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. 106
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 March 16, 1984, 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. 106, 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 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: August 24, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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

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* Added new page

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	2) Closure of containment isolation valves for the Type A test shall be accomplished by normal mode of actuation and without any preliminary exercising or adjustments.
	3) The containment test pressure shall be allowed to stabilize for a period of about 4 hours prior to the start of a leakage rate test.
	4) The reactor coolant pressure boundary shall be vented to the containment atmosphere prior to the test and remain open during the test.
	5) Test methods are to comply with ANSI N45.4-1972.
	6) The accuracy of the Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4-1972.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	7) Periodic Leakage Rate Tests Periodic leakage rate tests shall be performed at peak pressure (Pa). 8) Acceptance Criteria The maximum allowable leakage rate (Lam) is 0.75 La where La is defined as the design basis accident leakage rate of 2.0 weight percent of contained air per 24 hours at 54 psig. 9) Additional Requirements If any periodic Type A test fails to meet the applicable acceptance criteria the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission. If two consecutive periodic Type A tests fail to meet the acceptance criteria of 4.7.A.2.(a)(9) a Type A test shall be performed at each plant shutdown for major refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the subject acceptance criteria after which time the retest schedule of 4.7.A.2.(d) may be resumed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

b. Type B Tests

Type B tests refer to penetrations with gasketed seals, expansion bellows or other type of resilient seals as shown in Table 3.7-1.

1) Test Pressure

All Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

2) Acceptance Criteria

The combined leakage rate of all penetrations subject to Type B and C tests shall be less than 0.60 La.

c. Type C Tests

1) Type C tests shall be performed as specified in Table 3.7-2. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

2) Acceptance criteria - The combined leakage rate for all penetrations subject to Type B and C tests shall be less than 0.60 La.

3) The leakage from any one main steam isolation valve shall not exceed 11.5 scf/hr at an initial test pressure of 24 psig.

4) The leakage rate from any containment isolation valve whose seating surface remains water covered post-LOCA, and which is hydrostatically Type C tested, shall be included in the Type C test total. These valves are identified in Table 3.7-2 of this Technical Specification.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	d. Periodic Retest Schedule

1) Type A Test

After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. (These intervals may be extended up to eight months if necessary to coincide with refueling outages.) The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is nonoperational and secured in the shutdown condition under administrative control and in accordance with the plant safety procedures.

2) Type B Tests

a) Penetrations and seals of this type (except air locks) shall be leak tested at P_a (54 psig) during each reactor shutdown for major fueling or other convenient interval but in no case at intervals greater than two years.

b) The personnel airlock shall be pressurized to P_a (54 psig) and leak tested at least once every six (6) months. This test interval may be extended to the next refueling outage (up to a maximum interval between P_a tests of 24 months) provided there have been no airlock openings since the last successful test at P_a.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	c) Within three (3) days after securing the airlock when containment integrity is required, the airlock gaskets shall be leak tested at a pressure of P_a.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	3) Type C Tests Type C tests shall be performed during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years. e. Containment Modification Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test. f. Reporting Periodic tests shall be the subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of each test. The report will be titled "Reactor Containment Integrated Leakage Rate Test."

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	The report shall include a schematic arrangement or description of the leakage rate measurement system, the instrumentation used, the supplemental test method, the test program selected, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria. For each periodic test, leakage test results from Type A, B, and C tests shall be reported. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test. Leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria shall be reported in a separate accompanying summary report. The Type A test summary report shall include an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included. The Type B and C tests summary report shall include an analysis and interpretation of the data and the condition of the components which contributed to any failure in meeting the acceptance criteria.

TABLE 3.7-1

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TEST REQUIREMENTS

Penetration #	Type	Description
1	Testable Gaskets[2]	Personnel Lock Equipment Door
1	Personnel Lock[2]	Personnel Lock Doors and Penetrations
2	Testable Gaskets	Equipment Access
4	Testable Gaskets	Head Access
6	Testable Gaskets	CRD Removal Hatch
35A-D	Testable Gaskets	TIP Drives (4)
53	Testable Gaskets	Spare
----	Testable Gaskets	Drywell Head Flange
58 A-H	Testable Gaskets	Stabilizer Access Ports (8)
200A-B	Testable Gaskets	Torus Access Hatches (2)
100B,C,E,F,G	Electrical Canister	(B,C,E,F) Neutron Monitoring, (G) RPV Vibration Monitoring
101A,C	Electrical Canister	(C) (A) Recirc Pump Power
103	Electrical Canister	Thermocouples
104A-D	Electrical Canister	CRD Rod Position Indicator
105B,D	Electrical Canister	(B,D) Power & Control
106A,C	Electrical Canister	(A,C) Power & Control
230B	Electrical Canister	Vacuum Breakers Electrical Cables

TABLE 3.7-1 (Continued)

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TEST REQUIREMENTS

Penetration #	Type	Description
7A-D	Expansion Bellows	Steam to Turbine
9A,B	Expansion Bellows	RPV Feedwater
10	Expansion Bellows	Steam to RCIC Turbine
11	Expansion Bellows	Steam to HPCI Turbine
12	Expansion Bellows	Shutdown Pump Supply RHR
13A,B	Expansion Bellows	RHR Pump Discharge
15	Expansion Bellows	RWCU Supply
16A,B	Expansion Bellows	Core Spray Pump Discharge
17	Expansion Bellows	RPV Head Spray
201A-H	Expansion Bellows	Vent Lines
25	Flange "O" Rings[1]	Drywell Purge Outlet CV-4302
26,220	Flange "O" Rings	Drywell & Torus Purge Supply, CV-4307, CV-4308
205	Flange "O" Rings	Torus Purge Outlet, CV-4300
213A, B	Flange "O" Rings	Torus Drain Lines
231	Flange "O" Rings	Torus Vacuum Breakers, CV-4304, CV-4305

[1] Test inboard flange of designated valves.

[2] Testing to be in accordance with Technical Specification Section 4.7.A.2.d.2.

TABLE 3.7-2

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
7A	Main Steam Line	CV-4412[4], 4413
7B	Main Steam Line	CV-4415[4], 4416
7C	Main Steam Line	CV-4418[4], 4419
7D	Main Steam Line	CV-4420[4], 4421
8	Main Steam Line Drain	MO-4424
9A	Feedwater & HPCI Feed	V-14-3
9A[2]	Feedwater & HPCI Feed	MO-4441, MO-2312
9B	Feedwater	V-14-1
9B[2]	Feedwater & RCIC Feed & RWCU Return	MO-2740, MO-4442, MO-2512
10	RCIC Condensate Return	CV-2411
10	Steam to RCIC Turbine	MO-2401
11	Steam to HPCI Turbine	MO-2239
11	HPCI Condensate Return	CV-2212
15	RWCU Supply	MO-2700, MO-2701
16A	Core Spray Pump Discharge	MO-2115, MO-2117
16B	Core Spray Pump Discharge	MO-2135, MO-2137
19	Drywell Floor Drain Discharge	CV-3704, CV-3705
20	Demineralized Water Supply	V-09-65, V-09-111
21	Service Air Supply	V-30-287, Blind Flange
22, 229	Containment Compressor Discharge	CV-4371A, CV-4371C, V-43-214
23A[3], B[3]	Well Cooling Water Supply	CV-5718A, CV-5718B, CV-5719A, CV-5719B,
24A[3], B[3]	Well Cooling Water Return	CV-5704A, CV-5704B, CV-5703A, CV-5703B,
25	Drywell Purge Outlet	CV-4302[4], CV-4303, CV-4310
26, 220	Drywell and Torus Purge Supply	CV-4306, CV-4307[4], CV-4308[4]
26, 220	Drywell and Torus Nitrogen Makeup	CV-4311, CV-4312, CV-4313

TABLE 3.7-2 (Continued)

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
32D	Containment Compressor Suction	CV-4378A, CV-4378B
32E	Recirc Pump "A" Seal Purge	V-17-96, CV-1804B
32F	Recirc Pump "B" Seal Purge	V-17-83, CV-1804A
35A,B,C,D	T.I.P Drives	T.I.P. Ball Valves and Check Valve on X-35A
36[1]	CRD Return	V-17-53, V-17-52
39A	Containment Spray/CAD Supply	SV-4332A, SV-4332B
39B	Containment Spray/CAD Supply	SV-4331A, SV-4331B
40D	Post-Accident Sampling/Jet Pump Sample	SV-4594A, SV-4594B
41	Recirc Loop Sample	CV-4639[4], CV-4640
42	Standby Liquid Control	V-26-8, V-26-9
46E	O_2 Analyzer	SV-8105B, SV-8106B
48	Drywell Equipment Drain Discharge	CV-3728, CV-3729
50B	O_2 Analyzer	SV-8101A, SV-8102A,
50E	O_2 Analyzer	SV-8103A, SV-8104A,
50D	O_2 Analyzer	SV-8105A, SV-8106A
54[3]	Reactor Building Closed Cooling Water Return	MO-4841A
55[3]	Reactor Building Closed Cooling Water Supply	MO-4841B
56C	O_2 Analyzer	SV-8101B, SV-8102B,
56D	O_2 Analyzer	SV-8103B, SV-8104B
205	Torus Purge Outlet	CV-4300[4], CV-4301, CV-4309
211A	Torus Spray/CAD Supply	SV-4333A, SV-4333B
211B	Torus Spray/CAD Supply	SV-4334A, SV-4334B
212[1]	RCIC Turbine Exhaust	V-24-8[4], V-24-23 V-24-46, V-24-47
214[1]	HPCI Turbine Exhaust	V-22-16, V-22-17[4] V-22-63, V-22-64

TABLE 3.7-2 (Continued)

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
219	HPCI/RCIC Exhaust Vacuum Breaker	MO-2290A, MO-2290B
222[1]	HPCI Condensate	V-22-21, V-22-22[4]
229B	O_2 Analyzer	SV-8107A, SV-8108A,
229C	O_2 Analyzer	SV-8109A, SV-8110A,
229G	O_2 Analyzer	SV-8107B, SV-8108B,
229F	O_2 Analyzer	SV-8109B, SV-8110B
229H	Post-Accident Sampling System Liquid Sample Return	SV-8772A, SV-8772B
231	Torus Vacuum Breakers	CV-4304[4], V-43-169
231	Torus Vacuum Breakers	CV-4305[4], V-43-168

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 7A-D), test volumes are pressurized to 54 psig with air or nitrogen for test.

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

[3] In accordance with 10 CFR 50, Appendix A, General Design Criterion 57, the redundant barriers are a single isolation valve outside containment and a closed system inside. Testing of the single isolation valve only is required.

[4] Tested in reverse direction.

with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 90% for particulate iodine, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 2 rem and the maximum thyroid dose is about 32 rem at the site boundary over an exposure duration of two hours. The resultant thyroid dose that would occur over the course of the accident is 98 rem at the boundary of the low population zone (LPZ). Thus. these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate is conservative and provides additional margin between expected offsite doses and 10 CFR 100 guidelines.

The design basis accident leak rate (L_a) at the peak accident pressure of 54 psig (P_a) is 2.0 weight percent per day. To allow a margin for possible leakage deterioration during the interval between Type A tests, the maximum allowable containment operational leak rate (L_{am}), is 0.75 L_a.

Type B and Type C tests are performed on testable penetrations and isolation valves during the interim period between Type A tests. This provides assurance that components most likely to undergo degradation between Type A tests maintain leaktight integrity.

The containment leakage testing program is based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels, (Reference 4).

5. Drywell Interior

The interiors of the drywell and suppression chamber are coated to prevent corrosion and for ease of decontamination. The inspection of the coating during each major refueling outage, assures the coating is intact. Experience with this type of coating at fossil fueled generating stations indicates that the inspection interval is adequate.

6. Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e., the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. The CAD system serves as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 50,000 scf of liquid N_2 in the storage bank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available.

The Post-LOCA Containment Atmosphere Dilution System design basis and description are presented in the response to Question G.7.3 and Question G.7.4 of the FSAR. In summary, the limiting criteria, based on the assumptions of Safety Guide No. 7 are:

1. Maintain oxygen concentration in the containment during post-LOCA conditions to less than 4 Volume %.

3.7.A & 4.7.A REFERENCES

1. Section 14.6 of the FSAR.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR Part 50, Appendix J, Reactor Containment Testing Requirements, Federal Register, April 19, 1976.

5. DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-065, August 1976.

6. Supplement to DAEC Short-Term Program Plant Unique Analysis, NUTECH Doc. No. IOW-01-071, October 1976.