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April 25, 1983

Docket No. 50-331

Mr. Duane Arnold
Chairman of the Board and Chief
Executive Officer
Iowa Electric Light and Power Company
P. O. Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

The Commission has issued the enclosed Amendment No. 88 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment is in response to your application dated January 14, 1983 as supplemented by your letter of April 7, 1983. Concerning reload for cycle 7.

This amendment revises the Technical Specifications to incorporate the limiting conditions for operation during fuel cycle 7 by (1) adding the Maximum Axial Planar Linear Heat Generation Rate (MAPLHGR) operating limits based on the General Electric (GE) analysis for the two new bundle types (P8DRB299 & P8DRB284H) added to the core, and (2) changing Minimum Critical Power Ratio (MCPR) operating limits, based on the GE analysis for both 8x8 and P8X8R fuel types.

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

Sincerely,

ORIGINAL SIGNED BY

Frank L. Apicella, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 88 to DPR-49
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

Legal review of Amdt + F.R. notice only

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	AD:OR:DL	OELD		
SURNAME	SNorris	FApicella:pg	DVassallo	GLainas	W.D. Paton		
DATE	4/21/83	4/21/83	4/21/83	4/21/83	4/22/83		

Mr. Duane Arnold
Iowa Electric Light & Power Company

cc:

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Harold F. Reis, Esquire
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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. 88
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 January 14, 1983, as supplemented April 7, 1983, 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. 88 , 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 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 25, 1983.

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by removing the pages listed below and inserting new pages attached. The revised area is identified by a vertical line.

List of Pages Affected

vii	3.12-5
1.0-5	3.12-5a
1.1-2	3.12-9a
3.5-14	3.12-11
3.5-26	3.12-13
3.12-1	3.12-14
3.12-2	3.12-15
3.12-4	3.12-19*
	3.12-20*

* New pages

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

19. ALTERATION OF THE REACTOR CORE (CORE ALTERATION)

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

20. REACTOR VESSEL PRESSURE

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

21. THERMAL PARAMETERS

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

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

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

d. Deleted

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

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

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

h. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 1593 MWth.

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

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

C. Power Transient

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

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

Where: S = Setting in percent of rated power (1,593 MWt)

W = Recirculation loop flow in percent of rated flow. Rated recirculation loop flow is that recirculation loop flow which corresponds to $49x10^6$ lb/hr core flow.

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

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

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

2. APRM High Flux Scram

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

3.5 BASES

A. Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant accident (LOCA) evaluation models described in General Electric Topical Report NEDO-20566 (Ref. 2), the results of the LOCA analysis given in Reference 3 and Subsection 6.3 of the Updated FSAR and in accordance with the acceptance criteria of 10CFR50.46, any of the following cooling systems provides sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel clad temperature to less than 2200°F to assure that core geometry remains intact, and to limit clad metal-water reaction to less than 1%; either of the two core spray subsystems and the LPCI subsystem.

The limiting conditions of operation in Specification 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above.

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-21082-02-1A, Rev. 2, June 1982.

LIMITING CONDITIONS FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability

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

Objective

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

Specifications

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

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-5, -6, -7, -8 and -9. If at at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 2 hours, reduce reactor power to < 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENTS

4.12 CORE THERMAL LIMITS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

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

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2. During operation with a limiting control rod pattern, the MAPLHGR shall be determined at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Linear Heat Generation Rate (LHGR)	B. Linear Heat Generation Rate (LHGR)
1. During reactor power operation the linear heat generation rate (LHGR) of any rod in any 8x8 fuel assembly shall not exceed 13.4 KW/ft. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 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 LHGR as a function of core height shall be checked daily during reactor operation at $>$ 25% 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 LHGR shall be determined at least once per 12 hours.

3.12 BASES: CORE THERMAL LIMITS

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

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50.46.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50.46 limit.

The calculational procedure used to establish the MAPLHGRs is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10CFR Part 50.

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 eneration rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient 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% condifence, 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 > 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 Maximum Total Peaking Factor (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 [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 a any time during the transient assuming instrument trip settings given in Specification 2.1.

TABLE 3.12-2

MCPR LIMITS

Fuel Type	
8 x 8	1.25
8 x 8R/P8 x 8R	1.27

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Rev.2, June 1982.

2. "Generic Reload Fuel Application," NEDE-24011-P-A**.

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

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

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

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

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

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

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

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

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

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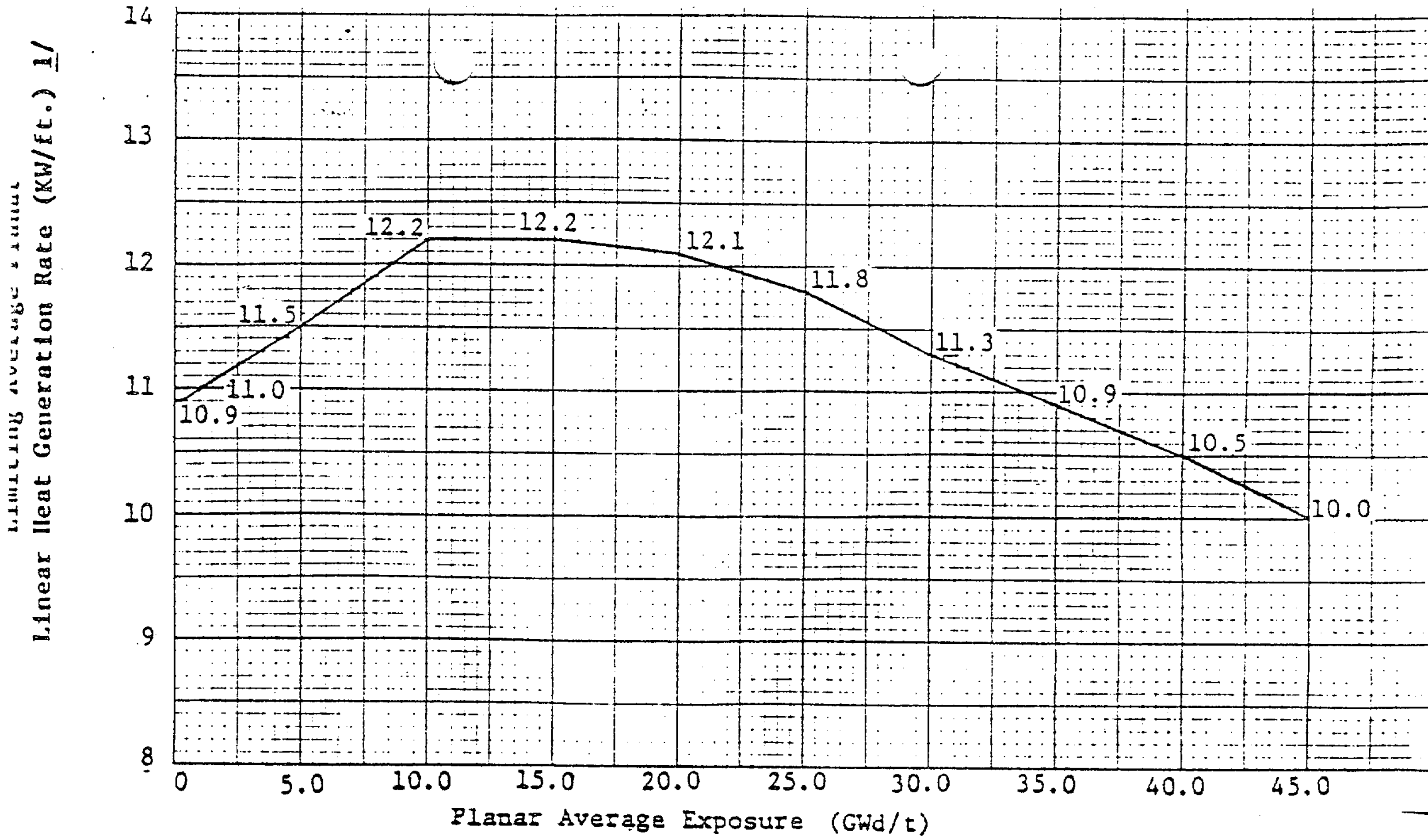
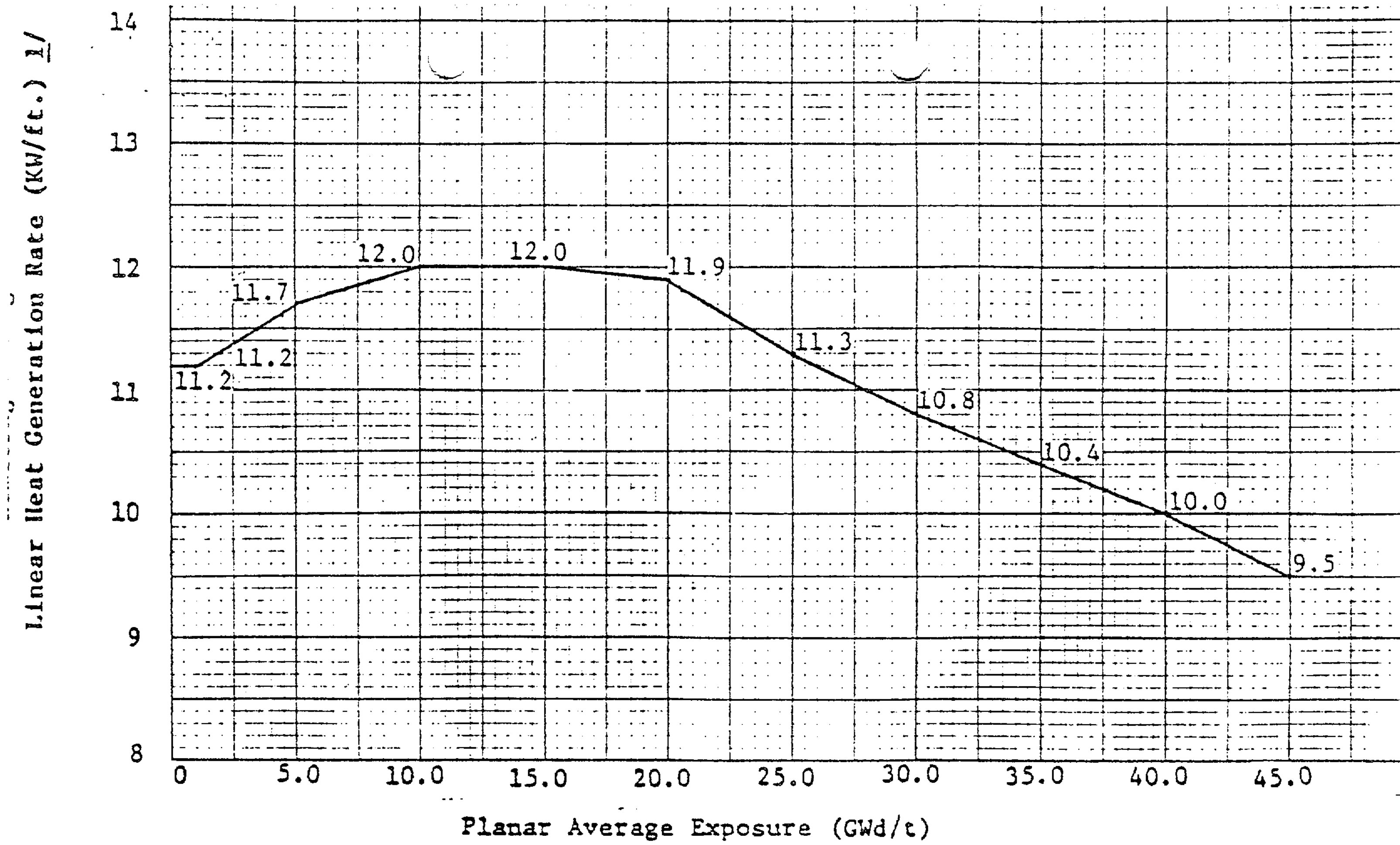
Amendment No. 88



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: P8DRB299 FIGURE 3.12-8



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

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT AND POWER COMPANY TECHNICAL SPECIFICATIONS
LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE FUEL TYPE: P8DRB284H FIGURE 3.12-9

Amendment No. 88

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 88 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 14, 1983 (Ref. 1), the Iowa Electric Light & Power Company (licensee) made application to amend the Technical Specifications of Operating License DPR-49 for the Duane Arnold Energy Center (DAEC) in order to operate the plant for fuel Cycle 7. In support of this application, the licensee also provided a supplemental reload licensing submittal (Ref. 2). We have reviewed these submittals and our evaluation follows.

2.0 Evaluation

2.1 Fuel Design Evaluation

The reload application contains four fuel-design related issues: (1) the replacement of all 7X7 fuel assemblies with the newer P8X8R fuel assemblies, (2) the analysis of safety considerations involved in the determination of Cycle 7 operating limits, (3) the reanalysis of the loss-of-coolant accident (LOCA) with the incorporation of extended maximum average planar linear heat generation rate (MAPLHGR) limits, and (4) the reanalysis for the control rod drop accident (RDA).

Replacement of 7X7 Fuel Assemblies

The Cycle 7 reload fuel is comprised of 128 standard-design P8X8R fuel assemblies. These assemblies will replace the last of the 7X7 predecessor fuel assemblies. The Cycle 7 core inventory is given in Table 1.

TABLE 1

DUANE ARNOLD CYCLE 7 CORE INVENTORY

Assembly Designation	Cycle Loaded	Number*
8DB274H	4	68
P8DB289	5	88
P8DPB289	6	84
P8DRB284H	7	88
P8DRB299	7	40
		368

*All assemblies are drilled.

Cycle 7 Operating Limits

The licensee's analysis of the safety considerations involved in the determination of Cycle 7 operating limits is set forth in the reload report (Ref. 2). In all fuel-design-related areas, except those separately identified, the reload report relies on the generic report, General Electric Standard Application for Reactor Fuel (Ref. 3), which we previously reviewed and approved (Ref. 14).

Loss-of-Coolant Accident

The licensee has submitted (Ref. 4) the results of a LOCA analysis that addresses the fresh Cycle 7 fuel and revised MAPLHGR limits for all Cycle 7 fuel types.

The LOCA analysis was performed using the General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K (Ref. 5) as amended (Ref. 6) in 1977. In 1981, the NRC conditioned (Ref. 7) the use of the GE emergency core cooling system (ECCS) evaluation model (EM) to require that plant analyses performed with the GE evaluation model be accompanied by supplemental calculations performed with a specified set of material correlations from NUREG-0630

(Ref. 8). This condition was subsequently removed (Ref. 9) following a GE modification to the cladding rupture temperature model. The DAEC LOCA analysis did not utilize the new "adjusted" GE rupture temperature model. Nevertheless, the licensee has addressed (Ref. 10) this issue and determined that for the DAEC Cycle 7 application there would have been no difference in LOCA analysis if the "adjusted" model had been employed. This is because the ruptures predicted for DAEC are in the temperature regime where the "adjusted" model coincides with the original GE model. Consequently, we conclude that the licensee's use of the original GE rupture temperature model is acceptable.

The licensee's submittal also provided revised MAPLHGR limits that have been extended to accommodate an exposure of 45 GWd/MTU. These limits were generated by methods previously approved (Ref. 6). Although the methodology used is generically applicable for the MAPLHGR limit determination, we believe that the effects of enhanced fission gas release in high-burnup fuel (i.e., greater than 20 GWd/MTU) were not adequately considered in the fuel performance model. In response to this concern, GE requested (Refs. 11 and 12) that credit for approved, but unapplied, ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at higher burnups. We found this proposal acceptable (Ref. 13), provided that certain plant-specific conditions were met. The licensee has stated (Ref. 10) that the GE proposal is applicable to the Duane Arnold analysis. On this basis, we conclude that the MAPLHGR limits proposed for Cycle 7 operation of Duane Arnold are in conformance with the requirements of 10 CFR 50.46 and are acceptable.

Control Rod Drop Accident

A reanalysis for the control rod drop accident was performed with generic-bounding and plant-specific inputs. The resultant peak enthalpy was found to be 236.6 cal/g. The calculated value is less than the acceptance criterion (i.e., 280 cal/g) given in Section 15.4.9 of the Standard Review Plan (NUREG-0800). We, therefore, conclude that the analysis meets the pressure boundary integrity and coolability requirements of the General Design Criterion 28 of Appendix A to 10 CFR 50 and is, hence, acceptable.

Changes to the Technical Specifications

Changes to the Technical Specifications, concerning the replacement of 7x7 fuel assemblies, Cycle 7 operating limits, and loss-of-coolant accident are acceptable, based on the above evaluation.

2.2 Thermal and Hydraulic Design Evaluation

The objective of the review is to confirm that the thermal-hydraulic design of the core has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and that the core is not susceptible to thermal-hydraulic instability.

The review includes the following areas: (1) safety limit minimum critical power ratio (MCPR), (2) operating limit MCPR, (3) thermal-hydraulic stability, and (4) changes to Table 3.12-2 of the Technical Specifications.

The licensee has submitted an analysis report for Cycle 7 operation (Ref. 2). This report relies on a generic document (Ref. 3) which has been reviewed and approved (Ref. 14) by the staff. Discussion of our review concerning the thermal-hydraulic design for Cycle 7 operation follows.

Safety Limit MCPR

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during normal and anticipated operational transients. As stated in Reference 3, the approved safety limit MCPR is 1.07. This safety limit MCPR of 1.07 is used for the DAEC Cycle 7 operation.

Operating Limit MCPR

The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Item 9 of Reference 2 are plant-specific values calculated by including the ODYN computational method. The calculated ΔCPRs are adjusted to reflect Option A ΔCPRs by employing the conversion method described in Reference 15. The MCPR values are determined by adding the adjusted ΔCPRs to the safety limit MCPR. The maximum cycle MCPR values (Option A) in Item 11 of Reference 2 are specified as the operating limit MCPRs and incorporated into the Technical Specifications. Since an approved method was used to determine the operating limit MCPRs to avoid violation of the safety limit MCPR in the event of any anticipated transients, we conclude that these limits are acceptable.

Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Ref. 2) show that the maximum reactor core stability decay ratio is 0.85, which is less than the calculated value for some operating reactors which have been previously approved. Since the operation in the natural circulation mode will be prohibited by Technical Specification 3.3.E(LCO), there will be added margin to the stability limit and we, therefore, conclude that the thermal-hydraulic stability results are acceptable for Cycle 7 operation.

Changes to the Technical Specifications

Changes to the Technical Specifications concerning the safety limit minimum critical power ratio (MCPR) and the operating limit MCPR for Cycle 7 are acceptable, based on the above evaluation.

2.3 Evaluation Summary

We have reviewed the fuel and thermal-hydraulic design related issues submitted, and we find, based on the above, that the reload safety analysis for Cycle 7 operation of DAEC, including the necessary changes to the Technical Specifications are acceptable.

3.0 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.

4.0 Conclusion

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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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 25, 1983

Principal Contributors: D. Powers, T. Huang

5.0 References

1. L. D. Root (IEL&P) letter to H. R. Denton (USNRC), January 14, 1983.

2. "Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center Reload 6," GE Report Y1003J01A46, January 1983.

3. "General Electric Standard Application for Reactor Fuel," GE Report NEDE-24011-P-A-4, April 1978.

4. "General Electric, Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant)," GE Report NEDO-210282-02-1A, Revision 2, June 1982.

5. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," GE Report NEDO-20566, 1974.

6. D. G. Eisenhut (USNRC) letter to E. D. Fuller (GE), June 30, 1977.

7. R. L. Tedesco (USNRC) letter to G. G. Sherwood (GE), February 4, 1981.

8. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analyses," NRC Report NUREG-0630, April 1980.

9. H. Bernard (USNRC) letter to G. G. Sherwood (GE), May 11, 1982.

10. L. D. Root (IEL&P) letter to H. R. Denton (USNRC), April 7, 1983.

11. R. E. Engel (GE) letter to T. A. Ippolito (USNRC), May 6, 1981.

12. R. E. Engel (GE) letter to T. A. Ippolito (USNRC), May 28, 1981.

13. L. S. Rubenstein (USNRC) memorandum for T. M. Novak, "Extension of General Electric Emergency Core Cooling Systems Performance Limits," June 25, 1981.

14. Letter from D. G. Eisenhut (USNRC) to R. Gridley (GE), May 12, 1978.

15. NEDE-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

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. 88 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 revises the 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.

This amendment revises the Technical Specifications to incorporate the limiting conditions for operation during fuel cycle 7 by (1) adding the Maximum Axial Planar Linear Heat Generation Rate (MAPLHGR) operating limits based on the General Electric (GE) analysis for the two new bundle types (P8DRB299 & P8DRB284H) added to the core and (2) changing Minimum Critical Power Ratio(MCPR) operating limits based on the GE analysis for both 8x8 and P8x8R fuel types.

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 14, 1983, as supplemented April 7, 1983, (2) Amendment No.88 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 Licensing.

Dated at Bethesda, Maryland, this 25th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

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