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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

June 25, 1979

IE Bulletin No. 79-13

CRACKING IN FEEDWATER SYSTEM PIPING

Description of Circumstances:

On May 20, 1979, Indiana and Michigan Power Company notified the NRC of cracking in two feedwater lines at their D. C. Cook Unit 2 facility. The cracking was discovered following a shutdown on May 19 to investigate leakage inside containment. Leaking circumferential cracks were identified in the 16-inch feedwater elbows adjacent to two steam generator nozzle elbow welds. Subsequent radiographic examination revealed crack indications in all eight steam generator feedwater lines at this location on both Units 1 and 2.

On May 25, 1979, a letter was sent to all PWR licensees by the Office of Nuclear Reactor Regulation which informed licensees of the D. C. Cook failures and requested specific information on feedwater system design, fabrication, inspection and operating histories. To further explore the generic nature of the cracking problem, the Office of Inspection and Enforcement requested licensees of PWR plants in current outages to immediately conduct volumetric examination of certain feedwater piping welds.

As a result of these actions, several other licensees with Westinghouse steam generators reported crack indications. Southern California Edison reported on June 5, 1979, that radiographic examination revealed indications of cracking in feedwater nozzle-to-piping welds on two of three steam generators of San Onofre Unit 1. On June 15, 1979, Carolina Power and Light reported that radiography showed crack indications in similar locations at their H. B. Robinson Unit 2. Duquesne Power and Light confirmed on June 18, 1979, that radiography has shown cracking in their Beaver Valley Unit 1 feedwater piping to vessel nozzle weld. Public Service Electric and Gas Company reported on June 20, 1979 that Salem Unit 1 also has crack indications. Wisconsin Public Service company decided on June 20, 1979 to cut out a feedwater nozzle to pipe weld which contained questionable indication, for metallurgical examination. As of June 22, 1979 and since May 25, 1979 seven other PWR facilities have inspected the feedwater nozzle-to-pipe welds without finding cracking indications.

The feedwater nozzle-to-pipe configurations for D.C. Cook and for San Onofre are shown on the attached figures 1 and 2. A typical feedwater pipe-to-nozzle weld joint detail showing the principal crack locations for D.C. Cook and San Onofre are shown on the attached figure 3.

On March 17, 1977, during heat-up for hot functional testing of Diablo Canyon Unit 1, a leak was discovered in the vessel nozzle-to-pipe butt weld joining the 16-inch diameter feedwater piping to steam generator 1-2. Subsequent nondestructive examination of all nozzle welds by radiography and ultrasonics revealed an approximate 6-inch circumferential crack originating in the weld root heat-affected zone of the leaking nozzle weld. The cause of this cracking was identified as either corrosion-fatigue or thermal fatigue initiating at small cracks probably induced by the welding and postweld heat treatment cycles. The system was repaired by replacing with a piping component employing greater controls on the welding including maintaining preheat temperature until postweld heat treatment.

Cracked weldments have been removed from the D. C. Cook Units 1 and 2 and San Onofre Unit 1 feedwater systems for extensive metallurgical investigation by Westinghouse. Based on preliminary analysis, Westinghouse stated the D. C. Cook failure may be "fatigue assisted by corrosion." The San Onofre cracking was stated to be characteristic of "stress assisted corrosion."

The cracking experienced at Diablo Canyon, D. C. Cook and San Onofre would appear to have different cause - effect relationships which are not fully understood at this time.

The potential safety consequences of the cracking is an increased likelihood of a feedwater line break in the event of a seismic event or water hammer. A feedwater line break results in a loss of one of the mechanisms of heat removal from the reactor core and would result in release of stored energy from the steam generator into containment. Although a feedwater line break is an analyzed accident, the identified degradation of these joints in the absence of a routine inservice inspection requirement of these feedwater nozzle-to-piping welds is the basis for this Bulletin.

