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TO: Mr. Edson G. Case

FROM: Iowa Elec. Light & Pwr. Co.
Cedar Rapids, Iowa
Lee Liu

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PLANT NAME: Duane Arnold (1-P)
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ENCLOSURE

License No. DPR-49 Appl for Amend: tech specs proposed change concerning revision RTS-88, RTS-93 and RTS-96 (Appendix A) and ETS-23 (Appendix B).......

(12-P)

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IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office
CEDAR RAPIDS, IOWA
October 26, 1977
IE-77-1974

LEE LIU
VICE PRESIDENT – ENGINEERING

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Case:

Transmitted herewith in accordance with the requirements of 10 CFR 50.59 and 50.90 is an application for amendment of DPR-49 (Appendices A and B to License) for the Duane Arnold Energy Center.

This application consisting of proposed Technical Specification changes RTS-88, RTS-93 and RTS-96 (Appendix A) and ETS-23 (Appendix B) has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee. This application does not involve a significant hazards consideration.

Three signed and notarized originals and 37 additional copies of this application are transmitted herewith. This application consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

By [signature]
Lee Liu
Vice President-Engineering

Subscribed and sworn to before me this 26 day of October, 1977.

Jean R. Smith
Notary Public in and for the State Iowa.

Jean R. Smith
NOTARY PUBLIC
STATE OF IOWA
Commission Expires
September 30, 1978

LL/OCS/D
Enc.
cc: D. Arnold
K. Meyer
R. Lowenstein
J. Keppler (NRC)
R. Clark (NRC)
L. Root
File A-117

773120101

PROPOSED CHANGE RTS-88 DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Table 3.2-G states, among other things that the trip level setting for Reactor High Pressure is "$\leq$1135 psig".

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Change the trip level setting to "$\leq$1120 psig".

III. Justification for Proposed Change

The design setpoint per the instrument data sheet for the Reactor High Pressure switch is 1135 psig. Adding the instrument tolerance of ±15 psig to the trip level setting as stated in Specification 1.0.2 could make the trip point as high as 1150 psig and still be within tolerance. Approval of this proposed change would eliminate that possibility.

This proposed change revises the setpoint of the subject instrument in the conservative direction. A trip of both recirculation pumps has been analyzed in the FSAR.

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action
1	Reactor High Pressure	$\leq$ 1120 psig	4	(2)
1	Reactor Low Water Level	$\geq$ -38.5 in. indicated level	4	(2)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for each operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector-switch in a mode other than the RUN Mode.

PROPOSED CHANGE RTS-93 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specifications 3.12.A, B and C state that: "If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (or LHGR or MCPR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (or LHGR or MCPR) is not returned to within the prescribed limits within two hours, the reactor shall be brought to the cold shutdown condition within 36 hours."

Specification 4.12.A states: "The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power."

Specification 4.12.B states: "The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power."

Specification 3.12.D (Reporting Requirements) states: "If any of the limiting values identified in Specifications 3.12.A, B or C are exceeded, a Reportable Occurrence report shall be submitted. If the corrected action is taken, as described, a thirty-day written report will meet the requirements of this specification."

Specification 4.12.D (Reporting Requirements) states: "The Limiting Conditions for Operation associated with monitoring the fuel rod operating conditions are required to be met at all times; i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR and MCPR. It is a requirement, as stated in Specifications 3.12.A, B and C, that if at any time during reactor power operation it is determined that the limiting values for MAPLHGR, LHGR or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicated that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable."

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

In Specifications 3.12.A, B and C described above, delete "two hours, the reactor shall be brought to the cold shutdown condition within 36 hours" and insert "4 hours, reduce reactor power to $\leq$ 25% of Rated Thermal Power within the next 4 hours".

To Specifications 4.12.A and B described above, add "and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2".

Delete the Limiting Conditions for Operation and Bases for Specification 3.12.D, Reporting Requirements.

III. Justification for Proposed Change

This change is proposed in order to clarify the Technical Specifications and bring them into agreement with the most recently approved Standardized Technical Specifications.

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

LIMITING CONDITION FOR OPERATION

3.12 CORE THERMAL LIMITS

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

During reactor power operation, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-2, 3.12-3, 3.12-4 and 3.12-5. When core flow is equal to or less than 70% of rated core flow, the MAPLHGR shall not exceed 95% of the limiting value shown in Figures 3.12-2 through 3.12-5. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 4 hours, reduce reactor power to $\leq$ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

SURVEILLANCE REQUIREMENT

4.12 CORE THERMAL LIMITS

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Linear Heat Generation Rate (LHGR)	B. Linear Heat Generation Rate (LHGR)
During reactor power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:	The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2.

$$LHGR_{max} \leq LHGR_d \left[1 - \left\{(\Delta P/P)_{max} (L/LT)\right\}\right]$$

$LHGR_d$ = Design LHGR = 18.5 KW/ft (7 x 7 array) or 13.4 KW/ft (8 x 8 array)

$(\Delta P/P)_{max}$ = Maximum power spiking penalty = 0.026

LT = Total core length - 12 feet

L = Axial position above bottom of core.

If at any time during reactor power operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 4 hours, reduce reactor power to $\leq$ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

During reactor power operations, MCPR shall be ≥ values as indicated in Table 3.12-2 at rated power and flow. If at any time during reactor power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within 4 hours, reduce reactor power to ≤ 25% of Rated Thermal Power within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be ≥ values as indicated in Table 3.12-2 times K_f, where K_f is as shown in Figure 3.12-1.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.2.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.12-1 are conservative for Duane Arnold operation because the operating limit MCPR of values as indicated in Table 3.12-2 is greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

PROPOSED CHANGE RTS-96 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specification 6.6, Reportable Occurrence Action, states as follows:

"1. Any reportable occurrence shall be reported immediately to the Chief Engineer and to the Vice President-Generation, and promptly reviewed by the Operations Committee.

2. The Operations Committee shall prepare a separate report for each reportable occurrence. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and also recommendations for appropriate action to prevent or reduce the probability of a recurrence.

3. Copies of all such reports shall be submitted to the Safety Committee for review and to the Vice President-Generation for review and approval of any recommendations."

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Change Specifications 6.6.1, 6.6.2 and 6.6.3 as indicated on the attached sheet.

III. Justification for Proposed Change

This change is proposed in order to bring the subject Technical Specifications concerning review of Reportable Occurrences into agreement with the Standardized Technical Specifications and Specifications 6.5.1.6.f and 6.5.2.7.g of the DAEC Technical Specifications.

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 Any reportable occurrence shall be reported immediately to the Chief Engineer and to the Vice President-Generation. In addition, any reportable occurrence requiring 24-hour notification to the Commission shall be promptly reviewed by the Operations Committee.

6.6.2 A separate report shall be prepared for each reportable occurrence. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and also recommendations for appropriate action to prevent or reduce the probability of a recurrence.

6.6.3 All reportable occurrence reports requiring 24-hour notification to the Commission shall be reviewed by the Operations Committee. In addition, copies of all such reports shall be submitted to the Safety Committee for review and to the Vice President-Generation for review and approval of any recommendations.

PROPOSED CHANGE ETS-23 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix B of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Table 3.3-1, Radioactive Liquid Waste Sampling and Analysis, provides, in part, as follows:

Sample Type	Sample Frequency	Sample Analysis	Sample Detectable Limit
Proportional Composite of Batches	Monthly	Sr^{89} Tritium Gross Alpha	5×10^{-8} µCi/ml 1×10^{-5} µCi/ml 5×10^{-7} µCi/ml
Proportional Composite of Batches	Quarterly	Sr^{90}	5×10^{-8} µCi/ml

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Delete Sr^{89} from the monthly analysis schedule and add it to the quarterly analysis schedule.

III. Justification for Proposed Change

The Sr^{89} and Sr^{90} detection method involves a complicated analysis which takes approximately a month to complete. Deletion of Sr^{89} from the monthly schedule and its addition to the quarterly schedule brings it into agreement with Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants".

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

TABLE 3.3-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS

Sample Type	Sample Frequency	Sample Analysis	Sample Detectable Limit (5) (2)
Waste Tank to be released	Each Batch	Gamma Scan (3)	5×10^{-7} µCi/ml
Proportional Composite of Batches	Monthly	Tritium Gross alpha	1×10^{-5} µCi/ml 5×10^{-7} µCi/ml (4)
Proportional Composite of Batches	Quarterly	Sr^{90}, Sr^{89}	5×10^{-8} µCi/ml
One Batch	Monthly	Dissolved noble gases	1×10^{-5} µCi/ml

Notes:

1. Certain mixtures of radionuclides may cause interference in the measurement of individual radionuclides at their detectable limit especially if other radionuclides are at much higher concentrations. Under these circumstances use of known ratios of radionuclides will be appropriate to calculate the levels of such radionuclides.

2. The above sample detectable limits are applicable to grab samples used to determine liquid waste release levels. Reported data shall reflect any improvement in detectable limits as such improvements are achieved.

3. Significant radionuclides are to be identified and where possible, quantitative values obtained.

4. Self absorption will result in a higher detectable limit for alpha counting.

5. Sample detectable limits are subject to revision. The values listed are believed to be attainable.