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. 181
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, et al., dated September 20, 1991 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 Appendix A, as revised through Amendment No. 181 , 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: March 11, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 181

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove	Insert
vi	vi
3.2-36a	3.2.-36a
3.2-38	3.2-38
3.2-39	3.2-39
3.7-2	3.7-2
3.7-5 through 3.7-7	3.7-5 through 3.7-7
3.7-18 through 3.7-29a	3.7-18 through 3.7-20
3.7-38	3.7-38
3.7-47	3.7-47
3.7-48	3.7-48

TECHNICAL SPECIFICATIONS

LIST OF TABLES (Continued)

Table Number	Title	Page
3.7-1	Deleted	
3.7-2	Deleted	
3.7-3	Deleted	
4.7-1	Summary Table of New Activated Carbon Physical Properties	3.7-50
4.10-1	Summary Table of New Activated Carbon Physical Properties	3.10-7
3.12-1	Deleted	
3.12-2	Deleted	
3.13-1	Fire Detection Instruments	3.13-11
3.13-2	Required Fire Hose Stations	3.13-12
3.14-1	Radioactive Liquid Effluent Monitoring Instrumentation	3.14-5
4.14-1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	3.14-7
4.14-2	Radioactive Liquid Waste Sampling and Analysis Program	3.14-9
3.15-1	Radioactive Gaseous Effluent Monitoring Instrumentation	3.15-7
4.15-1	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	3.15-9
4.15-2	Radioactive Gaseous Waste Sampling and Analysis Program	3.15-11
3.16-1	Radiological Environmental Monitoring Program	3.16-6
3.16-2	Maximum Values of the Lower Limit of Detection for Environmental Sample Analysis	3.16-8
3.16-3	Reporting Levels for Radioactivity Concentrations in Environmental Samples	3.16-10
6.2-1	Minimum Shift Crew Personnel and License Requirements	6.2-3
6.9-1	Deleted	
6.11-1	Reporting Summary - Routine Reports	6.11-8
6.11-2	Deleted	
6.11-3a	Semiannual Radioactive Material Release Report Liquid Effluents	6.11-10
6.11-3b	Semiannual Radioactive Material Release Report Gaseous Effluents	6.11-11

The low water level instrumentation set to trip at 170" above the top of the active fuel closes all isolation valves except those in Groups 1, 6, 7 and 9. For valves which isolate at this level this trip setting is

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Group 6.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and consequently main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel clad temperatures peak at approximately 1000°F and release of radioactivity to the environs is below 10 CFR 100 guidelines. Reference Subsection 15.6.5 of the Updated FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel and turbine building to detect leaks in this area. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting is 200°F for the main steam line tunnel detector. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. For the performance of a Hydrogen Water Chemistry pre-implementation test, the scram setpoint may be changed based on a calculated value of the radiation level expected during the test. Hydrogen addition will result in an approximate one- to five-fold increase in the nitrogen (N-16) activity in the steam due to increased N-16 carryover in the main steam. Reference Subsection 15.4.7 of the Updated FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 850 psig. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in Subsection 15.6.3 of the Updated FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN Mode is not required.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.	
(4) During reactor isolation conditions, the reactor shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.	
2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t). Compliance with Subsection 3.7.D.2 satisfies the requirement to maintain primary containment integrity.	2. The primary containment integrity shall be demonstrated as follows:
	a. <u>Type A Test</u>
	Primary Reactor Containment Integrated Leakage Rate Test
	1) The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of deterioration. In addition, the external surfaces of the torus below the water level shall be inspected on a routine basis for evidence of torus corrosion or leakage.
	Except for the initial Type A test, all Type A tests shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test.
	If a Type A test is completed but the acceptance criteria of Specification 4.7.A.2.a.(9) is not satisfied and repairs are necessary, the Type A test need not be repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

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.

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 on containment isolation valves. 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.

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 greater than or equal to 43 psig (P_a) 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 greater than or equal to 43 psig (P_a) 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

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.

4) Additional Periodic Tests

Additional purge system isolation valve leakage integrity testing shall be performed at least once every three months in order to detect excessive leakage of the purge isolation valve resilient seats. The purge system isolation valves will be tested in three groups, by penetration: drywell purge exhaust group (CV-4302 and CV-4303), torus purge exhaust group (CV-4300 and CV-4301), and drywell/torus purge supply group (CV-4307, CV-4308 and CV-4306).

e. Seal Replacement & Mechanical Limiter

The T-ring inflatable seals for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 shall be replaced at intervals not to exceed four years.

During Type C testing, it shall be verified that the mechanical modification which limits the maximum opening angle for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 is intact.

The baseline for this requirement shall be established during the Cycle 6/7 refuel outage.

f. 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.

2. If Specification 3.7.C.1 cannot be met:

a. Suspend reactor building fuel cask and irradiated fuel movement, and

b. Restore secondary containment integrity within one hour; or,

c. Be in COLD SHUTDOWN within the following 24 hours.

D. Primary Containment Power Operated Isolation Valves

1. During reactor power operating conditions, all primary containment isolation valves and all instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.

maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

D. Primary Containment Power Operated Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the OPERABLE isolation valves* that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. At least once per quarter:

1) All normally open power operated isolation valves** shall be fully closed and reopened.

2) With the reactor power less than 75%, trip main steam isolation valves individually and verify closure time.

c. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

*Due to operation limitations, the Main Steam Line Isolation Valves are exempt from Subsection 4.7.D.1.a.

**Due to plant operational limitations, the Well Cooling Water Supply/Return Valves, Reactor Building Closed Cooling Water Supply/Return Valves and the Containment Compressor Discharge and Suction Valves are exempt from the requirements of Subsection 4.7.D.1.b.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

2. With one or more of the primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE* or ISOLATED** and within 4 hours either:

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one automatic valve locked or electrically deactivated in the isolated position,** or

c. Isolate each affected penetration by use of at least one manual valve locked in the isolated position or blind flange.**

3. If Specification 3.7.D.1, and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

*This valve may be locked or electrically deactivated as noted in Subsection 3.7.D.2.b.

**Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

PAGES 3.7-20 THROUGH 3.7-29a THAT

CONTAINED TABLES 3.7-1, 3.7-2, AND 3.7-3

ARE DELETED IN THEIR ENTIRETY

Next page is 3.7-30

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. A controlled list of the testable penetrations and isolation valves subject to Type B and Type C testing is located in the plant Administrative Control Procedures.

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, atmosphere. The maximum closure times for these valves are selected in consideration of the design intent to contain released fission products following pipe breaks inside containment. Several of the automatic isolation valves serve a dual role as both reactor coolant pressure boundary isolation valves and containment isolation valves. The function of such valves on reactor coolant pressure boundary process piping which penetrates containment (except for those lines which are required to operate to mitigate the consequences of a loss-of-coolant accident) is to provide closure at a rate which will prevent core uncovery following pipe breaks outside primary containment. A controlled list of the primary containment power operated isolation valves is located in the plant Administrative Control Procedures.

In order to assure that the doses that may result from a steam line break are within 10 CFR 100 guidelines, it is necessary that no fuel rod perforation results from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of 5 seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided. Redundant valves in each line insure that isolation will meet the single failure criteria.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. The excess flow check valves in these lines shall be tested once each operating cycle.

Containment vent/purge valves (CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307, and CV-4308) have been mechanically modified to limit the maximum opening angle to 30 degrees. This has been done to ensure these valves are able to close against the maximum differential pressure expected to occur during a design basis accident.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.