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Docket No. 50-331

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

The Commission has issued the enclosed Amendment No. 48 to Facility License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications and is in response to your applications dated July 7, 1977 (IE-77-1305), June 21, 1978 (IE-78-926) and July 26, 1978 (IE-78-1133) and to our generic letter of July 29, 1977 on respiratory protective equipment.

This amendment changes the Technical Specifications to: (1) revise the critical power ratio limits as justified by your reanalysis of the fuel loading error utilizing new methods which more accurately model the postulated event and which have been approved generically by the Commission, (2) reduce the set point on one of the six safety relief valves from 1090 psig to 1080 psig to minimize the potential for multiple actuation of these valves, (3) delete the requirement to change the names on the plant staffing organizational chart with a change in one of the incumbents, (4) delete reference to a hydraulic snubber that was on a line that has been removed from the plant and (5) deletes Section 6.9.1 on respiratory protection in accordance with our generic letter of July 29, 1977.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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GD

Enclosures:
1. Amendment No. 48
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
see next page

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OFFICE	ORB#3	ORB#3	OELD	ORB#3
SURNAME	SSheppard	RClark:acr	W D Paton	TIppolito
DATE	1/9/78	1/9/78	1/10/78	1/10/78

Mr. Duane Arnold

cc:

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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. 48
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensees) dated July 7, 1977, June 21, 1978 and July 25, 1978 comply 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.

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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. 48, 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

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: January 11, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 48

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 area of change.

Remove	Replace
1.2-1	1.2-1
3.6-41	3.6-41
3.12-9a	3.12-9a
3.12-11	3.12-11
Figure 6.2-1	Figure 6.2-1
6.9-1	6.9-1
6.9-2	6.9-2
6.9-3	6.9-3
6.9-4	6.9-4
6.9-5	6.9-5
6.9-6	6.9-6
6.9-7	6.9-7
6.9-8	6.9-8
6.9-9	6.9-9
6.9-10	6.9-10
6.9-11	6.9-11

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

1. The reactor vessel dome pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification:

1. The limiting safety system settings shall be as specified below:

Protective Action/Limiting Safety System Setting

A. Scram on Reactor Vessel high pressure

1035 psig

B. Relief valve settings
1080 psig ± 11 psi (1 valve)
1090 psig ± 11 psi (1 valve)
1100 psig ± 11 psi (2 valves)
1110 psig ± 11 psi (2 valves)

TABLE 4.6-3

HYDRAULIC SNUBBERS ACCESSIBLE DURING NORMAL OPERATION

Identification No.	System	Bldg. Location	Vendor Dwg. No.
GBC-1-SS-56	RHR Service Water	Reactor	6156
GBC-1-SS-57	RHR Service Water	Reactor	6157
GBC-2-SS-62	RHR Service Water	Reactor	6162
HCC-8-SS-11	Core Spray Pump Suction	Reactor	1787
HCC-8-SS-12	Core Spray Pump Suction	Reactor	1788/1
EBB-16-SS-231	RHR	Reactor	2084
EBB-16-SS-232(2 ea.)	RHR	Reactor	2085
EBB-16-SS-233	RHR	Reactor	2086
EBB-16-SS-234(2 ea.)	RHR	Reactor	2087
GBB-3-SS-235	RHR	Reactor	2088
GBB-3-SS-236	RHR	Reactor	2089
GBB-3-SS-237	RHR	Reactor	2090
GBB-3-SS-238	RHR	Reactor	2091
GLE-8-SS-239	RHR	Reactor	2092
GLE-8-SS-240	RHR	Reactor	2093
GBB-10-SS-241	RHR	Reactor	2094
GBB-10-SS-242(2 ea.)	RHR	Reactor	2095
GBB-10-SS-243	RHR	Reactor	2096
GBB-4-SS-210	RHR	Reactor	2063
GBB-4-SS-211	RHR	Reactor	2064
GBB-4-SS-212	RHR	Reactor	2065
GBB-4-SS-213	RHR	Reactor	2066
GBB-16-SS-214	RHR	Reactor	2067
GBB-5-SS-215	RHR	Reactor	2068
GBB-4-SS-216 (2 ea.)	RHR	Reactor	2069
GBB-4-SS-217 (2 ea.)	RHR	Reactor	2070
HBB-21-SS-218(2 ea.)	RHR	Reactor	2071
HBB-23-SS-219	RHR	Reactor	2072
HBB-23-SS-220	RHR	Reactor	2073
HBB-24-SS-221	RHR	Reactor	2074
HBB-24-SS-222	RHR	Reactor	2075
GBB-7-SS-223	RHR	Reactor	2076
GBB-7-SS-224	RHR	Reactor	2077
GBB-6-SS-225	RHR	Reactor	2078
GBB-6-SS-226	RHR	Reactor	2079
HBB-24-SS-227(2 ea.)	RHR	Reactor	2080
HBB-24-SS-228(2 ea.)	RHR	Reactor	2081
HBB-24-SS-229	RHR	Reactor	2082
HBB-29-SS-199	RHR	Reactor	2052
HBB-30-SS-205	RHR	Reactor	2058
HBB-30-SS-206	RHR	Reactor	2059
HBB-30-SS-245	RHR	Reactor	2098

(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 3.12-2

MCPR LIMITS

Fuel Type	Exposure Remaining to End of Cycle B.O.C. to >2000 MWD/T	≤ 2000 MWD/T to >1000 MWD/T	≤ 1000 MWD/T to > 500 MWD/T	≤500 MWD/T to E.O.C.
7 x 7	1.22	1.22	1.26	1.30
8 x 8	1.21	1.29	1.34	1.38

Amendment No. ~~42~~, 48

3.12 REFERENCES

1. Duane Arnold Energy Center Loss-of-Coolant Accident Analysis Report, NEDO-21082-02-1A, Class I, July 1977, Appendix A.

2. General Electric BWR Generic Reload Application for 8 x 8 Fuel, NEDO-20360, Revision 1, November 1975.

3. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-19735, August 1973.

4. Supplement 1 to Technical Reports on Densifications of General Electric Reactor Fuels, December 14, 1973 (AEC Regulatory Staff).

5. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

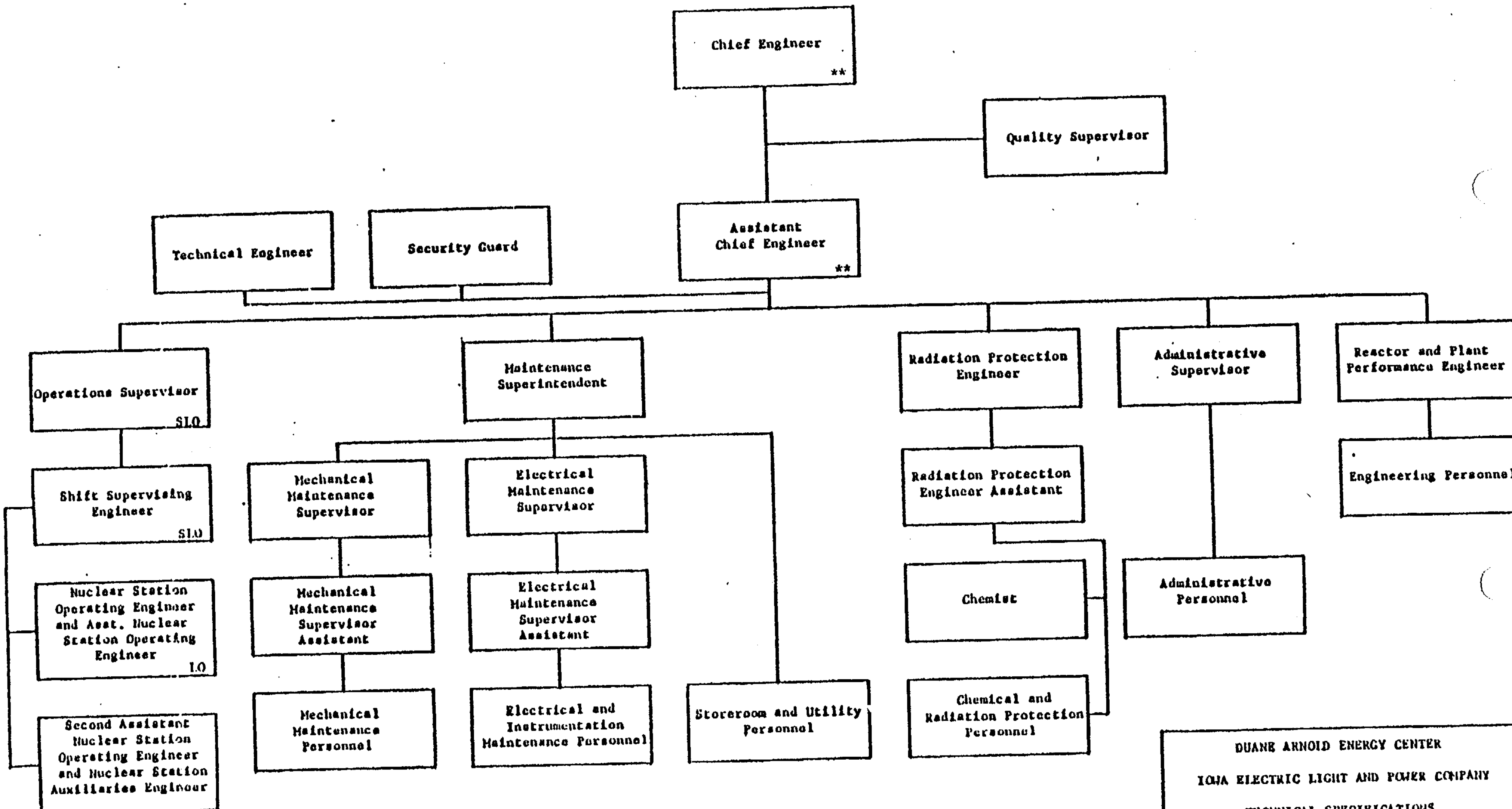
6. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

7. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR Part 50, Appendix K, NEDE-20566 (Draft), August 1974.

8. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, NEDO-24087, 77 NED 359, Class 1, December 1977.

9. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

10. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 5: Revised Operating Limits for Loss of Feedwater Heating, NEDO-24087-5.



DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

TECHNICAL SPECIFICATIONS

DAEC NUCLEAR PLANT STAFFING

FIGURE 6.2-1

SLO - Senior Licensed Operator.

LO - Licensed Operator.

** - Chief or Assistant Chief to Meet ANSI N.18.1-1971 License Requirements.

6.9 RADIOLOGICAL PROCEDURES

6.9.1 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 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

1.1 Revised Fuel Loading Error and Loss of Feedwater Heating Analyses

By letter dated June 21, 1978, Iowa Electric Light and Power Company (Iowa Electric or the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The changes would (1) revise the MCPR limits in Table 3.12-2 (page 3.12-9a) for the 7x7 and 8x8 fuel in the core for four fuel exposure ranges during the current fuel Cycle (Cycle No. 4) and (2) add references 9 and 10 to page 3.12-11. To support the request, the licensee submitted a revised plant specific analysis for the fuel loading error utilizing methods which more accurately model the postulated event. These methods have recently been accepted for generic application to General Electric boiling water reactor fuels. The licensee also submitted a revised analysis for the loss of feedwater heating based on a critical power ratio (CPR) for four specific fuel exposure ranges rather than operate throughout the fuel cycle with the CPR corresponding to the conservative, end-of-cycle conditions. The minimum critical power ratio (MCPR) needs to be the most conservative (highest) at the very end of core (EOC) life when the control rods are the most fully withdrawn. While it is ultraconservative to operate with the EOC MCPR throughout the entire fuel cycle, this places unnecessary restrictions on plant operations. Therefore, the licensee has proposed (and submitted an analysis to justify) a MCPR of 1.22 from the beginning of the fuel cycle (B.O.C.) up to the

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last 2000 megawatt days per ton of fuel exposure (>2000 MWD/T), a MCPR of 1.22 for $\leq$ 2000 MWD/T to >1000 MWD/T, a MCPR of 1.26 for $\leq$ 1000 MWD/T to >500 MWD/T and a MCPR of 1.30 for $\leq$500 MWD/T to EOC. The Duane Arnold facility operated with similar fuel exposure dependent MCPR limits in the previous fuel cycle (cycle 3). (See Amendment No. 40 issued December 30, 1977).

1.2 Routine Reporting on Plant Staff

On December 2, 1975, we issued Amendment No. 12 to Facility Operating License No. DPR-49; the Amendment changed the DAEC Technical Specifications to bring all plant reporting requirements into accordance with Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications". One of the changes in Section 6.11.1 of the DAEC Technical Specifications ("Routine Reports") was to delete the requirement to report the name of the individual whenever there was a change in incumbents. Inadvertently, however, a notation was left on figure 6.2-1 that refers to this deleted requirement in Section 6.11.1. Specifically, figure 6.2-1, entitled, "DAEC Nuclear Plant Staffing" depicts the organizational structure for the plant. The figure contains an asterisk by the positions of Chief Engineer, Assistant Chief Engineer, Operations Supervisor, Shift Supervising Engineer, Maintenance Superintendent, Mechanical Maintenance Supervisor, Electrical Maintenance Supervisor, Radiation Protection Engineer and Reactor and Plant Performance Engineer with a notation that the asterisk refers to, "Routine Reporting Requirements on Change in Incumbents (Ref. Spec. 6.11.1)".

By letter dated July 7, 1977, the licensee requested to delete the asterisk from the positions described above in figure 6.2-1 and to delete the notation to which the asterisk refers. As stated above, this change is an omission that should have been made in Amendment No. 12. The change has no safety significance. Deletion of this reporting requirement does not adversely affect the quality of DAEC supervisory personnel since the qualifications of plant members and replacements must meet or exceed the qualifications referenced for comparable positions in ANSI N18.1-1971, as noted in Specification 6.3.1.

1.3 Deletion of Snubber

On November 26, 1976, we issued Amendment No. 24 to Facility License No. DPR-49. This amendment changed the DAEC Technical Specifications to add new Sections 3.6.H and 4.6.H which provided Limiting Conditions for Operation and Surveillance Requirements associated with safety related shock suppressors (snubbers). The change included two new tables - Tables 4.6-3 and 4.6-4. Table 4.6-3 listed 73 hydraulic snubbers that are accessible for inspection during normal operation, along with the system on which each is installed and their location. Table 4.6-4 lists 83 snubbers which are inaccessible during normal operation by identification number, system on which installed and location. When Amendment No. 24 was issued, Table 4.6-3 inadvertently included one snubber on the Reactor Water Clean-up Return Line to the Feedwater Line (Snubber DCA-14-55-73) which does not exist in the plant anymore.

When the DAEC was constructed, the Reactor Water Clean-Up Return Line to the Feedwater Line was routed through the MG Set room and contained five hydraulic snubbers. However, by being routed through the MG Set room, the line went outside of the secondary containment. When the line was rerouted to keep it inside secondary containment, analysis showed that the five hydraulic snubbers were no longer required. The five hydraulic snubbers were, therefore, not used. One of the snubbers, DCA-14-SS-73, was inadvertently left on the list which was submitted by the licensee with the proposed hydraulic snubber Technical Specifications in May 1976. To correct this error, the licensee in their letter of July 7, 1977 proposed to delete snubber DCA-14-SS-73 from Table 4.6-3. We agree that the snubber should be deleted and that there is no safety significance to the change.

1.4 Reduction of Set-Point for One Safety Relief Valve

By letter dated July 25, 1978, the licensee requested to reduce the set point of one of the six safety relief valves from 1090 psig to 1080 psig. At present, two of the safety relief valves (SRV) are set to relieve at 1090 ± 11 psi, two SRVs are set at 1100 ± 11 psi and two SRVs are set at 1110 ± 11 psi. The purpose of the change is to prevent simultaneous second actuation of the two low set point SRVs, thus minimizing stresses on the torus from multiple sequential actuations of relief valves.

1.5 Respiratory Protection Program

On July 29, 1977, the Commission issued a generic letter addressed to the licensee with respect to the respiratory protection program described in Section 6.9.1-1 through 6.9.1-3 of the Technical Specifications for the Duane Arnold Energy Center. The letter called attention to the fact that on November 29, 1976, the Commission published in the Federal Register an amended Section 20.103 of 10 CFR 20, which became effective on December 29, 1976. One effect of this revision is that in order to receive credit for limiting the inhalation of airborne radioactive material, respiratory protective equipment must be used as stipulated in Regulatory Guide 8.15. Another requirement of the amended regulation is that licensees authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976, must bring the use of their respiratory protective equipment into conformance with Regulatory Guide 8.15 by December 29, 1977.

The Duane Arnold Technical Specifications anticipated the above Amendment to Section 20.103 of 10 CFR 20; section 6.9.1-3 contains a revocation provision stating that "these specifications with respect to the provisions of 20.103 shall be superseded if, in the future, 10 CFR 20.103 shall assign protection factors for respiratory and other protective equipment". In our letter of July 29, 1977, we advised Iowa Electric that "In view of the provisions of Section 6.9.1 of your Technical Specifications, which require conformance with 10 CFR 20, the fact that Section 20.103 no longer requires specific authorization to employ respiratory protective equipment, and the revocation provisions of subsection 6.9.1-3, we conclude that the necessary amendment to your facility's Technical Specifications can be effected by merely deleting Section 6.9.1-1 through 6.9.1-3".

In the letter, we also advised Iowa Electric that "Based on the revocation provision of your current specification on respiratory protection and in the absence of prior written objection from you, we will include deletion of this specification in an amendment of your Technical Specifications approved after December 28, 1977. No response to this letter is required".

This amendment will delete Sections 6.9.1-1, 6.9.1-2 and 6.9.1-3 in accordance with our letter of July 29, 1977. There is no safety significance since these sections are in effect revoked by 10 CFR 20.103.

2.0 <u>Evaluation</u>

2.1 <u>Revised Fuel Loading Error and Loss of Feedwater Heating</u>

The Present OLMCPR and the limiting transients are listed in Table 1. The fuel loading error (FLE) is limiting for the 7x7 fuel until 500 Megawatt Days per Tonne (MWD/T) before end of cycle (EOC) and for the 8x8 fuel until after 2000 MWD/T before EOC.

The licensee has presented a reanalysis of the FLE.[2] This reanalysis uses the methods of references 3 and 4. These methods have been reviewed and found acceptable by the staff.[5] For the fuel misorientation analysis, the actual fuel bundle tilt is modelled rather than the more conservative modelling assumption that the fuel bundle is not tilted. For the fuel mis-location analysis, the fuel bundle minimum critical power ratio (MCPR) is modelled as a function of the core reference loading, rod pattern, and burnup rather than the modelling assumption that the mislocated bundle is at the core OLMCPR. These methods more accurately model potential FLE events. A more detailed discussion of these methods is presented in the staff generic safety evaluations, reference 5.

The reanalysis demonstated that the limiting FLE is within the acceptance criteria as specified in reference 6. Thus, the FLE is no longer the most limiting transient for the OLMCPR evaluation. The OLMCPRs may be revised to the values as proposed in the previous analysis of abnormal operational transients (reference 7). The reference 7 analysis has been reviewed and found acceptable for DAEC cycle 4 operation.[6]

The licensee has presented an additional transient analysis to justify a further change to DAEC OLMCPRs [8] from those of reference 7. With the FLE re-analysis, the limiting transient for the 8x8 fuel in the exposure range from beginning of cycle (BOC) to EOC minus 2000 MWD/T is the loss of feedwater heat-ing. The analysis in Reference 7 of the loss of feedwater heating assumed an initial critical power ratio (CPR) of 1.38 (the EOC value). This resulted in a change in critical power ratio (ΔCPR) of -0.16, which was conservatively applied throughout the cycle. The revised analysis of the loss of feedwater heating (reference 8) was performed with a lower CPR value which corresponds to the condition from BOC to EOC - 2000 MWD/T. The ΔCPR for this analysis was reduced to -0.15. Therefore, the OLMCPR for 8x8 fuel from BOC to EOC -2000 MWD/T may be reduced to 1.21 from 1.22. The methods were the same as previously reviewed and found acceptable in reference 6.

These analyses justify the adjustment to the OLMCPR as listed in Table 2. The staff has reviewed these proposed changes to the OLMCPR technical speci-fication. The proposal utilizes approved methods and analyses. The pro-posed change does not result in violation of the safety limit MCPR for the most limiting transients. Based on the presented analyses and the use of previously accepted methods, the proposed operating MCPR limits are acceptable for application to DAEC cycle 4.

2.2 Routine Reporting on Plant Staff

As discussed in Section 1.2 above, there is no safety significance to this change. The change is necessary to correct an error in the Technical Specifications.

2.3 Deletion of Snubber

As discussed in Section 1.3 above, there is no safety significance to this change. The change is necessary to correct an error in Table 4.6-3 of the Technical Specifications.

2.4 Reduction of Set-Point for One Safety Relief Valve

In conjunction with the Mark I Containment Short Term Program, Iowa Electric performed an evaluation of the potential effects of multiple sequential actuations of relief valves on the torus and torus support system of the Duane Arnold Energy Center. The licensee's letter of July 25, 1978 summarizes the applicable information developed as part of the Mark I Containment Short Term Program. This letter, which was in response to our letter of March 20, 1978, provided a plant unique reevaluation of the effects of an isolation event with respect to potential multiple sequential actuations of the safety relief valves. The analyses indicates that if the set point on one of the two relief valves with the lowest set points (namely 1090 psig) is reduced to 1080 psig, no more than two safety relief valves will experience a second actuation. The letter also provided an analysis of the stresses on the torus support system if the two relief valves with the lowest set points experience a simultaneous second actuation.

We have evaluated the licensee's submittal and conclude that: (1) the proposed change in the set point for one of the relief valves will reduce the potential for multiple sequential actuation of the relief valves if an isolation event occurs and (2) the strength ratio for the torus support system will not exceed 0.5 during an isolation event. From the standpoint of system thermal-hydraulic parameters a decrease in a relief valve set point decreases the simmer margin and increases the minimum critical power ratio (MCPR) margin. Thus, a decrease in the set point increases the conservatism in the present analyses and from the safety standpoint is of no concern.

2.5 Respiratory Protection Program

As discussed in Section 1.5 above, Sections 6.9.1-1 through 6.9.1-3 of the Technical Specifications are being deleted as the Commission's initiative since they have been revoked by 10 CFR 20.103.

3.0 Environmental Considerations

We have determined that this 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 this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 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 accidents previously considered and does not involve a significant decrease in a safety margin, 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.

Dated: January 11, 1979

TABLE 1

CURRENT OPERATING MCPR LIMITS AND LIMITING TRANSIENTS

Fuel Type	Exposure Remaining to End of Cycle			
	BOC to >2000 MWD/T	<2000 MWD/T to >1000 MWD/T	<1000 MWD/T to >500 MWD/T	<500 MWD/T to EOC
7x7 (limiting transient)	1.27 (FLE)	1.27 (FLE)	1.27 (FLE)	1.30 (LR W/o B)
8x8 (limiting transient)	1.27 (FLE)	1.29 (LR W/o B)	1.34 (LR W/o B)	1.38 (LR W/oBO

FLE - Fuel Loading Error

LR W/o B - Load Rejection without Bypass

TABLE 2

PROPOSED OPERATING MCPR LIMITS AND LIMITING TRANSIENTS

	Exposure Remaining to End of Cycle			
Fuel type	BOC to >2000 MWD/T	<2000 MWD/T to >1000 MWD/T	<1000 MWD/T to >500 MWD/T	<500 MWD/T to EOC
7X7 (limiting transient)	1.22 (RWE)	1.22 (LR w/o B and RWE)	1.26 (LR w/o B)	1.30 (LR w/o B)
8X8 (limiting transient)	1.21 (LFH)	1.29 LR w/o B)	1.34 (LR w/o B)	1.38 (LR w/o B)

RWE - Rod Withdrawal Error

LFH - Loss of Feedwater Heating

LR w/o B - Load Rejection without Bypass

Reference

1. Letter from L. Liu (Iowa Electric Light and Power Company) to E. Case (NRC), IE-78-926, June 21, 1978.

2. Boiling Water Reactor Reload-3 Licensing Amendment for Duane Arnold Energy Center, Supplement 2: Revised Fuel Loading Accident Analysis, NEDO-24087-2.

3. Letter from R. E. Engel (General Electric Company) to D. G. Eisenhut (NRC), MFN-219-77, June 1, 1977.

4. Letter from R. E. Engel (GE) to D. G. Eisenhut (NRC), MFN-457-77, November 30, 1977.

5. Letter from D. G. Eisenhut (NRC) to R. E. Engel (GE), May 8, 1978.

6. Letter from G. Lear (NRC) to L. Liu (Iowa Electric Light and Power Company) dated April 20, 1978.

7. Boiling Water Reactor, Reload 3, License Amendment for Duane Arnold Energy Center, NEDO-24087, December 1977.

8. Boiling Water Reactor Reload - 3 Licensing Amendments for Duane Arnold Energy Center, Supplement 5: Revised Operating Limits for Loss of Feedwater Heating, NEDO -24987-5.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 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 revised Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County, Iowa. The amendment is effective as of the date of issuance.

The amendment consists of changes to the Technical Specifications to: (1) revise the operating limit-minimum critical power ratio restrictions as a result of reanalysis of the postulated fuel loading error and loss of feedwater heating, (2) reduce the set point on one of the six safety relief valves from 1090 psig to 1080 psig to reduce potential stresses on the torus support system, (3) delete the requirement to change the names on the plant staffing organization chart with a change in one of the incumbents, (4) delete reference to a hydraulic snubber that has been removed from the plant and (5) delete the section on respiratory protection since these requirements have been superseded by 10 CFR 20.103.

The applications for the amendment comply 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 Section 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 applications for amendments dated July 7, 1977, June 21, 1978 and July 25, 1978 and the Commission's generic letter to Iowa Electric Light and Power Company of July 29, 1977 on respiratory protective equipment (2) Amendment No. 48 to License No. DPR-49, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission'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 Operating Reactors.

Dated at Bethesda, Maryland, this 11 day of January 1979.

Tom Ippolito

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Division of Operating Reactors