UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 14, 1994

Docket Nos. 50-269, 50-270, 50-287,
50-289, 50-302, 50-313
and 50-346

GROUP: BABCOCK & WILCOX OWNERS' GROUP

SUBJECT: SUMMARY OF MEETING HELD ON MAY 27, 1994, TO DISCUSS ULTRASONIC TESTING IN LIEU OF SURFACE EXAMINATION OF REACTOR VESSEL NOZZLE WELDS

REFERENCES:
1. Inspection Report 50-302/89-21 issued by the NRC on August 25, 1989
2. Letter to U.S. NRC from Duke Power Company dated November 18, 1993
3. Letter from U.S. NRC to Duke Power Company dated March 1, 1994
4. Letter to U.S. NRC from The B&W Owners' Group dated April 13, 1994

On May 27, 1994, NRC staff members met at Rockville, Maryland, with members of the Babcock & Wilcox Owners' Group (B&WOG) at a public meeting to discuss an alternative to surface examination of reactor vessel nozzle welds. A list of people attending the meeting is provided in Enclosure 1.

The B&WOG made presentations on the demonstration history, demonstration objectives, analysis and calculations to support flaw size used in the demonstration. Slides presented at the meeting, except for those designated as proprietary, are included in Enclosure 2. The slides designated as proprietary were previously submitted under affidavit on October 13, 1993.

During the late 1980's and early 1990's, relief requests were conditionally approved in safety evaluation reports (SERs) with the provision that the licensee's demonstrate the ultrasonic examination techniques and procedures applied on the inside diameter surface to detect surface connected defects on the outside diameter in the circumferential orientation in a laboratory test block. The defects were supposed to be cracks and not machine notches. A demonstration conducted in 1989 on a core flood nozzle with machined notches

did not detect flaws in Inconel material. In 1991, other B&W plants requested that B&WOG conduct a demonstration that could be used for all B&W plants (a generic relief request). A second demonstration conducted in 1993 on core flood nozzle and reactor coolant nozzle mockups was capable of detecting cracks larger than the maximum acceptable for the surface examination method.

In a telephone conversation on November 10, 1993, the NRC stated that the demonstration was inadequate to satisfy the condition stipulated in the SERs. A licensee submittal (Reference 2) described the reasons for believing the 1993 demonstration fulfilled the conditions stipulated by the SERs and requested the NRC reconsider its position on the 1993 demonstration. The NRC reaffirmed its position and described its concerns in additional questions to the licensee, (Reference 3). The response to these additional questions was submitted (Reference 4). The subject meeting was requested to clarify what questions or concerns remained.

B&WOG stated that for the 1993 demonstration, one official set of data collected by one individual, qualified per ASME Code, was used to identify all the flaws. The procedure that was used addressed the essential variables. The procedures that would be used for plant specific examinations would contain these same essential variables, but may have administrative changes.

Although B&WOG did not have a precise minimum detection surface flaw length, they estimated that the indexing used in the 1993 demonstration would pick-up a 1/2-inch long flaw in at least 4 scans. This would be the same for both the reactor coolant and core flood systems.

In reviewing the fracture mechanics calculation for the core flood nozzle submitted in Reference 4, B&WOG identified at least one other plant with a 40 percent higher stress value.

In summary, the NRC is amenable to granting relief from performing surface examinations of reactor vessel nozzles to reduce radiation exposure to non-destructive examination personnel. The NRC understands that the demonstration held in 1993 was not to demonstrate an equivalency to a surface examination. The demonstration was a capability demonstration to show that flaws located in the test block could be detected and was not intended to be a reliability demonstration. The presentation clarified that a fracture mechanics analysis (based on minimum detectable flaw size and maximum propagation rates) plus a reliability safety factor, combined with ultrasonic testing may prove to be an acceptable alternative and meet the intent in the SER. However, the NRC would need a detailed submittal on the fracture mechanics analysis including the loadings, basis for any assumptions, and the inclusion of margins and uncertainties. The additional information needs to specify which nozzles (size, location/identification, plant) are included and state if the analyses are bounding for all plants.

At the conclusion of the meeting, the B&WOG stated they would submit additional information in four to six weeks. The NRC said this should allow sufficient time to support the first unit affected that needs the exemption by

February 1995. The NRC encouraged B&WOG to work with ASME Code Committees to allow this alternative.

Original signed by

Linda L. Gundrum, Acting Project Manager
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Division of Reactor Projects III/IV
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Enclosures:
1. List of Attendees
2. Presentation Slides (nonproprietary)

cc w/enclosures:
See next page

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DATE	6/9/94	6/9/94	6/10/94	6/13/94

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GWest (13-E-14)
GKalman (13-H-3)
RHernan (14-C-12)
PRush, Region II
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ENCLOSURE 2

DEMONSTRATION HISTORY

1989 Demonstration

-- Proved shear wave technique was inadequate for exams performed in 1985 for OD surface connected reflectors.

-- Proved longitudinal wave technique is capable of detecting OD surface connected reflectors.

1993 Demonstration

-- Proved the longitudinal wave technique is capable of detecting OD surface connected thermal fatigue flaws.

-- Proved the technique used provided accurate flaw sizing information.

RELIEF REQUESTS

Individual B&WOG utilities have submitted relief requests to the NRC and have received SERs with the following compliance requirements:

OCONEE AND TMI-1

1. Examine entire volume of weld and heat affected zone.

2. Demonstrate capability to detect circumferential OD surface connected cracks, not machined notches.

CRYSTAL RIVER UNIT 3

Perform a demonstration to show the capability of the ultrasonic examination to detect and size surface flaws.

ARKANSAS NUCLEAR ONE

Demonstrate the ability to detect OD surface flaws from the ID.

ISI INTERVAL END DATES FOR RELIEF REQUESTS

Arkansas Nuclear One Unit 1	February 1995
Crystal River Unit 3	March 1997
Oconee Unit 1	July 1994 [1, 3]
Oconee Unit 2	December 1994 [1, 3]
Oconee Unit 3	December 1994 [1, 3]
TMI - 1	April 2001 [2]

1 Examination already completed

2 Two of ten examinations completed

3 Relief granted to extend subject inspection (Unit 1--10/95, Unit 2--1/96, Unit 3--5/95)

B&WOG MEETING OBJECTIVES

- Obtain NRC concurrence that we have successfully demonstrated compliance with NRC requests as stated in the individual utilities SERs.

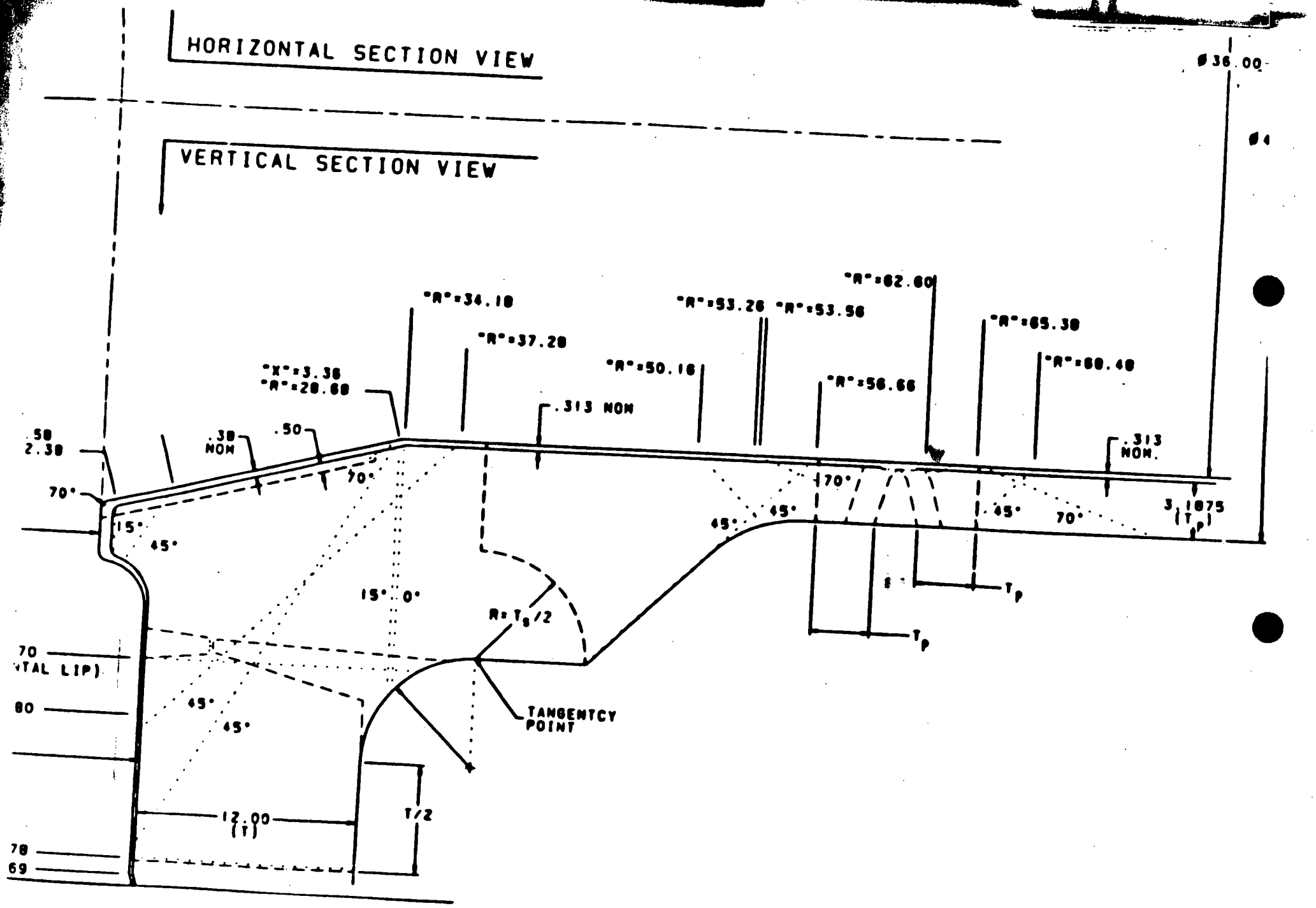
B&WOG Conclusion & Summary

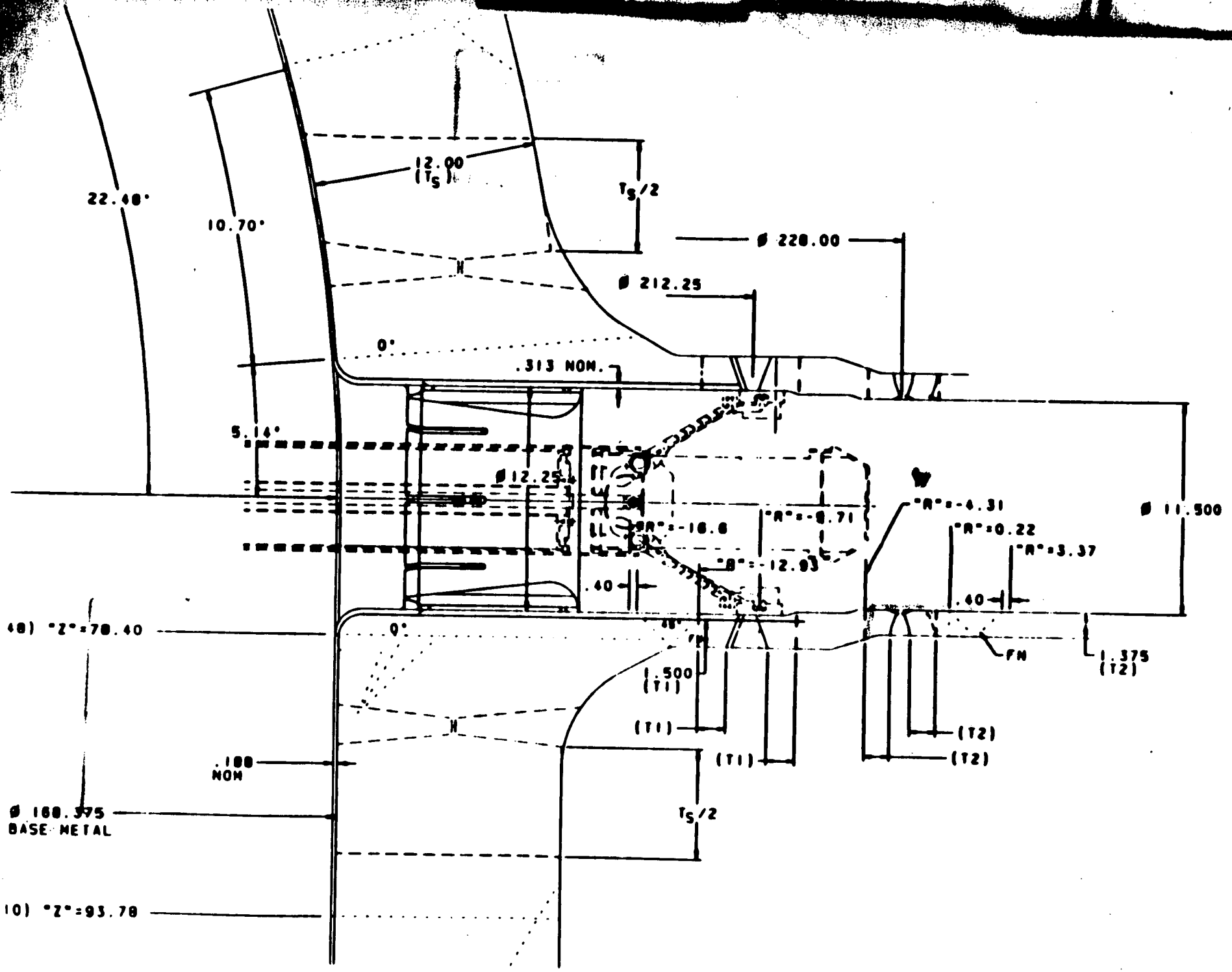
Mockup Demonstrations verify that OD flaws of interest can be detected and sized by using the demonstrated UT techniques.

FM evaluations support the conclusion that required safety margins are maintained between minimum detectable defects and maximum expected allowable flaw size for the design conditions.

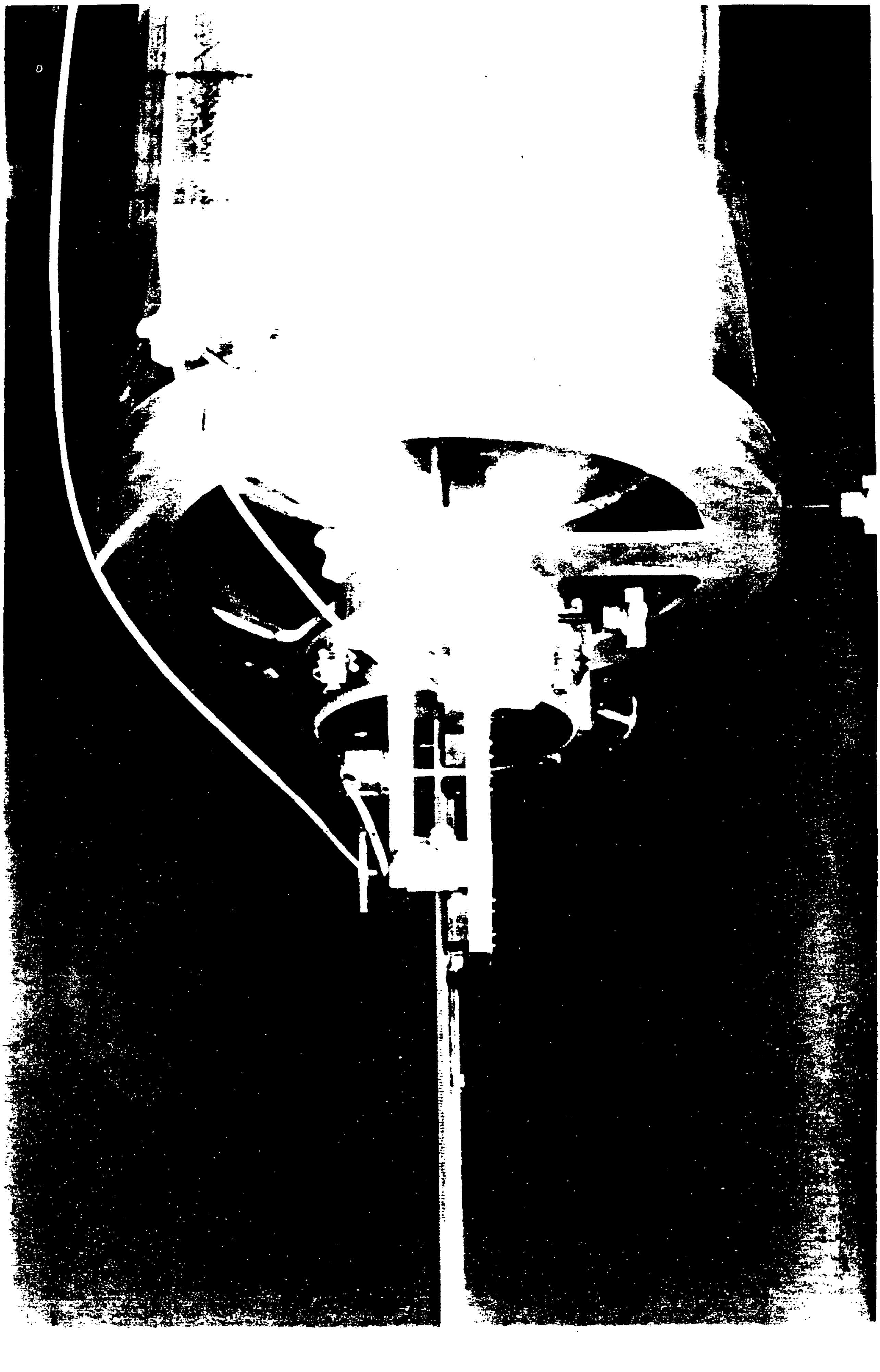
Core Flood Nozzle Safend Welds

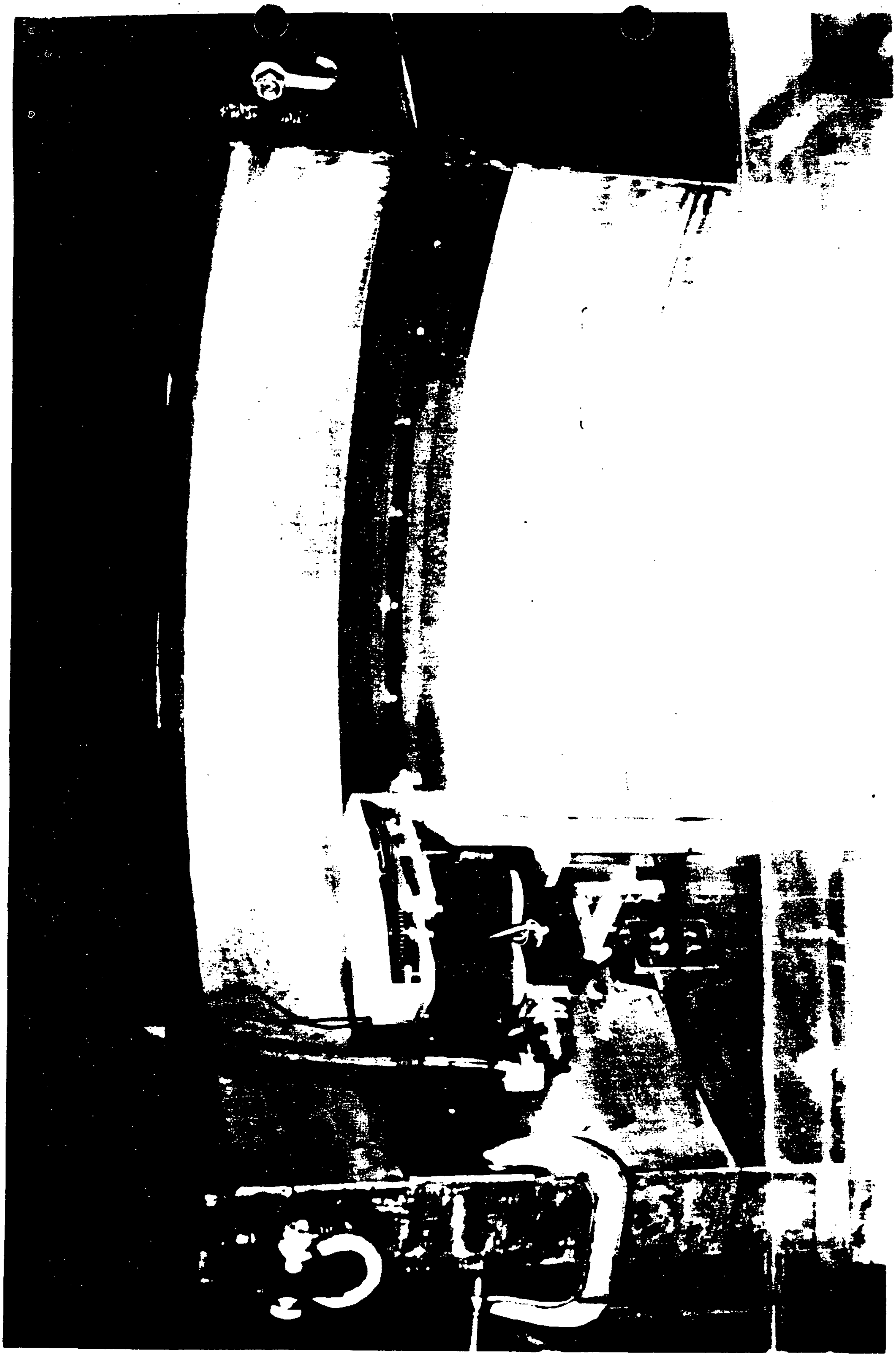
# of Flaws	1	0	2	2	2	1
Flaw #	1		2,3	4,7	8,9	10


HORIZONTAL SECTION VIEW
VERTICAL SECTION VIEW
Ø36.00
"R"=34.18
"R"=37.28
"X"=3.38
"R"=28.68
"R"=50.18
"R"=53.28
"R"=53.58
"R"=62.60
"R"=56.66
"R"=65.38
"R"=60.48
.313 NOM
.313 NOM.
.38 NOM
.50
2.38
70°
15°
45°
15° 0°
R=T/2
TANGENTCY POINT
45°
45°
45°
70°
.45°
70°
3.1875 (Tp)
Tp
Tp
70
VTAL LIP)
80
78
69
12.00 (T)
T/2


22.48°
10.70°
12.00
(Ts)
Ts/2
ø 228.00
ø 212.25
H
0°
.313 NOM.
5.14°
ø 12.25
"R"=-16.8
"R"=-8.71
"R"=-4.31
"R"=0.22
"R"=3.37
"B"=-12.93
ø 11.500
.40
.40
(8) "Z"=78.40
0°
FN
1.375
(T2)
.500
(T1)
(T1)
(T2)
(T1)
(T2)
H
.188
NOM
ø 168.375
BASE METAL
Ts/2
10) "Z"=93.78

UP SUPERIEURE ⬆ OBEN 参考





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DEMONSTRATION OBJECTIVES

1989

DEMONSTRATE THE FLAW DETECTION CAPABILITIES OF THE ULTRASONIC (UT) EXAMINATION TECHNIQUES FOR INSPECTION of PLANAR REFLECTORS OPPOSITE SURFACE FLAWS.

1993

DEMONSTRATE THE FLAW DETECTION and SIZING CAPABILITIES OF THE ULTRASONIC (UT) EXAMINATION TECHNIQUES FOR INSPECTION of REAL OPPOSITE SURFACE FLAWS.

B&WOG UT MOCKUPS

CORE FLOOD NOZZLE PIPING WELDS

1989 MOCKUP

CORE FLOOD SAFE-END TO NOZZLE WELD SIMILAR (MATERIAL AND GEOMETRY) TO ALL OPERATING B&W DESIGNED PLANTS

CONTAINS FOUR SETS OF PLANAR REFLECTORS (EDM NOTCHES) DISTRIBUTED THROUGHOUT THE VARIOUS MATERIALS OF THE WELD JOINT (NOZZLE BASE MATERIAL, INCONEL BUTTERING, INCONEL WELD AND STAINLESS STEEL SAFE-END)

NOTCH SETS CONSISTED OF FOUR THROUGH-WALL SIZES: 2.3%, 5.6%, 11.3% AND 16.9%

1993 MOCKUP

CONSISTS OF A CORE FLOOD PIPE TO SAFE-END AND SAFE-END TO NOZZLE WELDS SIMILAR (MATERIAL AND GEOMETRY) TO ALL OPERATING B&W DESIGNED PLANTS

B&WOG UT MOCKUPS continued

CORE FLOOD PIPE TO SAFE-END WELD

CONTAINED FOUR O.D. SURFACE, THERMAL FATIGUE CRACKS DISTRIBUTED THROUGHOUT THE WELD AND BASE MATERIAL

CRACK SIZES RANGE FROM 11% TO 71.8% THROUGH-WALL

CORE FLOOD SAFE-END TO NOZZLE WELD

CONTAINED FOUR O.D. SURFACE, THERMAL FATIGUE CRACKS DISTRIBUTED THROUGHOUT THE VARIOUS MATERIALS OF THE WELD JOINT

CRACK SIZES RANGE FROM 12.6% TO 62.9% THROUGH-WALL

B&WOG UT MOCKUPS continued

REACTOR COOLANT NOZZLE TO PIPE WELD

1993 MOCKUP

CONSISTS OF A CLAD FERRITIC DESIGN SIMILAR TO ALL OPERATING B&W OPERATING PLANTS

CONTAINS FIVE THERMAL FATIGUE CRACKS RANGING FROM 7.6% TO 78.7% THROUGH-WALL

FLAWS DISTRIBUTED IN BOTH THE WELD AND BASE MATERIALS

DEMONSTRATION RESULTS

1989

SUCCESSFUL IN DEMONSTRATING THE DETECTABILITY OF THE UT EXAMINATION TECHNIQUE FOR SMALL (LENGTH AND THROUGH-WALL) PLANAR REFLECTORS (EDM NOTCHES) LOCATED IN THE VARIOUS MATERIALS OF THE MOCKUP

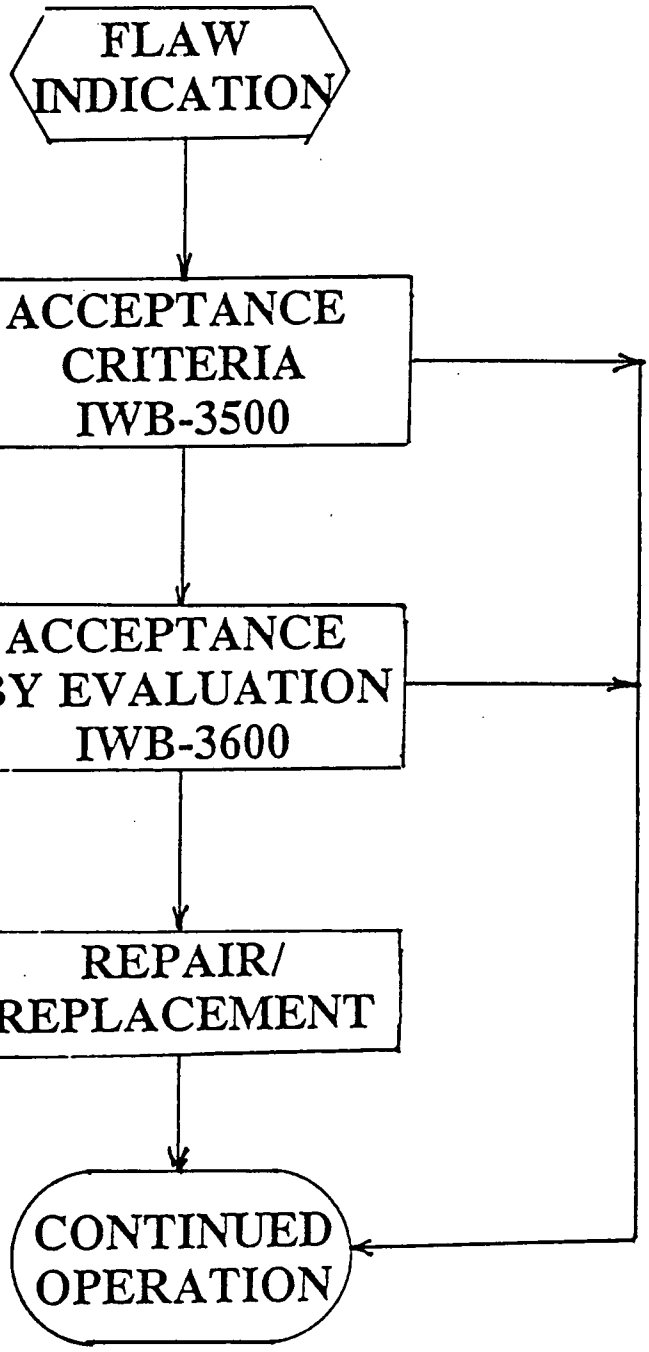
DETECTED REFLECTORS AS SMALL AS 2.3% THROUGH-WALL IN ALL THE MATERIALS OF THE WELD JOINT (NOZZLE BASE MATERIAL, INCONEL BUTTERING, INCONEL WELD AND STAINLESS STEEL SAFE-END)

1993

SUCCESSFULLY DEMONSTRATED the 100% DETECTION OF ALL THERMAL FATIGUE FLAWS IMPLANTED IN BOTH MOCKUPS (CORE FLOOD AND REACTOR COOLANT)

ACCURATELY DEPTH SIZED ALL THERMAL FATIGUE FLAWS WITHIN A RMS ERROR OF 0.076 INCHES

INSERVICE INSPECTION



FLAW EVALUATION

- FLAW TYPE -- EXTERNAL CIRC. FLAW ON A CYLINDER

- LOAD/STRESSES
  -- DESIGN BASIS LOADS FROM STRESS REPORT
  HEATUP AND COOLDOWN STRESSES

  -- INSPECTION INTERVAL 10 YEARS
  90 CYCLES/ 10 YEARS

- MATERIAL - STAINLESS STEEL
  da/dN from Appendix C of Section XI

2.1 Weld Geometry

The geometry of the two weld locations is given in Reference 5. The weld geometries are cylindrical. The pertinent dimensions for each of the weld locations are given below.

a) For the Nozzle to Safe-end Weld

Outside radius of weld, R_o = 7.8125 inches

Inside radius of weld, R_i = 6.125 inches

Thickness of weld, t = 1.6875 inches

b) For the Safe-end to Pipe Weld

Outside radius of weld, R_o = 7.125 inches

Inside radius of weld, R_i = 5.75 inches

Thickness of weld, t = 1.375 inches

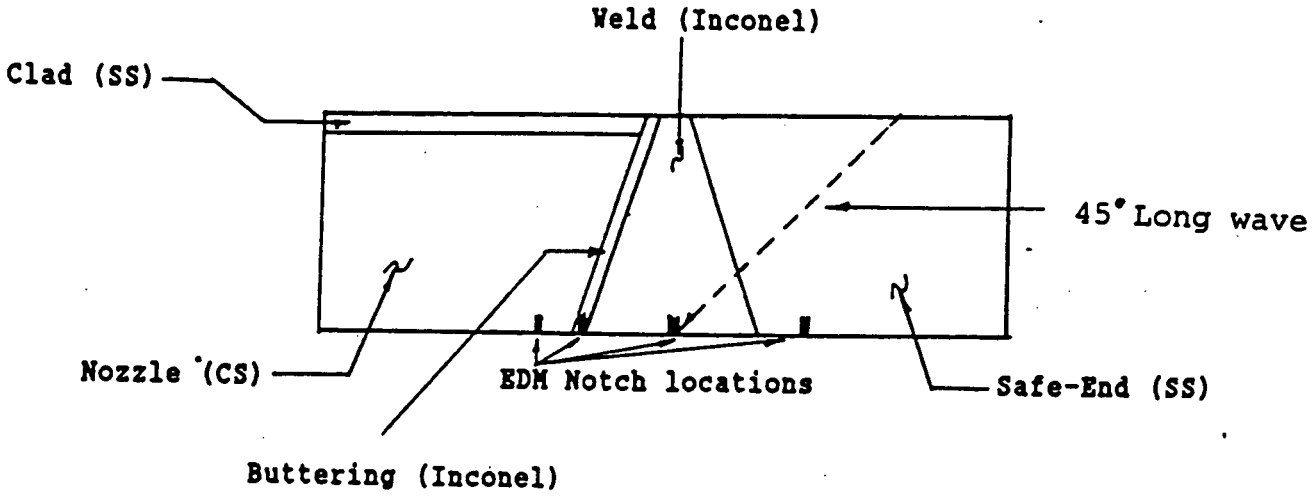
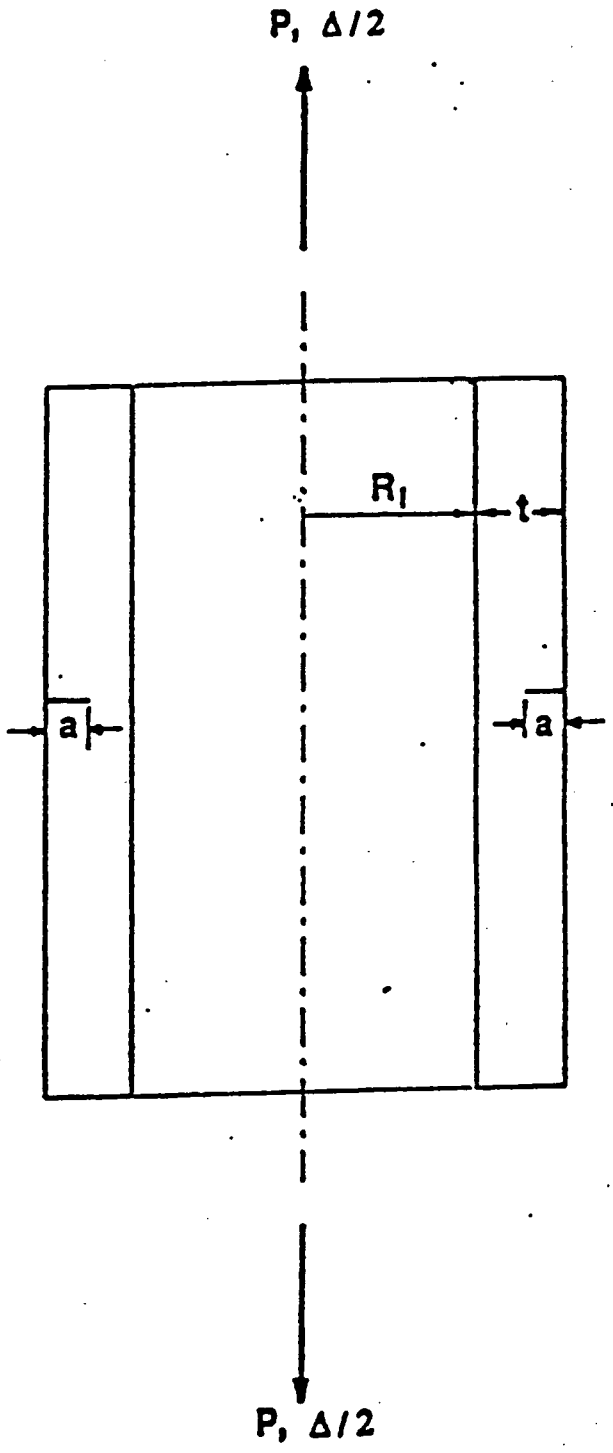
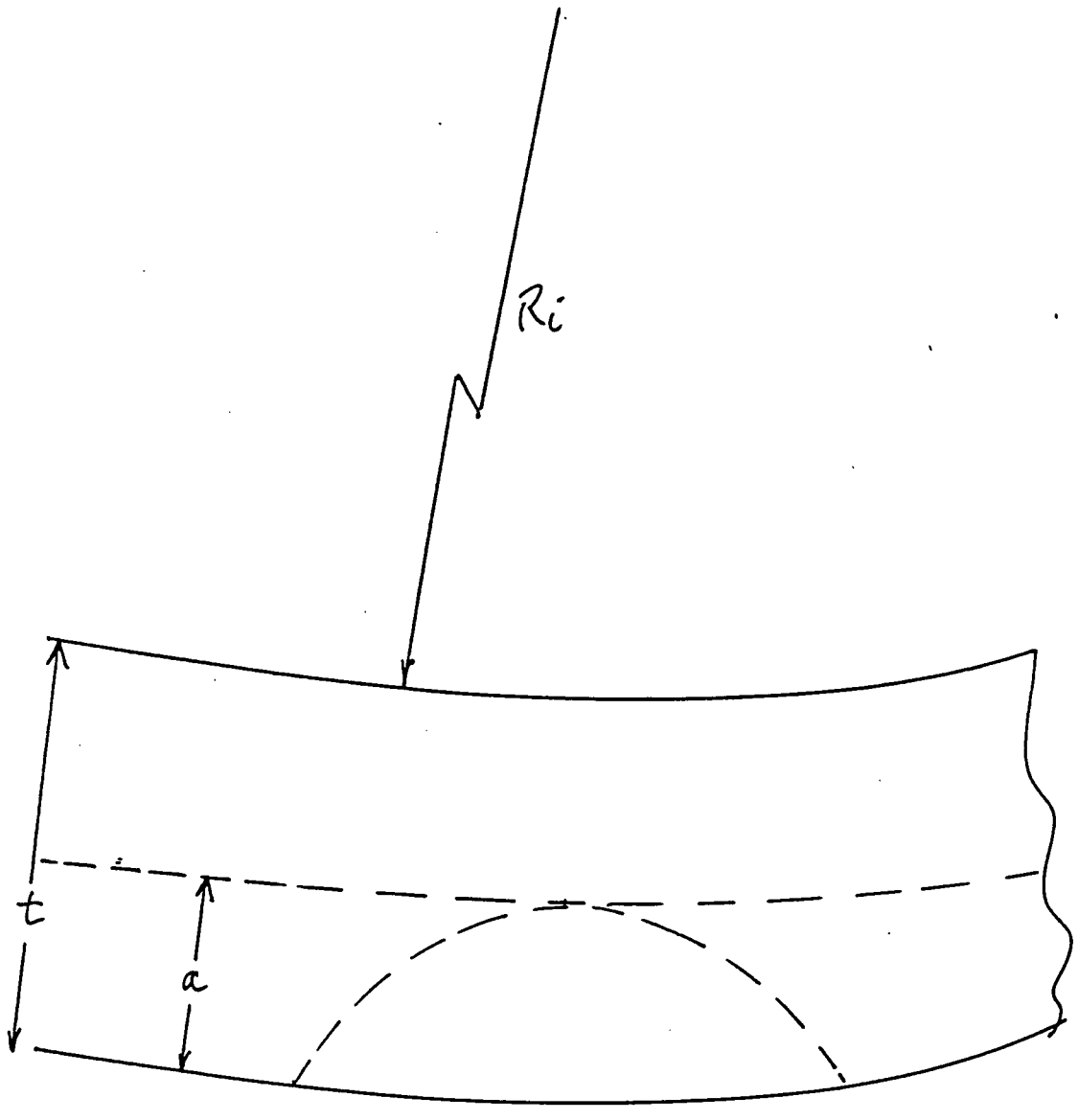

Weld (Inconel)
Clad (SS)
45° Long wave
Nozzle (CS)
EDM Notch locations
Safe-End (SS)
Buttering (Inconel)

Figure 1. Full Circumference Part-throughwall Flaw subjected to Axial Loading

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FLAW EVALUATION
OF
CORE FLOOD NOZZLE WELDS

I. NOZZLE TO SAFE-END WELD

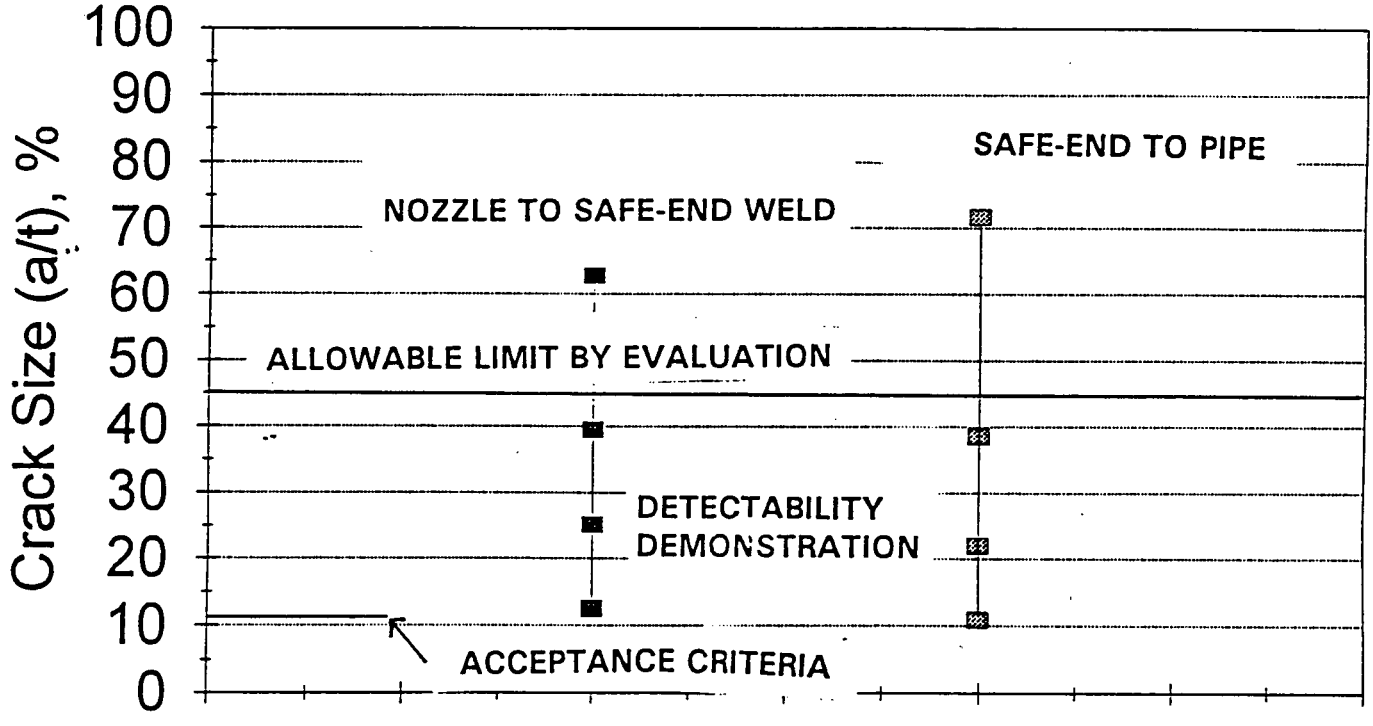
MAX. ALLOWABLE FLAW $a/t = 0.45$

2. SAFE-END TO PIPE WELD

MAX. ALLOWABLE FLAW $a/t = 0.56$

Core Flood Nozzle to Safe-End				
Flaw No.	a/l	a/t%	IWB-3514 Max. a/t%	t (inches)
1	.22	12.6	10.9	1.80
2	.25	25.2	11.0	1.80
3	.28	39.5	11.1	1.80
4	.45	62.9	11.8	1.80
Core Flood Safe-End to Pipe				
7	.17	22.0	11.1	1.10
8	.25	38.6	11.5	1.10
9	.29	71.8	11.6	1.10
10	.16	11.0	11.1	1.10
Reactor Coolant Outlet Nozzle to Pipe				
1	.25	7.6	11.7	3.0
2	.50	78.7	13.2	3.0
3	.27	16.3	12.3	3.0
4	.45	37.1	13.2	3.0
5	.28	10.9	12.6	3.0

ALLOWABLE FLAW SIZE



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