Distribution
Docket
ORB #3
NRR Reading
Local PDR
NRC PDR
HDenton
VStello
DEisenhut
WRussell
BGrimes
RVollmer
TJCarter
TIppolito
RClark
SSheppard
Atty, OELD
OI&E (5)
BJones (4)
BScharf (10)
STSG
BHarless
ACRS (16)
OPA (CMiles)
RDiggs
TERA
JRBuchanan

JULY 5 1979

Docket No. 50-331

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

Dear Mr. Arnold:

The Commission has issued the enclosed Amendment No. 52 to Facility License No. DPR-49 for the Duane Arnold Energy Center. This amendment:

1. Changes the Technical Specifications in response to your application dated November 30, 1977 (RTS-100) to require that all inservice inspection and testing at the Duane Arnold Energy Center be performed in accordance with the ASME Boiler and Pressure Vessel Code except where specific written relief has been granted by the Commission.

2. Pursuant to 10 CFR 50, Section 50.55a(g)(6)(i) and your letter of October 13, 1978, grants relief from performing the Inservice Inspection Program in accordance with the ASME Boiler and Pressure Vessel Code for certain components where we have determined that the Code requirements are impractical. Specifically, for the Inservice Inspection Program, the Commission hereby grants the relief requested in your letter of October 13, 1978 in request for relief No. 2, 3, 4, 6 and 8, subject to certain modifications in the alternate inspection programs discussed in the accompanying safety evaluation.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 52 to DPR-49
2. Safety Evaluation
3. Notice

cc w/enclosures:
See page 2

as to amendment + F.R. Notice CP 1 GD

OFFICE	ORB #3	ORB #3	AD:E&P	OELD	ORB #3
SURNAME	SSheppard	RClark:mjf	BGrimes	W.D. Paton	TIppolito
DATE	5/23/79	5/24/79	6/22/79	6/29/79	5/29/79

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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cc:

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Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, 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 & Power Company
ATTN: Ellery L. Hammond
P. O. Box 351
Cedar Rapids, Iowa 52406

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
US EPA
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region VII
ATTN: EIS COORDINATOR
1735 Baltimore Avenue
Kansas City, Missouri 64108

Cedar Rapids Public Library
426 Third Avenue, S. E.
Cedar Rapids, Iowa 52401

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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. 52
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, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensees) dated November 30, 1977, 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.

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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. 52, 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: July 5, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 52

TO THE TECHNICAL SPECIFICATIONS

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
3.6-8	3.6-8
3.6-9	3.6-9
3.6-10	3.6-10
3.6-10a	3.6-10a
3.6-30	3.6-30
3.6-31	3.6-31
3.6-32	3.6-32
3.6-34	3.6-34
3.6-35	3.6-35
3.6-36	3.6-36
3.6-37	3.6-37
3.6-38	3.6-38
3.6-39	3.6-39
3.6-40	3.6-40

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3. Following 1-pump operation, the discharge valve of the lower speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

G. Structural Integrity

The structural integrity of the pressure boundaries shall be maintained at the level required by the original acceptance standard throughout the life of the plant.

G. Structural Integrity

1. In-service inspection of ASME Code Class I, Class II and Class III Components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. In-service testing of ASME Code Class I, Class II and Class III pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3. The second 40-month inspection period updating our 10-year program, in accordance with paragraphs 1 and 2 above, commences June 1, 1978. The 10-year interval program commenced February 1, 1975.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

H. Shock Suppressors (Snubbers)

1. During all modes of operation, except Cold Shutdown and Refuel, all safety related snubbers listed in Tables 4.6-3 and 4.6-4 shall be operable, except as noted in 3.6.H.2 through 3.6.H.4 below.

H. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Tables 4.6-3 and 4.6-4:

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or

3.6.G & 4.6.G BASES:

REACTOR COOLANT SYSTEM

Structural Integrity

A pre-service inspection of Nuclear Class I Components was conducted to assure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the reactor coolant system as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal
18 problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no LOCA would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the reactor coolant system, portions of the ECCS, and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II Components because it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

The engineering and design effort associated with the Duane Arnold Energy Center predates the availability of the ASME Inspection Code. However, this Code, including subsequent Addendum through the Winter 1972 Addenda, dated December 31, 1972, has been used as a guide in the preparation of the DAEC In-Service Inspection Plan for Nuclear Class I and Class II Components for the first 40-month interval of the 10-year program, and maximum access has been provided to the extent drywell design and radiation levels permit.

