April 29, 1983

Docket No. 50-331

Mr. Lee Liu, President
and Chief Executive Officer
Iowa Electric Light and Power Company
P. O. Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your application dated March 10, 1983.

This amendment to the Technical Specifications revises the following; (1) the Group I Containment Isolation signal from reactor vessel low-low water level to reactor low-low-low water level, (2) the Group I isolation reactor pressure setpoint from 880 psig to 850 psig and (3) the low-low set logic to the two non-ADS safety relief valves.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Frank L. Apicella, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 89 to DPR-49
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

DIST:	Docket File	NRC PDR	LPDR	ORB#2 Rdg	DEisenhut
SNorris	FApicella	OELD	SECY	LJHarmon-1	TBarnhart-4
LSchneider-1		DBrinkman	ACRS-10	OPA-CMiles	RFerguson
RDiggs	RBallard	NSIC	Gray	ASLAB	EXTRA-5

no legal objection to FR notice or amendment.

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:AD:OR	OELD	
SURNAME	SNorris	FApicella:pob	DVassallo	GLainas	LJCHANDLER	
DATE	4/27/83	4/21/83	4/27/83	4/1?/83	4/?/83	
		MC-1700				

Mr. Lee Liu
Iowa Electric Light & Power Company

cc:

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Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

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Des Moines, Iowa 50319

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Region VII Office
Regional Radiation Representative
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U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, Iowa 52324

James G. Keppler
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
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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. 89
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 10, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This 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: April 29, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix A Technical Specifications by removing current pages and inserting revised pages listed below. The revised area is identified by a vertical line.

List of Affected Pages

3.1-6
3.2-5
3.2-5a
3.2-7
3.2-37
3.2-39
3.7-27

3. A main steam line isolation valve closure trip bypass is effective when the reactor mode switch is in the shutdown, refuel or startup positions.

4. Bypassed when turbine first stage pressure is less than 192 psig or less than 30% of rated.

5. IRM's are bypassed when APRM's are on-scale and the reactor mode switch is in the run position.

6. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode switch in shutdown

   b. Manual scram

   c. High flux IRM

   d. Scram discharge volume high level - may be bypassed in the refuel and shutdown modes for the purpose of resetting the scram.

   e. APRM 15% flux

Amendment No. 89

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Valve Groups Operated by Signal	Action (2)
2 (6)	Reactor Low Water Level	>+170" Indicated Level (3)	4	2,3,4,5 (Sec. Cont., 3	A E)
1	Reactor Low Pressure (Shutdown Cooling Isolation)	< 135 psig	2	4	C
2	Reactor Low-Low-Low Water Level	At or above +18.5" indicated level (3)	4	1	A
2 (6)	High Drywell Pressure	< 2.0 psig	4	2,3,4,8,9* (Sec. Cont., 3	A E)
2	High Radiation Main Steam Line Tunnel	< 3 X Normal Rated Power Background	4	1	B
2	Low Pressure Main Steam Line	> 850 psig (7)	4	1	B
2 (5)	High Flow Main Steam Line	< 140% of Rated Steam Flow	4	1	B
2	Main Steam Line Tunnel/Turbine Bldg. High Temperature	< 200° F.	4	1	B
1	Reactor Cleanup System High Diff. Flow	< 40 gpmd	2	5	D

*Group 9 valves isolate on high drywell pressure combined with reactor steam supply low pressure

TABLE 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION (continued)

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Valve Groups Operated by Signal	Action (2)
1	Reactor Cleanup Area Ambient High Temperature	130°F	3	5	D
1	Reactor Cleanup Area Differential High Temperature	Δ 14°F*	3	5	D
2	Loss of Main Condensor Vacuum	≤ 10 in Hg Vacuum	4	1	B
2	Reactor Low-Low Water Level	At or above +119.5" indicated level (3)	4	8	A

*Note: The actual setpoint shall be Δ14°F above the 100% operation ambient temperature conditions as determined by DAEC Plant Test Procedure.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

adequate to prevent uncovering the core in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low-low reactor water level instrumentation is set to trip when reactor water level is 119.5" above top of the active fuel. This trip initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18.5" above the top of the active fuel. This trip activates the remainder of the CSCS subsystems, closes Group 7 valves, closes Main Steam Line Isolation Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1) and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For large breaks up to the complete circumferential break of a 22-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Paragraph 6.5.4 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. See Spec 3.7 for Valve Group. 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 6 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. Reference Subsection 14.6.2 of the FSAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steeam 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 14.5 of the FSAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN Mode is not required.

NOTES FOR TABLE 3.7-3

1. Isolation Signals are as follows:

Group 1:

The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor vessel low-low-low water level.
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line tunnel high temperature.
5. Low main steam line pressure at turbine inlet (run mode only).
6. Main condenser low vacuum.

Group 2:

The valves in Group 2 are closed upon any of the following conditions:

1. Reactor vessel low water level.
2. High drywell pressure.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 Introduction

The low-low set (LLS) relief logic modification for BWRs with Mark I Containments is designed to prevent multiple subsequent actuations of safety relief valves (SRV) which might normally be expected during a transient. This in turn will reduce or prevent the discharge loads on the containment and suppression pool structures resulting from subsequent SRV actuations. The discharge loads from subsequent actuations tend to be higher due to the condensation of trapped steam in the safety relief valve discharge line (SRVDL), which results in a higher water leg in the SRVDL, and hence, larger thrust loads on subsequent actuations.

The LLS design modification is an automatic SRV actuation system which, upon initiation, will assign preset opening and closing setpoints to two preselected SRVs. These setpoints are selected such that the LLS controlled SRVs will stay open longer, thus releasing more steam (energy) to the suppression pool, and hence more energy (and time) will be required for repressurization and subsequent SRV openings. The LLS increases the time between (or prevents) subsequent actuations sufficiently to allow the high water leg created from the initial SRV opening to return to or below its normal water level, thus, reducing thrust loads from subsequent actuations to within their design limits. In addition, since the LLS is designed to limit SRV subsequent actuations to one valve, torus loads will also be reduced.

The lower MSIV water level trip causes the MSIV closure actuation to be changed from a reactor water level two signal to a reactor water level one signal. This design modification maintains the main condenser availability for a longer time which allows more energy to be released to the main condenser and will result in a slower repressurization rate. The lower MSIV water level trip reduces isolations, SRV challenges and provides some benefit to SRV subsequent actuations.

In a letter dated, March 10, 1983 from Iowa Electric Light and Power Company (licensee) Technical Specification Changes were requested that incorporate these logic changes into the DAEC Mark I Containment design.

2.0 Evaluation

2.1 System Transient and Accident Performance and Overall Plant Safety Aspects

The General Electric (GE) generic evaluation submittals (Refs. 1 & 2) considered abnormal operational transients, design basis accidents and the anticipated transient without scram (ATWS) events to determine the impact of these design modifications on overall plant safety margins. The safety evaluation for abnormal operational transients included the following considerations to determine that the design modifications will not produce any adverse effects on safe plant systems operation and plant safety margins: (1) Reduction in Minimum Critical Power Ratio (MCPR); (2) Increase in Peak Pressure; (3) Increase in Radiation Release; (4) Cause for Equipment Damage; (5) Reduction in Plant Shutdown Capability; and (6) Decrease in Core Cooling Capability.

The limiting transient events, such as MSIV trip with flux scram, and turbine trip from high power without bypass were evaluated. General Electric concluded that the LLS will not affect the MCPR or peak pressure, because these conditions occur early in the transients before the reactor pressure response is affected by the LLS SRVs subsequent actuations.

The effects of the LLS on LOCAs were evaluated using the approved GE Appendix K evaluation models for the entire break spectrum. The evaluation showed that the LLS logic has no effect on the limiting Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), because the rapid reactor depressurization precludes the actuation of LLS SRVs during the design basis accident.

For ATWS events, General Electric determined that the LLS logic has no effects because this logic would not affect the reactor pressure until after the ATWS-associated short-term pressure transient is over.

It was also concluded that lowering the MSIV water level trip will not have any effects on the limiting MCPR,peak pressure and MAPLHGR during abnormal operational transients, LOCAs and ATWS events.

The impact of these design modifications on other effects such as, increase in radiation release, cause for equipment damage, reduction in plant shut-down capability, pool heat-up and decrease in core cooling capability were also considered in this generic evaluation.

The General Electric plant specific submittal (Ref. 3) identified that only non ADS-SRVs are used in the LLS logic system to reduce subsequent plant transients and that the LLS logic would extend SRV subsequent actuation time sufficiently to clear the water columns in the SRV discharge line.

Based on our review of both the General Electric generic and plant specific submittals, we have concluded that these design modifications are acceptable, since they will not adversely affect overall plant performance and safety considerations.

2.2 Low-Low Set (LLS) Circuitry

The LLS initiation circuitry consists of two independent and redundant channels, each of which controls power to a different SRV solenoid. There are six SRVs at Duane Arnold, four of which are actuated by the Automatic Depressurization System (ADS). The two non-ADS-SRVs (B21-F031B and F031F) will be used for the LLS function. Each of the two LLS controlled SRVs will open when their respective solenoid becomes energized. In order for either LLS channel to energize its solenoid, both an arming logic and an initiation logic must be satisfied. The arming logic is satisfied when any SRV has opened and reactor pressure (two-out-of-two logic for each LLS channel) has exceeded the high pressure setpoint (this setpoint is selected above the reactor protection system high reactor pressure scram setting to assure that a scram has occurred). Four separate reactor high pressure channels (two for each LLS channel) are used. These instrument channels are part of the existing nuclear boiler instrumentation that provide inputs to the reactor protection system (RPS). Inputs to the LLS arming circuitry from these channels are through isolation relays so that independence of the RPS is maintained. Once the arming logic for either LLS channel is satisfied, it is sealed in and annunciated in the control room, and remains sealed in until manually reset by the operator. In addition, the arming logic in either LLS channel will seal in the arming logic in the other LLS channel provided the reactor high pressure permissive in that channel is satisfied. Separation between LLS channels for this arming signal is provided by coil-contact separation. Initial SRV actuations are detected by pressure switches located in the SRV discharge lines. These pressure switches are set slightly above the normal pressure expected in the discharge line (35 psig).

Once armed, the LLS actuation/control logic uses nuclear boiler system reactor pressure instrumentation to control the LLS SRV solenoids, thus opening and closing these SRVs at their assigned LLS setpoints. This control logic remains in effect as long as the arming logic is sealed in.

Both LLS logic channels can be tested at power. Test status lights in the control room indicate when the arming logic and control logic relays have operated satisfactorily during testing. These test lights can also be used to verify proper operation of LLS seal in and reset circuits. The SRV discharge line pressure switches (three switches per SRV) and reactor pressure instrument channels (used both for the arming logic premissive and LLS SRV control) must be tested separately. Each LLS channel provides annunciation in the control room upon loss of power. Test switches are provided to verify operability of this power monitor function. Additional status lights are provided in the control room to indicate that the two-out-of-two reactor high pressure permissives have been satisfied. The licensee has indicated that the LLS logic will be tested consistent with the test interval for the ADS: this was not included in the proposed Technical Specification changes. However, the licensee in its letter dated April 21, 1983 committed to incorporate into the Technical Specifications within four months of the date of issuance of this amendment, surveillance testing requirements on the LLS logic circuitry consistent with the test interval for the ADS.

In addition, the licensee has initiated surveillance procedures consistent with the Technical Specification changes that will be submitted. We find that the proposed test frequency and commitment to incorporate this testing requirement into the Technical Specifications within four months of the date of issuance of this amendment acceptable.

Power to each of the two LLS channels may be provided by one of two separate sources. Normally, LLS channel "A" will be powered from 125 Vdc Distribution Panel No.1. If this supply should fail, power will automatically be supplied from 125 Vdc Distribution Panel No. 2. Similarly, LLS Channel "B" normally powered from Panel No. 2 will be automatically powered from Panel No. 1, if power from Panel No. 2 is interrupted. Each LLS channel contains circuitry (consisting of one relay and four associated contacts to disconnect the normal supply if it fails and connect the backup supply in its place) to perform this automatic transfer function. Since this automatic transfer feature is part of the existing protection system (ADS) circuitry and redundant protective devices (fuses and circuit breakers) are located between the automatic transfer and the safety buses, we find their design to be acceptable.

The LLS circuitry contains no channel or operating bypasses. The circuitry added for the LLS function is located in the ADS cabinet (in a back panel room next to the control room) and is separated in accordance with IEEE 384-1974. The components of the LLS system (including power supplies) are classified as class 1E. The LLS will remain operable in the event of loss of offsite power. LLS components located inside the drywell are qualified for the environmental condition associated with a small break LOCA.

Based on our review of the General Electric generic submittals, we have concluded that the LLS circuitry is designed in accordance with the requirements of IEEE Standard 279-1971 to perform its intended function given a single failure and that no single failure could be found which could cause a spurious SRV actuation. Therefore, we find the LLS logic as designed, acceptable.

2.3 Evaluation Summary

System Transient and Accident Performance Overall Plant Safety Aspects.

We find these design modifications, LLS logic and lowered MSIV water level trip, acceptable because they will not adversely affect plant performance or safety margins. These modifications are compatible with normal plant operation and other safety systems.

LLS Circuitry

Based on our review, we have determined that the LLS modification installed at Duane Arnold is designed to perform its intended function given a single failure. In addition, no single failure in the electrical circuits could be found which would cause a spurious SRV actuation. The LLS is designed in accordance with the requirements of IEEE Standard 279-1971 and therefore, is acceptable.

We have reviewed the system transients and accident performance and overall plant safety aspects and LLS circuitry related issues submitted, and we find, based on the above, that the design modifications including the necessary changes to the Technical Specifications are acceptable.

3.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

5.0 References

1. General Electric letter MFN-176-82 dated November 19, 1982.
2. General Electric Report NEDE-22223 dated September 1982.
3. General Electric Report NEDE-30021 dated January 1983.

Dated: April 29, 1983

Principal Contributor: K. Desai
R. Kendall

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 89 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, which revises the Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County Iowa. The amendment is effective as of its date of issuance.

This amendment to the Technical Specification revises the following: (1) the Group I Containment Isolation signal from reactor vessel low-low water level to reactor low-low-low water level, (2) the Group I isolation reactor pressure setpoint from 880 psig to 850 psig and (3) the low-low set logic to the two non-ADS safety relief valves.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 10, 1983 (2) Amendment No. 89 to License No. DPR-49, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commissions's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Cedar Rapids Public Library, 426 Third Avenue, S. E., Cedar Rapids, Iowa 52401. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

D. Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing