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Docket No. 50-331

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The amendment consists of changes to the Technical Specifications in response to your application dated May 30, 1975, and supplemented by letters dated January 8, 1976, February 13, 1976, March 9, 1976, and March 12, 1976.

The amendment will modify the Technical Specifications to permit operation of the facility (1) using a partial fuel loading of 8 x 8 fuel assemblies and (2) with modification to the rod sequence control system.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 19
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

See Jacket for Enclosures

OFFICE	ORB#3	ORB#3	ORB#3	OELD	AD:ORs/DOR	
SURNAME	CParrish:kmf	WPaulson	GLear	H D Paton	KRGoller	
DATE	3/17/76	3/17/76	3/17/76	3/18/76	3/19/76	

Iowa Electric Light & Power Company - -

cc:

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

Cedar Rapids Public Library
426 Third Avenue, S. E.
Cedar Rapids, Iowa 52401

Office for Planning and Programming
523 East 12th Street
Des Moines, Iowa 50319

Mr. Dudley Henderson
Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

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

March 19, 1976

Docket No. 50-331

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ATTN: Mr. Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

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George Lear

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

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensees) dated May 30, 1975, and supplements submitted January 8, 1976, February 13, 1976, March 9, 1976, and March 12, 1976, 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Date of Issuance: March 19, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 19 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, which revised Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County, Iowa. The amendment is effective as of its date of issuance.

The amendment will modify the Technical Specifications to permit operation of the facility (1) using a partial fuel loading of 8 x 8 fuel assemblies and (2) with modification to the rod sequence control system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on August 15, 1975 (40 F.R. 34485). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant

to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 30, 1975 (supplemented by letters dated January 8, 1976, February 13, 1976, March 9, 1976, and March 12, 1976), (2) Amendment No. 19 to License No. DPR-49, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Cedar Rapids Public Library, 426 Third Avenue, S. E., Cedar Rapids, Iowa 52401.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 19 day of March 1976

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

3/19/78

ATTACHMENT TO LICENSE AMENDMENT NO. 19

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace pages v, vii, 1.1-1, 1.1-2, 1.1-3, 1.1-6, 1.1-10 through 1.1-13, 1.1-15, 1.1-17, 1.1-18, 1.1-20, 1.1-21, 1.1-24, 1.2-5, 1.2-6, 3.1-3, 3.2-16, 3.2-17, 3.2-41, 3.2-42, 3.3-1 through 3.3-18, 3.6-23, 3.6-24, 3.12-1 through 3.12-9, 3.12-11, and 3.12-16 with the attached revised pages. No change has been made on Figure 2.1-1.

Delete Figure 2.1-2.

TECHNICAL SPECIFICATIONS

LIST OF TABLES

Table Number	Title	Page
1.1-1	Uncertainties Used in the Determination of the Fuel Cladding Safety Limit	1.1-10
1.1-2	Nominal Values of Parameters Used in the Statistical Analysis of Fuel Cladding Integrity Safety Limit	1.1-11
1.1-4	R-Factor Distribution Used in GETAB Statistical Analysis	1.1-13
3.1-1	Reactor Protection System (SCRAM) Instrumentation Requirements	3.1-3
4.1-1	Reactor Protection System (SCRAM) Instrument Functional Tests	3.1-8
4.1-2	Reactor Protection System (SCRAM) Instrument Calibration	3.1-12
3.2-A	Instrumentation that Initiates Primary Containment Isolation	3.2-5
3.2-B	Instrumentation that Initiates or Controls the Core and Containment Spray Systems	3.2-8
3.2-C	Instrumentation that Initiates Control Rod Blocks	3.2-16
3.2-D	Radiation Monitoring Systems that Initiate and/or Isolate Systems	3.2-19
3.2-E	Instrumentation that Monitors Drywell Leak Detection	3.2-20
3.2-F	Surveillance Instrumentation	3.2-21
3.2-G	Instrumentation that Initiates Recirculation Pump Trip	3.2-23
4.2-A	Minimum Test and Calibration Frequency for PCIS	3.2-24
4.2-B	Minimum Test and Calibration Frequency for CSCS	3.2-26
4.2-C	Minimum Test and Calibration Frequency for Control Rod Blocks Actuation	3.2-38

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

Figure Number	Title
1.1-1	Power/Flow Map
2.1-1	APRM Flow Biased Scram and Rod Blocks
2.1-2	(DELETED)
4.1-1	Instrument Test Interval Determination Curves
4.2-2	Probability of System Unavailability Vs. Test Interval
3.4-1	Sodium Pentaborate Solution Volume Concentration Requirements
3.4-2	Saturation Temperature of Sodium Pentaborate Solution
3.6-1	DAEC Operating Limits
6.2-1	DAEC Nuclear Plant Staffing
3.12-1	K_f as a Function of Core Flow
3.12-2	Limiting Average Planar Linear Heat Generation Rate (Fuel Types 1 and 3)
3.12-3	Limiting Average Planar Linear Heat Generation Rate (Fuel Type 2)
3.12-4	Limiting Average Planar Linear Heat Generation Rate (Fuel Type 4)

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. APRM High Flux Scram When In Run Mode.

For operation with a peaking factor less than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram trip setpoint shall be as shown on Fig. 2.1-1 and shall be:

$$S \leq (0.66W + 54)$$

with a maximum setpoint of 120% rated power at 100% rated recirculation flow or greater.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
C. Power Transient To ensure that the Safety Limits established in Specification 1.1.A and 1.1.B are not exceeded, each required scram shall be initiated by its primary source signal. A Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal. D. With irradiated fuel in the reactor vessel, the water level shall not be less than 12 in. above the top of the normal active fuel zone.	Where: S = Setting in percent of rated power (1,593 MWt). W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to 49×10^6 lb/hr core flow. MTPF = Actual Maximum Total peaking factor. For a peaking factor greater than 2.61 (7 x 7 array) or 2.43 (8 x 8 array), the APRM scram setpoint shall be: $S \leq (0.66\ W + 54)\ \frac{(\ *\ )}{MTPF}$ NOTE: These settings assume operation within the basic thermal design criteria. These criteria are LHGR $\leq$ 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array) and MCPR $\geq$ 1.40 K_f (7 x 7 array) or $\geq$ 1.50 K_f (8 x 8 array), 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. 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. * 2.61 (7 x 7 array) or 2.43 (8 x 8 array)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	3. APRM Rod Block When in Run Mode. For operation with combinations of power and peak flux below the curve for total peaking factor of * in Fig. 2.1-2 the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be: $S \leq (0.66\ W + 42)$ The definitions used above for the APRM scram trip apply. For a 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: $S \leq (0.66\ W + 42)\ \frac{(\ *\ )}{MTPF}$
	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.

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

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass Flux:	0.1 to 1.25 10^6 lb/hr	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Max./Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array:	16, 64 Rods in an 8 x 8 array	
	49 Rods in a 7 x 7 array	

The required input to the statistical model is the uncertainties listed in Table 1.1-1, the nominal values of the core parameters listed in Table 1.1-2, and the relative assembly power distribution shown in Figure 1.1-2. Table 1.104 shows the R-Factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The basis for the uncertainties in the core parameters is given in NEDO-20340[2], and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical

TABLE 1.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
Bypass Void Effect on TIP	3.85 (core midplane) 5.05 (core exit)
R-Factor	1.6
Critical Power	3.6

TABLE 1.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3,293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (7 x 7 Array) 1.100 (8 x 8 Array)

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TABLE 1.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7 x 7 Rod Array

R-Factor	Rod Sequence No.
1.098	1
1.083	2
1.075	3
1.062	4
1.052	5
1.042	6
1.042	7
≤ 1.027	8 thru 49

8 x 8 Rod Array

1.100	1
1.100	2
1.095	3
1.095	4
1.093	5
1.093	6
1.092	7
≤ 1.077	8 thru 63

insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 25.6% and 46.4% insertion. By the time the rods are 67.2% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 4.7% and 88.1% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of 1.40 (7 x 7 array) or 1.50 (8 x 8 array) is conservatively assumed to exist prior to initiation of the transients.

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

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 1658 MWt.

ii. The licensed maximum power level is 1658 MWt.

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

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

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 $\geq$ 1.40 (7 x 7 array) or $\geq$ 1.50 (8 x 8 array).

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

For operation in these modes the APRM scram setting of 15 percent of rated power and the IRM High Flux Scram provide adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod

worth minimizer and the Rod Sequence Control System. Worths of individual rods are very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

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

3. APRM Rod Block (Run Mode)

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given power level at constant recirculation flow rate, and thus prevents a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents excessive reactor power level increase resulting from control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases

In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents has been analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is by-passed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

B. Scram and Isolation on Reactor Low Water Level

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. Analyses show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 21 inches below the normal operating range and is thus adequate to avoid spurious scrams.

C. Scram - Turbine Stop Valve Closure

The turbine stop-valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves.

With a scram setting at 10 percent of valve closure, the resultant increase in surface heat flux is such that MCPR remains above 1.07 even during the worst case transient that assumes the turbine bypass is closed. This scram is by-passed when turbine steam flow is below 30 percent of rated, as measured by the turbine first stage pressure.

D. Turbine Control Valve Fast Closure (Loss of Control Oil Pressure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection. It prevents MCPR from becoming less than 1.07 for this transient.

E. F. and J. Main Steam Line Isolation on Low Pressure, Low Condenser Vacuum, and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 880 psig has been provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity. Operation of the reactor at pressures lower than 880 psig requires that the reactor mode switch be in the STARTUP position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients

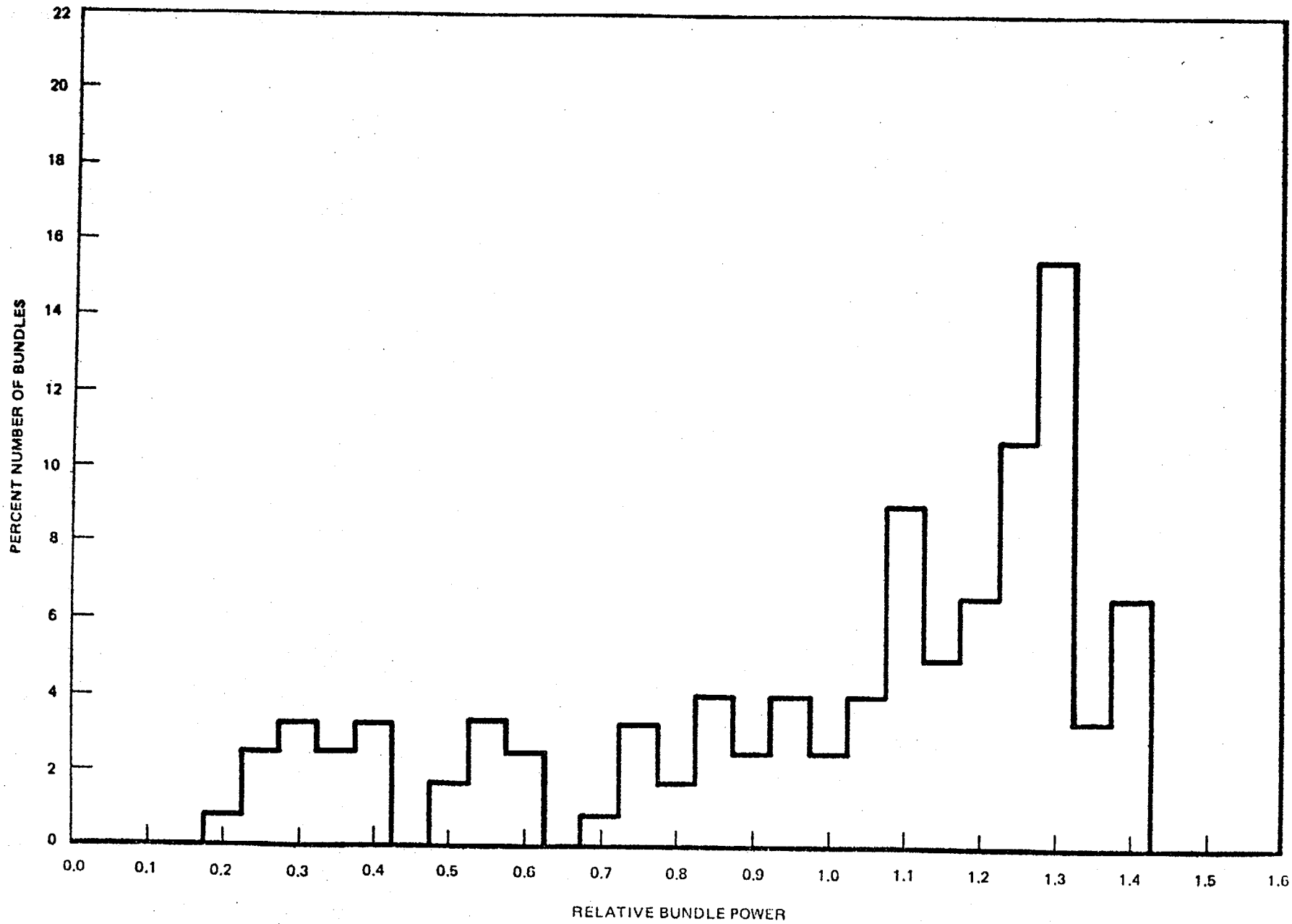


Figure 1.1-2. Relative Bundle Power Histogram for Power Distribution Used in Statistical Analysis

2.2 BASES

Reactor Coolant System Integrity

The pressure relief system for DAEC has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis in Subsection 4.4 of the FSAR which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME Code requirements are presented in Subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Report in FSAR Amendment No. 3 (Response to AEC Question N.1.1) and DAEC Reload Number One License Submittal.

Six relief valves and two safety valves have been installed. The 6 relief valves have a total capacity of 68.4% of design flow calculated at a pressure of 1090 psig (the lowest release valve setpoint). The 2 safety valves have a total capacity of 18.7% calculated at a pressure of 1240 psig (the safety valve setpoint). The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation

valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1291 psig (at vessel bottom) if a flux scram is assumed. This results in 80 psig to the code allowable overpressure limit of 1375 psig. In addition, the same event was analyzed to determine the number of installed valves which must open to limit peak pressure to 1350 psig (25 psig margin). The results of this analysis show that four valves must open if a neutron flux scram is assumed.

To meet the second design basis, the total safety/relief capacity has been divided into 6 relief valves and 2 safety valves. The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraph 14.5.1.2 and Figure 14.5-3. This analysis shows that the 6 relief valves limit pressure at the safety valves to 1196 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1234 psig which is 141 psig below the allowed vessel overpressure of 1375 psig.

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	(.66W+54) (* /P.F.) (11) (12)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤ 1035 psig	X(8)	X	X	4 Instrument Channels	A

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

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)	$\leq (0.66W + 42)\ (\frac{*}{P.F})$ (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)	$\leq (0.66W + 41)\ (\frac{*}{TPF})$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

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

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 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 design. Trip level setting is in percent of rated power (1593 MWt). Refer to Limiting Safety System Settings for variation with peaking factors. Peaking factor is applicable only when it exceeds 2.61 (7 x 7 array) or 2.43 (8 x 8 array).

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

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

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, six IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criterion is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation

at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RMB rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 5 indicated on scale for APRM's and 5/125 full scale for IRM's.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3 REACTIVITY CONTROL	4.3 REACTIVITY CONTROL
Applicability:	Applicability:
Applies to the operational status of the control rod system.	Applies to the surveillance requirements of the control rod system.
Objective:	Objective:
To assure the ability of the control rod system to control reactivity.	To verify the ability of the control rod system to control reactivity.
Specification:	Specification:
Reactivity Limitations	A. Reactivity Limitations
Reactivity margin - core loading	1. Reactivity margin - core loading
A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.	Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38 Δ k/k that the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.
2. Control Rod Exercise	2. Control Rod Exercise
a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing.	a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30% power. This test shall be performed at least once per 24 hours in the event power operation is ontinuing above 30% power with two or more inoperable control rods or in the event power operation is continuing above 30% power with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than two and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such positions that Specification 3.3.A.1 is met.	b. A second licensed operator shall verify the conformance to Specification 3.3.A.2d before a rod may be bypassed in the Rod Sequence Control System.
c. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.	c. Once per week when the plant is in operation, check status of pressure and level alarms for each CRD accumulator.
d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full-in on insertion or full-out on withdrawal).	
e. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.	
f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
1) In addition, whenever the reactor is in the startup or run mode no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this Specification cannot be met, the reactor shall not be started, or if at power, the reactor shall be brought to a cold shutdown condition within 24 hours.	
2) All rods within a notch group containing an inoperable rod will be positioned within 1 (one) notch of the inoperable rod whenever the Rod Sequence Control System is required.	
g. During reactor power operation the number of inoperable control rods shall not exceed 8. Specification 3.3.A.1 must be met at all times.	
B. <u>Control Rods</u>	B. <u>Control Rods</u>
1. Each control rod shall be coupled to its drive and have rod position indication available for the "full in" and "full out" position or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.	1. The coupling integrity shall be verified for each withdrawn control rod as follows:
	a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation and rod position indication for the "full in" and "full out" position. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 30% power shall be performed to verify instrumentation response.
	b. When the rod is fully withdrawn the first time after each refueling outage or after CRD maintenance, observe that the drive does not go to the overtravel position.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	c. During each refueling outage observe that any drive which has been uncoupled from and subsequently recoupled to its control rod does not go to the overtravel position.
2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.	2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.
3.a Whenever the reactor is in the startup or run mode below 30% rated power, the rod sequence control system shall be operable. If the system is determined to be inoperable during power descent in accordance with checks in Specification 4.3.B.3, power may be increased above 30% rated power.	3. Prior to the start of control rod withdrawal towards criticality and prior to attaining 30% rated power during rod insertion at shutdown, the capability of the Rod Worth Minimizer and Rod Sequence Control System to properly fulfill their functions shall be verified by the following checks:
b. During the shutdown procedure no rod movement is permitted following the testing performed between 35% and 20% power level and the automatic reinstatement of the RSCS restraints at the preset power level. Alignment of rod groups shall be accomplished prior to the reinstatement of the Rod Sequence Control System restraints.	a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests: Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences. Group Notch Portion - Test the six comparator circuits. Go through each comparator inhibit, initiate test, verify error, and reset. After comparator checks initiate test and observe completion of cycle indicated by illumination of test complete light.
c. Whenever the reactor is in the startup or run modes below 30% rated power the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.	
d. Control rod withdrawal sequence shall be established such that the drop of any increment of any one in-sequence control rod would not result in a peak fuel enthalpy greater than 280 cal./gm.	b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 30% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.
	2) The RWM computer on line diagnostic test shall be successfully performed.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
	4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.
	c. When required, the presence of a second licensed operator to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).
a. Both RBM channels shall be operable: or	
b. Control rod withdrawal shall be blocked: or	
c. The operating power level shall be limited so that the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single operable control rod.	

LIMITING CONDITION FOR OPERATION

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Rod Position	% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
46	4.7	0.37
36	25.6	1.10
26	46.4	1.87
06	88.1	3.41

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Rod Position	% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
46	4.7	0.39
36	25.6	1.17
26	46.4	1.98
06	88.1	3.62

3. Maximum scram insertion time for 90% insertion of any operable control rod should not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. After each refueling outage all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature) and the requirements of Specification 3.3.B.3.a met. This testing shall be completed prior to exceeding 40% power. Below 30% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. During all scram time testing below 30% power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

2. Whenever such scram time measurements are made (such as when a scram occurs and the computer is operable) an evaluation shall be made to provide reasonable assurance that propose control rod drive performance is being maintained.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.D Reactivity Anomalies The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.	4.3.D Reactivity Anomalies During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.
E. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.	

3.3 and 4.3 BASES

1. Reactivity Limitation

a. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in Subsection 3.4 of the FSAR, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38% $\Delta k/k$ with the analytically determined strongest control rod fully withdrawn.

The value of "R", in units of % $\Delta k/k$, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local K_∞ . Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is

not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% $\Delta k/k$. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

b. Reactivity margin - inoperable control rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically*, it is in a safe position of

*To disarm the drive electrically, four Amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest operable control rod does not insert. Inoperable bypassed rods will be limited within any group to not more than one control rod of a (5 x 5) twenty-five control rod array. If damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially servered collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

In order to perform shutdown margin and control rod drive scram time tests subsequent to any fuel loading operation as

required by the Technical Specifications, the relaxation of the following Rod Sequence Control System restraints is required: (a) The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range. (b) Verify that subsequent to the use of the rod position bypass switches rod movement in the 50% rod density to preset power level range is restricted to the single notch mode.

2. Control Rod Withdrawal

a. Control rod drop accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the rod sequence control system and the rod worth minimizer (RWM).

b. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR and the safety evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated,

since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

c. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal./gm. An enthalpy of 280 cal./gm. is well below the level at which rapid fuel dispersal could occur (i.e., 425 cal./gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 3.6.6, 7.16.5.3, 14.6.2, J.4.13 and Amendments 14 and 15 to the FSAR, NEDO-10527 including Supplements 1, 2 and 3 to NEDO-10527 and the DAEC Reload #1 Licensing Submittal dated January 8, 1976 as amended.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the NRC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed

by power distribution requirements as defined in Sections 3.3.B.5 and 3.3.C.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 10% of full power, the nominal instrument setting is 30% of rated power. Power level for automatic cut-out of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full-out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control rods have been withdrawn (e.g., groups A_{12} and A_{34}), it is demonstrated that the Group Notch mode for the control

drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 30% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

d. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

e. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be

bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit [MCPR = 1.40 (7 x 7 array) or 1.50 (8 x 8 array) and LHGR = 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array)]. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the DAEC Chief Engineer.

3. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the

MCPR from becoming less than 1.07. The limiting power transient is that resulting from a turbine trip without bypass. Analysis of this transient shows that MCPR remains greater than 1.07.

After initial fuel loading and subsequent refuelings when operating above 950 psig all control rods shall be scram tested within the constraints imposed by the Technical Specifications and before the 40% power level is reached.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on DAEC.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an

3.6.D & 4.6.D BASES:

Safety and Relief Valves

The pressure relief system has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of Subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in Subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Report in FSAR Amendment No. 3 (response to AEC Question H.1.1) and DAEC Reload Number One License Submittal.

Six relief valves and two safety valves have been installed as described previously in Subsection 2.2 of these Technical Specifications. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting

the direct scram (valve position scram) results in a maximum vessel pressure of 1291 psig (at vessel bottom) if a flux scram is assumed. This results in 80 psig margin to the code allowable overpressure limit of 1375 psig. In addition, the same event was analyzed to determine the number of installed valves which must open to limit peak pressure to 1350 psig (25 psig margin). The results of this analysis show that five valves must open if a neutron flux scram is assumed or six valves must open if a pressure scram is assumed.

To meet the power generation design basis, the total safety/relief capacity has been divided as described previously in Subsection 2.2. The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraphs 14.5.1.2 and 14.5.1.3. This analysis shows that the 6 relief valves limit pressure at the safety valves to 1196 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1234 psig which is 141 psig below the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to

LIMITING CONDITION FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

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

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-2, 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.

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 $\geq$ 25% rated thermal power.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Linear Heat Generation Rate (LHGR) During reactor power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation: $LHGR_{max} \leq LHGR_d \left[1 - \left\{(\Delta P/P)_{max}(L/LT)\right\}\right]$ $LHGR_d$ = Design LHGR = 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array) $(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026 LT = Total core length - 12 feet L = Axial position above bottom of core. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR 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.	B. Linear Heat Generation Rate (LHGR) The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR) During reactor power operations, MCPR shall be ≥ 1.40 (7 x 7 array) or ≥ 1.50 (8 x 8 array) at rated power and flow. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met. For core flows other than rated the MCPR shall be ≥ 1.40 (7 x 7 array) or ≥ 1.50 (8 x 8 array) times K_f, where K_f is as shown in Figure 3.12-1.	C. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during reactor power 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.
D. Reporting Requirements If any of the limiting values identified in Specifications 3.12.A, B or C are exceeded, a Reportable Occurrence report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.	

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.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 3.12-1.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 3 and in References 4 and 5, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient which determines the required steady state MCPR limit is the turbine trip with failure of the turbine bypass. This transient yields the largest MCPR. When added to the safety limit MCPR of 1.07 the required minimum operating limit MCPR of Specification 3.12.C are obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360[2] and on core parameters shown in Table 4.2.4 of Supplement 1 to Reference 1.

The evaluation of a given transient begins with the system initial parameters shown in Table 4.1 (page 4-7) of Reference 1 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[6]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[7]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

2. MCPR Limits for Core Flows Other than Rated Flow

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.40 (7 x 7 array) or 1.50 (8 x 8 array) will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.12-1 were developed generically and are applicable to all BWR/2, BWR/3 and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the value of K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.12-1 are conservative for Duane Arnold operation because the operating limit MCPR of 1.40 (7 x 7 array) or 1.50 (8 x 8 array) is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

D. Reporting Requirements

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

TABLE 3.12-1

SIGNIFICANT INPUT PARAMETERS TO THE DUANE ARNOLD

LOSS-OF-COOLANT ACCIDENT ANALYSIS

COOLANT PARAMETERS:

Core Thermal Power	1655 MWt* which corresponds to 105% of vessel rated steam output	
Vessel Steam Output	7.16×10^6 Lbm/h which corresponds to 105% of vessel rated steam output	
Vessel Steam Dome Pressure	1055 psia	
Recirculation Line Break Area for Large Breaks (ft^2)	2.641 (DBA)**	1.0
Recirculation Line Break Area for Small Breaks (ft^2)	1.0	0.05 (WSB)***

FUEL PARAMETERS:

Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (KW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio****
Initial Core Type 1 & 3	7 x 7	18.5	1.5	1.18
Initial Core Type 2	7 x 7	18.5	1.5	1.18
7D230 Type 4	7 x 7	18.5	1.5	1.18
8D274	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

* This power level exceeds the Appendix K requirements of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its maximum (technical specification) linear heat generation rate.

** The DBA area includes: the area of the recirculation suction line (2.18 ft^2), the throat area of 8 jet pumps (0.38 ft^2) plus the area of the clearup line (0.08 ft^2).

*** Worst Small Break.

**** Initial value for calculations.

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.

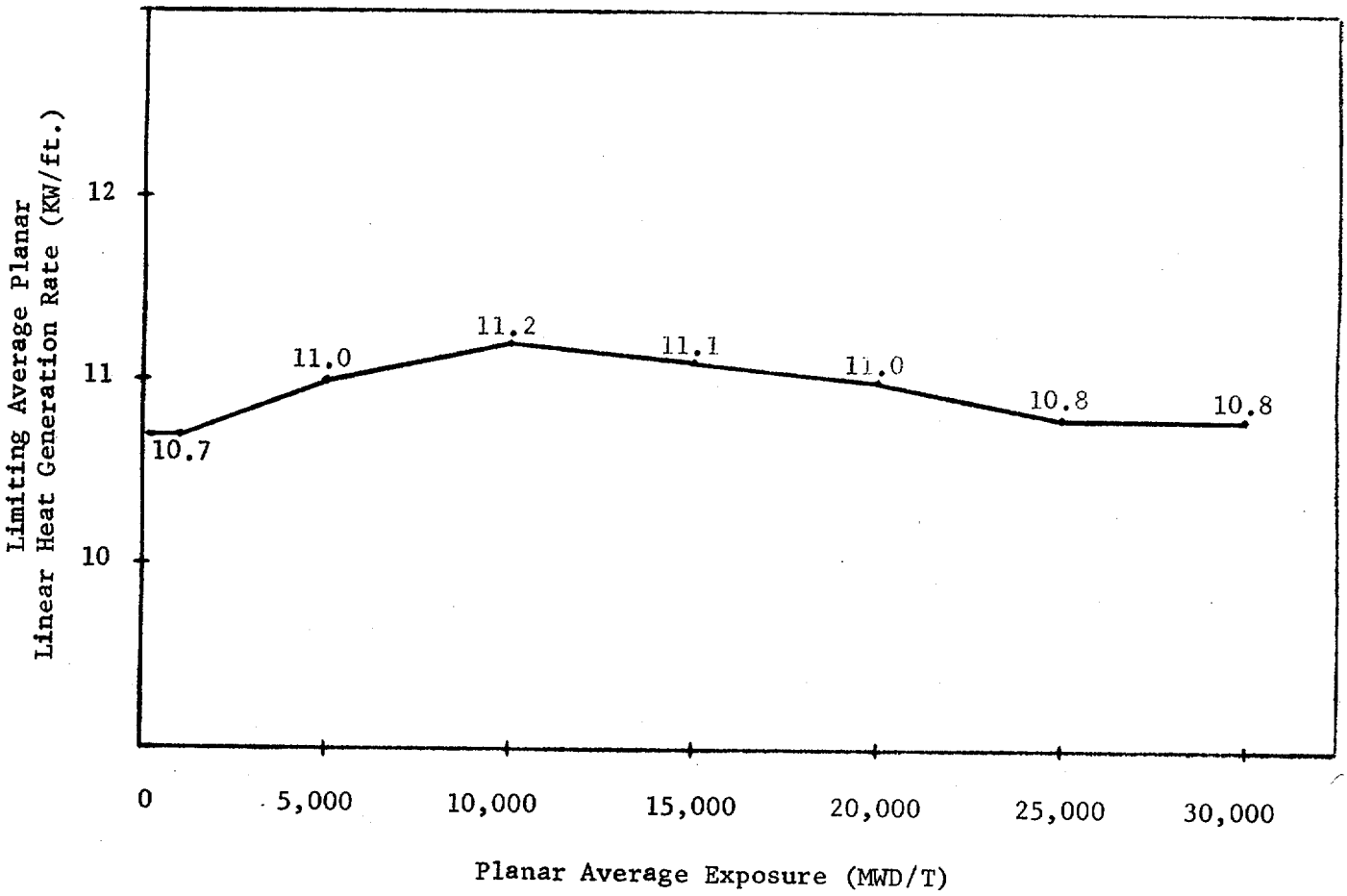
4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

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.



DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

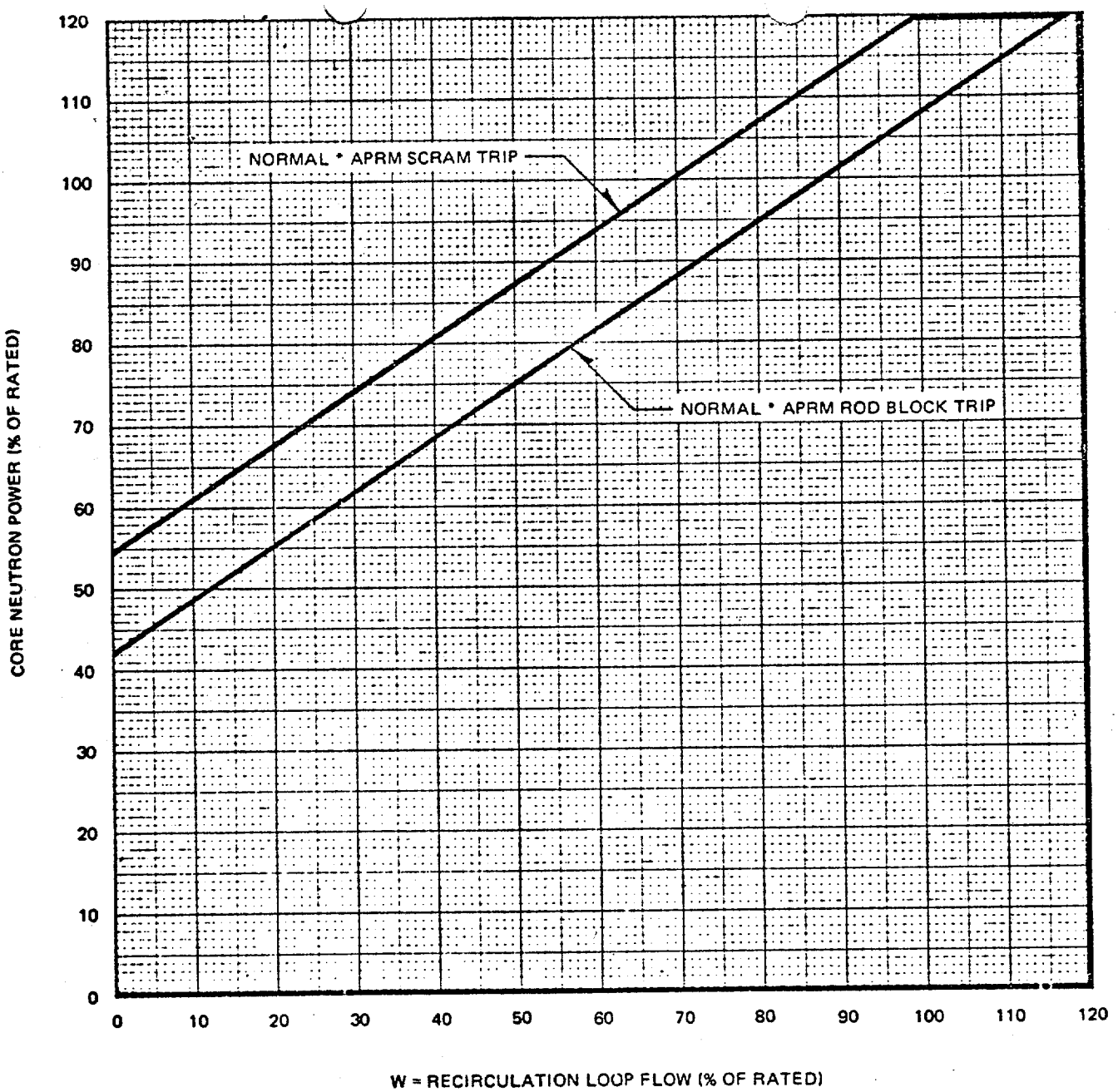
TECHNICAL SPECIFICATIONS

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

FUEL TYPE: 8D274

FIGURE 3.12-5

Amendment No. 19



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

Core Power Vs Recirc Loop Flow

FIGURE 2.1-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 Introduction

By letter dated May 30, 1975 and supplemented by letters dated January 8, 1976, February 13, 1976, March 9, 1976, and March 12, 1976, Iowa Electric Light and Power Company, Central Iowa Power Company, and Corn Belt Power Cooperative (the licensees) requested an amendment to Facility Operating License No. DPR-49. The amendment would modify the Technical Specifications for the Duane Arnold Energy Center (DAEC) to permit operation of the facility (1) using a partial fuel loading of 8 x 8 fuel assemblies, and (2) with modifications to the rod sequence control system (RSCS). Amendment 15 to the DAEC Final Safety Analysis Report (FSAR) describes the proposed changes to the rod sequence control system.

2.0 Evaluation

2.1 Reload with 8 x 8 Fuel Assemblies

The licensees have proposed to reload the Duane Arnold Energy Center Reactor for Cycle Number 2 with up to 96 General Electric Company 8 x 8 fuel assemblies having an average enrichment of 2.74 wt% U-235. The documentation submitted or referenced in support of the proposed reload consisted of: (a) GE BWR Reload-1 licensing submittal for DAEC, NEDO-21082 and supplementary information (References 1 and 2), (b) GE BWR Generic Reload Application for 8 x 8 fuel, NEDO-20360 (Reference 3), and (c) proposed Technical Specification changes (References 4 and 5). During the course of our review, the licensees informed us that 84 8 x 8 fuel assemblies were actually loaded into the core during refueling operations. We have evaluated the licensees submittals regarding Cycle 2 operation.

2.2 Nuclear Characteristics

The information presented in the licensing submittal for the reconstituted core closely follows the guidelines of Appendix A

of Reference 3. The licensees rely heavily on this licensing topical report (NEDO-20360). Although later supplements to this report are undergoing review by us, this topical report has been found acceptable by us for use in support of licensing actions for reactors containing General Electric Company 8 x 8 reload fuel. Eighty-four GE 8x8 reload fuel bundles have been loaded throughout the core. Fifty-two of the reload fuel bundles have a high gadolinia content; thirty-two have a low gadolinia content. The reload fuel incorporates the improved water rod design described in Reference 3. The core contains a total of 368 fuel bundles; thus, about 23 percent of the fuel bundles have been replaced for this reload.

Previously, during an interim reload, four (2.30% average enrichment) 7 x 7 fuel bundles had been loaded. The loading pattern for Cycle 2 may be described as follows: (1) the 8 x 8 reload fuel will be basically scatter loaded, (2) the two rows and columns of fuel bundles which intersect at the center of the core will contain reload fuel and in the reloaded region of the core, about half of the four bundle arrays surrounding a control rod will have two in-line reload fuel bundles, (3) the high gadolinia reload bundles are loaded in the interior of the core while the low gadolinia reload bundles are loaded near the outer periphery of the core, and (4) the interim reload 7 x 7 fuel bundles are located near the core periphery. The data in Reference 1 indicate that the nuclear characteristics of the reload core is similar to the previous core. Thus, the total control system reactivity worth and the temperature, and void dependent behavior of the reconstituted core will not differ significantly for those values which were previously reported for the DAEC reactor.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38%Δk subcritical in the most reactive operating state with the largest worth control rod fully withdrawn and with all other control rods fully inserted. The minimum shutdown margin for Cycle 2 is 1.2%Δk and it occurs at the beginning of the cycle. Thus there is no difference between the beginning of Cycle 2 shutdown margin and the minimum shutdown margin.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.30Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the standby liquid control system.

The Technical Specification requirement for the storage of fuel for the DAEC reactor is that the effective multiplication factor, keff, of the fuel as stored in the fuel storage rack is equal to or less than 0.90 for normal conditions. This is achieved if the uncontrolled k_∞ of a single fuel bundle is less than 1.30 at 65°C (Reference 3). The 8 x 8 fuel bundle types 8D274H and 8D274L at both the zero exposure and the peak reactivity point, have a maximum k_∞ of 1.216 and 1.238 respectively and, therefore, meet the fuel storage requirement for DAEC.

The full power scram reactivity curves used for Cycle 2 are shown in Figure 6.6 of Reference 1 and Figure 1 of Reference 2. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8 and are more stringent than those predicted for this reload; therefore, the scram curves are acceptable.

The void and Doppler coefficients of reactivity for Cycle 2 are given in Table 5-1 of Reference 1. The void coefficient of reactivity at the core average void fraction of 41 percent varies from -15.5 to -18.9 x 10^{-4} $\Delta k/k/\Delta\%V$. The Doppler coefficient of reactivity at a fuel temperature of 650°C varies during the operating cycle from -1.096 to -1.251 x 10^{-5} $\Delta k/k/\Delta T$. Also the effective delayed neutron fraction varies from 0.00545 to 0.00627 over the fuel cycle.

Thus, based on our review of the information presented in the DAEC licensing submittal, and the generic 8 x 8 reload report (Reference 3), we conclude that the nuclear characteristics and performance of the reconstituted core for Cycle 2 will not differ significantly from that of the first fuel cycle and therefore, they are acceptable.

2.3 Mechanical Design

The reload fuel consists of 84 General Electric 8 x 8 fuel bundles with average bundle enrichment of 2.74 wt%, with high and low gadolinia content. These two types of reload fuel have the same mechanical configuration and fuel bundle enrichments as the 8D274H and 8D274L types of fuel assemblies described in the 8 x 8 generic reload report (Reference 3). This generic report, which is under review, has been found acceptable for use for reactors containing 8 x 8 reload fuel, when supplemented with information required by our status report (Reference 6) on the GE generic report evaluation.

Mechanical and operating parameters for the 8 x 8 assemblies are compared to the 7 x 7 assemblies in Table 1 (enclosed). The smaller diameter rods, with lower linear heat generation rate and increased cladding thickness/diameter ratio for the 8 x 8 fuel design as compared to the 7 x 7 fuel assemblies, result in increased safety margins with respect to maximum design linear heat generation rate.

On the basis of our review of the generic 8 x 8 reload report and current operating experience with the 8 x 8 reload design in similar plants, we conclude that the mechanical design of the DAEC reload fuel is acceptable.

2.4 Thermal-Hydraulics

The generic GE 8 x 8 fuel reload topical report (Reference 3) and the GE BWR Thermal Analysis Basis GETAB (Reference 7) are referenced by the licensees to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins.

Application of the GETAB involves:

1) establishing the fuel damage safety limit,

2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and

3) establishing limiting conditions for operation such that the initial conditions assumed in the accident analyses are satisfied.

The following sections present the results of our review of the DAEC Cycle 2 thermal margins based on the GETAB report (Reference 7) and plant specific input information provided by the licensees.

2.4.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding integrity safety limit minimum critical power ratio (MCPR) is 1.07. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during abnormal operational transients.

The uncertainties in (1) the core and system operating parameters, and (2) the GEXL correlation (Table 4-1 of Reference 1), combined with the relative bundle power distribution in the core, form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for the DAEC Cycle 2 are found to be the same or more conservative than those used in GETAB (revision of Table IV-1 of NEDO-10958 – Reference 8). The analysis performed for DAEC includes an additional uncertainty in the standard deviation for the TIP readings (3.85-5.05%) due to the bypass void content which results from plugging of the bypass holes.

The specific core design selected for the generic GETAB statistical analysis is a typical 251/764 core while the DAEC is a 183/368 core. The generic GETAB statistical analysis results are conservative since the bundle power distribution used for GETAB has more high power bundles than the distribution expected during the second cycle of operation of the DAEC reactor. For DAEC, this results in a conservative value of the Minimum Critical Power Ratio (MCPR) which meets the 99.9% criterion discussed above.

We conclude that the proposed fuel integrity safety limit (MCPR of 1.07) is acceptable for DAEC Cycle 2.

2.4.2 Operating Limit MCPR

Various transient events will reduce the Minimum Critical Power Ratio (MCPR) below the specified operating limit MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.07) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (ΔMCPR). The licensees have submitted the results of analyses of those transients which produce a significant decrease in MCPR (Reference 1). The types of transients evaluated were overpressure, feedwater temperature decrease, coolant flow increase, etc. The most limiting transient in these categories was the turbine trip without bypass assuming end of cycle (EOC) scram reactivity insertion rates and initial conditions of 104% of rated power and 100% rated flow. The turbine trip transient results in ΔMCPR's of 0.33 (7 x 7 fuel) and 0.43 (8 x 8 fuel). Addition of these ΔMCPR's to the safety limit MCPR (1.07) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur, i.e., MCPR (OL) $\geq$ ΔMCPR + MCPR (SL). Therefore, the operating limit MCPR's are 1.40 for 7 x 7 fuel and 1.50 for 8 x 8 fuel.

The calculated change in MCPR for the second most severe abnormal operating transient, the loss of feedwater heating, is 0.16 for 7 x 7 fuel and 0.19 for 8 x 8 fuel.

The transient analyses were evaluated with scram reactivity insertion rates that include a design conservatism factor of 0.80. The design conservatism factor for the void coefficient used was 1.25 and the design conservatism factor for Doppler coefficient was 0.90. The initial conditions (Reference 1) used for the worst operational transient are acceptable. The MCPR's assumed in the transient analyses were equal to or greater than the initial operating limit MCPR's.

A GE study (Reference 7) has shown that the required operating limit MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses, the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located at the midplane (axial peak-to-peak average of 1.40).

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values used are 1.098 for 7 x 7 fuel and 1.100 for 8 x 8 fuel. During the cycle the local peaking, and therefore, the R-factor is reduced while the peak in the axial power shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane; however, this compensatory effect does not alter the conservatism applied in determining the operating limit MCPR.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

Analyses have shown that the operating limit MCPR's of 1.40 for 7 x 7 fuel and 1.50 for 8 x 8 fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. Hence, we conclude that the operating limit MCPR's of 1.40 for 7 x 7 fuel and 1.50 for 8 x 8 fuel are acceptable.

2.4.3 Rod Withdrawal Error

The rod withdrawal error transient is discussed in Reference 1 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 3. The

information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and initiate an alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop the rod withdrawal while the critical power ratio is equal to 1.21. Rod block thus occurs when the actual MCPR is greater than the 1.07 MCPR safety limit and the cladding strain is under the one percent plastic strain limit. This rod withdrawal error transient is not limiting for the DAEC Cycle 2 with the RBM setting at 107% of its initial level. We conclude that the analysis performed for this localized transient and the consequences of this localized transient are acceptable.

2.4.4 Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensees will conform to Technical Specification limiting conditions for operation; (Paragraph 3.12.c of the Technical Specifications). This requires that for core flows less than the rated flow, the licensees maintain the MCPR greater than the operating minimum values (1.40 for 7 x 7 fuel and 1.50 for 8 x 8 fuel).

The minimum MCPR values for less than rated flow are the rated flow values multiplied by the respective K_f factors shown in Figure 3.12-1 of the Technical Specifications. The K_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.07. We conclude that the calculated consequences of the anticipated abnormal transients will not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

2.5 Accident Analysis

2.5.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". One of the requirements of the Order was that prior to any license amendment authorizing any core reloading" . . . the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50, 50.46". The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

By letter dated July 8, 1975 (Reference 10), the licensees submitted a reevaluation of the Duane Arnold ECCS performance. This submittal referenced an appropriate lead plant analysis (Reference 9), to show compliance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

The licensees also submitted a request for an amendment to the operating license that would modify the Technical Specifications to implement the results of the ECCS evaluation (Reference 10). We reviewed the licensees' submittals and authorized operation of the facility in accordance with the proposed Technical Specifications as discussed in our Safety Evaluation dated December 16, 1975, issued in support of Amendment No. 15 to the DAEC Facility Operating License. These Technical Specification changes, which included Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) curves for the 7 x 7 fuel, remain applicable to this reload.

By letter dated January 8, 1976 (Reference 4), the licensees submitted a request for an amendment to the Facility Operating License, requesting modifications to the Technical Specifications for DAEC to implement the results of their evaluation of ECCS performance based upon an appropriate lead plant analysis (Reference 9). Based on our review, we find that the analyses are wholly in conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50. The Technical Specification changes which were submitted with Reference 4 include a new MAPLHGR curve for the 8 x 8 fuel to be used during Cycle 2. Operation of the DAEC reactor will meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of (1) figures 3.12-2 through 3.12-4 (7 x 7 fuel) discussed in our Safety Evaluation dated December 16, 1975, and (2) figure 3.12-5 of Iowa Electric Light and Power's January 8, 1976 submittal (Reference 4) and to a minimum critical power ratio (MCPR) greater than 1.18. The analysis of ECCS performance assumed that reactor operation will be limited to a MCPR of 1.18. However, the operating MCPR limits at the plant will be more restrictive.

We conclude that operation of the reactor in accordance with the Technical Specifications will meet the requirements of 10 CFR Part 50, Section 50.46.

2.5.2 Steamline Break Accident

The licensees referenced the steam line break analysis presented in the GE Topical Report NEDO-20360 (Reference 3). Based on our review of this analysis, we find that the introduction of 8 x 8 fuel into the DAEC reactor will not affect the consequences of this accident as discussed in the DAEC Safety Analysis Report. We conclude that the dose resulting from this accident will be well below the guidelines of 10 CFR Part 100, and therefore, it is acceptable.

2.5.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 1 for an 8 x 8 fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in Reference 1 indicates that a fuel loading error results in a peak linear heat generation rate (LHGR) of less than 16.9kW/ft and a minimum critical power ratio (MCPR) greater than 1.15 in the misplaced 8 x 8 fuel bundle during steady state operation.

The peak LHGR is less than that required to cause a 1% plastic strain in the cladding. The MCPR of more than 1.15 in the misplaced bundle is higher than the core wide fuel integrity safety limit of 1.07 and consequently, the misplaced fuel bundle will avoid boiling transition. Fuel bundles adjacent to a misplaced fuel bundle will be negligibly affected. We conclude that the consequences of a fuel loading error are acceptable.

2.5.4 Control Rod Drop Accident

The control rod drop accident for the Duane Arnold reloaded core is within the bounding analysis presented in Reference 3. The Doppler coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 3. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Reload 1 fuel cycle, at both cold (20°C) and hot (286°C) startup conditions. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 1 that the maximum values of the parameters for this reloaded core will not exceed the bounding values.

Based on our review, we conclude that the consequences of a control rod drop accident from any insequence control rod during startup will be below the design limit of 280 cal/gm.

2.5.5 Fuel Handling Accident

With respect to fuel handling accidents discussed in Reference 1, the licensees noted that the description and analyses of this event provided in the FSAR and discussed in the generic 8 x 8 reload report (Reference 3) are applicable to this reload. That is, the total activity released to the environment and the radiological exposures for the 8 x 8 fuel will be less than those values presented in the FSAR for the 7 x 7 core. As identified in the FSAR, the radiological exposures for this accident with 7 x 7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, we conclude that the consequences of this accident for the 8 x 8 fuel will also be well below the 10 CFR 100 guidelines.

2.5.6 Overpressure Analysis

In reference 4, the licensees presented the results of an overpressure analysis to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure. The transient analyzed was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 104% power with the end of cycle scram reactivity insertion rate curve, scram initiated by high neutron flux, void reactivity applicable to this reload, no credit for the relief function of the safety/relief valves, and all safety valves operable. The results of this analysis indicate that the peak pressure at the vessel bottom would be 1291 psig. Furthermore, in Reference 11, the licensees stated that an overpressure sensitivity analysis was done for the Duane Arnold plant and that analysis showed that for the aforementioned overpressure event, the failure of one safety valve would cause the maximum vessel pressure to increase by 19 psig. Hence, the maximum peak pressure at the vessel bottom for MSIV closure with flux scram, no relief function of the safety relief valves and one failed safety valve is calculated to be 1310 psig. This results in a 65 psi margin below the ASME Code limit of 1375 psig. We conclude that the overpressure protection meets the ASME Code criteria and is, therefore, acceptable.

2.6 Modifications to the Rod Sequence Control System (RSCS)

We have reviewed the licensees' submittals regarding the proposed modifications to the RSCS. Amendment 15 to the FSAR provides a description of the proposed modifications to the existing RSCS (Simple Notch Control) at Duane Arnold. The proposed modifications will upgrade the RSCS at Duane Arnold to the RSCS/BWR-4 design (Group Notch Control for plants with 25 wire probes) as reviewed on the Browns Ferry Dockets 50-259, 50-260 and 50-296. Amendment 15 to the FSAR describes the system criteria, operational requirements, concept description, technical basis, and design details.

The physics of the control rod reactivity worths available for a rod drop accident is unchanged by the electrical-mechanical changes involved in the replacement of the simple notch control by the group notch control RSCS. Both systems are intended to maintain the same type of rod group patterns during zero and low power operation. These patterns were developed to assure that control rod worths of magnitude sufficiently large to exceed the NRC staff criteria of 280 cal/gm., as a result of a rod drop accident, will not occur. These patterns were reviewed during our review of the single notch control system for the Duane Arnold first cycle operation and, as indicated in Supplement 3 to the SER, were approved. The pattern for the withdrawal of the first 50% of the rods is the same checkerboard-like pattern of fully withdrawn rods which has been used and approved on previous BWR reactors. The patterns for the second 50% of the rods (until RSCS restrictions are removed at about 30% power) involve the withdrawal of symetric groups of rods arbitrary distances, but with all rods in a group withdrawn equally. With such a pattern, if a single rod is inadvertently left behind, perhaps to be involved in a drop accident, the neutron flux and importance would not be shifted toward that rod and, thus, relatively low reactivity worth is associated with that rod.

Amendments 14 and 15 to the Duane Arnold FSAR present the results of GE calculations for rod interaction parametric studies as well as Duane Arnold patterns which demonstrate that rod worths well below the required limit of about 1.4%Δk will be maintained by the group patterns. We have reviewed these calculations and find that the results are reasonable and we agree that the patterns will assure suitably low rod worths.

The electrical information and design details described in Amendment 15 to the FSAR were reviewed and compared to the design accepted on the Browns Ferry dockets. We conclude, based on our review, that the proposed design modifications will upgrade the existing DAEC design so that it is equal to the design accepted on the Browns Ferry dockets. We further conclude that the implementation of the design modifications will meet the requirements specified in our Safety Evaluation Report for Duane Arnold Energy Center, Supplement 3, dated February 20, 1974.

During the course of our review of the proposed Technical Specifications, we suggested and the licensees agreed to modify sections 4.3.B.3 and 4.3.C.1 to provide consistency

with power levels and surveillance requirements specified in other sections of the Technical Specifications. We have reviewed the proposed Technical Specifications and we conclude that they provide reasonable assurance that the RSCS will operate as designed.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not 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.

Dated: March 19, 1976

TABLE 1

COMPARISON OF PARAMETERS FOR 8x8 AND 7x7 ROD FUEL ASSEMBLY DESIGN

	7x7	8x8
Pellet Outside Diameter (in.)	0.477	0.416
Rod Outside Diameter (in.)	0.563	0.493
Rod-to-Rod Pitch (in.)	0.738	0.640
Water-Fuel Ratio (cold)	2.53	2.60
U Bundle Weight (pounds)	411.8	403.3
Cladding Thickness (mils)	37	34
Active Fuel Length	144	144

REFERENCES

1. "General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Duane Arnold Energy Center With Bypass Flow Holes Plugged." License No. DPR-49 Docket No. 50-331. NEDO 21082 Class I January 1976. Appendix A to NEDO-20360 Revision 1 Supplement 3.

2. Iowa Electric Light and Power Letter (Liu) to NRC (Rusche), IE-76-403 dated March 9, 1976.

3. "General Electric Generic Reload Licensing Application for 8x8 Fuel," Revision 1 Supplement 3, September 1975 (NEDO-20360).

4. Iowa Electric Light and Power Letter (Liu) to NRC (Rusche), "Proposed Changes to Appendix A Technical Specifications and Bases. Duane Arnold Energy Center Operating License DPR-49 - Conformance with Reload Number One License Submittal," IE-76-32 dated January 8, 1976.

5. Iowa Electric Light and Power Letter (Liu) to NRC (Rusche), amendment to Licensing Application ... IE-76-232 dated February 13, 1976.

6. Status Report on the Licensing Topical Report "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360 Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

7. "General Electric BWR Thermal Analysis (GETAB): Data, Correlation and Design Application," NEDO-10958, 73NED9, Class I, November 1973.

8. General Electric Letter (Hinds) to AEC (Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing reports, NEDO-10558 and NEDO-10958, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," dated July 24, 1974.

9. Quad Cities Station Special Report No. 15, Supplement C, Unit 2 and Attachment A (Proprietary).

10. Duane Arnold Energy Center, "Loss-of-Coolant Accident Analysis Conformance with 10 CFR Part 50 Appendix K, July 1975", submitted by letter dated July 8, 1975.

11. Letter from K. Shea (attorney for IELP) to B. Rusche (NRC) dated June 10, 1975.