September 19, 1986

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. 137 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications in response to your application dated October 12, 1984.

The amendment revises the DAEC Technical Specifications to incorporate changes reflecting the elimination of differential pressure between the drywell and the wetwell of the DAEC containment.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by
[illegible]

Mohan C. Thadani, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 137 to
 License No. DPR-49
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	SNorris	BGrimes	OPA
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SNorris:	MThadani	S H Lewis	DMuller
9/10/86	9/10/86	9/12/86	9/17/86

No legal objection.

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PDR ADOCK 05000331
P PDR

Mr. Lee Liu
Iowa Electric Light and Power Company

Duane Arnold Energy Center

cc:
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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. 137
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, dated October 12, 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. 137, 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 and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W Rivenbark for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 137

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 areas are indicated by marginal lines.

Pages	
3.2-21	
3.2-31	
3.7-14	
3.7-14a	(deleted)
3.7-41	
3.7-49	

TABLE 3.2-F

SURVEILLANCE INSTRUMENTATION

Total No. Channels Provided	Minimum No. Channels Required	Instrument	Type Indication and Range	Action
3	2	Reactor Water Level	Recorder, Indicator 158"-218"*	(1) (2) (3)
3	2	Reactor Pressure	Recorder, Indicator 0-1200 psig	(1) (2) (3)
2	2	Drywell Pressure	Recorder -10 to +90 psig	(1) (2) (3)
8	2	Drywell Temperature	Recorder 0-350°F	(1) (2) (3)
2	2	Torus Water Temperature	Recorder 0-350°F	(1) (2) (3)
2	2	Torus Water Level	Recorder -10"/0/+10" H_2O	(1) (2) (3)
2	2	Containment Water Level	Recorder, Indicator -20 to +80 feet	(1) (2) (3)
2	1	Control Rod Position	Process Computer, Full Core Display, Four Rod Group Display	
4	3	Neutron Monitoring	SRM***(10^{-1} to 10^{6} CPS)	(1) (2) (3) (4)
3(per Trip System)	2(per Trip System)	Neutron Monitoring	IRM,*** APRM 0 to 125% power	(1) (2) (3) (4)
1	1	Drywell Pressure	Local Indicator,** 0-100 psia	(5)
1	1	Torus Pressure	Local Indicator, ** 1-100 psia	(5)

*Indicator scale is referenced to the Top of Active Fuel (TAF), defined as 344.5 inches above vessel zero.
**Capable of ±0.1 psi
***Not required when in the Run mode.

Amendment No. 40, 137

TABLE 4.2-F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Level	Once/6 months	Once Each Shift
2) Reactor Pressure	Once/6 months	Once Each Shift
3) Drywell Pressure	Once/6 months	Once Each Shift
4) Drywell Temperature	Once/6 months	Once Each Shift
5) Suppression Chamber Temperature	Once/6 months	Once Each Shift
6) Suppression Chamber Water Level	Once/6 months	Once Each Shift
7) Control Rod Position	NA	Once Each Shift
8) Neutron Monitoring	Prior to Reaching 20% Power and once per day when in Run Mode (APRM Gain Adjust when in Run Mode)	Once Each Shift (When in Startup or Run Mode)
9) Drywell Pressure	Once/Operating Cycle	Once Each Shift
10) Torus Pressure	Once/Operating Cycle	Once Each Shift

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
7. (Deleted)	7. (Deleted)
8. If the specifications of 3.7.A.1 through 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	
9. <u>Purging</u> The time which containment vent/purge valves (CV-4302, CV-4303, CV-4300, CV-4301 and CV-4307) can be open is limited to a maximum of 90 hours per calendar year, not including the 24 hour period prior to shutdown and the 24 hour period subsequent to placing the reactor in the run mode following a shutdown as specified in 3.7.A.5.b. This restriction applies whenever primary containment integrity is required.	
10. If Specification 3.7.A.9 cannot be met, prepare and submit a Special Report to the Commission pursuant to Specification 6.11.3 within the next 30 days outlining the cause of the limits being exceeded and the plans for limiting the time which these valves will be open.	

Due to the nitrogen addition, the pressure in the containment after a LOCA could possibly increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 70 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment System in order to minimize the offsite dose.

Following a LOCA, periodic operation of the drywell and torus sprays may be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

7. Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shut down and the drywell is

3.7.A & 4.7.A REFERENCES

1. "Duane Arnold Energy Center Power Uprate", NEDC-30603-P, May, 1984 and Attachment 1 to letter L. Lucas to R.E. Lessly, "Power Uprate BOP Study Report," June 18, 1984.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR Part 50, Appendix J, Reactor Containment Testing Requirements, Federal Register, April 19, 1976.

5. Deleted

6. Deleted

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 137 TO FACILITY OPERATING 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 letter dated October 12, 1984, the Iowa Electric Light and Power Company (the licensee) requested revision to the Technical Specifications for the Duane Arnold Energy Center (DAEC) reflecting elimination of the requirement for a differential pressure system between the drywell and the wetwell of the DAEC Mark I containment. The differential pressure system was established as an interim measure as a part of the Mark I Containment Short Term Program, for the purpose of providing a reduction in potential loads during a postulated loss of coolant accident (LOCA) and an associated restoration of the margins of safety to approximately two. Subsequently, as a part of Mark I Containment Long Term Program, the licensee modified the containment to establish acceptable structural safety margins to withstand dynamic loads under conditions of loss of coolant accident (LOCA). Therefore, the differential pressure system is no longer needed.

2.0 EVALUATION

In July 1980 the NRC issued NUREG-0611, "Safety Evaluation Report, Mark I Containment Long-Term Program," to address the NRC acceptance criteria for the Mark I Containment Long-Term Program. As a result of additional evaluation work, the NRC issued Supplement 1 to NUREG-0611 in August 1982.

On December 30, 1982, the licensee submitted to the NRC a Plant Unique Analysis Report (PUAR) for the DAEC Mark I Containment Long-Term Program. The PUAR documented the licensee's reassessment of the DAEC containment to withstand the additional design loads postulated to occur during a LOCA, or a safety relief valve (SRV) discharge event. The reanalysis was performed in accordance with NUREG-0611 and its supplement. The licensee concluded that upon completion of all modifications addressed in the PUAR the containment system design safety margin was restored to the safety margin intended at the time of plant licensing for operation. The PUAR analyses were performed with a drywell to wetwell differential pressure of zero.

By letter dated September 11, 1985, the NRC transmitted to Iowa Electric Light and Power Company the Safety Evaluation (SE) covering the PUAR for the DAEC Mark I Containment Long-Term Program. In the SE the NRC concluded that acceptable safety margins had been established for all pool dynamic loads under LOCA and SRV discharge loads, and that the established margins for structural integrity under LOCA conditions are acceptable. Therefore, the licensee plans to eliminate the differential pressure system and change the Technical Specifications accordingly.

The NRC Senior Resident Inspector at the DAEC has informed the staff that all the facility modifications addressed by the licensee in the PUAR have been completed. The completion of these modifications have restored the design safety margin of the DAEC containment system to the safety margin intended at the time the plant was licensed for operation. The staff has also reviewed the requested Technical Specification changes and finds that they are consistent with the elimination of the differential pressure systems and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 the 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Maura

Dated: September 19, 1986