Jlc

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

July 12, 1979

Mr. J. G. Keppler, Director
Office of Inspection and Enforcement
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO IE BULLETIN 79-13
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On May 25, 1979, the Office of Nuclear Reactor Regulation informed all PWR licensees of the occurrence of cracking in welds in the feedwater nozzle-to-piping welds at the D. C. Cook Unit 2 facility and requested information concerning similar welds at other PWR plants. Our letter of June 20, 1979, provided the requested information and stated our intent to inspect the steam generator feedwater nozzle welds of both Point Beach units during the week of July 1, 1979. On June 25, 1979, we informed you by telegram of our intent to proceed with ultrasonic examination only of the feedwater line welds during hot shutdown of the Point Beach units beginning June 30, 1979. Subsequently, NRC issued IE Bulletin 79-13 on June 25, 1979, which required radiographic examinations of feedwater nozzle-to-piping welds and adjacent pipe and nozzle areas within 90 days of the date of the bulletin. Upon receipt of the bulletin and after identification of a packing leak in a Unit 2 ten-inch Residual Heat Removal System valve, it was decided to proceed with Unit 2 to cold shutdown on June 30 which then allowed repair of the RHR valve and provided the opportunity to perform radiographic and ultrasonic examinations of the feedwater line welds.

Eight Unit 2 feedwater piping welds have been inspected, including the weld connecting each three-inch auxiliary feedwater pipe to each main feedwater pipe. Linear indications and several small cracks were found during radiographic and ultrasonic examinations of the Unit 2 welds. These indications are described in our July 6, 1979, 24-hour notification letter. Representatives of Wisconsin Electric Power Company, Southwest Research Institute, and Bechtel Power Corporation met with NRR and IE personnel in Washington on July 6 to discuss the results of our inspections and the expected extent of the repairs. The enclosure to this letter summarizes information presented at our meeting and presents additional feedwater piping system information requested by NRC personnel during the meeting, including results of all testing and metallurgical examination undertaken to date.

593 274

JUL 13 1979

7908090416

As discussed in more detail in the enclosure, the Unit 2 feedline welds are being repaired. While completing these repairs, the feedwater piping in the inspected area is being reconstructed in a manner similar to the original construction. No significant changes are being made from the original construction requirements and we have determined that the repair work does not constitute an unreviewed safety question. A 10 CFR 50.59 review has been performed by the Manager's Supervisory Staff and the results of this review have been documented. Further, metallurgical laboratory examinations of the welds which were removed for replacement indicate that the welds did not present an unsafe condition and, had this knowledge been available earlier, the repairs would not have had to be made.

At the completion of these repairs, baseline radiographic and ultrasonic examinations of the affected welds will be performed. These examinations will be the reference for the next inspection of these welds which is tentatively scheduled for the Spring 1980 refueling of Unit 2.

It is anticipated that the repair effort will be completed by July 15 and the unit will be returned to power generation by July 17, 1979.

As we discussed at the July 6 meeting, we intend to perform the inspection of the Unit 1 feedwater piping welds during the scheduled Fall refueling outage for Unit 1. This outage is tentatively scheduled to begin on September 28, 1979.

It is our intention to continue to follow developments relating to feedwater line cracking and to review the experience of other operating PWR plants.

Very truly yours,

Sol Burstein

Sol Burstein

Executive Vice President

Enclosures

Copy to: Office of Inspection and Enforcement
Division of Reactor Operations Inspection

Mr. A. Schwencer, Chief
Office of Nuclear Reactor Regulation

593 275

ENCLOSURE

Response to
IE Bulletin 79-13
July 10, 1979

This attachment provides Wisconsin Electric Power Company's response for Point Beach Nuclear Plant Units 1 and 2 to NRC IE Bulletin No. 79-13 entitled "Cracking in Feedwater System Piping". This attachment contains analytical and operating information for both Point Beach Units and inspection results for Unit 2. Unit 1 inspection results will be provided at a later date.

In addition to responses to each of the bulletin items (in paraphrased form), additional information is also provided as requested by the NRC staff during our meeting of July 6, 1979. Also enclosed are simplified isometric sketches of each steam generator and adjacent feedwater piping; see Figures 1 thru 4 attached.

1. Examine feedwater nozzle-to-piping welds and piping supports.

RESPONSE

The Unit 2 welds identified on Figures 3 and 4 have been inspected and repaired.

Appendix A, attached, provides details of the radiographic (RT) and ultrasonic (UT) examinations on the main feedwater piping welds. The RT evaluation and UT examination and evaluation were performed by Southwest Research Institute personnel.

Appendix B, attached, provides a summary of indications found during other examinations, such as the steam generator feedwater nozzle I.D. surface, and the corrective actions that have occurred due to this inspection.

Appendix C, attached, provides an interim report of the preliminary metallurgical evaluation of indications found in the "A" loop reducer. The laboratory examinations were performed at the Southwest Research Institute.

Figures 1 thru 4 tabulate the design stress levels from the steam generator nozzle weld (data point 1) thru the check valve located closest to the steam generator. The original piping analysis did not consider the reducer; the stress results for data point 1 are for a 16 inch diameter pipe. Thus, the actual stresses are less than those tabulated. However, based upon the original piping analysis, there are no known weld locations in the feedwater piping where either the thermal stress or the summation of the seismic stresses exceed 90% of the code allowable stresses. Thus, no other welds have been examined.

The Unit 2 feedwater piping supports and restraints have been visually inspected for conformance to the design requirements as shown on Bechtel drawing M-2409, revision 7 which was provided to NRC by our letter of June 20, 1979. The supports appear to be in conformance with the original design requirements, but some additional verification of spring constants is still being pursued.

593 276

2. Perform additional piping weld inspections.

RESPONSE

The Unit 2 inspections will be performed during the Spring 1980 refueling. The Unit 1 inspections will be performed during the Fall 1979 refueling.

3. Effects of cracking indications on multiple nuclear unit facilities.

RESPONSE

As shown in Appendix C, by the metallurgical examination, the indications discovered during the inspection of Unit 2 are generally on the order of 0.043 inches deep on the inside surface of the weld. The difference between the piping mill nominal and minimum wall thickness is 0.117 inches for 18 inch, schedule 80 pipe (t_{min} is 0.820 inches) and 0.104 inches for 16 inch, schedule 80 pipe (t_{min} is 0.739 inches) per ASTM A106 -75a, Table A2. In addition, the analytical minimum wall thickness required to withstand the operating pressure of 1050 psi (per ANSI B31.1, paragraph 104.1.2, equation 3) is only 0.625 inches, including an 0.08 inch corrosion allowance.

The general indications discovered during the Unit 2 examination do not reduce the sound metal wall thickness below the mill minimum wall thickness or the analytically required minimum wall thickness and thus do not constitute a safety hazard.

At full power with all feedwater heaters in use, the feedwater temperature entering containment is about 425°F. In the event of a unit trip, inlet water temperatures may drop to about 80°F. The resultant pipe metal temperatures would still be above the range of carbon steel brittle-to-ductile transition temperatures. Thus, the feedwater pipe system is not believed to operate in a non-ductile temperature environment.

During the July 6, 1979 meeting with NRC the Unit 2 inspection results were discussed, with the laboratory investigations showing that the Point Beach Nuclear Plant indications were not serious, and do not have any apparent safety significance. On this basis, it was generally agreed that the Point Beach indications were not serious. The Unit 1 annual Fall refueling outage is tentatively scheduled for only a few days beyond the expiration of the 90 day inspection interval specified in IE Bulletin 79-13, and, as discussed at the meeting, the Unit 1 inspections will be performed during that refueling outage.

4. Report indications to NRC within 24 hours of identification.

RESPONSE

The existence of linear indications and cracks in the Unit 2 welds (as discussed in Appendices A and B) was positively identified late on July 5 by the Southwest Research Institute metallurgical laboratory. NRC was advised of this during our meeting of July 6 and a written 24-hour notification was sent to NRC Region III on July 6.

593 271

5. Provide a report identifying the inspection schedule, adequacy of applicable operating and emergency procedures, and ability to detect feedwater leaks inside containment.

RESPONSE

The intiial inspections required by Item 1 of the Bulletin have been completed for Unit 2. Comparable Unit 1 inspections will be performed during the Fall 1979 refueling outage.

The operating and emergency procedures in use at the Point Beach Nuclear Plant are adequate to recognize and respond to a feedwater line break accident. The indications and symptoms of the accident are dependent upon the location and size of the break. A feedwater line break upstream of the final check valve at the steam generator feedwater nozzle would result in a reactor trip. The reactor trip would probably be caused by a steam flow/feed flow mismatch coincident with a low water level in either steam generator or by low-low water level in either steam generator. Upon the trip of the reactor, the plant operators would follow emergency operating procedure EOP-5A for an emergency shutdown of the reactor.

