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TJCarter

NOV 5 1974

Docket No. 50-331

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility License No. DPR-49 for Duane Arnold Energy Center. This amendment includes Change No. 5 to the Technical Specifications, Appendix A, and is in response to your request dated June 21, 1974.

This amendment changes the Technical Specifications pertaining to the Reactor Protection System Trip to specify a time interval following start of Turbine Control Valve Fast Closure rather than an oil pressure setting. It also includes changing the specified water level in the Suppression Chamber at which transfer of the High Pressure Coolant Injection pump suction occurs.

Copies of the Safety Evaluation and the Federal Register Notice relating to this action are enclosed.

Sincerely,

Original Signed

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 4 to License No. DPR-49
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

See next yellow sheet

OFFICE ➤	ORB#3	ORB#3	ORB#3	OGC	L:AD/OR	Const
SURNAME ➤	SATeets:kmf	GOwsley	GLear		KRGoller	
DATE ➤	9/ 18 /74	9/ 19 /74	9/ 23 /74	9/ /74	9/ /74	

cc: w/enclosure

Jack R. Newman, Esquire
Harold F. Reis, Esquire
Newman, Reis & Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

Mr. Dudley Henderson
Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

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

Reference Service
Cedar Rapids Public Library
426 3rd Avenue, S. E.
Cedar Rapids, Iowa 52041

OFFICE ➤						
SURNAME ➤						
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SVarga
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extra cps
CHebron

Docket No. 50-331

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility License No. DPR-49 for Duane Arnold Energy Center. This amendment includes Change No. 5 to the Technical Specifications, Appendix A, and is in response to your request dated June 21, 1974.

This amendment changes the Technical Specifications pertaining to the Reactor Protection System Trip to specify a time interval following start of Turbine Control Valve Fast Closure rather than an oil pressure setting. It also includes changing the specified water level in the Suppression Chamber at which transfer of the High Pressure Coolant Injection pump suction occurs.

The Safety Evaluation and the Federal Register Notice relating to this action are enclosed.

Sincerely,

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 4 to License No. DPR-49
2. Federal Register Notice
3. Safety Evaluation

cc: See next page

Subject to indicated typo correction on FIR notice here

OFFICE	ORB#3	ORB#3	ORB#3	OGC	L:AD/ORs KRG
SURNAME	SATeets Hmf	GOwsley	GLear	W.D. Paton	KRGoller
DATE	10/15/74	10/15/74	9/ /74	10/23/74	10/4/74

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

cc: w/enclosure

Jack R. Newman, Esquire
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1025 Connecticut Avenue, N. W.
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Des Moines, Iowa 50319

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Cedar Rapids Public Library
426 3rd Avenue, S. E.
Cedar Rapids, Iowa 52041

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

IOWA ELECTRIC LIGHT AND POWER COMPANY, CENTRAL IOWA POWER COOPERATIVE

AND CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-49

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by Iowa Electric Light & Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative (the licensee) dated June 21, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 5."

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

FOR THE ATOMIC ENERGY COMMISSION

Original Signed

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 5 to the
Technical Specifications

Date of Issuance: NOV 5 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 4
CHANGE NO. 5 TO TECHNICAL SPECIFICATIONS
FACILITY OPERATING LICENSE NO. DPR-49

Replace pages 1.1-3 & 1.1-4, 3.1-3 & 3.1-4, 3.1-11 & 3.1-12, 3.1-13 & 3.1-14, 3.1-21 & 3.1-22 and 3.2-13 & 3.2-14 with the attached pages (revisions contained in pages 1.1-4, 3.1-4, 3.1-12, 3.1-14, 3.1-21 and 3.2-13 only).

SAFETY LIMIT		LIMITING SAFETY SYSTEM SETTING	
B.	When the reactor pressure is less than 600 psig, or core coolant flow is less than 10 percent of design, the reactor thermal power shall not exceed 268 MWt (about 17 percent of rated power).	B.	APRM Rod Block For operation with combinations of power and peak flux below the curve for total peaking factor of 2.62 in Fig.2.1-2 the APRM Control Rod Block setpoint shall be as shown on Fig. 2.1-1 and shall be: $S \leq (0.66\ W + 42)$ The definitions used above for the APRM scram trip apply. For combinations of power and peak flux above the curve for total peaking factor of 2.62 the APRM Control Rod Block setpoint shall be: $S \leq (0.66\ W + 42)\ \frac{(2.62)}{TPF}$
C.	To ensure that the Safety Limit established in Specification 2.1.A and 2.1.B is not exceeded, each required scram shall be initiated by its primary source signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the Primary Source Signal.	C.	Scram and Isolation reactor low water $\geq$ 514.5 in. above vessel zero (+12" on level instruments
D.	Whenever the reactor is in the shutdown condition 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.	D.	Scram - turbine stop $\leq$ 10 percent valve closure

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING		
	E.	Turbine Control Valve Fast Closure - scram shall occur within 30 milliseconds of the start of Turbine Control Valve Fast Closure.	5
	F.	Scram - main steam line isolation	≤10 percent valve closure
	G.	Main steam isolation valve closure - nuclear system low pressure	≥880 psig
	H.	Core spray & LPCI actuation - reactor low water level	≥363 in. above vessel zero (-139.5 in. indicated level)
	I.	HPCI & RCIC actuation - reactor low water level	≥464 in. above vessel zero (-38.5 in. indicated level)
	J.	Main steam isolation valve closure - reactor low water level	≥464 in. above vessel zero (-38.5 in. indicated level)

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
2	IRM High Flux	≤ 120/125 of Full Scale	X	X	(5)	6 Instrument Channels	A
2	IRM Inoperative		X	X	(5)	6 Instrument Channels	A
2	APRM High Flux	(.66W+54) (2.62/P.F.) (11) (12)			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B
2	APRM Downscale	≥ 5 Indicated on Scale			(9)	6 Instrument Channels	A or B
2	APRM High Flux in Startup	≤ 15% Power	X	X		6 Instrument Channels	A
2	High Reactor Pressure	≤ 1035 psig	X(8)	X	X	4 Instrument Channels	A

TABLE 3.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels for Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6)	Startup	Run	Number of Instrument Channels Provided by Design	Action (1)
2	High Drywell Pressure	≤ 2.0 psig	X(7)	X(8)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥ +12" Indicated Level	X	X	X	4 Instrument Channels	A
2	High Water Level in Scram Discharge Volume	≤ 60 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Main Steam Line High Radiation	≤ 3 X Normal Rated Power Background *(14)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3)(13)	X(3)(13)	X (13)	8 Instrument Channels	A or C
2	Turbine Control Valve Fast Closure (Loss of Control oil Pressure)	Within 30 milliseconds of the start of Control Valve Fast Closure.			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure			X(4)	8 Instrument Channels	A or D
2	Turbine Control Valve - Fast Closure	Valve 10% closed			X(4)	4 Instrument Channels	A or D

*Alarm setting ≤ 1.5 X Normal Rated Power Background

Change No. 5
Dated:

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after completion of the functional test program.

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	On Controlled Shutdown
APRM High Flux			
Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	With Standard Pressure Source	Every refueling
LPRM Signal	B	TIP System Traverse	Every 1,000 EFPH
High Reactor Pressure	A	Standard Pressure Source	Every 3 months
High Drywell Pressure	A	Standard Pressure Source	Every 3 months
Reactor Low Water Level	A	Pressure Standard	Every 3 months
High Water Level in Scram Discharge Volume	A	Water Column	Every refueling
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Oil Pressure Trip	A	Note (6)	Once per operating cycle

Change No. 5
Dated:

TABLE 4.1-2 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration (4)	Minimum Frequency
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Reactor Pressure Permissive	A	Standard Pressure Source	Every 6 months
Turbine Control Valve Fast Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the bases of this Specification.

2. Calibration test is not required on the parts of the system that are not required to be operable or are tripped but is required prior to return to service.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Response time is not a part of the routine instrument channel test but will be checked once per operating cycle.

5. Physical inspection and actuation of these position switches will be performed during the refueling outages.

6. Measure time interval base line data for each operating cycle as follows: From energization of fast acting solenoid, measure time interval to response of oil pressure switch, HFA relay (RPS) and position response of control valves.

Change No. 5
Dated: NOV 5 1974

to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

Turbine stop valve closure occurs at 10% of valve closure. Below 205 psig turbine first stage pressure (30% of rated), the scram signal due to turbine stop valve closure is by-passed because the flux and pressure scrams are adequate to protect the reactor

Turbine control valve fast closure scram trip shall initiate within 30 milliseconds of the start of control valve fast closure. The trip level setting is verified by measuring the time interval from energizing the fast acting solenoid (from valve test switch) to pressure switch response; the measured result is compared to base line data taken during each refueling outage. Turbine control valve fast closure is sensed by measuring disc dump electro-hydraulic oil line pressure (Relay Emergency Trip Supply) which decreases rapidly upon generator load rejection. This scram is only effective when turbine steam flow is above 30% of rated as measured by turbine first stage pressure (approximately 206 psia).

