MAY 6 1977

Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
KGoller
GLear
JWetmore
CParrish
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
DRoss
JMcGough
DEisenhut
ACRS (16)
OPA (Clare Miles)
TBAbernathy
JRBuchanan

Docket No. 50-331

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 33 to Facility License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications and is in response to your application dated January 31, supplemented by letters dated March 10, March 15, March 28, and May 5, 1977.

This amendment changes the Technical Specifications to allow operation of DAEC in Core Cycle 3 by appropriately modifying the core thermal limits and adding a restriction on operations in the natural circulation mode.

Copies of the related Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 33
2. Safety Evaluation
3. FEDERAL REGISTER Notice

cc w/enclosures:
See next page

Const. 1
GL

OFFICE➤	ORB #3	ORB #3	OELD	ORB #3		
SURNAME➤	CParrish	JWetmore:mjf	WD Paton	GLear GL		
DATE➤	5/5/77	5/5/77	5/6/77	5/6/77		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc:

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

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

Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Iowa Electric Light & Power Company
ATTN: Ellery L. Hammond
P. O. Box 351
Cedar Rapids, Iowa 52406

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VII
ATTN: EIS COORDINATOR
1735 Baltimore Avenue
Kansas City, Missouri 64108

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

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. 33
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 January 31, supplemented by letters dated March 10, March 15, March 28 and May 5, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 33, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: May 6, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 33

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
i	i
1.1-1	1.1-1
1.1-10	1.1-10
1.1-17 thru 1.1-20	1.1-17 thru 1.1-20
1.2-5 thru 1.2-7	1.2-5 thru 1.2-7
3.2-16	3.2-16
3.2-14 & 3.2-42	3.2-14 & 3.2-42
3.3-5	3.3-5
3.3-7	3.3-7
3.3-18	3.3-18
3.6-23 & 3.6-24	3.6-23 & 3.6-24
3.12-5 & 3.12-6	3.12-5 & 3.12-6
3.12-9a	3.12-9a
3.12-11	3.12-11
3.12-16	3.12-16
5.2-1	5.2-1

TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS

Amendment No. ~~15~~, 33

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the inter-related variables associated with fuel thermal behavior

Objective:

To establish limits which ensure the integrity of the fuel cladding.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 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.

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	3.53 to 4.15
R-Factor	1.6
Critical Power	3.6

Amendment No. ~~19~~, 33

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 greater than or equal to safety limit when the transient is initiated from MCPR $\geq$ values as indicated in Table 3.12.2.

2. APRM High Flux Scram (Refuel or Startup & Hot Standby Mode).

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

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

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

as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds the safety limit, thus preserving the APRM rod block safety margin.

4. IRM

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that the heat flux is in equilibrium with the neutron flux, and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

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

B. <u>Scram and Isolation on Reactor Low Water Level</u>

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

C. <u>Scram - Turbine Stop Valve Closure</u>

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

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

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

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

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

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

2.2 BASES

Reactor Coolant System Integrity

The discussion in section 3.6.D and 4.6.D Bases is applicable for discussion of pressure relief.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 135 psig.

THIS SIDE INTENTIONALLY LEFT BLANK

THIS SIDE INTENTIONALLY LEFT BLANK

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)	$\leq (0.66W + 42) \left(\frac{*}{P.F}\right)$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W + 39) \left(\frac{*}{TPF}\right)$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

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

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 safety limit. 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 safety limit.

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

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.

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.

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.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

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 safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

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.

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. 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. 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).

Amendment No. ~~19~~, 33

LIMITING CONDITIONS FOR OPERATION

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% Δ k. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. Recirculation Pumps

When the reactor mode switch is in the startup or run position, the reactor shall not be operated in the natural circulation flow mode.

A recirculation pump shall not be started while the reactor is in natural circulation flow and reactor power is greater than 1% of rated thermal power.

F. If Specifications 3.3.A through D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

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.

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

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 is reverified in individual reload analyses.

Six relief valves and two safety valves are installed. 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 greater than a 79 psi margin to the code allowable overpressure limit of 1375 psig if a flux scram is assumed. In addition, the generic analyses have been conducted which show an approximate 36 psi sensitivity increase for each relief valve failure.

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 and is evaluated in each reload analyses. These analyses show that the six relief valves assure greater than 36 psi margin below the setting of the safety valves. Therefore, the safety valves will not open. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to 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

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 value, and an analysis of abnormal operational transients[1]. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). 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 value 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.

