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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 Introduction

Since May 1, 1978, the Duane Arnold Energy Center had been monitoring a slowly increasing rate of leakage from an unidentified source in the drywell. On June 14, 1978, the leakage rate increased from about 1 gpm to 3 gpm. On June 17, 1978, an automatic scram occurred during the weekly control valve testing due to problems in reactor protection system relays associated with the testing. Although the leak rate was within the Technical Specification limit of five gpm (for leakage from an unidentified source), the licensee took advantage of the unplanned shutdown to deinert the containment and identify the source of the leakage.

During the inspection of the reactor coolant system piping, a through-wall crack was found in one of the eight recirculation system inlet nozzle safe-ends. The safe-ends are transition pieces that join the ten-inch recirculation system piping to the inlet nozzles on the reactor vessel and to the thermal sleeve to the jet pumps.

The purpose of the reactor recirculation system is to provide forced circulation of water through the reactor core. Forced circulation permits a higher specific power than natural circulation and permits control of flow distribution to all channels. The recirculation system consists of two separate, parallel pump loops which operate simultaneously but independent of each other (see figure 1). Each loop consists of a variable speed, motor driven recirculation pump, two motor-operated gate valves (for isolation of the pumps), 16 jet pumps, piping and instrumentation. The recirculation system, which is part of the primary system pressure boundary, is located inside the drywell containment structure. The jet pumps are located inside the reactor vessel, between the core shroud and vessel wall. Each of the recirculation pumps withdraw water from the reactor vessel annulus area through a 22" suction line and discharge the water into a 16" manifold containing four 10" riser pipes per recirculation loop. Each riser penetrates the reactor

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vessel and supplies water to two jet pumps. The jet pumps mix the high velocity water from the recirculation system with water in the annulus and circulates this through the core. As noted above, the leak was in one of the 10" riser pipes at the point where the piping connects to the reactor vessel, specifically in the safe-end or nozzle N2A. These safe-ends contained an area in which a machining error was weld repaired during their original manufacture.

At the time the facility was shutdown on June 17, 1978, the leak rate through the crack was about 3 gpm. The leaking water was collected in the containment structure drain system, from whence it was pumped to the plant's radioactive waste treatment system for processing. There was no release of radioactivity to the environment as a result of the crack.

Ultrasonic testing and radiographic examinations were performed to determine the crack extent in the leaking safe-end and to check the other seven safe-ends. Iowa Electric reported that linear UT indications in excess of Code limits were recorded on five safe-ends (including the leaking one); later discussion revealed that the UT data showed indications, interpreted as originating from cracks, from all eight safe-ends. Based on the initial NDE results, Iowa Electric decided to replace all eight safe-ends with an improved design and so notified the Commission.

Prior to initiation of any cutting operations to remove the safe-ends, Iowa Electric held a meeting with the NRC staff on July 7, 1978 to discuss the proposed repair program. (See minutes of that meeting issued July 10, 1978.) At that meeting, the licensee proposed that the safe-end with the through-wall crack be sent to an independent consulting laboratory for detailed metallurgical analysis. The staff requested, and the licensee agreed, to provide one of the remaining 7 safe-ends to the staff so we also could perform a confirmatory metallurgical analysis. The licensee sent the safe-end with the through-wall crack (safe-end N2A) to Southwest Research Institute for examination. The staff selected safe-end N2E, which we sent to Battelle-Columbus Laboratory for examination.

By letter dated July 31, 1978, Iowa Electric provided a report describing the repair program, the cutting and welding procedures, the steps to be taken to insure that the reactor system was maintained in a safe configuration during the outage, the methods to be used to maintain radiation exposures as low as reasonably achievable and the design of the replacement safe-ends.

By letter dated October 27, 1978, Iowa Electric forwarded the interim report by Southwest Research Institute (SRI) titled "Metallurgical Investigation of Cracking in a Reactor Vessel Nozzle Safe-End". Supplemental hardness analyses performed by SRI were supplied by Iowa Electric's letter of November 27, 1978.

In parallel with Iowa Electric's efforts, the NRC staff initiated an examination of safe-end N2E at Battelle Columbus Laboratories (BCL) through our consultants, Parameter, Inc., to independently confirm results obtained at SRI. An interim report dated November 8, 1978 titled "Examination of Inconel Safe-End from Duane Arnold" was submitted by BCL which discusses results of their metallurgical evaluation and planned additional work. The additional work resulted from a technical meeting which the staff held at BCL on October 26, 1978; the additional work included more chemical analyses of the sulfur contamination and grey phase identified in the samples.

All BCL investigative efforts are now complete and a final report is expected by mid-January, 1979. The results in the interim report plus our recent discussion with Parameter and BCL indicate that the BCL findings and conclusions as to cause of cracking are in reasonably good agreement with the SRI results discussed in their interim report and Iowa Electric Company's report of December 8, 1978.

On November 14, 1978, the NRC staff held a technical meeting open to the public with Iowa Electric Light and Power Company and their consultants in Cedar Rapids, Iowa. A followup meeting was held at the NRC offices in Bethesda, Maryland on December 6, 1978, to discuss several items which were not completely resolved at the November 14th meeting. The information discussed at these meetings was essentially that which was documented in Iowa Electric's report of December 8, 1978.

2.0 Discussion

Failure Analysis

Nondestructive tests of all eight safe-end segments, after they were removed from the vessel, showed all to be cracked at the safe-end to thermal sleeve weld, 360° around the inside surface. Two segments, N2A and N2E, were selected for destructive examination. N2A, with the leaking through-wall crack, was sent to the Southwest Research Institute where work proceeded under the direction of Iowa Electric. N2E was chosen by the staff because the original UT results had identified cracking that extended around the entire inner circumference and, by subsequent UT, slag indications were identified. It was examined at the Battelle Columbus Labs under the direction of the NRC.

Both laboratories reached the same conclusion regarding both the nature and the mechanism of the cracking. Briefly, they observed that: (1) the cracking was entirely on the safe-end side of the crevice formed by the welding of the thermal sleeve to the safe-end, originating near the crevice tip; (2) the cracks began in the weld HAZ; (3) the cracks propagated entirely by intergranular stress corrosion with an absence of evidence of mechanical fatigue; (4) the weld repair on the outside surface of the safe-end was not involved in crack initiation (and had little to do with the later states of cracking; (5) the cracks did encircle the safe-end ID (at the crevice) around the full 360º; (6) there were a significantly large number of particles on the fracture surface which were compounds high in sulphur from an as yet unidentified source.

Although some small cracks were observed in the vicinity of the slag inclusions examined metallographically at Battelle, neither the slag or the tears played a role in the N2E safe-end cracking. The extensive evidence assembled at both laboratories supported the conclusion that the cause of failure was intergranular stress corrosion cracking (GSCC).

In their examination of the safe-ends, both SRI and BCL detected sulfur contamination. Additional investigations were conducted by both SRI and BCL to determine whether the sulfur contamination was associated with sulfur segregation at grain boundaries of the initial Inconel matrix or whether it resulted from progressive concentration in the fractures by transport mechanisms from other sources in the system. While the sulfur ion species was not conclusively identified, extensive analysis by BCL utilizing selective grain boundary etching techniques revealed no evidence of sulfur segregation at grain boundaries. This suggests that the sulfur was probably from other sources in the system. SRI concluded, that if a sulfur species was entrapped from the environment, it could lead to acidification of the crevice and contribute to cracking. The EDAX* analysis of crud deposits and pH values (approximately 4-6; indicated by qualitative litmus tests within the fractures) tend to further support this conclusion. BCL was unable to make a quantitative evaluation of the extent of the contribution of sulfur to the cracking.

Description of the Replacement Design

The modification consists of replacing the existing eight cracked recirculation inlet nozzle safe-ends with redesigned safe-ends and thermal sleeve adapters. The safe-ends were fabricated from SB-166 (Alloy 600) material.** They function as transition pieces between the stainless steel reactor recirculation piping and the carbon steel reactor vessel nozzles, and as attachment points for the internal thermal sleeves, that carry the recirculation flow to the jet pump risers (see Figure 1).

*EDAX = Energy Dispersive X-ray Analysis

**The alloy also is known as Inconel 600. "Inconel" is a registered trade mark of the International Nickel Company.

The new safe-end design removes the thermal sleeve attachment weld from the primary pressure boundary and eliminates the sharp crevice in the high residual tensile stress area. In the new design, the thermal sleeve is welded to the safe-end at a point away from the pressure boundary wall (see Weld #3, Figure 2). The new safe-end design improves the in-service inspection access at the nozzle attachment end by maintaining longer straight inside and outside surfaces, thus simplifying ultrasonic angle beam examinations.

Installation

The installation of the new assemblies involves five separate welds at each safe-end (see Figure 2). Prior to installation, each weld was mocked up to demonstrate reproducibility of welding and dimensional control of weld shrinkage. Because the root of the production weld between the thermal sleeve and the safe-end could not be examined after welding to confirm that complete fusion was achieved, the mockup weld was sectioned, etched and examined. Although it was observed that the backing ring deformed against the thermal sleeve creating a crevice-like condition, the joint is not located on the primary pressure boundary of the safe-end, therefore if the condition were to induce IGSCC the pressure boundary integrity would be unaffected.

The installation procedure consisted of machining weld preps on the reactor vessel nozzle and thermal sleeve using witness marks to obtain accurate dimensional tolerances between the safe-end and the thermal sleeve adapter. The weld prep on the vessel nozzle was made on the existing Ni-Cr-Fe weld butter to avoid dissimilar metal field welding during installation and the necessity for post-weld heat treatment. The weld root pass was made with a consumable insert in place thus minimizing the likelihood of forming a crevice through incomplete penetration. While the root pass of the nozzle to safe-end weld was being laid down, the annular region between the weld joint and the thermal sleeve was flushed with an inert gas mixture. The inert gas was used as a precaution against the formation of oxide inclusions on the I.D. surface. After completing the weld, the two small (< 1/4 in. diameter) chamfered purge gas holes in the thermal sleeve adapter were seal welded.

After the adapter to thermal sleeve and safe-end to nozzle welds were completed, the weld gap at the root of the thermal sleeve adapter to safe-end was measured. The thermal sleeve was then positioned to compensate for the weld shrinkage prior to welding, thereby minimizing the net residual tensile stress at the joint. Once the safe-end and thermal sleeve were welded in place, the closure spool piece was templated and machined for fitup, and the final two closure welds were made.

All safe-end welds were subjected to radiographic, liquid penetrant and ultrasonic examinations in accordance with Section XI of the ASME Code. The results of the radiographic and liquid penetrant examinations were evaluated in accordance with the acceptance criteria set forth in Section III of the ASME Code. The ultrasonic examinations performed after welding were done with the test sensitivity increased beyond the Code requirements and the results were recorded to serve as detailed baseline comparisions for future ISIs.

Structural and Mechanical Design

Analyses of the recirculation inlet nozzle replacement safe-end and thermal sleeve adapter for all loads, including seismic and thermal transient loadings, were performed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, with addenda through Summer 1967. The analyses also make use of the simplified elastic plastic analysis rules of NB 3228.2 of the 1977 ASME Code.

The recirculation inlet nozzle safe-end and thermal sleeve adapter were analyzed using an axisymmetric finite element computer model to obtain the maximum thermal gradients through the section during the postulated plant operational transients. The results of the thermal analysis were used in a similar axisymmetric finite element computer model to obtain the maximum thermal stress intensities. The applied piping loads from the original piping analysis, in conjunction with the internal pressure, were used in a shell of revolution computer program to obtain the primary stress intensities. The results of these analyses were combined for appropriate ASME Code evaluations. In additio , for the fatigue evaluation, stress concentration factors were applied to the areas where local geometric discontinuities exist.

An additional analysis, not required for Code evaluation, was performed to determine the residual welding stresses in the area of the thermal sleeve adapter to safe-end weld. The method used incorporates transient thermal analysis of a point heat source moving through a body, followed by an axisymmetric, elastic-plastic stress evaluation of the resulting temperature distributions. The analysis was performed as a time history employing actual welding parameters such as weld heat input, travel speed and number of passes and employing a temperature dependent material stress-strain relation. The analysis predicted a compressive residual stress in the region of the weld where the potential for a crevice existed. Compressive residual stresses will reduce the susceptibility to stress corrosion cracking. An analysis of weld residual stress also was performed for the original thermal sleeve to safe-end weld. The results of the analysis showed a high tensile residual stress developed in the crevice region where cracks initiated. Since initiation of IGSCC depends on a relatively high (with respect to the yield strength) tensile stress, the residual stress analysis of the original design helps to explain the observed cracking. By comparison, the analysis of the safe-end to thermal sleeve weld in the replacement design suggested that IGSCC will not be a problem since the compressive weld residual stress will result in a lower net tensile stress when combined with the other loadings.

Recognizing that the weld joining the thermal sleeve adapter to the safe-end may include a crevice-like configuration in the HAZ, an evaluation was performed to determine the potential safety affect of assuming a complete circumferential fracture of this weld which would separate the thermal sleeve from the safe-end. The evaluation showed that the thermal sleeve could move radially inward towards the vessel approximately two inches. Yielding would occur in the jet pump riser elbows and the riser brace. The maximum stress in the diffuser would be below the normal allowable value. Other than the localized elbow and brace yielding, no damage would be expected on the reactor vessel or internals, and the primary pressure boundary integrity would not be compromized.

The separation of the sleeve would cause some recirculation flow to leak into the vessel through the thermal sleeve annulus, reducing the flow through the jet pumps. Flow in the two affected jet pumps would drop to approximately 76% of rated flow, resulting in a reduction in core flow of approximately 3%. The reduction in jet pump flow would be detected in the control room by the core flow measurement indicators and reactor shut-down would be required by the Technical Specifications.

3.0 Evaluation

The analyses, design, fabrication and installation of the recirculation nozzle inlet safe-end and thermal sleeve adapter replacements are in accordance with accepted criteria as stated below. The structural loads including dynamic, static and thermal loadings applied by the attached piping and the acceptance criteria for the appropriate loading conditions are in accordance with the appropriate portions of the previously approved Duane Arnold Final Safety Analaysis Report. The allowable stress limits for the combined loading conditions are in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

The QA program used by the licensee meets or exceeds the requirements of the ASME Code and other criteria employed by the NRC including the requirements of Appendix B to 10 CFR 50 and is acceptable. The ultrasonic examinations of the safe-end welds described by Iowa Electric are adequate to serve as detailed baselines for future ISI. Final judgement of the non-destructive examinations (NDE) is reserved until the licensee provides details of the inspections in a report to the NRC. Also, the licensee will be required to submit a proposed inspection program, which can be considered an augmented ISI effort, to the NRC at least 90 days prior to the next scheduled refueling outage. As part of the submittal under review herein, the licensee proposed that one of the replacement safe-ends be examined by UT at each refueling outage until all eight safe-ends have been examined. However, until the results of the baseline UT examination have been reviewed by the staff and consultants, and the programs now underway involving Alloy 600 safe-ends at other BWR facilities have matured, the need for an augmented ISI program above that proposed by the licensee's program cannot be determined. As soon as this information is available we intend to inform the licensee of the nature of an acceptable ISI program.

The safe-end and thermal sleeve adapter replacements are fabricated from Alloy 600, the same type of material as the original safe-ends. A review of BWR operating experience showed that the safe-end cracking at Duane Arnold is the first example of IGSCC in Alloy 600 exposed at the BWR water environment. Moreover, because the original safe-end to thermal sleeve weld joint created a relatively long, sharp, crevice, the cracking actually occurred under unusual environmental considerations indigenous to the stagnant, contaminated, crevice conditions in an area of high residual stress. At other areas, there was no sign of distress at the welds, HAZs or base metal exposed to circulating water on any of the metallographic sections made during failure analysis. Further, laboratory tests have shown that very high tensile stresses (above yield) and tight crevice conditions, both of which were present in the original design, are significant factors in initiating stress corrosion cracking in Alloy 600. The currently available evidence from operating experience with Alloy 600 in several BWR's has shown that the Duane Arnold safe-end cracks are the only example of stress corrosion cracking.

The new safe-end design has removed from the primary pressure boundary the weld which caused both the tight crevice and the high residual stresses in the original design. Therefore, there is reasonable assurance that stress corrosion cracking will not occur in the pressure boundary of the new design. Although the annulus region between the safe-end and the thermal sleeve will restrict fluid flow, the machined gap will allow enough circulation to prevent the build-up of detrimental chemical species as would occur in a tight crevice.

We find that the safe-end replacement proposed by the licensee is acceptable and satisfies the applicable requirements of the Commission's Regulations.

In their letter of December 8, 1978, the licensee committed that "the DAEC will not be operated until repairs have been completed to the satisfaction of the NRC". There are three items to be resolved between the licensee and the NRC's Office of Inspection and Enforcement to complete the repair program - namely, (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) resolution of any discrepancies identified by the licensee's audit of the safe-end repair program. In view of the licensee's committment, it is evident that these matters will be resolved in a manner satisfactory to the NRC staff prior to resumption of operation of the Duane Arnold facility.