Inspections and testing concluded subsequent to the first 40-month interval are as required by 10 CFR 50, Section 50.55a(g).

Visual inspections for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC and is delineated by Section XI of the ASME Code. These studies show that it requires thousands of stress cycles at stresses beyond those expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results, only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Appendix J of the DAEC FSAR provides details of the inspection program for the first 40-month cycle.

3.6.H & 4.6.H BASES:

Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly

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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. 52 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. Introduction

On April 26, 1976, the Commission sent a generic letter to Iowa Electric Light and Power Company (the licensee) advising them that the inservice inspection and testing requirements for ASME Code Class 1, 2 and 3 components for nuclear power plants delineated in 10 CFR Part 50.55a were changed by a revision to the regulations published on February 27, 1976. The revised regulations require inservice inspection and testing to be performed in accordance with the examination and testing requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda thereto. To avoid potential conflicts between the ASME Code requirements and the Technical Specifications presently in effect for the Duane Arnold Energy Center, we also advised the licensee that he should apply to the Commission for amendment of the Technical Specifications. Sample language for such Technical Specifications changes was provided as an enclosure to our letter of April 26, 1976.

By letter dated November 30, 1977, the licensee requested a change to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). The proposed amendment and revised Technical Specifications would delete the present inspection and testing requirements in Section 3.6.6 of the Technical Specification and substitute therefore - verbatim - the sample language enclosed with our letter of April 26, 1976. The proposed Technical Specification would require all inspection and testing to be performed in accordance with the ASME Code except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(d)(6)(i).

Our letter of April 26, 1976 also advised the licensee that if he determines that conformance with certain ASME Section XI inservice inspection and testing requirements is impractical, he should submit information to the Commission to support his determination in accordance with 50.55a(g)(5)(iii) and (iv). By letters dated November 22, 1977 and January 5, 1978, we provided additional guidance in preparing inservice inspection

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and testing program descriptions and associated relief requests. In response to our letters, the licensee submitted a proposed Inservice Inspection and Testing Program by letter dated March 1, 1978, supplemented by letter dated March 15, 1978. This submittal also included requests for relief from examining certain components where the licensee determined that it was impossible or impractical to examine or test the specific component because of design, geometry or materials of construction. As part of our review, a meeting was held with the licensee and his consultants on June 13, 1978. In response to staff comments, the licensee submitted a revised Inservice Inspection and Testing Program by letter dated October 13, 1978. This letter also revised the requests for relief where the licensee determined that certain requirements of the ASME Code cannot be implemented at the Duane Arnold facility because of component or system design, geometry or materials of construction.

2.0 Evaluation

2.1 Technical Specifications

The changes proposed by the licensee to the Technical Specifications are identical to the sample Technical Specifications enclosed with our letter of April 26, 1976. The revised Technical Specifications require all inspections and testing to be performed in accordance with the ASME Boiler and Pressure Vessel Code and are acceptable.

2.2 Requests for Relief

As required by 10 CFR 50.55a(g), Iowa Electric Light and Power Company has updated the Inservice Inspection Program for the Duane Arnold Nuclear Generating Plant to the requirements of the 1974 Edition through Summer 1975 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code (B&PV Code). Based on information contained in the submittal dated March 1, 1978 and the revised submittal dated October 13, 1978, it has been determined that certain requirements of the Code cannot be implemented at the facility because of component or system design, geometry, or materials of construction. Requested reliefs from those requirements have been reviewed and evaluated by the staff and our determinations to grant or deny the requests, pursuant to 10 CFR 50.55a(g)(6)(i), are documented below.

2.2.1 Class 1 Components

A. Relief is requested from volumetric examination of each meridional weld in the bottom head of the reactor vessel. (Item B 1.2, Category B-B)

Code Requirement

The examinations performed during each inspection interval shall cover at least 10% of each meridional head weld.

Licensee Basis for Requesting Relief and Alternate Examination Proposed

These welds are located within the array of control rod drives and are not accessible for volumetric examination.

These welds will be visually examined for leakage or deposits caused by leakage during the leak testing after each refueling outage and during the hydrostatic test to be performed near the end of the 10-year interval.

Evaluation

Because of the design of the vessel, access to these welds is limited from the exterior of the vessel by the control rod drives and from the interior by the core shroud and core plate and preclude volumetric or surface examination with currently available technology.

The licensee has proposed to visually examine these welds for leakage or deposits caused by leakage during leak testing after each refueling outage and during the hydrostatic test performed near the end of the 10-year interval.

In addition, the staff recommended, and the licensee agreed, that the proposed examination be supplemented by a surface examination at areas of the welds accessible for surface examination.

The reactor vessel was designed, fabricated, examined and tested to the rules of Section III of the ASME Boiler and Pressure Vessel Code. The inaccessible meridional head welds are in an area of the vessel which is subjected to a lower neutron flux than that of the beltline region and is therefore less susceptible to radiation damage and brittle fracture. Areas of the vessel which are subjected to the higher neutron flux will be examined to Code requirements and the on-going material surveillance program will provide data to determine the condition or change in the properties of materials in the most severe locations.

The Technical Specifications and plant operating procedures require that certain leak detection systems be functioning during operation and impose limits on the amount of leakage that may be permitted. Specifically, the plant must be shut down for inspection and corrective action whenever the leakage system indicates, within a period of four hours or less, an increase in the rate of unidentified leakage in excess of 2 gpm or when the total unidentified leakage exceeds 5 gpm. The staff concludes that design, leakage monitoring requirements, examinations being performed on other welds on the vessel, and the examination proposed by the licensee will provide adequate assurance of the vessel's structural integrity and therefore relief from the volumetric examination requirement for the meridional head welds may be granted.

B. Relief is requested from the volumetric examination of the drain nozzle weld in the bottom head of the reactor vessel. (Item B 1.4, Category B-D).

Code Requirement

The examination of each nozzle shall cover 100% of the volume shown in Figure IWB-2500 D during each inspection interval.

Licensee Basis for Requesting Relief and Alternate Examination Proposed

This weld is located within the array of control rod drives and is not accessible for volumetric examination.

This weld will be visually examined for leakage or deposits caused by leakage during the leak testing after each refueling outage and during hydrostatic test to be performed near the end of the 10 year interval.

Evaluation

Complete failure of the weld attaching the 1 3/4 inch O.D. drain nozzle to the reactor vessel has been demonstrated by the licensee as a failure which would not cause a net loss of coolant because of the normal makeup capacity of the reactor coolant makeup system, assuming a simultaneous loss of offsite power. This weld may be exempted from volumetric examination as allowed by IWB-1220 (b)(1) and subjected to the requirements of Examination Category B-P of Table IWB-2500. The alternate examination proposed by the licensee exceeds the Code requirements for Category B-P and the staff concludes that the proposed alternate examinations will provide adequate assurance of the structural integrity of this weld.

C. Relief is requested from the visual examination for the internal pressure boundary surfaces of the main recirculation system valves. (Item B 6.7, Category B-M-2)

Code Requirements

One valve in each group of valves of the same constructural design (e.g., globe, gate or check valve, manufacturing method and manufacturer) that performs similar functions in the system shall be visually examined internally during each inspection interval.

Licensee Basis for Requesting Relief and Alternate Examinations Proposed

The main recirculation valves are located in piping which penetrates the reactor pressure vessel and cannot be isolated for disassembly and visual examination. To accomplish the required examination would entail drainage of the reactor vessel as well as removal of the core.

These valves will be examined should valve maintenance be required. For those intervals when valve maintenance does not occur Leak Tests and Pressure Tests will be performed in accordance with Category B-P (a pressure test once in 10 years).

Evaluation

Because of the design of the reactor coolant pressure boundary, the internal pressure boundary surface of the main recirculation system valves are not accessible for visual examination since the valves cannot be isolated from the reactor vessel to allow disassembly. In order to inspect the valves, the fuel must be removed from the reactor core and the reactor vessel must be drained. The licensee has committed to inspect the valves when valve maintenance is required and to conduct visual examination when the system pressure tests (IWA-5000) are conducted in accordance with the requirements for Category B-P.

For those intervals when the valves are not disassembled and inspected, the staff recommended, and the licensee agreed, that thickness measurements (ultrasonic examination) be performed on one valve of the group. The staff has determined that measurement of valve body wall thickness to the minimum requirements established by the ASME B&PV Code, Section III will provide similar information to that obtained in a visual examination, i.e., degradation of the wall by general corrosion, cavitation or erosion, and that this information is adequate in providing assurance of the continued material and structural acceptability of these valves. We conclude that relief from the internal visual examination should be granted.

2.2.2 Class 2 Components - Pressure Tests

A. Relief is requested from pressure testing the piping from the main steam relief valves to the torus.

Code Requirement

Pressure test near the end of the inspection interval. The system test pressure shall be at least 1.25 times the system design pressure.

Licensee Basis for Requesting Relief and Alternate Testing Proposed

The pipe is open-ended in the torus.

The pipe and supports will be visually examined. If there are indications of structural distress in any component or indications that the component had leaked during operation of the relief valve, the component will be surface or volumetrically examined.

Evaluation

The relief valve discharge line to the torus is an open-ended line and therefore may be exempted from the system pressure test requirement as per IWC-5220(d) of Section XI ASME Code.

B. Relief Request

Relief is requested from Code required distribution of pressure tests for Class 2 components.

Code Requirement

The tests required shall be distributed as follows:

(a) Between 25 and 33 1/3% of the required tests shall be completed by the expiration of one-third of each inspection interval.

(b) Between 50 and 66-2/3% of the required tests shall be completed by the expiration of two-thirds of each inspection interval.

(c) The remaining required tests shall be completed by the end of each inspection interval.

Licensee Basis for Requesting Relief and Proposed Alternate Testing

It is impractical to schedule tests at these intervals inasmuch as the systems cannot be isolated at the boundaries created by the NDE exemption. Redundant pressure tests would be performed that would serve no useful purpose. These systems are normally pressurized for pump or valve functional tests which would reveal any degradation of the system.

These components will be pressure tested at or near the end of each 10-year inspection interval. This proposal is in accordance with Section XI Subcommittee interpretation of Class 2 pressure test requirements.

Evaluation

The licensee has proposed that all components be pressure tested at or near the end of each inspection interval (10 years) instead of pressure testing some of the exempted components during the inspection interval. The staff has evaluated the licensee's basis for requesting relief and concluded that this request should not be granted. However, the staff concludes that the following examinations may be conducted. A system functional test may serve as a system pressure test and at least one visual examination shall be conducted at or near the end of each inspection period coinciding with a system functional test. In addition a system hydrostatic test shall be conducted at or near the end of each inspection interval. These requirements are consistent with the Winter 77 Addenda Section XI requirements for all Class 2 components and will maintain an acceptable level of quality during the 10-year interval. The licensee concurs with the staff's recommendations.

2.2.3 Class 1, 2 & 3, Components - Pressure Tests

Relief is requested from maintaining pressure four hours for all Class 1, 2 and 3 leakage and hydrostatic tests.

Code Requirement

The test pressure and temperature shall be maintained for at least four hours prior to performance of the examinations.

Licensee Basis for Relief and Proposed Alternate Testing

The intent of the Code is to hold the pressure for four hours during testing of components that are covered by insulation. No useful purpose would be achieved in holding the pressure for four hours where the components to be examined are exposed.

The test pressure and temperature will be maintained for a minimum of four hours as required by IWA-5210 (a) where areas of examination are not exposed and accessible for visual examination. The test pressure and temperature will be maintained for a minimum of 10 minutes where areas are exposed and accessible for visual examination.

Evaluation

The staff concludes that this relief should be granted with the following conditions.

(a) When performing a system pressure test the entire system must be directly visible. This includes the welds and all base materials.

(b) Following a repair the repaired area must be accessible for a direct visual examination.

(c) When the areas are exposed, the pressure and temperature shall be maintained for a minimum time of 10 minutes and for such additional time as may be necessary to conduct the examinations.

The above conditions are consistent with the rules of Section XI Winter Addenda which the staff finds acceptable and which will not decrease the quality or safety of the facility.

The licensee has accepted the above conditions and will revise the inspection procedures accordingly.

2.2.4 Overall Evaluation

We have evaluated requests for relief from Code requirements which the licensee has determined to be impractical for implementation at the facility and granted relief from the requirements in those cases where our evaluation deems that such relief will not endanger life or property and is in the public interest giving due consideration to the burden placed on the licensee if the requirements were imposed.

We conclude that the Inservice Inspection Program meets the requirements of the 1974 Edition through Summer 1975 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, to the extent practical and thus meets the requirements set forth in 10 CFR 50.55a(g).

2.3 Recirculation Inlet Safe Ends

On June 17, 1978, the licensee discovered a through-wall crack in a safe-end on one of the recirculation inlet lines. The safe-end is a short transition piece (approximately 8 inches long) joining the recirculation inlet line to the nozzle on the reactor vessel and to the internal inlet line to the jet pumps. Nondestructive testing of the other seven identical safe-ends revealed that all had indications of cracks or weld irregularities; however, these flaws did not penetrate to the surface of the safe-ends. The licensee removed all eight safe-ends and replaced them with safe-ends of an improved design. The new design minimizes the tight crevice formed by the fit up of the safe-end and an internal thermal sleeve; such crevices are known to enhance the possibility of stress corrosion cracking in an adverse chemical environment.

On March 5, 1979, we issued Amendment No. 49 to Facility License No. DPR-49 authorizing the Duane Arnold Energy Center to resume normal power operations following installation of the eight new replacement recirculation inlet safe-ends. Amendment No. 49 also changed the Technical Specifications to incorporate augmented inservice inspection of the modified safe-ends.

All pressure boundary welds (designated as numbers 2, 6 and 7) were subjected to an ultrasonic examination to provide a base line for future examinations. Complete recordings were made of these examinations to ensure that any changes in ultrasonic results indicative of cracking during service will be identified. The specific program that will be followed will be to ultrasonically examine all three welds in one half (four) of the eight safe end assemblies every refueling outage. This program will continue at least until every weld involved will have been inspected twice. As the Duane Arnold plant is on a yearly refueling schedule, this means that welds in four of the safe-end assemblies will be inspected after one and three years of operation, the remaining four will be inspected after two and four years of operation. The requirements in Amendment No. 49 are not changed by this amendment and supersede the less frequent examinations that would be required by the ASME Boiler and Pressure Vessel Code.

During the outage associated with replacement of the safe ends on the recirculation inlet lines, Iowa Electric performed ultrasonic examinations of the core spray, the feedwater (FW) and the control rod drive hydraulic system return (CRD HSR) nozzle safe ends. These are the only locations on the DAEC reactor vessel where a thermal sleeve is welded to a safe end and that contain potentially creviced Alloy 600 material on the primary pressure. There are significant differences, however, between the above safe ends and the recirculation system inlet nozzle safe ends. The maximum calculated stress index values for the core spray, feedwater and CRD HSR safe ends is 0.71, 1.19 and 0.88, respectively, compared to 2.24 for the recirculation inlet safe ends. The pressure boundary material for the FW safe end is carbon steel with Ni-Cr-Fe weld metal inlay at the crevice location, whereas, the material in the heat affected zone of the recirculation inlet safe end (where the crack occurred) is wrought Alloy 600. The crevice in the FW safe end is 0.125 inches maximum as opposed to the 0.50 inch length in the former recirculation inlet safe ends. As documented in Report No. 50-331/79-10 dated March 27, 1979 issued by NRC's Office of Inspection and Enforcement Region III, a full volumetric examination of the above safe ends was performed, with emphasis in the ultrasonic examination on the areas of the safe ends where thermal sleeves are welded to the safe ends. No significant inidcations of cracking or reflections were noted in the areas of interest on any of the safe ends. Adequate base lines for future ultrasonic examinations were established for the core spray, feedwater and CRD HSR nozzle safe ends. These safe ends will be examined by ultrasonic inspection techniques during subsequent refueling outages.

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 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: (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: July 5, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND

NOTICE OF GRANTING RELIEF FROM ASME SECTION XI INSERVICE INSPECTION (TESTING) REQUIREMENTS

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 52 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County, Iowa to require that all inservice inspection and testing at the facility be performed in accordance with the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code (ASME Code) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). The letter transmitting Amendment No. also grants relief from certain requirements of the ASME Code where the Commission has determined it is impossible or impractical to examine or test a specific component because of design, geometry or materials of construction. For those requirements of the ASME Code for which the Commission has granted relief, the Commission has required alternate, compensatory examinations and tests that will achieve the objectives of the ASME Code.

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

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

For further details with respect to this action, see (1) the application for amendment dated November 30, 1977, (2) request for relief dated October 13, 1978, (3) Amendment No. 52 to License No. DPR-49, (4) the Commission's related Safety Evaluation, and (5) the Commission's letter to the licensee dated
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 (3), (4) and (5) 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 5th day of July 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

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