5

The requirement that the IRM's be inserted in the core when the APRM's read 5 as indicated on the scale in the Startup and Refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

Change No. 5
Dated: NOV 5 1974

4.1 BASES

1. The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in Reference 6. This concept was specifically adapted to the one out of two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failures such as blown fuses, ruptured bourdon tubes, faulted amplifiers, and faulted cables, which result in "upscale" or downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1-1 and 4.1-2 are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

TABLE 3.2-B (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
5 2	Suppression Chamber High Level	$\leq$5" above normal water level.	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber 5
1	RCIC Turbine High Flow	$\pm$ 180" H_2O (2)	2 Inst. Channels	
2	RCIC Turbine Equipment Room High Ambient Temperature	$\leq$ 175 deg. F (2)	4 Inst.	
2	RCIC Vent High Differential Temperature	$\leq \Delta$ 50 deg. F (2)	4 Inst.	
2	RCIC Steam Line Low Pressure	100 > P > 50 psig (2)	4 Inst.	
1	HPCI Turbine Steam Line High Flow	$\pm$ 225" H_2O (3)	2 Inst. Channels	
2	Suppression Pool Area High Ambient Temperature	150°F	4 Inst. Channels	1
2	Suppression Pool Area High Diff. Temperature	50°F	4 Inst. Channels	
1	HPCI Leak Detection Time Delay	15 min.	2 Inst.	

Change No. 5
Dated:

TABLE 3.2-B (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Leak Detection Time Delay	30 min.	2 Inst.	
2 (5)	HPCI Steam Line Low Pressure	100 > P > 50 psig (3)	4 Inst.	
2	HPCI Equipment Room High Ambient Temperature	≤ 175 deg. F	4 Inst.	
2	HPCI Equipment Room High Diff. Temperature	≤ Δ 50 deg. F (3)	4 Inst.	
1 per 4 kV Bus	4 kV Emergency Bus Undervoltage Relay	20% of Rated Voltage		1. Trips all loaded breakers 2. Fast transfer permissive 3. Dead bus start of
2 per 4kV Bus	Emergency Transformer Undervoltage	85% of Rated Voltage		1. Trips emergency transformer feed to 4KV emergency bus 2. Fast transfer permissive

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 4 TO LICENSE NO. DPR-49

(CHANGE NO. 5 TO TECHNICAL SPECIFICATIONS)

IOWA ELECTRIC LIGHT AND POWER COMPANY

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

Introduction

By letter dated June 21, 1974, Iowa Electric Light and Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). The proposed changes concern the Reactor Protection System Trip from Turbine Control Valve Fast Closure and the Suppression Chamber water level at which transfer of HPCI pump suction to the Suppression Chamber occurs. The licensee proposes that the Turbine Control Valve Fast Closure Trip be revised to specify a time interval following start of Control Valve Fast Closure rather than an oil pressure setting; specification of a time interval provides that the trip will occur in response to the critical safety related parameter rather than to the parameter (oil pressure) which is associated with valve operation. With respect to transfer of HPCI pump suction, the licensee proposes a higher allowable suppression chamber water level as the trip setting, a change which is consistent with practice of other operating BWR's and which the licensee believes necessary for operating flexibility.

Discussion

A. Turbine Control Valve Fast Closure

The licensee has requested that the following changes be made with respect to the Reactor Protection System trip from Turbine Control Valve Fast Closure:

Radiological Technical Specification Proposed Change No. 15

p. 3.1-4, Table 3.1-1: Change Trip Level Setting for Turbine Control Valve Fast Closure (Loss of Control Oil Pressure) from "SetPoint $\geq$ 800 psig Control Oil Pressure" to "within 30 milliseconds of the start of Control Valve Fast Closure".

p.3.1-12, Table 4.1-2: Change Calibration for Turbine Control Valve Oil Pressure Trip from "Standard Pressure Source" to "Note 6".

p.3.1-14, Notes for Table 4.1-2:

Add the following Note 6, "Measure time interval base line data for each operating cycle as follows: From energization of fast acting solenoid, measure time interval to response of oil pressure switch, HFA relay (RPS) and position response of control valves."

p.3.1-21, Second paragraph of bases:

Delete the paragraph beginning with "Turbine control valve fast closure---" and substitute the following: "Turbine control valve fast closure scram trip shall initiate within 30 milliseconds of the start of control valve fast closure. The trip level setting is verified by measuring the time interval from energizing the fast acting solenoid (from valve test switch) to pressure switch response and compared to base line data taken during each refueling outage. Fast closure trip level setting is sensed by measuring disc dump electro-hydraulic oil line pressure (Relay Emergency Trip Supply) which decreases rapidly upon generator load rejection, and pressure switch response is detected prior to fast closure of the control valves. This scram is only effective when turbine steam flow is above 30% of rated as measured by turbine first stage pressure (approximately 206 psia)".

The licensee stated that the present pressure trip setting is not directly enough associated with essential criteria that reactor scram occurs within the required time. The important point is that reactor scram occurs within 30 milliseconds of control valve fast closure; not at some certain pressure.

The licensee states as a further justification for this change that the monthly surveillance requirement (Technical Specification 4.1.A.1) is not possible for Turbine Control Valves which are not open as would occur if the reactor is not at sufficiently high power level that all four Turbine Control Valves are open. The proposed changes provide a means of demonstrating correct operation of the subject trips.

We find that the Licensee's proposed change is acceptable on the basis that control is specified on the critical safety parameter (time interval following

start of closure), the proposed time interval (30 milliseconds) corresponds to the staff's requirement for this trip since it is also the time interval for Reactor Protection System trip assumed in the applicable transient calculation presented in the Final Safety Analysis Report, the proposed monthly testing program will provide assurance that the portion of the Reactor Protection System circuitry is functioning properly, and testing of the valve performance characteristics related to oil pressure during each refueling outage is considered an adequate frequency since this trip is redundant to other trip functions which provide adequate over-pressure protection to abnormal occurrences. Therefore, the proposed change does not involve a significant decrease in safety margin and does not involve a significant increase in either the probability or consequences of an accident.

B. Suppression Chamber High Water Level

The licensee has requested that the following change be made to the Suppression Chamber High Water Level Trip Setting:

p. 3.2-13, Table 3.2-B: Change the Trip Level Setting for Suppression Chamber High Level from "≤2 inches above normal water level" to "≤5 inches above normal water level".

The licensee states that the design criteria for the suppression chamber High Level for DAEC has always been 5". The 5" setting assures adequate margin in the torus free volume relative to that used in the containment pressure response analysis in Section 14 of the FSAR. The setting of ≤2 " above normal water level which is now in the Technical Specifications is actually the fluctuation of water level during normal plant operations; the setting of 2" was inadvertently used in the Technical Specifications when the license was issued. We find that the licensee's proposed change is desirable to provide adequate operating flexibility and is acceptable in that it will not result in the plant operating with an insufficient air volume in the suppression chamber and is consistent with other operating BWR's of similar size such as Vermont Yankee. Therefore, the proposed change does not involve a significant decrease in safety margin and does not involve a significant increase in either the probability or consequences of an accident.

Conclusion

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

Original Signed

Gerald Owsley
Operating Reactors Branch #3
Directorate of Licensing

Original Signed

George Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT & POWER COMPANY, CENTRAL IOWA POWER COOPERATIVE

AND CORN BELT POWER COOPERATIVE

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Atomic Energy Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. DPR-49 issued to Iowa Electric Light & Power Company, Central Iowa Power Cooperative and Corn Belt Power Cooperative which revised Technical Specifications for operation of the Duane Arnold Energy Center, located near Palo in Linn County, Iowa. The amendment is effective as of its date of issuance.

The amendment permits a change to the Technical Specifications pertaining to the Reactor Protection System Trip to specify a time interval following start of Turbine Control Valve Fast Closure rather than an oil pressure setting and to the Suppression Chamber Water level at which the High Pressure Core Injection Pump Suction transfers to the Suppression Chamber.

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.

For further details with respect to this action, see (1) the application for amendment dated June 21, 1974, (2) Amendment No. 4 to License No. DPR-49, with any attachments, 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 Reference Service, 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. Atomic Energy Commission, Washington, D. C. 20545. Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 5th day of November, 1974.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed

George Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing