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Docket No. 50-331

OCT 8 1975

Iowa Electric Light and 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. 14 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The amendment includes Change No. 15 to the Technical Specifications in accordance with your application dated September 4, 1975, and supplements dated September 9 and 15, 1975.

The amendment modifies the Technical Specifications to delete the trip level settings for the bus power monitor relays for the following: residual heat removal (RHR) system-low pressure coolant injection mode; core spray system; automatic depressurization system; high pressure coolant injection system; and the reactor core isolation cooling trip system. The amendment will also add new trip functions that will indicate undervoltage conditions on the instrument a.c. and battery busses and will add a surveillance requirement to verify that there is power on the RHR valve bus.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 14
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OFFICE ➤	ORB #3	ORB #3	AD	OELD	ORB #3	
SURNAME ➤	SATeets	WPaulson	Stello		GLear	
DATE ➤	9/24/75	9/24/75	9/29/75	10/6/75	10/8/75	

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

OCT 8 1975

Iowa Electric Light and Power Company

cc:

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

Cedar Rapids Public Library
426 Third Avenue, S. E.
Cedar Rapids, Iowa 52401

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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. 14
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, Central Iowa Power Cooperative, and the Corn Belt Power Cooperative (the licensees) dated September 4, 1975, and supplements dated September 9 and 15, 1975, comply 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; and

   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.

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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 licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 15."

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Change No. 15 to the
Technical Specifications

Date of Issuance: OCT 8 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 14

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace pages 3.2-11, 3.2-12, 3.2-15, 3.2-16, 3.2-25, 3.2-26, 3.2-45, 3.2-46, 3.5-3, 3.5-4, 3.5-31 and 3.5-32 with the attached revised pages. (No change made on pages 3.2-16, 3.2-25, 3.2-46, 3.5-3 and 3.5-31.)

Add pages 3.2-14a and 3.2-45a.

TABLE 3.2-B (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channel Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	Auto Blowdown Timer	120 sec ± 5 sec	2 timers	In conjunction with Low Reactor Water Level, High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown
2	RHR (LPCI) Pump Discharge Pressure Interlock	≥100 psig	4 channels	Defers ADS actuation pending confirmation of Low Pressure core cooling system operation (LPCI or Core Spray Pump running interlock)
2	Core Spray Pump Discharge Pressure Interlock	100 ± 5 psig	4 channels	"
1.	RHR (LPCI) Trip System bus power monitor	Not applicable (6)	2 Inst. Channels	Relay which continuously monitors availability of power to logic systems and annunciates upon loss of power
1	Core Spray Trip System bus power monitor	Not applicable (6)		"

15

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	ADS Trip System bus power monitor	Not applicable (6)	2 Inst. Channels	Relay which continuously monitors availability of power to logic systems and annunciates upon loss of power	15
1	HPCI Trip System bus power monitor	"	2 Inst. Channels		15
1	RCIC Trip System bus power monitor	"	2 Inst. Channels		15
2	Recirculation Pump A d/p	$\leq$ 2 psid	4 Inst. Channels	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop	
2	Recirculation Pump B d/p	$\leq$ 2 psid	4 Inst. Channels		
2	Recirculation Riser d/p A > B	$0.5 < p < 1.5$ psid	4 Inst. Channels		
1	Core Spray Sparger to Reactor Pressure Vessel d/p	0.74 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break	
2	Condensate Storage Tank Low Level	$\geq$12" above tank bottom (10,000 gallons)	2 Inst. Channels	Provides interlock to HPCI pump suction valves	

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
2	125 volt undervoltage	105 volts ± 5% (6)	2	2 relays, 1 per bus
1	250 volt D.C. under-voltage relay	210 volts ± 5% (6)	1	1 relay, 1 per bus
4	± 24 volt under-voltage relay	21 volts ± 5% (6)	4	4 relays, 2 per bus
1	240 volt uninterrup-tible A.C. under-voltage relay	220 volts ± 5% (6)	1	1 relay, 1 per bus
2	240 volt instrument A.C. undervoltage relay	220 volts ± 5% (6)	2	2 relays, 1 per bus

15

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. Instrument setpoint corresponds to 18.5" above the 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 15
cycle and the data examined for evidence of relay deterioration.

TABLE 3.2-C

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action
2	APRM Upscale (Flow Biased)	$\leq (0.66W + 42) \left(\frac{2.62}{P.F}\right)$ (2)	6 Inst. Channels	(1)
2	APRM Upscale (Not in Run Mode)	$\leq$ 12 indicated on scale	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 5 indicated on scale	6 Inst. Channels	(1)
1 (7)	Rod Block Monitor (Flow Biased)	$\leq (0.66W + 41) \left(\frac{2.62}{TPF}\right)$ (2)	2 Inst. Channels	(1)
1 (7)	Rod Block Monitor Downscale	$\geq$ 5 indicated on scale	2 Inst. Channels	(1)
2	IRM Downscale (3)	$\geq$ 5/125 full scale	6 Inst. Channels	(1)
2	IRM Detector not in Startup Position	(8)	6 Inst. Channels	(1)
2	IRM Upscale	$\leq$ 108/125	6 Inst. Channels	(1)
2 (5)	SRM Detector not in Startup Position	(4)	4 Inst. Channels	(1)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.	4 Inst. Channels	(1)

TABLE 4.2-A (Continued)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

<u>Logic System Functional Test</u> (4) (6)	<u>Calibration Frequency</u>
4) Drywell Isolation Valves TIP Withdrawal Atmospheric Control Valves Sump Drain Valves	Once/6 months
5) Standby Gas Treatment System Reactor Building Isolation	Once/6 months

TABLE 4.2-B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check	
1) Reactor Water Level	(1)	Once/3 months	Once/day	
2) Drywell Pressure	(1)	Once/3 months	None	
3) Reactor Pressure	(1)	Once/3 months	None	
4) Auto Sequencing Timers	N/A	Once/operating Cycle	None	
5) ADS - LPCI or CS Pump Discharge Pressure Interlock	(1)	Once/3 months	None	
6) Trip System Bus Power Monitors	(1)	Not applicable	None	15
7) Recirculation System d/p	(1)	Once/3 months	Once/day	
8) Core Spray Sparger d/p	(1)	Once/3 months	Once/day	
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None	
10) Steam Line High Temp. (HPCI & RCIC)	(1)	Once/operating cycle	Once/day	
11) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None	
12) HPCI Suction Source Levels	(1)	Once/3 months	None	
13) 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 years	None	
14) Instrument A.C. and battery bus undervoltage relays	(1)	Once per operating cycle	None	15

timer is set to annunciate before the values specified in Specification 3.6.C are exceeded. An air sampling system is also provided to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report, NEDO-10349, dated March, 1971.

Trip function settings are included for instrument a.c. and battery busses
for surveillance of undervoltage relays. The undervoltage relays are required to 15
sense a reduction in the power source voltage so that the subject instruments can
be transferred to an alternate power source.

Surveillance tests other than a monthly functional check of the bus power monitors for the RHR, Core Spray, ADS, HPCI and RCIC trip systems are not required since they serve as annunciators for complete loss of power and do 15
not monitor reduction of voltage. The subject functional check consists of opening the appropriate circuit breakers and observing the loss of power annunciator activation.