TABLE 3.12-2

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle		
	$>$ 2000 MWD/T	$\leq$ 2000 MWD/T, $>$ 1000 MWD/T	$\leq$ 1000 MWD/T to E.O.C.
7 x 7	1.27	1.34	1.35
8 x 8	1.35	1.42	1.43

Amendment No. ~~21~~, ~~25~~, 33

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.

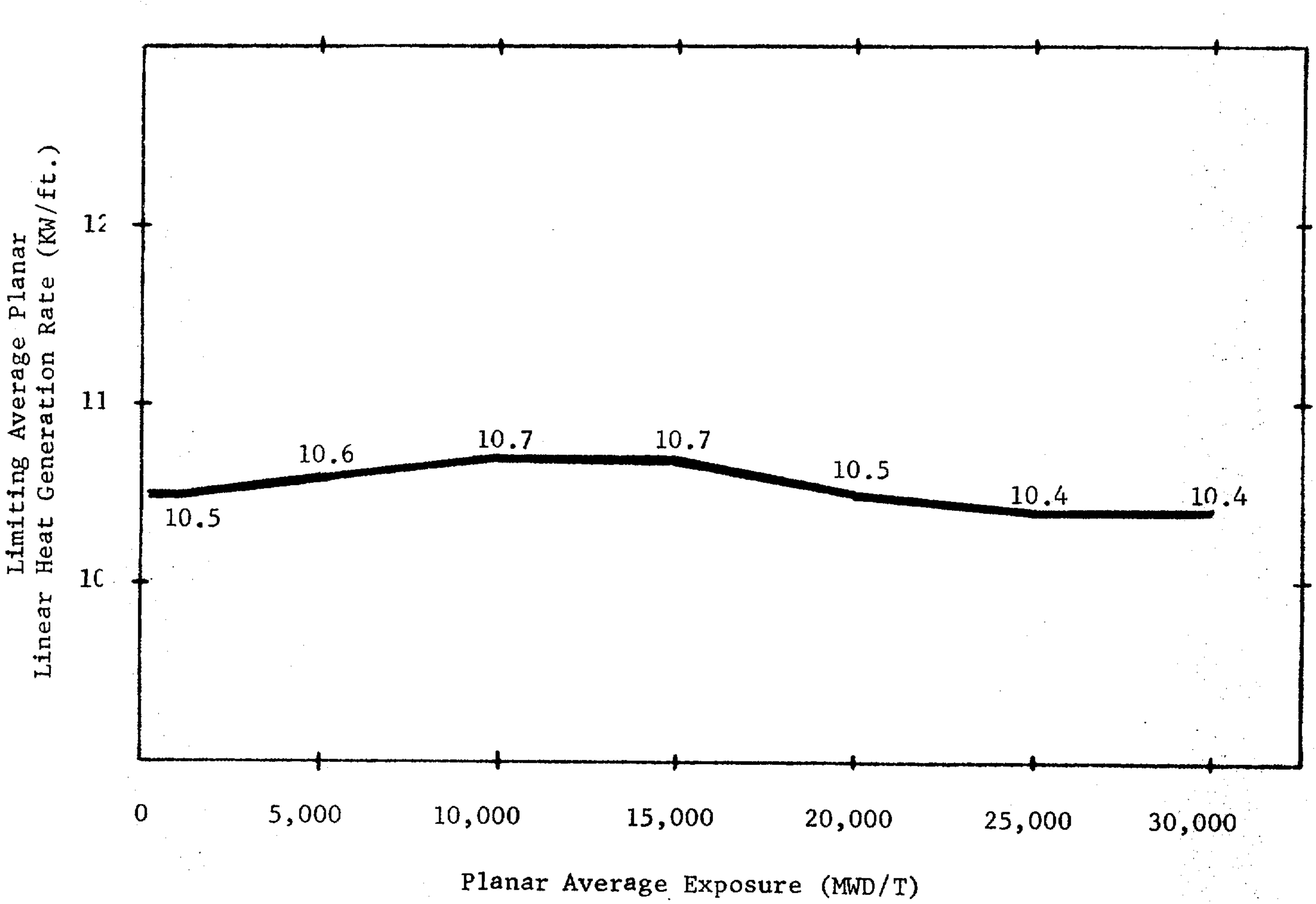
4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

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 Two Licensing Submittal, NEDO-21082-02, Class I, January 1977.



