APRIL 20 1978

Docket No. 50-331

APRIL 0 1978

Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
BGrimes
GLear
SSheppard
RClark
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
OPA-CMiles
DRoss
TBAbernathy
JRBuchanan

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. 42 to Facility License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your application dated January 4, 1978.

This amendment changes the Technical Specifications to allow operation of the Duane Arnold Energy Center in Core Cycle 4. With your concurrence, we have revised the MCPR limits in Table 3.12-2 from those submitted in your application of January 4, 1978.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 42
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Provided changes made as shown on the safety evaluation

Const. 1
GD

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	SSheppard:mjf	RClark	BMBoardman	GLear		
DATE	4/ /78	4/ /78	4/12/78	4/20/78		

NRC FORM 318 (9-76) NRCM 0240

☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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. 42
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, et al (the licensee) dated January 4, 1978, 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. 42, 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: April 20, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 42

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	Replace
3.6-24	3.6-24
3.12-9a	3.12-9a
3.12-11	3.12-11
3.12-13	3.12-13
3.12-14	3.12-14
3.12-15	3.12-15
3.12-16	3.12-16
3.12-17	3.12-17

the direct scram (valve position scram) results in an 88 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 20 psi sensitivity increase for each relief valve failure.

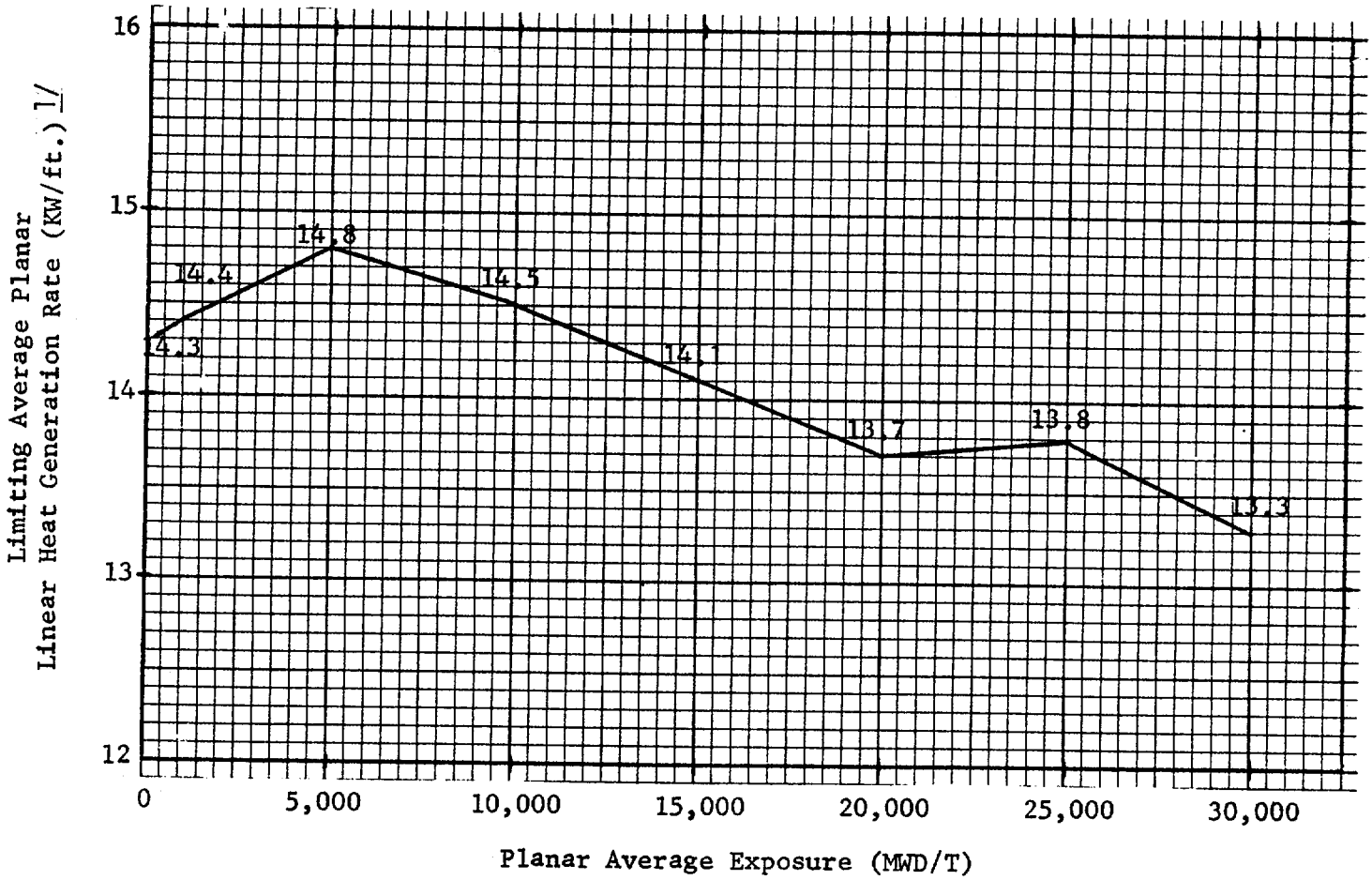
The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraphs 14.5.1.2 and 14.5.1.3 and is evaluated in each reload analyses. These analyses show that the six relief valves assure a 25 psi or greater margin 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

TABLE 3.12-2

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle			
	B.O.C. to >2000 MWD/T	≤ 2000 MWD/T to >1000 MWD/T	≤ 1000 MWD/T to > 500 MWD/T	≤ 500 MWD/T to E.O.C.
7 x 7	1.26	1.26	1.26	1.30
8 x 8	1.26	1.29	1.34	1.38



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

DUANE ARNOLD ENERGY CENTER

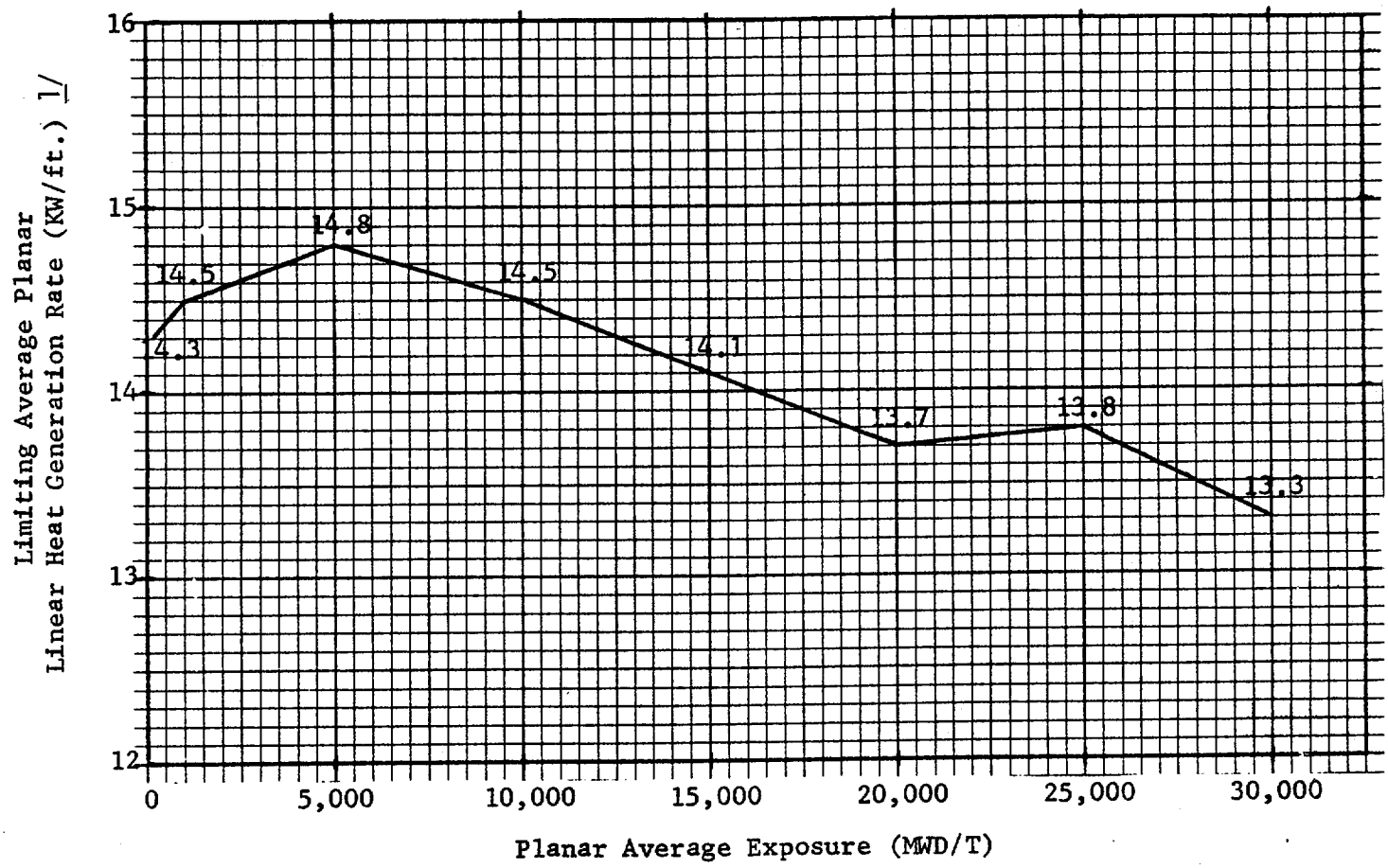
IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

