April 17, 1985

Docket No. 50-331

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

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 117 to Facility Operating 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 August 17, 1984.

The amendment revises the Technical Specifications to incorporate the changes to support the reload and restart for Cycle 8 operation. The Technical Specifications will incorporate changes to (1) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), (2) Minimum Critical Power Ratio (MCRP), and (3) identification of the fuel type.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

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

Enclosures:
1. Amendment No. 117 to License No. DPR-49
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Lee Liu
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Duane Arnold Energy Center

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

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. DPR-49

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

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

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 17, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

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

LIST OF AFFECTED PAGES

vi	3.12-3	3.12-11
vii	3.12-5a	3.12-13
3.3-6	3.12-6	3.12-16*
3.3-18	3.12-7	3.12-17
3.5-26	3.12-8	3.12-19
3.12-1	3.12-9a*	

*These pages are being deleted.

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

LIMITING CONDITION FOR OPERATION

C. Scram Insertion Times

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

% Inserted from Fully Withdrawn	Rod Position	Average Scram Insertion Times (Sec)
5	44	0.375
20	38	0.900
50	24	2.000
90	04	3.500

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

% Inserted from Fully Withdrawn	Rod Position	Average Scram Insertion Times (Sec)
5	44	0.398
20	38	0.954
50	24	2.120
90	04	3.710

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

SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

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

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 requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

4. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poision in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1%ΔK. Deviations in core reactivity greater than 1% ΔK are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3.5 REFERENCES

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

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

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

4. General Electric Company, Analysis of Reduced RHR Service Water Flow at the Duane Arnold Energy Center, NEDE-30051-P, January 1983.

5. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.12 CORE THERMAL LIMITS	4.12 CORE THERMAL LIMITS
Applicability	Applicability
The Limited Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.	The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.
Objective	Objective
The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.	The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.
Specifications	Specifications
A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)	A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-6, -7, -8 and -9. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 2 hours, reduce reactor power to $<$ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.	The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern, the MAPLHGR shall be determined at least once per 12 hours hours.

LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

During reactor power operations, MCPR shall be $\geq$ values as indicated in Figure 3.12-2 at rated power and flow. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within 2 hours, reduce reactor power to $\leq$ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be $\geq$ values as indicated in Figure 3.12-2 times K_f, where K_f is as shown in Figure 3.12-1.

SURVEILLANCE REQUIREMENT

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern, the MCPR shall be determined at least once per 12 hours.

derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients (2). 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 transient which yields the largest ΔCPR. The minimum operating limit MCPR of Specification 3.12.C bounds the sum of the safety limit MCPR and the largest ΔCPR.

The required minimum operating limit MCPRs are determined by the methods described in References 11 and 12. These limits were derived by using the GE 67B scram times, given in Section 3.3.C, which are based upon extensive operating plant data, as well as GE test data. The ODYN Option B scram insertion times were statistically derived from the 67B data to ensure that the resulting Operating Limit from the transient analysis would, with 95% probability at the 95% confidence level, result in the Safety Limit MCPR not being exceeded. The scram time parameter (τ), as calculated by the following formula, is a measure of the conformance of the actual plant control rod drive performance to that used in the ODYN Option-B licensing basis:

$$\tau = \frac{\tau_{ave} - \tau_b}{\tau_a - \tau_b}$$

where: τave = average scram insertion time to Notch 38, as measured by surveillance testing

τ_b = scram insertion time to Notch 38 used in the ODYN Option-B Licensing Basis.

τ_a = 67B scram insertion time to Notch 38

As the average scram time measured by surveillance testing (τave),exceeds the ODYN Option B scram time (τ_b), the Operating Limit MCPRs must be adjusted using Figure 3.12-2.

2. MCPR Limits for Core Flows Other than Rated Flow

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

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of values as indicated in Figure 3.12-2 will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

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

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

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

The K_f factors shown in Figure 3.12-1 are conservative for Duane Arnold operation because the operating limit MCPR of values as indicated in Figure 3.12-2 is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

DELETED

3.12 REFERENCES

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

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

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

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

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

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

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

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

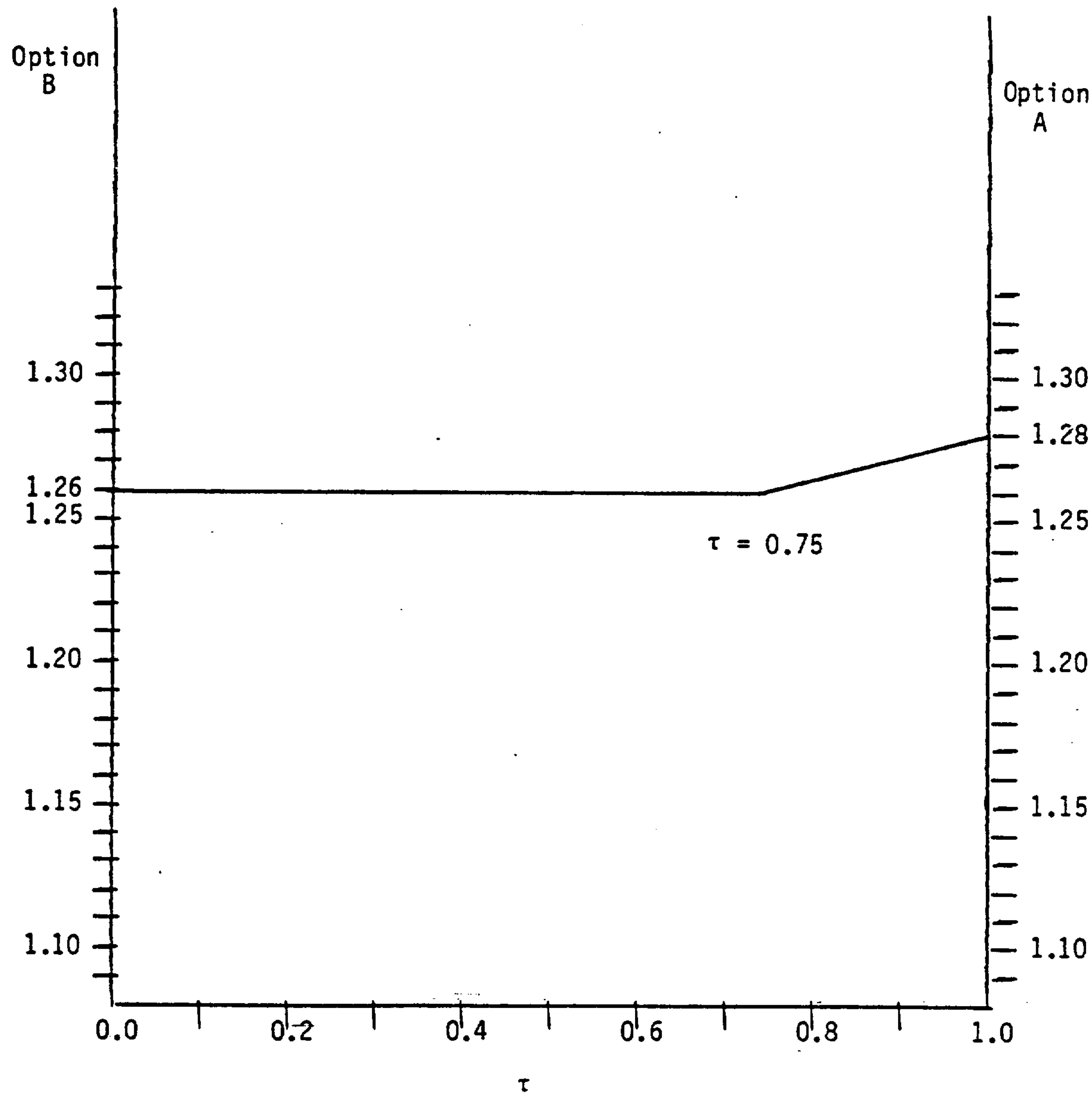
9. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

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

11. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

12. Letter, R. H. Buchholz (GE) to P. S. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits," January 19, 1981.

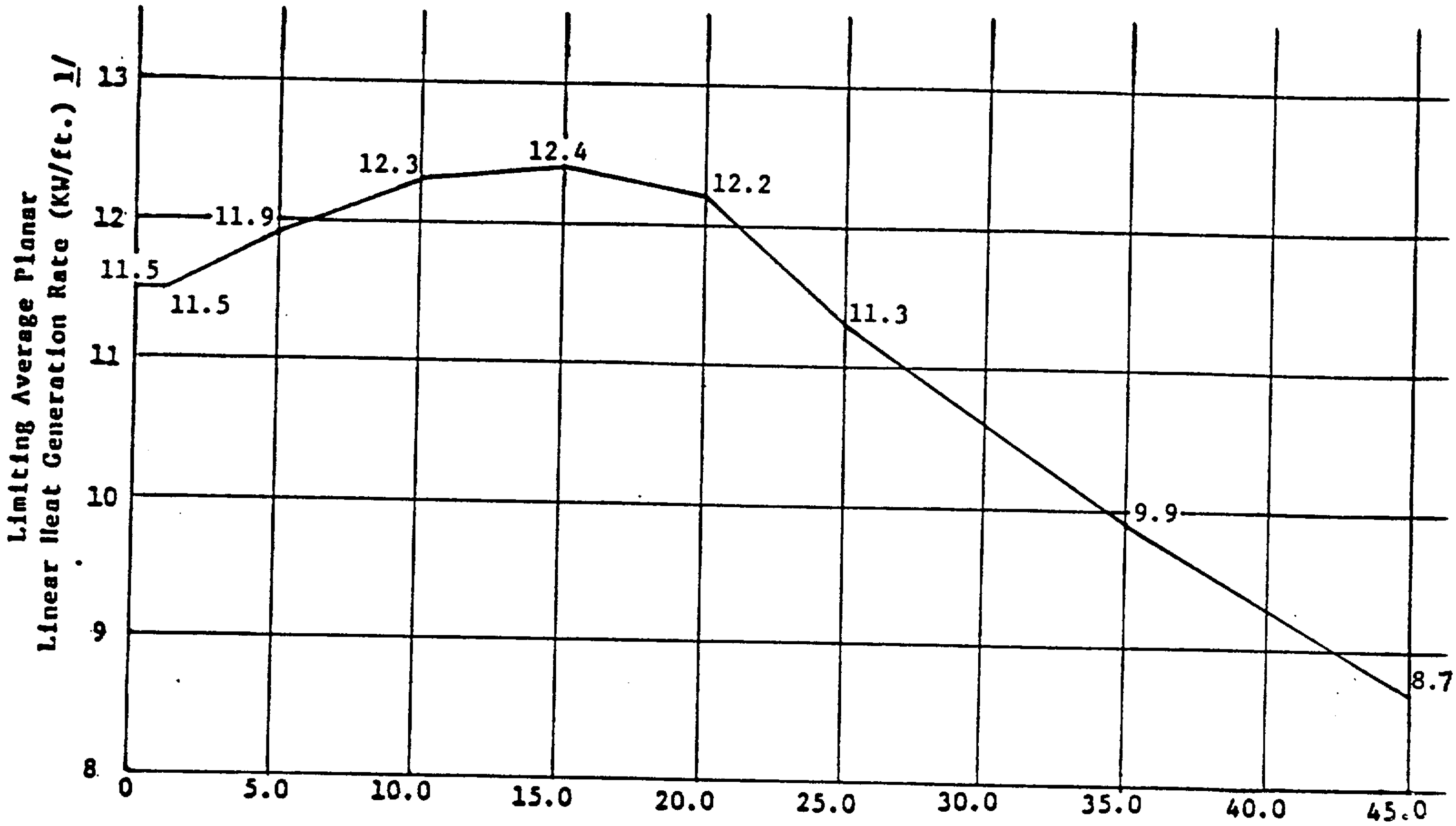
**Approved revision number at time reload fuel analyses are performed.



(based on tested measured scram time as defined in Reference 11)

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ FUEL TYPE: BP/P8X8R FIGURE 3.12-2

DELETED

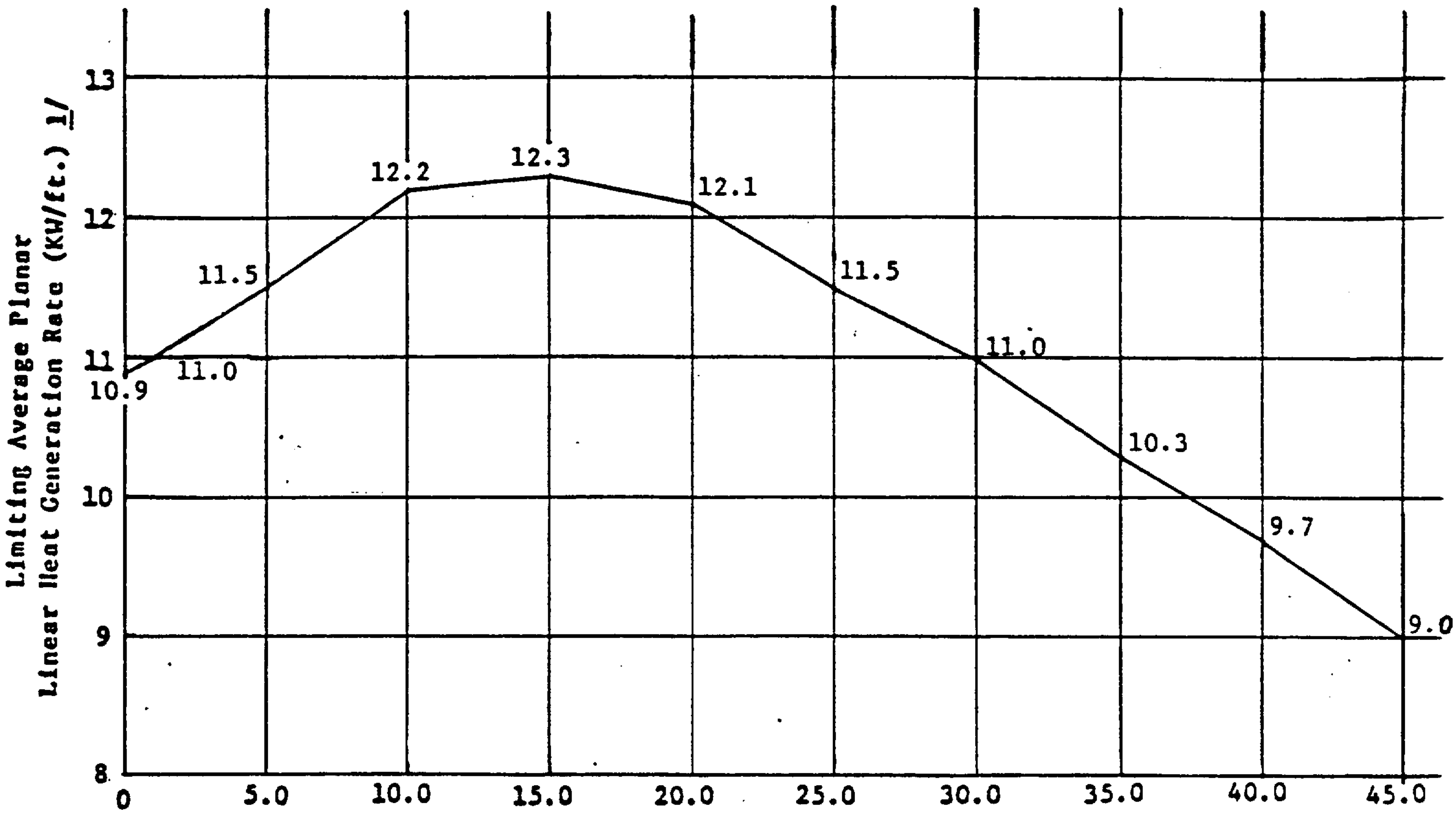


Planar Average Exposure (GWd/t)*

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

* 1 GWd/t = 1000 MWd/t

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



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

* 1 GWd/t = 1000 MWd/t

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: BP/P8DRB299 FIGURE 3.12-8

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 117 TO LICENSE NO. DPR-49

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

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By a letter dated August 17, 1984, the Iowa Electric Light and Power Company (the licensee/IELP) submitted an application to amend the Duane Arnold Energy Center (DAEC) Technical Specifications. The changes were proposed to support the DAEC reload and operation for Cycle 8, and to incorporate administrative changes reflecting revision to figure numbers, table of contents, references, and correction of errors.

During Cycle 8, the licensee proposes to add to the reactor core new types of fuel bundles which are similar to the other fuel except that a thin Zirconium liner has been added to the inner surface of the cladding to reduce cladding failures due to pellet clad interactions. The use of the barrier fuel does not significantly affect the thermal hydraulic performance of the fuel. Based on the analysis of the design basis Loss-of-Coolant Accident (LOCA) and the analysis of the transients, the licensee proposes to revise the Technical Specifications to change (1) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), (2) Minimum Critical Power Ratio (MCPR) operating limits for the facility, and (3) identify the barrier type fuel to be used during Cycle 8. The proposed changes are intended to assure that the fuel performance limits and margins specified in the Final Safety Analysis Report (FSAR) are maintained, and the barrier fuel identified in the Technical Specifications.

The licensee has also proposed miscellaneous administrative changes to achieve consistency throughout the Technical Specifications, update references, and correct typographical errors. Most of these changes reflect the Cycle 8 reload.

2.0 EVALUATION

The staff review of the licensee's August 17, 1983 submittal is summarized as follows:

Fuel Mechanical Design

The fresh fuel which is to be loaded into DAEC for Cycle 8 is of the standard General Electric reload design. This fuel is described in

Reference 3. That reference has been approved for such use and we conclude that no further review of the fuel design is required. The licensee, in a separate action, proposes to install five lead test assemblies in the core for Cycle 8. That proposal is the subject of a separate evaluation.

Nuclear Design

The nuclear design of the Cycle 8 reload core has been performed with methods and techniques that are described in Reference 3. The results of the analyses are given in Reference 1. Those results are within the range of those usually encountered for BWR reloads and are acceptable. In particular, the shutdown margin is greater than 0.01 in K_{eff} and the Standby Liquid Control System is capable of providing a shutdown K_{eff} of 0.97. These results are acceptable and since they have been obtained by previously approved methods, we conclude that the nuclear design of the Cycle 8 reload core is acceptable.

Thermal Hydraulic Design

The thermal hydraulic analysis of the Cycle 8 core was performed with methods and techniques described in Reference 3. Analyses were done at a power level of 1658 thermal megawatts with the assumption of an extended load line limit.

A safety limit value of 1.07 for the core-wide minimum critical power ratio is used for Cycle 8. This value is generic for BWR reloads and is acceptable for Cycle 8.

The operating limit MCPR is obtained by performing analyses of anticipated events in order to determine the reduction in critical power (ΔCPR) resulting from them. Analysis methods, including treatment of uncertainties, are described in Reference 3. The operating limit MCPR is established by adding the largest value of ΔCPR to the safety limit value.

The stability analysis has been performed for the extended load line limit and at 1658 thermal megawatts. The effect of extended load line operation and single loop operation for Cycle 8 will be addressed in a separate evaluation.

Transient and Accident Analyses

Transient and accident analysis methods, described in Reference 3, are the same methods that have been used in previous cycles for DAEC and are acceptable for Cycle 8.

