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

November 17, 1975

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. 13 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The amendment includes Change No. 14 in the Technical Specifications in accordance with your application dated February 21, 1975. By letter dated May 28, 1975 you withdrew four of the proposed changes and provided further clarification of your original request.

This amendment will (1) delete the calibration of the Average Power Range Monitor (APRM) when the reactor is in the startup mode; (2) delete the functional test of certain IRM and APRM trip channels before each startup if a successful functional test has been accomplished during the preceeding seven days; (3) delete the calibration of the turbine control valve fast closure position trip which was not included in the DAEC design and is not required; (4) add specifications that were inadvertently omitted from the DAEC Technical Specifications; (5) provide clarification to several specifications; and (6) correct typographical errors. During our review of the proposed change, we found that certain modifications to the application were necessary. These latter modifications were found mutually acceptable to you and the NRC staff and have been incorporated into the Technical Specifications.

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

Sincerely,

for Walter A. Paulson

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

Enclosure and cc:
See next page

cc: w/enclosure

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

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

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

Mr. Ed Vest
Environmental Protection Agency
Region VII Office
1735 Baltimore Avenue
Kansas City, Missouri 64108

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

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. 13
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 Corn Belt Power Cooperative (the licensees) dated February 21, 1975 and supplement submitted May 28, 1975, 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; 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.

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. 14."

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. 14 to
Technical Specifications

Date of Issuance: November 17, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 13

CHANGE NO. 14 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace page 3.1-4, page 3.1-6, page 3.1-8, page 3.1-9, page 3.1-13, page 3.1-21, page 3.2-5.a, page 3.2-24, page 3.2-28, page 3.2-31, pages 3.5-6 thru 3.5-8, page 3.7-23, page 3.7-34, page 3.8-2, and page 6.9-8, with the attached revised pages. No change has been made on page 3.1-5, page 3.1-7, page 3.1-10, page 3.2-5, page 3.2-23, page 3.2-27, page 3.2-32, page 3.5-5, page 3.7-24, page 3.7-33, page 3.8-1, and page 6.9-7.

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			Number of Instrument Channels Provided by Design	Action (1)
			Refuel (6)	Startup	Run		
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 First Stage Pressure Permissive	Bypass below 192 psig at shell	X	X	X	4 Instrument Channels	A or D

*Alarm setting ≤ 1.5 X Normal Rated Power Background

NOTES FOR TABLE 3.1-1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable sensor or instrument channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   a. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   b. Reduce power level to IRM range and place mode switch in the startup position within 8 hours.

   c. Reduce turbine load and close main steam line isolation valves within 8 hours.

   d. Reduce power to less than 30% of rated.

2. Permissible to bypass, in refuel and shutdown positions of the reactor mode switch.

3. A main steam line isolation valve closure trip bypass is effective when the reactor mode switch is in the shutdown, refuel or startup positions and reactor pressure is below 1035 psig.

4. Bypassed when turbine first stage pressure is less than 14
192 psig or less than 30% of rated.

5. IRM's are bypassed when APRM's are on-scale and the reactor mode switch is in the run position.

6. When the reactor is subcritical and the reactor water temperature is less than $212^{\circ}F$, only the following trip functions need to be operable:

   a. Mode switch in shutdown

   b. Manual scram

   c. High flux IRM

   d. Scram discharge volume high level - may be bypassed in the refuel and shutdown modes for the purpose of resetting the scram.

   e. APRM 15% flux

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 LPRM 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. The trip setting and alarm setting for the Main Steam Line High Radiation Monitor shall be $\leq$ 6 X and $\leq$ 3 X, respectively, Normal Rated Power Background during the period prior to achieving 50 per cent rated power for the first time.

TABLE 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shut down	Each refueling outage.
Manual Scram	A	Trip Channel and Alarm	Every 3 months
RPS Channel Test Switch	A	Trip Channel and Alarm	Every refueling outage or after channel maintenance
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup unless a satisfactory test has been accomplished during the preceeding 7 days.
Inoperative	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup unless a satisfactory test has been accomplished during the preceeding 7 days.
APRM			
High Flux in Run	B	Trip Output Relays (4)	Once/week (While in Run Mode)
Inoperative	B	Trip Output Relays (4)	Once/week
Downscale *	B	Trip Output Relays (4)	Once/month (1)
Flow Bias	B	Trip Output Relays (4)	Once/month (1)
High Flux in Startup or Refuel	C	Trip Output Relays	Once per week during refueling or startup and before each startup unless a satisfactory test has been accomplished during the preceeding 7 days.
High Reactor Pressure	A	Trip Channel Alarm	Every 1 month (1)

*With companion IRM Hi-Hi or Inoperable.

TABLE 4.1-1 (Continued)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency (3)
High Drywell Pressure	A	Trip Channel and Alarm	Every 1 month (1)
Reactor Low Water Level (5)	A	Trip Channel and Alarm	Every 1 month (1)
High Water Level in Scram Discharge Volume	A	Trip Channel and Alarm	Every 3 months
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/week
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
Turbine Control Valve EHC Oil Pressure	A	Trip Channel and Alarm	Every 1 month
Turbine First Stage Pressure Permissive	A	Trip Channel and Alarm	Every 3 months (1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Every 1 month (1)
Reactor Pressure Permissive	A	Trip Channel and Alarm	Every 3 months

14

NOTES FOR TABLE 4.1-1

1. Initially once every month, the compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of DAEC. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

2. A description of the three groups is included in the Bases of this Specification.

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped.

If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

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

14

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

Turbine stop valve closure trip occurs at approximately 10% of valve closure. Below approximately 192 psig turbine first stage 14
pressure (30% of rated power), the scram signal due to turbine stop valve closure is bypassed 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).

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.

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	Action(2)
2 (6)	Reactor Low Water Level	$\geq$ +12" Indicated Level (3)	4 Inst. Channels	A
	Reactor Low Pressure (Shutdown Cooling Isolation)	$\leq$ 135 psig	2 Inst. Channels	C
2	Reactor Low-Low-Water Level	At or above -38.5" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	$\leq$ 2.0 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	$\leq$ 3 X Normal Rated Power Background (8)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	$\geq$ 880 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	$\leq$ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel/Turbine Bldg. High Temperature	$\leq$ 200 deg. F	4 Inst. Channels	B
1	Reactor Cleanup System High Diff. Flow	$\leq$ 40 gpmd	2 Inst. Channel	D
1	Reactor Cleanup System High-High Temperature	$\leq$ 140°F	1 Inst. Channel	D

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION (Continued)

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action (2)	
1	Reactor Cleanup Area Ambient High Temp.	130°F	3 Inst. Channels	D	
1	Reactor Cleanup Area Differential High Temp.	Δ14°F	3 Inst. Channels	D	
2	Loss of Main Condenser Vacuum	$\leq$ 10 in. Hg vacuum	4 Inst. Channels	B	14

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	The HPCI pump shall deliver at least 3000 gpm for a system head corresponding to a reactor pressure of 1020 to 150 psig. 14

Additional page submitted on 1-15-76

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT		
		Item	**Frequency**
		than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1020 psig.	
	e.	At reactor pressure of 150±10 psig demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection.	Once/operating cycle
		The RCIC pump shall deliver at least 400 gpm for a system head corresponding to 1020 to 150 psig.	
2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.	2.	When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.	
3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 150 psig within 24 hours.			

Corrected page submitted on 1-15-76

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	Reactor High Pressure	$\leq$ 1135 psig	4	(2)
1	Reactor Low Water Level	$\geq$ -38.5 in. indicated level	4	(2)

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 each 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.

TABLE 4.2-A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

Instrument Channel (5)	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Low Pressure (Shutdown Cooling Permissive)	(1)	Once/3 months	None
2) Reactor Low-Low Water Level	(1)	Once/3 months	Once/day
3) Main Steam High Temp.	(1)	Once/operating cycle	Once/day
4) Main Steam High Flow	(1)	Once/3 months	None
5) Main Steam Low Pressure	(1)	Once/3 months	None
6) Reactor Water Cleanup High Flow (7)	(1)	Once/3 months	Once/day
7) Reactor Water Cleanup High Temp. (7)	(1)	Once/3 months	None
8) Reactor Cleanup Area High Temp. (8)	(1)	Once/Operating cycle	None
9) Loss of Main Condenser Vacuum	(1)	Once/Operating Cycle	None 14
Logic System Functional Test (4) (6)			
1) Main Steam Line Isolation Valves Main Steam Line Drain Valves Reactor Water Sample Valves		Once/6 months	
2) RHR - Isolation Valve Control Shutdown Cooling Valves Head Spray		Once/6 months	
3) Reactor Water Cleanup Isolation		Once/6 months	

TABLE 4.2-B (Continued)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Logic System Functional Test (4) (6)	Calibration Frequency
1) Core Spray Subsystem	Once/6 months
2) Low Pressure Coolant Injection Subsystem	Once/6 months
3) Containment Spray Subsystem	Once/6 months
4) HPCI Subsystem	Once/6 months
5) HPCI Subsystem Auto Isolation	Once/6 months
6) ADS Subsystem	Once/6 months
7) RCIC Subsystem Auto Isolation	Once/6 months
8) Area Cooling for Safeguard System	Once/6 months

TABLE 4.2-C

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
1) APRM - Downscale	(1) (3)	Once/3 months	Once/day
2) APRM - Upscale	(1) (3)	Once/3 months	Once/day
3) IRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
4) IRM - Downscale	(2) (3)	Startup or Control Shutdown	(2)
5) RBM - Upscale	(1) (3)	Once/6 months	Once/day
6) RBM - Downscale	(1) (3)	Once/6 months	Once/day
7) SRM - Upscale	(2) (3)	Startup or Control Shutdown	(2)
8) SRM - Detector Not in Startup Position	(2)	Refuel	N/A
9) IRM - Detector Not in Startup Position	(2)	Refuel	N/A

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) 14

NOTES FOR TABLES 4.2-A THROUGH 4.2-F

1. Initially once every month. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of DAEC. The failure rate data must be reviewed and approved by the AEC prior to any change in the once-a-month frequency.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed prior to each startup or prior to controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	Item / **Frequency** b) Flow Rate Test-Each RHR service water pump shall deliver at least 2400 gpm at a TDH of 610 ft or more. / after major pump maintenance and every 3 months
2. From and after the date that one of the RHR Service Water subsystem pumps is made or found to be inoperable for any reason, reactor operation must be limited to thirty days unless operability of that pump is restored within this period. During such thirty days all other active components of the RHR Service Water subsystem are operable.	2. When it is determined that one RHR Service Water pump is inoperable, the remaining components of that subsystem and the other subsystems shall be demonstrated to be operable immediately and daily thereafter.
3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is limited to seven days unless operability of that subsystem is restored within this period. During such seven days all active components of the other RHR Service Water Subsystem and its associated diesel-generators required for operation of such components (if no external source of power were available), shall be operable.	3. When on RHR Service Water subsystem becomes inoperable, the operable 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 REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

D. HPCI Subsystem

1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 150 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 and 3.5.D.3 below.

SURVEILLANCE REQUIREMENTS

D. HPCI Subsystem

1. HPCI Subsystem testing shall be performed as follows:

	Item	Frequency
a)	Simulated Automatic Actuation Test	Once/operating cycle
b)	Pump Operability	Once/month
c)	Motor Operated valve Operability	Once/month
d)	At rated reactor pressure demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1020 psig.	Once/3 months
e)	At reactor pressure of 150$\pm$10 psig demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection.	Once/operating cycle

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.	2. When it is determined that the HPCI Subsystem is inoperable, the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be demonstrated to be operable immediately. the RCIC system and ADS subsystem logic shall be demonstrated to be operable daily thereafter.
3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	
E. <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>	E. <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>
1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 150 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.E.2 below.	1. RCIC Subsystem testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	At rated reactor pressure demonstrate ability to deliver rated flow at a discharge pressure greater	Once/3 months

14

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT		
		Item	Frequency
		than or equal to that pressure required to accomplish vessel injection if vessel pressure were as high as 1020 psig. 14	
	e.	At reactor pressure of 150 ± 10 psig demonstrate ability to deliver rated flow at a discharge pressure greater than or equal to that pressure required to accomplish vessel injection.	Once/operating cycle
2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.	2. When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.		
3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 150 psig within 24 hours.			

TABLE 3.7-2 (Continued)

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PEN #		BOUNDARY VALVES	
25	Drywell Purge Outlet	CV-4302, CV-4303, CV-4310	
26,220	Drywell and Torus Purge Supply	CV-4306, CV-4307, CV-4308	
26,220	Drywell and Torus Nitrogen Makeup	CV-4311, CV-4312, CV-4313	
32D	Containment Compressor Suction	CV-4378A, CV-4378B	
32E	Recirc Pump "A" Seal Purge	V-17-96, V-17-84	14
32E	Recirc Pump "A" Seal Purge	CV-1804B, V-17-84	
32F	Recirc Pump "B" Seal Purge	V-17-83, V-17-80	
32F	Recirc Pump "B" Seal Purge	CV-1804A, V-17-80	
36[1]	CRD Return	V-17-54, V-17-52	
36[1]	CRD Return	V-17-54, V-17-53	
41	Recirc Loop Sample	CV-4639, CV-4640	
46E	O_2 Analyzer	SV-8105B, SV-8106B	
48	Drywell Equipment Drain Discharge	CV-3728, CV-3729	
50B,E,D	O_2 Analyzer	SV-8101A, SV-8102A, SV-8103A, SV-8104A, SV-8105A, SV-8106A	
54	Reactor Bldg Closed Cooling Water Return	MO-4841A, V-12-64, V-12-65, V-12-68	
55	Reactor Bldg Closed Cooling Water Supply	MO-4841B, V-12-62, V-12-63, V-12-66	
56C,D	O_2 Analyzer	SV-8101B, SV-8102B, SV-8103B, SV-8104B	
205	Torus Purge Outlet	CV-4300, CV-4301, CV-4309	
229B,C, G,F	O_2 Analyzer	SV-8107A, SV-8108A, SV-8109A, SV-8110A, SV-8107B, SV-8108B, SV-8109B, SV-8110B	
231	Torus Vacuum Breakers	CV-4304, ZS-4329	
231	Torus Vacuum Breakers	CV-4305, ZS-4330	
219[3]	HPCI/RCIC Exhaust Vacuum Breaker	MO-2290A, MO-2290B, V-22-60	

NOTES TO TABLE 3.7-2

[1]Test volume is filled with demineralized water then pressurized to 54 psig with air or nitrogen for test.

For all other penetrations (except Main Steam Lines) test volumes are pressurized to 54 psig with air or nitrogen for test.

[2]MO-4441, MO-4442 will be remote manually closed.

[3]Subject isolation valves to be installed at earliest practicable date per FSAR P. 6.4-10.C, Dated 9/73.

Guide No. 7 flammability limit. By keeping oxygen concentrations less than 5% (AEC has recommended 4%), Safety Guide No. 7 requirements are satisfied. The Containment Atmosphere Dilution System further assures that a combustible hydrogen/ oxygen atmosphere will not be created in a post-LOCA condition.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

3. Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psi, the external design pressure. One valve may be inoperable, either non-fully closed or inoperable for opening subject to the requirements as stated in Specifications 14
3.7.A.4.b and c and 4.7.A.4.b and c. If these specifications cannot be met, the reactor coolant system is brought to a condition where vacuum relief is not required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8 AUXILIARY ELECTRICAL SYSTEM	4.8 AUXILIARY ELECTRICAL SYSTEM
Applicability:	Applicability:
Applies to the auxiliary electrical power system.	Applies to the periodic testing requirements of the auxiliary electrical systems.
Objective:	Objective:
To assure an adequate supply of electrical power for operation of those systems required for safety.	Verify the operability of the auxiliary electrical system.
Specification:	Specification:
A. Auxiliary Electrical Equipment	A. Auxiliary Electrical Equipment
The reactor shall not be made critical unless all of the following conditions are satisfied:	
1. Both off-site sources and the startup transformers and standby transformers are available and capable of automatically supplying power to the 4kV emergency buses.	1. Diesel-Generators
2. The two diesel-generators shall be operable and there shall be a minimum of 35,000 gallons of diesel fuel in the diesel fuel oil tank.	a. Each diesel-generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at rated load.
3. All station 24,125 and 250 volt battery systems shall be operable. The associated battery chargers for the 24 volt batteries, two of the three battery	During the monthly generator test the diesel-generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
chargers for the 125 volt station batteries, and one of the two 250 volt battery chargers shall be operable.	b. Once per operating cycle the condition under which the diesel-generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.
4. The emergency 4160 volt 14 buses 1A3 and 1A4 and 480 volt buses 1B3, 1B4, 1B9 and 1B20 shall be energized and operable.	c. The quantity of diesel fuel available shall be logged monthly and after each use of the diesels.
	d. Once a month a sample of diesel fuel shall be checked for viscosity, water and sediment. The values for viscosity, water and sediment shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.
	e. Each diesel-generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.
	f. A sample test and record shall be made of each oil delivery before it is placed in the storage tank.
	2. Unit Batteries
	a. Every week the specific gravity, the voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