FUEL TYPE: INITIAL CORE TYPE 2

FIGURE 3.12-2



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

DUANE ARNOLD ENERGY CENTER

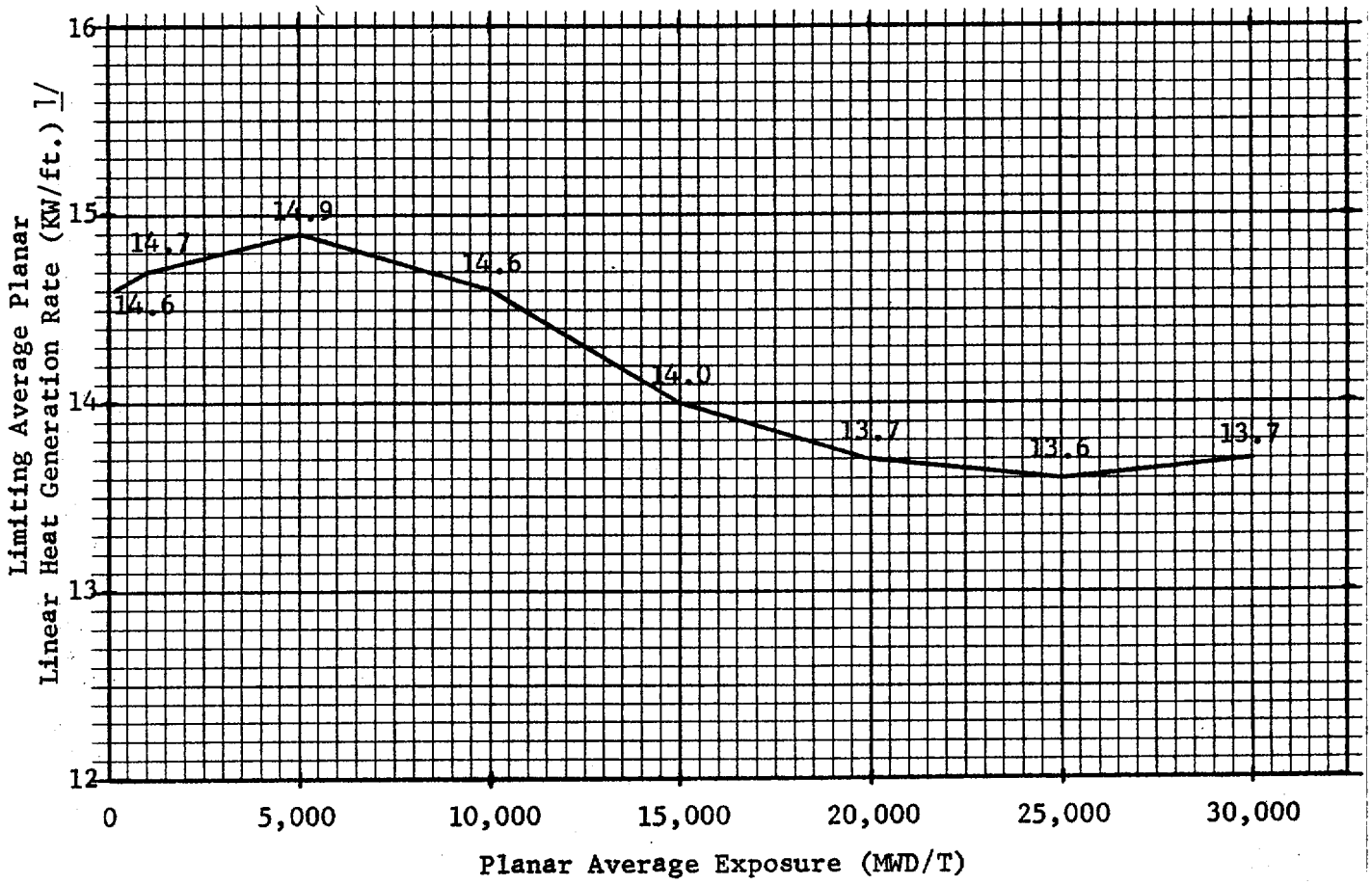
IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

