Revision 1
April 30, 1985

INSERVICE INSPECTION EXAMINATION REPORT

YANKEE ATOMIC ELECTRIC COMPANY
YANKEE NUCLEAR POWER STATION

MARCH 31, 1984 THROUGH JUNE 9, 1984

PREFACE

This summary report covers the inservice inspection of Yankee Nuclear Power Station during the period March 31, 1984 through June 9, 1984.

Included in this report is the NIS-1 form as required by the provisions of ASME Section XI.

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FORM NIS-1 OWNERS' DATA REPORT FOR INSERVICE INSPECTIONS

As required by the Provisions of the ASME Code Rules

1. Owner Yankee Atomic Electric Co., 1671 Worcester Road, Framingham, Ma. 01701
(Name and Address of Owner)

2. Plant Yankee Nuclear Power Station, Rowe, Massachusetts 01367
(Name and Address of Plant)

3. Plant Unit Yankee Rowe 4. Owner Certificate of Authorization (if required) DPR-3

5. Commercial Service Date 7/1/61 6. National Board Number for Unit Reactor #NB-23964

7. Components Inspected

Component or Appurtenance	Manufacturer or Installer	Manufacturer or Installer Serial No.	State or Province No.	National Board No.
Reactor Vessel	B&W	610-0011	N/A	23964
Steam Generator	W	N/A	N/A	404E(Z)1
Piping	S&W	N/A	N/A	N/A
Pressurizer	B&W	610-0011	N/A	2397 4E 22

Note: Supplemental sheets in form of lists, sketches, or drawings may be used provided (1) size is 8½ in. x 11 in., (2) information in items 1 through 6 on this data report is included on each sheet, and (3) each sheet is numbered and the number of sheets is recorded at the top of this form.

This form (E00029) may be obtained from the Order Dept., ASME, 345 E. 47th St., New York, N.Y. 10017

FORM NIS-1 (back)

8. Examination Dates 3/31/84 to 6/9/84 9. Inspection Interval from 12/1/74 to 12/1/84

10. Abstract of Examinations. Include a list of examinations and a statement concerning status of work required for current interval. Page 3 of 28

11. Abstract of Conditions Noted Page 6 of 28

12. Abstract of Corrective Measures Recommended and Taken Page 7 of 28

We certify that the statements made in this report are correct and the examinations and corrective measures taken conform to the rules of the ASME Code, Section XI.

Date 4/5 .. 84 Signed Yankee Atomic By N N St Laurent
Owner

Certificate of Authorization No. (if applicable) DOR-3 USNRC Facility License No. Expiration Date 11/4/97

CERTIFICATE OF INSERVICE INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State or Province of Massachusetts and employed by HSBI & I Co. of Connecticut have inspected the components described in this Owners' Data Report during the period 3/31/84 to 6/9/84, and state that to the best of my knowledge and belief, the Owner has performed examinations and taken corrective measures described in this Owners' Data Report in accordance with the requirements of the ASME Code, Section XI.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the examinations and corrective measures described in this Owners' Data Report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date 9/6 1984

R. L. Lane
Inspector's Signature

Commissions Mass. 1182
National Board, State, Province and No.

Revision 1

Date 5/1 1985 Signed Yankee Atomic By N N St Laurent
Owner

Date 5/16 1985

Tom L. Hulme
Inspector's Signature

Commissions NB 9948 MASS 1396
National Board, State, Province and No.

10. Abstract of Examinations (Safety Class 1)

ASME Code Category	No.	Components Examined	Method
B-A		Closure Head Flange Circulation Weld	UT
		Vessel to Flange Weld	UT
B-B	(2)	Welds Steam Generator #3	UT
	(3)	Pressurizer Nozzles	UT
B-D	(4)	Outlet Nozzles	UT,VT
	(4)	Inlet Nozzles	UT,VT
	(4)	Outlet Nozzles to Reducer	UT,VT
	(4)	Inlet Nozzles to Reducer	UT,VT
B-E	(1)	Feed/Bleed Heat Exchanger Nozzle	VT
B-F	(1)	Pressurizer Safe End Weld	PE,UT
B-G-1	(52)	Rx Head Studs, Nuts, and Washers	VT,MT,UT
		Flange Ligaments	UT
B-G-2	(2)	Safety Valves SV-181/182 Bolting	VT
	(20)	Studs/Nuts Main Coolant Check Valve #316	UT,VT
	(16)	Stator Cap Bolts, #2 Main Coolant Pump	UT,MT,VT
	(12)	Studs/Nuts Pressurizer Primary Manway	UT,VT
	(40)	Studs, #2 Steam Generator Primary Manway	UT,MT,VT
	(38)	Studs, #3 Steam Generator Primary Manway	UT,MT,VT
	(40)	Studs, #4 Steam Generator Primary Manway	UT,MT,VT
B-H	(4)	Welded Supports Feed/Bleed Heat Exchangers	PE
B-J	(34)	Safety Class 1 Pipe Welds	UT,PE,VT
B-K-1	(3)	Integrally Welded Pump Supports	PE
B-K-2	(5)	Nonwelded Supports	VT
B-M-2	(1)	Main Coolant Check Valve #316 Internals	VT
B-N-1		Interior of Reactor Vessel	VT
B-N-3		Upper Core Support Barrel	VT
		Lower Core Support Barrel O.D.	VT
		Flow Baffle Assembly, Spacer Blocks, and Source Vanes	VT
		Shroud Tubes from I.D. of Lower Core Support Barrel	VT
		Bolting Upper Core Support Plate to Baffle Assembly	VT
		Shroud Tube Tie-Plate Bolting	VT
		Lower Core Support Barrel I.D.	VT

ASME Code Category	No.	Components Examined	Method
		Core Barrel Lateral Support Pads	VT
		Thermal Shield Surface, Spacer Pins, Seam Clamps, and Support Lugs	VT
B-O		Control Rod Drive Housings	PT
B-P		System Leakage Test Conducted on all Safety Class 1 Systems	VT

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10. Abstract of Examinations (Safety Classes 2 and 3)

ASME Code Category	No.	Components Examined	Method
C-A	(1)	Weld Low Pressure Surge Tank Heat Exchanger	UT
C-B	(1)	Nozzle Low Pressure Surge Tank Heat Exchanger	PE
CC-CE	(2)	Welded Supports #3 Steam Generator	PE
	(1)	Mechanical Support #1 Steam Generator	VT
	(3)	Charging Pump Supports	VT
	(2)	Nonwelded Supports #3 Steam Generator	VT
	(30)	Nonwelded Supports Shutdown Cooling System	VT
	(1)	Welded Support Shutdown Cooling System	PE
	(15)	Nonwelded Supports, Main Steam Line	VT
	(17)	Nonwelded Supports, Feedwater Line	VT
	(1)	Vapor Container Heating, Welded Support	VT
	(1)	Low Pressure Surge Tank SV Discharge Support	VT
C-F	(14)	Shutdown Cooling System Piping Welds	PE
	(11)	Safety Injection System Piping Welds	PE
	(5)	Feedwater System Piping Welds	MT, VT
	(10)	Main Steam System Piping Welds	UT, MT, VT
C-H	(9)	Safety Class 2 System Leakage Tests	VT
	(13)	Safety Class 2 System Hydrostatic Tests	VT
D-A	(5)	Safety Class 3 System Leakage Tests	VT
	(3)	Safety Class 3 System Hydrostatic Tests	VT

11. Abstract of Conditions Noted

B-D Ultrasonic inner radius examination of pressurizer nozzles PRZN-2, 5, and 6 revealed 1 recordable indication on nozzle PRZN-2.

Ultrasonic examination of RPV outlet nozzle FF (I.D.) revealed 1 reportable indication in the weld region.

B-F An unacceptable linear indication was identified by visual testing on the pressurizer drain line weld PRZ-SE-2.

B-G-1 Examination of 52 reactor head studs revealed a degraded condition of two threads on reactor stud #4.

B-G-2 Inspection of 16 bolts on #2 main coolant pump stator cap identified a break in the plating on 1 bolt.

B-J Examination of main coolant piping weld MC-1-4 identified an unacceptable linear indication.

B-K-2 Inspection of main coolant pump support CRM-H-12 identified one bolt/nut having lack of thread engagement.

B-N-3 Visual inspection of the flow baffle assembly (lower core barrel I.D.) revealed a reportable condition - cracked tack weld.

CC-CE Visual examination identified a total of 15 supports with discontinuities on the Shutdown Cooling System.

Visual examination identified a total of 4 supports with discontinuities on the main steam line.

Visual examination identified a total of 4 supports with discontinuities on the Feedwater System.

C-F Ultrasonic examination of main steam line welds MS-1-15 and MS-2-11 identified two recordable indications.

Examination of 4 nonnuclear safety main steam line welds identified a nonservice-induced discontinuity (LAP).

12. Abstract of Corrective Measures Recommended and Taken

B-D The ultrasonic indication recorded on PRZN-2 was subsequently evaluated in accordance with IWB-3514.5 and determined to be a geometric reflector.

The indication found in RPV outlet nozzle FF was evaluated as defined in IWB-3600. Appendix B to this report contains that evaluation. Subsequent evaluation of the sizing method was performed. Based on the data developed by this sizing study, the indication is acceptable per IWB-3512-1. Appendix C contains the sizing study and all pertinent information. 1

B-F The linear indication identified on weld PRZN-SE-2 was subsequently removed by light buffing and re-examined as satisfactory.

B-G-1 The two degraded threads on reactor stud #4 were subsequently evaluated by engineering and found to be acceptable for continued service.

B-G-2 The bolt on #3 main coolant pump stator cap which had a break in the plating was replaced in kind.

B-J The indication on main coolant piping weld MC-1-4 was subsequently removed by light buffing and re-examined satisfactory.

B-K-2 The lack of thread engagement condition identified on main coolant pump support CRM-H-12 was subsequently corrected and re-examined as satisfactory.

B-N-3 The indication identified on the flow baffle assembly was subsequently evaluated and deemed inconsequential to the structure.

CC-CE The 15 supports on the Shutdown Cooling System identified as having discontinuities were subsequently corrected and re-examined as satisfactory.

The 4 supports on the main steam line identified as having discontinuities were subsequently corrected and re-examined as satisfactory.

The 4 supports on the Feedwater System identified as having discontinuities were subsequently corrected and re-examined as satisfactory.

C-F The ultrasonic indications recorded on main steam line welds MS-1-5 and MS-2-11 were subsequently evaluated in accordance with IWB-3514.5 and determined to be geometric reflectors.

The discontinuity on #2 nonnuclear safety main steam line weld was subsequently determined to be nonservice induced and accepted as is.

1.0 INTRODUCTION

This report covers the inservice inspection of Yankee Nuclear Power Station during the period of March 31, 1984 to June 9, 1984.

The examinations performed are those of the third period of the second interval.

The required ten-year reactor pressure vessel examinations were performed this refueling. A summary of the examinations performed and conditions noted are included as part of this report along with attachments.

With the exception of several hydrostatic pressure tests, the examinations performed this refueling complete the required inspections for the third period of the second interval.

2.0 NONDESTRUCTIVE EXAMINATION PROCEDURES

Nondestructive examinations were performed in accordance with the procedures contained in the Yankee Atomic Electric Company Engineering Guidelines, Book III, "Inservice Inspection NDE Procedures", and Nuclear Energy Services examination procedures.

The examination procedures were reviewed and approved by personnel certified to Level III per SNT-TC-1A, 1975 Edition, and the authorized Nuclear Inservice Inspector.

These procedures conform to the requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1977 Edition, Summer 1978 Addenda.

The following procedures were used during the 1984 inservice inspections:

1. YA-PE-2 Rev. 3 "Liquid Penetrant Examination"
2. YA-MP-111 Rev. 0 "Procedure for Magnetic Particle Examination"
3. YA-VT-11 Rev. 1 "Visual Examination Procedure"
4. YA-RT-111 Rev. 0 "Radiographic Examination"
5. OP-4200 Rev. 10 "Main Coolant System Leak Inspection"
6. YA-UT-1 Rev. 2 "Ultrasonic Examination, General Requirements"
7. YA-UT-7 Rev. 2 "Ultrasonic Examination of Bolting"
8. YA-UT-8 Rev. 1 "Ultrasonic Examination of Reactor Closure Nuts"
9. YA-UT-9 Rev. 2 "Ultrasonic Examination of Piping - Ferritic Welds"
10. YA-UT-10 Rev. 4 "Ultrasonic Examination of Piping - Austenitic Welds"
11. YA-UT-13 Rev. 1 "Ultrasonic Examination of Vessel Nozzles - Inner Radius"
12. YA-UT-14 Rev. 2 "Ultrasonic Examination of Piping - Base Metal and HAZ"
13. YA-UT-22 Rev. 0 "Ultrasonic Examination of Vessels - Circumferential, Longitudinal, Meridional, and Flange Welds"
14. YA-UT-44 Rev. 0 "Ultrasonic Examination of Vessels - Nozzle to Vessel Welds"
15. YA-UT-116 Rev. 0 Ultrasonic Examination of Full Penetration Welds"
16. YA-UT-6 Rev. 2 "Ultrasonic Examination of Flange Ligaments"

17. 83A0314 Rev. 0 "Automated UT Examination for RPV Nozzle Welds from Nozzle Bore"

18. 83A0313 Rev. 0 "UT Examination of RPV Closure Head to Flange Weld"

19. 83A0312 Rev. 0 "UT Examination of RPV Upper Shell to Flange Weld from the Flange Mating Surface"

20. 83A0311 Rev. 0 "Operation of NES Mini-Scanner System"

21. 83A0317 Rev. 0 "Visual Examination Procedure"

22. 83A0318 Rev. 0 "Visual Examination Scan Plan"

The following technique sheet was used to perform the subject examination:

Technique Sheet No.	Rev.	Subject
YA-UT-3, TS-1	1	Flange to Vessel Weld From Flange Face

3.0 EVALUATION OF DATA

All inservice examinations were performed, evaluated, and reviewed by personnel certified to Level II in accordance with SNT-TC-1A, 1975 Edition and ASME Section XI, 1977 Edition, Summer 1978 Addenda.

The examination methods, volumes, and evaluation of indications were in accordance with ASME Boiler and Pressure Vessel Code, Section XI, 1977 Edition, Summer 1978 Addenda, except for Class 1 piping ultrasonic calibration. This was conducted in accordance with Article III-2000 of Appendix III, ASME Section XI, Summer 1976 Addenda, as required per Plant Technical Specifications.

Summaries of the examinations that were performed are contained in Section 4.0 of this report. The detailed examination data along with the calibration records, procedures, personnel, and equipment certifications are maintained at the plant site.

Attached is a summary of the examination methods, volumes, and the results and evaluation of test data thereof, including any corrective measures taken.

4.0 SUMMARY REPORT

The following is a summary of all the examinations performed, the conditions noted and corrective measures taken during the 1984 inservice inspections, with the exception of the ten-year RPV examination that is included as part of this report.

CODE CATEGORY B-B — **PRESSURE RETAINING WELDS IN VESSELS OTHER THAN REACTOR VESSELS**

Ultrasonic examination was performed on two welds on steam generator #3. The longitudinal shell to barrel weld SG-3-8 and the longitudinal barrel to transitional piece SG-3-9 were inspected with no recordable indications.

CODE CATEGORY B-D — **FULL PENETRATION WELDS OF NOZZLES IN VESSELS**

Ultrasonic examination was performed on 3 pressurizer nozzle welds. Nozzle welds PRZN-2, 5, and 6 were examined with no recordable indications. A nozzle inner radius examination was also performed on all 3 nozzles. Only one indication was recorded, on nozzle PRZN-2, which was subsequently evaluated and determined to be a geometric reflector in accordance with IWB-3514.5.

Ultrasonic examination was performed on eight (8) RPV outlet and inlet nozzle I.D.s. Outlet nozzle FF revealed one reportable indication. This was 1
subsequently evaluated and determined to be acceptable (see Attachment A, B and C).

CODE CATEGORY B-E — **PRESSURE RETAINING PARTIAL PENETRATION WELDS IN VESSELS**

No. 1 feed and bleed heat exchanger nozzle FB-1-3 was visually examined during the Main Coolant System leakage inspection (OP-4200). No leakage was noted.

CODE CATEGORY B-F — **PRESSURE RETAINING DISSIMILAR METAL WELDS**

An ultrasonic and liquid penetrant examination was performed on pressurizer safe end weld PRZ-SE-1. No indications were recorded. During the examination of PRZ-SE-1, a visual inspection of weld PRZ-SE-2 identified two linear indications.

The areas were lightly buffed and re-examined with liquid penetrant which revealed the indications had been removed. A liquid penetrant examination was also performed on the weld adjacent to PRZ-SE-2. This examination resulted in no unacceptable indications.

CODE CATEGORY B-G-1 — **PRESSURE RETAINING BOLTING LARGER THAN 2" IN DIAMETER**

Fifty-two (52) reactor head closure st[illegible], nuts, and washers (Set No. L-43) including 2 reduced diameter studs #1-638-1 and #1-674-1 were inspected. A visual examination of the washers and a visual, magnetic particle, and ultrasonic examination of the nuts resulted in no recordable indications.

A visual and ultrasonic examination was performed on all 52 studs. During the visual examination of stud #4, two threads were found to be degraded. An engineering evaluation was conducted which determined that the stud was still acceptable for use. The subsequent ultrasonic examination did not reveal any indications.

CODE CATEGORY B-G-2 PRESSURE RETAINING BOLTING 2" AND SMALLER IN DIAMETER

A base line visual examination was performed on 16 new bolts and nuts prior to installation on pressurizer safety valves SV-181 and 182. This inspection resulted in no unacceptable indications noted.

A base line visual examination was also conducted on 6 new bolts installed on PR-MOV-191. No unacceptable indications were noted.

The following bolting/studs and nuts were inspected per I&E Bulletin 83-02, "Bolting in RCPB Closure Connections Greater Than Six Inches":

A visual and ultrasonic examination was performed on 20 studs and nuts from #3 main coolant loop check valve #316. No unacceptable indications were noted.

A visual, magnetic particle and ultrasonic examination was performed on 16 bolts removed from #2 main coolant pump stator cap. A linear indication was noted on one bolt which was determined to be a break in the plating. A new bolt was installed in its place, prior to which a visual, magnetic particle, and ultrasonic examination was performed on it, with no unacceptable indications noted.

Twelve (12) pressurizer primary manway studs were visually and ultrasonically inspected with no unacceptable indications noted.

A visual, ultrasonic, and magnetic particle examination was conducted on 40 studs from steam generator #2 primary manways, 38 studs from steam generator #3 primary manways, and 40 studs from steam generator #4 primary manways. These examinations resulted in no unacceptable indications.

CODE CATEGORY B-H VESSEL SUPPORTS

Feed and bleed heat exchanger integrally welded supports BL-H-6, 8, and 10 were liquid penetrant inspected with no unacceptable indications noted.

CODE CATEGORY B-J PRESSURE RETAINING WELDS IN PIPING

The following Main Coolant System piping welds were inspected as follows:

Weld No.

MC-1-10 Liquid penetrant examination performed - no unacceptable indications noted.

MC-1-13 Ultrasonic and liquid penetrant examination performed - no unacceptable indications noted.

MC-1-16 Liquid penetrant examination performed - no unacceptable indications noted.

Weld No.	
MC-1-17	Liquid penetrant examination performed - no unacceptable indications noted.
MC-1-15	Liquid penetrant examination performed - no unacceptable indications noted.
MC-2-15	Liquid penetrant examination performed - no unacceptable indications noted.
MC-2-10	Liquid penetrant examination performed - no unacceptable indications noted.
MCB-2-5	Liquid penetrant and ultrasonic examination performed - no unacceptable indications noted.
MCB-4BR-2	Liquid penetrant and ultrasonic examination performed - no unacceptable indications noted.
MC-1-4	Liquid penetrant examination was performed which identified several unacceptable linear indications. The areas were subsequently buffed and re-examined which verified the indications had been removed. An additional weld was examined in accordance with IWB-2430, which identified no unacceptable indications.
MC-2-3	Liquid penetrant examination performed - no unacceptable indications noted.
Safety Valve 182 Weld	Ultrasonic and liquid penetrant base line examination performed - no unacceptable indications noted.
PRS-206-23	Liquid penetrant base line examination - no unacceptable indications noted.
PRS-206-26	Liquid penetrant base line examination - no unacceptable indications noted.
PRS-206-29	Liquid penetrant examination - no unacceptable indications noted.
PRS-206-39	Visual and liquid penetrant examination - no unacceptable indications noted.
PRS-43	Liquid penetrant examination - no unacceptable indications noted.
PRS-52	Liquid penetrant examination - no unacceptable indications noted.
PRS-50	Liquid penetrant examination - no unacceptable indications noted.

Weld No.

PRS-206-24A Liquid penetrant base line examination - no unacceptable indications noted.

PRS-206-24B Liquid penetrant base line examination - no unacceptable indications noted.

PRS-206-22A Liquid penetrant base line examination - no unacceptable indications noted.

PRS-206-22B Liquid penetrant base line examination - no unacceptable indications noted.

PRS-206-40 Liquid penetrant examination - no unacceptable indications noted.

Pressurizer safety valve discharge header weld caps #1 and #2 - liquid penetrant examination - no unacceptable indications noted.

SAFETY INJECTION PIPING INSPECTIONS

Weld No.

SI-3-7 Ultrasonic and liquid penetrant examination performed - no unacceptable indications noted.

SI-3-7A Ultrasonic and liquid penetrant examination performed - no unacceptable indications noted.

SI-3-8 Ultrasonic and liquid penetrant examination performed - no unacceptable indications noted.

SI-4-1 Ultrasonic and liquid penetrant examination performed - no unacceptable indications noted.

SI-147 Liquid penetrant examination performed - no unacceptable indications noted.

SI-153 Ultrasonic examination performed - no unacceptable indications noted.

SI-156 Ultrasonic examination performed - no unacceptable indications noted.

SI-137 Liquid penetrant examination performed - no unacceptable indications noted.

CODE CATEGORY B-K-1 SUPPORT MEMBERS FOR PIPING, PUMPS, AND VALVES

A liquid penetrant examination was performed on 3 main coolant pump integrally welded supports, CRM-H-5, CRM-H-12, and CRM-H-26. No unacceptable indications were identified.

CODE CATEGORY B-K-2 COMPONENT SUPPORTS FOR PIPING PUMPS AND VALVES

A visual examination (VT-3 and VT-4) was performed on the following component supports:

MCB1-H-1	CRM-H-26
CRM-H-5	SDC-H-1
CRM-H-12	BL-H-1

Only one discontinuity was noted. CRM-H-12 was rejected due to a nonservice-induced condition (lack of thread engagement), which was subsequently corrected and re-examined as satisfactory.

CODE CATEGORY B-M-2 PUMP CASINGS AND VALVE BODIES

Main Coolant System loop #3, check valve 316, was disassembled and a visual inspection was performed on the internals. This resulted in no unacceptable indications.

CODE CATEGORY B-P ALL PRESSURE RETAINING COMPONENTS

A system hydrostatic pressure test was conducted on all repaired/replaced sections of the Main Coolant System prior to startup.

Plant Procedure OP-4200, Rev. 10, "Main Coolant System Leak Inspection or ISI Pressure Test", was performed at 2040 psi at 514°F for two hours. The areas inspected were uninsulated. The visual inspection (VT-2) was acceptable.

In conjunction with the above test, the remaining portions of the Reactor Coolant System was subjected to the required system leakage tests.

No serious degradation was noted during the inspection other than normal packing leaks which were corrected at the time of the inspection.

SAFETY CLASS 2 COMPONENTS

CODE CATEGORY C-A PRESSURE RETAINING WELDS IN PRESSURE VESSELS

An ultrasonic examination was performed on the low pressure surge tank heat exchanger head circumferential weld (LPST-Hx-H-1). No recordable indications were observed.

CODE CATEGORY C-B PRESSURE RETAINING NOZZLE WELDS IN VESSELS

The low pressure surge tank heat exchanger nozzles to shell welds #1 and #2 (LPST-Hx-N1 and 2) were subjected to a liquid penetrant examination. No unacceptable indications were recorded.

An ultrasonic inner radius examination was attempted on #3 steam generator main steam outlet nozzle SG-3-S01. Due to physical limitations of the component, no relevant data could be obtained, therefore the examination was deleted.

CODE CATEGORY CC-CE

Due to access limitations, a "best effort" liquid penetrant examination was performed on #3 steam generator integrally welded supports (SG-3-E and SG-3-W). No unacceptable indications were identified.

Mechanical support SG-1-236 was functionally tested (VT-4) after removal from steam generator #1. Functional acceptability was verified as satisfactory.

Component supports for charging pumps P-15-1, 2, and 3 and steam generator #3 supports SG-3-SE and SG-3-SW were subjected to a visual examination (VT-3) and found to be acceptable.

The following Shutdown Cooling System support members were subjected to visual examinations with results as follows:

Support No.	
CRT-H-66	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-related discontinuity (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory.
CRT-H-59	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-related discontinuity (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory. VT-4 identified an incorrect support setting. This was evaluated by Engineering and found to be acceptable as is. An additional examination was performed per IWC-2430 and found acceptable.
SDC-H-1	Subjected to a VT-3/VT-4 visual examination and found acceptable.
PRCL-H-43	Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable condition (loose nuts). This condition was corrected and re-examined as satisfactory. An additional examination was performed per IWC-2430 and found to be acceptable.
PRCL-H-53	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-induced unacceptable condition (lack of thread engagement). This condition was corrected and subsequently re-examined and found to be acceptable.
CRT-H-71	Subjected to a VT-3/VT-4 visual examination and found to be acceptable.
CRT-H-62	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-induced unacceptable condition (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory.

Support No.

PRCL-H-54 Subjected to a VT-3/VT-4 visual examination. This was an additional examination as required by IWC-2430(a). This examination resulted in an unacceptable VT-3/VT-4 condition. In accordance with IWC-2430(b) the remaining number of supports within the system were subjected to applicable VT-3/VT-4 examinations with results as follow:

PRCL-H-60 - Subjected to VT-3/VT-4 examination. VT-3 identified an unacceptable condition (missing bolt). Subsequently corrected and re-examined as satisfactory.

PRCL-H-57 - Subjected to VT-3/VT-4 examination. VT-3 identified an unacceptable condition (loose nuts). VT-4 identified incorrect support settings. Both conditions were subsequently corrected and reinspected as satisfactory.

PRCL-H-58 - Subjected to a VT-3/VT-4 visual examination. VT-4 identified an incorrect support setting which was subsequently corrected and re-examined as satisfactory.

PRCL-H-61 - Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice related unacceptable condition (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory.

PRCL-S-55 - Subjected to a VT-3 visual examination and found acceptable.

PRCL-H-51 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-H-44 - Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable nonservice-induced condition (lack of thread engagement). VT-4 identified an incorrect support setting. Both items were subsequently corrected and reinspected as satisfactory.

PRCL-H-64 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-S-51 - Subjected to a VT-3 visual examination and found acceptable.

PRCL-S-51B/S-52 - Subjected to a VT-3 visual examination which identified an unacceptable condition (loose nuts). Subsequently corrected and re-examined as satisfactory.

Support No.

CRT-H-65 - Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable nonservice-induced condition (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory.

CRT-H-67 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

CRT-H-68 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

CRT-H-69 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

CRT-H-70 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-H-40 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-H-42 - Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable condition (loose bracket) which was subsequently corrected and re-examined as satisfactory.

PRCL-H-45 - Subjected to a VT-3/VT-4 visual examination. VT-4 identified an incorrect support setting which was subsequently corrected and re-examined as satisfactory.

PRCL-H-52 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-H-56 - Subjected to a VT-3/VT-4 visual examination and found acceptable.

PRCL-H-63 - Subjected to a VT-3/VT-4 visual examination. VT-4 identified an unacceptable support setting which could not be adjusted. A new support was replaced in kind. A base line VT-3/VT-4 was conducted and found acceptable.

PRCL-S-51A- Subjected to a VT-3 visual examination and found acceptable.

PRCL-H-41 - Subjected to a VT-3/VT-4 visual examination. VT-4 identified an incorrect support setting which was subsequently corrected and re-examined as satisfactory.

A reinspection will be conduct d on selected supports on the Shutdown Cooling System next refueling to assure no excessive movement has occurred.

The following main steam line supports were visually examined as follows:

Support No.	
SHP-H-83	Subjected to a VT-3/VT-4 visual examination and found to be acceptable.
SHP-H-81	Subjected to a VT-3/VT-4 visual examination and found to be acceptable.
SHP-H-91	Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable condition (loose bolts) which was subsequently corrected and re-examined as satisfactory. In accordance with ASME IWC-2430 an additional examination was conducted and found to be acceptable.
SHP-H-3	Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable condition (lack of thread engagement) which was determined not to be inservice-induced. This condition was corrected and subsequently re-examined as satisfactory.
SHP-H-65	Subjected to a VT-3/VT-4 visual examination and found acceptable.
SHP-H-74	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-induced unacceptable condition (lack of thread engagement) which was subsequently corrected and re-examined as satisfactory.
SHP-H-71	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-induced unacceptable condition (loose bolt) which was subsequently corrected and re-examined as satisfactory.
SHP-RH-3	Subjected to a VT-3 visual examination and found acceptable.
SHP-H-66	Subjected to a VT-3/VT-4 visual examination and found acceptable.
SHP-H-92	Subjected to a VT-3/VT-4 visual examination and found acceptable.
SHP-H-91R	Subjected to a VT-3 visual examination and found acceptable.
SHP-H-81R	Subjected to a VT-3 visual examination and found acceptable.
SHP-H-65R	Subjected to a VT-3 visual examination which identified an unacceptable nonservice-induced condition (lack of thread engagement). Subsequently corrected and re-examined as satisfactory.
SHP-H-71R	Subjected to a VT-3 visual examination and found acceptable.
SHP-H-75	Subjected to a VT-3 visual examination and found acceptable.

The following Feedwater System supports were visually examined as follows:

Support No.	
WCBD-H-12	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-139	Subjected to a VT-3/VT-4 visual examination. VT-3 identified an unacceptable condition (loose nuts) which was subsequently corrected and re-examined as satisfactory. In accordance with IWC-2430, an additional examination was conducted and found acceptable.
WCBD-H-130-1	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-3	Subjected to a VT-3/VT-4 visual examination and found acceptable.
WCBD-H-138	Subjected to a VT-3/VT-4 visual examination. VT-3 identified a nonservice-induced unacceptable condition (improper support installment) which was subsequently corrected and re-examined as satisfactory.
WCB[illegible]-131	Subjected to a VT-3 visual examination and found acceptable.
WCBD-RH-122	Subjected to a VT-3 visual examination and found acceptable.
WCBD-RH-119	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-114	Subjected to a VT-3/VT-4 visual examination and found acceptable.
WCBD-H-6	Subjected to a VT-3/VT-4 visual examination and found acceptable.
WCBD-RH-117	Subjected to a VT-3 visual examination which identified an unacceptable condition (loose nuts) which was subsequently corrected and re-examined as satisfactory. An additional examination was performed in accordance with IWC-2430 and found to be acceptable.
WCBD-H-8	Subjected to a VT-3/VT-4 visual examination and found acceptable.
WCBD-H-126	Subjected to a VT-3 visual examination. VT-3 examination identified a nonservice induced unacceptable condition (lack of thread engagement) which was subsequently corrected and reinspected satisfactory.
WCBD-H-129	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-134-1	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-134-4	Subjected to a VT-3 visual examination and found acceptable.
WCBD-H-120-4	Subjected to a VT-3 visual examination and found acceptable.

OTHER CC-CE SUPPORTS

VC Heating A-1	Subjected to a VT-1 visual examination and found acceptable.
Penetration #4-6" BRL (EDCR 83-10)	Subjected to a base line VT-3 visual examination and found acceptable.

CODE CATEGORY C-F — PRESSURE RETAINING WELDS IN PIPING

The following Shutdown Cooling System piping welds were subject to liquid penetrant examination with no unacceptable indications noted:

SDC-L-15	SDC-4-6
SDC-3-16	SDC-4-21
SDC-3-19	SDC-4-24
SDC-3-22	SDC-4-27
SDC-3-25	SDC-3L-21
SDC-3L-20	SDC-4-39
	SDC-3L-22

The following Safety Injection System piping welds were liquid penetrant examined with no unacceptable indications noted:

SI-003	SI-093
SI-006	SI-096
SI-009	SI-117
SI-018	SI-120
SI-019	SI-126
SI-027	

Pressurizer piping weld PR-2-26 was liquid penetrant examined with no unacceptable indications noted.

The following Feedwater System piping welds were subjected to magnetic particle examination with no unacceptable indications noted:

FW-4-15	FW-4-18
FW-4-17	94-8D

No. 3 feedwater nozzle to pipe weld (MT and VT).

MAIN STEAM SYSTEM PIPING WELDS

MS-4-20	Subjected to a ultrasonic, magnetic particle, and visual examination with no unacceptable indications noted.
MS-1-15	Subjected to an ultrasonic, magnetic particle, and visual examination. One indication was recorded during the ultrasonic examination which was subsequently evaluated as a geometric reflector.
MS-2-6	Subjected to an ultrasonic, magnetic particle, and visual examination - no indications noted.

MS-2-7 Subjected to an ultrasonic, magnetic particle, and visual examination - no indications noted.

MS-2-11 Subjected to an ultrasonic, magnetic particle, and visual examination. One indication was recorded during the ultrasonic examination which was subsequently plotted and evaluated as being a geometric reflector in accordance with IWB-3514.5.

MS-2-12 Subjected to an ultrasonic, magnetic particle, and visual examination - no indications noted.

In accordance with the Integrated Safety Assessment Systematic Evaluation Program for Yankee Nuclear Power Station - NUREG 0825 (3-5.6). The first weld downstream of the non-return valves (all four loops) was magnetic particle and visually examined.

Only one indication was noted. During the magnetic particle examination of loop #2, a lap-like indication was revealed. Subsequently, an area of the indication was blended out to verify in fact it was a processing-related discontinuity. This indication was evaluated and determined to be nonservice-induced and accepted as is.

CODE CATEGORY C-H ALL PRESSURE RETAINING COMPONENTS

The following systems were subjected to system leakage tests:

1. Safety Injection System

2. Low Pressure Surge Tank Cooling

3. Shutdown Cooling

4. Service Water System

5. Vapor Container Heating System

6. Main Coolant Drain System

7. Charging and Volume Control System

8. Purification System

9. Chemical Shutdown System

No serious degradation or leakage was noted other than packing leaks which were subsequently corrected.

The following system hydrostatic tests were conducted after modification or repair to safety class systems. Testing was performed in accordance with IWA/IWC-5000; no serious degradation or leakage was identified:

System	Test Pressure	Duration	Procedure
Main Coolant Vent System	2200 psi	10 min	OP-2000.136
Low Pressure Safety Injection Header	787 psi	10 min	OP-2000.130
Charging and Volume Control	52 psi	10 min	OP-2000.130
Vapor Container Drain Header	190 psi	10 min	OP-2000.112.4
Pressure Control and Relief System	300 psi	10 min	OP-2000.123
Feedwater System	1214 psi	10 min	OP-2000.126
Bleed Line	3000 psi	10 min	OP-2000.127
Post Accident Hydrogen Vent System	102 psi	10 min	OP-2000.128/129
Valve Stem Leakoff	375 psi	10 min	OP-2000.132
Low Pressure Surge Tank	94 psi	10 min	OP-2000.134
Main Steam System	1035 psi	10 min	OP-2000.135
EBFP Steam Inlet	1170 psi	10 min	OP-2000.137

An Appendix "J" test was conducted on modifications to the low pressure surge tank safety valve discharge header. The test was conducted at 32 psi. Reference plant Procedure OP-2000.133. The test was conducted and found acceptable.

SAFETY CLASS 3

CODE CATEGORY D-A PRESSURE RETAINING COMPONENTS

Portions of the following systems were subjected to either inservice or system leakage tests and found acceptable:

1. Emergency Boiler Feedwater System
2. Main Coolant Vent System
3. Spent Fuel Pit Cooling
4. Primary Pump Seal Water
5. Service Water

System hydrostatic tests were conducted on the following systems after repair or modification and found acceptable:

System	Test Pressure	Duration	Procedure
1. Emergency Boiler Feed Pump Steam Inlet	188 psi	10 min	OP-2000.137
2. Primary Pump Seal Water	345/414 psi	15 min	OP-2000.124
3. Demineralized Water/Service Water Crosstie	165 psi	10 min	OP-2000.125

ADDITIONAL EXAMINATIONS

In addition to the code-required examinations, the following inspections were conducted:

A visual examination was performed on the reactor head underside to monitor the known flaws. This inspection revealed that no change has occurred.

A visual inspection was also performed on the cladding of #3 steam generator hot leg and cold leg. The visual inspection resulted in no unacceptable indications noted.

The pressurizer internals were subjected to a visual inspection. This inspection concluded that no change has occurred since the last visual inspection.

5.0 SAFETY VALVE TESTING

The following safety valves were subjected to testing and found to be acceptable:

SV-215A	Low Pressure Surge Tank Safety
SV-409A	Main Steam Safety
SV-409B	Main Steam Safety
SV-409C	Main Steam Safety
SV-409D	Main Steam Safety
SV-409E	Main Steam Safety
SV-409F	Main Steam Safety
SV-409G	Main Steam Safety
SV-409H	Main Steam Safety
SV-409I	Main Steam Safety
SV-409L	Main Steam Safety
PRSV-181	Pressure Code Safety Relief Valve

PRSV-182	Pressure Code Safety Relief Valve replaced this refueling. During plant operation, valve setpoint was found to be out of tolerance. See LER 84-11.

6.0 CONCLUSIONS

With the exception of several hydrostatic pressure tests (which will be completed within this second interval), the examinations performed during this outage complete the inservice inspection requirements of the Yankee Nuclear Station Technical Specifications for the third period of the second interval.

ATTACHMENT A

Procedure No. 53A0314
Subject: RPV [illegible]
Page 64 of 253
Plant/Unit YANKEE ROWE
Cal.Data Pkg. 0314-1

EVALUATION SHEET FOR RECORDABLE INDICATIONS

Supplement C

A. Zone number N/A   Evaluation Proc. No. 80A5535

B. Raster No. FE1P011A   Weld number RPV-FE-1

C. Indication number R2/3

D. Applicable ASME Code Standards used for evaluation:
Section XI SUMMER 1978   Article IWB 3000

E. Size of indication:
Length SEE BELOW   Depth 1.48 TO CLOSEST POINT
Width SEE BELOW   Plate thickness (t) 9.75" NOM

F. Characterization of Flaw Indication per Para. IWA-3320
1. Type of Flaw: SUBSURFACE PLANAR
2. Sketch of indication:
CIRC DEPTH of C5 x AX CORRECTION FACTOR 14.51 x .57 = 8.27
CIRC DEPTH of C1 x AX CORRECTION FACTOR 2.27 x .81 = 1.84
6.43

$\frac{6.43"}{5.35SD} = 1.2"/SD$

3. Flaw characteristic calculations:
$a = \frac{1.2 \Delta SD}{2} = .6 \times .7 = .42$ $\frac{a}{l} = .11$ $\frac{a}{t} = 4.3\%$

$l = 20.28° \times \frac{[(1.38)^2 + 19.12]\pi}{360°} = 3.9"$

G. Comparison of pertinent evaluation standard (Para. IWB 3512.1)
to actual flaw size. ALLOWS 2.6% WITH AN $\frac{A}{l}$ of .10

☐ Acceptable   ☒ Reportable SEE ATTACHED SHEET

Prepared By: [signature]   Approval [signature] SNT-TC-1A Level III

FORM 028-120/1 8/83   nes NUCLEAR ENERGY SERVICES, INC.

[illegible]ster No. FFIPCIIA Indication No R2/3

THIS COULD HAVE BEEN SIZED AS A LAMINAR INDICATION IN ACCORDANCE WITH FIG. IWB 2500-7(c) AND TABLE IWB-3512-2. THIS WOULD PUT IT WELL WITHIN ACCEPTABLE LIMITS.

THE a WOULD BE REDUCED TO .18 WITH A REALISTIC SIZING CRITERIA USING BEAM SPREAD.

$\frac{a}{l} = \frac{.18}{3.9} = .05 \qquad \frac{a}{t} = \frac{.18}{9.75} = 1.85\%$

THIS WOULD BE ALLOWABLE.

[signature] III

ATTACHMENT B

ABSTRACT

Finite element stress analyses for the inlet and outlet nozzles of the Yankee Pressurized Water Reactor (PWR) vessel subjected to postulated Level A, Level B, and most limiting Level C and D Transients were carried out at Westinghouse for the Yankee Atomic Electric Company. Complete data and calculation results are stored on microfiche.

Two-dimensional finite element models were constructed for the inlet and outlet nozzles, and the stresses due to the pressure and temperature loadings were computed for all transients. This report presents detailed stress results for critical regions in the two nozzles for selected transients.

During the inservice inspection of April-May 1984, an indication was discovered in the outlet nozzle of loop 1 which required further evaluation under the provisions of the ASME Code, Section XI. The stress analysis results which were calculated for the outlet nozzle were then used to perform a fracture evaluation to demonstrate the acceptability of the indication without repair.

ATTACHMENT B

SECTION 7

FRACTURE EVALUATION OF DETECTED INDICATION

7-1 INTRODUCTION

A fracture analysis per Section XI of the ASME Code has been performed to investigate the acceptability of the indication found in the outlet nozzle of loop 1 during the recent inservice inspection.

The indication is shown in Figure 7-1, and was found at the boundary between the nozzle forging material and the weld by which it is attached to the vessel. The indication was characterized as an embedded ellipse oriented circumferentially with its plane being directed nearly radially from the centerline of the nozzle, as shown in the figure. The indication was centered at an angle of 232° from top dead center with respect to the centerline of the nozzle.[8]. To evaluate the acceptability of the indication, it must be assumed to be a flaw.

The evaluation consists of three parts:

(1) Crack Growth Analysis - the final crack width of the initial flaw is to be determined as the basis for the calculation of maximum stress intensities K_I for normal/upset/test conditions.

(2) Evaluation of the critical depth for level A and B conditions.

(3) Evaluation of the critical flaw depth for level C and D conditions.

The acceptability is to be evaluated by the criteria of applied stress intensity factor of the Code Section XI and its appendices,[9] as described in Section 7-2.

7.2 CODE ACCEPTANCE CRITERIA

There are two sets of flaw acceptance criteria for continued service without repair in paragraph IWB-3600 of ASME Code Section XI. Namely,

1. Acceptance Criteria Based on Flaw Size (IWB-3611)
2. Acceptance Criteria Based on Stress Intensity Factor (IWB-3612)

Both criteria are comparable in accuracy for surface flaws in thick sections. The acceptance criteria (2) have been assessed by past experience to be less restrictive for thin sections, as well as for evaluation of embedded flaws in either thick or thin sections. For completeness, however, both criteria will be given.

7.2.1 CRITERIA BASED ON FLAW SIZE

The code acceptance criteria stated in IWB-3611 of Section XI are:

$a_f < .1\ a_c$ For Level A and B Conditions

and

$a_f \leq .5\ a_i$ For Level C and D Conditions

where

a_f - The maximum size to which the detected flaw is calculated to grow at the end of design life, or until the next inspection time.

a_c The minimum critical flaw size under level A and B conditions (upset and test conditions inclusive)

a_i - The minimum critical flaw size for initiation of nonarresting growth under level C and D conditions

To determine whether a flaw is acceptable for continued service without repair, both criteria must be met simultaneously.

7.2.2 CRITERIA BASED ON STRESS INTENSITY FACTOR

As mentioned in the proceeding paragraphs, the criteria used for the evaluation of embedded flaws in thin or thick sections, and surface flaws in thin sections are from IWB-3612 Section XI. Namely,

$$K_I \leq \frac{K_{Ia}}{\sqrt{10}} \quad \text{For level A and B conditions}$$

$$K_I \leq \frac{K_{Ic}}{\sqrt{2}} \quad \text{For level C and D conditions}$$

where

K_I = The maximum applied stress intensity factor for the flaw size a_f to which a detected flaw will grow, during the conditions under consideration, to the next inspection.

K_{IA} = Fracture toughness based on crack arrest for the corresponding crack tip temperature.

K_{Ic} = Fracture toughness based on fracture initiation for the corresponding crack tip temperature.

7.3 FATIGUE CRACK GROWTH ANALYSIS

7.3.1 EMBEDDED FLAW VS. SURFACE FLAW

Since the indication discovered by inspection is near the inside surface of the nozzle wall, prior to crack growth analysis being performed, the first item that has to be determined is whether the flaw should be considerrd as embedded as it apparently is, or, be considered as a surface flaw.

The dimensions and the location of the flaw are shown in Figure 7-1. The rule for treating the flaw is found in Section XI, and is:

$$\frac{2e}{t} < 1 - (2)\ (\frac{2a}{t})$$

Where the nomenclature is defined in Figure 7-1.

Dimensions as measured:

$t = 8.06"$ $S = 1.48"$ $e = 2.13$

$a = .42"$ $\ell = 3.9"$

(refer to Figure 7-1 for definitions)

then,

$$\frac{2e}{t} = 0.528$$

whereas,

$$1 - (2)\ (\frac{2a}{t}) = 0.792$$

$$\therefore \frac{2e}{t} < 1 - (2)\ (\frac{2a}{t})$$

therefore, the flaw can be considered as embedded.

7.3.2 OUTLET NOZZLE TRANSIENTS AND STRESSES

The transients used in the fatigue crack growth evaluation are the Level A and B transients developed in Section 2 of this report. The number of cycles for each of the transients has been provided earlier in Table 2-1 for one year of service, and these occurrances were used directly in the analysis.

The stresses used in the analysis were obtained from the detailed finite element results for the outlet nozzle as presented in Section 6 of this report. The stress values used as input to the analysis are provided in Table 7-1.

7.3.3 FATIGUE CRACK GROWTH ANALYSIS

In applying code acceptance criteria as introduced in Section 7.2, the final flaw size a_f used in criteria (2) is defined as the flaw size to which the detected flaw is calculated to grow at the end of 40 years design life, or until the next inspection time. In this work, periods in increments of ten years were used for reporting crack growth results.

The analysis procedure involves postulating an initial flaw at the location of interest and predicting the growth of that flaw due to an imposed series of loading transients. The input required for a fatigue crack growth analysis is basically the information necessary to calculate the parameter ΔK_I which depends on crack and structure geometry and the range of applied stresses in the area where the crack exists. Once ΔK_I is calculated, the growth due to that particular stress cycle can be calculated by equations given in Figure 7-2. This increment of growth is then added to the original crack size, and the analysis proceeds to the next transient. The procedure is continued in this manner until all the transients known to occur in the period of evaluation have been analyzed.

The transients considered in the analysis are all the design transients contained in the vessel equipment specification as shown in Section 2. These transients are spread equally over the design lifetime of the vessel. Faulted conditions are not considered because their frequency of occurrence is too low to affect fatigue crack growth.

Crack growth calculations were carried out for a range of flaw sizes, beginning with the detected indication size. For all cases the flaw was assumed to maintain a constant shape as it grew.

The stress intensity factor expression provided in Appendix A of Section XI was used directly, which requires linearizing the stresses. The flaw shape was set as described in Figure 7-1 with the eccentricity set at 2.13, as described earlier.

section 3) for the main steam line break, and are shown in Figure 7-4. For reference, the stress distribution for the small break LOCA is shown in Figure 7-5.

7.4.2 STRESS INTENSITY FACTOR CALCULATIONS

One of the key elements of the critical flaw size calculations is the determination of the driving force or stress intensity factor. This was done using expressions available from the literature. In all cases the stress intensity factor for the critical flaw size calculations utilized a representation of the actual stress profile rather than a linearization. This was necessary to provide the most accurate determination possible of the critical flaw size, and is particularly important for consideration of Level C and D conditions, where the stress profile is generally nonlinear and often very steep. The stress profile was represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1 \frac{x}{t} + A_2 \left(\frac{x}{t}\right)^2 + A_3 \left(\frac{x}{t}\right)^3$$

where x is the coordinate distance into the wall

t = wall thickness

σ = stress perpendicular to the plane of the crack

The stress intensity factor calculation for an embedded flaw was taken from work by Shaw and Kobayashi [10] which is applicable to an embedded flaw in an infinite medium, subjected to an arbitrary stress profile. This expression has been shown to be applicable to embedded flaws in a thick-walled pressure vessel in a recent paper by Lee and Bamford [11].

7.4.3 FRACTURE TOUGHNESS

The other key element in the determination of critical flaw sizes is the fracture toughness of the material. The fracture toughness has been taken directly from the reference curves of Appendix A, Section XI. In the transition temperature region, these curves can be represented by the following equations:

The crack growth rate curves used in the analysis were taken directly from Appendix A of Section XI of the ASME Code. The air environment curve was used for the embedded flaws analyzed, and is shown in Figure 7-2.

$$\frac{da}{dN} = (0.0267 \times 10^{-3})\ K_I^{3.726}$$

where, $\frac{da}{dN}$ = Crack growth rate, micro-inches/cycle

K_I = stress intensity factor range, ksi $\sqrt{in}$

$= (K_{Imax} - K_{Imin})$

7.3.4 FATIGUE CRACK GROWTH RESULTS

The analysis of all the embedded flaws showed negligible crack growth, as shown in Table 7-2. This result is not unexpected, since the crack growth rate for air environments is relatively low.

7.4 CRITICAL FLAW SIZE CALCULATIONS

7.4.1 SELECTION OF KEY TRANSIENTS

The key parameters used in the evaluation of any indications discovered during inservice inspection are the critical flaw parameters. These are as follows:

- (a_c) for the governing Level A and B conditions
- (a_i) for the governing Level C and D conditions.

The selection of the governing transient for Level A and B conditions can be done easily based on the results of the fatigue crack growth analyses, where all the transients were considered, and stress intensity factors calculated for each one. The governing transient was found to be the reactor trip (with loss of MCP) and the stress distribution in the section of interest is provided in Figure 7-3. For the Level C and D conditions the two most limiting transients are the small break LOCA and the main steam line break. For the outlet nozzle region, the highest stresses were found (in cross

$$K_{Ic} = 33.2 + 2.806 \exp. [0.02 (T-RT_{NDT} + 100°F)]$$

$$K_{Ia} = 26.8 + 1.233 \exp. [0.0145 (T-RT_{NDT} + 160°F)]$$

where K_{IC} and K_{IA} are in ksi $\sqrt{in}$.

The upper shelf temperature regime requires utilization of a shelf toughness which is not specified in the ASME Code. A value of 200 ksi in has been used here. This value is consistent with general practice in such evaluations, as shown for example in reference [12] which provides the background and technical basis of Appendix A of Section XI.

The toughness used for both the governing conditions was 200 ksi $\sqrt{in}$, because the temperature throughout the wall thickness remains above 400°F at all times for both. The temperature distributions at the governing time step for each transient are shown in Figures 7-6 and 7-7, for the Reactor Trip and Main Steam Line Break respectively. The RT_{NDT} for the Yankee Rowe outlet nozzles was estimated at 60°F, but does not play a role, because of the high metal temperature.

7-5 RESULTS OF FRACTURE EVALUATION

The fracture and fatigue evaluation of the indication found in the Yankee Rowe Outlet Nozzle of Loop 1 showed that the indication meets the acceptance criteria of IWB 3600 of the ASME Code Section XI, therefore, no repair will be necessary.

The criteria are summarized below, along with the results of the fracture evaluation. The first step in the evaluation is the calculation of fatigue crack growth according to the methods suggested in Section XI. The reference crack growth law for air environments was used, and the resulting crack growth was negligible for the entire remaining life of the reactor vessel, as shown in Table 7-2. Therefore, the indication as detected was used in the evaluation. The configuration of the indication is shown in Figure 7-1.

For level A and B conditions the allowable stress intensity factor is:

$$K_I < \frac{K_{Ia}}{\sqrt{10}} = \frac{200}{\sqrt{10}} = 63.2 \text{ ksi } \sqrt{in}$$

As seen in Figure 7-8, the maximum applied K_I for all the Level A and B transients is 5.2 ksi $\sqrt{in}$ found in the reactor trip transients, which is far below the allowable.

For level C and D conditions the allowable stress intensity factor is:

$$K_I < \frac{K_{Ic}}{\sqrt{10}} = \frac{200}{\sqrt{10}} = 141 \text{ ksi } \sqrt{in}$$

The maximum applied stress intensity factor for the governing Level C and D condition (the main steam line break), is shown in Figure 7-9, and is found to be 4.8 ksi $\sqrt{in}$, which is well below the allowable as well.

Therefore the indication easily meets both the acceptance standards of Section XI, IWB 3600, for acceptance without repair, for the entire remaining life of the plant.

TABLE 7-1

STRESS RESULTS USED FOR FATIGUE CRACK GROWTH ANALYSIS

TRANSIENT	LOAD STEP max/min	i (Inside Stress)	o (Outside Stre
1. Reactor Trip with Loss of MCP	max	19.544	7.249
	min	0.0	0.0
2. Heatup and Cooldown 50°F/Hr	max	3.742	0.0
	min	0.0	2.320
3. Heatup and Cooldown 100°F/Hr	max	6.435	0.0
	min	0.0	2.354
4. Unit Loading and Unloading (0-50%)	max	0.0	2.845
	min	0.0	2.625
5. Unit Loading and Unloading (50-100%)	max	0.0	3.376
	min	0.0	2.845
6. Step Load Decrease	max	0.0	3.514
	min	0.0	3.124
7. Steady State Fluctuations Temp. Period Used	max	0.0	3.347
	min	0.0	3.280
8. Steady State Fluctuations Pressure Period Used	max	0.0	3.345
	min	0.0	3.279
9. Feedwater Cycling	max	0.0	2.507
	min	0.0	2.281
10. Loss of Load	max	0.0	3.999
	min	0.0	3.022
11. Loss of Power	max	0.0	4.137
	min	0.0	2.715
12. MCP Startup & Shutdown RCS Venting	max	0.0	0.389
	min	0.0	0.6
13. MCP Startup & Shutdown Pump Restart Conditions	max	0.0	2.269
	min	0.0	2.160
14. MCP Startup & Shutdown Hot Plant Conditions	max	0.0	2.247
	min	0.0	2.074
15. Reactor Trip from Full Power (Normal)	max	1.343	3.70
	min	0.0	3.477
16. Reactor Trip	max	4.968	3.513
	min	0.0	3.088
17. Inadvertant Steamline NRV Closure (Affected Loop)	max	2.496	4.559
	min	0.0	3.315
18. Inadvertent Steamline NRV Closure (Other Loops)	max	3.293	3.912
	min	0.0	2.983
19. Excessive Feedwater Flow (Variation 1)	max	9.776	1.749
	min	0.0	[illegible]
20. Execssive Feedwater Flow (Variation 2)	max	11.039	[illegible]
	min	0.0	[illegible]
21. Primary Side Leakage	max	0.0	[illegible]
	min	0.0	[illegible]

TABLE 7.2

RESULTS OF FATIGUE CRACK GROWTH ANALYSIS

(note crack length "a" is half of embedded crack width

INITIAL CRACK LENGTH	CRACK LENGTH AFTER YEAR 10	20	30	40
.420*	.42008	.42016	.42024	.42032
.500	.50011	.50023	.50034	.50045
1.000	1.00051	1.00103	1.00154	1.00205

*Actual detected size

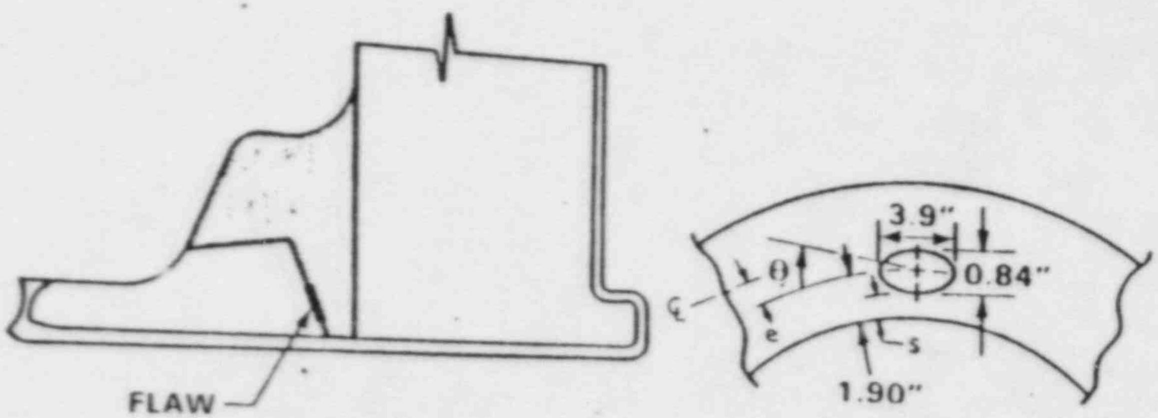


Figure 7.1 Geometry of Indication Outlet Nozzle, Loop 1

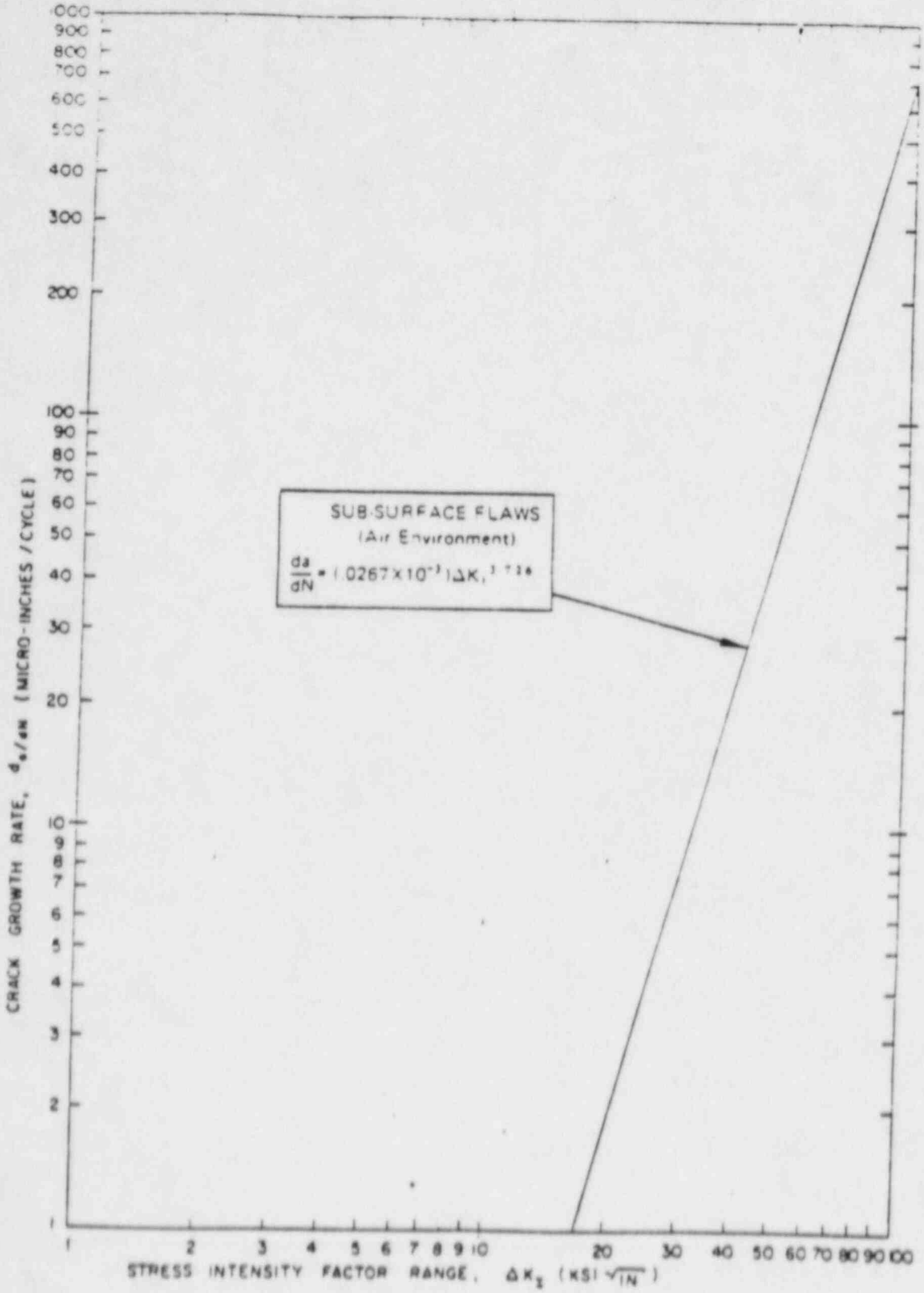


Figure 7.2 Upper Bound Fatigue Crack Growth Data for Reactor Vessel Steels

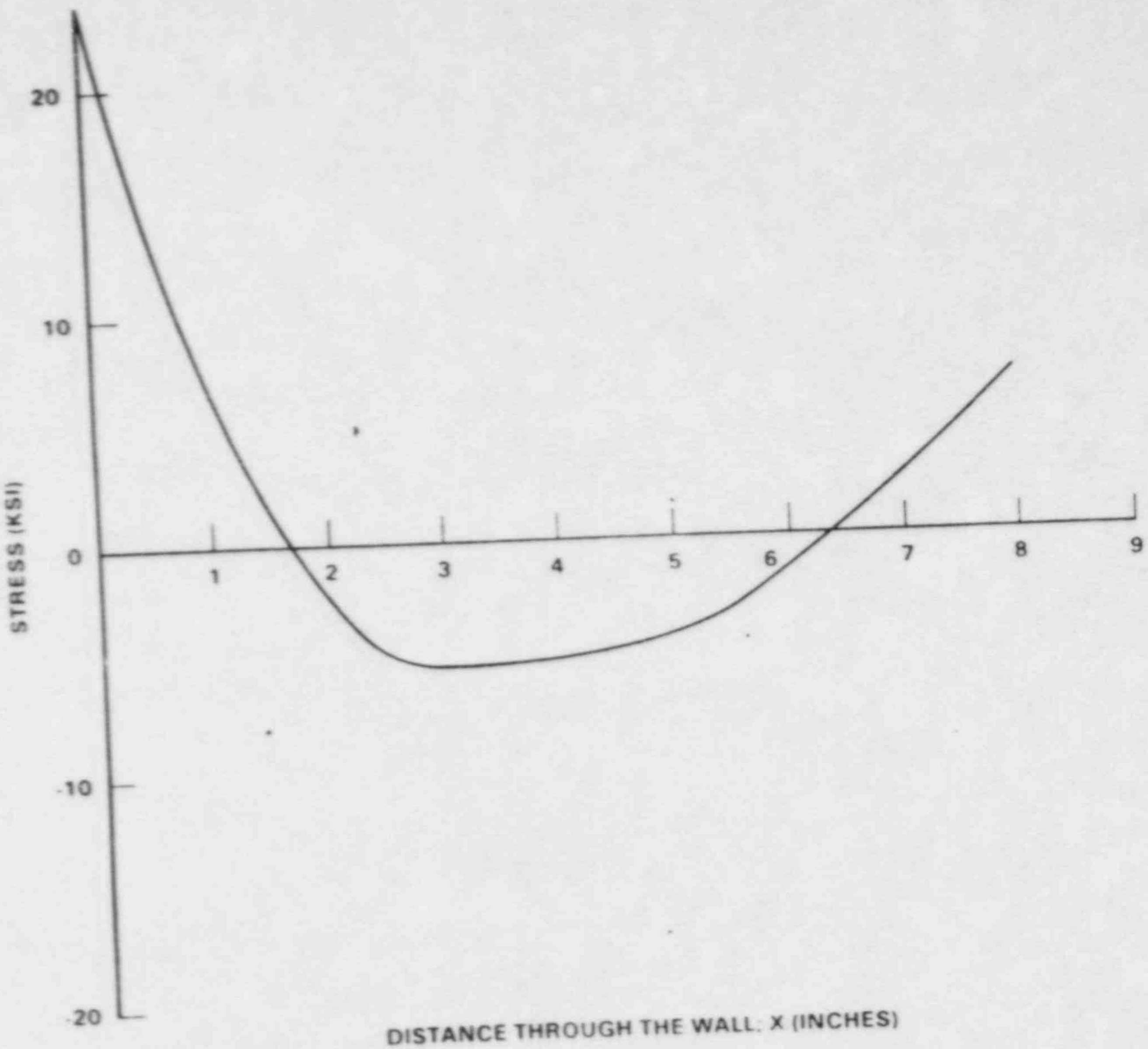
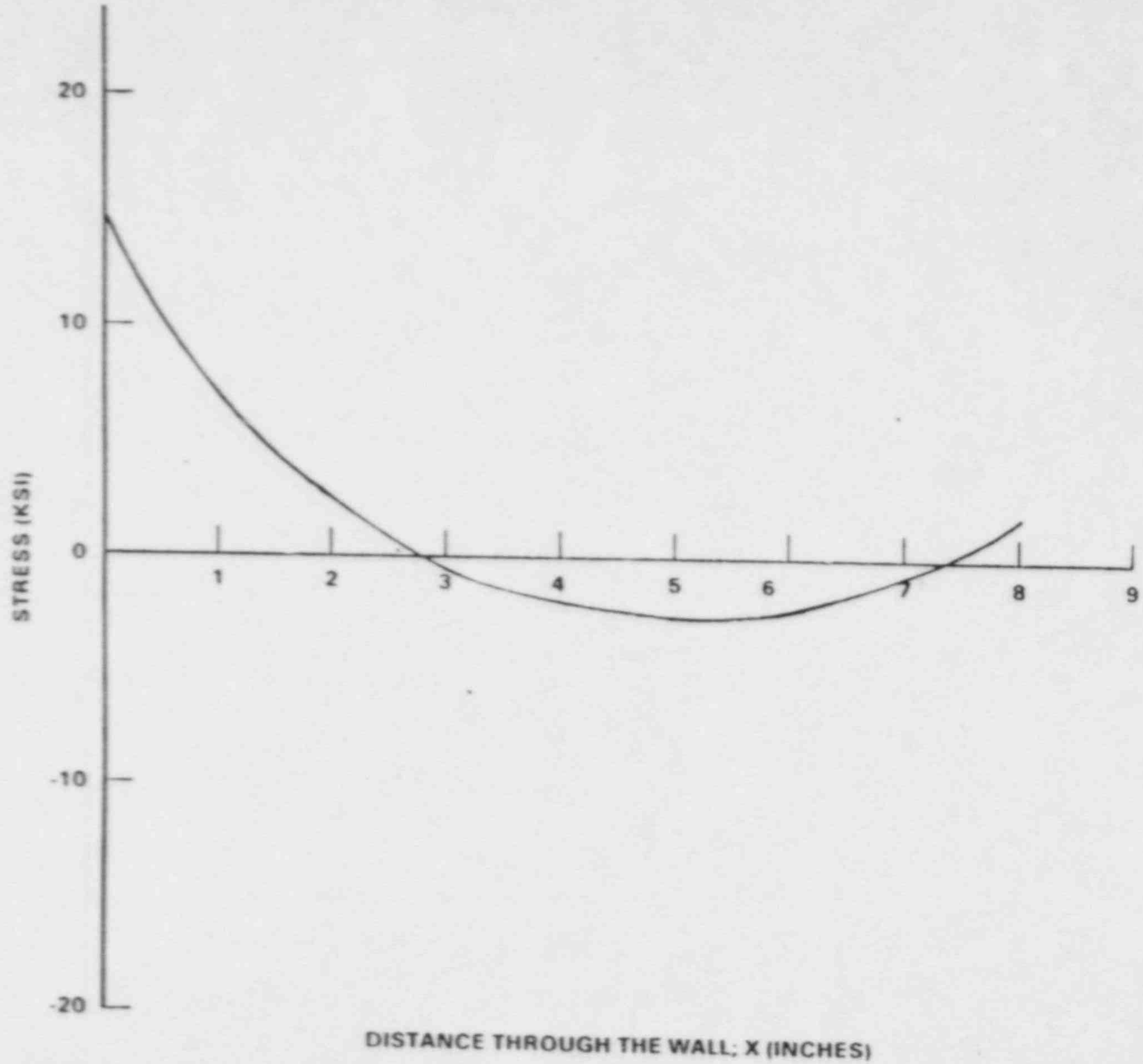


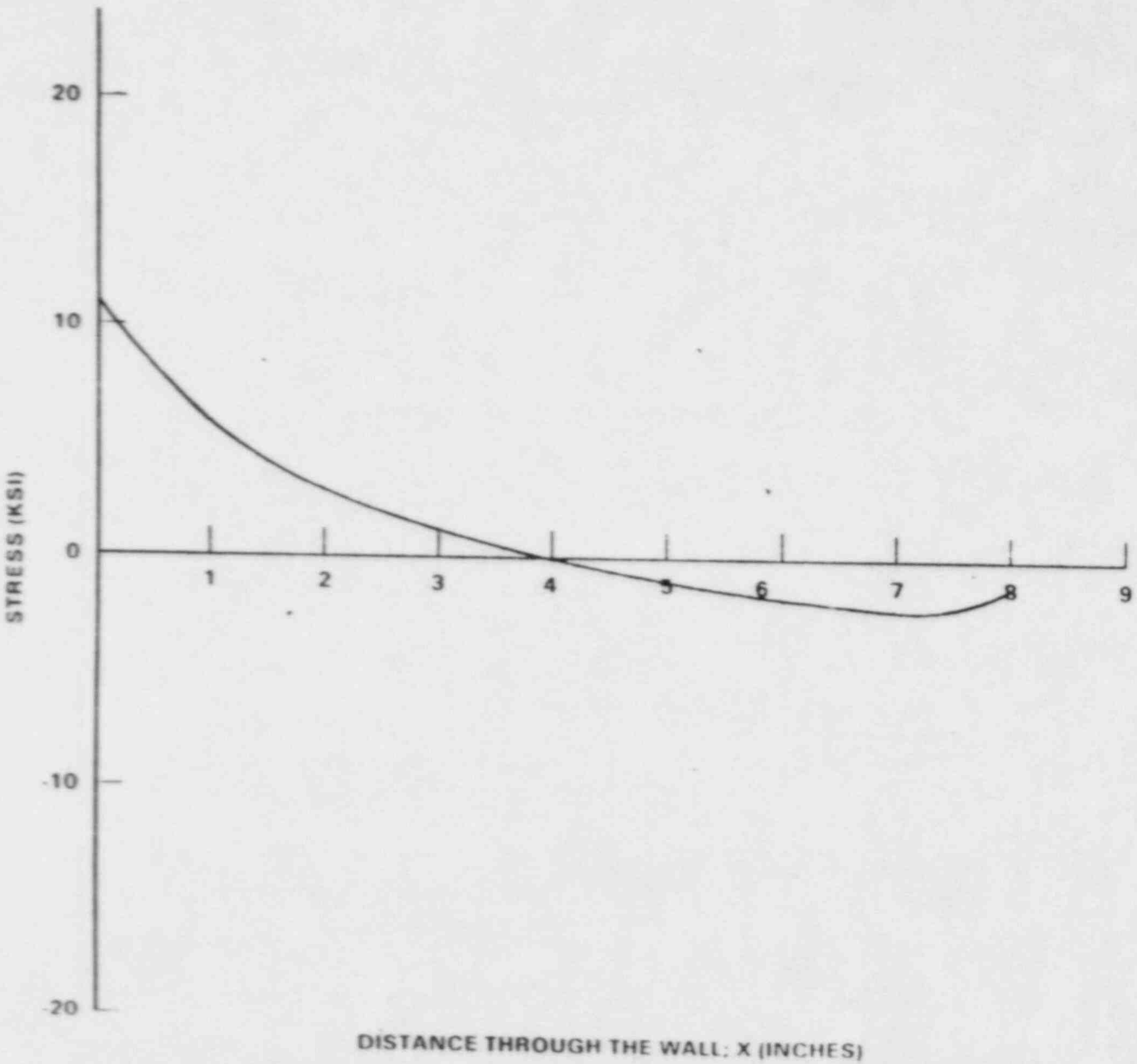
Figure 7.3 Stress Distribution at Section 3 during Reactor Trip Transient at 3.33 Minutes

Figure 7.4 Stress Distribution at Section during Main Steamline Break at 9.167 Minutes



9070 3

Figure 7.5 Outlet Nozzle Stresses at Section 3 at 50 minutes into Small LOCA



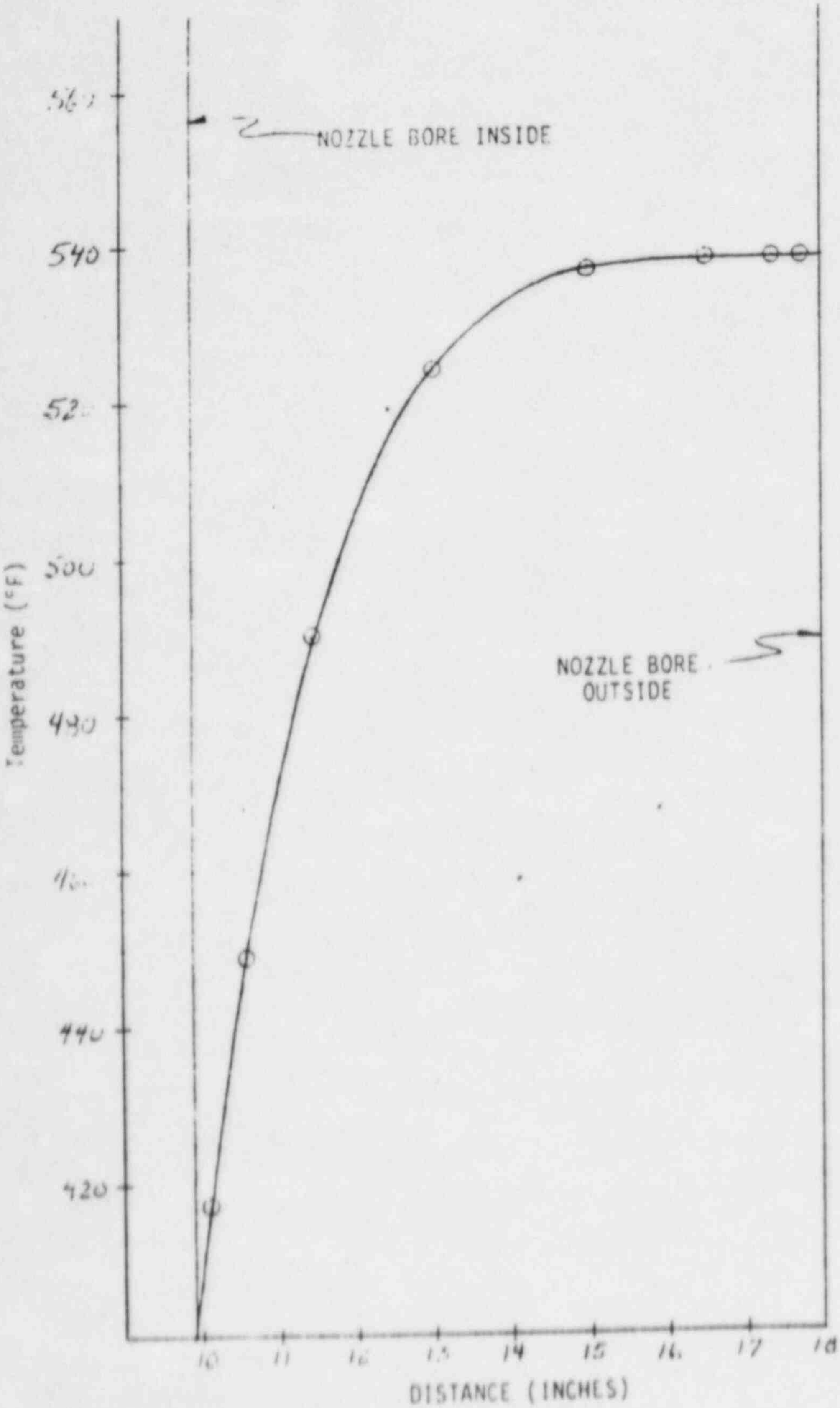


Figure 7.6 Temperature Distribution at Section 3 at 3.33 Minutes into Reactor Trip with Loss of MCP - Outlet Nozzle

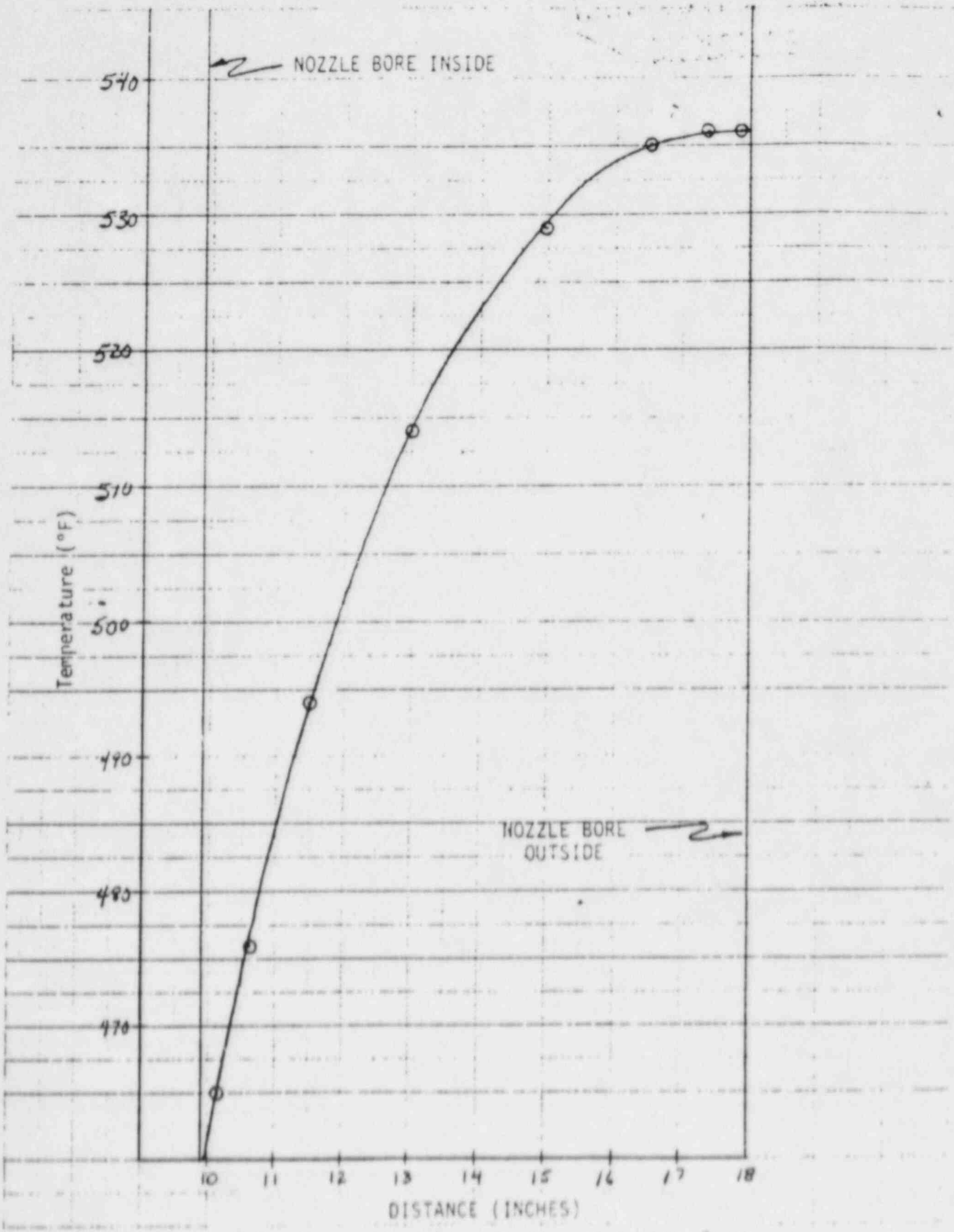


FIGURE 7.7 Temperature Distribution at Section 3 at
9.167 minutes into Main Steamline Break

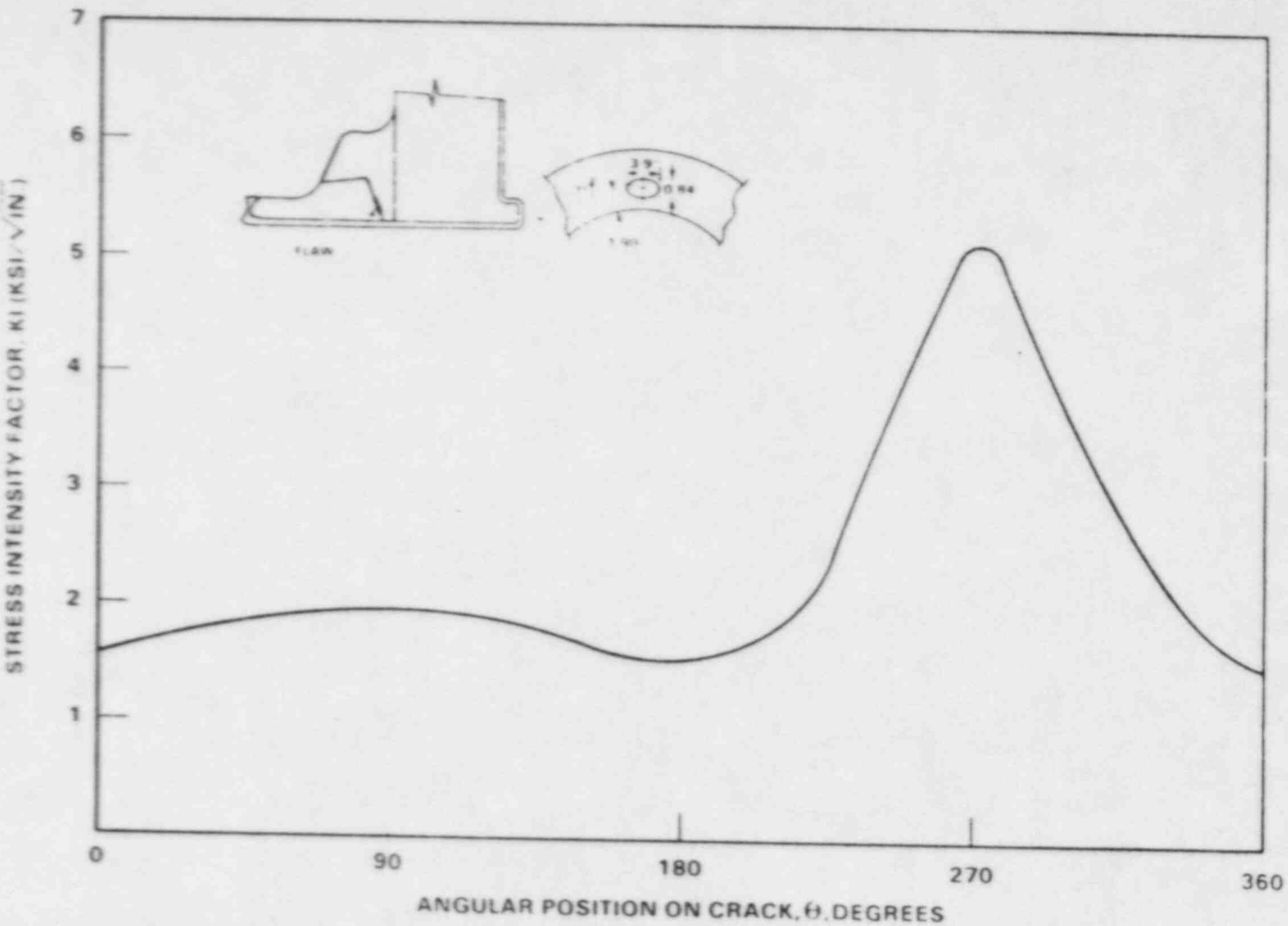


Figure 7-8 Applied Stress Intensity Factor around the Periphery of the Indication - Governing Level A and B Condition (Reactor Trip)

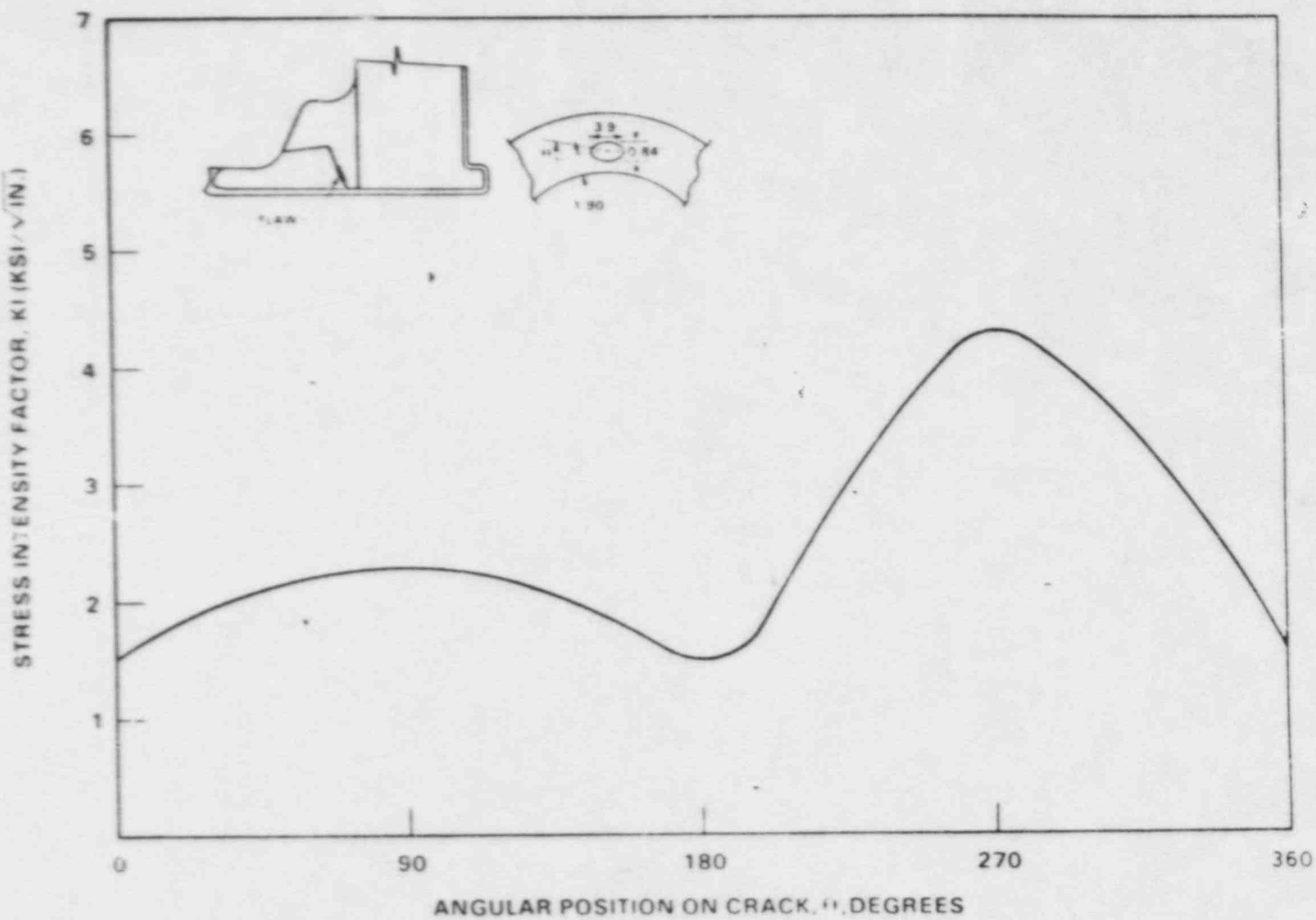


Figure 7.9 Applied Stress Intensity Factor around the Periphery of the Indications - Governing Level C & D Condition (Main Steamline Break at 9.167 minutes

SECTION 8

SUMMARY AND CONCLUSIONS

Applicable transients were developed for the thermal and stress evaluation of the inlet and outlet nozzles of the Yankee Pressurized Water Reactor Vessel. Using all the pressure and temperature data from these developed transients and finite element models developed for each nozzle, the stresses at 154 time steps were calculated for each finite element run for Levels A and B transients. Thirty two time steps were calculated for Level C and D conditions.

These stress analysis results were used to perform fracture and fatigue analyses to demonstrate the acceptability of the indication discovered in the outlet nozzle of loop 1 during the inservice inspection of April 1984.

The maximum applied stress intensity factors for all transients analyzed are well below the allowable stress intensity factors. The indication easily meets both the requirements of Section XI, IWB 3600, for acceptance without repair, for the entire remaining life of the plant.

7YR 84-96

COMMITMENT TRACKING SYSTEM

PROCTR Applicability ☐ Y ☒ N

Commitment Responsibility:

☐ NSD ☐ Plant ☐ Both

Outgoing Letters:

Responsibility: ____________________

Due Date: ____________________

NSD Service Request Required ☐ Y ☒ N

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T. Rennell
N. Fetherston (Rowe)
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SUPPLEMENT 1
ATTACHMENT A

UT

IMS DOCUMENT INPUT FORM - L

REQUIRED INFORMATION

2. PLANT YR 3. CLASSIFICATION TYPE L

4. RECORD TYPE NO. Ø2 CØ2 ØØ1

5. IMS SUBJECT NO. BØ3 Ø1 Ø1

6. DATE WRITTEN 84-Ø914

7. DOCUMENT FORM

8. DOCUMENT LOCATION

SUPPLEMENTAL INFORMATION

9. PRIMARY DOCUMENT NO. FYR 84-96

11. TITLE Inservice Inspection Examination Report

12. KEYWORDS ISI; Report

13. ORIGINATOR

14. RECEIVER

17. REFERENCE DOCUMENT

21. WORK ORDER NO.

ACTION* ADD/CHANGE/DELETE (CIRCLE ONE)

1. ACCESSION NUMBER 171602

* If ADD, provide all document information and circle ADD.
If CHANGE, provide document Accession Number, corrected document characteristic information and circle CHANGE. If deleting all document information for a particular characteristic, write DELETE in appropriate entry.
If DELETE, provide document Accession Number and circle DELETE.

ATTACHMENT C

NOTE: This section added-Revision 1.

nes

EVALUATION OF INDICATION
REPORTED IN YANKEE ROWE
NOZZLE NUMBER RPV-FF-1

Prepared by: G D Martens
George Martens
Technical Consultant

1. SCOPE

1.1 This document summarizes a study to determine the size of the flaw that produced reportable indication #R2/CH.3 in nozzle number RPV-FF-1 at the Yankee Rowe Nuclear Power Station.

1.2 The determination of flaw size is based on comparison of data obtained from the nozzle with data from a special test block.

2. BACKGROUND

2.1 At examination time, NES advised Yankee Atomic that the indication was sized as reportable pursuant to ASME Code/Reg. Guide 1.150 criteria, but that the actual size of the flaw was probably within acceptable limits. NES reasoned that the conservative indication sizing criteria applied to a longitudinal wave examination more than doubles the actual flaw size in the specific location and orientation of detection.

2.2 NES subsequently demonstrated the oversizing parameter on an NES test block and estimated the actual flaw size to have a 2a dimension of 0.31", resulting in an allowable a/t of 1.59%.

2.3 Yankee Atomic had SWRI fabricate a special test block to the original Yankee Rowe specification to assure accurate data. The SWRI test block certifications are in Addendum B. This block was fabricated with 0.200" wide and 0.400" wide slots to bracket the NES estimate of flaw size. This report covers the analysis of the flaw size based on data from this block.

3. EXAMINATION OF THE SPECIAL TEST BLOCK

3.1 A dimensional inspection of the special test block YR-D-00-056 (SWRI Drawing No. D-8328-600) was performed. All important dimensions are within specified limits except some from the clad surface. This is realistic and totally acceptable. The data is in Addendum C.

3.2 The equipment used for this examination is the same as that used at Yankee Rowe except for the coaxial cables and examination head (search unit) which are in hot storage. Identical cable type and lengths were used. The examination head used is the identical spare that was taken to Yankee Rowe but was not put in service.

3.3 The calibration was in total conformance to the procedure used at Yankee Rowe (NES Document No. 83A0314, Rev. 1), with two exceptions:

-the sensitivity was increased 2 dB to adjust for the measured sensitivity loss between curved and flat entry

-the sweep was adjusted to expand the time of interest for more accurate measurement

Examination documentation is in Addendum D.

3.4 The block was scanned in the same orientation and with the same index (.2") as the nozzles at Yankee Rowe. Examination data is in Addendum E.

4. DATA ANALYSIS

4.1 The composite data sheet is attached. The plots depict apparent depth in the material from left to right. This represents Code/ Reg. Guide indication sizing of the 2a dimension.

4.2 The flaw was initially considered to be at 20° because it was located at a 20° fusion line, the amplitude was into saturation, and it was not detected with any other angles. The test block data substantiates this, however, it shows that the flaw is not located as deep as measurement indicated. The data clearly shows that the angle of the reflector strongly influences the apparent depth by changing the effective detection angle. This effect is greater with the longitudinal beam because of its wider beam divergence. The peak amplitude points are at 2.02" for the 0.2" 30° reflector, 2.52" for the 0.2" 20° reflector, and 2.82" for the 0.4" 20° reflector. It is understandable that the larger reflector at the same angle has a stronger peak effect because the reflected energy has less divergence.

4.3 No attempt is made to establish correction for the ℓ dimension because of the beam divergence difference between flat and curved entry in this orientation. NES estimates a possible oversize in this dimension of up to 0.75", however, this would not substantially effect the aspect ratio.

5. FLAW SIZE

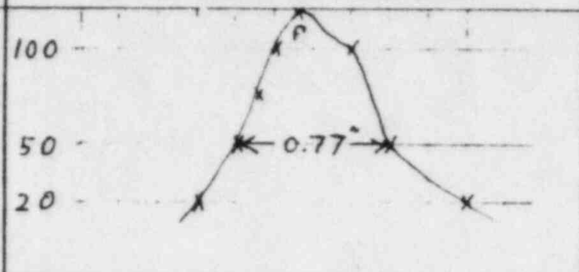
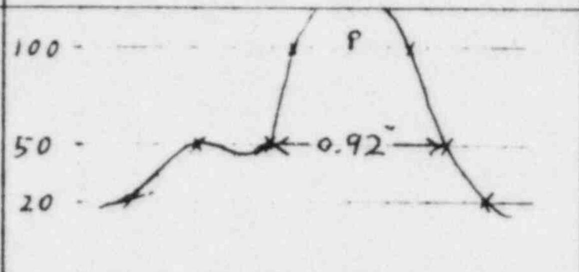
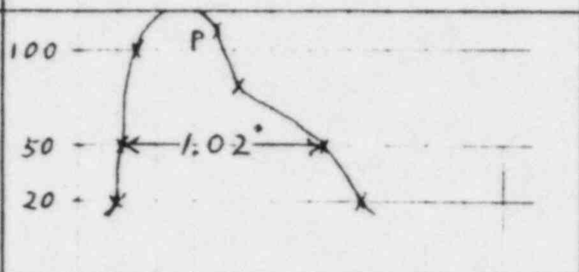
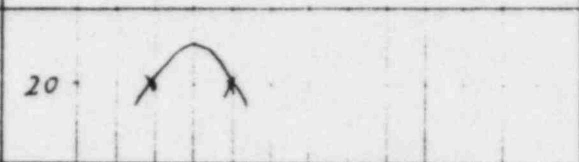
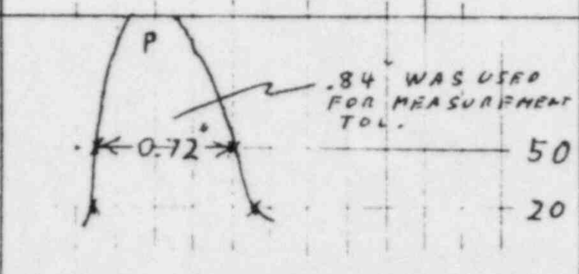
5.1 The 20° slots, 0.2" and 0.4" wide, bracket the data from the flaw. The 0.2" slot was oversized by 0.58", and the 0.4" slot was oversized by 0.54". The average oversize dimension of 0.56" subtracted from the 0.84" flaw data results in a 2a dimension of 0.28". This results in an a/t of 1.4%. The decrease in depth discussed in paragraph 4.2 changes the center of the flaw depth from 1.9" to 1.35". The estimated decrease in oversize dimension due to decreased effective beam width is 15%. This results in a 2a dimension of 0.36" and an a/t of 1.8%. This is the estimated flaw size. The tolerances in measurement of ± .1" result in 1.3% to 2.4% outside limits.

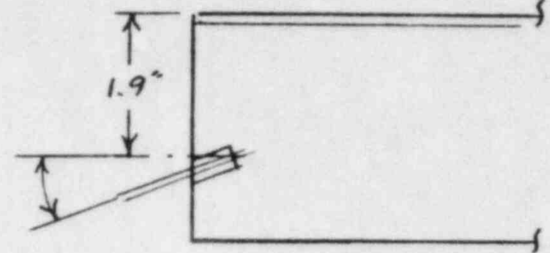
6. CONCLUSION

6.1 All data clearly indicates that the flaw is within the allowable limit of 2.4% (IWB 3512.1). The flaw had to be at least 20° to produce the peak amplitude in excess of 150% DAC. If the flaw is over 20°, the "oversizing" would have been even greater.

6.2 The possibility of this flaw being "service induced" is highly remote. The shape, orientation, location, and size indicate that the flaw is lack of fusion to the parent material at part of one weld pass.

COMPOSITE DATA

TARGET	PERPENDICULAR PLOT OF APPARENT 2a DIMENSION	KNOWN 2a	AMOUNT OVERSIZED	Estimated 2a	,a/l	a/t	Allowable
Special Test Block 20°, .2" Wide		Cos. 20° x 0.2 = .19	0.77 -0.19 = 0.58				
20°, .4" Wide		Cos. 20° x 0.4 = .38	0.92 -0.38 = 0.54				
30°, .2" Wide		Cos. 30° x 0.2 = .17	1.02 -0.17 = 0.85				
10°, .2" Wide		Cos. 10° x 0.2 = .20	< 50% would have been sized differently				
Apparent Depth in Inches	1.6 2.0 2.4 2.8 3.2 3.6 1.4 1.8 2.2 2.6 3.0 3.4						
Indication in Nozzle		N.A.	20° avg. x .85 depth correction 0.48	0.84 -0.48 = 0.36	.18/ 3.9 = .046	.18/ 9.75 = 1.8%	with an aspect ratio of 0.05 2.4%



Depth to the center of all slots is 1.9"

ADDENDA

A. YANKEE ROWE INDICATION DATA

B. SPECIAL TEST BLOCK DATA FROM SWRI

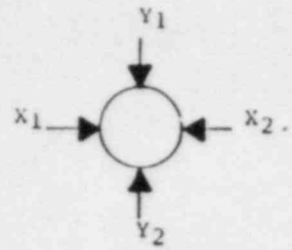
C. SPECIAL TEST BLOCK DIMENSIONAL TEST DATA

D. DOCUMENTATION ON EXAMINATION OF SPECIAL TEST BLOCK

E. DATA FROM SPECIAL TEST BLOCK

ADDENDUM A

YANKEE ROWE INDICATION DATA



SUPPLEMENT A

INDICATION REPORT SHEET

Cal. Data Pkg. 0314-1

Raster No. FF1PC2I1A

Video Tape No. 33

Weld No. RPV-FF1 Weld ℄ N/A

IND. NO.	CH. NO.	BM. DIR.	∡	0° BR SD	MAXIMUM PEAK % DAC	SD	X	Y	LOCATION OF ___ % DAC END POINTS X_1	SD	X_2	SD	Y_1	SD	Y_2	SD	LOCATION OF ___ % DAC END POINTS X_1	SD	X_2	SD	Y_1	SD	Y_2	SD
R5	4	⊥ OUT	60	N/A	80	1.4	46.9	63.9	45.49	1.4	54.08	1.4	63.6	1.7	63.4	1.9	45.47	1.9	50.23	1.4	63.6	1.7	63.4	1.4
R6	4	⊥ OUT	60	N/A	130	3.6	50.27	62.8	55.95	3.6	54.65	3.6	63.6	3.9	62.6	3.9				N/A				
R7	4	⊥ OUT	60	N/A	75	1.7	13.03	63	11.93	1.7	13.03	1.7	63	1.7	63	1.7	9.55	1.7	14.12	1.7	63	1.7	63	1.7
R11	4	⊥ OUT	60	N/A	120	4.9	135.55	57.5	135.07	4.4	137.93	4.9	59.9	3.8	57.5	4.1				N/A				
R22	3	⊥ IN	60	N/A	150+	1.6	224.9	66.5	242.2	1.6	221.92	1.6	67.5	2.1	66.0	1.9	255.15	1.6	259.63	1.6	67.4	2	66.4	1.4

N/A

REMARKS Correction to previous Supplement A dated 5/3/84

EXAMINERS
1 [signature] Level II Date 5/10/84
2 ______ Level ___ Date ______
3 ______ Level ___ Date ______
REVIEWER ______ Date ______

NUCLEAR ENERGY SERVICES, INC.

FORM 028-116/1 1/83

Procedure No. 83A0314
Subject: RPV NOZZLES
Page 64 of 253
Plant/Unit YANKEE ROWE
Cal.Data Pkg. C314-1

EVALUATION SHEET FOR RECORDABLE INDICATIONS

Supplement C

A. Zone number N/A Evaluation Proc. No. 80A5535

B. Raster No. FF1FCT1A Weld number RPV-FF-1

C. Indication number R2/3

D. Applicable ASME Code Standards used for evaluation:
Section XI Summer 1978 Article IWB-3000

E. Size of indication:
Length SEE BELOW Depth 1.48 TO CLOSEST POINT
Width SEE BELOW Plate thickness (t) 9.75" NOM

F. Characterization of Flaw Indication per Para. IWA-3320
1. Type of Flaw: SUBSURFACE PLANAR
2. Sketch of indication:
CIRC DEPTH OF C5 x AX CORRECTION FACTOR 14.51 X.57 = 8.27
CIRC DEPTH OF C1 x AX CORRECTION FACTOR 2.27 X.81 = 1.84
6.43

$\frac{6.43''}{5.35 SD} = 1.2''/SD$

3. Flaw characteristic calculations:
$a = \frac{1.2 \Delta SD}{2} = .6 \times .7 = .42$ $\frac{a}{l} = .11$ $\frac{a}{t} = 4.3\%$

$l = 26.28° \times \frac{[(1.38)^2 + 14.12]\pi}{360} = 3.9''$

G. Comparison of pertinent evaluation standard (Para. IWB-3512.1) to actual flaw size. ALLOWS 2.6% WITH AN $\frac{a}{l}$ OF .10

☐ Acceptable ☒ Reportable SEE ATTACHED SHEET

Prepared By: [signature] III Approval [signature] SNT-TC-1A Level III

NES NUCLEAR ENERGY SERVICES, INC.

FORM 028-120/1 8/83

) Raster No. FFIPOIIA Indication No. R2/3

THIS COULD HAVE BEEN SIZED AS A LAMINAR INDICATION IN ACCORDANCE WITH FIG. IWB 2500-7(c) AND TABLE IWB-3512-2. THIS WOULD PUT IT WELL WITHIN ACCEPTABLE LIMITS.

THE a WOULD BE REDUCED TO .18 WITH A REALISTIC SIZING CRITERIA USING BEAM SPREAD.

$\frac{a}{\ell} = \frac{.18}{3.9} = .05 \qquad \frac{a}{t} = \frac{.18}{9.75} = 1.85\%$

)

THIS WOULD BE ALLOWABLE.

[signature] LIII

)

ADDENDUM B

SPECIAL TEST BLOCK DATA FROM SWRI

Yankee Rowe Nuclear Power Station
Special Test Block
Certifications

Purchase Order No. 105016
Dated February 12, 1985

Identification No. YR-D-00-056

SOUTHWEST RESEARCH INSTITUTE
SAN ANTONIO HOUSTON

SOUTHWEST RESEARCH INSTITUTE

P-NUMBER CLASSIFICATION FOR CALIBRATION BLOCKS

CALIBRATION BLOCK YR-D-00-056 IS HEREBY CLASSIFIED AS P-NUMBER 3 GROUP 3 IN ACCORDANCE WITH SECTION IX 1980 EDITION OF THE ASME BOILER AND PRESSURE VESSEL CODE. THE P-NUMBER CLASSIFICATION FOR THIS CALIBRATION BLOCK IS SUBSTANTIATED WITH THE ATTACHED CHEMICAL ANALYSIS REPORT FOR SA508, CL2, HT Q2Q 106 NQT IN ACCORDANCE WITH THE MATERIALS SPECIFICATION SECTION II OF THE ASME BOILER AND PRESSURE CODE.

DESIGN CRITERIA

This special test block was designed by Yankee Atomic Electric Company. This is a verification that the block was fabricated in accordance with Yankee Atomic Electric Company, Purchase Order No. 105016 requirements and to the requirements specified on Southwest Research Institute Drawing No. D-8328-600.

* Due to the availability of the ultrasonic procedure Southwest Research Institute was unable to perform the final ultrasonic calibration on the machined reflectors. Final ultrasonic acceptance to be performed by Yankee Atomic Electric Company.

ATTACHMENTS

MILL TEST REPORT/CHEMICAL ANALYSIS REPORT
PRELIMINARY UT DATA SHEETS (SwRI)
DIMENSIONAL DATA SHEETS (MACHINE SHOP - QC)
* FINAL UT ACCEPTANCE DATA SHEETS (SwRI) * N/A
DRAWING (SwRI) D-8328-600
Welding Electrode Certifications
Postweld Heat Treat Certification

[signature]	Project Manager	March 5, 1985
SIGNATURE	TITLE	DATE

Robert L. Edwards

SWRI FORM PN-1-1

G+W **Lenape Forge**
Gulf + Western Manufacturing Company
P.O. BOX 536
WEST CHESTER, PENNSYLVANIA 19380

Phone: (215) 793-1500
TWX: 510-663-0372
Telex: 83-5453
Telecopier: (215) 793-1500 Ext. 264

MATERIAL TEST REPORT S.O. NO. C1351-03 DATE 8/23/83

PURCHASER SOUTHWEST RESEARCH DISTRIBUTOR

PURCHASER'S ORDER NO. 11288 DISTRIBUTOR'S ORDER NO.

ITEM NO.	QTY.	PRODUCT	SPEC	HEAT NO.	M/O No.	BHN
4 1	2	Rgh. Frg. Blk to Fin. 8 X 14 X 16	A508-74A CL. 2	Q2Q106NQT	133J	
3 2	1	Rgh. Frg. Blk to Fin. 8 X 14 X 36		Q2Q106NQT	134J	

CHEMICAL ANALYSIS AND MECHANICAL PROPERTIES

HEAT NO.	C	MN	P	S	SI	CR	NI	MO	V		HEAT TREATMENT
Q2Q106NQT	.23	.82	.008	.007	.27	.33	.72	.59	.04	LADLE	NORM, QUENCH & TEMPER

SwRI
P. O. 11288
P. R. 338469
LOG 1811

HEAT NO.	TENSILE	YIELD	ELONG % in 2"	R.A. %	BHN	IMPACT V@ +40°F	REMARKS LAT EXP	% SHEAR
Q2Q106NQT	Long'l							
0°	85,000	63,500	26	68		144-128-110	.085-.094-.086	70-70-60
180°	93,500	72,000	23	68		150-139-198	.091-.090-.103	70-60-100

We hereby certify the above results to be correct as contained in the records of the Company.

Charles R. Alakin Jr.

Sw.R.I. SONIC INSTRUMENT CALIBRATION RECORD FOR ATTENUATION/LAMINATION EXAMINATION

PROJECT No.: 17-8328-206
SITE: SwRI (YANKEE ROWE)
DATE: (DAY-MON.-YR.) 20 FEB. 85
TIME: (24 HR CLOCK) 1420
SHEET No. YR-001

EXAMINER (SIGNATURE) Tommy E. Jackson
SNT LEVEL II
PROCEDURE No. IX-FE-120 REV N/A DEV. N/A
INSTRUMENT SONIC MARK I ☒ II ☐ III ☐
SERIAL No. 04334E
EXAMINATION AREA(S): CAL BLOCK ID. NO. YR-D-00-056 TJ TJ

EXAMINER (OPERATOR) N/A
SNT LEVEL N/A
COUPLANT: GLYCERINE ☒ WATER ☐ OTHER (SPECIFY) ______

REFERENCE BLK No.: SWRI-IIW-10

PRELIMINARY UT CALIBRATION ONLY TJ
FOR LAMINATION SCAN TJ

SEARCH UNITS

BRAND	SERIAL NUMBER 0° (L.S.)	0° (ATT.)
SWRI	1346	N/A
SIZE	1" Rd.	
NOMINAL FREQ (MHz)	2.25	

CALIBRATION VERIFICATION

TIME	1451				
INITIALS	TJ	F	I	N	AL
TIME					
INITIALS					

INSTRUMENT SETTINGS

REJECT	0	N/A
DEC:	OFF	
FREQUENCY	2	
DELAY	099-1	
MATL. CAL:	097	
RANGE	10	
DAMPING:	MIN	
REP RATE:	3K	
VIDEO	NORM	
FILTER:	HI	
JACK USED:	R	
MODE OF TRANS:	NORMAL	

REMARKS: NO RECORDABLE INDICATIONS TJ

PRELIMINARY UT DATA

10 SCREEN DIVISIONS = 10 INCHES OF METAL

MODE: LONGITUDINAL

LONGITUDINAL ATTENUATION

BASIC CALIBRATION BLOCK No. N/A

1ST ECHO ____ dB ____ LINES OF AMPLITUDE
2ND ECHO ____ dB ____ LINES OF AMPLITUDE
Δ dB ____ (2ND ECHO - 1ST ECHO) N/A

BASIC CALIBRATION BLOCK No. N/A

1ST ECHO ____ dB ____ LINES OF AMPLITUDE
2ND ECHO ____ dB ____ LINES OF AMPLITUDE
Δ dB ____ (2ND ECHO - 1ST ECHO) N/A

CABLE TYPE
RG 174 ☐
RG 62 ☐ LENGTH ____ IN N/A
OTHER ______

BASIC CALIBRATION BLOCK No. N/A

1ST ECHO ____ dB ____ LINES OF AMPLITUDE
2ND ECHO ____ dB ____ LINES OF AMPLITUDE
Δ dB ____ (2ND ECHO - 1ST ECHO) N/A

REVIEWED BY Jerry McFarland SNT LEVEL: II DATE: 22 Feb 85

Sw. R.I. STRAIGHT BEAM LAMINATION EXAMINATION RECORD

PROJECT No.: 17-8328-206	SITE: SWRI (YANKEE ROWE)	DATE: (DAY – MON. – YR.) 20 FEB 85	TIME (24-HR CLOCK) SHEET STARTED: 1431 SHEET ENDED: 1447	SHEET No. YR-002
EXAMINATION AREA: (SYSTEM/COMPONENT) N/A	(LINE/SUBASSEMBLY) N/A	(IDENTIFICATION) CAL BLOCK ID. NO. YR-D-00-056	L_0 LOCATION N/A	W_0 LOCATION N/A

EXAMINER: T. E. JACKSON	SNT LEVEL: II	PROCEDURE No. IX-FE-120 REV N/A	CALIBRATION SHEET (S): YR-001	MEASURED THICKNESS UP: N/A	℄: N/A	DOWN: N/A	CROWN HEIGHT: N/A	ATTENUATION UP: N/A	DOWN: N/A	WELD TYPE (←FLOW→): N/A
EXAMINER: N/A	SNT LEVEL: N/A						CROWN WIDTH: N/A			WELD LENGTH: N/A

IND No	% LOSS OF BW	IND AMP % F.S.	POSITION 1 L_1	W_1	W_2	MP	POSITION L	W_1	W_2	MP	POSITION L	W_1	W_2	MP	POSITION 2 L_2	W_1	W_2	MP	SEARCH UNIT LOCATION	REMARKS	INI.
	NO RECORDABLE INDICATIONS																		NON CLAD SIDE		TJ
																				PRELIMINARY UT DATA	

REMARKS:

EXAMINATION AREA LIMITATIONS: (IF NONE, SO STATE): NONE TJ

REVIEWED BY: [illegible]	SNT LEVEL: II	DATE: 20 Feb. 85	PAGE 1 OF 1

DOSCO, INC.
P. O. BOX 20227
4900 HIGHWAY 90 EAST
SAN ANTONIO, TEXAS 78220

NO. 2

QUALITY CONTROL INSPECTION REPORT

Contract No. 85017 Item Special test block

P/N YR-D-00-056 Qty 1 ea NSN S.W.R.I.

D-8328-600

			Acc	Rej			Acc	Rej			Acc	Rej
1		1	/		16	.200 ±.005	/		31			
2		2.080	/		17	.500	/		32			
3		1.900 Ref.	/		18	2.080	/		33			
4		20°	/		19	1.900 Ref.	/		34			
5		.400 ±.005	/		20	20°	/		35			
6		.500	/		21	.200 ±.005	/		36			
7		1 1/2	/		22	12	/		37			
8		6	/		23	.500	/		38			
9		3	/		24	.200 ±.005	/		39			
10		3 ± 1/32	/		25	30°	/		40			
11		2.000	/		26	2.200	/		41			
12		1.900	/		27	1.900 Ref	/		42			
13		1/4	/		28				43			
14		.500	/		29				44			
15		10°	/		30				45			

Tolerances on Dimensions
(unless otherwise specified)

Fractions ± 1/16
.X ±
.XX ±
.XXX ± .010
Ang. ± 30'

REMARKS:

P.O. 64632
LOT 1959

ACCEPTED

LOT:

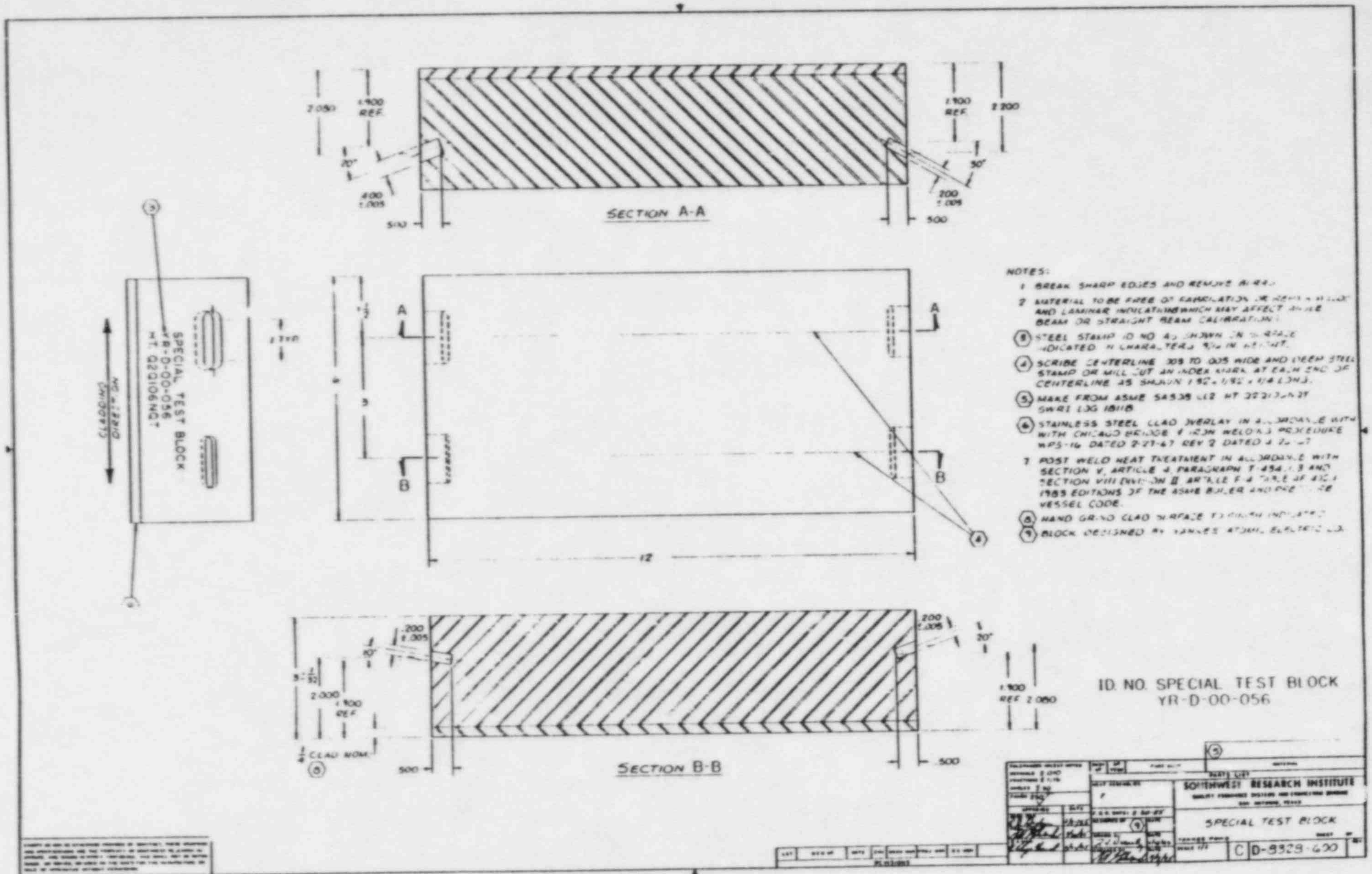
REJECTED

Number of Samples 1
Number ACCEPTED 1
Number REJECTED 0

Signed [signature]

Date MAR 6 1985

Page 1 of 1



SECTION A-A
SECTION B-B
NOTES:
1 BREAK SHARP EDGES AND REMOVE BURRS
2 MATERIAL TO BE FREE OF FABRICATION OR ... AND LAMINAR INDICATIONS WHICH MAY AFFECT ANGLE BEAM OR STRAIGHT BEAM CALIBRATION.
3 STEEL STAMP ID NO AS SHOWN ON SURFACE INDICATED IN CHARACTERS IN HEIGHT.
4 SCRIBE CENTERLINE .005 TO .005 WIDE AND DEEP STEEL STAMP OR MILL CUT AN INDEX MARK AT EACH END OF CENTERLINE AS SHOWN
5 MAKE FROM ASME SA533 ... SWRI LOG 18118
6 STAINLESS STEEL CLAD OVERLAY IN ACCORDANCE WITH WITH CHICAGO BRIDGE & IRON WELDING PROCEDURE WPS-16 DATED 2-27-67 REV 2
7 POST WELD HEAT TREATMENT IN ACCORDANCE WITH SECTION V, ARTICLE 4, PARAGRAPH T-434 ... AND SECTION VIII DIVISION II ARTICLE F-4 ... 1983 EDITIONS OF THE ASME BOILER AND PRESSURE VESSEL CODE.
8 HAND GRIND CLAD SURFACE TO FINISH INDICATED
9 BLOCK DESIGNED BY YANKEE ATOMIC ELECTRIC CO.
SPECIAL TEST BLOCK
YR-D-00-056
HT. Q2Q106NQT
CLADDING DIRECTION
ID. NO. SPECIAL TEST BLOCK
YR-D-00-056
SOUTHWEST RESEARCH INSTITUTE
SPECIAL TEST BLOCK
C D-9329-420

MATERIAL TEST REPORT
Welding Products Division

TELEDYNE McKAY

P.O. Box 1509 • 850 Grantley Road • York, Pa 17405

CSS-309 11/79
WELDERS SUPPLY CO
SAN ANTONIO TX 78238

Your Order No 1936
Marked For

Our Order No. G7680
Date Shipped 04/27/83

Material Description and Specifications

Item	Weight lb	Size, in	Classification	Coating
1	60	5/32	E309-15	DC Lime
2			E309-16	AC-DC
3			E309-16	DCT
4				
5				
6				

Item	Heat No	Lot No	Specification
1	626218	2237797	AWS A5.4-78; ASME SFA5.4
2			AWS A5.4-78; ASME SFA5.4
3			AWS A5.4-78; ASME SFA5.4
4			
5			
6			

Typical Deposit Chemistry

Item	% C	% Mn	% P	% S	% Si	% Cr	% Ni	% Mo	%Cu	%V	%Cb+Ta	Ferrite
1		1.8										7 FN
2	.07	1.0	.022	.015	.35	23.4	12.5	.15	.11			7 FN
3					.75	24.2						10 FN
4												
5												
6												

Typical Deposit Mechanical Properties

Item	Tensile psi	Yield, psi	% Elong	Charpy V-Notch Impact, ft.lb.
1				
2	88,000	67,000	37	
3				
4				
5				
6				

5/32" DCT needs the following statement:
The 5/32" DCT electrode is not recommended for use in the vertical and overhead positions.

For Notarization

SwRI
P. O. 00942
P. R. 320608
LOG 1767 C

We hereby certify that the above product has been classified in accordance with the listed specifications and conforms to all applicable requirements thereof

TELEDYNE McKAY

Glenn Villemy 4/27/83

CSS-308
MATERIAL TEST REPOR
Welding Products Division

TELEDYNE McKAY
P.O. Box 1509 • 850 Grantley Road • York, Pa. 17405

CSS-308 11/79
WELDERS SUPPLY CO
SAN ANTONIO TX.

Your Order No. 2279
Marked For

Our Order No. C9365
Date Shipped 09/28/83

Material Description and Specifications

Item	Weight, lb	Size, in	Classification	Coating
1				
2	60	3/16	E308-16	AC-DC
3				
4				
5				
6				

Item	Heat No	Lot No	Specification
1			
2	J1475	2538052	AWS A5.4-78; ASME SFA5.4
3			
4			
5			
6			

Typical Deposit Chemistry

Item	%C	% Mn	% P	% S	% Si	% Cr	% Ni	% Mo	%Cu	%V	%Cb+Ta	Ferrite	
1													
2	.06	1.0	.022	.015	.40	20.2	9.6	.15	.10			7 FN	
3													
4													
5													
6													

Typical Deposit Mechanical Properties

Item	Tensile, psi	Yield, psi	% Elong	Charpy V-Notch Impact, ft.lb.	5/32" DCT needs the following statement: The 5/32" DCT electrode is not recommended for use in the vertical and overhead positions
1					
2	86,000	65,000	45		
3					
4					
5					
6					

For Notarization

CP&I
P. O. 20035
P. R. 320683
LOG 1823

We hereby certify that the above product has been classified in accordance with the listed specifications and conforms to all applicable requirements thereof

TELEDYNE McKAY

Glenn Villarreal 9/28/83

HEAT TREATMENT LOG

Welding Research & Development
Quality Assurance Systems and Engineering Division
Southwest Research Institute

PROJECT NO. 17-8328-206 DATE 2/19/85

PART NO./ID NO. Cal. Block / Yankee Atomic

DRAWING NO. D-8328-600

STRESS-RELIEF	TEMPERATURE°F	TIME AT TEMPERATURE
	1150°F	2 hours

SPECIAL INSTRUCTIONS °F/PER HOUR

HEAT-UP-RATE From 800°F — 95°/hr.

COOL-DOWN RATE To 800°F — 87.5°/hr

REMARKS None

PERSON PERFORMING STRESS RELIEF:

WR Schick
SIGNATURE

2/19/85
DATE

Date 5 MARCH 85 SwRI QA Signoff R. Ward

Project Number 17-8328-206 Utility Atomic Yankee.

Plant —

QUALITY ASSURANCE REVIEW - PREOPERATIONAL ACTIVITY
PRIOR TO SHIPMENT OF TEST EQUIPMENT AND
DEPARTURE OF PERSONNEL

List Ultrasonic Calibration Blocks:

Serial Number	P#	Mill Test	Insp. Rept.	Dwg.	Remarks
YR-D-00-056	✓	✓	✓	✓	U.T. DATA

List Magnetic Particle Equipment (yokes):

Yoke Serial Number	Yoke Calibration Date	Serial Number of Block

Yoke Calibration valid for 6 months

POA-1-C
12/79

ADDENDUM C

SPECIAL TEST BLOCK DIMENSIONAL TEST DATA

nes

CALIBRATION BLOCK DIMENSIONAL INSPECTION

Project Name and No.: Yankee Rowe 5610-100

Calibration Block No.: Special Test Block YR-D-00-056

Drawing No. and Rev.: SWRI Drawing D-B32A-600 Rev Ø

Dimensional Specification No. and Rev.: N/A

Inspected by: Vincent Pravitz Level II Date 3-29-85

Reviewed by: Level Date

Spool/Blank ( ) Prelim. Machining ( ) Fabricated Block (✓)

No.	Drawing Dimension		Limits	Act. Dim.	Remarks
1	12	Block Length	± 1/16	12.012	OK
2	6	Block Width	± 1/16	6.030	OK
3	3	Block Thickness	± 1/32	3.030 — 3.092	NG
4	.200	Width of notch right side view A-A	±.005	0.195 go 0.205 no go	OK
5	.500	Depth of notch right side view A-A	±.010	0.497	OK
6	30°	Angle of cut of notch at right side of A-A	± 30'	30° 0'	OK
7	2.200	Depth from clad surf. of notch ℄ Right A-A	±.010	2.270	NG
8	1	Length of notch at right side of view A-A	± 1/16	0.999	OK
9	1½	Location of notch from block edge to center. (Notch at right side of view A-A)	± 1/16	1.51	OK
10	.200	Width of notch right side of B-B	±.005	0.195 go 0.205 no go	OK
11	.500	Depth of notch right side of B-B	±.010	0.495	OK
12	20°	Angle of cut of notch at right side B-B	± 30'	20° 15'	OK
13	2.080	Depth from clad surf. of notch ℄ Right B-B	±.010	2.122	NG
14	1	Length of notch at right side of B-B	± 1/16	1.005	OK
15	3	Location of notch from adjacent notch to center (Notch at right side of view B-B)	± 1/16	3.006	OK
16	.400	Width of notch left side of A-A	±.005	0.397	OK
17	.500	Depth of notch left side of A-A	±.010	0.491	OK
18	20°	Angle of cut of notch at left side of A-A	± 30'	20° 20'	OK
19	2.080	Depth from clad surface of notch ℄ left A-A	±.010	2.110	NG
20	1	Length of notch at left side of A-A	± 1/16	0.993	OK

CALIBRATION BLOCK DIMENSIONAL INSPECTION

INSPECTION RECORD

Block No: YR-D-00-056

ITEM NO.	DRAWING	DIMENSION	LIMITS	ACTUAL DIM.	REMARKS
21	$1\frac{1}{2}$	Location of notch from block edge to center (Notch at left side of A-A)	$\pm\frac{1}{16}$	1.508	ok
22	.200	Width of notch left side of B-B	±.005	0.195 go 0.205 no go	ok
23	.500	Depth of notch left side of B-B	±.010	0.493	ok
24	2.000	Depth from clad surface of notch ℄ left B-B	±.010	1.998	ok
25	1	Length of notch at left side of B-B	$\pm\frac{1}{16}$	0.997	ok
26	3	Location of notch from adjacent notch to center (notch at left side of B-B)	$\pm\frac{1}{16}$	3.005	ok
27	10°	Angle of cut of notch at left side of B-B	±30'	10° 0'	ok
28	3	Distance between scribe lines	$\pm\frac{1}{16}$	3	ok
29	$\frac{3}{16}$	BLOCK IDENT. LETTERING	$\pm\frac{1}{16}$	$\frac{3}{16}$	ok
30	N/A	SPECIAL TEST BLOCK, YR-D-00-056 HT Q2Q106HQT	N/A	N/A	ok
31	$\frac{1}{4}$	Cladding Thickness	NOM.	$\frac{17}{64}$ – $\frac{22}{64}$	ok
32					
33					
34					
35					
36		NOTE: SECTION A-A IS DRAWN			
37		INCORRECTLY ON THIS DRAWING.			
38					
39					
40					
41					
42					
43					
44					
45					

ADDENDUM D

DOCUMENTATION ON EXAMINATION OF SPECIAL TEST BLOCK

Report No. 3310484

ULTRASONIC SEARCH UNIT QUALIFICATION

SEARCH UNIT DATA

Manufacturer: NES

Serial No.: C- 4

Mfgr. Designation: 80D8120-1 spare

Nominal Angle: 45 ° (Check One)

☒ Long. ☐ Shear

Nominal Frequency: 2.25 MHz

Element Dimensions: 1x1

Element Material: LTZ-2

Tuning: (Check One)

☒ Tuned ☐ Untuned

Intended Use: (Check One)

☐ Contact ☒ Immersion ☐ Other

Connector: BNC attached

Peak Distance:

NA in NA

(Check One)

Flat Y_o NA Peak(for dual) NA

Line Focus NA Point Focus NA

____ Other: NA

TEST DATA

Test Block S/N: 4608

Block Material: Carbon Steel

Block Dimension Used: 4" radius

Cable Type: Mini noise coax

Cable Length: 75'

Pulser/Receiver

Gain: 40 Energy: 4

Receiver Attenuation: 6 db

Receiver Damping: 50 Ohms

3 db Limits: Lower 1.96 MHz

Upper 2.35 MHz

Center Freq.: 2.155 MHz

6 db Limits: Lower 1.88 MHz

Upper 2.44 MHz

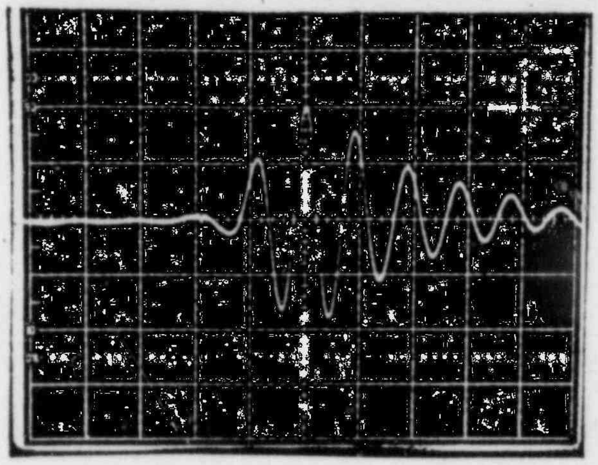
...sured Peak Freq.: 2.155 MHz

... idth: .56 MHz

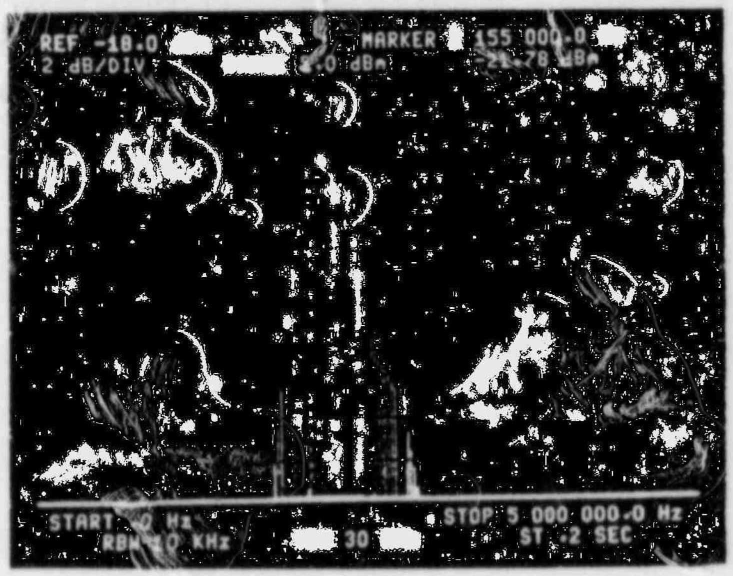
...ed Angle: 45 ° in H_2O

REAL TIME WAVEFORM

.5 uS/Div. .5 Volts/Div.



SPECTRUM ANALYSIS



REMARKS

CERTIFICATION PERFORMED BY: [signature]

DATE: 4/14/84 ACCEPTED REJECTED

BY: [signature] UT LEVEL III

RECERTIFY BEFORE: 4/14/85

Form 028-137/2 8/83 REF: SOA5536

nes NUCLEAR ENERGY SERVICES, INC.

RF WAVEFORM RECORD SHEET

CHECK ONE: ☑ PRE-EXAM FIELD WAVEFORM, SITE: NA

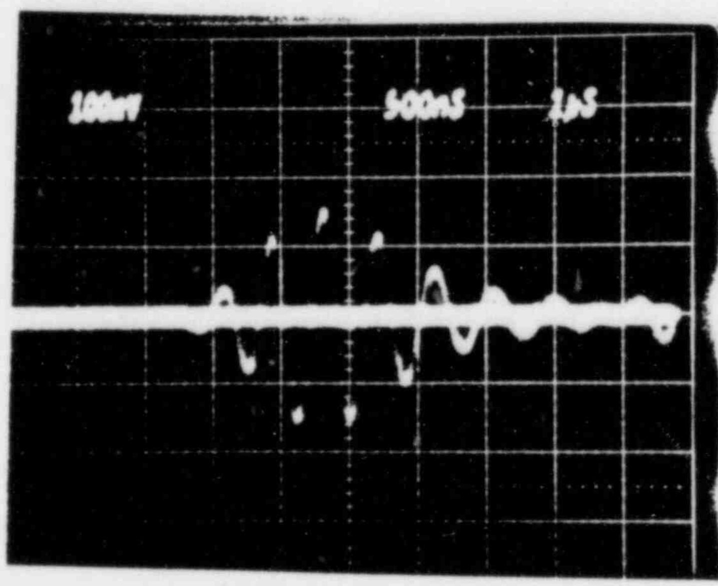
INFO ~~☒~~ POST-EXAM FIELD WAVEFORM, SITE: ~~Yankee Rowe~~ NES DWS 4/03/85 DWS 4/03/85

TRANSDUCER	PULSER RECEIVER	ULTRASONIC INSTRUMENT
S/N: C-4	S/N: 11528-9	S/N: 10244-9
MFGR.: NES	MODEL NO.: PR35	MFGR.: AI
MFGR. DESIGNATION: NA	**PULSER RECEIVER SETTINGS**	STYLE: S-80
ELEMENT DIMENSIONS: 1X1	ATTEN. db: 48	90-DAY LINEARITY DUE DATE: 3-19-85
CABLE TYPE: RG174/u	REF. (Fine Gain): MAX	**REFERENCE BLOCK**
CABLE LENGTH: 300' + 75'	FILTER: 2	S/N: 80D8842-1
FIXTURE OR HEAD	MODE: FW	TYPE: Carbon Steel
S/N: 80D8120-1	FREQ.: 2.25	MATERIAL: Carbon Steel
ANGLE: 60° L	Q: 4	REFLECTOR USED: A
	PULSE LENGTH: MAX	COMMENTS:
	JACK (T or R): T	
	TEST (Thru or Normal): Thru	

RPP 1832 5000-3
CH #1

VOLTS/DIV. 100 mV

SEC./DIV. 500 nS



PERFORMED BY: [signature], LEVEL: II DATE: 3-19-85

REVIEWED BY: ____________, LEVEL: ______ DATE: ______

nes

ULTRASONIC INSTRUMENT LINEARITY RECORD

ULTRASONIC INSTRUMENT

MODEL NO. S-80 SERIAL NO. 10244-9
PR 35 #11528-9

CALIBRATION BLOCK

TYPE Mini Pump SERIAL NO. 6060-83

TRANSDUCER

BRAND KB Aerotech FREQUENCY 2.25
BRAND ________ FREQUENCY ________
B20939

SIZE 1" STRAIGHT BEAM (✓)
SIZE ________ ANGLE BEAM ( )

HORIZONTAL LINEARITY

BACK REFLECTOR	GRID LOC.	ACCEPT LIMITS
1	1	1
2	2.0	1.90-2.10
3	3.0	2.85-3.15
4	4.0	3.80-4.20
5	5.0	4.75-5.25
6	6.0	5.70-6.30
7	7.0	6.65-7.35
8	8.0	7.60-8.40
9	9.0	8.55-9.45
10	10	10

VERTICAL LINEARITY

SIGNAL AMPLITUDES IN % FSH

NO.	ACTUAL HIGHER SIGNAL	(CALCULATE) 1/2 OF HIGHER	ACCEPT. LIMITS*	ACTUAL LOWER SIGNAL
1	100	(50)	(45)-(55)	50
2	80	(40)	(35)-(45)	40
3	62	(31)	(26)-(36)	31
4	50	(25)	(20)-(30)	25
5	40	(20)	(15)-(25)	20
6	30	(15)	(10)-(20)	15
7	24	(12)	(7)-(17)	12
8	20	(10)	(5)-(15)	10
9	16	(8)	(3)-(13)	8
10	12	(6)	(1)-(11)	6

*ACCEPTANCE LIMITS ARE 1/2 OF THE HIGHER SIGNAL ± 5% FSH

AMPLITUDE CONTROL LINEARITY

INITIAL AMPLITUDE	db CHANGE	RESULT	LIMIT
80% FSH	DOWN 6	40	32% - 48%
80% FSH	DOWN 12	20	16% - 24%
40% FSH	UP 6	80	64% - 96%
20% FSH	UP 12	80	64% - 96%

THIS INSTRUMENT IS CONSIDERED: (✓) ACCEPTABLE
( ) NOT ACCEPTABLE

SIGNED David [illegible] LEVEL II DATE 3-19-85

NUCLEAR ENERGY SERVICES, INC.

DAILY LINEARITY SHEET

Plant/Unit YANKEE ROWE (INFO)

Page ____ of ____

Channel 1

INSTR. LINEARITY RESPONSE

	Amplitude High	Low		High	Low
1	100	50	5	40	20
2	80	40	6	30	15
3	62	31	7	24	12
4	50	25	8	20	10

AMPL. CONTROL LINEARITY

Init.	db	Result	Limits
80	-6	40	32-48%
80	-12	20	16-24%
40	+6	80	64-96%
20	+12	80	64-96%

Mfg/Model No. AI/S80
Serial No. 10244-9
Date 3-21-85
Time 0800
Examiner David W Senenz
Level II

Channel 1

INSTR. LINEARITY RESPONSE

	Amplitude High	Low		High	Low
1	100	50	5	40	21
2	80	40	6	32	16
3	62	31	7	26	13
4	50	27	8	20	10

AMPL. CONTROL LINEARITY

Init.	db	Result	Limits
80	-6	39	32-48%
80	-12	20	16-24%
40	+6	80	64-96%
20	+12	80	64-96%

Mfg/Model No. AI/S80
Serial No. 10244-9
Date 4/3/85
Time 13 20
Examiner David W Senenz
Level II

Channel 1

INSTR. LINEARITY RESPONSE

	Amplitude High	Low		High	Low
1	100	52	5	40	20
2	80	40	6	30	16
3	70	35	7	24	12
4	54	27	8	20	10

AMPL. CONTROL LINEARITY

Init.	db	Result	Limits
80	-6	40	32-48%
80	-12	20	16-24%
40	+6	90	64-96%
20	+12	90	64-96%

Mfg/Model No. AI S80
Serial No. 10244-9
Date 4/9/85
Time 11:00
Examiner [illegible]
Level II

Channel ____

INSTR. LINEARITY RESPONSE

	Amplitude High	Low		High	Low
1			5		
2	80	40	6		
3			7		
4			8		

AMPL. CONTROL LINEARITY

Init.	db	Result	Limits
80	-6		32-48%
80	-12		16-24%
40	+6		64-96%
20	+12		64-96%

Mfg/Model No. ____
Serial No. ____
Date ____
Time ____
Examiner ____
Level ____

Reviewer ____ Date ____

FORM 028 103/1 2/85

NES NUCLEAR ENERGY SERVICES, INC.

CALIBRATION DATA SHEET SEE NOTE ON ATTACHED SHEET RJ

Cal. Blk. No. 80D8842-1 Page ______ of ______

Surface Temp. NA °F Cal. Data Pkg. NA

Block Temp. NA °F Procedure No 83A0314 Rev. ______

RPP 1832 CH #1 C-1 TRANSDUCER 60°L CIRC

Instrument		Hole Ident	Depth In.	Sweep Div.	Amp% (FSH)
Mfg.: AI	Model: S80	C-1	2.27	1.5	80
Serial No.: 10244-9	Channel: 1	C-2	4.27	2.4	42
Nom. Angle: 60	Mode: LONG	C-3	6.27	3.1	28
Transducer		C-4	10.26	4.3	15
Mfg.: NES	Model: NA	C-5	14.51	6.0	12

		Calibration	Date	Time	By
Serial No.: NA	Size: 1X1				
Freq.: 2.25 MHz	Measured Angle: 60°L	Initial	4/9/85	12 00	DWS
Cable Type: RG174/U	Length: 300' + 75''	Intermediate	4/9/85	16 30	DWS
Head No.: 80D8120-1	Couplant: H2O	Intermediate	4/11/85	11 00	DWS
Main Sweep Range: 20 in.		Intermediate			
Delay/Setup Range: 25 sec.		Intermediate			
Main Sweep Calib.: 8.48		Intermediate			
Delay: 0.0	Sync: DELAY	Final	4/11/85	15 00	DWS

Rep Rate Range: 1.2	KHz	Var: CAL
Display: 1	Pulser Sync: ALT	
Fltr: 2	Mode: FW	
Q: 4	Freq.: 2	
Jack: T		
Ref.:	Min.:	Max.
Pulse Length:	Min.:	Max.

Primary Reference Sensitivity 53 FOR 60 CIRC db

Equalized Response: 60° CIRC 30 db db

Exam Sensitivity: 27 db - 3 - 24 db

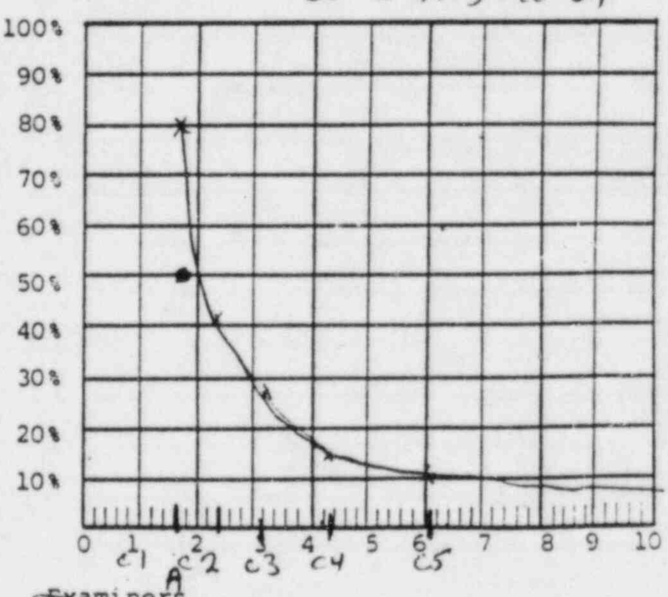
Screen Gated: NA sd to NA sd

Front Interface Signal Position: NA sd

1 Major Screen Division: NA in.

TUC Card No.:	Pulses:
Trigger: + -	Freq.: A B C
Pulse Period: C	F
Pulse Delay: C	F
Attenuation: C	F
Resistance Added:	

60°L CIRC 60°L AX 50% FSH



Examiners

1 [signature] Level II Date 4/11/85

2 [signature] Level II Date 4/11/85

3 ____________ Level ____ Date ______

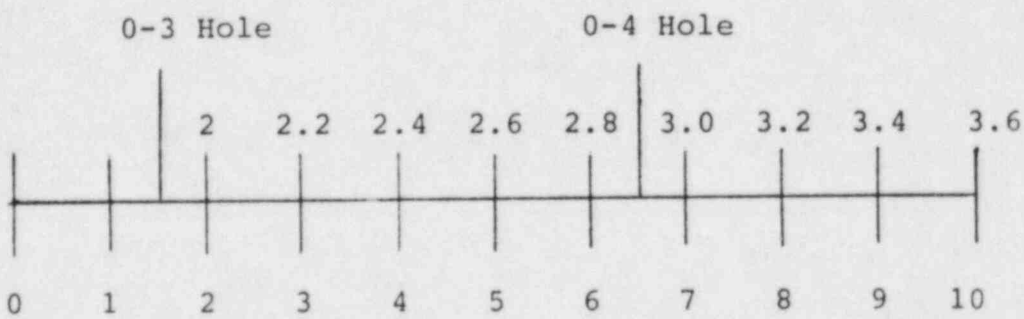
Reviewer ____________ Date ______

nes NUCLEAR ENERGY SERVICES, INC.

FORM 028-102/1 11/82

NOTE: After calibration documented on calibration data sheet, sweep was adjusted from long side of cal block no. 80D8842-1 on holes 0-3 and 0-4, for more accurate readings, as follows:

DEPTH IN MATERIAL - INCHES



HORIZONTAL SCREEN DIVISIONS

Sensitivity was increased 2dB to a setting of 22dB to compensate for curved vs. flat entry, as measured by the difference in amplitude between Hole A and Hole 0-3 (from the long side of block 80D8842-1).

Level II 4/11/85

nes

ULTRASONIC INSTRUMENT LINEARITY RECORD

ULTRASONIC INSTRUMENT

CALIBRATION BLOCK

MODEL NO. 580 SERIAL NO. 10244-9 TYPE Linearity SERIAL NO. 85C653

TRANSDUCER

BRAND Aerotech - Gamma HP FREQUENCY 2.25 MHz SIZE 1" ø STRAIGHT BEAM (✓)
BRAND N/A FREQUENCY N/A SIZE N/A ANGLE BEAM ( )

VERTICAL LINEARITY

SIGNAL AMPLITUDES IN % FSH

NO.	ACTUAL HIGHER SIGNAL	(CALCULATE) 1/2 OF HIGHER	ACCEPT. LIMITS*	ACTUAL LOWER SIGNAL
1	100	(50)	(45)-(55)	49
2	90	(45)	(40)-(50)	47
3	80	(40)	(35)-(45)	40
4	70	(35)	(30)-(40)	36
5	60	(30)	(25)-(35)	30
6	50	(25)	(20)-(30)	26
7	40	(20)	(15)-(25)	20
8	30	(15)	(10)-(20)	16
9	20	(10)	(5)-(15)	10
10	10	(5)	(0)-(10)	6

HORIZONTAL LINEARITY

BACK REFLECTOR	GRID LOC.	ACCEPT LIMITS
1	1	1
2	2.0	1.90-2.10
3	3.0	2.85-3.15
4	4.0	3.80-4.20
5	5.0	4.75-5.25
6	6.0	5.70-6.30
7	7.0	6.65-7.35
8	8.0	7.60-8.40
9	9.0	8.55-9.45
10	10	10

*ACCEPTANCE LIMITS ARE 1/2 OF THE HIGHER SIGNAL ± 5% FSH

AMPLITUDE CONTROL LINEARITY

INITIAL AMPLITUDE	db CHANGE	RESULT	LIMIT
80% FSH	DOWN 6	39	32% - 48%
80% FSH	DOWN 12	20	16% - 24%
40% FSH	UP 6	82	64% - 96%
20% FSH	UP 12	80	64% - 96%

THIS INSTRUMENT IS CONSIDERED: (✓) ACCEPTABLE
( ) NOT ACCEPTABLE

SIGNED Zane D. Dorgats LEVEL II DATE 17 APR 85

NUCLEAR ENERGY SERVICES, INC.

RF WAVEFORM RECORD SHEET

CHECK ONE: ☐ PRE-EXAM FIELD WAVEFORM, SITE:

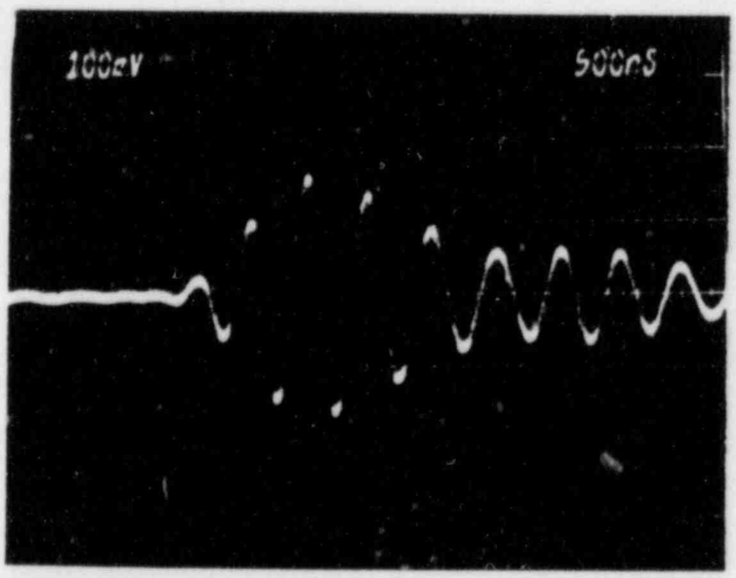
☑ POST-EXAM ~~FIELD~~ WAVEFORM, SITE: N/A

Z 17Apr85

TRANSDUCER	PULSER RECEIVER	REMOTE PULSER PREAMP:
S/N: C-4	S/N: 11528-9	S/N: 1832 Ch 1
MFGR.: NES	MODEL NO.: PR35	PE/THRU: Thru
MFGR. DESIGNATION: None	**PULSER RECEIVER SETTINGS**	**ULTRASONIC INSTRUMENT**
ELEMENT DIMENSIONS: 1" x 1" dual	ATTEN. db: 48	S/N: 10244-9
CABLE TYPE: RG 174/U	REF. (Fine Gain): Max	MFGR.: AI
CABLE LENGTH: 300' + 75'	FILTER: 2	STYLE: 580
	MODE: FW	90-DAY LINEARITY DUE DATE: 16 July 85
	FREG.: 2.25 MHz	**REFERENCE BLOCK**
FIXTURE OR HEAD	Q: 4	S/N: 80D8842-1
S/N: 80D8120-1	PULSE LENGTH: Max	TYPE: Calibration
ANGLE: 60° L	JACK (T or R): T	MATERIAL: Carbon Steel
	TEST (Thru or NORMAL): Thru	REFLECTOR USED: "A"

VOLTS/DIV. 100 MV

SEC./DIV. 500 nS



PERFORMED BY: Zane D. Dargatz LEVEL: II DATE: 17 Apr 85

REVIEWED BY: LEVEL: DATE:

ADDENDUM E

DATA FROM SPECIAL TEST BLOCK

R. Lucas Level II
4/9/85

PARALLEL SCAN OF SLOT

20° .4"

PARALLEL LOCATION OF THE FOLLOWING % DAC AND SWEEP POSITION (S.D.) AT ALL 20% + 50% POSITIONS

PERP. DIST. (0.2" INC)	PEAK AMP. % DAC OR DBIF ABOVE 100	20	S.D.	50	S.D.	100	PEAK	100	50	S.D.	20	S.D.
7.9			.2				3.95					
7.7		4.3	1.1								3.45	19
7.5		4.5	18								3.4	1.6
7.3		4.4	13	4.0	2.0		3.9		3.6	17		
7.1	50%	4.0	2.6		2.2		3.8				325	2.2
6.9			3.4		2.2		43					
6.7		4.0	3.5								3.5	37
6.5	60%	4.7	4.5	4.0	4.2		3.8		355	3.9	3.2	4.3
6.3	+3db 100%	485	5.1	4.6	5.1		3.8		34	4.7	3.0	4.7
6.1	+4db 100%	4.6	5.7	4.5	5.4		3.6		3.0	4.9	2.9	5.0
5.9	+3db 100%	4.7	6.2	4.5	6.2		3.7		3.0	5.2	2.9	5.6
5.7	+7db 100%	4.8	6.2	4.7	6.7		3.9		3.0	5.8	2.9	5.8
5.5	+5db 105%	4.8	7.1	4.6	6.9		3.9		3.1	6.5	2.9	6.3
5.3	+4db 100%	4.7	7.2	4.6	7.3		3.9		3.1	6.9	295	7.1
5.1	100%	4.8	8.0	4.3	79		385		3.2	7.5	3.0	
4.9	85%	4.6	8.5	4.4	8.7		425		3.8	8.2	3.8	
4.7	60%	4.4	7.2	3.9	8.5		3.8		3.7	8.5	3.2	8.2
4.5		4.5	9.6								43	9.5

Zane D. Dorgaty UTL II 9 APR 85

PARALLEL SCAN OF SLOT

10° .2"

PARALLEL LOCATION OF THE FOLLOWING % DAC AND SWEEP POSITION (S.D.)
AT ALL 20% + 50% POSITIONS

PERP. DIST. (0.2"/NC)	PEAK AMP. % DAC OR DBIF ABOVE 100		20	S.D.	50	S.D.	100	PEAK	100	50	S.D.	20	S.D.		
8.0			6.8	0.9											
7.8			7.0	1.3								6.5	1.2		
7.6			7.1	1.8								6.7	1.75		
7.4			7.2	2.3								6.7	2.2		
7.2												7.8	2.8		

Zane D. Dougats UTL II 9 APR 85

PARALLEL SCAN OF SLOT

$\boxed{20^\circ\ .2''}$

PARALLEL LOCATION OF THE FOLLOWING % DAC AND SWEEP POSITION (S.D.)
AT ALL 20% + 50% POSITIONS

PERP. DIST. (0.2"/NC)	PEAK AMP. % DAC OR DBIF ABOVE 100	20	S.D.	50	S.D.	100	PEAK	100	50	S.D.	20	S.D.
0.1		5.7	2.2									
0.3		5.9	2.7								5.4	2.7
0.5		6.0	3.2		3.2		5.7				5.4	3.2
0.7	75	5.9	3.6	5.8	3.6		5.6		5.4	3.6	5.25	3.6
0.9	100	6.0	4.1	5.9	4.1		5.5		5.2	4.2	4.45	4.2
1.1	+2dB	6.1	4.6	5.9	4.6	6.8	5.45	5.4	5.3	4.7	4.6	4.75
1.3	100	6.1	5.05	5.9	5.1		5.65		5.2	5.2	4.5	5.2
1.5	100	6.1	5.55	5.85	5.5		5.5		5.1	5.6	4.6	5.65
1.7	100	6.0	6.0	5.8	6.0		5.5		5.1	6.05	4.75	6.1
1.9	75	6.1	6.5	5.8	6.5		5.45		5.2	6.6	4.8	6.6
2.1		6.0	7.1		7.05		5.7				5.2	7.1
2.3	35	5.8	7.5				5.65				5.2	7.5
2.5		5.9	8.1								5.25	8.1
2.7		6.0	8.4								5.4	8.5
2.9											5.1	9.0

Zane D. Dargatz UTL II 11 APR 85

PARALLEL SCAN OF SLOT

30° .2"

PARALLEL LOCATION OF THE FOLLOWING % DAC AND SWEEP POSITION (S.D.) AT ALL 20% + 50% POSITIONS

PERP. DIST. (0.2"/NC)	PEAK AMP. % DAC OR DBIF ABOVE 100	20	S.D.	50	S.D.	100	PEAK	100	50	S.D.	20	S.D.
−1.0		1.2	0.0									
−0.8	64	2.1	0.3	1.6	0.3		1.3		1.1	0.3	0.9	0.5
−0.6	100	2.3	0.5	1.8	0.45		1.55		1.1	0.5	0.5	0.8
−0.4	+2dB	2.35	0.9	1.85	1.0	1.55	1.4	1.05	0.7	1.0	0.4	1.2
−0.2	+2dB	2.2	1.65	2.05	1.5	1.7	1.4	0.9	0.7	1.55	0.4	1.7
0	+4dB	2.1	2.15	2.0	2.0	1.55	1.2	0.9	0.6	2.1	0.4	2.2
0.2	+1dB	2.0	2.75	1.8	2.6	1.45	1.2	0.95	0.65	2.55	0.4	2.7
0.4	80	1.8	3.1	1.4	3.1		1.3		0.9	3.0	0.5	3.2
0.6		1.7	3.6	1.3	3.6						0.6	3.6
0.8		1.7	4.2	1.4	4.0						0.8	4.05
1.0		1.7	4.6	1.4	4.55						0.7	4.6
1.2	56	1.6	5.2	1.6	5.1		1.4		1.2	4.9	0.7	5.0
1.4		1.8	5.45	1.5	5.4						0.95	5.55
1.6		1.5	6.1								0.95	6.05
1.8		2.2	6.5								1.1	6.1

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