Actions to be Taken by Licensees:

For all pressurized water reactor facilities with an operating license:

1. Facilities which have steam generators fabricated by Westinghouse or Combustion Engineering that have not conducted volumetric examination of feedwater nozzles since May 1979 shall complete the inspection program described below at the earliest practical time but no later than 90 days after the date of this Bulletin.

   a. Perform radiographic examination, supplemented by ultrasonic examination as necessary to evaluate indications, of all feedwater nozzle-to-piping welds and of adjacent pipe and nozzle areas (a distance equal to at least two wall thicknesses). Evaluation shall be in accordance with ASME Section III, Subsection NC, Article NC-5000. Radiography shall be performed to the 2T penetrameter sensitivity level, in lieu of Table NC-5111-1, with systems void of water.

b. If cracking is identified during examination of the nozzle-to piping weld, all feedwater line welds up to the first piping support or snubber and high stress points in containment shall be volumetrically examined in accordance with 1.a. above. All unacceptable code discontinuities, other than cracking, shall be subject to repair unless justification for continued operation is provided.

c. Perform a visual inspection of feedwater system piping supports and snubbers in containment to verify operability and conformance to design.

2. All pressurized water reactor facilities shall perform the inspection program described below at the next outage of sufficient duration or at the next refueling outage after the inspection required by item 1.

a. For steam generator designs having a common nozzle for both main and auxiliary (emergency) feedwater systems, perform volumetric examination of all feedwater nozzle-to-pipe weld areas and all feedwater pipe weld areas inside containment in accordance with item 1 above.1/ In addition, conduct an examination of welds connecting auxiliary feedwater piping to the main feedwater line outside containment. This examination should include an area of at least one pipe diameter on the main feedwater line downstream of the connection.

b. For steam generator designs with separate nozzles for main feedwater and auxiliary feedwater, perform volumetric examination (in accordance with item 1 above) of all welds inside containment and upstream of the external ring header or vessel nozzle for each steam generator. If an external ring header is employed, also inspect all welds of one inlet riser on each feed ring of each steam generator.1/

c. Perform a visual inspection of all feedwater system piping supports and snubbers in containment to verify operability and conformance to design.

3. Identification of cracking indications in feedwater nozzle or piping weld areas in one unit of a multi-unit facility shall require shutdown and inspection of other similar units which have not been inspected since May 1979, unless justification for continued operation is provided.

1/ Welds in the feedwater system, (other than the feedwater nozzle-to-pipe welds) that have been examined since May 1979 need not be re-examined.

4. Any cracking or other unacceptable code discontinuities identified shall be reported to the Director of the appropriate NRC Regional Office within 24 hours of identification.

5. Provide a written report to the Director of the appropriate NRC Regional Office within 20 days of the date of this Bulletin addressing the following:

   a. Your schedule for inspection if required by item 1.

   b. The adequacy of your operating and emergency procedures to recognize and respond to a feedwater line break accident.

   c. The methods and sensitivity of detection of feedwater leaks in containment.

6. A written report of the results of examinations, in accordance with requests by Regional Offices preceding this Bulletin and with Bulletin item 1 and 2 including any corrective measures taken, shall be submitted within 30 days of the date of this Bulletin or within 30 days of completion of the examination, whichever is later, to the Director of the appropriate NRC Regional Office with a copy to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072), clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Attachments:
Figures 1, 2 and 3