FUEL TYPE: INITIAL CORE TYPE 3

FIGURE 3.12-3



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

DUANE ARNOLD ENERGY CENTER

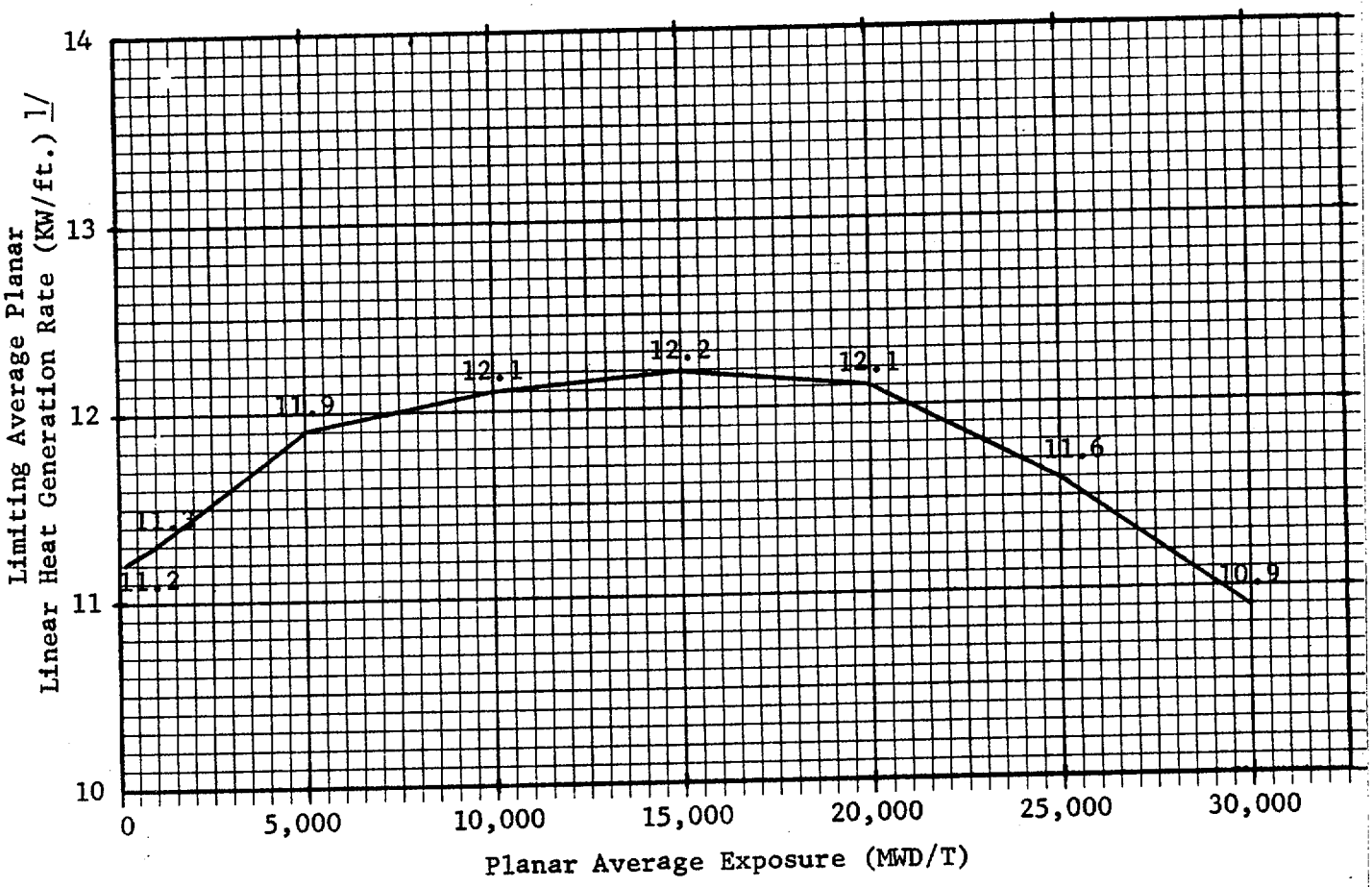
IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

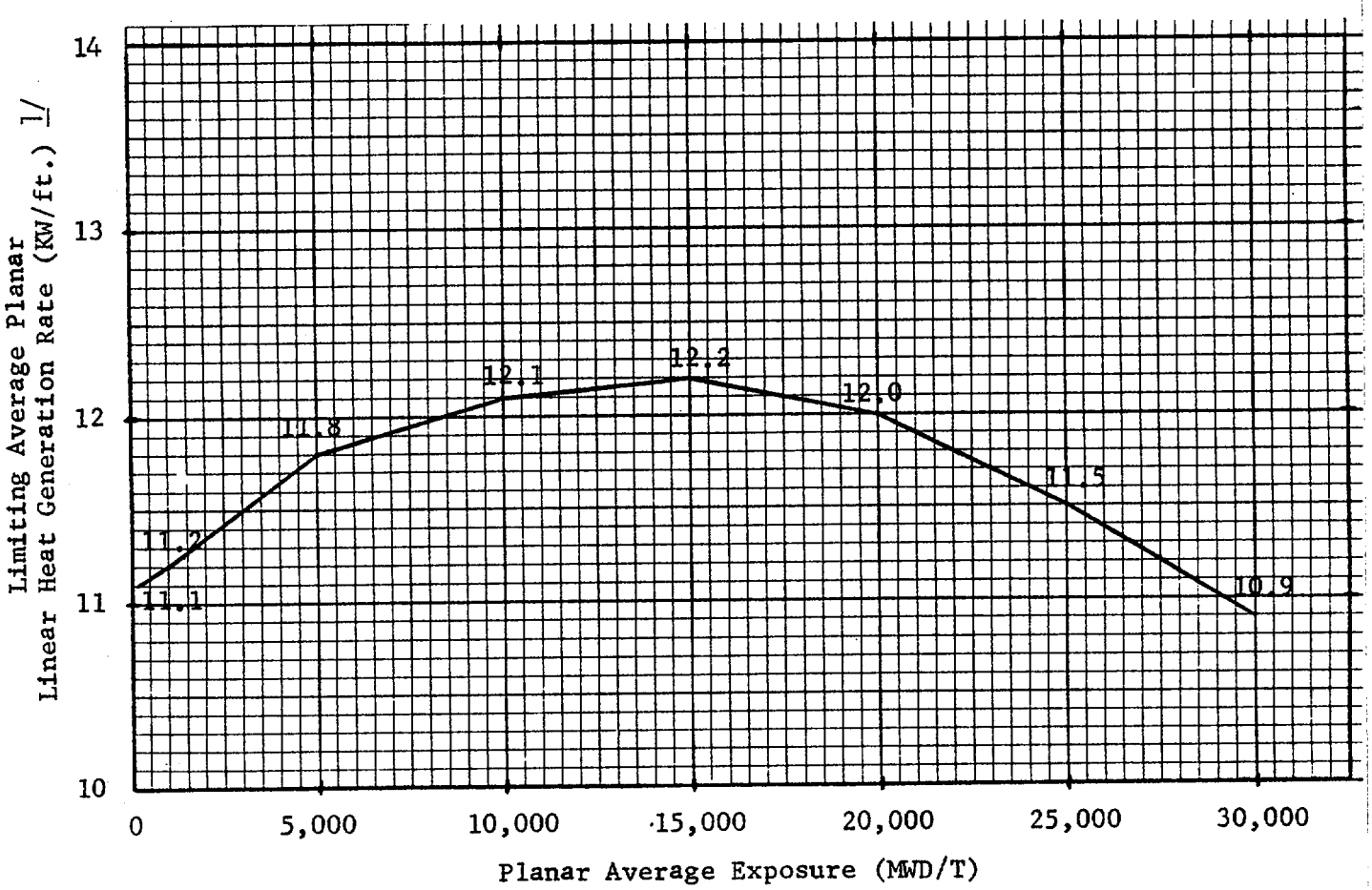
FUEL TYPE: 7D230 TYPE 4

FIGURE 3.12-4



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

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: 8D274L FIGURE 3.12-5



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

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

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

FUEL TYPE: 8D274H

FIGURE 3.12-6

Amendment No. 39, 42

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 42 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 Introduction

By letter dated January 4, 1978, Iowa Electric Light and Power Company (the licensee) proposed changes to the Technical Specifications of Provisional Operating License DPR-49 for Duane Arnold Energy Center (DAEC). The proposed changes relate to the replacement of 88 fuel assemblies which constitutes refueling of the core for fourth cycle operation at power levels up to 1593 MWt (100% power). Iowa Electric is not proposing any change in the type of fuel used in cycle 4 vs that used in cycle 3; the core for cycle 4 will still be 7x7 and 8x8 fuel bundles. The transient and accident analyses performed by the licensee in support of cycle 4 operation are the same as those we have previously reviewed and approved in Amendments Nos. 33, 37, 38, 39 and 40 related to cycle 3 operation.

In support of the reload application the licensee has provided the GE BWR Reload 3 licensing submittal for DAEC (Reference 1) and proposed Technical Specification changes (Reference 2).

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (Reference 3). Although later supplements to this report are undergoing review by the staff, portions of this topical have been found applicable for reactors containing 8x8 reload fuel and are acceptable to the staff when supplemented with information required by our status report (Reference 4). The supplemental information provided by the licensee and the staff's evaluation thereof are summarized below.

During our review of the proposed Technical Specifications we determined that certain changes were necessary to conform with Regulatory requirements. These changes have been accepted by the licensee.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4 operation of DAEC, 88 fresh 8x8 fuel bundles (8D274H) with an enrichment of 2.74% U235 by weight will be loaded into the core. In addition, 96 7x7 assemblies from the initial core loading,

84 8D274 assemblies from Reload 1, and 100 8D274 assemblies from Reload 2 will remain in the core. Thus, for Cycle 4 (Reload 3) approximately 24% of the 368 fuel bundles will be fresh fuel.

As indicated by the loading diagram presented in Reference 1 (Figure 2-1), the fresh fuel will be distributed symmetrically throughout the core.

The nuclear characteristics of the reload 8D274H fuel bundles are identical to those previously loaded in the core and discussed in Reference 3. The licensee therefore states, as substantiated in Reference 3, that the total control system worth, and the temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported.

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 condition throughout the cycle when the highest worth control rod is fully withdrawn and all other rods are fully inserted. The licensee has calculated the minimum shutdown margin to be 0.012 Δk for this cycle. This occurs at the beginning of cycle. The effect of settling of B_4C in inverted poison tubes in control rods will not have a significant effect on this cycle's margin.

The calculational results presented in Reference 1 indicate that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by at least 0.030 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria (i.e., that the reactor is subcritical at cold conditions) is met by the Standby Liquid Control System.

The full power scram reactivity curves used for the reload submittal analyses are shown in Figures 6.6a-d of Reference 1. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8 which is acceptable to the staff as discussed in Reference 4.

Based on our review of the information presented in the licensing submittal (Reference 1) as supplemented by applicable portions of the generic 8x8 reload report (Reference 3) and the staff's acceptance thereof (Reference 4), we have determined that the nuclear characteristics and expected performance of the reconstituted core for this reload are acceptable.

2.2 Mechanical Design

The reload fuel has the same mechanical configuration and fuel bundle enrichments as the 8D274 assemblies described in the 8x8 generic reload report (Reference 3). The improved water rod design described in Section 3 of Reference 3 has been adopted by the licensee for this reload.

The generic 8x8 reload report (Reference 3), supplements of which are under review, has been found acceptable for use for reactors containing 8x8 reload fuel, when supplemented with information required by our status report (Reference 4) on the GE generic report evaluation. As previously discussed, the proposed reload does not use any new type of fuel than that used in a previous reload. The proposed loading is similar to the core configuration previously found acceptable for cycle 3 operation. On the basis of our review of the generic 8x8 reload report, the reload submittal which is the subject of this safety evaluation and the cumulative operating experience of the 8x8 fuel assemblies in other BWRs, we conclude that the fuel mechanical design is acceptable for this cycle.

2.3 Thermal-Hydraulics

The GE generic 8x8 fuel reload topical report (Reference 3) and the General Electric Thermal Analysis Basis (GETAB) (Reference 5) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit Minimum Critical Power Ratio (MCPR),

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

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

We have evaluated the thermal margins based on the GETAB report and plant specific input information provided by the licensee. The staff evaluation of these margins is reported in the following subsections.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding safety limit MCPR of 1.06 has been established, based on the GETAB (Reference 5) statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients (Reference 6). This limit is applied for both core-wide and localized transients or perturbations to the expected CPR distribution.

The uncertainties in core and system operating parameters and the GEXL correlation uncertainties are the same as those used for the original statistical analysis on which the fuel cladding safety limit MCPR is based. The bundle power distribution is expected to include fewer high power bundles than the distribution assumed for the original statistical analysis as is indicated by comparing Figure 4-1 and 4-2 of Reference 1 with Figures 4-4.1 through 4-4.4 of Reference 3. Therefore, it is conservative to apply the fuel cladding safety limit of 1.06.

2.3.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 4 operation. The most limiting operational transients and the fuel loading error have been analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR.

The transients evaluated were the generator load rejection without bypass, the turbine trip with failure of the bypass valves, loss of 100°F of feedwater heating, feedwater controller failure, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 4-4, 6-1a and b of Reference 1 were assumed.

The input to the transient calculations and the application of the analysis methods of Reference 3 have been reviewed and determined to provide appropriate conservatism for determination of the operating limit MCPR.

From BOC to EOC-2 GWD/T, the rod withdrawal error results in the largest CPR for the 7x7 fuel, and the loss of feedwater heating for the 8x8 fuel. For the rest of the cycle the largest CPR results from the load rejection without bypass for both fuel types. The calculated reductions in CPR during each of the operational transients have been tabulated in Reference 1. Addition of these CPR's to the safety limit MCPR of 1.06 would give the operating limit MCPR's for each fuel type through different cycle exposure levels which would protect against boiling transition during plant transients. The licensee has also analyzed fuel loading errors. Should a fuel bundle loading error occur, an even higher operating limit MCPR for the 7x7 and 8x8 fuel through the exposure interval from BOC to EOC-2GWD/T and for the 7x7 fuel for the exposure intervals from EOC-2GWD/T to EOC-1 GWD/T and from EOC-1GWD/T to EOC-.5 GWD/T would be required to ensure that for this localized perturbation the CPR at the misloading site would not be below the safety limit of 1.06 during steady state operation. Therefore, the operating limits for the previously specified fuel and exposure points should be increased by 0.04 and the operating limit MCPR's should be as specified in Table 1.

