October 29, 1984

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications in response to your application dated July 20, 1983, as supplemented January 27, 1984 and August 8, 1984.

The amendment revises the Technical Specifications and the bases to permit operation of the Residual Heat Removal (RHR) system with reduced water flow and corrects a discrepancy between the bases to the Technical Specifications and the Updated Final Safety Analysis Report (UFSAR) for DAEC.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Mohan C. Thadani, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 108 to
   License No. DPR-49
2. Safety Evaluation

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cc w/enclosures:
See next page

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Mr. Lee Liu
Iowa Electric Light and Power Company
Duane Arnold Energy Center

cc:

Jack Newman, Esquire
Harold F. Reis, Esquire
Newman and Holtzinger
1615 L Street, N. W.
Washington, D. C. 20036

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

Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

Iowa Electric Light and Power Company
ATTN: D. L. Mineck
Post Office Box 351
Cedar Rapids, Iowa 52406

U. S. Environmental Protection
Agency
Region VII Office
Regional Radiation Representative
324 East 11th Street
Kansas City, Missouri 64106

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, Iowa 52324

James G. Keppler
Regional Radiation Representative
Region III Office
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Thomas Houvenagle
Regulatory Engineer
Iowa Commerce Commission
Lucas State Office Building
Des Moines, Iowa 50319

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-255-77 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	SURVEILLANCE REQUIREMENT
3.7 PLANT CONTAINMENT SYSTEMS	4.7 PLANT CONTAINMENT SYSTEMS
Applicability:	Applicability:
Applies to the operating status of the primary and secondary containment systems.	Applies to the primary and secondary containment system integrity.
Objective:	Objective:
To assure the integrity of the primary and secondary containment systems.	To verify the integrity of the primary and secondary containments.
Specification:	Specification:
A. Primary Containment	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.	1.a. The pressure suppression pool water level and temperature shall be checked once per day.
a. Maximum water volume - 61,500 cubic feet	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.
b. Minimum water volume - 58,900 cubic feet	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.
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.	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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 108 TO LICENSE NO. DPR-49

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

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 Introduction

By a letter dated July 20, 1983, Iowa Electric Light and Power Company (the licensee/IELP) proposed a change to the Duane Arnold Energy Center (DAEC) Technical Specifications to permit Residual Heat Removal Service Water (RHRSW) flow reduction. This change would allow the excess capacity of the service water flow (above the design basis performance requirements) currently required for operation of the Residual Heat Removal (RHR) system to be eliminated. Subsequently, the licensee by a letter dated January 27, 1984, revised the July 20, 1983 submittal to correct a discrepancy between the bases of the Technical Specifications and the Updated Final Safety Analysis Report (UFSAR), discovered subsequent to the original application.

The current Technical Specification Bases state that only one RHRSW pump is required to be operable to meet the design bases requirements, while the UFSAR states that two pumps are required to provide the necessary coolant flow. The licensee's investigation shows that the UFSAR analysis is correct and the current Technical Specifications need to be revised to require at least two RHR pumps to be operable. Furthermore, the licensee's analysis has shown that if one pump is operable in each of the two RHR systems, the resulting condition is similar to having one RHR system operable and adequate RHRSW flow is achieved. The licensee has also proposed to modify the diesel generator surveillance requirements for the RHRSW system eliminating the daily testing requirements.

2.0 Evaluation

The licensee has requested that the service water system flow be reduced below the currently documented and approved rated value of 4800 gpm to each RHR heat exchanger for design basis heat removal. There is no change in RHR system flow in the primary side. Only the service water flow is reduced. This change was prompted by a number of instances of failures to meet the above flow rate during surveillance testing.

The primary function of the RHR service water system is to provide cooling water to the RHR system heat exchangers during various modes of operation of the RHR system. The design specification for the RHR system states that the shutdown cooling mode is considered to be the limiting case for design basis heat removal, but that the steam condensing mode should also be evaluated as it may sometimes govern heat removal requirements. While not a limiting mode of operation, the RHR system is also used for suppression pool cooling during certain plant transients.

The licensee contracted General Electric Company (GE) to analyze the RHR service water system to determine the minimum flow rate required to meet the design basis conditions. In support of the licensee's requested change, GE performed analysis and provided a licensing letter report, "Duane Arnold Energy Center Reduced RHR Service Water Flow and Suppression Pool Temperature Response." The GE analysis considered the operation of the RHR system in both the shutdown cooling and steam condensing modes. The analysis verified that with both RHR heat exchangers operating and with 30% reduced RHR service water flow, the shutdown cooling subsystem meets its performance requirement of cooling the reactor to 125° within 20 hours following reactor trip.

Our position, Reactor Systems Branch (RSB) Technical Position 5-1, requires that the RHR system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure. (The cold shutdown condition, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 200°F for a PWR, 212°F for a BWR.)

In a telephone conference with the licensee on November 15, 1983, we requested clarification regarding the number of RHR heat exchangers assumed in the licensee's analysis to be in operation. The licensee confirmed that with a single RHR heat exchanger, the reactor coolant temperature would be less than 212°F in 20 hours following reactor trip. This satisfies RSB Technical Position 5-1.

Our review of the analysis indicted that the RHR service water system flow can be reduced and still satisfy design basis heat removal requirements. The analysis showed that the RHR service water flow rate to each RHR heat exchanger may be reduced by approximately 30% in the shutdown cooling mode and still meet the design basis heat removal performance requirements. Further, the analysis determined that the steam condensing mode was not limiting as a large excess capacity existed. Operation of the RHR system in the suppression pool cooling mode was analyzed only for a 15% reduction in service water flow. Based on the above, the licensee has requested a 15% total reduction in the limiting RHR service water system flow to bound the cooling requirements in each mode of operation. The reduction of 15% in the minimum required flow rate does not require any system hardware changes.

Based on our evaluation of the results of the supporting analysis presented by the licensee, we conclude that with the 15% reduction of the RHR service water flowrate, the RHR system is adequate to meet our RSB Technical Position 5-1. The licensee's request is, therefore, acceptable.

The licensee proposed to relax the requirements of daily testing of the diesel generators required for the operation of the RHR service water system. The daily testing requirement is not consistent with the diesel generators testing in relation to other Emergency Core Cooling System (ECCS) subsystems. The licensee has proposed to change the Technical Specifications to eliminate the daily testing of the diesel generators when the RHR service water system becomes inoperable. Instead the licensee proposes to demonstrate that the diesel generators will be operable only immediately after a RHR service water system becomes inoperable. The diesel generators will not be tested daily thereafter. The staff, as a part of the evaluation of Generic Issue B-56, has concluded that excessive testing of diesel generators results in degradation of diesel engines. Therefore, the staff is considering a generic reduction, from the plant Technical Specifications, of unnecessary test starts of diesel generators when ECCS systems are inoperable. The licensee's request for reduced diesel generator testing is consistent with the current staff position on this issue, and is therefore acceptable.

3.0 Environmental Considerations

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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 or to the health and safety of the public.

Principal Contributor: George Thomas and Mohan Thadani

Dated: October 29, 1984