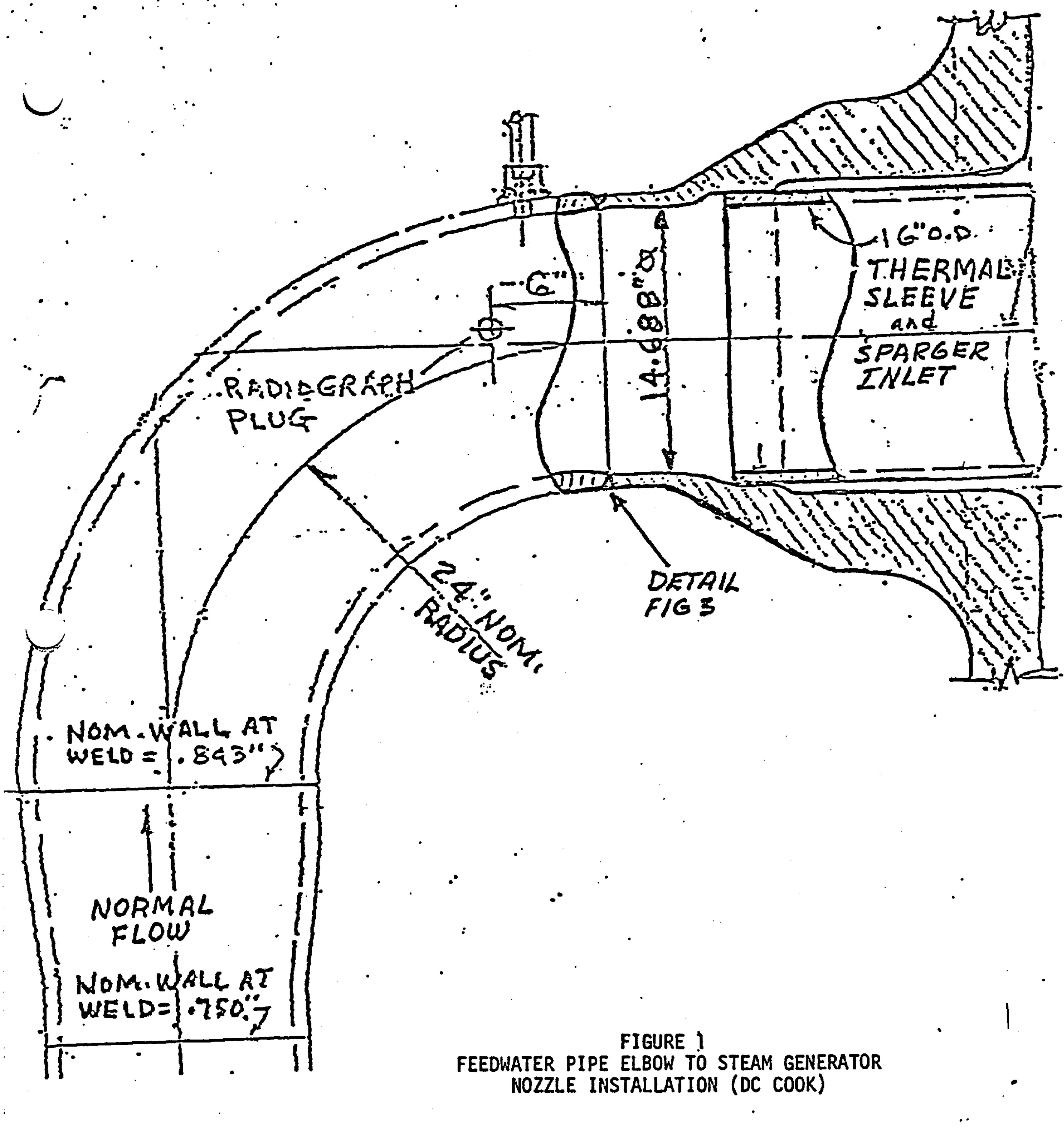

FIGURE 1
FEEDWATER PIPE ELBOW TO STEAM GENERATOR
NOZZLE INSTALLATION (DC COOK)