Furthermore, should there be no fuel loading error, when with the proposed operating limit MCPR, 99.9% of the fuel rods will avoid transition boiling by an extra margin during any operational transient. We have reviewed these analyses and find the MCPR values acceptable.

2.3.3 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 licensee will conform to the limiting conditions for operation stated in the Technical Specifications. This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum values. The MCPR values for less than rated flow are the rated flow values multiplied by the respective K_f factors appearing in 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 1.06. We conclude that the calculated consequences of the anticipated operational transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 ECCS Appendix K Analysis

The generic Loss of Coolant Accident (LOCA) analysis for Reference 7 is to be applied to this reload's fuel. Since the core will contain more drilled bundles than the generic analysis assumed, the previous LOCA analysis is conservative for this reload. No new fuel types will be inserted into the core, therefore, new MAPLHGR's are not required.

2.4.2 Main Steam Line Break Accident

Steam line break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in Section 2.4.1. The analysis of steam line break accidents occurring outside containment as presented by the licensee is acceptable based on our generic review of NEDO-20360 (References 3 and 4).

2.4.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 1 for a fuel bundle placed in an improper location or rotated 180°. The worst potential fuel loading error was analyzed from an initial MCPR of 1.22. This resulted in a MCPR of 1.02 and a peak linear heat generation rate of 18.6 KW/ft. The staff proposed and the licensee agreed to increase the MCPR by at least 0.04. Therefore, the plant operating limit for the applicable fuel types and exposures as discussed in Section 2.3.2 have been increased. The staff finds this acceptable based on the sensitivity of ΔCPR to initial MCPR seen to date. The implications of

the MCPR have been discussed previously in Section 2.3.2. The peak LHGR associated with a fuelloading error is not large enough to cause fuel damage.

2.4.4 Control Rod Drop Accident

The control rod drop accident is defined as a power excursion caused by accidental removal of a control rod from the core at a more rapid rate than can be achieved by the use of the contol rod drive mechanism. In the control rod drop accident, a fully inserted control rod is assumed to fall out of the core after becoming disconnected from its drive and after the drive has been removed to the fully withdrawn position. There are many design safeguards which minimize the risk of this accident, e.g., the control rod design minimizes probability of sticking in the core and separation from a control rod drive, rod coupling is verified by changes in neutron flux during criticality and a rod bottoming out indication before criticality, the rod velocity limiter limits rod drop velocity, and the control rod worth minimizer interlock system consists of a computer program which monitors the control rod withdrawal sequence and actuates interlocks to prevent abnormal control rod patterns and high rod worths.

For reloads, the significant parameters of the control rod drop accident are compared to values used in a bounding analysis. If the reload specific parameters are conservatively compared to the bounding analysis values, the consequences of the control rod drop are less severe than those of the bounding analysis. In Figures 6-1 through 6-5 of Reference 1 the licensee has shown that the Doppler Coefficient as a function of fuel temperature is more negative than the bounding curve presented in Reference 3. However, the reactivity insertion rate versus rod position was not bounded by the generic analyses presented in Reference 3. Therefore, a plant specific analysis was performed. This analysis demonstrated compliance with the 280 cal/gm design limit for this accident.

2.4.5 Fuel Handling Accident

The fuel handling accident was addressed in the staff's Safety Evaluation Report (SER) on the FSAR and in the staff's evaluation of the topicals on the generic reload for 8x8 fuel (Reference 3). In the generic reload evaluation, the staff stated that the mechanical analysis of the fuel handling accident should be better justified. However, the conclusions drawn in the staff's evaluation of the generic reload that the amount of fission products released from 8x8 assemblies in a refueling accident would not be significantly greater than from the 7x7

assemblies is not changed by this reload; and the conclusions of the SER that the dose consequences of a fuel handling accident are appropriately within 10 CFR Part 100 guidelines are not changed.

2.5 Overpressure Analysis

The licensee has presented an analysis to demonstrate that during the most severe overpressure event an adequate margin (88 psi) exists between the peak vessel pressure and the ASME code allowable vessel pressure which is 110% of the vessel design pressure (Reference 1). The event analyzed was the closure of all main steam line isolation valves with indirect (high flux) scram.

The input to the calculations is listed in Table 6-1a and b of Reference 1 at end of cycle conditions for void coefficient, Doppler coefficient and scram characteristics.

The licensee referenced a sensitivity study which demonstrates that should the transient be initiated at the maximum pressure permitted by the high pressure trip point rather than that assumed for the analysis there would be a reduction in the margin to the pressure limit of approximately 20 psi. It has also been shown that the increase in peak vessel pressure during an MSIV closure due to a failed safety valve would not reduce the margin to the limit by more than approximately 15 psi (Reference 8).