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

5.2 REACTOR

1. The core shall consist of not more than 368 fuel assemblies of an approved fuel design.

2. The reactor core shall contain 89 cruciform shaped control rods. The control material shall be boron carbide powder (B_4C).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 33 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

Introduction

By application for license amendment dated January 31, 1977,[1] Iowa Electric Light and Power Company requested changes to the Technical Specifications for Duane Arnold Energy Center (DAEC). The January 31, 1977 application was supplemented by reference to General Electric Company (GE) DAEC reload 2 licensing submittal,[2] by submittal dated March 10, 1977 on drilling of irradiated fuel assembly's lower tie plates,[3] and by letters dated March 15, 1977[4] and March 28, 1977[15] in response to NRC's request for additional information. Additionally, the licensee will replace relief valves during this reload outage and has supplied information for this change by letter dated May 5, 1977.[17] The requested amendment would revise Technical Specifications for fuel cycle 3 operation of DAEC by appropriately modifying the core thermal limits and adding a restriction on operations in the natural circulation flow mode.

Discussion

Iowa Electric has proposed to replace 100 of the cycle 2 fuel assemblies in DAEC with 100 General Electric (GE) 8 x 8 fuel assemblies which have an average enrichment of 2.74 weight percent (wt%) uranium-235 for cycle 3 operation. Sixty-eight (68) of the assemblies will have a high gadolinia poison content and the remaining 32 will have a low gadolinia content. These fuel assembly types are designated as 8D274H and 8D274L, respectively. The 100 new fuel assemblies will replace 14 type 2 and 74 type 3 initial core 7 x 7 fuel assemblies and 12 reload number one 8D274H fuel assemblies. Approximately 27% of the core fuel assemblies are being replaced for this reload.

The neutronic, thermal-hydraulic, and mechanical designs of 8 x 8 fuel assemblies during normal operation, operational transients, and postulated accidents were evaluated, and found acceptable by the NRC staff in a Directorate of Licensing report entitled "Technical Report on the General Electric Company 8 x 8 Assembly" dated February 5, 1974.(6) The use of 8 x 8 fuel assemblies for reloads was also reviewed, and found acceptable, by the Advisory Committee on Reactor Safeguards and discussed in its annual report dated February 12, 1974.(7) License Amendment No. 19, issued on March 11, 1976, specifically approves the use of 8 x 8 fuel assemblies in DAEC. The technical support for the proposed reload is documented in both the specific documentation for the DAEC as previously referenced and the GE Boiling Water Reactor reload licensing application for 8 x 8 fuel assemblies.(8) Although this report on 8 x 8 reloads is still undergoing NRC review, the report describes many safety analyses which were previously found acceptable and provides an acceptable analytical basis for the evaluation of DAEC reload with GE 8 x 8 fuel assemblies.

The new safety/relief valves have been considered from the standpoint of normal operations, abnormal transients, and accidents. These new valves will decrease the safety/relief valve discharge capacity by approximately 3.6 percent. The major effects of the valve capacity change are on the overpressurization protection analysis, turbine trip without bypass, and the small break Loss-of-Coolant-Accident (LOCA) with High Pressure Coolant Injection system failure. The staff agrees with the licensee that the change in safety/relief valve capacity does not significantly affect any other safety aspects(17), and our review was conducted on this basis.

Evaluation

1. Nuclear Characteristics

The information presented in Iowa Electric's submittals follows the guidelines of NEDO-20360.(8) The fuel assembly pattern consists of previous core and reload assemblies in a symmetrical pattern throughout the core. The low gadolinia reload assemblies are loaded in the exterior of the core and the high gadolinia reload assemblies are loaded in the interior regions of the core. The data in reference 2 indicate that the nuclear characteristics of the cycle 3 core are similar to that of the previous cycle. The total control system worth and the temperature behavior of the reload core will not differ significantly from those values previously reported for DAEC.

The shutdown margin meets the Technical Specifications requirements that the core be at least 0.38%ΔK subcritical in its most reactive state with the most reactive rod fully withdrawn and all others fully inserted. For cycle 3 a minimum shutdown margin of 1.38%Δk was calculated. Reference 1 calculational results indicate that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by 3.0%Δk at 20°C under a xenon free condition. Therefore, the alternate shutdown requirement of the General Design Criterion 26 of Appendix A to 10 CFR Part 50 is met by the Standby Liquid Control System which contains the borated solution.

The effect of lower tie plate drilling on reactor shutdown margins and other nuclear characteristics of the core will not be significant, except for the void coefficient which will be somewhat less negative. This change in void coefficient is in the conservative direction for power increase transients and is in the non-conservative direction for power decrease transients. This increase in void coefficient would result in greater thermal and safety margins for cycle 3 operation for the limiting thermal-hydraulics, transient, and accident analyses. Power decrease transient analyses, have been performed with a void coefficient which bounds the potential void coefficients due to the drilling change. From the standpoint of nuclear characteristics, the drilling of the fuel assembly lower tie plates has been conservatively considered; and thus, is acceptable.

Therefore, in summary, based on the information presented in Iowa Electric's submittals and supplemented by the applicable sections of the generic 8 x 8 reload report,[8] the nuclear characteristics and performance of the reload core for cycle 3 operation of DAEC are acceptable.

2. Mechanical Design

The two types of Reload 2 fuel assemblies have the same mechanical configuration and fuel assembly enrichments as the 8D274L and 8D274H fuel assemblies described in the 8 x 8 generic reload report. On the bases of the licensee's submittals, the staff safety evaluation on the results of drilling the lower tie plates (to eliminate in-core vibration of TIP instrument tubes)[10], and the substantial operating experience of this fuel in operating reactors,[2] the staff concludes that the mechanical design of the fuel proposed for cycle 3 operation at DAEC is acceptable.

3. Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report[8] and GETAB[11] Licensing Topical Report are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins for cycle 3. Application of thermal-hydraulic analyses involves:

1. The determination of the fuel damage safety limit Minimum Critical Power Ratio (MCPR); and

2. The determination of the operating limit MCPR, such that, for any anticipated transient the safety limit MCPR will not be violated.

We have evaluated the DAEC cycle 3 thermal margins based on the GETAB report and plant specific input information provided by the licensee. Our evaluation is summarized below.

a. Fuel Cladding Integrity Limit Minimum Critical Power Ratio (MCPR)

The fuel cladding integrity safety limit MCPR is 1.06. 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 normal operation or transients that are anticipated to occur with moderate frequency".[12] The uncertainties in the core and system operating parameters[2] and the GEXL correlation form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for DAEC cycle 3 are the same as those used in the initial cycle except for the bypass void effect on TIP uncertainty. The analysis includes revised uncertainty for the TIP readings due to the decreased voiding in the bypass region resulting from drilling of the bypass holes in fuel assemblies.

This drilling was accomplished as part of a generic task to eliminate BWR TIP instrument tube vibrations. The licensee has additionally proposed to drill some of the residual assemblies and has provided a safety analyses for this task.[3] The staff has reviewed and has concluded that the operation with both the completely drilled assemblies and completely plugged core support plate to be acceptable (reference 10). The staff's position on the operation of partially drilled cores, documented in reference 10, was as follows:

"For those reactors in which the 1-inch bypass flow holes are plugged but not all fuel bundles are drilled, we conclude that the out reactor flow test sufficiently demonstrated that the modification will reduce significantly in-core tube vibration and hence channel box damage. However, the allowable power level after such modification must be reviewed individually for each reactor considering normal operation, anticipated transients and accidents."[10]

Since DAEC cycle 3 operation will utilize a partially drilled core it was "reviewed individually". The thermal-hydraulic analyses were performed as described in reference 11. This conservatively predicts the thermal margins for individual fuel bundles based on the predicted bypass flow rates. The amount of bypass region voiding is calculated using a core hydraulics model which calculates the flow distribution in both the bundles and the bypass region assuming a given core pressure drop and total flow rate. The bypass region voiding is conservatively calculated by assuming no mixing with bypass flow from cooler regions.

The staff has determined that the thermal hydraulics performance with bypass region voiding as a result of loading with drilled fuel has been conservatively considered for cycle 3 operation for DAEC. The calculational methods[11] and input data[2] have been conservatively represented and used in this analysis. Therefore, we find the safety limit MCPR of 1.06 to be acceptable for DAEC cycle 3 operation.

b. Operating Limit MCPR

To assure that the fuel cladding integrity safety limit MCPR of 1.06 is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determined which one results in the largest reduction in MCPR (ΔCPR). Iowa Electric has submitted the results of analyses of these limiting transients.[2,3] Addition of these ΔCPR's to the safety limit MCPR gives the minimum operating limit MCPR required to avoid violation of the safety limit, should the limiting transient occur.

