Central File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

OCT 17 1979

In Reply Refer To:
RII:JPO
50-438, 50-439
50-259, 50-260
50-296, 50-518
50-519, 50-520
50-521, 50-553
50-554, 50-328
50-566, 50-567

Tennessee Valley Authority
Attn: H. G. Parris
Manager of Power
500A Chestnut Street Tower II
Chattanooga, Tennessee 37401

Gentlemen:

The enclosed Bulletin No. 79-13, Revision 2 is forwarded to you for information. No written response is required. If you desire additional information regarding this matter, please contact this office.

Sincerely,

James P. O'Reilly

James P. O'Reilly
Director

Enclosures:
1. IE Bulletin No. 79-13, Revision 2
2. Listing of IE Bulletins
Issued in Last Six Months

7911070 210

A02

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SSINS: 6830
Accession No.:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

October 16, 1979

IE Bulletin No. 79-13
Revision 2

CRACKING IN FEEDWATER SYSTEM PIPING

Description of Circumstances:

This revision to IE Bulletin No. 79-13 is based on the results of the radiographic examinations and ongoing investigation of the subject problem to date since the initial Bulletin was issued. The revision reduces in scope the number and extent of the piping system welds required to be examined. The requirements for reporting and action time frame remain unchanged. R2

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-pipe 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.

<u>NOTE</u>: R1 and R2 indicates lines revised or added.

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 nozzle-to-pipe 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.

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-pipe welds formed the basis of this Bulletin.

To date the radiographic examinations, supplemented by ultrasonic methods, have identified cracking in the steam generator nozzle to feedwater piping weldments at the following W, and C. E. plants.

D. C. Cook Units 1 & 2
Diablo Canyon*
San Onofre Unit 1
H. B. Robinson Unit 2
Beaver Valley Unit 1
Kewaunee
Point Beach Unit 2
Salem Unit 1
Surry Unit 1
R. E. Ginna
Millstone Unit 2
Palisades**
Yankee Rowe**
Maine Yankee

* Found during hot functional testing
** Confirmatory evaluation incomplete

R2

An extensive metallurgical investigation has been conducted by Westinghouse on a substantial number of cracked weldments removed from the above plants. Results of the metallurgical analysis lead to the conclusion that a corrosion fatigue phenomenon is the probable failure mechanism, except for the San Onofre piping which has been characteristized as stress assisted corrosion.

In parallel with the above ongoing analysis, the feedwater piping at D. C. Cook, H. B. Robinson, R. E. Ginna, Salem 1 and other plants have been instrumented (Thermocouples, accelerometers, strain gages, and transducers) to collect data on the potential forcing functions contributing to cracking under steady state and transient conditions. Preliminary unchecked results of temperature data has identified cyclic thermal gradients may exist due to stratified feedwater temperature conditions in the feedpipe weld region during zero and low power operations. This gradient tends to support the fatigue aspect of the postulated failure mechanism. No further unexpected operation loading or forcing functions have been identified by other instrumentation.

In regard to B&W plants a total of 95 welds in the main and separate auxiliary feedwater piping, risers and, steam generator nozzles regions have been examined at Crystal River Unit 3 and Davis Besse. No indications of a cracking problem was found. R2

In view of the findings to date, the revised inspections outlined below is considered acceptable to meet this intent of IE Bulletin No. 79-13. R2

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 Bulletin No. 79-13.

   a. Perform radiographic examination, supplemented by ultrasonic examination as necessary to evaluate indications, of all feedwater nozzle-to-pipe 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. In the event cracking is identified during examination of the nozzle-to-pipe weld, all feedwater line welds up to the first piping support or snubber outboard of the nozzle shall be volumetrically examined in accordance with 1.a above. All unacceptable code discontinuities 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 with a common nozzle for both main and auxiliary feedwater systems, perform volumetric examination of the feedwater nozzle-to-pipe welds, the feedwater piping welds to the first support, and the feedwater line-to-containment penetration welds in accordance with Item 1 above. In addition, examine an area of at least one pipe diameter of the main feedwater line downstream at the auxiliary feedwater to main feedwater connection.

R2

b. For steam generator designs utilizing auxiliary feedwater systems connected by means of welded nozzle connections, perform volumetric examination of all auxiliary feedwater nozzle to piping welds and the first adjacent outboard pipe-to-pipe welds (risers) in accordance with item 1 above.

For designs utilizing auxiliary feedwater systems connected to the steam generator by means of bolted flange connections, perform volumetric examination of the flanged nozzle to piping and first outboard pipe-to-pipe welds (risers) in accordance with item 1 above.

The examinations specified in 2.b above are not required provided that during startup, hot standby or cold shutdown operations, the feedwater level within the steam generator is maintained essentially constant and no intermittent cold auxiliary feedwater injection is utilized; i.e., auxiliary feedwater injection where used, is preheated during the forementioned operating modes.

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.

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 the orginal Bulletin (June 25, 1979) 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 examination, in accordance with requests by Regional Offices preceding this Bulletin and with Bulletin items 1 and 2 including any corrective measures taken, shall be submitted within 30 days of the date of the original Bulletin No. 79-13 (June 25, 1979) 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.

Actions to be Taken by Designated Applicants for Operating Licenses:

1. On completion of the hot functional testing program and prior to fuel loading, perform the inspections described in item 1 above.

2. During the first refueling outage, perform the inspections described in item 2 above. R1

3. Submit reports as described in Items 4, 5, amd 6 above based on the date of Revision 1 to Bulletin No. 79-13 (August 30, 1979)

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

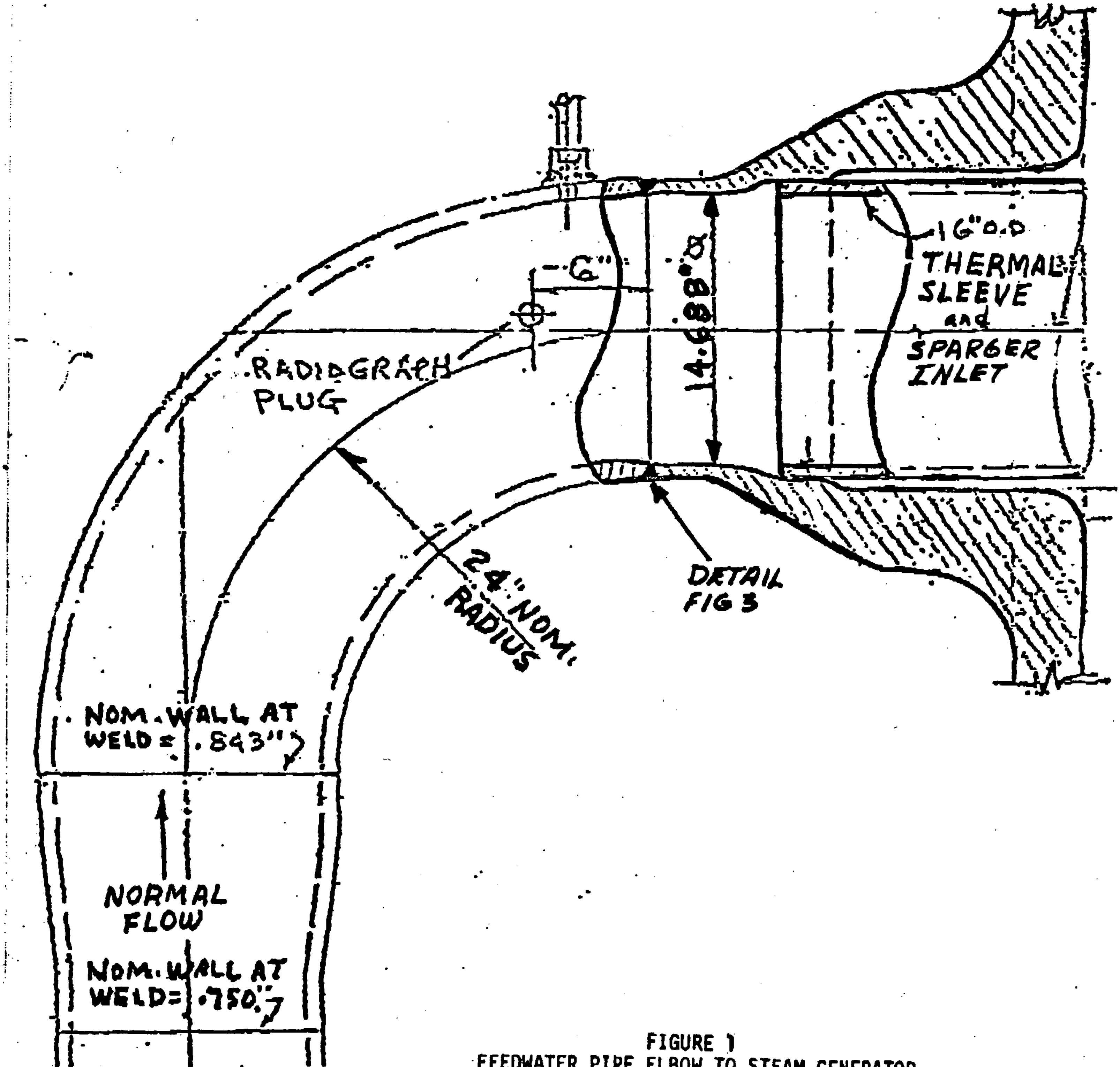

FIGURE 1
FEEDWATER PIPE ELBOW TO STEAM GENERATOR
NOZZLE INSTALLATION (DC COOK)

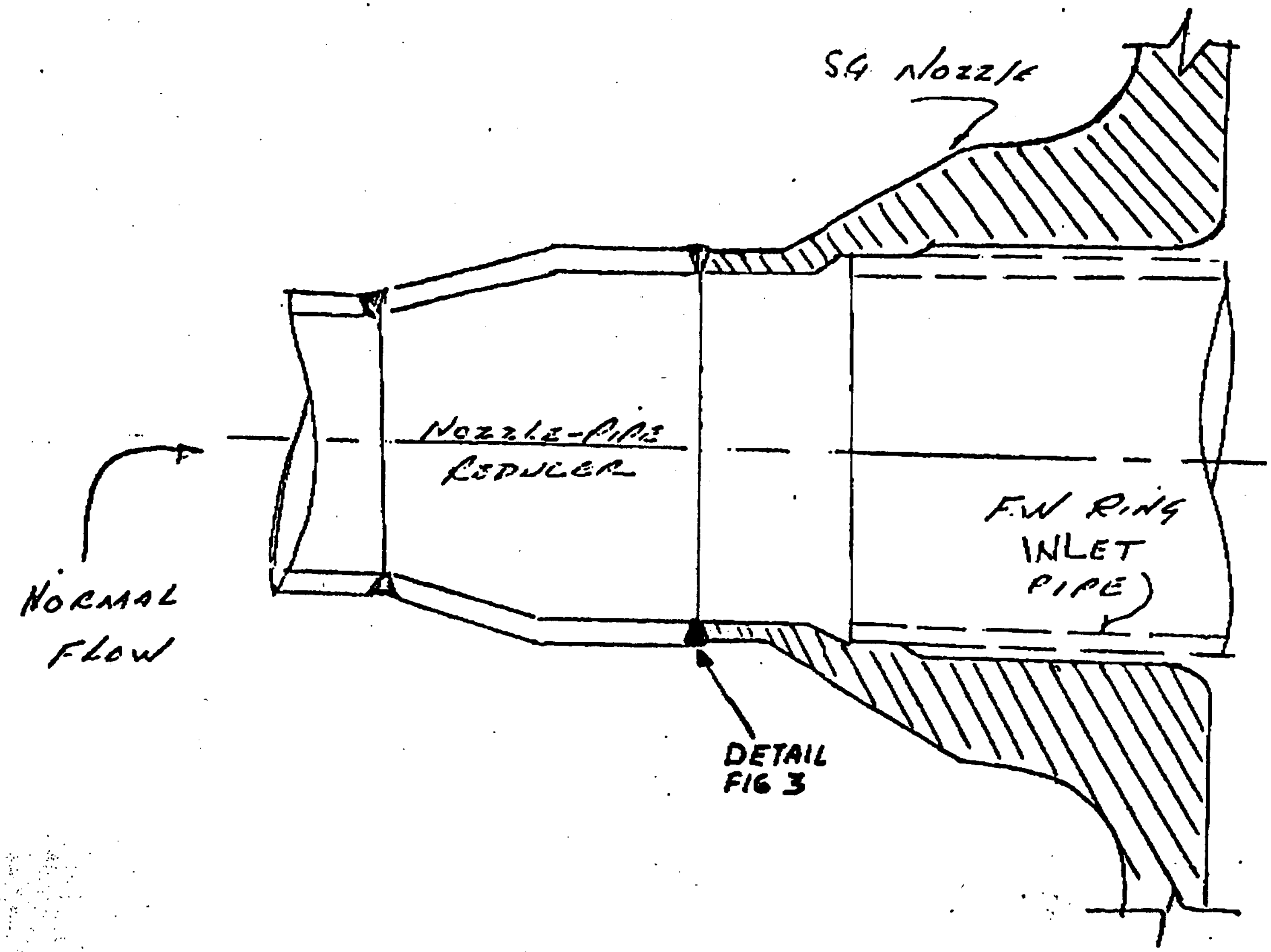

FIGURE 2
FEEDWATER PIPE REDUCER TO STEAM GENERATOR
NOZZLE INSTALLATION (SAN ONOFRE)

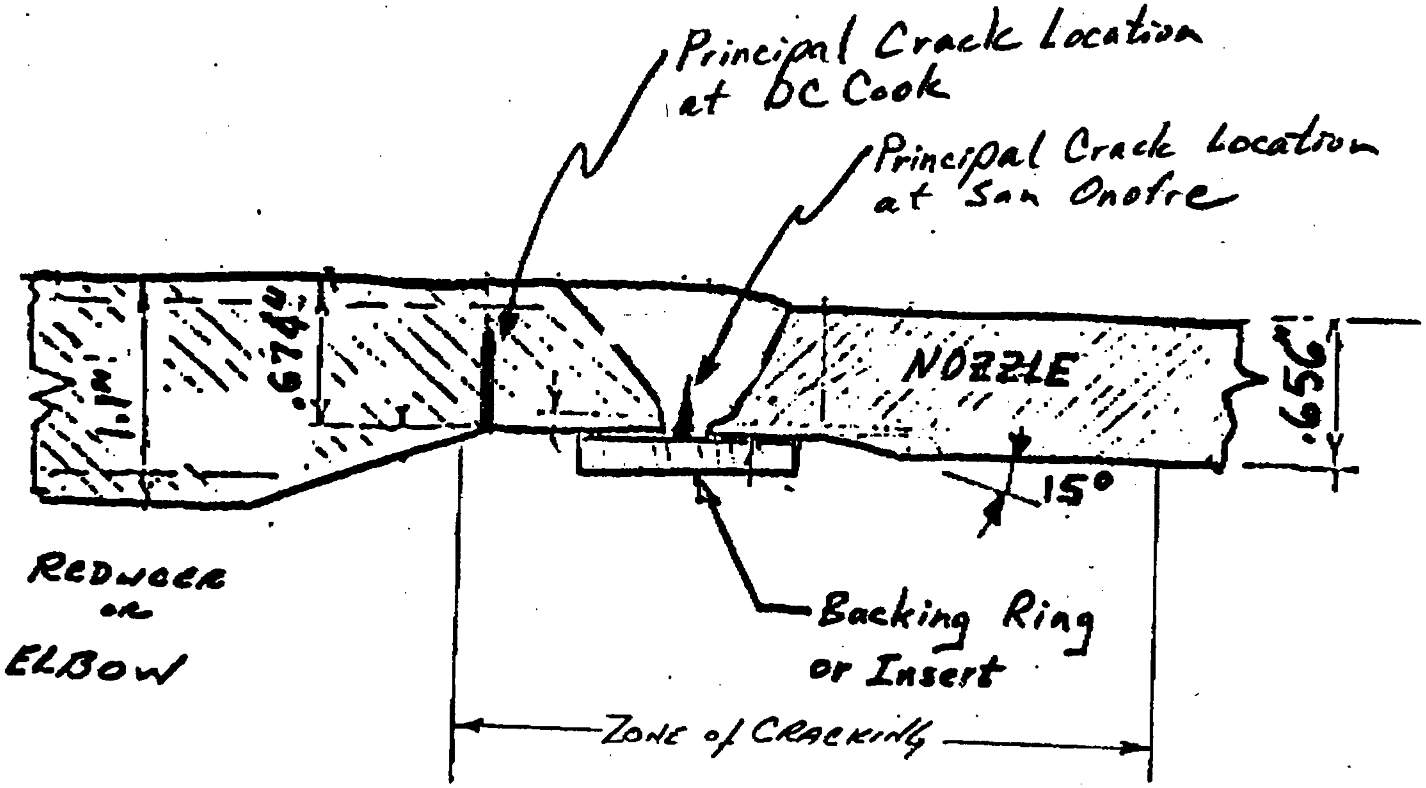

FIGURE 3
TYPICAL FEEDWATER PIPE TO NOZZLE
WELD JOINT DETAIL
(W ,CE)

DESIGNATED APPLICANTS FOR OPERATING LICENSES

Salem 2

North Anna 2

Diablo Canyon 1 & 2

Sequoyah 1 R1

McGuire 1

San Onofre 2

Summer

Watts Bar 1 & 2

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-24	Frozen Lines	9/27/79	All Power Reactor Facilities with an OL or a CP
79-23	Potential Failure of Emergency Diesel Generator Field Exciter Transformer	9/12/79	All Power Reactor Facilities with an OL or a CP
79-22	Possible Leakage of Tubes of Tritium Gas Used in Timepieces for Luminosity	9/5/79	Each Licensee who Receives Tubes of Tritium Gas in Time-pieces for Luminosity
79-21	Temperature Effects on Level Measurements	8/13/79	All PWR's with an Operating License
79-20	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Materials Licensees who did not receive Bulletin No. 79-19
79-19	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Power and Research Reactors with OLs, Fuel Facilities except uranium mills, and certain materials licensees
79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas	8/7/79	All OLs for Action All CPs for Information
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWRs with Operating License
79-16	Vital Area Access Controls	7/26/79	All Holders of and applicants for Power Reactor Operating Licenses who Antici-page loading fuel prior to 1981
79-15 (Supp. 1)	Deep Draft Pump Deficiencies	7/18/79	All Power Reactor Licensees with a CP and/or OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Licensees with a CP and/or OL
79-14 (Correction)	Seismic Analyses for As-Built Safety-Related Piping System	7/27/79	All Power Reactor Facilities with an OL or a CP
79-14 (Supp. 2)	Seismic Analyses for As-Built Safety-Related Piping System	9/7/79	All Power Reactor Facilities with an OL or a CP
79-14 (Rev. 1)	Seismic Analyses for As-Built Safety-Related Piping System	7/18/79	All Power Reactor Facilities with an OL or a CP
79-14	Seismic Analyses for As-Built Safety-Related Piping System	7/2/79	All Power Reactor Facilities with an OL or a CP
79-13 (Rev. 2)	Cracking in Feedwater System Piping	10/17/79	All PWR's with an Operating License
79-13 (Rev. 1)	Cracking in Feedwater System Piping	8/30/79	All PWR's with an Operating License
79-13	Cracking in Feedwater System Piping	6/25/79	All PWR's with an OL for action. All BWRs with a CP for information
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
79-06C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-05C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL
79-02 (Rev. 1) (Supp. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	8/20/79	All Power Reactor Facilities with an OL or a CP
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
79-01A	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 a CP