Furthermore, it has been demonstrated that should the MSIV transient be initiated at a value of reactor power slightly above the value assumed for the analysis (because of uncertainties in monitoring of power) there would not be a significant reduction in margin (approximately 10 psi at 102% power) (Reference 9).

Based on the analysis and the sensitivity studies submitted, the overpressure analysis has been found acceptable.

2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in References 3 and 1, respectively. The results show that for both the 7x7 and 8x8 fuel the channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the General Electric Company's 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 afore-mentioned 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. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs change. 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.

Until this issue has been resolved generically, planned operations in the natural circulation flow mode have been restricted by Technical Specifications (Reference 10). The restriction will provide a significant increase in the reactor core stability margins (Reference 10). On the basis of the foregoing, the NRC staff considers the thermal-hydraulic stability to be acceptable.

2.7 Recirculation Pump Startup From The Natural Circulation Operational Mode

During a recent BWR reload review (Reference 10), 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 have not been previously evaluated. Therefore, authorization to operate in this fashion would require additional analyses as to this accident sequence and its consequences. In the absence of this information, the Technical Specifications were previously amended by Amendment No. 33 to eliminate the potential for such an accident (Reference 10). We find this to be acceptable for this cycle of operation as it was for cycle 3 which ended March 17, 1978.

3.0 Physics Startup Testing

The licensee will conduct physics startup tests which in addition to verifying the predicted shutdown margin, are designed to provide assurance that the incore monitoring instrumentation is functioning properly, that the process computer is programmed correctly, and that the core is loaded as intended. The results of these tests will provide additional assurance that the core as loaded is consistent with the physics input submittal (Reference 1). The results of the tests will be submitted to the staff within 90 days of startup. The staff finds the licensee's plan for confirmatory testing and documentation acceptable.

4.0 Maximum Average Planar Linear Heat Generation Rates (MAPLHGR)

In Amendment No. 39 to Facility License No. DPR-49 (DAEC) issued September 30, 1977, we approved increases in MAPLHGR limits for cycle 3 which ended March 17, 1978. The same MAPLHGR limits will apply for cycle 4. In Amendment No. 37 to the DAEC license issued June 20, 1977, we had added a note to the MAPLHGR curves stating: "When core flow is equal to or less than 70% of rated, the MAPLHGR shall not exceed 95% of the limiting values shown". This note was not included, but should be included, on the revised figures 3.12-2, 3.12-3, 3.12-4, 3.12-5 and 3.12-6 that were issued with Amendment No. 39. The licensee operated the Duane Arnold facility throughout cycle 3 in accordance with the requirements of this note. With this amendment, we are reissuing the above figures (pages 3.12-3 thru 3.12-17) with the note added. There is no change in these figures from those issued with Amendment No. 39. The only change is the addition of the note, worded as above, requiring that a reduction in MAPLHGR limits accompany a reduction is core flow for cycle 4 operation.

5.0 Environmental Consideration

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.

6.0 Conclusions

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with operation in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable.

We have concluded, based on the considerations discussed above, that: (1) because the amendment 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 amendment 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: April 20, 1978

TABLE 1

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle			
	B.O.C. to >2000 MWD/T	≤2000 MWD/T to >1000 MWD/T	≤1000 MWD/T to >500 MWD/T	≤500 MWD/T to E.O.C.
7x7	1.26	1.26	1.26	1.30
8x8	1.26	1.29	1.34	1.38

REFERENCES

1. "General Electric Boiling Water Reactor Reload 3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, December 1977.

2. Letter from L. Liu, Iowa Electric Light and Power Company, January 4, 1978.

3. "General Electric Boiling Water Reactor Generic Reload Application For 8x8 Fuel," NEDO-20360, Rev. 1, Supp. 4, April 1, 1976.

4. 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, U. S. Nuclear Regulatory Commission, April 1975.

5. "General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958, November 1973.

6. Letter from J. A. Hinds, GE, to W. Butler, AEC, transmitting Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, dated July 24, 1974.

7. Loss of Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant), NEDO-21082-02-1A, Appendix A, Revision 1, July 1977.

8. Letter from I. F. Stuart, GE, to V. Stello, NRC, dated December 34, 1975.

9. Letter from R. L. Gridley, GE, to D. G. Eisenhut, NRC, daed September 12, 1977.

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 33 to License No. DPR-49, Duane Arnold Energy Center (Docket No. 50-331), dated May 6, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 42 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 changes the Technical Specifications to allow operation of the Duane Arnold Energy Center in Core Cycle 4.

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 4, 1978, (2) Amendment No. 42 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 20th day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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