The one-dimensional transient code ODYN has been used to analyze the pressurization events. The licensee has elected to use ODYN Option B in which measured rod scram times are used. For this option the pressurization events are not limiting. If Option A scram times are used, the Load

Rejection Without Bypass event is limiting. Use of the Option B mode is widespread in boiling water reactors and its use is acceptable for DAEC.

The licensee has elected to use the generic analysis results for the rod withdrawal event. This had been approved by the staff for BWR reloads and is acceptable for DAEC. The fuel misorientation event is the limiting event for Option B and establishes the operating limit MCPR for Cycle 8 of 1.26. The analysis of this event has been performed by the approved methods of Reference 3 and is acceptable.

The loss-of-coolant accident (LOCA) has been reanalyzed for Cycle 8 of DAEC at a power of 1691 thermal megawatts (1658 X 1.02) as required by 10 CFR 50.46 and with a full core of assemblies having drilled lower tie plates. The analysis has been approved in a separate evaluation (Amendment No. 115).

A cycle specific rod drop accident analysis has been performed for Cycle 8 of DAEC for the hot shutdown case since the parameters of the generic analysis were not bounding for this case. The result shows that the NRC limit of 280 calories per gram for the peak enthalpy is satisfied. This meets our criterion for the rod drop accident event.

Technical Specifications

The changes to be made to the Technical Specifications are due to the following circumstances:

1. Removal of the last of the 8X8 fuel from the core.
2. Reanalysis of the Loss-of-Coolant Accident with a full core of assemblies having drilled lower tie plates.
3. Reanalysis of the transients and accidents for Cycle 8.
4. Use of the ODYN Option B scram times.
5. Elimination of the requirement for end-of-cycle scram testing.
6. Change of the format for the MCPR values from the present tabular to a graphical presentation.
7. Administrative changes reflecting the above changes and correction of errors.

Each of these items is discussed below:

Removal of 8X8 Fuel

The Technical Specifications are changed in several places to remove references to and operating limits for the 8X8 fuel. These changes are editorial in nature and are acceptable.

LOCA Reanalysis

The new curves of MAPLHGR as a function of exposure (Figures 3.12-6 and 3.12-8) have been obtained from the approved LOCA reanalysis and are acceptable.

Use of ODYN Option B and MCPR Specification Format

The minimum critical power ratio Technical Specification has been rewritten to accommodate use of the ODYN Option B analysis (Reference 3). The tabular form of the permitted value of MCPR has been replaced with a curve of MCPR as a function of the parameter τ. The ODYN transient analyses have been performed for both the Option B scram time and the Option A (Technical Specification) scram time in order to establish end points for the curves.

The curve of MCPR as a function of τ is consistent with the results of the safety analyses and is acceptable. The format of the Technical Specification is similar to that of other plants using Option B and is acceptable.

Reanalysis of Transients and Accidents for Cycle 8

The limiting transient for low values of (scram times near those for Option B) is the fuel misorientation event. The Technical Specification value of MCPR is consistent with the results for this event and is acceptable.

End-of-Cycle Scram Testing

In order to verify that scram time degradation was not occurring between refueling outages DAEC was required, by Amendment No. 54, to perform end-of-cycle scram testing. The additional testing was required to be conducted only through Cycle 6. Therefore, the deletion of this requirement for Cycle 8 is acceptable.

As a result of our review, we conclude that the proposed reload and Technical Specification changes are acceptable. This conclusion is based on the following:

1. Previously approved analysis methods and techniques are employed.
2. The consequences of the transients and accidents which are affected by the reload are acceptable for Cycle 8.
3. The administrative revisions to the Technical Specifications have been found to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

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

4.0 CONCLUSION

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

5.0 REFERENCES

1. General Electric Report 23A1739, Supplemental Reload Licensing Submittal for Duane Arnold Energy Center Unit 1, Reload 7, June 1984.

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

3. NEDE-24011-P-A-6, General Electric Standard Application for Reactor Fuel, April 1983.

Principal Contributor: W. Brooks

Dated: April 17, 1985