MARCH 14 1979

MEMORANDUM FOR: T. A. Ippolito, Chief, Operating Reactors Branch #3, DOR

FROM: R. J. Clark, Project Manager, Operating Reactors Branch #3, DOR

SUBJECT: SUMMARY OF MEETING WITH IOWA ELECTRIC ON FEBRUARY 21, 1979

On February 21, 1979, the NRC staff (NRR and IE) met with representatives of Iowa Electric Company and their consultants to discuss the acceptability of welds on the recirculation system inlet safe-ends, the recovery of the shield plug from nozzle B, the effect on plant operations of portions of the shield container remaining in the reactor, the licensee's proposed inservice inspection program, the effect of possible lead contamination on system materials and augmented leak detection procedures. The list of attendees is attached as Enclosure 1.

The basis for the discussion was a draft report prepared by the licensee. An outline of the subject material is attached as Enclosure 2. Following the meeting, the draft report was modified to incorporate the comments from the NRC staff and formally submitted to the NRC by the licensee's letter of February 22, 1979.

The significant items discussed were as follows:

1. The licensee presented their basis for why they conclude that the welds are acceptable under ASME Code requirements. Based on the UT and RT examinations of the welds, they also presented a stress and fatigue analysis of the welds. The extensive ultrasonic and radiographic examinations provice a baseline that permits them to detect any crack if one should develop.

2. There are 16 small carbon steel tabs and a fragment of the aluminum backing plate left in the reactor. These pieces broke off the cannister which held the 10 lead blocks used as a shield plug in nozzle B. The cannister was swept up the recirculation inlet riser into the jet pumps when they started the recirculation pumps. The aluminum will be dissolved within a short period of time at reactor operating temperature. General Electric analysis indicated that the Duane Arnold facility could be operated at a power level of 29% with complete

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blockage of a fuel bundle orifice without causing a departure from nucleate boiling in the bundle. The carbon steel tabs are too small to block an orifice. The fragment of aluminum is small enough that blocking of an orifice is very unlikely.

3. General Electric has conducted tests to determine the amount of lead that might be deposited on a stainless steel or Alloy 600 surface by heavily rubbing a piece of lead across the metal surfaces. The maximum weight of lead deposited was 1.8 mg/cm^2, equivalent to a thickness of about 0.08 mils. Autoclave tests of lead smeared samples showed that the lead would be oxidized (corroded) off the stainless steel and Alloy 600 surfaces within 48 hours at 500^{o}F at a dissolved oxygen level of 0.2 ppm. The lead smeared specimens of Type 304 stainless steel and Alloy 600 were stressed by bending prior to the autoclave tests. There was no evidence of stress corrosion cracking. If there is any lead smeared on the Inconel safe-ends, it will be oxidized off the surfaces within 48 hours at elevated temperatures. The resulting concentration in the reactor coolant will be less than 1 ppm. The lead oxide compounds will be removed in the reactor water cleanup system by the filter-demineralizers.

4. The licensee has analysed the plant water systems for lead. The analyses disclosed 10 ppb (0.010 ppm) in the makeup water, 80 ppb in the reactor coolant and 200 ppb in the condensate storage tank. Lead is soluble in water up to 20 ppm at room temperature. The temperature of the reactor system has not been above 212^{o}F since the safe-end repairs were completed. Tests conducted by International Nickel and Westinghouse show that it takes several weeks exposure to high concentrations of lead (in the order of 280 ppm) at high temperatures (500^{o}F to 600^{o}F) to initiate cracking of stressed Ni-Cr-Fe Alloy 600 (Inconel) specimens in the mill annealed or sensitized conditions.

5. The licensee has proposed to ultrasonically examine 50% of the three pressure boundry welds on the 8 safe-ends during the next refueling outage. The other 50% of the welds will be ultrasonically examined during the subsequent refueling outage (scheduled for 1981). This cycle will be repeated during the two subsequent refueling outages. Thus, 100% of the welds will be inspected within a two year period (vs. a 10 year period as required by the ASME Code).

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6. Besides the requirements in the Technical Specifications on unidentified leakage within containment, the DAEC has implemented the criteria as requested in IE Bulletin No. 74-10B.

R. J. Clark

R. J. Clark, Project Manager
Operating Reactors Branch #3
Division of Operating Reactors

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~~XXXXXX~~
See attached list

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DATE➤	3/ 12 /79					

ENCLOSURE 1

ATTENDANCE LIST

MEETING WITH IOWA ELECTRIC ON PIPE REPAIRS

FEBRUARY 21, 1979

IOWA ELECTRIC

H. W. Rehrauer
L. D. Root
E. Hammond
R. A. Meyer
K. V. Harrington

GENERAL ELECTRIC

F. P. Felini
G. Gordon
R. L. Gridley

NUTECH

P. Riccardella

LOWENSTEIN, NEWMAN, REIS & AXELRAD

R. Lowenstein
D. Bernstein

NRC - STAFF

B. Grimes
R. E. Johnson
J. R. Fair
K. R. Wichman
W. S. Hazelton
R. W. Klecker
D. Eisenhut*
V. S. Noonan
E. L. Jordan
W. J. Collins
D. Clark
T. Ippolito

*Denotes Part-Time

<u>ENCLOSURE 2</u>

I. INTRODUCTION

II. WELD ACCEPTABILITY

A. Weld Acceptance
B. Oxide Film Effects
C. Stress and Fatigue Evaluation
D. UT Inspectability

III. LEAD SHIELD PLUGS

A. Recovery of Nozzle B Plug
B. Retrieval of Other Shield Plugs
C. Lead Stress Corrosion Cracking
D. Lead Removal

IV. LOOSE PARTS ANALYSIS

V. CRACK GROWTH

A. Alloy 600 Stress Corrosion Crack Propagation
B. Stress Intensity & Crack Propagation Calculations
C. Maximum Flaw Acceptability
D. Leak Before Break

VI. INSERVICE INSPECTION

VII. LEAK DETECTION

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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President
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G. Knighton
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D. Ziemann
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