June 20, 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. 124 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 June 14, 1985.

This amendment revises the Amendment No. 121 effective date from May 28, 1985 to July 31, 1985.

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

cc w/enclosures:
See next page

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SNorris:ajs	MThadani	DVassallo		GLainas
06/18/85	06/18/85	06/18/85	06/18/85	06/19/85

no changes in STS & license.

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PDR ADOCK 05000331
P PDR

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

cc:

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

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

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

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

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

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

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

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. 124
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 June 14, 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 revising the effective date of Amendment No. 121 from May 28, 1985 to July 31, 1985.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

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

Date of Issuance: June 20, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 124

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.

LIST OF AFFECTED PAGES

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

* Remove these pages from the Technical Specifications after July 31, 1985.

** Effective on July 31, 1985.

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed $100^{o}F/hr$ when averaged over a one-hour period.

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 Fig. 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years new curves will be submitted.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

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 $5^{o}F$.

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. 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.

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

LIMITING CONDITION FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

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.

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.

SURVEILLANCE REQUIREMENT

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

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. 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
	Test specimens of the reactor vessel base, weld and heat affected zone metal subjected to the highest fluence of greater than 1 MeV neutrons shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-66 or ASTM E-185-70 to the degree discussed in Section 5.3.1.6 of the Updated FSAR. Samples shall be withdrawn at one-fourth and three-fourths service life in accordance with 10 CFR 50, Appendix H.
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 100°F.	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 flange shall be permanently recorded.
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.	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.
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.	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.

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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.

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.

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.

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.

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.

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.

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.

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.

Amendment No. ~~114~~, ~~121~~ 124

EFFECTIVE ON JULY 31, 1985.

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.

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

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

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. (3)

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 nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (> 1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel base metal above the initial value. Extensive tests have established the magnitude of changes as a function of the integrated neutron exposure.

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

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

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/cm^2 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

EFFECTIVE ON JULY 31, 1985.

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.

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

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. 113

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 Boiling Water Reactor Increased Safety/Relief Valve Simmer Margin Analysis for Duane Arnold Energy Center," NEDC-30606, May, 1984. 115

Correction letter of 4-29-85

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

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.

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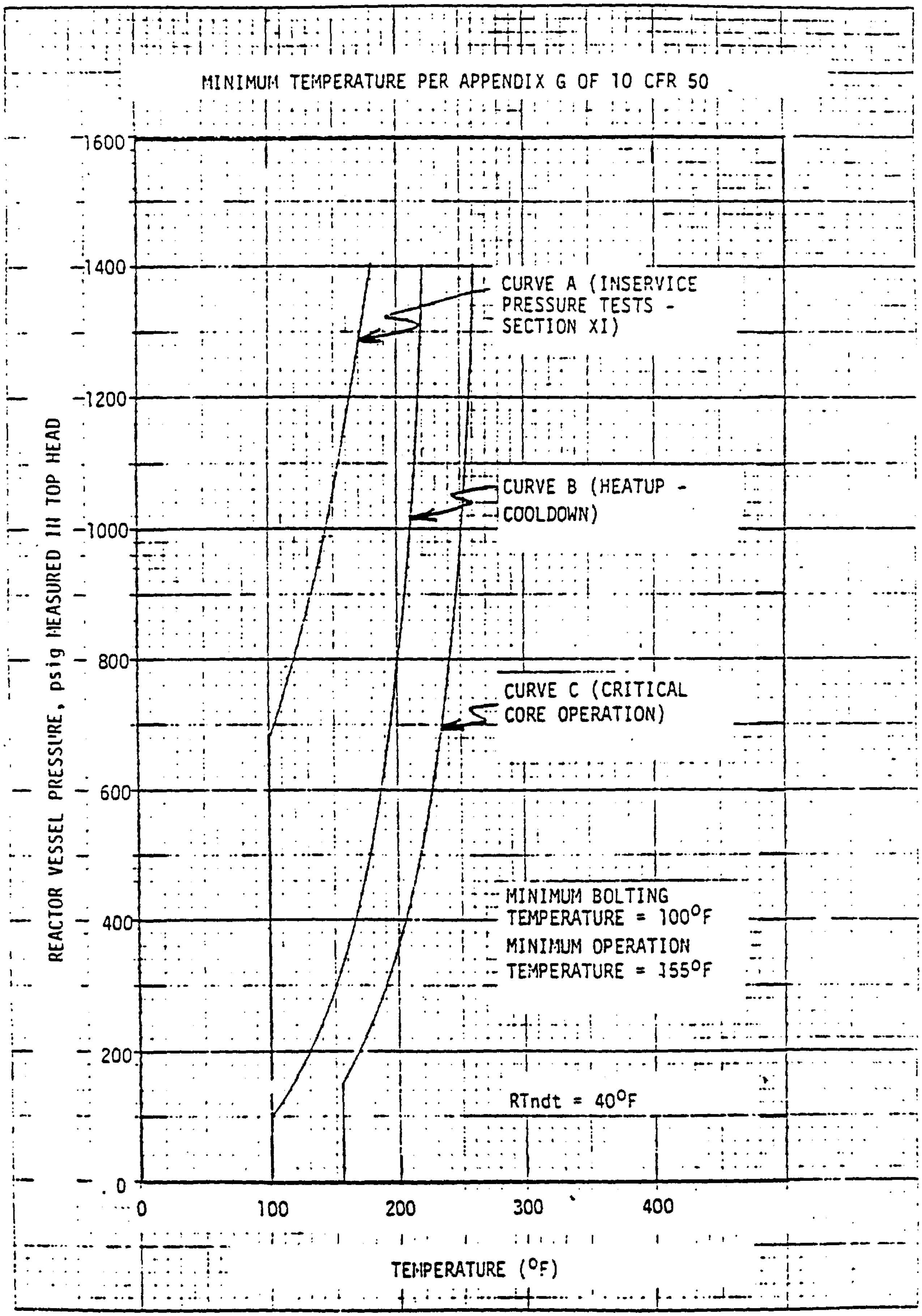
2) "General Electric Boiling Water Reactor Increased Safety/Relief Valve Simmer Margin Analysis for Duane Arnold Energy Center," NEDC-30606, May, 1984.

3) 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



Amendment No. ~~113~~, 124

REMOVE FROM THE TECHNICAL SPECIFICATIONS ON JULY 31, 1985.

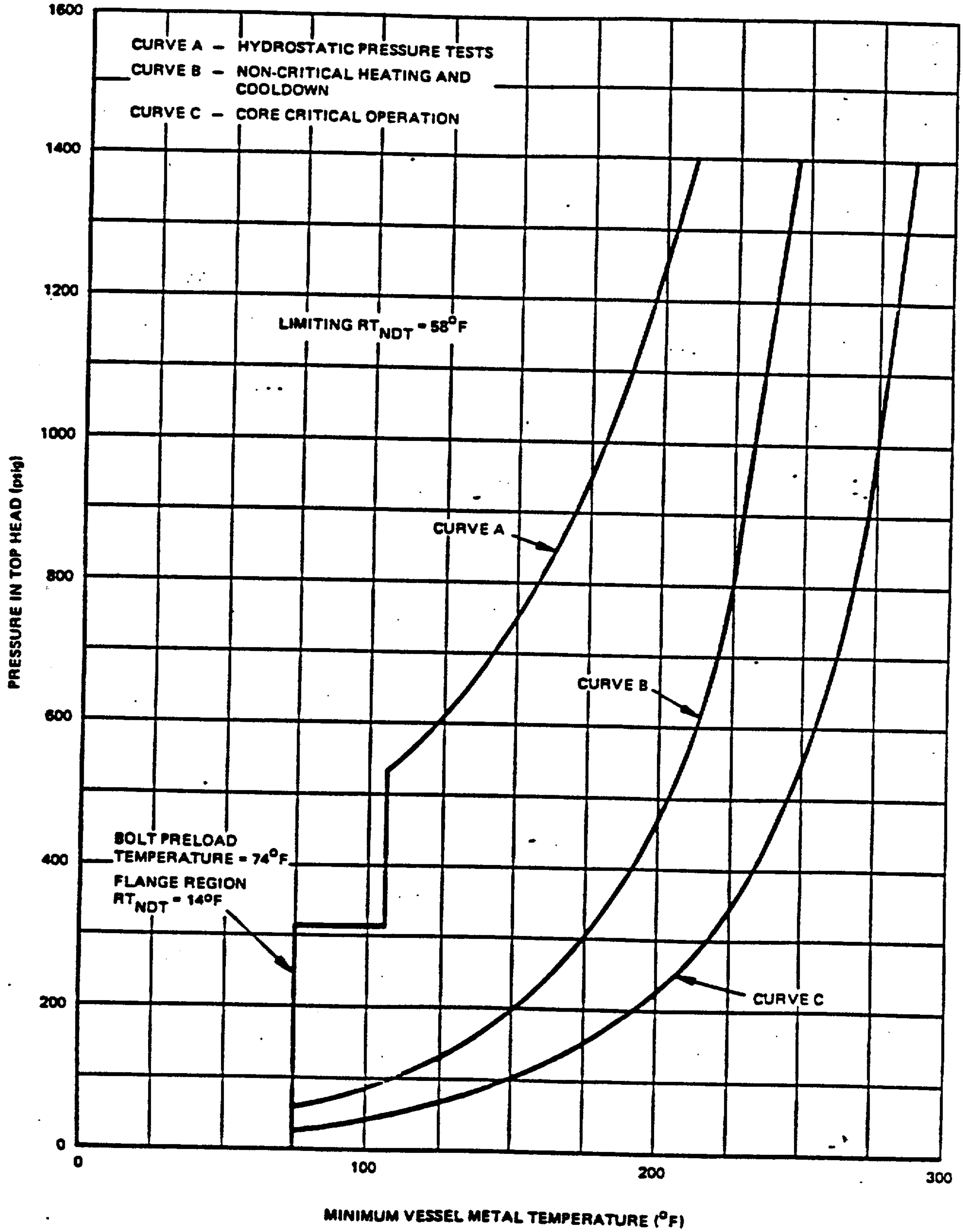


Figure 3.6-1. Pressure versus Minimum Temperature Valid to Twelve Full Power Years, per Appendix G of 10CFR50

EFFECTIVE ON JULY 31, 1985.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 124 TO LICENSE NO. DPR-49

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

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By a letter dated June 14, 1985, the Iowa Electric Light and Power Company (the licensee) requested that the effective date of the Duane Arnold Energy Center (DAEC) Amendment No. 121 be revised from May 28, 1985 to July 31, 1985.

By a letter dated January 11, 1985, the Iowa Electric Light and Power Company requested revision of the DAEC Technical Specifications to incorporate revised reactor vessel pressure-temperature operating limits. The proposed limits accounted for minor estimated changes in fracture toughness due to neutron fluence on the vessel during the first six effective full power years (EFPY) of operation and were intended to cover operation during the second such six EFPY. In retrospect, it would have been appropriate to request that the amendment be made effective upon restart of the plant. The licensee stated that it anticipated that restart would occur in May and that NRC review of the requested Technical Specification changes would not be completed until early July--i.e., six months after the submittal.

At the time of the licensee's January 11, 1985 submittal, the DAEC Cycle 7/8 refueling outage was scheduled to begin on February 1, 1985, and be completed by May 20, 1985. During the outage, the licensee was required to perform the 10-year hydrostatic test of the reactor vessel and that test was scheduled to be done after the fuel had been loaded (approximately May 6).

The outage began on schedule but, during the outage, pipe cracks were discovered. The associated repair work and other unanticipated problems have extended the outage. The licensee now expects to restart the plant on July 3, 1985, some six weeks later than was scheduled in January 1985. Meanwhile, the NRC completed its review of the requested amendment incorporating revised pressure-temperature operating limits earlier than expected by the licensee and issued the amendment on May 28, 1985.

Application of the revised limits would, in effect, make it impossible to perform the hydrostatic test while the fuel is in the reactor vessel. The

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licensee has, therefore, requested that the effective date of Amendment No. 121 be revised from May 28, 1985 to July 31, 1985 (after the June 21, 1985 scheduled hydrostatic test date).

2.0 EVALUATION

In order to comply with Amendment No. 121, the licensee will have to unload the fuel before performing the hydrostatic test. Alternatively the licensee can perform that test while adhering to the limits which were in effect prior to Amendment No. 121. The reactor has not yet achieved six EFPY of operation and, therefore, the old limits are still valid. The revised limits are based on the vessel's estimated fracture toughness at completion of twelve EFPY. Those limits are therefore extremely conservative for use at this time and require unloading and reloading of the fuel. The licensee states that such a requirement will further delay completion of the current outage.

DAEC is currently in the Cycle 7/8 refueling outage and will undergo a 10-year hydrostatic test in accordance with 10 CFR 50, Appendix G, and with the constraints of applicable pressure-temperature limits on the pressure vessel. The licensee, in a letter dated June 14, 1985, stated that the current (Cycle 7/8) refueling outage has been extended beyond the time scheduled (by the licensee) when application for Amendment No. 121 was submitted to the NRC. As a result of the extension of the current refueling outage, and the NRC approval of Amendment No. 121, the licensee's 10-year hydrostatic test will be subject to the pressure-temperature limits of Amendment No. 121 and would require the reactor water to be heated above 212°F in order to achieve the requisite vessel metal temperatures. Exceeding the 212°F water temperature after the fuel has been loaded in the reactor would actuate DAEC Technical Specification sections which require that the safety relief valves and Emergency Core Cooling System (ECCS) functions should be operable. The licensee states that with safety relief valves operable and set at their setpoints, requisite hydrostatic pressure cannot be achieved. The only way that the hydrostatic tests can be conducted subject to the Amendment No. 121 limits, is by removing the fuel from the reactor. The licensee states that the removal of fuel requires unnecessary fuel-handling operations which would contribute to additional lengthening of the current refueling outage.

The licensee states that the first six effective full power years of operation will be completed approximately 45 days after the Cycle 8 operation is commenced. Therefore, compliance with Amendment No. 121 is not required for the safe and satisfactory conduct of vessel hydrostatic tests prior to restart. The licensee requests that Amendment No. 121 be revised by changing the effective date of the amendment from May 28, 1985 to July 31, 1985, thus permitting the 10-year vessel hydrostatic tests to be conducted subject to the first six effective full power years pressure-temperature limits which assure that the water temperature will not exceed 212°F and fuel removal will not be required to satisfactorily complete the tests.

The staff evaluation indicates that revising the effective date of Amendment No. 121 would permit the licensee to perform the 10-year hydrostatic tests of the DAEC pressure vessel in accordance with the pressure-temperature limits approved (in accordance with the regulations) by the NRC for the first six effective full power years of operation. The evaluation of the fracture toughness properties of the DAEC pressure vessel, taking into account the neutron and thermal environment of six effective full power years of operation, showed that the hydrostatic tests performed with the previously-approved pressure-temperature limits will assure adequate margins against nonductile failure of the pressure vessel until the next hydrostatic test.

The staff therefore finds that the hydrostatic testing, conducted in accordance with the pressure-temperature limits associated with the first six effective full power years of operation, meets the Commission's regulations and is acceptable, and the effective date of Amendment No. 121 can be revised from May 28, 1985 to July 31, 1985.

3.0 CONCLUSIONS

3.1 Final No Significant Hazards Consideration Determination

3.1.1 State Consultation

In accordance with the Commission's regulations, consultation was held with the State of Iowa Commerce Commission, by telephone. The State expressed no concern over the proposed revision to the Amendment No. 121 effective date, in view of the fact that the licensee will perform the vessel hydrostatic tests in accordance with valid pressure-temperature limits.

3.1.2 Response to Comments

No comments were received. A notice of the proposed amendment was not published in the FEDERAL REGISTER due to the lack of sufficient time for public comment prior to the expected plant startup date (July 3, 1985).

3.1.3 No Significant Hazards Consideration Determination

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a proposed license amendment involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The information in this Safety Evaluation provides the basis for evaluating the proposed license amendment against these criteria. The licensee will perform the hydrostatic tests in accordance with the pressure-temperature limits previously approved and still applicable until July 31, 1985 in accordance with the NRC regulations. Therefore, the staff concludes that:

(1) Operation of the facility in accordance with the proposed amendment would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Accordingly, we conclude that the amendment to Facility Operating License No. DPR-49 revising the effective date of Amendment No. 121 from May 28, 1985 to July 31, 1985, involves no significant hazards consideration.

4.0 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 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. 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.

5.0 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: Mohan Thadani

Dated: June 20, 1985