The most limiting transient was the turbine trip without turbine bypass to the main condenser. The transient was analyzed at various times during the cycle using the appropriate scram reactivity curves. Since the scram reactivity curve decreases with increasing burnup, the ΔCPR generally increases with burnup to end-of-cycle (EOC). The licensee has performed the turbine trip without bypass for three burnup intervals, e.g., from Beginning-of-cycle (BOC) to 2 GWD/T1/ before End-of-Cycle (EOC), from 2 GWD/T before EOC to 1 GWD/T before EOC, and from 1 GWD/T before EOC to EOC.

We have reviewed the calculational methods referenced by Iowa Electric(8) as well as the input data and safety analysis(2)(3) for calculation of operational transients. Based on our review, we have concluded that the operating limit MCPRs proposed for Technical Specification Table 3.12-2 for DAEC cycle 3 operation are acceptable.

4. Accident and Transient Analysis

a. Anticipated Transients

Iowa Electric has stated that "all transients which are the basis of the existing license were reviewed, and those transients which have been limiting in the past with respect to safety margins and are significantly sensitive to the core transient parameter deviations (effected by this reload) were reanalyzed."(2) The methods of these analyses are described in reference 8. The conservatisms for these analyses as related to drilling of the residual fuel in the core are discussed in reference 3. The input parameters and functions which characterize this cycle are analyzed under EOC conditions, and also at the two intermediate exposures previously discussed. The results of the analyses are given in Section 6 of reference 2. The highest ΔCPR occurs for the 8 x 8 fuel during the turbine trip without bypass transient. A brief description of the transients that were reanalyzed and reported in reference 2 are presented in the following sections.

(1) Overpressure Analysis

Reference 17 presents 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 for DAEC. This analysis demonstrates the adequacy of the safety/relief valves which are to be in service for cycle 3 operation. The transient analyzed was the closure of all main steam

1/ Gigawatt-Day per Metric Tonne Uranium isolation valves (MSIVs) with high neutron flux scram. The analysis was performed for 104% power, scram initiated by high neutron flux, void reactivity conservatively applicable to this reload, credit for the relief function of the safety/relief valves, with all safety valves operative. The results of this analysis indicate that the peak pressure of 1296 psig at EOC. Furthermore, generic analysis applied to DAEC showed that for this overpressure event, the failure of one relief valve would cause the maximum vessel pressure to increase by 20 psig.[15] Hence, the maximum peak pressure at the vessel bottom for MSIV closure with an indirect scram, and one failed relief valve is calculated to be 1316 psig; this results in about a 59 psig margin below the code allowable of 1375 psig which is acceptable.

(2) Rod Withdrawal Error

Iowa Electric has analyzed the Rod Withdrawal Error transient according to the assumptions given in reference 2. The analysis was performed for both the fully drilled configuration[3] and the configuration with only new fuel drilled.[2] The results show ΔCPRs of 0.15 for 7 x 7 fuel and 0.16 for 8 x 8 fuel. The rod block monitor (RBM) set point of 105% of reactor thermal power will stop rod withdrawal at a MCPR of >1.06, the MCPR safety limit. Based on this analysis of the spectrum of worst case conditions for DAEC, we have concluded that the proposed Rod Block Monitor flow biased set point relationship contained in Technical Specification Table 3.2-C is acceptable for cycle 3 operation.

(3) Turbine Trip with Bypass Failure

Fast closure of the turbine control valve (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The TCV's are required to close as rapidly as possible to prevent overspeed of the turbine-generator (T-G) rotor. This closing, concurrent with the failure of the bypass valve system, causes a sudden reduction in steam flow which results in a nuclear system pressure increase and shutdown of the reactor. The licensee has demonstrated that the analysis submitted for the partially drilled configuration[2] is limiting compared to the analysis for the fully drilled configuration[3]. Again, this is the most restrictive transient on a MCPR basis. The licensee has performed this analysis with the new safety/relief valve configuration and has found it limiting. The results have been discussed previously and are acceptable.

(4) Loss of 100°F Feedwater Heater, Manual Control

The loss of a feedwater heater is the most limiting cool water injection transient. A feedwater heater can be lost by (1) the steam extraction line to the heater being closed off which removes the heat supply to the heater and causes a gradual cooling down of the tubes or (2) the feedwater flow through the heater being switched to the bypass line. In either case, the reactor will receive cooler feedwater which results in an increase in core inlet subcooling, and an increase in core power due to a negative void coefficient. The results of this transient, documented in Table 6-3 of reference 2, are not limiting; and thus, are acceptable.

b. Accident Analysis

(1) Loss-of-Coolant Accident

The loss-of-coolant accident was reanalyzed for the 8D274 fuel with the results presented in Reference 2. The analysis indicated compliance with Section 50.46 criteria and Appendix K to 10 CFR Part 50. The Technical Specification changes[1] include MAPLHGR limits for the 8 x 8 fuel (Technical Specification Figure 3.12-5) which are appropriate for this analysis; and thus are acceptable.

The licensee has also provided evidence that the replacement of the safety/relief valves do not significantly impact on the ECCS analysis.[17] For the small break LOCA with a failure of the High Pressure Coolant Injection System, the Automatic Depressurization System uses these valves to reduce pressure, so that the Low Pressure Injection system may be used to remove core decay heat. To support the adequacy of the new safety/relief valves to provide sufficient flow capacity for this postulated event, the licensee referenced a generic sensitivity study on the effects of ADS capacity on peak clad temperature (PCT). This study showed that a 10% reduction in ADS capacity resulted in less than or equal to a 130°F increase in PCT for the small break LOCA. The installation of the new safety/relief valves will result in only a 3.6% reduction in ADS capacity. Since the PCT for the most limiting small break LOCA that has been previously analyzed is 1573°F, the maximum PCT with the new safety/relief valves is conservatively assumed to be less than 1703°F (1573°F + 130°F), which leaves a margin of 497°F to the 2200F PCT limit of 10 CFR 50.46. Therefore, we have concluded that the LOCA analysis conservatively accounts for the new safety/relief valves, and is acceptable.

We would also like to note that the NRC issued an Exemption to 10 CFR 50.46 for DAEC by letter dated March 11, 1977.[16] The exemption was issued after GE notified NRC that generic errors had been discovered in the GE ECCS evaluation model. The errors detected applied to other BWR's as well as DAEC and either were of the nature of inputs to computer codes used in the analyses or were due to numerical errors in the calculations performed. The March 11 Exemption identifies the errors and proposed changes in methods of analysis in the ECCS performance evaluation. The licensee has supplied the NRC with an evaluation of the ECCS errors identifed generically by GE. The Exemption confirms the appropriateness of the licensees voluntary action in agreeing to submit, on a timely basis, an ECCS reevaluation using a GE ECCS evaluation model approved by the NRC, and permits operation of DAEC during the interim period while the required calculations are carried out.

(2) Main Steam Line Break, Refueling Accident

The analyses of the following accidents were listed by the licensee as being covered by the generic analyses given in reference 8:

(a) Main Steam Line Break Accident, and

(b) Refueling Accident,

Based on our previous review (reference 9) of the referenced material for DAEC we conclude that the results provided by the generic analyses are applicable and acceptable.

(3) Control Rod Drop Accident

A plant specific analysis was performed for the control rod drop accident. The analysis indicated that the maximum fuel enthalpy resulting from the dropping of any in-sequence control rod would be below the 280 cal/gm safety limit.[2] The change in these results due to residual fuel drilling was evaluated and found to be negligible.[3] The staff concludes that this analysis demonstrates the adequacy of the core and safety protection from the rod drop accident, and is, therefore, acceptable.

(4) Fuel Loading Error Event

The fuel loading error event assumes that either a reload bundle is rotated 180 degrees in a location near the center of the core or a bundle is inserted in an improper location. For DAEC the case of a fuel bundle inserted in an improper location gave the most restrictive results. For this case the maximum linear heat generation rate (LHGR) is 16.5 kw/ft and the MCPR is 1.08 in the misplaced fuel bundle (adjacent fuel bundles are not affected). Based on the conservative treatment of this event and the satisfaction of the safety limit MCPR criteria as previously discussed as well as the fact that the LHGR will not cause excessive strain of the cladding, the staff finds the consequences of the fuel loading error to be acceptable.

(5) Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in reference 8 and 2, respectively. The results of the Cycle 3 analysis show that the 7 x 7 and 8 x 8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is well within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

The NRC staff has expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve. The staff concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a test program.

The proposed test program will be conducted at a BWR facility representative of DAEC and is expected to be a significant aid in the resolution of generic staff concerns on stability. The results from the testing program will be provided to the NRC staff by the General Electric Company. The results will be used to refine the reactor stability analysis safety margins.

In the interim, the staff has imposed a specific requirement on DAEC which will restrict operations in the natural circulation flow mode (Technical Specification 3.3.E). The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during Cycle 3. On the basis of the foregoing, the NRC staff considers the thermal-hydraulic stability of DAEC to be acceptable.

5. Recirculation Pump Startup From the Natural Circulation Operational Mode

During a recent BWR reload review, the question of recirculation pump startup from the natural circulation operational mode was raised. This pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence had not been previously evaluated, so that for this reload review, additional information was requested.

Iowa Electric was requested to provide analyses and startup test results, which prove that the startup of recirculation pumps from natural circulation conditions does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed. An option was also afforded to preclude power operations, i.e., at $>1\%$ rated thermal power, in the natural circulation mode by Technical Specification. The licensee has agreed to incorporate Technical Specifications (Technical Specification 3.3.E) which preclude recirculation pump startup at reactor power operation $>1\%$ rated power in the natural circulation mode. We find this measure to be acceptable.

6. Physics Startup Testing

As part of the review of cycle 3 Iowa Electric was requested to provide a description of the Physics Startup Testing program. In response to our request, this program was described in letter dated March 15, 1977. Additionally, the program was discussed with the licensee for clarification of the cold criticality measurement and comparison to predicted criticality. This measurement and comparison will be performed and documented per agreement with the licensee. The results of the physics startup test program will be reported within 90 days of startup. The Staff finds the Startup Physics Testing program and reporting schedule to be acceptable.

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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: May 6, 1977

References

1. Letter from L. Liu (Iowa Electric Light and Power Company) to B. C. Rusche (NRC) dated January 31, 1977 refer to IE-77-220.

2. "General Electric Boiling Water Reactor Reload Number 2 Licensing Submittal, Duane Arnold Energy Center License Number DPR-49 Docket Number 40-331," NEDO-21082-02, Class I; January, 1973.

3. Letter from L. Liu of (Iowa Electric Light and Power Company) to B. C. Rusche (NRC) dated March 10, 1977 refer to IE-77-515.

4. Letter from L. Liu (Iowa Electric Light and Power Company) to B. C. Rusche (NRC) dated March 15, 1977 refer to IE-77-551.

5. Letter from L. Liu (Iowa Electric Light and Power Company) to B. C. Rusche (NRC) dated March 28, 1977 refer to IE-77.

6. "Technical Report on the General Electric Company 8x8 Assembly", NRC Directorate of Licensing, February 4, 1974.

7. Annual Report by Advisory Committee on Reactor Safeguards, February 12, 1974.

8. GE/BWR Generic Reload Licensing Application for 8x8 fuel Rev. 1, Supplement 4, NEDO-20360, April, 1976.

9. Status Report on the Licensing Topical Report, "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel" NEDO 20360, Rev. 1, Supplement 1.

10. Safety Evaluation Report on the Reactor Modification to Eliminate Significant In-Core Vibration in Operating Reactors with 1-inch Bypass Holes in Core Support Plates," Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, February, 1976.

11. General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application," General Electric Company BWR Systems Department, NEDO-10958, November, 1973.

12. Standard Review Plan, Section 4.4, pgs, 4.2.-2 and 4.4.-3.

13. Letter from L. Liu (Iowa Electric Light and Power Company) to G. Lear (NRC) dated February 13, 1977 refer to IE-77-364.

14. Letter from G. K. Rhode (Niagara Mohawk Power Corporation) to G. Lear (NRC) dated March 14, 1977.

15. Letter I. Stuart (GE) to V. Stello (NRC) on Code Overpressurization Protection Analysis Sensitivity of Peak Vessel Pressure to Valve Operability, December 23, 1976.

16. Letter G. Lear (NRC) to D. Arnold (Iowa Electric), March 11, 1977.

17. Letter L. Liu (Iowa Electric) to G. Lear (NRC) dated May 5, 1977.

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 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 consists of changes to the Technical Specifications that will allow operation of DAEC in Core Cycle 3 by appropriately modifying the core thermal limits and adding a restriction on operations in the natural circulation mode.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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 impact statement or negative declaration and 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 January 31, supplemented by letters dated March 10, March 15, March 28 and May 5, (2) Amendment No. 33 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 6 day of May 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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