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April 1, 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. 86 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 December 13, 1982.

This change to the Technical Specifications establishes revised vessel level set points that are consistent with a new common instrument zero level.

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

Sincerely,

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

Enclosures:
1. Amendment No. 86 to DPR-49
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

Leg Rev. of Amdt + FRN only

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OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#4	DL:ORB#2	DL:OR	OELD	
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DATE	3/17/83	3/17/83	3/18/83	3/18/83	3/18/83	3/24/83	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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. 86
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 December 13, 1982 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. 86, 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 1, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by inserting revised pages listed below. The revised area is identified by the vertical line.

Revised Pages

1.1-3
1.1-4
3.1-4
3.1-7
3.2-5
3.2-6
3.2-8
3.2-9
3.2-15
3.2-23
3.2-36
3.2-37

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	3. APRM Rod Block when in Run Mode.

For operation with MFLPD less than or equal to FRP the APRM Control Rod Block setpoint shall be as shown on Figure 2.1-1 and shall be:

$S \leq (0.66\ W + 42)$

The definitions used above for the APRM scram trip apply.

For a MFLPD greater than FRP, the APRM Control Rod Block setpoint shall be:

$$S \leq (0.66\ W + 42)\ \frac{FRP}{MFLPD}$$

4. IRM - The IRM scram shall be set at less than or equal to 120/125 of full scale.

B.	Scram and Isolation on reactor low water level	$\geq$ 514.5 inches above vessel zero (+170" indicated level)
C.	Scram - turbine stop valve closure	$\leq$ 10 percent valve closure

D. Turbine control valve fast closure shall occur within 30 milliseconds of the start of turbine control valve fast closure.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT		
	E.	Scram - main steam line isolation valve	≤ 10 percent valve closure
	F.	Main steam isolation valve closure nuclear system low pressure	≥ 880 psig
	G.	Core spray & LPCI actuation - reactor low level	≥ 363 inches above vessel zero (+18.5 inches water indicated level)
	H.	HPCI & RCIC actuation - reactor low water level	≥ 464 inches above vessel zero (+119.5 inches indicated level)
	I.	Main steam isolation valve closure-reactor low water level	≥ 464 inches above vessel zero (+119.5 inches indicated level.)
	J.	Main steam isolation valve closure-loss of main condenser vacuum	≤ 10 inches Hg vacuum

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	≥ +170" Indicated Level (15)	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*	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	First Stage	Bypass below 192 psig	X	X	X	4 Instrument Channels	A or D

*Alarm setting ≤1.5 X Normal Rated Power Background

7. Not required to be operable when primary containment integrity is not required.

8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

9. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

10. To be considered operable, APRM's A, B, C and D must have at least 9 LPRM inputs while APRM's E and F must have at least 13 LRPM inputs. Additionally each APRM must have at least 2 LPRM inputs per level.

11. W is the recirculation loop flow in percent of rated.

12. See Subsection 2.1.A.1.

13. The design permits closure of any two lines without a scram being initiated.

14. Deleted.

15. Zero referenced to top of active fuel.

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Valve Groups Operated by Signal	Action (2)
2 (6)	Reactor Low Water Level	> +170" Indicated Level (3)	4	2,3,4,5 (Sec. Cont., 3	A E)
1	Reactor Low Pressure (Shutdown Cooling Isolation)	≤ 135 psig	2	4	C
2	Reactor Low-Low-Water Level	At or above +119.5" indicated level (3)	4	1, 8	A
2 (6)	High Drywell Pressure	≤ 2.0 psig	4	2,3,4,8,9* (Sec. Cont., 3	A E)
2	High Radiation Main Steam Line Tunnel	≤ 3 X Normal Rated Power Background	4	1	B
2	Low Pressure Main Steam Line	≥ 880 psig (7)	4	1	B
2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4	1	B
2	Main Steam Line Tunnel/Turbine Bldg. High Temperature	≤ 200° F.	4	1	B
1	Reactor Cleanup System High Diff. Flow	≤ 40 gpmd	2	5	D

*Group 9 valves isolate on high drywell pressure combined with reactor steam supply low pressure

NOTES FOR TABLE 3.2-A

1. Whenever Primary Containment integrity is required by Subsection 3.7, there shall be two operable or tripped systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

ACTION A - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION B - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION C - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION D - Be in at least STARTUP within 6 hours.

ACTION E - Isolate secondary containment and start the standby gas treatment system.

3. Zero referenced to top of active fuel.*

* Top of the active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

TABLE 3.2-B

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
2	Reactor Low-Low Water Level	≥ + 119.5 in. indicated level (4)	4 HPCI & RCIC Instrument Channels	Initiates HPCI & RCIC
2	Reactor Low-Low-Low Water Level	≥ + 18.5 in. indicated level (4)	4 Core Spray & RHR Instrument Channels 4 ADS Instrument Channels	1. In conjunction with Low Reactor Pressure initiates Core Spray and LPCI 2. In conjunction with confirmatory low level High Drywell Pressure, 120 second time delay and LPCI or Core Spray pump interlock initiates Auto Blowdown (ADS) 3. Initiates starting of Diesel Generator
2	Reactor High Water Level	≤ + 211 in. indicated level (4)	2 Instrument Channels	Trips HPCI and RCIC turbines

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	Reactor Low Level (inside shroud)	$\geq$ + 305.5 in. above vessel zero (2/3 core height)	2 Instrument Channels	Prevents inadvertent operation of contain-ment spray during accident condition
2	Containment High Pressure	$1 < p < 2$ psig	4 Instrument Channels	Prevents inadvertent operation of contain-ment spray during accident condition
1	Confirmatory Low Level	$\leq$ + 170 in. indicated level (4)	2 Instrument Channels	ADS Permissive
2	High Drywell Pressure	$\leq$ 2.0 psig	4 HPCI Instrument Channels	1. Initiates Core Spray LPCI; HPCI
2	Reactor Low Pressure	$\geq$ 450 psig	4 Instrument Channels	Permissive for open Core Spray and LCPI Injection valves. Coincident with high drywell pressure, start LPCI and Core Spray pumps

NOTES FOR TABLE 3.2-B

1. Whenever any CSCS subsystem is required by Subsection 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Zero referenced to top of active fuel.*

5. HPCI has only one trip system for these sensors.

6. The relay drop-out voltage will be measured once per operating cycle and the data examined for evidence of relay deterioration.

7. Four undervoltage relays with integral timers per 4KV bus. The relay output contacts are connected to form a one-out-of-two-twice coincident logic matrix. With one relay inoperable, operation may proceed provided that the inoperable relay is placed in the tripped condition within one hour.

*Top of active fuel zone is defined to be 344.5 inches above vessel zero (see Bases 3.2).

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (1)
1	(ATWS) Reactor High Pressure	≤ 1120 psig	4	(2)
1	(ATWS) Reactor Low-Low Water Level	≥ +119.5 in indicated level (5)	4	(2)
1	(EOC) RPT Logic	N/A	2	(3)
1	(EOC) RPT System (Response Time)	≤ *msec (4)	2	(3)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

3. Two EOC RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 consecutive hours, an orderly power reduction shall be initiated and the reactor power shall be less than 85% within four hours.

4. This response time is from initiation of turbine control valve fast closure to actuation of the breaker auxiliary contact.

5. Zero referenced to top of active fuel.

* To be determined by testing after installation. (Valve to be design requirement for breaker opening less difference between cycle time for loaded vs. unloaded breaker.)

** Top of active fuel zone is defined to be 344.5" above vessel zero (see Bases 3.2).

explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2-A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

Many of the reactor water level trip settings are defined or described in terms of "inches above the top of the active fuel." In the new reload fuel the column of fuel pellets in each fuel pin of a bundle is 150 inches long; whereas in the initial core load and first few reloads it was 144 inches long. Thus, during the period of reloads until all of the 144 inch bundles are replaced with bundles with 150 inches of fuel pellets the core will be composed of fuel bundles with fuel pins containing differing lengths of fuel pellet columns and the term "top of active fuel" no longer has a precise physical meaning. Since the basis of all safety analyses is the absolute level (inches above vessel zero) of the trip settings, the "top of the active fuel" has been arbitrarily defined to be 344.5 inches above vessel zero. This definition is the same as that given by the FSAR for the initial core and maintains the consistency between the various level definitions given in the FSAR and the technical specifications.

adequate to prevent uncovering the core in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 119.5" above the top of the active fuel. This trip closes Main Steam Line Isolation Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1), initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18.5" above the top of the active fuel. This trip activates the remainder of the CSCS subsystems, closes Group 7 valves, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 22-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Paragraph 6.5.4 FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 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 December 13, 1982 Iowa Electric Light and Power Company (the licensee) forwarded a proposed Technical Specification change that establishes revised vessel level setpoints that are consistent with a new common instrument zero level. The proposed common reference level is 344.5" above vessel zero. Establishment of the common zero level for all reactor vessel level instrumentation is addressed in TMI Action Item II.K.3.27 in NUREG-0737.

2.0 Evaluation

Different reference points of the various reactor vessel water level instrument may cause operator confusion. Therefore, all level instruments should be referenced to the same point. The licensee defines this common reference level as the absolute level (inches above vessel zero) of the trip settings; the "top of active fuel" has been defined to be 344.5 inches above vessel zero.

We have reviewed each of the proposed revised setpoints and find them to be consistent with the previously established safety settings. However, we require and the licensee has committed by its letter dated February 3, 1983, that all operators will be trained on the new level setpoints by the completion of the refueling outage commencing February 1983. The required changes to procedures and Technical Specifications will be entered prior to operating with the new setpoints installed.

Since no changes in actual water level for any function is involved in the proposed Technical Specification revisions, and no instrumentation is being changed, and since the proposed Technical Specification revisions conform to the guidelines of Item II.K.3.27 of NUREG-0737 we find the revisions 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 Section 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 safety margin, the amendment does not involve a significant hazards consideration (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 1, 1983

Principal Contributor: Frank L. Apicella

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. 86 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 change to the Technical Specifications establishes revised vessel level setpoints that are consistent with a new common instrument zero level.

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 December 13, 1982 (2) Amendment No.86 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 1st day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

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