Leak Detection

As discussed in the Introduction, Duane Arnold detected the leakage from the cracked safe-end when it was less than 1 gpm. On June 14, 1978 the leakage increased from 1 gpm to 3 gpm and remained essentially constant at this rate until the plant was shutdown on June 17, 1978. As discussed in Section IIA of the report forwarded by the licensee's letter of December 8, 1978, "the increased leakage was immediately observed by six independent methods". The Technical Specifications for DAEC (Section 3.6.C) require that the facility be shutdown if reactor coolant leakage into the primary containment from unidentified sources exceeds 5 gpm. This specification is consistent with NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors". The Technical Specifications for DAEC require that the sump and air sampling systems (two of the systems for monitoring leakage) be operable during reactor power operation and that "reactor coolant leakage shall be checked by the sump and air sampling system and recorded at least once per day". In Sections III.A.2 and III.A.3 of their December 8th report, the licensee discussed the adequacy of the present Technical Specifications with respect to reactor coolant leakage. All reactor systems generally have a small amount of unidentified leakage, primarily from packing gland "weepage" from the hundreds of packed valves in containment. The leakage is usually lowest following startup from a refueling outage, since during the extended outages the packing glands are generally inspected and tightened. The total leakage generally increases toward the end of each fuel cycle. A review of DAEC operating experience showed that total leakage in containment from valve packing has reached about 3 gpm, on four occasions; leakage from this source is not a significant concern. A leakage rate of 3 gpm is about the lowest practical limit that could be incorporated in the Technical Specifications without requiring unnecessary shutdowns of the facility.

In 1975, NRC established a Pipe Crack Study Group, which published its report (NUREG-75/067) in October 1975. Based on operating experience since 1975 and the continuing investigation of the potential for IGSCC in sensitized stainless steel pipes, the NRC reconstituted its Pipe Crack Study Group in September 1978. The reconstituted group will consider the 1975 recommendations and recommend any changes deemed appropriate in light of experience and other new information developed since that time. The Group's report is expected to be available in early 1979. The applicability of the Pipe Crack Study Group's recommendations to DAEC will be evaluated, including the adequacy of the present standard limit of 5 gpm for unidentified leakage from BWR reactor systems, and any necessary changes made to the DAEC Technical Specifications. In the interim, the present DAEC Technical Specifications are considered adequate.

Occupational Radiation Exposure

The total occupational radiation exposure associated with repair of the safe-ends has been about 800 man-rem; the licensee estimates that about another 10 to 20 man-rem exposure will be incurred to complete the repair program. By maintaining water in the reactor vessel and through use of shielding from lead bricks and lead wool around the nozzle inlet to the reactor vessel, installation of lead blankets on the recirculation line piping and insertion of lead plugs in open pipes, the licensee reduced the general work area dose rates from approximately 20 R/hr to about 100 mr/hr. A fire resistant tent was constructed around the penetration area to prevent the release of airborne radioactivity from the repair operations into containment. The tent was equipped with a special smoke filter and a blower to maintain a negative pressure. The blower discharge was processed through a HEPA filter and charcoal filter to remove any particulates, aerosols and fission products before discharging into containment. Using mock-ups in the shop areas, all personnel were trained in operations to be conducted in containment (i.e., inspection, welding, cutting, installation of shielding, et al) in order to minimize the time spent in the radiation areas. Automatic cutting and welding was extensively used throughout the repair which further served to minimize occupational exposures. All personnel scheduled to work in radiation areas were given a minimum of 8 hours training in radiation safety, mask and clothing usage contamination control procedures, etc. Full protection clothing and full face respirators were required to work in the nozzle areas. Due to the personnel training and radiation control procedures, there were no over exposures to any personnel and no significant spread of contamination outside the immediate work areas. Although the total occupational exposures for the repair were relatively high when compared to plant operation without major maintenance activities, the exposures are common for similar primary system repair efforts. Primary system repair efforts involving BWR feedwater nozzles/ spargers have ranged from several hundred to over 1000 man-rems. Considering the complexity of the Duane Arnold activities, the exposure of 800 man-rems is reasonable.

We have reviewed the licensee's actions taken to reduce radiation exposures. Based on the licensee's use of training, mock-ups, automatic equipment and shielding, we conclude that the licensee has made a reasonable effort to maintain occupational exposures as low as reasonably achievable.

4.0 Environmental Considerations

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

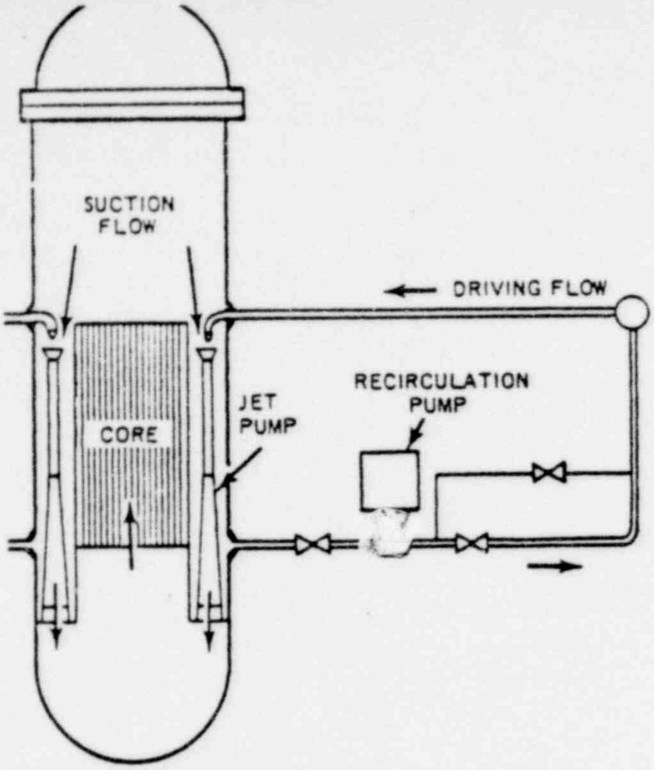
The exposures involved in the repair have been consistent with the range of exposures generally estimated for maintenance of large nuclear power reactors.

5.0 Conclusion

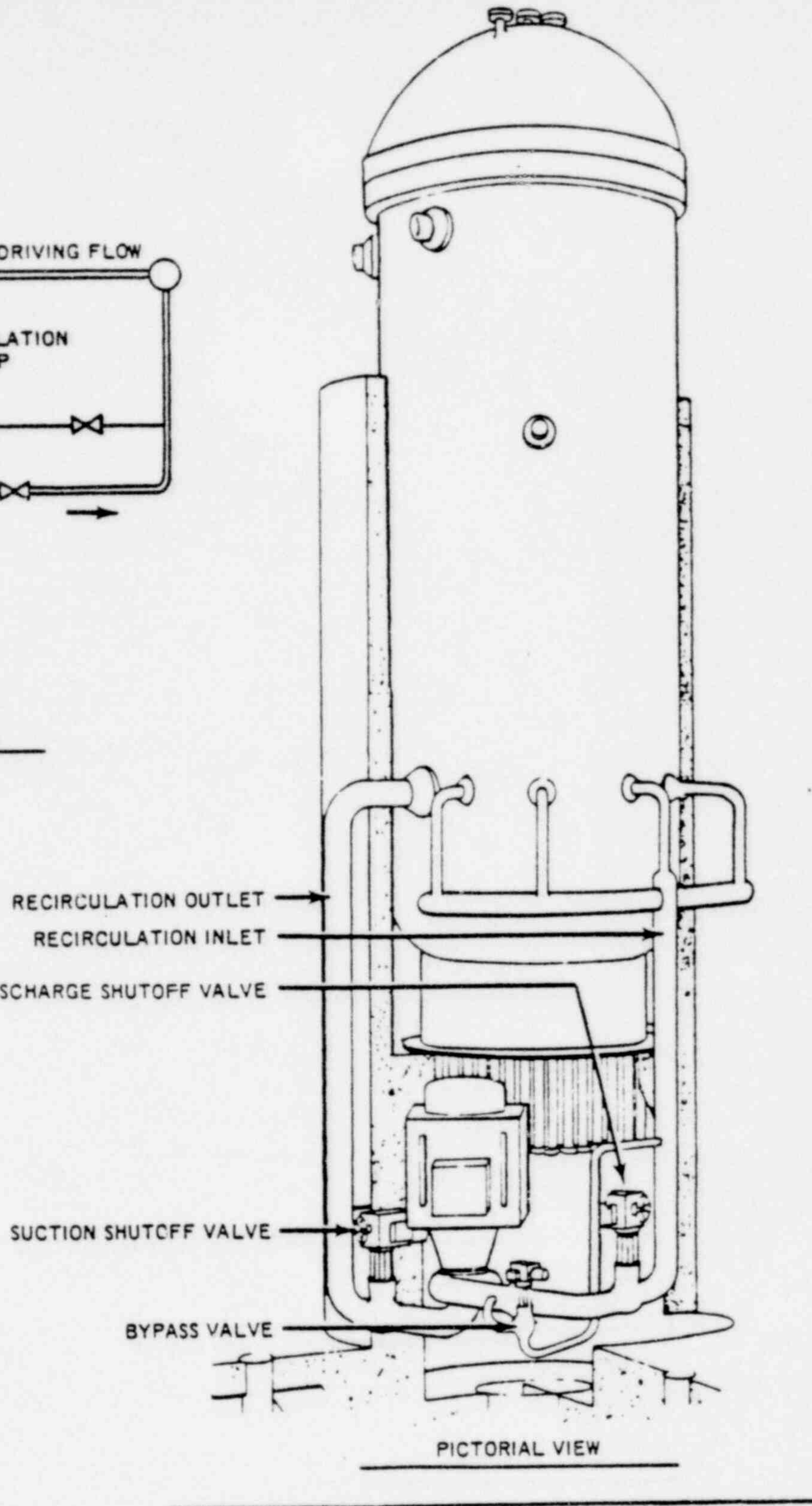
Basically, this action involves the replacement of a damaged component with a similar component, but one in which the design has been improved to eliminate the causes contributing to the damage to the safe-ends of the original design. Although the change is in one sense slight, moving the thermal sleeve attachment weld location away from the primary pressure boundary wall is an important improvement in eliminating the causes of stress corrosion cracking of the pressure boundary wall. This results in an overall improvement in plant safety.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 8, 1979

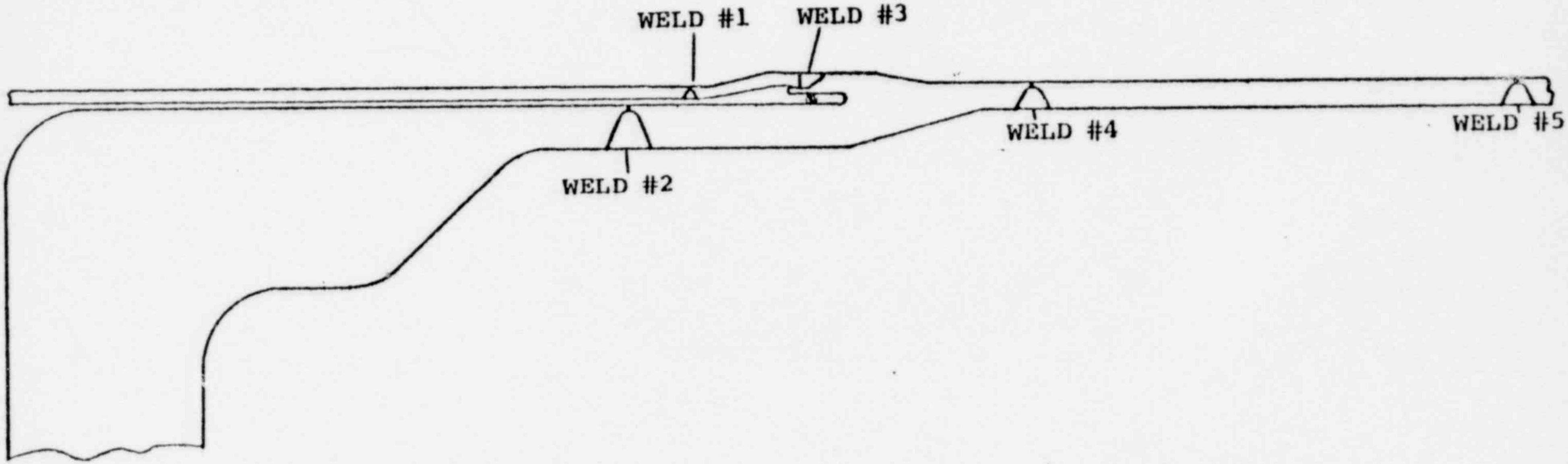


SIMPLIFIED SCHEMATIC



PICTORIAL VIEW

DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY
Recirculation System Elevation and Isometric Figure -1



<u>WELD SEQUENCE FOR REPLACEMENT SAFE END ASSEMBLY</u>

Figure 2