3. These specifications with respect to the provisions of 20.103 shall be superseded if, in the future, 10 CFR 20.103 shall assign protection factors for respiratory and other protective equipment.

TABLE 6.9-1

PROTECTION FACTORS FOR RESPIRATORS

Description	Modes 1/	Protection Factors 2/ Particulates and Vapors and Gases Except Tritium Oxide 3/	Guides to Selection of Equipment: Bureau of Mines Approval Schedules* For Equipment Capable of Providing at Least Equivalent Protection Factors *or Schedule Superseding For Equipment of Type Listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b) (4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline Respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR 12.2(c) (2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c) (2) Type C(iii)
Hood	CF	5/	6/
Suit	OF	5/	6/
2. Self-Contained Breathing Apparatus (SCBA)			
Facepiece, full 7/	D	100	13E 30 CFR 11.4(b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b) (2) (ii)
Facepiece, full	R	100	13E 30 CFR 11.4(b) (1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR 12.2(3) or applicable schedules as listed above

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 13 TO FACILITY OPERATING

LICENSE NO. DPR-49

CHANGE NO. 14 TO THE TECHNICAL SPECIFICATIONS

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

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

Introduction

By letter dated February 21, 1975, Iowa Electric Light and Power Company (IELP) proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). By letter dated May 28, 1975, IELP withdrew four of the proposed changes and provided further clarification of their original request. The proposed changes will (1) delete the calibration of the Average Power Range Monitor (APRM) when the reactor is in the startup mode, (2) delete the functional test of certain IRM and APRM trip channels before each startup if a successful functional test has been accomplished during the preceding seven days, (3) delete the calibration of the turbine control valve fast closure position trip which was not included in the DAEC design and is not required, (4) add specifications that were inadvertently omitted from the DAEC Technical Specifications, (5) provide clarification to several specifications, and (6) correct typographical errors.

This safety evaluation addresses only the proposed changes to Appendix A (Radiological Technical Specifications). The proposed changes to Appendix B (Environmental Technical Specifications) are the subject of a separate evaluation.

Discussion and Evaluation

1. Deletion of the APRM Gain Adjust Calibration when in the Startup Mode

   The current DAEC Technical Specifications require that the APRM gain adjust calibration be performed once per day when the reactor is in the startup or run mode. IELP has requested a change to the Duane Arnold Energy Center Technical Specifications that would delete the requirement to calibrate the APRM when the reactor is in the startup mode. The APRM is calibrated by performing a reactor heat balance and then adjusting the APRM gain so that the power determined from the APRM is in agreement with that determined from the reactor heat balance.

IELP has requested this change to the Technical Specifications because heat balances at the low reactor power levels associated with the startup mode are inherently inaccurate.

The APRM subsystem is based on the LOCAL Power Range Monitor (LPRM) neutron detectors. These LPRM detectors are calibrated every 1000 effective full power hours or less using the Traversing Incore Probe (TIP) subsystem. Once this calibration of the LPRM subsystem has been performed, the relative power distribution of the reactor can be determined. Then, a normalization of the APRM subsystem reactor power determination to the reactor power obtained from a reactor heat balance allows a determination of the absolute reactor power distribution by the LPRMs and the core thermal power by the APRM subsystem.

The procedure outlined in the preceding paragraph is applicable when the reactor power determined from a heat balance is accurate. For the Duane Arnold reactor, the calibration of the APRM subsystem with the reactor heat balance is viable when the reactor power is greater than about 12% of rated power. When the reactor power is greater than 12%, the reactor mode switch will be in the RUN position. Below 12% power the reactor heat balance is inherently inaccurate and the reactor mode switch will be, of necessity, in the STARTUP position.

With the reactor mode switch in the STARTUP position, the primary neutron flux scram signal to the reactor protection system is provided by the Intermediate Range Monitor (IRM) subsystem. The APRM system provides an additional neutron flux scram signal when the reactor power reaches a fixed value of 15% of rated. When the reactor power is less than 12%, IELP proposes to leave the APRM gain adjustments at the last calibration setting made while above 12% power. If new LPRMs are installed during the course of a refueling outage, IELP proposes to adjust the gain settings conservatively high. Subsequently, on a return to power levels above 12%, both the APRM and LPRM systems will be calibrated.

The IRMs, on the other hand, provide an indication of regional core power changes. The readings of the IRMs, however, can be roughly calibrated to indicate reactor power by, for example, a previous calibration with the APRM subsystem.

The abnormal operating transient of importance below 12% power is a reactivity insertion by the removal of a control rod of high worth. In the Duane Arnold reactor both the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer (RWM) will limit the reactivity worth of a control rod from the 50% rod density to the present power level. These two systems will also prevent movement of out-of-sequence control rods.

In the unlikely event of failure of both of these systems, a rod withdrawal error occurring below 12% power will be turned around by the Doppler and moderator coefficients of reactivity such that the fuel temperature is below the melting point of UO_2, the clad temperature is below operating temperatures, and the clad strain is less than the design limit of 1%.

Based on our review of the information provided by IELP, we conclude that the Technical Specifications may be changed to delete the APRM calibration based on a reactor heat balance when the reactor mode switch is in the STARTUP position. During the course of our review, we informed IELP that we will require that an APRM calibration based on a reactor heat balance be made as soon as possible after the reactor mode switch is placed in the RUN position. IELP stated that the APRM subsystem will be calibrated prior to reaching 20% power and once per day when the reactor mode switch is in the RUN position. We find that this calibration frequency is acceptable. We conclude that the revised APRM calibration frequency will provide adequate accuracy of the APRM subsystem during power operation.

2. <u>Modification of Functional Test Frequency for IRM and APRM of Trip Channels</u>

The current DAEC Technical Specifications require that the IRM High Flux and IRM Inoperative Trip channels and alarms and also the APRM High Flux trip output relays (in startup or refuel modes) be functionally tested once per week during refueling or startup, and before each startup. IELP has proposed modifying the frequency for the functional tests to once per week during refueling or startup and before each startup unless successful functional tests were accomplished during the preceding seven days. The functional test period of once per week was established in order to satisfy the safety objectives derived from the accident analyses for this plant. This technical specification change does not alter the requirement for weekly functional tests. Based on our review of the proposed changes we find that the functional test frequency, as modified, will provide adequate assurance of the operability of the trip channels. We conclude that the proposed change is acceptable.

3. <u>Deletion of the Turbine Control Valve Fast Closure Position Trip</u>

The current DAEC Technical Specifications include a turbine control valve fast closure position trip function that has a trip level setting "valve 10% closed". However, the DAEC design does not include this trip function. This trip function is not required for the safe operation of the facility. However, the DAEC design does include a turbine control valve fast closure trip based on loss of electrohydraulic control (EHC) system oil pressure and this trip is included in the Technical Specifications. Based on our review, we find that the turbine control valve fast closure position trip (valve 10% closed) was inadvertently included in the DAEC Technical Specifications and that this trip is not required for safe operation of the facility. We conclude that the deletion of the turbine control valve fast closure position trip from the DAEC Technical Specifications is acceptable.

4. Specifications Inadvertently Omitted from Technical Specifications

A. The Technical Specifications currently list a limiting safety system setting for "main steam isolation valve closure - loss of main condenser vacuum"; however, the Limiting Condition for Operation (LCO) and the surveillance requirements were inadvertently omitted. IELP has proposed LCO and surveillance requirements for the protective instrumentation. We have reviewed the proposed specifications and find that the instrument settings and surveillance requirements are acceptable. We conclude that the proposed additions to the Technical Specifications will provide adequate assurance of the operability of the (loss of main condenser vacuum) protective instrumentation.

B. The specification of two face mask protection factors and also the modes of use of self-contained breathing apparatus were inadvertently omitted from the DAEC Technical Specifications. IELP has proposed protection factors taken from the draft of WASH-1287, "Manual of Respiratory Protection Against Airborne Radioactive Materials". We have reviewed the proposed protection factors and we conclude that they will provide conservative factors for self-contained breathing apparatus.

C. The turbine first stage pressure permissive trip function was inadvertently omitted from the limiting condition for operation in the DAEC Technical Specifications. The functional test and calibration requirements for this trip function are currently specified in the Technical Specifications. IELP has proposed an instrumentation requirement for this trip function. We have reviewed the proposed specification and we conclude that the proposed reactor protection system instrumentation requirement for the turbine first stage pressure permissive trip function will provide adequate assurance that the trip will perform its intended function.

In addition, IELP has proposed reducing the turbine first stage pressure level, at which the scram signal due to turbine stop valve closure is bypassed, from 205 psig to 192 psig. IELP stated that this change is necessary because the value of 205 psig in the original design control document was found to be in error. We have reviewed this proposed change and find that it would provide an unbypassed scram function over a wider range of reactor operation than the current specification. Based on our review, we conclude that the proposed reduction in the pressure level below which the turbine stop valve closure scram will be bypassed, is acceptable.

5. Clarification of Existing Technical Specifications

A. The bases pertaining to vacuum relief (specification 3.7.A.3 Bases) in the current Technical Specifications are not consistent with the limiting conditions for operation (LCO) and the surveillance requirements (specifications 3.7.A.4.b and c, and 4.7.4.b). During the preparation of the Duane Arnold Technical Specifications, these LCO and surveillance requirements were changed to their present state with our concurrence; however, the bases were inadvertently not changed. IELP has proposed modifying the bases to be consistent with the LCO and the surveillance requirements. We have reviewed the proposed change to the Technical Specifications bases and find that there is no significant hazards consideration relating to this change. We conclude that the proposed modification to the bases is acceptable.

B. IELP has proposed clarifications to the Technical Specifications regarding testing of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems. The clarifications (1) provide a tolerance band of $\pm$ 10 psig about the nominal test pressure of 150 psig currently specified in the Technical Specifications for low pressure testing, and (2) remove the inference that the reactor pressure must be 1020 psig in order to test the systems at high pressure. The high pressure tests for both systems are performed by pumping water into the condensate storage tank at a discharge pressure of 1100 psig. We find that these clarifications do not involve a significant hazards consideration. We conclude that these proposed changes are acceptable.

C. The current DAEC Technical Specifications exempt the "SRM-Detector Not in Startup Position" and "IRM-Detector Not in Startup Position" instrument channels from the functional test definition. IELP has proposed to eliminate the exemptions because this instrumentation can be functionally tested as described in the Definitions section of the Technical Specifications. We conclude that this proposed change should be made.

6. Typographical Errors

IELP has proposed correcting two typographical errors which do not involve significant hazards considerations.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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 of the health and safety of the public.

Dated: NOV 1 7 1975

UNITED STATES 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. 13 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 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 will (1) delete the calibration of the Average Power Range Monitor (APRM) when the reactor is in the startup mode; (2) delete the functional test of certain IRM and APRM trip channels before each startup if a successful functional test has been accomplished during the preceeding seven days; (3) delete the calibration of the turbine control valve fast closure position trip which was not included in the DAEC design and is not required; (4) add specifications that were inadvertently omitted from the DAEC Technical Specifications; (5) provide clarification to several specifications; and (6) correct typographical errors.

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 February 21, 1975 and supplement submitted May 28, 1975, (2) Amendment No. 13 to License No. DPR-49, with Change No. 14, 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 17th day of November, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter A. Paulson

Walter A. Paulson, Acting Chief
Operating Reactors Branch #3
Division of Reactor Licensing