

May 28, 1985

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. 121 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 January 11, 1985.

This amendment revises the Technical Specifications to incorporate the updated reactor pressure vessel pressure-temperature limits, minimum boltup temperature, and reactor vessel capsule withdrawal schedule as required by 10 CFR 50, Appendices G and H.

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. 121 to License No. DPR-49
- 2. Safety Evaluation

cc w/enclosures:
See next page

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

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 121
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Iowa Electric Light and Power Company, et al, dated January 11, 1985, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121, 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: May 28, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise the Appendix "A" Technical Specifications by removing the current pages and inserting the revised pages listed below. The revised areas are identified by vertical lines.

AFFECTED PAGES

3.6-1
3.6-2
3.6-16
3.6-17
3.6-18
3.6-40
3.6-41

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6 PRIMARY SYSTEM BOUNDARY	4.6 PRIMARY SYSTEM BOUNDARY
<u>Applicability:</u>	<u>Applicability:</u>
Applies to the operating status of the reactor coolant system.	Applies to the periodic examination and testing requirements for the reactor cooling system.
<u>Objective:</u>	<u>Objective:</u>
To assure the integrity and safe operation of the reactor coolant system.	To determine the condition of the reactor coolant system and the operation of the safety devices related to it.
<u>Specification:</u>	<u>Specification:</u>
A. <u>Thermal and Pressurization Limitations</u>	A. <u>Thermal and Pressurization Limitations</u>
1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.	1. During heatups and cooldowns, the following temperatures shall be logged at least every 15 minutes until 3 consecutive readings at each given location are within 5°F.
	a. Reactor vessel shell adjacent to shell flange.
	b. Reactor vessel bottom drain.
	c. Recirculation loops A and B.
	d. Reactor vessel bottom head temperature.
2. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6-1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. Figure 3.6-1 is effective through 12 effective full power years. At least six months prior to 12 effective full power years new curves will be submitted.	2. Reactor vessel metal temperature at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to shell flange shall be recorded at least every 15 minutes during inservice hydrostatic or leak testing when the vessel pressure is >312 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
<p>3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 74°F.</p>	<p>Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 MeV neutrons were installed in the reactor vessel adjacent to the vessel wall at the core midplane level at the start of operation. The specimens and sample program shall conform to ASTM E 185-66 to the degree discussed in the FSAR.</p>
<p>4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50°F of each other.</p>	<p>Samples shall be withdrawn at 6 and 15 effective full power years in accordance with 10 CFR 50, Appendix H. Neutron flux wires were installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires were removed and tested during the second refueling outage to experimentally verify the calculated values of neutron fluence at one-fourth of the beltline shell thickness that are used to determine the NDTT shift. Results of the flux wire test and the effects of copper and phosphorus on the beltline are reflected in Figure 3.6-1.</p>
<p>5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.</p>	<p>3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flang shall be permanently recorded.</p> <p>4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.</p> <p>5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.</p>

3.6.A and 4.6.A BASES:

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel meet the requirements of 10 CFR 50, Appendix G, revised May 1983. (2)

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time period for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

Chicago Bridge and Iron Company performed detailed stress analysis as shown in the Updated FSAR Appendix 5A, "Site Assembly of the Reactor Vessel." This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling rate applied continuously over a 100°F to

550°F range. The differential is due to the sluggish temperature response to the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The operating limits in Figure 3.6-1 are derived in accordance with 10 CFR 50 Appendix G, May 1983 and Appendix G of the ASME Code. Conditions in three regions influence the curves: the closure flange region, the non-beltline region which includes most nozzles and discontinuities, and the beltline region which is irradiated with fluence above 10^{17} n/cm² during the vessel operating life. Irradiation causes an increase in the nil-ductility temperature (RT_{NDT}) of the beltline materials, possibly to the point where the beltline region impacts the pressure-temperature limits for the vessel. However, for Figure 3.6-1, effective to 12 EFPY, the beltline which has an

RT_{NDT} of 40°F is less limiting than the non-beltline regions which generally experience higher stresses at nozzles and discontinuities. The limiting RT_{NDT} of 58°F for the Standby Liquid Control Nozzle (N10) is the highest RT_{NDT} of any component in the non-beltline region.

The closure flange region, with RT_{NDT} = 14°F, has a bolt preload and minimum operating temperature of 74°F. This exceeds original requirements of the ASME Code (Winter 1967 Addendum) and provides extra margin relative to current ASME Code requirements.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested according to 10 CFR 50 Appendix H. Results of these analyses will be used to adjust Figure 3.6-1 as appropriate.

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of age and operating conditions. Due to implementation of the snubber service life monitoring program after several years of plant operation, the historical records to date may be incomplete.

The records will be developed from engineering data available. If actual installation data is not available, the service life will be assumed to commence with the initial criticality of the plant. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.6 and 4.6 References

- 1) General Electric Company, Low-Low Set Relief Logic System and Lower MSIV Water Level Trip for the Duane Arnold Energy Center, NEDE-30021-P, January, 1983.
- 2) General Electric Company, Duane Arnold Energy Center Reactor Pressure Vessel Fracture Toughness Analysis to 10 CFR 50 Appendix G, May 1983, NEDC-30839, December, 1984.

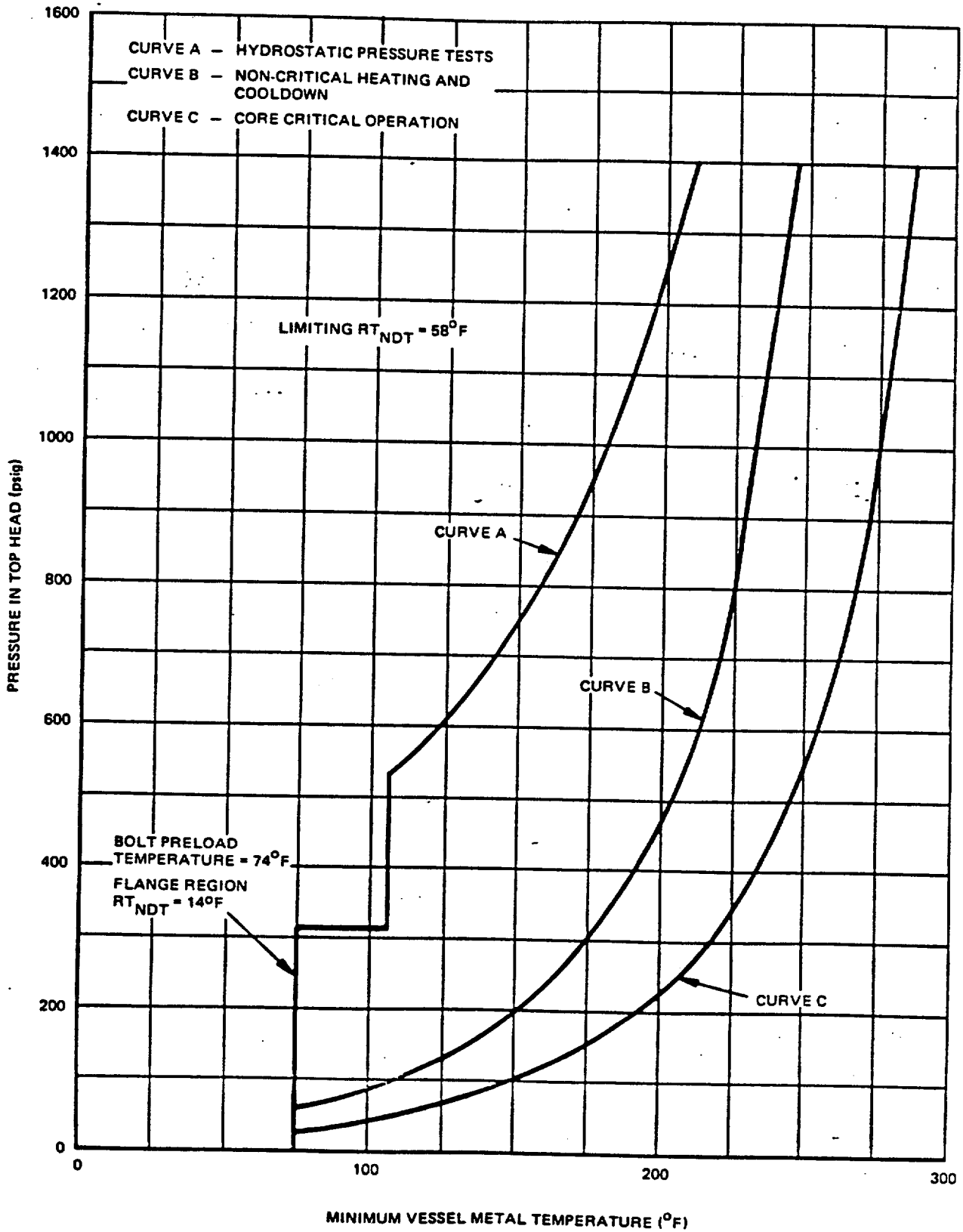


Figure 3.6-1. Pressure versus Minimum Temperature Valid to Twelve Full Power Years, per Appendix G of 10CFR50



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 121 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

In a letter dated January 11, 1985, the Iowa Electric Light and Power Company (the licensee/IELP) requested an amendment to the Technical Specifications for the Duane Arnold Energy Center (DAEC). The amendment proposes (1) to increase the effectiveness of the pressure-temperature operating limits for DAEC to 12 effective full power years (EFPY), (2) to adjust the minimum vessel head bolting stud temperature, and (3) to revise the reactor vessel surveillance capsule withdrawal schedule. Submitted in the January 11, 1985 letter was a General Electric (GE) Report NEDC-30839, titled "Duane Arnold Energy Center Reactor Pressure Vessel Fracture Toughness Analysis to 10 CFR 50 Appendix G, May 1983," which contained the analysis in support of the proposal.

2.0 EVALUATION

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial reference temperature defined in the ASME Code, Section XI, (RT_{NDT}) for the limiting materials in the beltline, nozzle discontinuities, and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The DAEC reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness to determine the initial RT_{NDT} for each reactor vessel material. Hence, the initial RT_{NDT} for materials in the closure flange, nozzle discontinuities, and beltline regions of the DAEC reactor vessel could not be determined in accordance with the test requirements of the ASME Code. Therefore, the initial RT_{NDT} for these materials must be estimated from material test data from other similar materials used for fabrication of the reactor vessels in the nuclear industry.

The licensee, in developing pressure-temperature limit curves, has estimated the initial RT_{NDT} for the limiting closure flange (Shell No. 4)

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material as 14°F, the initial RT_{NDT} for the limiting nozzle discontinuity (Standby Liquid Control Nozzle N10) material as 58°F, and the initial RT_{NDT} for the limiting beltline (Plate Course No. 1) material as 40°F. These values were determined using the available drop weight and Charpy V-notch test data and the GE Analytical Procedure Y1006A006, which is explained in GE Report NEDC-30839. This method of analysis was developed by GE from an evaluation of nuclear industry reactor vessel materials. The procedure was previously approved by the staff in its Safety Evaluation for LaSalle Unit Nos. 1 and 2 (NUREG-0519, March 1, 1981 and Supplement No. 1, June 1981).

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the empirical relationship documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and phosphorus in the limiting beltline material. According to GE Report NEDC-30839, the limiting beltline material is Plate B0673-1, which has .012 percent phosphorus and .15 percent copper. Using flux wire measurements, a two-dimensional computer program (DOT) to solve Boltzman transport equation, and a one-dimensional computer code (SN1D) to calculate radial flux distributions, GE Report NEDC-30839 indicates that the peak neutron fluence at the 1/4 T location at the end of life of DAEC is 4.4×10^{18} n/cm² (E greater than 1MeV).

The staff used the method of calculating pressure-temperature limits in US NRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to determine the amount of time that the proposed pressure-temperature limits are effective. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50 for 12 EFPY and may be incorporated into the plant's Technical Specifications.

The minimum vessel head bolting stud temperature must comply with Appendix G, 10 CFR 50. Appendix G, 10 CFR 50 requires that this minimum boltup temperature comply with the requirements in Appendix G of Section III, Division 1 of the ASME Code. Based on the initial RT_{NDT} for the limiting closure flange material, the proposed minimum bolt preload temperature meets these requirements, and is acceptable.

The DAEC reactor vessel material surveillance program must comply with Appendix H, 10 CFR 50. This requires that the capsule withdrawal schedule comply with ASTM E 185-82. Based on the amount of predicted neutron irradiation damage, the proposed capsule withdrawal schedule complies with these requirements. Hence, it may be incorporated into the plant's Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20.

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: B. Elliot and J. Kim

Dated: May 28, 1985