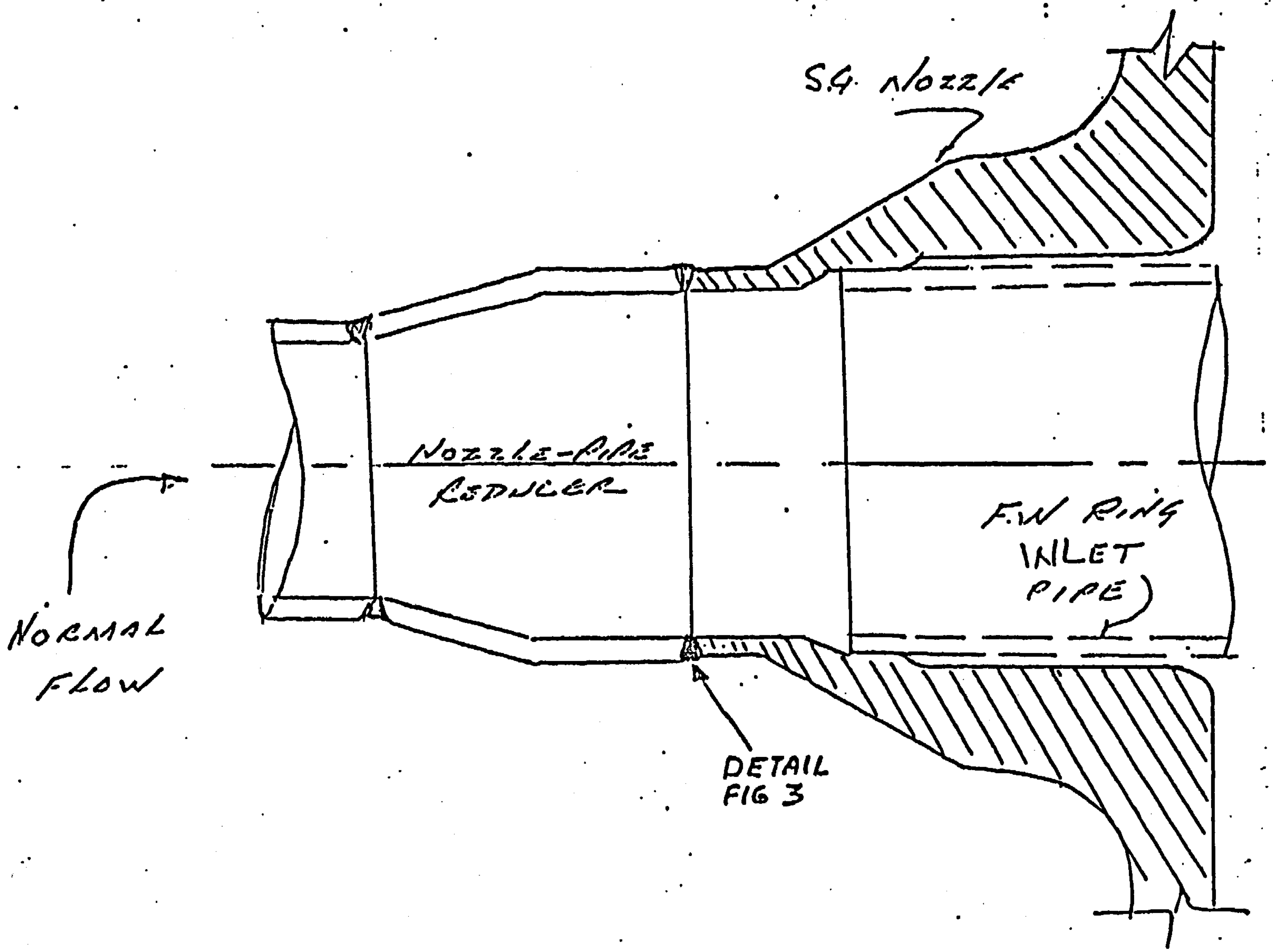

FIGURE 2
FEEDWATER PIPE REDUCER TO STEAM GENERATOR
NOZZLE INSTALLATION (SAN ONOFRE)

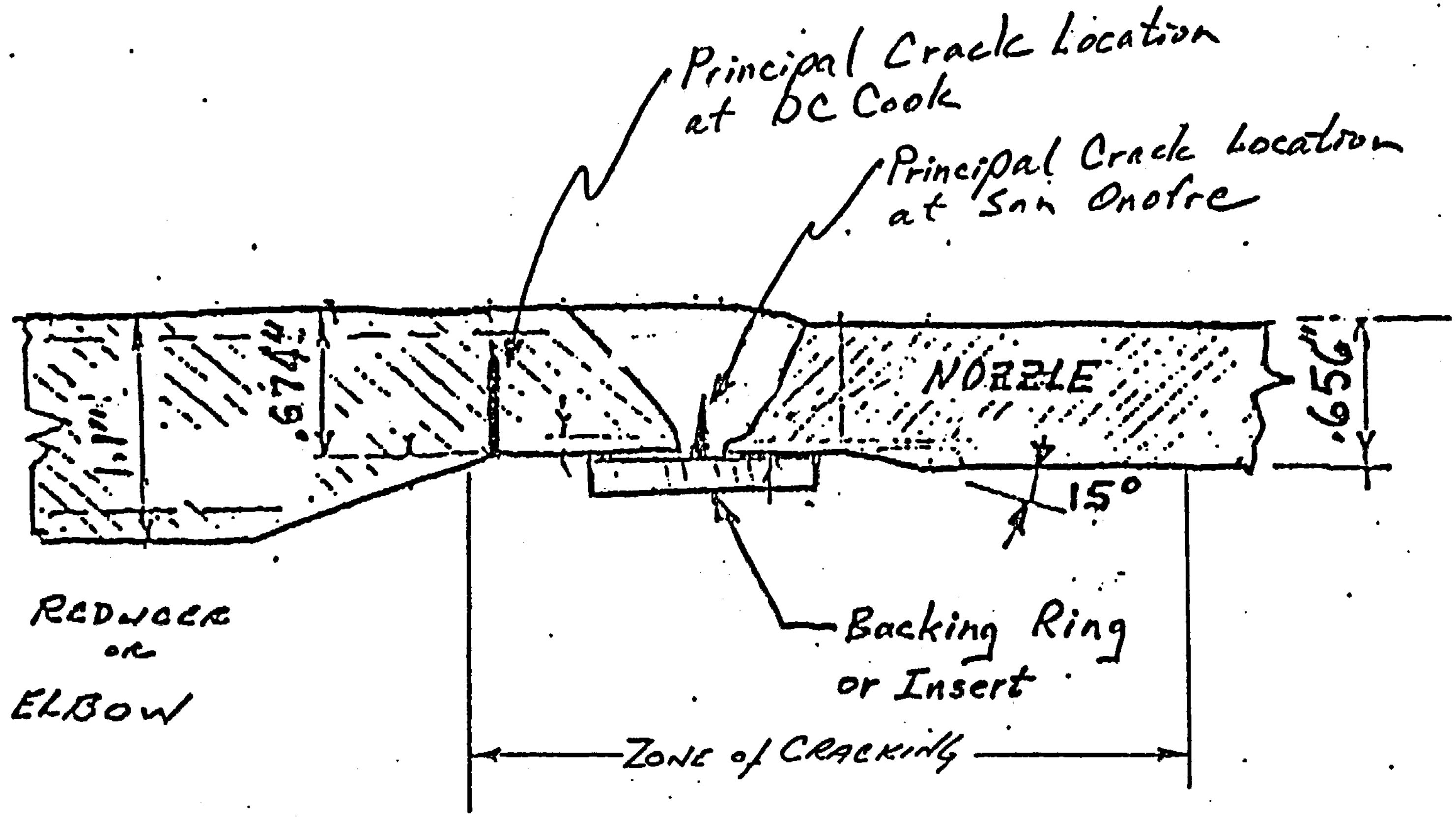

FIGURE 3
TYPICAL FEEDWATER PIPE TO NOZZLE
WELD JOINT DETAIL
(W ,CE)

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LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an OL or a CP
79-12	Short Period Scrams at BWR Facilities	5/31/79	All GE BWR Facilities with an OL
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an OL or a CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL or CP
79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP
79-06B	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident	4/14/79	All Combustion Engineering Designed Pressurized Water Power Reactor Facilities with an Operating Licensee

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Bulletin No.	Subject	Date Issued	Issued To
79-06A (Rev 1)	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06A	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/14/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/11/79	All Pressurized Water Power Reactors with an OL except B&W facilities
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL
79-05	Nuclear Incident at Three Mile Island	4/2/79	All Power Reactor Facilities with an OL and CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP

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Bulletin No.	Subject	Date Issued	Issued To
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Welds Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/70	All Power Reactor Facilities with an OL or CP
79-01A	Environmental Qualification of Class 1E Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	6/6/79	All Power Reactor Facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP
78-14	Deterioration of Buna-N Component In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP
78-13	Failures in Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges

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Bulletin No.	Subject	Date Issued	Issued To
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP