review Plan.

2.0 Evaluation

The staff reviewed the licensee's application and the supporting analyses and evaluations. The staff's evaluations are summarized as follows:

PLANT HEAT BALANCE

To determine the necessary plant initial conditions and input parameters, heat balances at the proposed power level of 1658 MWt and 1691 MWt were performed. Heat balance parameters are used as initial conditions and input parameters in various plant analyses-LOCA, thermal limits, reactor overpressure protection, reactor internal pressure differences and containment evaluation.

An increase in thermal power from 1593 MWt to 1658 MWt will require an increase in the reactor nominal operating pressure from 1005 psig to 1025 psig. This increase in pressure will compensate for the additional pressure drop caused by the increased steam flow to the turbine and will provide sufficient control margin for the turbine control valves so that continuous stable operation will be maintained.

This new reactor dome pressure will affect reactor systems which are pressure dependent. Reactor high pressure scram setpoint, ATWS reactor recirculation pump trip and SRV setpoints are affected and will be addressed as the Technical Specification Changes later.

LOSS-OF-COOLANT ACCIDENT

The LOCA analyses were performed using Appendix K approved ECCS evaluation models/methodology to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 at uprated power conditions. The ECCS performance was evaluated for the entire LOCA break spectrum. The ECCS performance evaluation included the most limiting break size, break location, and single failure combinations. The LOCA analysis results improved from the previous analyses due to credit taken for a full-core of Drilled Lower Tie Plates which permit a substantial amount of backflow leakage over the range of differential pressures expected. The licensee has demonstrated compliance with the ECCS acceptance criteria as follows.

(1)	Peak Cladding Temperature (PCT)	1959°F	(2200°F allowable)
(2)	Maximum cladding oxidation	1.1%	(17% allowable)
(3)	Maximum total hydrogen generation	0.08%	(1.0% allowable)

(Recirculation Suction Line Break-LPCI Injection Valve Failure)

(4) A coolable geometry is demonstrated by the compliance with the criteria for the PCT and the maximum cladding oxidation.

(5) Long-term cooling is ensured by the use of redundant systems that have adequate water sources available to remove the decay heat generated within the reactor core and transfer the heat to the ultimate heat sink.

The results of the LOCA analyses demonstrate that the ECCS will perform its function in an acceptable manner and meet the 10 CFR 50.46 acceptance criteria.

REACTOR VESSEL OVERPRESSURE PROTECTION

The DAEC pressure relief system includes two spring safety valves and six dual function safety/relief valves. The pressure relief system was designed in compliance with ASME Code Section III. Article NB-7000 which requires that the maximum pressure reached during the most severe pressure transient be less than the 110% of the reactor vessel design pressure. This pressure limit is 1375 psig.

The two spring safety valves are set to actuate at 1240 psig. Due to the increase in dome pressure of 10 psi, the proposed setpoints for the six dual function SRVs are 1110, 1120, 1130 (two valves) and 1140 (two valves) psig. The DAEC nominal operating pressure is 1025 psig.

The MSIV closure with flux scram (assuming position switch scram failure) transient was analyzed to determine the peak system pressure. The SRVs open to limit the pressure rise at the bottom of the vessel to 1275 psig which is below the allowable maximum pressure of 1375 psig.

The DAEC pressure relief system has adequate simmer and overpressure protection margin during plant operation at uprated power conditions.

ABNORMAL OPERATIONAL TRANSIENTS AND MCPR OPERATING LIMITS

To maintain fuel cladding integrity, the reactor core is designed with appropriate margin during any conditions of normal operation including the effects of anticipated operating occurrences. The minimum value of the critical power ratio reached during the transient should be such that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. The limiting value of the minimum critical power ratio is called the safety limit.

The design calculation of the safety limit MCPR is based on a Monte Carlo analysis (ODYN) of core performance in a limiting configuration and takes into account both performance monitoring uncertainties and calculation uncertainties. The safety limit MCPR is 1.07 for the DAEC.

To determine the MCPR operating limit for the DAEC at the uprated conditions, the most limiting abnormal operational transients were considered in the analysis.

These transients are:

(1) Load Rejection without Bypass
(2) Turbine Trip without Bypass
(3) Loss of Feedwater Heating
(4) Feedwater Controller Failure
(5) Inadvertent Startup of HPCI Pump

The DAEC uprated MCPR operating limit is obtained by addition of the absolute maximum CPR value (including any imposed adjustment factors) for the most limiting transient postulated to occur at the plant from uprated conditions, to the fuel cladding integrity safety limit.

The ODYN Option A MCPR operating limit for the DAEC, Cycle 8, is 1.28 and the ODYN Option B MCPR operating limit is 1.26.

The MCPR operating limit ensures that the fuel cladding integrity would be maintained for any abnormal operational transient.

SRV LOW-LOW-SET SYSTEM (LLS)

The LLS logic system employs two non-ADS SRVs to reduce subsequent actuations of SRVs during plant abnormal transients or small break LOCAs. The LLS system would mitigate the induced thrust loads on the SRV discharge line resulting from SRVs subsequent actuations.

For the DAEC uprated power, the two limiting events, (1) Isolation by MSIV closure and (2) Small Break with Isolation Due to Loss of Offsite Power were analyzed. The results of the analysis indicate that the minimum time between SRV actuation for both events exceeds the minimum acceptable value of 3.7 seconds mitigating thrust loads.

We conclude that there is no adverse effect on the LLS logic system due to power uprate.

FUEL MECHANICAL DESIGN

The fuel used in the DAEC is the standard General Electric design which has been described in the GESTAR document (Reference 3). This fuel design has been approved for use in BWR reactors from BWR/2 through BWR/6 designs with power densities which encompass that of DAEC at the uprated power. We therefore find its use acceptable. The safety limits for the fuel

(MCPR and clad strain limits) are established generically as described in Reference 3. This report has been reviewed by the staff and approved (Reference 4). We conclude that these limits apply to DAEC with uprated power.

NUCLEAR DESIGN

The methods and techniques employed in the nuclear design of DAEC at uprated power are described in Reference 3. These methods have been approved by the staff for use in the design and analysis of BWR cores, including those having core power densities in the range of uprated DAEC. We conclude that the nuclear design of the uprated core is acceptable.

THERMAL-HYDRAULIC DESIGN

The methods and techniques used to perform the thermal-hydraulic design of the uprated DAEC core are described in Reference 3 which has been approved by the staff for such application.

The value of 1.07 for the MCPR safety limit is a generic value applicable to reloaded BWR reactors having 8x8, 8x8R, P8x8R, and/or BP8x8R fuel. The methods and techniques used to obtain the safety limit value are described in Reference 3. The value of the safety limit MCPR depends on uncertainties in the thermal hydraulic parameters of the core and on the uncertainty in the critical heat flux correlation (GEXL). Since these quantities are not affected by the power uprate, we conclude that the safety limit MCPR value of 1.07 is still acceptable for DAEC.

The operating limit MCPR is obtained by an analysis of the transients to obtain their effect on the core critical power ratio (ΔCPR). The maximum value of ΔCPR is then added to the safety limit MCPR to obtain the operating limit. The maximum value of ΔCPR includes multipliers which are required to account for uncertainties in the transient calculation methods.

The methods and techniques employed in obtaining the operating limit MCPR are described in Reference 3. These methods are applicable generically to BWR reload analysis. We find that they are acceptable for the uprated DAEC.

The K_f curves are plots of multiplying factors to be applied to the operating limit MCPR for core conditions less than rated flow. The rated flow is the same for the uprated as for the present DAEC. However, the steam flow will be greater by a factor of $\sim$1.048. Since the current K_f curves were calculated for 105 percent of the rated steam flow, the current values of the K_f factors are still applicable.

The thermal-hydraulic stability analysis of the DAEC core has been performed at the proposed uprated power level for previous cycles as well as for Cycle 8. Therefore the power uprate does not affect the analysis. The effects of single loop operation on core thermal-hydraulic performance will be addressed in a separate evaluation.

TRANSIENTS AND ACCIDENTS

Transient and accident analysis methods are described in Reference 3. These are the same methods that have been used in previous cycles and they continue to be acceptable. The applicability of these methods is not dependent on reactor power within the range of BWR designs. Accordingly, the transient and accident analysis are acceptable at the DAEC uprated power.

EXTENDED LOAD LINE LIMIT ANALYSIS

The operation with an extended load line would permit higher power at low flows than the power permitted by operation with the standard load line. This affects the core thermal hydraulic stability analysis and the initial conditions for certain transients. The power uprate will have an additional effect on these parameters. The effect of the power uprate on the Extended Load Line Analysis will be treated in the review of that analysis which will be the subject of a separate evaluation.

CONTAINMENT RESPONSE

The containment design basis accident (DBA) is an instantaneous double-ended guillotine break of the recirculation pump suction line which is postulated to occur. Analyses to determine the DAEC containment short-term accident response were performed at an initial power condition corresponding to 102% of the uprated power. The differences in the peak calculated values for the drywell and wetwell pressures and temperatures between the uprated power conditions of 102% (1691 MWt) and the current rated power at 102% (1625 MWt) are negligible.

The licensee did not perform any revised analyses for the long-term containment response transients at uprated power, because of the large margins in the original analyses between the predicted and the containment design temperatures and pressures. However, the licensee did reanalyze the maximum local pool temperature at the uprated power condition and determined that the local pool temperature increased by about 1°F.

The licensee performed the peak containment pressure and temperature analysis with improved containment mass and energy release rate methodology. The staff concludes that the release rate methodology is acceptable (see NUREG-0661, containment mass and energy release). The new peak calculated pressure and temperature values are lower than the previous analysis as presented in the updated FSAR.

Based on our review, we find that the post-LOCA containment environmental conditions are not significantly affected by the proposed increase in rated power. Moreover, the licensee's latest analysis shows an increase in the margin of safety due to use of the improved calculational method.

CONSEQUENCES OF DESIGN BASIS ACCIDENT

The staff evaluation of the DAEC design basis accidents and their radiological consequences were reported in the SER dated January 1973. In that report, the consequences of the design basis accidents were calculated for the higher core power level of 1658 MWt. The staff analysis of the consequences was based on the assumption that the fuel burnup would not exceed 3800 MWd/metric ton. The licensee has assured us that the burnup of the DAEC fuel bundles will not exceed 28,500 batch average MWd/metric ton. We therefore conclude that our conclusions for accident consequences reported in our January 1973 SER remain unchanged excepting the consequences of a LOCA. The model for LOCA calculations were revised in 10 CFR 50 Appendix K subsequent to the issuance of our SER. The Appendix K requires that the consequence calculations be based on an assumed stretched power of 102% of the rated power or 1691 MWt. The staff therefore recalculated the radiological consequences of a design basis LOCA and found that the increase in the doses resulting from a LOCA will be less than 2 rems, and the resulting doses would meet all dose guidelines of 10 CFR 100.

The staff therefore concludes that the engineered safety feature designs and performances are acceptable for DAEC operation at a power level of 1658 MWt.

THE ENVIRONMENTAL IMPACTS OF OPERATION AT 1658 MWt

The operating license stage Final Environmental Statement (FES-OL) of March 1973 evaluated the potential operational impacts of DAEC at a power level of 1658 MWt. In 1981, NRC published the results of a contractor study (Reference 5) that evaluated the observed impacts of DAEC during the period 1975-1980 and compared those against the predicted impact in the construction permit stage Final Environmental Statement (FES-CP). The operation of DAEC was not found to cause any major long-term changes in aquatic resources of the site vicinity. An observed acceptable level of impact at current power, along with a predicted acceptable impact at 1658 MWt power are sufficient for us to conclude that no further review of the environmental impacts is necessary.

TECHNICAL SPECIFICATION CHANGES

1. Reactor Systems Performance

Increase in thermal power from 1593 MWt to 1658 MWt will require an increase in the nominal operating pressure/reactor dome pressure from 1005 psig to 1025 psig. This 20 psi increase in reactor operating pressure will affect reactor systems which are pressure dependent. Reactor High Pressure Scram setpoint, ATWS reactor recirculation pump trip and SRV setpoints are affected and are modified to maintain the margin above reactor operating pressure listed below:

	Uprated Power 1658 MWt (psig)	Rated Power 1593 MWt (psig)
Nominal Operating Pressure or Reactor Dome Pressure	1025	1005
Reactor High Pressure Scram Setpoint	1055	1035
ATWS Reactor Recirculation Pump Trip	1140	1120
SRV Setpoints		
Valve No. 1	1110	1080
2	1120	1090
3	1130	1100
4	1130	1100
5	1140	1110
6	1140	1110

These pressure setpoints changes are necessary for the uprated power conditions and are acceptable. The SRV pressure setpoints will improve the SRV simmer margin.

In addition, the licensee has requested that the limiting condition for operation, in the case of one ADS valve inoperable, be extended from 7 days to 30 days.

GE performed a small break LOCA analysis assuming one ADS valve inoperable. The results of the analysis meet the ECCS acceptance criteria. We find the limiting condition for operation (LCO) request acceptable.

2. Performance of the Reactor Core

The most significant reactor core related changes arising from the proposed power uprate are those to the protection system setpoints. These setpoints are adjusted to restore the operating margins to their current values or, in some cases, to increase the margins. The following core related Technical Specification changes have been evaluated:

1. Revised flow-dependent APRM scram and rod block settings, and

2. Revised Rod Block Monitor Setting

   Note that the APRM fixed trip setpoint does not require resetting since it is expressed as a percent of rated power.

The APRM trip setpoints are consistent with the analysis described above and are acceptable. The licensee has opted to use a generic cycle independent Rod Block Monitor setpoint (105 percent of rated power at rated flow), resulting in a ΔCPR of 0.19 for this event. The generic analysis is applicable for all BWR designs which use the Rod Block Monitor and is acceptable for DAEC.

3. Containment Performance

The leak rate testing for the primary containment is based upon the analysis of containment response following a DBA LOCA. As a result of the licensee's reanalysis, the peak calculated pressure is now 43 psig (whereas 54 psig was indicated in the updated FSAR). The licensee wished to incorporate the new value in the Technical Specifications. Accordingly, the licensee proposed certain changes to Section 3.7 of the Technical Specifications for the Duane Arnold Energy Center. The proposed changes are dominated by the revisions to the leak rate test pressure.

Another proposed change to the Technical Specifications involves revision of the temperature limit for conducting visual inspections of the suppression pool to be consistent with the modifications made for the Mark I improvement program. The remaining changes pertain to Section 3.7, which are clarifications or enhancements to the text.

3.0 Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 References

1. Duane Arnold Energy Center Power Uprate, General Electric Report NEDC-30603-P-1, October 1984.

2. Duane Arnold Energy Center Power Uprate, General Electric Report NEDO-30603, July 1984.

3. GESTAR II, General Electric Standard Application for Reactor Fuel, NEDO-24011-A, Latest Approved Version.

4. Approval Letter, D. G. Eisenhut (NRC) to R. Gridely (GE) dated May 12, 1978 and Supplements thereto, forming Appendix C to Reference 3.

5. NUREG/CR-2337, "Aquatic impacts from operation of three midwestern nuclear power stations. Duane Arnold Energy Center, Unit No. 1. Environmental Appraisal Report," November 1981.

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