



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. 102  
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 August 26, 1983, 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:

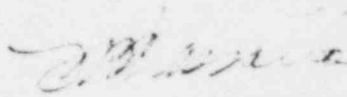
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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of its 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: June 25, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by removing the current pages and inserting the revised pages listed below. The revised areas are identified by vertical lines.

LIST OF AFFECTED PAGES

1.1-4

1.2-2

3.2-26

3.2-27

3.6-6

3.6-28

3.6-29

3.6-41

## LIMITING CONDITIONS FOR OPERATION

## SURVEILLANCE REQUIREMENT

E.	Scram - main steam line isolation valve	$\leq$ 10 percent valve closure
F.	Main steam isolation valve closure nuclear system low pressure	$\geq$ 850 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	$>$ 363 inches above vessel zero (+18.5 inches indicated level.)
J.	Main steam isolation valve closure- loss of main condenser vacuum	$\leq$ 10 inches Hg vacuum

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

- 2. The reactor vessel dome pressure shall not exceed 135 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

C. Relief Valve Settings - Low-Low Set Function

<u>Valve Group</u>	<u>Open</u>	<u>Close</u>
Low (1 valve)	1020 psig	900 psig
High (1 valve)	1025 psig	905 psig.

All settings are  $\pm 25$  psi.

D. Safety Valve settings

1240 psig + 12 psi  
(2 valves)

- 2. The shutdown cooling isolation valves shall be closed whenever the reactor vessel dome pressure is  $\geq 135$  psig.

TABLE 4.2-B  
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

<u>Instrument Channel</u>	<u>Instrument Functional Test (9)</u>	<u>Calibration Frequency (9)</u>	<u>Instrument Check</u>
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 - LPIC or CS Pump Discharge Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	Not Applicable	None
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 (HPIC & RCIC)	(1)	Once/3 months	None
10) Steam Line High Temp. (HPIC & 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) a. 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/operating cycle	None
b. 4KV Emergency Power System Voltage Relays (Degraded Voltage)	Once/month	Once/operating cycle	None
14) Instrument A.C. and battery bus undervoltage relays	(1)	Once/operating cycle	None
15) Low-Low Set Function	(1)	Once/6 months	Once/day

3.2-26

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TABLE 4.2-B (Continued)

## MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

<u>Logic System Functional Test (4) (6)</u>	<u>Calibration Frequency(9)</u>
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
9) Low-Low Set Function	Once/6 months

3.2-27

Amendment No. ~~29~~, 102

DAEC-1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
<p>2.</p> <p>a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.</p> <p>b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.</p>	<p>2. At least one of the relief valves shall be disassembled and inspected each refueling outage.</p>
<p>3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours.</p>	<p>3. With the reactor pressure <math>&gt; 100</math> psig and turbine bypass flow to the main condenser, each relief valve shall be manually opened and verified open by turbine bypass valve position decrease and pressure switches and thermocouple readings downstream of the relief valve to indicate steam flow from the valve once per operating cycle.</p>
	<p>4. The relief valve setpoints for the Low-Low Set function shall be as specified in Section 2.2.1.C. Instrumentation and system logic shall be functionally tested, calibrated, and checked as indicated in Table 4.2-8.</p>
<p>E. <u>Jet Pumps</u></p> <p>1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.</p>	<p>E. <u>Jet Pumps</u></p> <p>1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:</p> <p>a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.</p>



the direct scram (valve position scram) results in a peak vessel pressure less than the Code allowable overpressure limit of 1375 psig if a flux scram is assumed.

The analyses of the plant isolation transients are evaluated in each reload analyses. These analyses show that the six relief valves assure margin below the setting of the safety valves such that the safety valves would not be expected to open during any anticipated normal transient. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psia.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief and safety valves are benchtested every second operating cycle to ensure that their setpoints are within the  $\pm 1$  percent tolerance. Additionally, once per operating cycle, each relief valve is tested manually with reactor pressure above 100 psig and with turbine bypass flow to the main condenser to demonstrate its ability to pass steam. By observation of the change in position of the turbine bypass valve, the relief valve operation is verified.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The low-low set (LLS) function provides automatic relief mode setpoints on the two non-ADS safety/relief valves (S/RV's). The LLS function lowers the opening and closing setpoints after any S/RV has opened at its normal steam pilot setpoint when a concurrent high reactor vessel steam dome pressure scram signal is present. The purpose of the LLS is to mitigate the induced high frequency loads on the containment and thrust loads on the S/RV discharge lines. The LLS function increases the amount of reactor depressurization during an S/RV blowdown because the lowered LLS setpoints keep the two LLS S/RV's open for a longer time. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the LLS function such that failure of any one LLS valve to open or close at its reduced setpoint does not violate the design basis. (1)

The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

#### 3.6 and 4.6 References

- 1) General Electric Company, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center, NEDE-30021-P, January, 1983.