4.2 BASES

The instrumentation listed in Table 4.2-A through 4.2-F will be functionally tested and calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$I = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	Item — **Frequency**
	c. Motor Operated valve operability — Once/month
	d. Pump Flow Rate — Once/3 months Three LPCI pumps shall deliver 14,400 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests.
4. From and after the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days provided that during such thirty days the remaining active components of the LPCI subsystem, the containment cooling subsystem, and all active components of both core spray subsystems and the diesel-generators are operable.	4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operable, the remaining active components of the LPCI subsystem, the containment spray subsystem, both core spray subsystems and the diesel-generators shall be demonstrated to be operable immediately and the operable LPCI pump daily thereafter.
5. From and after the date that two RHR pumps (LPCI mode) are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless at least one of the inoperable pumps is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment spray subsystem and the diesel-generators required for operation of such components are operable.	5. When it is determined that the LPCI subsystem is inoperable, both core spray subsystems, the containment spray subsystem and the diesel-generators required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
6. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.	6. Once per shift visually inspect and verify that RHR valve panel lights and instrumentation are functioning normally. 15
B. <u>Containment Spray Cooling Capability</u>	B. <u>Containment Spray Cooling Capability</u> Surveillance of the drywell spray loops shall be performed as follows:
1. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.	1. During each five year period, an air test shall be performed on the drywell and suppression pool spray headers and nozzles.
2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.	
C. <u>Residual Heat Removal (RHR) Service Water System</u>	C. <u>Surveillance of the RHR Service Water System</u>
1. Except as specified in 3.5.C.2, 3.5.C.3, and 3.5.G.3 below, both RHR service water subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.	1. Surveillance of the RHR service water system shall be as follows: RHR Service Water Subsystem Testing: <u>Item</u> — <u>Frequency</u> a) Pump and motor operated valve operability. — Once/3 months

4.5 BASES

Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with frequent tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure due to a design deficiency, caused the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

The RHR valve power bus is not instrumented. For this reason surveillance
requirements require once per shift observation and verification of lights and 15
instrumentation operability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 14 TO FACILITY OPERATING

LICENSE NO. DPR-49

CHANGE NO. 15 TO THE 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 telecopy request dated September 4, 1975 and modifications dated September 9, and September 15, 1975, Iowa Electric Light and Power Company (IELP) requested authorization to change the Technical Specifications appended to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The proposed change involves deleting the trip level settings for the bus power monitor relays in the following: residual heat removal (RHR) system-low pressure coolant injection mode (LPCI); core spray system; automatic depressurization system (ADS); high pressure coolant injection (HPCI) system; and the reactor core isolation cooling (RCIC) trip system. The proposed change will add new trip functions that will indicate undervoltage conditions on the instrument a.c. and battery busses. The proposed change will also add a new surveillance requirement to visually inspect and verify that the RHR valve panel lights and instrumentation are functioning properly. In addition, the proposed change will correct the number of instrument channels for the ADS trip system bus power monitor from 3 to 2.

Discussion

The bus power relay is designed to trip on complete loss of power to the RHR-LPCI, ADS, HPCI, and RCIC trip logic systems. The current Technical Specifications incorrectly indicate that the trip setting for these relays is 80% $\pm$ 10% of rated voltage. IELP has proposed deleting the trip setting for these loss of power relays. However, these relays will be functionally tested monthly by opening the appropriate circuit breakers and observing the annunciators to verify that the relays are operable. (The bus power relay trips are annunciated.)

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To assure that the voltage levels on the instrument a.c. and battery busses are adequate to actuate the logic relays, IELP has proposed to add new surveillance requirements for existing undervoltage relays. The new surveillance requirements will establish a trip level setting of 80% ± 10% of the rated voltages on these busses. This voltage level is adequate to actuate the logic relays.

In addition, IELP has proposed to add surveillance requirements for the RHR panel valve lights and instrumentation. Because the RHR valve power bus is not instrumented, there is no annunciation on loss of power to the bus. Hence, the RHR panel valve lights and instrumentation will be visually inspected once per shift and it will be verified that the lights and instrumentation are functioning properly.

Based on our review of the proposed changes, we conclude that the modified Technical Specifications will assure that loss of power or undervoltage conditions in the electrical circuits or logic systems discussed above will be detected in a timely manner.

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) 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: OCT 8 1975

U. S. NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-49 issued to Iowa Electric Light And Power Company, Central Iowa Power Cooperative, and the Corn Belt Power Cooperative which revised 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.

The amendment modifies the Technical Specifications to delete the trip level settings for the bus power monitor relays for the following: residual heat removal (RHR) system-low pressure coolant injection mode; core spray system; automatic depressurization system; high pressure coolant injection system; and the reactor core isolation cooling trip system.

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 is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated September 4, 1975, and supplements dated September 9 and 15, 1975, (2) Amendment No. 14 to License No. DPR-49, with Change No. 15, 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.

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 Reactor Licensing.

Dated at Bethesda, Maryland, this 8th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing