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. 108
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 July 20, 1983, as supplemented January 27, 1984 and August 8, 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. 108, 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: October 29, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 108

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 pages are identified by Amendment number and contain vertical lines indicating the areas of change.

AFFECTED PAGES

3.5-4
3.5-5
3.5-17
3.5-18
3.5-26
3.7-1
3.7-2
3.7-32
3.7-32a
3.7-32b
3.7-49

LIMITING CONDITION FOR OPERATION

6. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

B. Containment Spray Cooling Capability

1. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in 3.5.C.2, 3.5.C.3, 3.5.C.4, 3.5.C.5, and 3.5.G.3 below, both RHR service water subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

SURVEILLANCE REQUIREMENT

6. Once per shift visually inspect and verify that RHR valve panel lights and instrumentation are functioning normally.

B. Containment Spray Cooling Capability

Surveillance of the drywell spray loops shall be performed as follows:

1. During each five year period, an air test shall be performed on the drywell and suppression pool spray headers and nozzles.

C. Surveillance of the RHR Service Water System

1. Surveillance of the RHR service water system shall be as follows:

RHR Service Water Subsystem Testing:

	Item	Frequency
a)	Pump and motor operated valve operability.	Once/3 months
b)	Flow Rate Test-Each RHR service water pump shall deliver at least 2040 gpm at a TDH of 610 ft. or more.	after major pump maintenance and every 3 months

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 pump in each subsystem is made or found to be inoperable for any reason, reactor operation is limited to seven days unless operability of at least one pump is restored within this period. During such seven days all active components of both RHR Service Water subsystems and their associated diesel generators required for operation of such components (if no external source of power were available), shall be operable.	3. When one RHR Service Water pump in each subsystem becomes inoperable, the remaining components of both subsystems and their associated diesel-generators required for operation of such components, shall be demonstrated to be operable immediately. The remaining components of both subsystems shall be demonstrated to be operable daily thereafter.
4. 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 one pump is restored within this period. During such seven days all active components of the other RHR Service Water subsystem, and its associated diesel-generator required for operation of such components (if no external source of power were available), shall be operable.	4. When one RHR Service Water subsystem becomes inoperable, the operable subsystem and the diesel-generator required for operation of such components shall be demonstrated to be operable immediately. The operable subsystem (excluding diesel generators) shall be demonstrated to be operable daily thereafter.
5. If the requirements of 3.5.C cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.	

1 LPCI pump must be available to fulfill the containment spray function. The 7 day repair period is set on this basis.

B&C Containment Spray and RHR Service Water

The containment spray subsystem for DAEC consists of 2 loops each with 2 LPCI pumps and 2 RHR service water pumps per loop. The design of these systems is predicted upon use of 1 LPCI, and 2 RHR service water pumps for heat removal after a design basis event. Thus, there are ample spares for margin above the design conditions. Loss of margin should be avoided and the equipment maintained in a state of operability so a 30-day out-of-service time is chosen for this equipment. If one loop is out-of-service, or one pump in each loop is out-of-service, reactor operation is permitted for seven days with daily testing of the operable loop(s) after testing the appropriate diesel generator(s).

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operabililty of the remaining cooling equipment. The degree of operabililty to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity, the other pumps of this type might be subjected to a capacity test. In any event, surveillance procedures, as required by Section 6 of these specifications, detail the required extent of testing.

The pump capacity test is a comparison of measured pump performance parameters to shop performance tests. Tests during normal operation will be performed by measuring the flow indication and/or the pump discharge pressure will be measured and its power requirement will be used to establish flow at that pressure.

Analyses were performed to determine the minimum required flow rate of the RHR Service Water pumps in order to meet the design basis case (Reference 4) and the NUREG-0783 requirements (Reference 5). (See Section 3.7.A.1 Bases for a discussion of the NUREG requirements.) The results of these analyses justify reducing the required flowrate to 2040 gpm per pump, a 15% reduction in the original 2400 gpm per pump requirement.

D. HPCI System

The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss-of-coolant, which

3.5 REFERENCES

1. Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Company, APED, April 1968 (APED 5736).

2. General Electric Company, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K, NEDO-20566, 1974, and letter MFN-265-77 from Darrell G. Eisenhut, NRC, to E.D. Fuller, GE, Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-lead Plants, dated June 30, 1977.

3. General Electric, Loss-of-Coolant Accident Analysis Report for Duane Arnold Energy Center (Lead Plant), NEDO-21082-02-1A, Rev. 2, June 1982.

4. General Electric Company, Analysis of Reduced RHR Service Water Flow at the Duane Arnold Energy Center, NEDE-30051-P, January 1983.

5. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

LIMITING CONDITION FOR OPERATION

3.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained with the following limits.

a. Maximum water volume - 61,500 cubic feet

b. Minimum water volume - 58,900 cubic feet

c. Maximum water temperature

(1) During normal power operation - 95F.

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

SURVEILLANCE REQUIREMENT

4.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1.a. The pressure suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160F or more and the primary coolant pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110F. 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).	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.

2. There is no significant thermal stratification in the condensation oscillation regime after LOCA with three feet submergence.

3. There is some thermal stratification in the chugging regime for all break sizes. However, this will not inhibit the pressure suppression function of the suppression pool.

4. Seismic induced waves will not cause downcomer vent uncovering with three feet submergence.

5. Post-LOCA pool waves will not cause downcomer vent uncovering with three feet submergence.

6. Maximum post-LOCA drawdown will not cause downcomer vent uncovering and condensation effectiveness of the suppression pool will be maintained.

Therefore, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Using a 50°F rise (Table 6.2-1, UFSAR) in the suppression chamber water temperature and a minimum water volume of 58,900 ft^3, the 170° temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 95°F will assure that the 170°F limit is not approached.

As part of the program to reduce the loads on BWR containments, the NRC issued NUREG-0783, which limits local suppression pool temperatures during Safety Relief Valve (SRV) actuations. Stable steam condensation is assured in the vicinity of T-type quenchers on SRV discharge lines if the following limits on local suppression pool temperatures are met:

1. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 94 lbm/ft^2-sec, the suppression pool local temperature shall not exceed 200°F.

2. For all plant transients involving SRV operations during which the steam flux through the quencher perforations is less than 42 lbm/ft^2-sec, the suppression pool local temperature shall be at least 20°F subcooled.

3. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 42 lbm/ft^2-sec, but less than 94 lbm/ft^2-sec, the suppression pool local temperature is obtained by linearly interpolating the local temperatures established under aforementioned items 1 and 2.

Maintaining the suppression pool temperature below the normal operating limit of 95°F, and scramming the reactor if the pool temperature reaches 110°F, will ensure that the local temperature limits outlined above are not exceeded during plant transients.(7)

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in Basis 3.5.G or the requirements of Specification 3.5.G.4 are met.

2. Inerting

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of the Safety

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 50.54, Appendix J, Reactor Containment Testing Requirements, Federal Register, August 27, 1971.

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.

7. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.