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

MAR 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 completed its review of the safe-end repair work audit results. We have concluded that the safe-end repair work is acceptable as discussed in the attached SER. Accordingly, you may startup and operate the Duane Arnold Energy Center in accordance with DPR-49. As proposed in your letter of March 3, 1979, initial operating power levels will not exceed 5% for 48 hours. Following satisfactory completion of a hydrostatic test of the repaired safe-ends in accordance with applicable ASME Code requirements, you are authorized to operate at a power level not to exceed 25% for an additional 108 hours. Following these reduced power operations (for a total period of 156 hours) you may resume normal power operations at the Duane Arnold Energy Center.

In addition, the enclosed Amendment No. 49 to Facility License No. DPR-49 for the Duane Arnold Energy Center, changes the Technical Specifications to incorporate the augmented inservice inspection of the modified safe-ends on the eight recirculation system inlet lines as proposed in your submittal of February 22, 1979. As discussed with your staff, we are treating your submittal of February 22, 1979, as supplemented by your letters of March 1, 1979 and March 3, 1979, as an application for amendment of License DPR-49.

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

Sincerely,

B. Grimes for Direct note to Mr. Clark 3/5/79

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

7904030004

Enclosures and ccs:
See next page

OFFICE	ORB #3	ORB #3	OELD	AD:E&P	ORB #3
SURNAME	SSheppard	RClark:mjf		BGrimes	TIppolito
DATE	3/ /79	3/05/79	3/ /79	3/5/79	3/5/79

(signed 10:45 pm March 5)

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

March 5, 1979

Docket No. 50-331

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

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Sincerely,

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

Enclosures and ccs:
See next page

Enclosures:
1. Amendment No. 49
2. Safety Evaluation
3. Notice

cc w/enclosures:
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Harold F. Reis, Esquire
Lowenstein, Newman, Reis and Axelrad
1025 Connecticut Avenue, N. W.
Washington, D. C. 20036

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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

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. 49
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 licensee) consisting of a letter dated February 22, 1979, as supplemented by letters dated March 1, 1979 and March 3, 1979, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 5, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 49

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

   3.6-10a
   3.6-10b
   3.6-10c

2. The underlined page is the page being changed. The revised page is identified by Amendment No. and contains a vertical line along the margin indicating the area of change.

3. Add the following page:

   3.6-10aa

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	7. During each plant refueling outage all eight RPV Seismic Stabilizer assemblies and attachments will be inspected as follows: a. Visually inspect stabilizer assembly parts for deformation and cracking. b. Verify that all clevis pin retainers are in place. c. Verify that all drawbar set screws are in place. d. Visually inspect stabilizer gusset plate welds. e. Visually inspect stabilizer support-to-PRV welds such that all four welds are inspected during the regular 10-year inservice inspection interval. 8. At the end of each 10-year inspection interval, a report shall be submitted to the NRC that defines which of the following examination categories, if any, could not be completed: a. Class 1 components - Categories N, L-2, and M-2. b. Class 2 components - Category C-H. 9. Starting with the next refueling outage (scheduled for 1980) the inspections required in Table 4.6-1 under Items 1.7, 4.1 and 4.4 are augmented with respect to the three pressure boundry welds identified as weld nos. 2, 6 and 7 in Figure 4.6-1 on the eight recirculation system inlet safe-ends. Each of the three welds on four of the safe-ends shall be ultrasonically examined during the next refueling outage. The three welds on the other four safe-ends shall be ultrasonically examined during the subsequent refueling outage. This cycle of inspections shall continue in the same sequence in subsequent refueling periods.

Amendment No. ~~18~~, ~~24~~, ~~32~~, 49

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	10. Following repair of the modified recirculation system inlet safe-ends, the facility shall be operated at a power level not to exceed 5% for 48 hours. Following satisfactory completion of a hydrostatic test of the repaired safe-ends in accordance with ASME Code requirements, the facility shall be operated at a power level not to exceed 25% for an additional 108 hours.

This page shall be removed when the NRC's Region III Office of Inspection and Enforcement has determined that the above requirements have been satisfactorily completed.

LIMITING CONDITIONS FOR OPERATION

H. Shock Suppressors (Snubbers)

1. During all modes of operation, except Cold Shutdown and Refuel, all safety related snubber 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.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.H.1 and 3.6.H.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to safety related systems without prior License Amendment to Tables 4.6-3 or 4.6-4 provided that a revision to Table 4.6-3 or 4.6-4 is included with the next License Amendment request.

SURVEILLANCE REQUIREMENTS

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 analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers are categorized in two groups, "accessible and inaccessible" based on their accessibility for inspection during reactor operation. These two groups will be inspected independently according to the above schedule.

LIMITING CONDISIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.
	3. The initial inspection shall be performed within six months ± 25% from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.H.1, it shall be assumed that the facility has been on a 6-month inspection interval.
	4. Once each refueling cycle a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock-up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten (10) hydraulic snubbers shall be so tested until notmore failures are found or all units per category tested have been tested. Snubbers of rated capacity greater than 50,000 lb. need not be functionally tested.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 49 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 Introduction

On January 8, 1979, the Nuclear Regulatory Commission (NRC) issued a Safety Evaluation Report (SER) pertaining to the safe-end cracking that occurred at the Duane Arnold Energy Center (DAEC) on June 17, 1978 and to the repair program initiated by the licensee to replace the damaged safe-ends with safe-ends of an improved design. This SER was transmitted to the Iowa Electric Light and Power Company by a letter also dated January 8, 1979.

There remained, however, three items requiring resolution before operation under the amendment could commence; these are (1) a finding that the testing had been conducted in conformance with approved procedures; (2) satisfactory completion of a hydrostatic test of the repaired safe-ends in accordance with applicable ASME Code requirements; and (3) completion of the licensee's audit of the safe-end repair work and resolution of any discrepancies identified in this audit. These three items were reiterated in a letter to the licensee dated January 16, 1979 from the NRC's Region III Office of Inspection and Enforcement. Item (1) and the part of item (3) regarding the licensee's audit of the safe-end repair work have been completed to the satisfaction of the Office of Inspection and Enforcement; this will be documented in a subsequent inspection report. Item (2) must be completed to the satisfaction of the NRC's Region III Office of Inspection and Enforcement before the facility is authorized to exceed 5% rated power.

In relation to those aspects of item (3), not completed by I&E there are several matters that required evaluation because of questions concerning certain discrepancies in the actual performance of the safe-end repair program. This safety evaluation assesses the safety significance of the following information:

A. Radiographic (RT) examination of the new pressure boundary welds indicated that some of the weld root inside surfaces are very irregular, possibly caused by difficulty with uniform melting of the insert used for the first welding pass. In addition, obvious oxidation in local areas was noted in the radiographs, indicating that the inert gas purge was not uniformly effective.

B. During the initial leak test, performed after all repair operations, a flow blockage was noticed in one recirculation riser and jet pump assembly. An investigation showed that the flow blockage was caused by a temporary radiation shield plug used to protect personnel during the welding operation and inadvertently left in the pipe near the nozzle. This plug consisted of a thin aluminum and carbon steel can filled with shaped lead blocks. During the leak test, water flow in the line pushed the plug into the jet pump assembly where it came apart. Retrieval operations recovered most of the can and all of the lead blocks. One small lead block went through the jet pump and down to the bottom of the reactor vessel, but it was recovered also. Because all the lead blocks were recovered, the only potential safety significance of this event is that Alloy 600* may be subject to stress corrosion cracking at reactor operating temperature due to residual lead smears or high concentrations of lead in the water. In addition to the Alloy 600 nozzle safe end and thermal sleeve that could have been contaminated by contact with the lead blocks, the control rod drive stub tubes in the lower head of the reactor vessel also are made of Alloy 600 and may have been contaminated by the lead block that fell to this region. All other hardware that could have been contaminated by contact with the lead blocks is made of stainless steel, which is not affected by lead contamination.

Several small metal tabs of carbon steel and a part of the thin aluminum backing plate from the can used to contain the lead blocks were not recovered.

This SER addresses the potential effects of the relatively poor weld root geometry of some welds, the effect of possible lead contamination, and the possible effects of the loose pieces of metal on the safe operation of the plant. These issues were addressed in the licensee's submittal of February 22, 1979, as supplemented by letters dated March 1, 1979 and March 3, 1979.

2.0 Discussion and Evaluation

2.1 Weld roots

2.1.1 Characterization of weld root conditons

Although there were differences of opinion regarding the detailed weld root geometry and conditons as determined from interpretation of the radiographs, it was agreed that the evidence clearly showed a wide range of conditions existed from weld to weld and even from place to place on some individual welds. Therefore, the licensee characterized the root conditions at weld numbers 2 and 6

* The Alloy also is know as Inconel 600.

(see Figure 1) by assuming and analyzing six different notch configurations (see Figures 2 and 3). These notch configurations were chosen by the licensee to conservatively bound, relative to mechanical strength, the worst root conditions that could be inferred from the interpretation of the radiographic films. The worst-case condition enveloping all the stated variations is a sharp cup-shaped intrusion, with a depth of about 1/10 the wall thickness. This condition is designated as Case E* in the licensee's analyses.
There are three possible adverse effects of this assumed worst-case root condition; adequate fatigue life, initiation of stress corrosion cracking, and initiation of brittle or unstable fracture without prior warning. These three possible effects are discussed below.

2.1.2 Fatigue analysis

The licensee performed detailed stress analyses for weld locations 2 and 6 in accordance with the requirements of Section III of the ASME boiler and Pressure Vessel Code assuming the notch configurations shown in Figures 2 and 3. The results of the original stress analyses were modified to account for the reduced wall thickness at the assumed notches and to obtain primary stress intensities and primary plus secondary stress ranges. No credit was taken for any weld reinforcement at the outer weld surface. In addition, theoretical stress concentration factors were determined for Cases A through D and F for the calculation of the fatigue usage factors. For Case E, a fatigue strength reduction factor of 4.0 was used which is the highest value required by the ASME Boiler and Pressure Vessel Code for a partial penetration weld.

The analysis of weld numbers 2 and 6 for each of the assumed notch configurations, considering the reduced wall thickness and the fatigue reduction factors, show that the calculated stresses and fatigue usage factors are within the limits prescribed by Section III of the ASME Boiler and Pressure Vessel Code.

The staff has evaluated the fatigue analysis and agrees with the licensee's evaluation.

* The letters A through F refer to the notch configurations analyzed by the licensee as shown in Figures 2 and 3.

2.1.3 Stress Corrosion

Because there were concerns that the weld root irregularities could be severe enough to act like crevices to initiate stress corrosion cracking, the licensee performed a stress corrosion crack growth rate analysis. In this analysis he assumed that stress corrosion cracks would be initiated, even though the worst case root condition postulated is not nearly as severe a crevice as was present in the original design. The results of these analyses show that the crack growth rate in terms of fractional depth through the wall as a function of time would be slower than in the old design.

The staff has considered the possibility of stress corrosion cracking initiating at the irregular weld roots, and considers that it is a fairly remote possibility, because the root irregularities are very unlikely to be deep enough and tight enough to cause the crevice chemistry conditions necessary to initiate stress corrosion cracking. Even the worst postulated case, that of a cusp-shaped defect 1/10 the wall thickness at local areas around the circumference, is nowhere near as severe a crevice from a stress corrosion standpoint as the built in deep crevice of the original design.

Although the staff has reservations regarding the bases for the stress corrosion crack growth predictions submitted by the licensee, we conclude that even if stress corrosion cracking should initiate at local spots around the circumference, crack propagation would be no faster than the original stress corrosion cracks, and most likely will be slower, because the nominal stress levels at welds 2 and 6 are lower than those at the cracked section of the original safe ends.

2.1.4 Brittle or Unstable Fracture

Alloy 600 is a very ductile and tough material that is very resistant to brittle or unstable fracture. The licensee performed analyses to justify the assumption that complete failure of the new safe end and pipe welds will not occur unless an extremely large crack is present, and that the worst case weld root irregularities could not initiate failure of the weldments. This was done using a net-section stress analysis of the new safe end design with a postulated crack. In this approach, it is assumed

that a pipe (or a safe end) of highly ductile material with a crack is at point of incipient failure when the stress in the remaining ligament ahead of the crack reaches the flow stress.* Numerous test have been performed by Battelle, General Electric and others using pre-cracked pipes under both pressure and bending loads. The results of these tests validate the efficacy of this approach and indicate that a typical pipe of tough material can tolerate a crack of sufficient size to make detection of a crack by inservice inspection of by the leak detection system highly likely before a critical crack size is reached.

The staff agrees with the method of analysis and with the licensee's conclusions. The staff also used information available from the recent Pipe Crack Study Group report. In this report, a more sophisticated analysis was performed utilizing the tearing stability concept and the associated tearing modulus stability criterion. Based on this analysis and the analyses performed by the licensee and the staff, we conclude that is is unlikely that safe end cracks or postulated cracks emanating from irregular weld root geometries, should they occur and be missed by inservice inspection, will result in unstable crack growth and excessive loss of coolant. In addition to the confidence gained from the above analyses, the inservice inspection program and the leak detection system, the emergency core cooling system is designed to keep the plant in a safe condition even postulating the failure of the largest pipe in the reactor system.

2.2 Lead contamination

The radiation shields used in the nozzles to protect welders from radiation emanating from the reactor vessel were made of shaped lead blocks enclosed in a thin sheet metal can. The circular end piece was made of 0.016" thick aluminum and the cylindrical portion of 0.016" thick carbon steel sheet metal.

A cause for concern about lead smear contamination results from the one shield plug that was not removed before the pipe closure welds were made. The plug was forced by water flow into the elbow and jet pump assembly where it came apart. Because some of the lead blocks fell back down the jet pump riser pipe, there is a possibility that lead came into contact with the Alloy 600 portion of the thermal sleeve. In the case of the single lead block that passed through the jet pump and was found in the bottom head of the vessel, contact between it and the Alloy 600 control rod drive stub tubes must be assumed. Although contact with stainless steel components could also result in smears of lead contamination, there is no concern regarding

* The flow stress has been shown by various experiments to be approximately the average of the yield stress and te ultimate stress of the material.

these components other than that the smears would add to the total inventory of lead in the system.

Tests were run by General Electric to determine the amount of lead that could be deposited on such components by forceful impact. They reported that the maximum smear was measured to be 0.0001 inch thick. It is unlikely that significantly thicker deposits would have occurred in the actual incident because of the cushioning effect of the water in the vessel and the comparatively small mass of the actual lead blocks. At plant operating temperatures (about 550°F) in flowing water the lead smears, if present, will dissolve in about two days. The resulting concentration of lead in the reactor water would be very low due to the large mass of water and the continuous clean-up.

Except for Alloy 600, all reactor and fuel materials (carbon steel, stainless steel and zircaloy) are not affected by the presence of lead or lead compounds in water. Although lead causes stress corrosion cracking in nickel-base alloys, relatively high concentrations are necessary. As determined from the open literature, the measured time to initiate SCC in Alloy 600 loaded to stresses above yield in BWR conditions is three weeks or more. Thus the smears will be oxidized or dissolved in less time than that necessary to influence the Alloy 600 corrosion behavior. The reactor water clean-up system will remove the lead or its compounds. We conclude that with the successful retrieval of all the lead blocks that any minor amounts of lead that may have been left in the system as lead smears will not represent a safety concern.

2.3 Loose Parts Evaluation

The licensee wishes to commence operations with some loose objects that cannot be accounted for and are assumed to be somewhere in the vessel. The objects consist of a fragment of aluminum, which were 16 small carbon steel tabs, torn from the protective cannister.

2.3.1 Aluminum Fragment

The aluminum section of the can used to contain the lead blocks has the shape of a segment of a circle of diameter 9.75" with a chord length of 5.25" and a thickness of 0.016". This section is presumed to have passed through one of the jet pumps into the bottom region of the reactor vessel. It may have done so as one deformed piece or as two or more fractured pieces. In any event, while the recirculation system is in operation, the water velocities and degree of turbulance in the bottom region of the vessel are sufficient to keep the aluminum piece or pieces levitated so that most likely it or they would follow the streams of water flow toward the fuel assemblies and either be jammed in the fuel support assembly or further broken up by hydraulic forces to pieces small enough to pass into the fuel.

Aluminum loses its strength at elevated temperatures and, at 540°F operating temperatures, is approximately as strong as, but more flexible than, typing paper. Because of this low strength, we estimate that approximately one pound of force would drive the fragment through a fuel support casting orifice. Actual hydraulic forces are about a factor of twenty greater than this, and therefore we conclude that the fragment, even if crumpled and wedged such that it had sufficient area to eclipse an orifice, would not be likely to obstruct orifice flow more than momentarily.

Once past the orifice, the fragment would come to rest against the lower tie plate of a fuel assembly. If completely flattened out, the fragment has an area of about 3 inches square. Thus, the fragment would not cause the 86% area restriction GE has determined necessary to cause DNB at full power. [NEDO-10174, rev. 1]

It is expected that the fragment will break up into smaller particles which would be drawn up into the fuel bundle. Such a particle, small enough to pass the lower tie plate, would not cause any significant change in local flow. Cross flows would eliminate any perturbation, and in any case the flow perturbation would be of the same order or smaller than normal perturbation caused by steam bubbles.

If a small particle were to become caught at a grid spacer, no problem would likely result since the grid spacers normally reduce power in fuel rods in their locality. Particles carried through to the core exit need not be considered further, as they would at worst be carried through to the lower plenum again.

In any case, the licensee has demonstrated, by autoclave tests of samples of aluminum recovered from the shield cannister and samples of stock from which the cannister was fabricated, that the aluminum fragment will completely corrode away to Al_2O_3 (as a fine powder) after exposure to water at BWR operating temperatures in about 156 hours or less. The licensee has proposed to operate first at 5% power (to dissolve the lead smears discussed earlier), then at 25% power until 156 hours at 540°F have been accumulated. Only then will power be increased to rated.

On the basis of the above discussion we conclude that the aluminum fragment will not cause a safety problem due to flow blockage.

2.3.2 Carbon Steel Tabs

The carbon steel tabs will retain their integrity and will not corrode away for several years. They are light enough to be carried up to the fuel support casting orifices, just like the aluminum fragment. Therefore, flow blockage at the orifices and at the lower tie plates must be considered.

Except in the case of peripheral assemblies, the tabs are too small to block the orifices and should pass freely through. The licensee has conseratively calculated the probability of peripheral orifice blockage by one tab to be 4% (peripheral orifices are smaller). One tab can obstruct up to 81% of the flow area, which is slightly greater than the 79% DNB threshold discussed in NEDO-10174, rev. 1). However, peripheral bundles operate at much lower power, generally 2/3 or less of that of the "average" bundle. Therefore, transition boiling is unlikely even if a tab does obstruct a peripheral assembly's orifice. We therefore conclude that local fuel damage is highly unlikely to result from orifice blockage.

Once past an orifice, a tab would remain at the lower tie plate. GE has calculated (NEDO-10174 rev. 1) that 78% area blockage at the lower tie plate is necessary to cause transition boiling in an assembly. This tolerance to blockage is due to the automatic reduction in power caused by reduced flow, due to the other structures and orifices also contributing to hydraulic resistance, and also due to some leakage flow to the assembly. Although we have not yet accepted [NEDO-10174, rev. 1], we note that all the tabs collected at one assembly in the worst assumed distribution would cause at most a 66% blockage, much less than the 78% calculated to cause transition boiling. The low probability of such a blockage occurring in a limiting assembly at the worst time of life simultaneous with an abnormal operational occurrence is low. Therefore, we find the conclusion that no adverse flow blockage will occur to be acceptable.

2.3.3 Other Loose-Object Considerations

The loose objects are too small to cause mechanical damage due to impacting or abrading the rather massive components located in the lower plenum. Coolant velocities vary, but average about 15 feet/second. The impact of a steel tab or aluminum fragment moving with the coolant is therefore equivalent to dropping it several feet in air.

When carried up into the chamber below a lower tie plate, a loose object can more credibly cause damage due to the

increased turbulence. However, test data indicates that one fuel rod can absorb 250 ft-lb without damage when struck from the end. [NEDO-24011, "Generic Reload Fuel Application," May 1977] The parts of concern would have impact forces of less than 1 ft-lb. We therefore conclude that mechanical impact damage will not pose a problem.

Mechanical interference with moving parts must also be considered. The only moving parts within the lower plenum and in contact with coolant are the control blades and their drives. The blades are located within guide tubes, the inside of which does not communicate with lower plenum water. Thus, there is no way for a loose object to enter the support column, and mechanical interference is precluded.

Similarly, the control blade drives do not communicate with lower plenum water except through a ball check valve which is too small to admit the tabs and which is normally closed in any case.

Therefore, there is no significant possibility of mechanical interference with the control blades and their drives.

Based upon the conclusions above, we conclude that the presence of the loose objects will not pose a safety problem when the plant is operated.

3.0 Augmented ISI

The repair and redesign of the safe end thermal sleeve assembly eliminated the primary cause of the original stress corrosion cracking, namely, the built-in crevice between the non-welded portion of the facing surfaces between the thermal sleeve and the safe end.

Nevertheless, to provide additional assurance that unrecognized factors resulting from irregularities in the root passes of the welds do not exist that could cause future cracking of the pressure boundary welds in the new safe end assembly, the staff has required, and the licensee has committed to, an augmented inservice inspection program for these welds. All pressure boundary welds (designated as numbers 2, 6 and 7) were subject 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 NRC staff will require that the licensee submit his criteria for each of the above inspections at least 30 days prior to the inspections. We will require that the safe end assemblies selected for the first inspection be those deemed to have the poorest weld root conditions. The license will be modified by changing the Technical Specifications to reflect these conclusions.

A decision on whether or not it will be necessary to continue or modify this program will be made after all welds have been examined twice, according to the schedule described above. The plan for the augmented inservice inspection of the safe end volume away from pressure boundary welds, will incorporate conclusions reached after analysis of the Pipe Crack Study Group report (to be issued) as it relates to Alloy 600 safe ends.

4.0 Leak Detection

The technical specifications governing nuclear facility operation require that certain leak detection systems be functioning during operation and imposes limits on the amount of leakage that may be permitted. When these conditions cannot be met, timely remedial measures are required, including possible shutdown of the facility. In NUREG-0313*; the NRC staff recommended that facilities that cannot meet the guidelines stated in Part II of that document (i.e., facilities with piping materials susceptible to IGSCC) have augmented leak detection requirements. Specifically, plant shutdown should be initiated for inspection and corrective action when 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 attains a rate of 5 gpm, whichever occurs first. The recent Pipe Crack Study Group reconsidered this requirement and concluded that it is still appropriate.

The licensee states that since February 1975 the DAEC has implemented the criteria as required in NRC IE Bulletin No. 74-10B which are essentially in agreement with the NUREG-0313 augmented leak detection recommendations.

The primary leakage detection system for the DAEC containment is the Drywell Floor Drain Sump system for which the licensee states a four-hour interval is required to obtain an accurate leak rate measurement. Based on information obtained via this system, the DAEC operators

* Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.

concluded that the June, 1978, safe end crack resulted in approximately a two gpm increase in unidentified leakage over about a twelve-hour period (not within a period of four hours or less) and hence immediate shutdown was not required. Information obtained from other detection systems, although not quantitative in terms of gpm, could be interpretated to indicate that the actual safe end leak developed over a period less than twelve hours and possibly less than four hours.

There are inherent time limitations associated with the various types of leak detection systems. For instance, assume the instantaneous development of a two gpm leak from the primary system at operating temperature and pressure. A portion of this two gpm will flash to steam and be absorbed by the atmosphere or condense on cooler structures and equipment and most likely will take some time to reach the sump. Even that portion of leakage exiting directly as water may have to traverse a long a tortuous path to the sump. Although other leakage detection systems can provide more prompt information, they too have limits such as associated with mixing times of steam and its associated entrained radioactive materials with the containment atmosphere.

The DAEC licensee has committed by letter dated March 1, 1979, to modify his present operating procedure by changing the words "...within a period of 4 hours or less..." to read "...within a period of 24 hours or less..." and not alter the other words. The sump level will continue to be monitored at four-hour intervals. This interim commitment will remain in effect until the licensee can satisfy the NRC that modifications or additions to his present leakage detection systems, which he may propose, will increase the sensitivity of his systems and aid in discriminating against non-critical sources of leakage such as through valve stem packings, or unless modified in accordance with the provisions of the applicable Technical Specifications.

5.0 Evaluation Summary

Except for the increase in the ISI, the matters discussed above do not affect the presently approved design and operating license amendment authorization. They relate in the main toward further evaluation performed concerning certain questions which arose in the completion of the repair work. The changes in ISI discussed above involve an amendment to the Technical Specifications.

6.0 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.

7.0 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.

Date: March 5, 1979

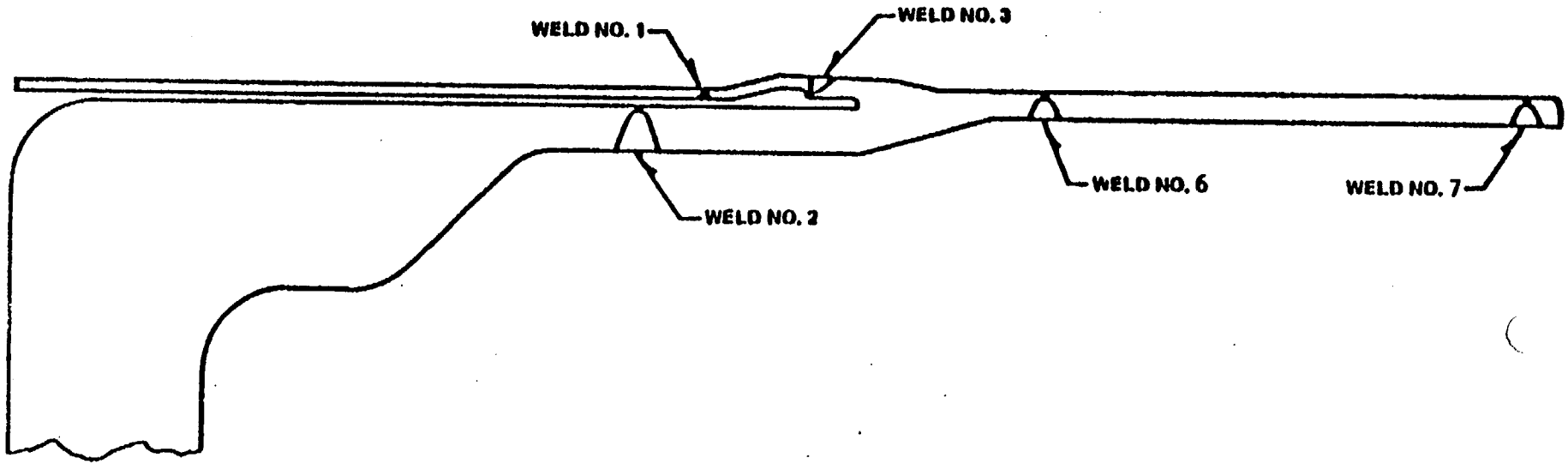


FIGURE 4.6-1

FIGURE 2: Analyzed Notch Configurations at Root of Safe End to Piping Weld (Weld #6)

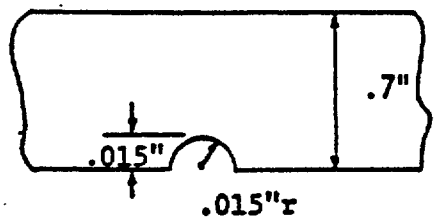
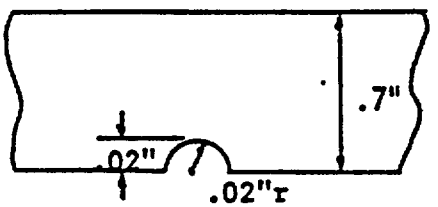
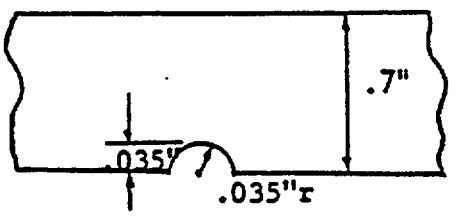
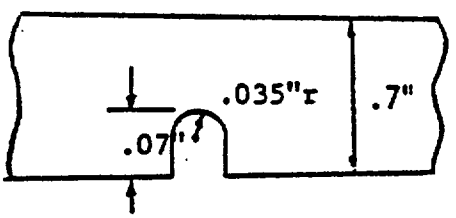
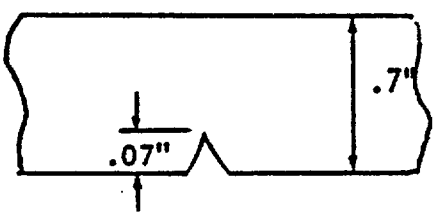
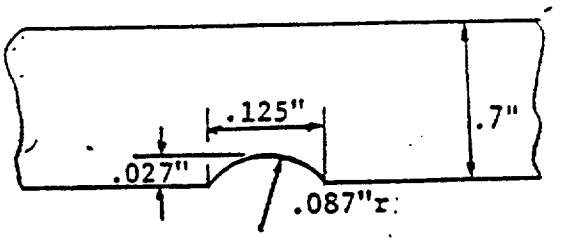
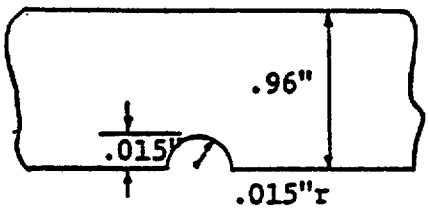
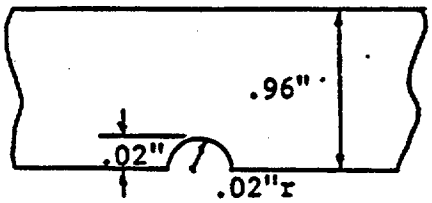
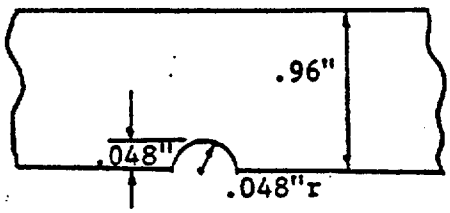
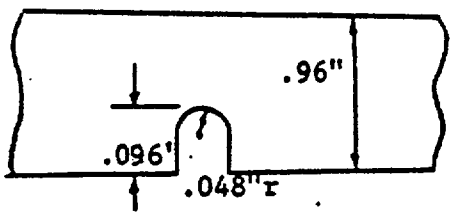
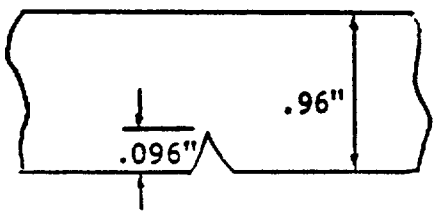
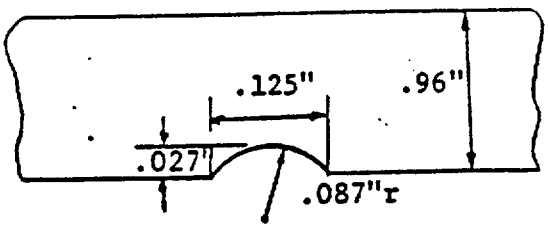
CASE	NOTCH CONFIGURATION	% OF WALL	STRESS CONC. FACTOR
A		2.14%	2.93
B		2.86%	2.90
C		5%	2.83
D		10%	3.3
E		10%	4.0
F		3.86%	2.02

FIGURE 3: Analyzed Notch Configurations at Root of Nozzle to Safe End Weld (Weld #2)

CASE	NOTCH CONFIGURATION	% OF WALL	STRESS CONC. FACTOR
A		1.56%	3.1
B		2.08%	2.9
C		5%	2.8
D		10%	3.3
E		10%	4.0
F		2.81%	2.1

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. 49 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 add augmented inservice inspection of the modified safe-ends on the eight recirculation system inlet lines and specifies a power ascension schedule.

The application for the amendment 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. Prior public notice of this amendment is 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 application for amendment dated February 22, 1979, as supplemented by letters dated March 1, 1979 and March 3, 1979, (2) Amendment No. 49 to License No. DPR-49, and (3) the Commission's related Safety Evaluation. The Safety Evaluation also discusses a number of other matters which arose during the completion of repair work at the facility. 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 5th day of March 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

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