A smaller break upstream of the check valve could be of insufficient magnitude to cause a sustained steam flow/feed flow mismatch. In such an eventuality, the reactor would be tripped on a two out of three low-low water level signal in either steam generator. The low-low level signal in either steam generator would also automatically start the motor driven feedwater pumps. As discussed in Section 14.1.11 of the Final Facility Description and Safety Analysis Report (FFDSAR), the loss of normal feedwater flow caused by a feedwater line break does not result in any adverse conditions.

In the event of a feedwater line break in the short run of piping between the final check valve and the feedwater nozzle of the steam generator, the same reactor trip functions would occur and the plant operators would be presented with symptoms and indications essentially identical to those of the rupture of a steam pipe. Plant EOP-2A, "Steam Line Break", addresses the operator action in the event of such a rupture.

Under certain circumstances, such as a feedwater pipe crack, the feedwater leakage may not be of sufficient magnitude to manifest itself in a reactor trip. In such a situation there are several excellent leakage detection methods to reveal the presence of significant leakage levels in containment. The humidity detection instrumentation offers one means of detecting low level leakage. Plots of containment air dew point variations above a base-line maximum, established by cooling water temperature to the containment air coolers, can determine incremental leakage equivalent to 2 to 10 gpm. The sensitivity of this method is dependent upon the cooling water temperature, containment air temperature variation, and containment air recirculation rate.

A second leak detection method is based on the principle that the condensate collected by the containment cooling coil matches, under equilibrium conditions, the leakage of water and steam from systems within containment. Measurement of the condensate drained from each of the fan cooler units can

503 278

be made to determine condensate rate and thus leak rate. Should a le... occur, the condensation rate will increase above the previous steady state due to the increased vapor content of the air. This condensate from the fan coolers is piped to sump A in both containments. A high liquid level in the sump is alarmed in the control room. It takes approximately 22 gallons to actuate the Unit 1 alarm and about 42 gallons for an alarm in Unit 2. Since sump contents removal requires manual operation, the operators would be alerted to an increase in total containment leakage rate. As with the containment humidity indications, if an increased total containment condensation rate is apparent, a containment entry and inspection would be made to determine the source of the leakage.

6. Submit a written report to NRC presenting the pipe weld inspection results.

RESPONSE

This submittal constitutes Wisconsin Electric's response to this IE Bulletin with respect to Item 1 for Point Beach Nuclear Plant Unit 2. When the metallurgical examination report for Unit 2 is completed it will be forwarded to NRC. This is tentatively scheduled for mid-August 1979. The Bulletin Item 2 results for Unit 2 will be provided after the Spring 1980 refueling.

The report for the Unit 1 inspections for both Items 1 and 2 of the bulletin will be provided within 30 days of the Unit 1 inspection. This is expected to be in early November 1979, following inspections during the Fall refueling.

The following are additional items discussed at the July 6, 1979 meeting with NRC staff.

7. Steam Generator Feedwater Chemistry

The Point Beach Nuclear Plant steam generator feedwater chemistry was originally based upon "phosphate control". However, the units were converted to the "all volatile treatment" (AVT) method in September 1974 for Unit 1 and in November 1974 for Unit 2. In Wisconsin Electric's letter of August 18, 1978 (Steam Generator Operating History Questionnaire) to Mr. K. R. Goller of the NRC, the feedwater chemistry specification for AVT control is presented. Table 1 attached hereto is the same as the table attached to the August 18, 1978 letter. The feedwater chemistry is maintained in accordance with specifications for AVT control.

8. Feedwater System Transients

Significant operating events that would have caused thermal transients to the feedwater piping system are listed in Tables 2 and 3. These tables have been developed from information contained in the Operations Section of the Monthly Semi-annual and Annual Operating Reports which are periodically submitted to the NRC.

593 279

AVT CONTROL, SECONDARY CHEMISTRY SPECIFICATIONS

STEAM GENERATOR BLOWDOWN	NORMAL POWER OPERATION	LIMITED NORMAL POWER OPERATION*	<15% POWER OPERATION	WET LAYUP
pH	8.5 - 9.0	8.5 - 9.2	8.0 - 10.0	10.0 - 10.5
Cation Conductivity, µmhos/cm	<2.0	<7.0	<7.0	------
Ammonia, ppm	<0.25	------	------	------
Sodium, ppm	<0.10	------	<0.5	------
Chloride, ppm	<0.15	------	<0.5	<0.5
Silica, ppm	<1.0	------	<5.0	------
Free Hydroxide, ppm as $CaCO_3$	<0.15	0.15 - 1.0**	<0.15	------
Suspended Solids, ppm	<1.0	------	<1.0	------
Blowdown Rate, gpm	Continuous as Required	Maximum Available	------	------
Oxygen, ppb	------	------	<5.0	<100
Hydrazine, ppm	------	------	------	75 - 150

FEEDWATER				
pH	8.8 - 9.2	Same As Normal Power Operation	Same as Normal Power Operation	As Required By Situation
Total Conductivity, µmhos,cm	<4.0			
Oxygen, ppb	<5.0			
Excess Hydrazine, ppb	5.0			
Copper, ppb	<5.0			
Iron, ppb	<10			

CONDENSATE				
Cation Conductivity, µmhos/cm	<0.2			

POOR ORIGINAL

* Period of operation within these limits should not exceed two weeks

** Period of operation within these limits should not exceed twenty-four hours.

593 280

TABLE 1

TABLE 2

LIST OF SIGNIFICANT OPERATING EVENTS FOR FEEDWATER PIPING SYSTEM TRANSIENTS
(Based upon Operating Reports)

POINT BEACH NUCLEAR PLANT UNIT 1

DATE	EVENT
7/70	Hot functional test completed.
11/2/70	Initial criticality.
11/22/70	Turbine trip from 40% power.
11/29/70	Power escalation to 70% and unit trip.
12/4/70	Turbine and reactor trip from 425 MWe.
12/5/70	Turbine trip from 70% power.
12/6/70	Unit trip from 70% power.
12/18/70	Unit trip from 80% power.
1/4/71	Auxiliary feedwater injection.
1/8/71	Turbine trip from 82% power.
1/9/71	Turbine trip from 70% power.
1/27/71	Unit trip from 90% power.
1/28/71	Turbine trip from 92% power.
2/3/71	Turbine trip from 90% power.
2/4/71	Unit trip from 90% power plus a turbine trip.
2/9/71	Unit trip from 80% power.
2/26/71	Load runback from 450 MWe to 200 MWe.
5/29/71	Reactor trip from no-load.
7/2/71	Reactor and turbine trip.
8/29/71	Reactor and turbine trip.

593 281

DATE	EVENT
9/7/71	Load runback from 425 MWe to 260 MWe.
9/18/71	Load runback from 480 MWe to 390 MWe.
12/3/71	Reactor and turbine trip.
1/3/72	Reactor and turbine trip.
1/19/72	100% load rejection test/reactor and turbine trip.
2/12/72	Reactor trip during startup.
2/13/72	Load runback of 20%.
4/13/72	Load runback of 20%.
4/21/72	Reactor and turbine trip.
7/3/72	20% step increase in power.
9/11/72	Turbine and reactor trip from 99% power.
7/2/73	Reactor and turbine trip.
8/13/73	Reactor and turbine trip.
1/11/74	Reactor and turbine trip.
1/18/74	Reactor and turbine trip.
2/3/74	Reactor and turbine trip.
8/2/74	Reactor and turbine trip.
9/25/74	Reactor trip from 99% power.
10/4/74	Reactor and turbine trip.
2/27/75	Emergency shutdown with reactor and turbine trip.
11/16/75	Emergency shutdown with reactor and turbine trip.

DATE	EVENT
1/10/76	Load runback of 20%.
1/14/76	Reactor trip.
11/30/76	Load runback of 20% from 90% power.
2/21/77	Reactor and turbine trip.
4/5/77	Reactor and turbine trip.
1/7/78	Reactor and turbine trip.
2/9/78	Reactor and turbine trip.
4/2/78	Reactor and turbine trip.

TABLE 3

LIST OF SIGNIFICANT OPERATING EVENTS FOR FEEDWATER PIPING SYSTEM TRANSIENTS
(Based upon Operating Reports)

POINT BEACH NUCLEAR PLANT UNIT 2

DATE	EVENT
12/22/71	Normal plant heatup.
5/30/72	Initial criticality.
8/4/72	Turbine trip from 20% power.
8/18/72	Unit trip from 10% power.
12/7/72	Turbine trip from 20% power plus a turbine and reactor trip from 10% power.
1/14/73	Reactor and turbine trip.
2/18/73	Reactor and turbine trip.
3/8/73	Received 100% power license.
3/9/73	Reactor and turbine trip.
3/10/73	Reactor and turbine trip.
3/14/73	Reactor and turbine trips - 2.
3/24/73	Reactor and turbine trip.
3/26/73	Reactor and turbine trip.
3/30/73	Reactor and turbine trips - 2.
4/8/73	Reactor and turbine trip.
5/30/73	Reactor and turbine trip.
6/19/73	Reactor and turbine trip.
10/13/73	20% load runback.
12/15/73	Reactor and turbine trip.

593 284

TABLE 3 UNIT 2 - CONT'D

DATE	EVENT
7/2/74	Reactor trip from 0% power.
12/27/74	Reactor and turbine trip.
2/11/75	Reactor and turbine trip.
2/24/75	Reactor trip from 10% power.
8/19/75	Reactor trip from about 10% power.
1/14/76	Manual unit trip.
2/21/76	Load runback of 20%.
4/8/76	Unit trip.
5/7/76	Load runback of 20%.
6/13/76	Load runback of 20% from 100% power.
9/3/76	Reactor trip.
1/12/77	Reactor and turbine trip.
6/28/77	Reactor and turbine trip.
7/7/77	Reactor and turbine trip.
1/10/78	Reactor and turbine trip.

CALCULATION SHEET

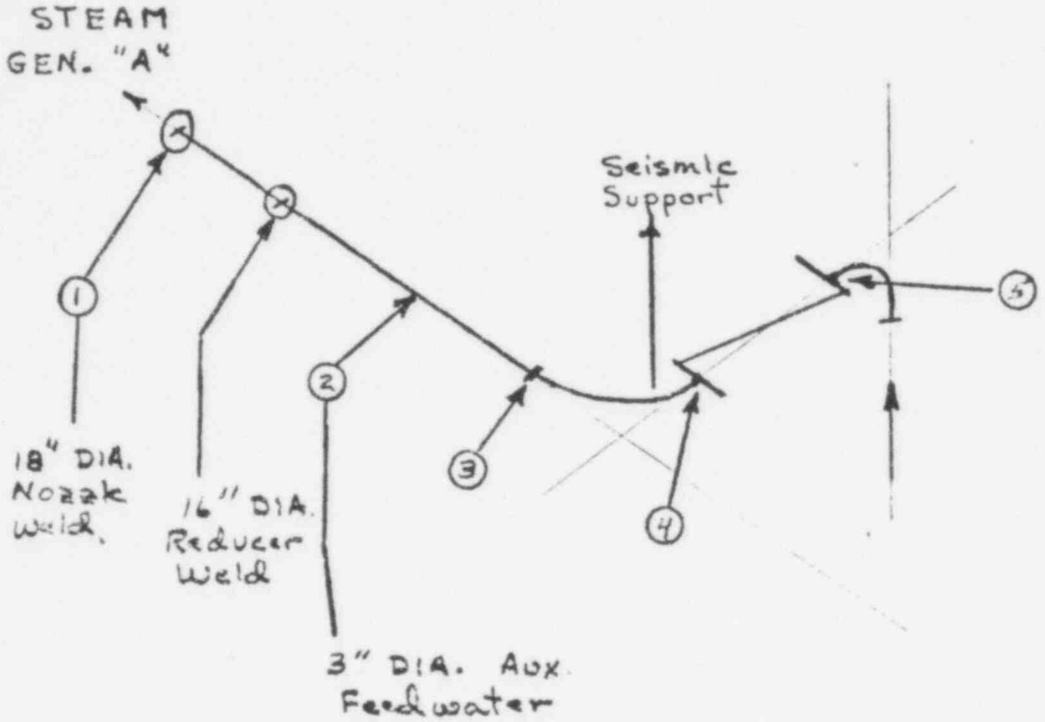
SHEET ______ OF ______

FILE No. ______

SUBJECT POINT BEACH NUCLEAR PLANT - Unit 1, Loop "A"

MADE BY DLD DATE 7/9/79

CHKD. BY ______ DATE ______



Data Point	Item	Thermal Stress, psi	Summation of OBE Stresses, psi	Summation of SSE Stresses, psi
1*	St. Gen. Nozzle to reducer weld	3,621	12,534	18,460
2	Aux. FW Conn.	—	12,280	17,952
3	Pipe to elbow	3,728	15,434	24,260
4	Elbow to check valve	5,706	11,232	15,856
5	Check valve to elbow	—	14,011	21,414
—	Allowable stress, psi	22,500	18,000	27,000

* - See text.

593 206

POOR ORIGINAL

FIGURE 1

SP 12041A

CALCULATION SHEET

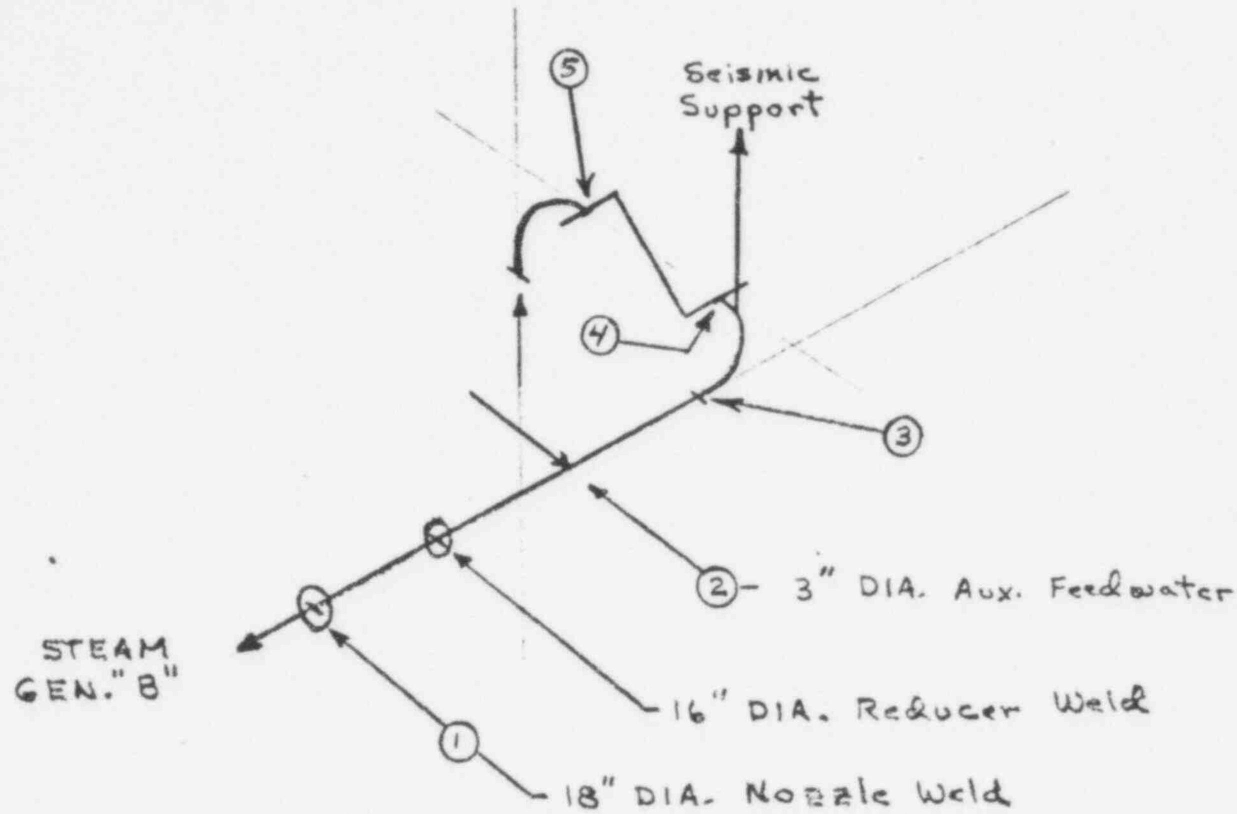
SHEET ________ OF ________

FILE No. ________

SUBJECT POINT BEACH NUCLEAR PLANT - Unit 1, Loop "B"

MADE BY DLDiaz DATE 7/9/79

CHKD. BY ________ DATE ________



Data Point	Item	Thermal Stress, psi	Summation of OBE Stresses, psi	Summation of SSE Stresses, psi
1*	St. Gen. Nozzle to reducer weld	1,730	7,955	9,302
2	Aux. FW Conn.	—	11,072	15,536
3	Pipe to elbow	12,860	7,832	9,056
4.	Elbow to check valve	—	8,130	9,652
5	Check valve to elbow	516	7,736	8,864
			593	~~285~~ 287
—	Allowable Stress, psi	22,500	18,000	27,000

* - See text.

FIGURE 2

CALCULATION SHEET

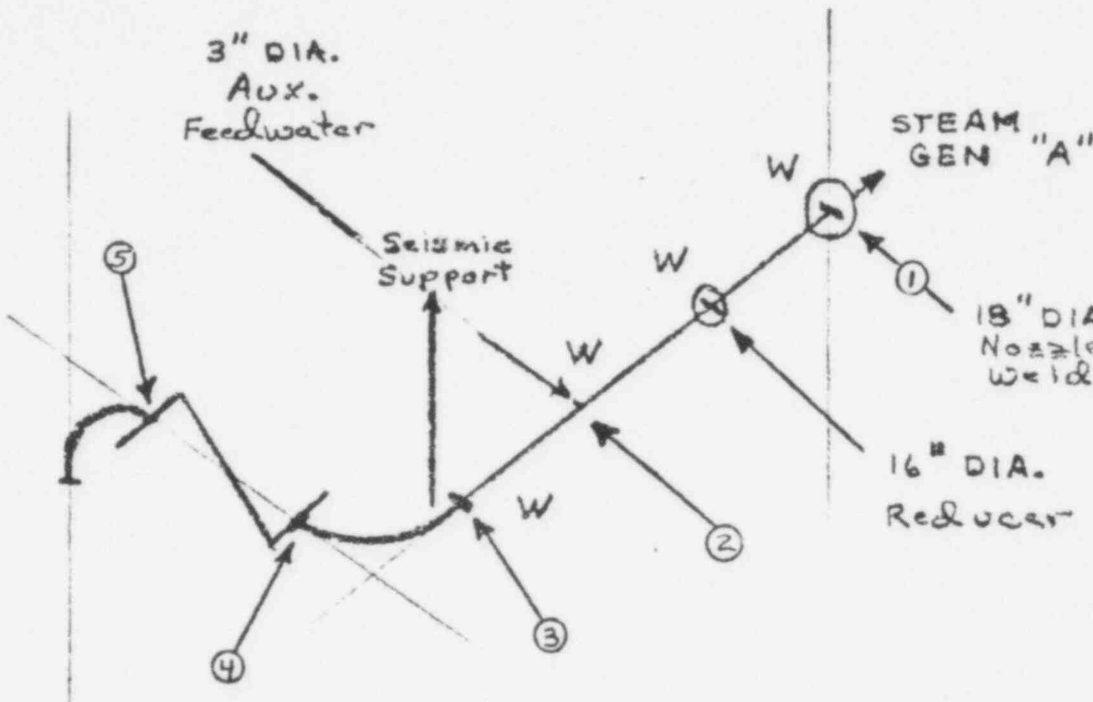
SHEET ______ OF ______

FILE No. ______

SUBJECT POINT BEACH NUCLEAR PLANT - Unit 2, Loop "A"

MADE BY DLD DATE 7/9/79

CHKD. BY ______ DATE ______



Data Point	Item	Thermal Stress, psi	Summation of OBE stresses, psi	Summation of SSE Stresses, psi
1*	St. Gen. Nozzle To Reducer Weld	7,925	9,689	12,770
2	Aux. FW Conn.	—	8,607	10,606
3	Pipe to elbow	12,860	11,144	15,680
4	Elbow to check Valve	—	8,020	9,432
5	Check valve to elbow	516	10,256	13,904
—	Allowable Stress, psi	22,500	18,000	27,000

* - See text.

FIGURE 3

593 280

POOR ORIGINAL

SP 12041A

CALCULATION SHEET

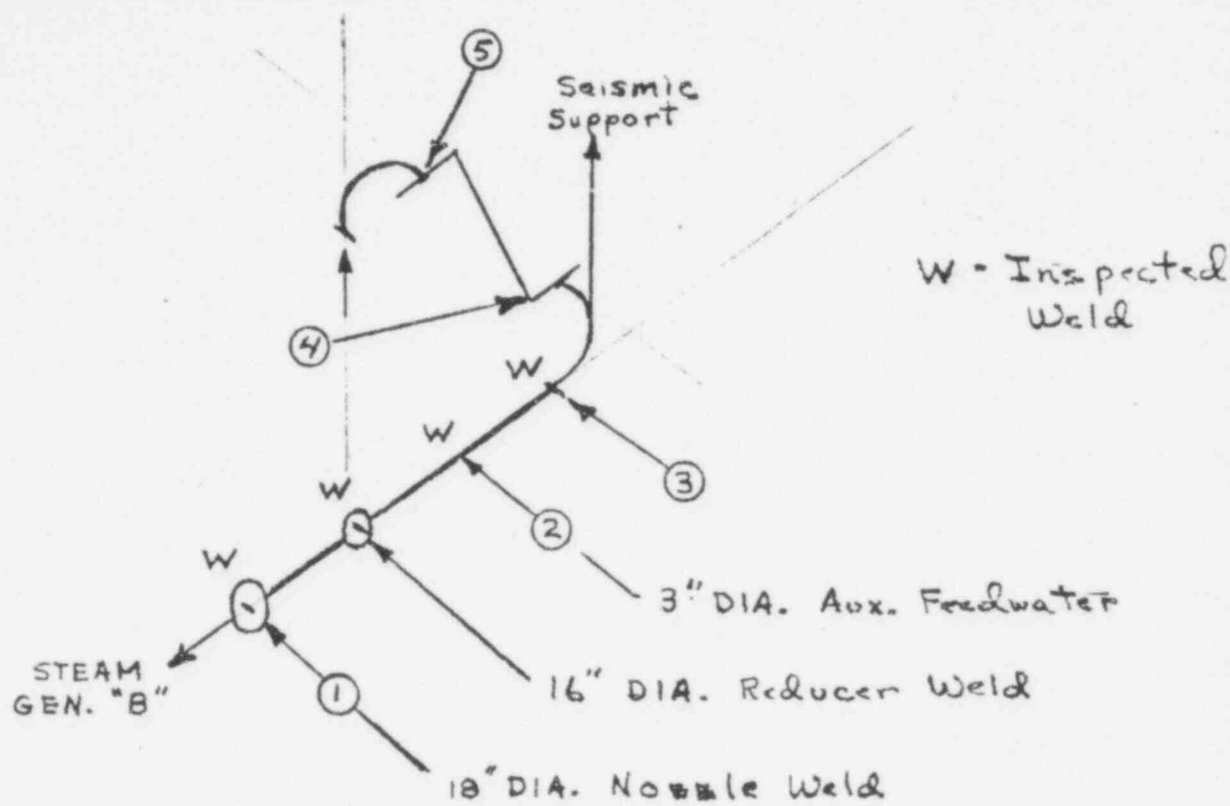
SHEET ______ OF ______

FILE No. ______

SUBJECT POINT BEACH NUCLEAR PLANT - Unit 2, Loop "B"

MADE BY DLD DATE 7/4/79

CHKD. BY ______ DATE ______



Data Point	Item	Thermal Stress, psi	Summation of OBE Stresses, psi	Summation of SSE Stresses, psi
1*	St. Gen. Nozzle To Reducer Weld	4,613	8,065	9,522
2	Aux. FW Conn.	5,127	7,736	8,864
3	Pipe to elbow	11,331	7,985	9,362
4	Elbow to check valve	971	8,639	10,670
5	Check valve to elbow	7,433	8,231	9,854
-	Allowable stress, psi	22,500	18,000	27,000

* - See text.

593 289

POOR ORIGINAL

FIGURE 4

APPENDIX A

Wisconsin Electric Power Company
Point Beach Nuclear Plant

SUMMARY OF RADIOGRAPHIC AND ULTRASONIC EXAMINATIONS FOR UNIT 2 MAIN FEEDWATER PIPING WELDS

July 1979

593 290

RADIOGRAPHIC REVIEW

radiographic testing

On July 2, 1979, S. A. Wenk, SwRI Level III (RT) reviewed the following radiographs at the Point Beach Unit 2 Generating Station. The radiography was performed by personnel from the Superior Industrial X-Ray Corp., and the sensitivity was in compliance with the requirements of NRC IE Bulletin 79-13.1a. Radiographic technique details are shown on the Superior Industrial X-Ray review form, a copy of which is contained in each weld film packet.

The results of the July 2, 1979, review are tabulated below:

Steam Generator A - 16-Inch Reducer-to-Pipe Weld

1. At radiographic station marker #32, a 1/2-inch long linear indication, 1/2 inch from the edge of the center bead on the pipe side of the weld.

2. At radiographic station marker #38, a 5/8-inch long linear indication on pipe side.

3. At radiographic station marker #40, a 1/2-inch long linear indication on pipe side.

Steam Generator A - 18-Inch Nozzle-to-Reducer Weld

1. Starting at radiographic station marker #10 and extending to station marker #16 a continuous linear indication which is crack-like in appearance.

2. Between radiographic station markers #42 and #43, there are two transverse linear indications each 1/4-inch long extending from the weld crown on the pipe side of the weld into the base metal.

Steam Generator B - 16-Inch Reducer-to-Pipe Weld

1. Linear indication between radiographic station markers #2 and #8, 3/4 inch from root bead centerline on the pipe side.

2. Linear indication between station markers #7 and #10, 3/4 inch from root bead centerline on the pipe side.

3. Linear indication between station markers #12 and #14, 3/4 inch from root bead centerline on the pipe side.

4. Linear indication between station markers #14 and #23, 5/8 inch from root bead centerline on the pipe side, very pronounced between station markers #19 and #23.

5. Linear indication between station markers #46 and #49, 5/8 - 3/4 inch from root bead centerline on pipe side.

Steam Generator B - 18-Inch Nozzle-to-Reducer Weld

1. Linear indication between radiographic station markers #10 and #12 at the edge of the cap weld 1/2 inch from the centerline of the weld bead closest to the reducer.

2. Linear indication between station markers #22 and #24 in the same area as described in (1) above.

3. Cluster of porosity at station marker #27, 3/8-inch long but separated.

4. Linear indication between station markers #30 and #32 in the same area as described in (1) above.

5. Linear indication between station markers #34 and #38 at the base of the bead on the reducer side.

6. Linear indication between station markers #41 and #43, 3/8 inch from centerline of the bead described in (1) above.

7. Linear indication at station marker #42 + 3/4 inch extending into the reducer transverse to the weld, the indication is 1/2-inch long.

8. Suck-back or concavity of root bead between station markers #43 and #44.

9. Linear indication between station markers #49 and #53 in the same area described in (1) above.

10. Transverse linear indication at station marker #56 + 1/2 inch. Due to nature of image, visual examination is recommended.

S. A. Wenk
SwRI Level III - RT

593 292

RADIOGRAPHIC REVIEW

On July 4, 1979, S. A. Wenk, SwRI Level III RT, reviewed the following radiographs at the Point Beach Unit 2 Generating Station. The radiography was performed by personnel from the Superior Industrial X-Ray Corp., and the sensitivity was in compliance with the requirements of NRC IE bulletin 79-13.1a. Radiographic technique details are shown on the Superior Industrial X-Ray review form, a copy of which is contained in each weld film packet.

Weld 1B-16 A Side Pipe-to-90° Elbow

1. Starting at radiographic station marker 0 and continuing through station marker #13 is a linear indication 7/16 inch from centerline of middle bead on the pipe side.

2. Linear indication between station markers #6 and #8, 7/16 inch from center line of middle bead on the elbow side.

3. Linear indication described in (1) above continuing from station marker #13 through station marker #26.

4. Linear indication between station markers #16 and #17 in the center of the middle bead.

5. Linear indication at station marker #37, 3/8 inch from center of middle bead on pipe side.

6. Linear indication described in (1) above between station marker #44 and station marker 0.

Weld 1B-16 B Side Pipe-to-90° Elbow

1. Linear indication between station markers #7 and #13, 1/2 inch from centerline of middle bead on the pipe side.

2. Linear indication described in (1) above for this weld continuing between station markers #13 and #26.

3. Crater at station marker #30, 5/8 inch from centerline of middle bead on elbow side.

4. Linear indication between station markers #28 and #39 on elbow side of middle bead.

5. Linear indication between station markers #28 and #39, a continuation of the indication described in (1) above.

6. Linear indication between station markers #39 and #49 on elbow side of middle bead.

7. Porosity at station marker #49 in center of middle bead, code acceptable.

593 293

The linear indications described above could possibly be I.D. machining marks, and should be verified by visual and penetrant examination.

S. A. Wenk
SwRI Level III - RT

593 294

ULTRASONIC EXAMINATION OF STEAM GENERATOR SECONDARY FEEDWATER NOZZLE WELDS AT POINT BEACH NUCLEAR GENERATING STATION NO. 2

On the first, second, and third of July 1979, the welds and adjacent base metal for the feedwater reducer-to-nozzle, pipe-to-reducer, and elbow-to-pipe welds, on the feedwater piping of the steam generators for Point Beach Nuclear Generating Station No. 2 were examined in accordance with the SwRI Procedure 600-3, Rev. 46 insofar as was practicable. This procedure required 0° longitudinal and 45° and 60° shear wave examination from both sides of the weld. Calibration was performed on Standard No. 18-CS-X-.688-CS-PTB.

This procedure requires, as a minimum, attenuation measurements, 0° longitudinal, 45° shear transverse to the pipe axis, and 45° and 60° shear examinations. The basic calibration block used for these examinations was 18-CS-X-.688-CS-PTB.

All three examination angles were used on each weld except as noted below.

All three examination angles was not performed on each weld. Exceptions were:

(1) The nozzle-to-reducer and reducer-to-pipe weld on steam generator A were examined using 0° longitudinal mode only. (The 45° and 60° shear wave examinations were not performed.)

(2) The nozzle-to-reducer and reducer-to-pipe weld on steam generator B were examined using 0° longitudinal and 45° shear wave. 60° shear wave examinations were not performed.

The not performed examinations were at the perogative of the management of Point Beach Nuclear Generating Station No. 2 because of the decision to remove the reducer.

Evaluation of Ultrasonic Indications on Steam Generator B 18 Inch Reducer-to-Nozzle Weld

The 45° shear wave examinations from the downstream nozzle side of the weld showed only two indications exceeding 50 percent of the calibration. These indications damped on the crown and were resolved as geometrical in nature, probably counter bore-to-crown reflections.

The 45° shear wave examinations from the reducer or upstream side of the weld disclosed a low level signal for 360° of the circumference with several indications exceeding the 50 percent recording level. However, in accordance with Paragraph 8.0 of the ultrasonic procedure used, indications regardless of their amplitude which, in the opinion of the operator, are caused by non-geometric conditions should be recorded. Precise locations of those portions of the signals which exceeded the mandatory 50 percent calibration were duly recorded. Resolution of these indications indicated that they were crack-like in nature not caused by geometry. These signals were also detected from the weld crown with the ultrasonic beam being directed from the nozzle toward the reducer. (This information was utilized in resolving the indications.)

593 295

Figure 1 is a polar ordinate plot showing the relative position and amplitude of the ultrasonic signals as well as the reported radiographic indications. It should be noted however, that the ultrasonic stations proceeded clockwise from top dead center when facing the direction of flow and the radiographic station markers proceeded counterclockwise from top dead center facing the director of flow. This difference has been corrected for all plots and the plots are clockwise facing the direction of the flow.

The inch stations on the outside of the polar plot Figure 1 are the ultrasonic station markers. The radiographic station markers were corrected to agree with the ultrasonic station markers and so plotted. This provides concise correlation between the radiographic indications and the ultrasonic indications.

No recordable indications were noted on either 0° longitudinal examination or the 45° transverse examinations.

Evaluation of Ultrasonic Examination of Steam Generator B Feedwater 16 Inches Pipe-to-Reducer Weld

The 45° shear wave examination from the reducer side of the weld indicated only one point where the ultrasonic signal reached an amplitude of 50 percent of the calibration amplitude. No signals were noted which appeared to travel for any significant length.

The 45° shear wave from the pipe side of the weld showed an indication which traveled for 360° of the weld circumference exceeding 50 percent of the calibration level at several places. This indication was also detectable when examined from the weld crown with the shear wave directed from the reducer to the pipe or opposite the mandatory scan. Figure 2 is a polar ordinate plot showing the position and relative amplitude of the ultrasonic signals. The radiographic indications were also plotted to show a correlation between the ultrasonic and radiographic indications.

No reportable indications were noted during either the 45° transverse scan or the 0° longitudinal scan.

Meetings, Discussions, and Further Examinations

A meeting was held between SwRI, the Superior Industrial X-Ray, and Point Beach Nuclear Generating Station management on the morning of 2 July 1979. The results of the examinations performed on the night and morning of 1 and 2 July were discussed.

During the discussion it was pointed out to the management of Point Beach Nuclear Generating Station that although the indications were indicative of crack like defects, more data would be necessary to positively identify the source of the ultrasonic signals as eminating from cracks. However, the results of the radiographic and ultrasonic examiantions were, in the opinion of Point Beach Nuclear Generating Station management, sufficiently conclusive to indicate the necessity for mechanical removal of the reducer. This was performed during the afternoon and evening of 2 July 1979.

593 296

Results of Ultrasonic Examination of the 16 Inches Elbow-to-Pipe Weld on Steam Generator B

On 3 July 1979 ultrasonic examinations were performed on the elbow-to-pipe weld for both steam generators A and B. During the 45° shear wave examination it was again noted that there was a singular signal trackable for 360° of the pipe circumference. However, in most cases this signal did not exceed 50 percent of the calibration amplitude. The one point at which the signal exceeded 50 percent of the calibration amplitude is clearly indicated in Figure 3. This plot is from the data taken from the pipe side of the weld.

Figure 3 is a polor ordinate plot showing the respective positions for the ultrasonic and radiographic examinations.

No recordable indications were noted on the 60° shear wave examination from the pipe side of the weld. No recordable indciations were noted using the 45° shear wave from the elbow side of the weld.

No recordable indications were noted udring the 45° transverse examination of the weld nor were any reportable indications noted during the 0° longitudinal examination of the weld.

Results of Ultrasonic Examination of the 16 Inches Elbow-to-Pipe Weld on Steam Generator A

The 45° shear wave examination of the pipe side of the elbow-to-pipe weld disclosed an ultrasonic signal preceding 360° around the circumference of the weld. Figure 4 is a polar ordinate plot showing the position where the ultrasonic signal exceeded 50 percent of the calibration amplitude as well as the positional location of the radiographic indications. 60° shear wave examination from the pipe side of the weld showed 2 places where the signal amplitude exceeded 50 percent of the calibration amplitude. It was noted that the 60° shear wave signal was not trackable through the full 360 degrees. An ultrasonic signal was also detected using 45° shear wave from the top of the weld crown toward the pipe side of the weld. Thus, confirming the absence of geometry as a source for the reflected signal.

The 45° shear wave examination of the elbow side of the weld showed a singular indication 360° around the circumference of the weld. This signal was less than 50 percent of the calibration amplitude for the majority of the circumference. Those areas where the signal amplitude exceeded 50 percent of the calibration amplitude were noted.

The 60° shear wave examination of the elbow side of the weld showed that the signal ampliutde exceeded 50 percent of the calibration amplitude in six places. The signal was not discernable except in the area adjacent to those areas exceeding 50 percent of the calibration amplitude. These locations were noted on Figure 5.

No recordable indications were found on the 45° transverse and 0° longitudinal scans of the weld.

593 297

Supplementary ultrasonic examinations were performed after comparing data with radiographic examinations particularly in the areas where radiographic examinations disclosed transverse indications. These areas were located and ultrasonic examinations indicated presence of low amplitude signals, however, these signals were not detected during a circumferential or transverse 45° examination. They were observed only when the ultrasonic transducer was placed at a high skew angle on the base material adjacent to the weld or on the weld crown itself. This information is included as supplementary data and was not plotted for reporting purposes.

593 298

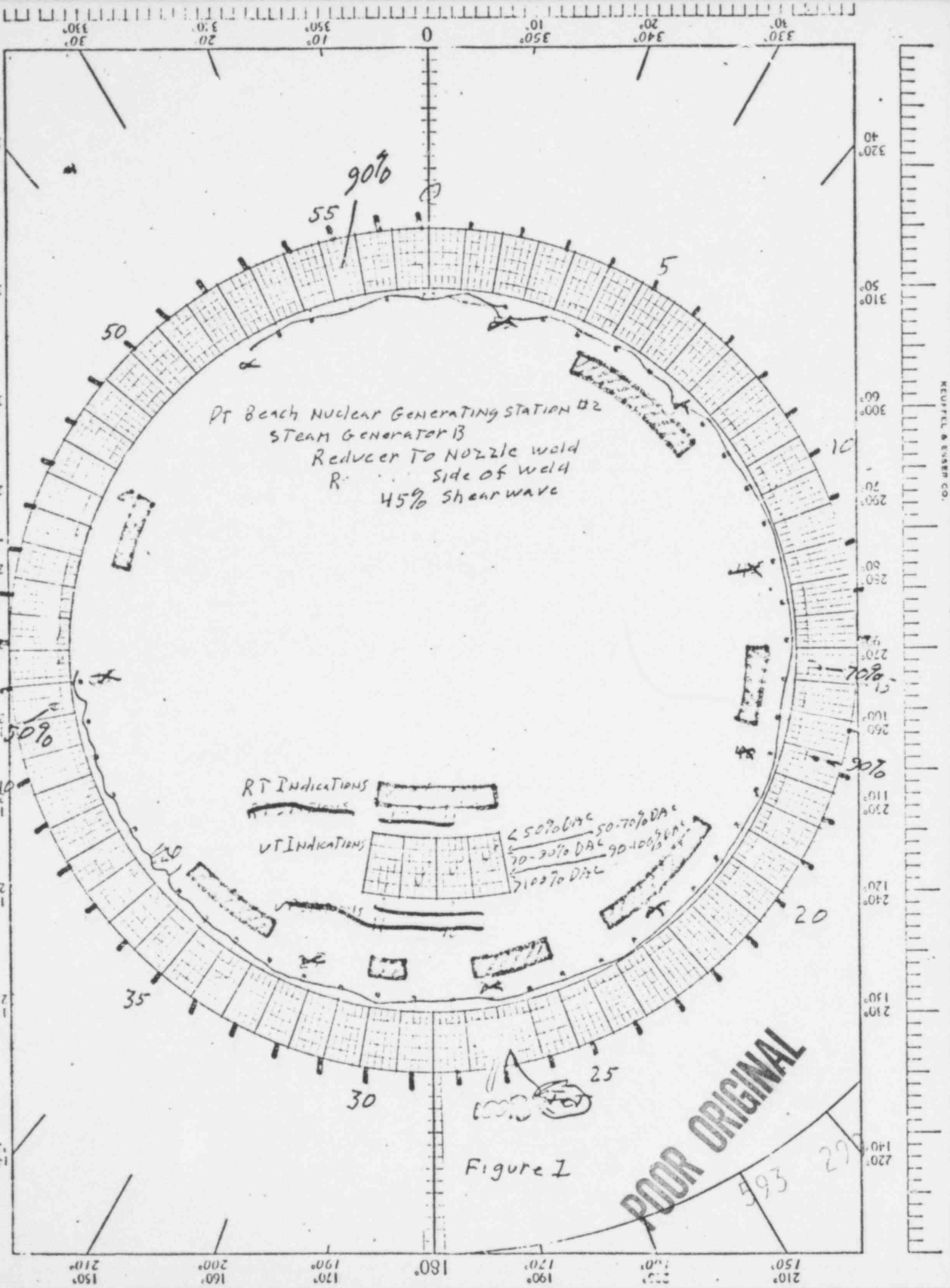

Pt Beach Nuclear Generating Station #2
Steam Generator B
Reducer To Nozzle weld
R Side of Weld
45% Shearwave
RT Indications
UT Indications
<50% DAC
50-70% DAC
70-90% DAC
90-100% DAC
>100% DAC
90%
55
5
50
10
45
70%
15
50%
90%
40
20
35
25
30
POOR ORIGINAL
593 290

Figure 1

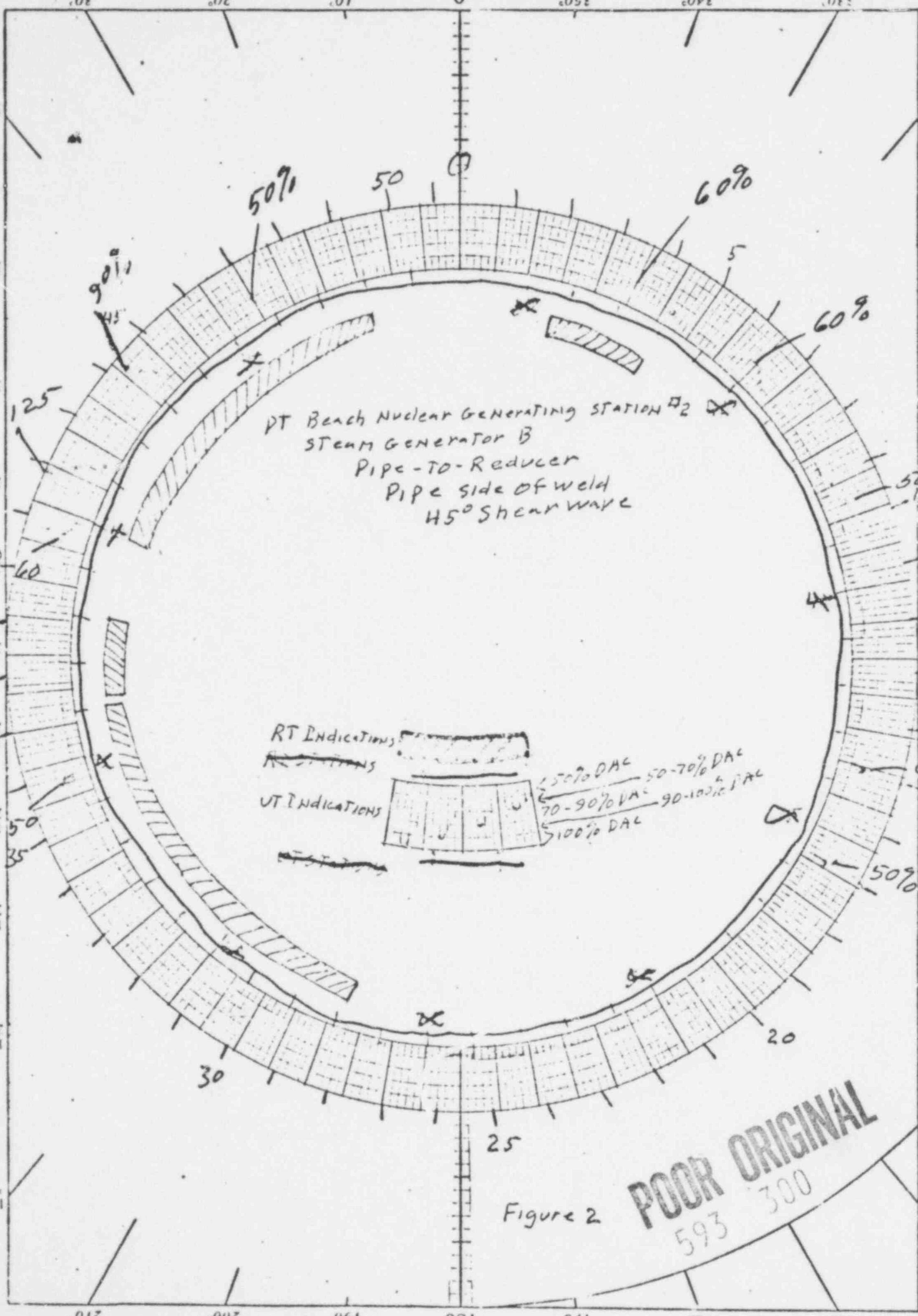


Figure 2

593 300

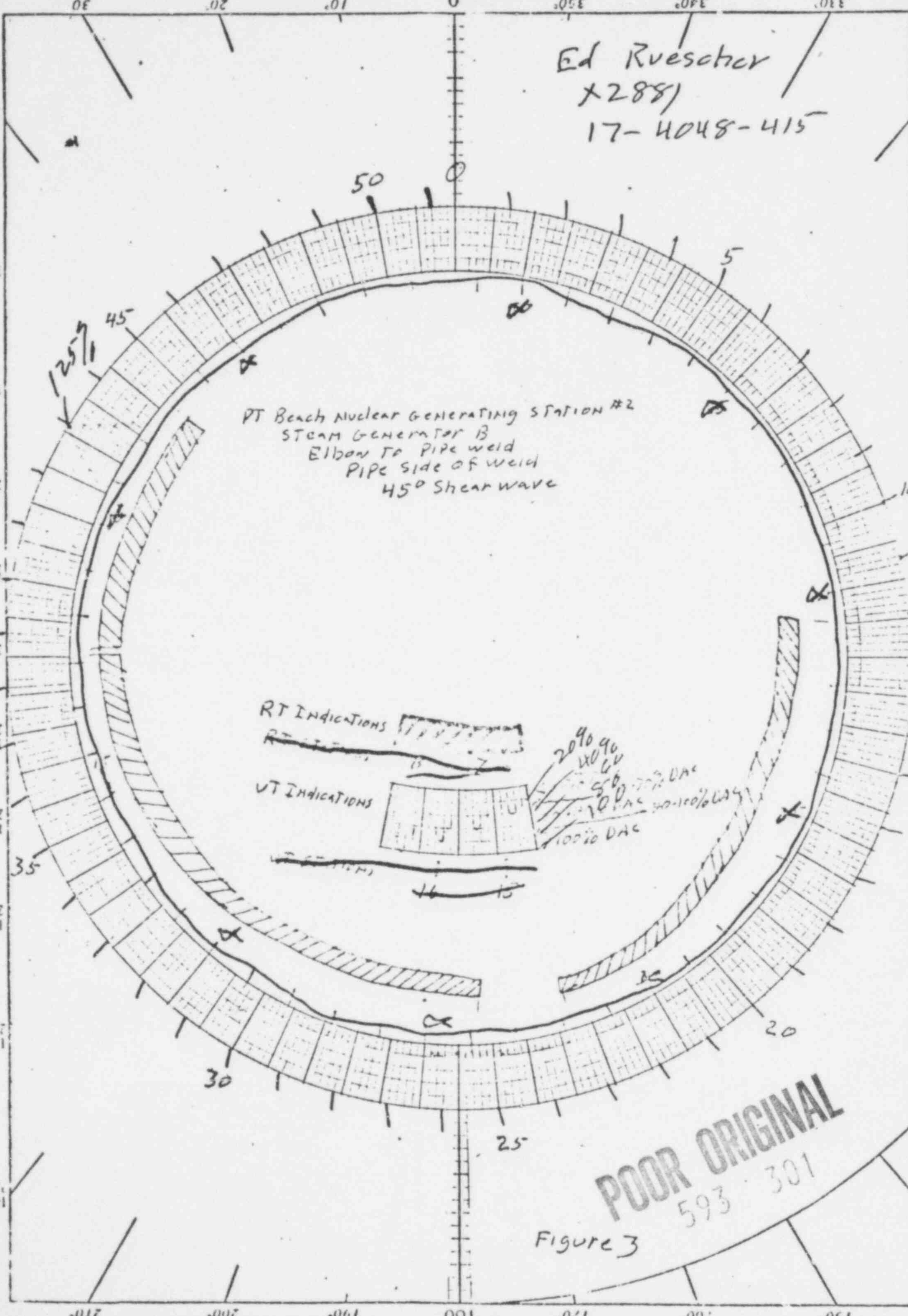

Ed Ruescher
x2881
17-4048-415
PT Beach Nuclear Generating Station #2
Steam Generator B
Elbow to Pipe weld
Pipe side of weld
45° Shear wave
RT Indications
UT Indications
20%
40%
50%
100% DAC
100% DAC
0
5
10
15
20
25
30
35
45
50

Figure 3

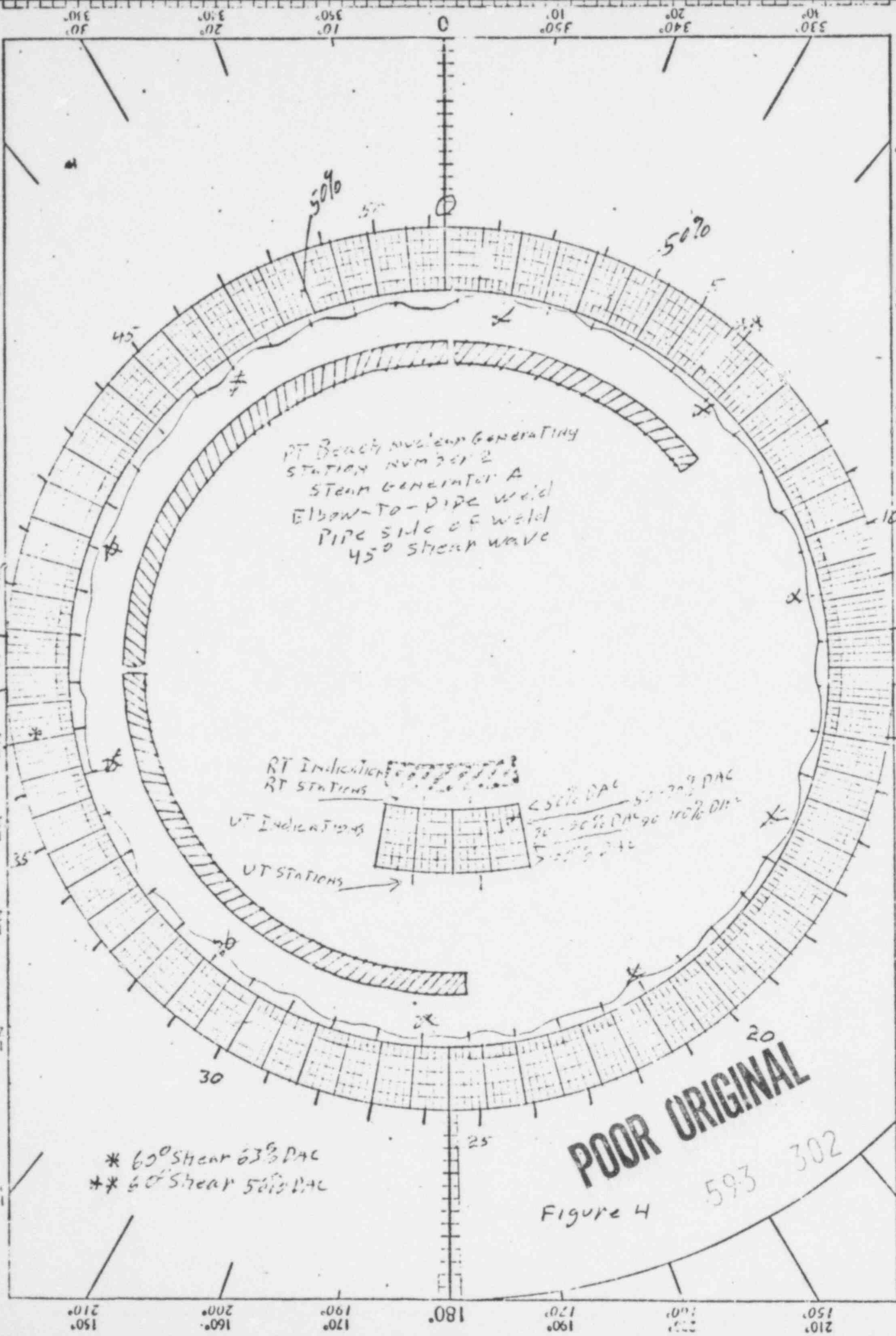


Figure 4

POOR ORIGINAL

593 302

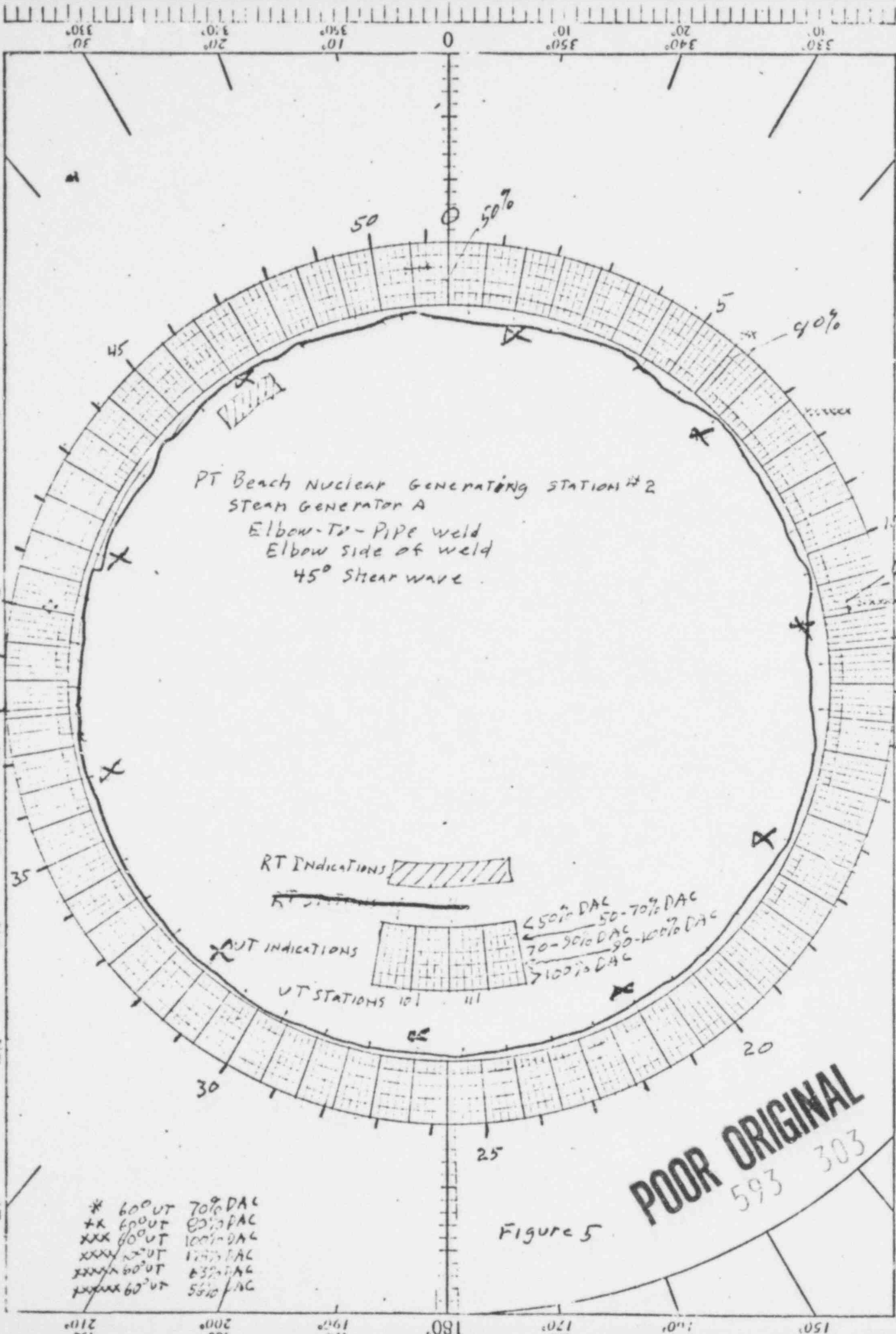


Figure 5

593 303

APPENDIX B

Wisconsin Electric Power Company
Point Beach Nuclear Plant

SUMMARY OF SURFACE INDICATIONS AND CORRECTIVE ACTIONS FOR UNIT 2 FEEDWATER PIPING WELDS

July 1979

593 304

1.0 Steam Generator "A" Feedwater Pipeline

1.1 Feedwater Nozzle

Numerous pits were observed on the inside surface of the feedwater nozzle. These were ground out and two areas were repair welded. These areas were each approximately 1/8" x 5/8". The pits were generally shallow, with one or two about .040 inches deep. The rest were less than .020 inches deep. Most were on the bottom half of the nozzle within about 2½" of the edge. On a relative basis, the "A" nozzle had less pitting than the "B" nozzle.

After repair of the nozzle, a dye penetrant examination of the outside and inside of the nozzle was performed with acceptable results. The weld preparation was then completed per Detail II in Phillips-Getschow welding procedure IA-MA-13, Revision 07-07-79. Prior to welding to the reducer, the nozzle weld preparation was radiographed with acceptable results.

1.2 Reducer

The weld preparation on the new reducer was made somewhat differently than on the original reducer. The counterbore was machined to a depth of 1½" in order to remove its end from the area of the weld. The transition angle from the counterbore to the inside of the reducer was machined to an angle less than ten degrees and a small radius was ground where the counterbore met the slope. The reducer ends were inspected with dye penetrant, ultrasonics and radiography with acceptable results.

1.3 Auxiliary Feedwater Connection

Visual examination of the three-inch branch connection weld from the inside of the pipe indicated a lack of penetration in the root pass of the weld. This connection was cut off completely and the weld redone. The three-inch pipe was shortened approximately 3/8" during this operation. A radius was ground on the inside edge of the penetration in the main feedwater piping.

Structural reinforcement (a Weldolet) is being used for this connection to meet the requirements of B31.1.

1.4 Elbow to Pipe Weld

The outside surface of the elbow to pipe weld was ground flat, polished and blended into the pipe in order to facilitate radiography. No outer diameter indications were observed. Linear indications were observed in the radiographs. Also two tranverse (perpendicular to the weld centerline) indications, one approximately ¼" deep by 3/8" long and another 1/8" away that was 1/8" deep by ¼" long were noted. These indications were repaired. The inside surface of the weld was then ground and polished and examined with

593 305

dye penetrant. A linear indication seven to eight inches long was observed near the top of the pipe on the pipe side of the weld. In the same plane, near the bottom of the pipe, a machining grove was noticed approximately four to five inches long. Both were removed by grinding.

After the inside of the weld had passed a dye penetrant examination, a radiograph revealed an intermittent linear indication from the 12 o'clock to 3 o'clock position facing upstream. This was found to be a combination of lack of fusion and a line of slag against one sidewall and another line of slag between weld beads. These were apparently left during the original submerged arc welding. Two areas, one approximately three inches long and one approximately four inches long were ground completely through the weld to remove the slag. The weld was then repair welded.

2.0 Steam Generator "B" Feedwater Pipeline

2.1 Feedwater Nozzle

Numerous pits were observed on the inside surface of the feedwater nozzle. These were ground out and repair welded. The repair welding covered approximately 25 square inches at the bottom of the nozzle extending from the edge to 2½" to 3" inward. The pitting was generally worse on this nozzle than on the "A" nozzle. Several pits were approximately 1/8" deep.

A dye penetrant examination of the inside and outside of the nozzle revealed three or four linear indications in the inside and eight on the outside. These were shallow and were ground out.

After repair of the nozzle, a dye penetrant examination of the inside and outside of the nozzle was performed with acceptable results. The weld preparation was then completed per Detail II in Phillips-Getschow's welding procedure, IA-MA-13, Revision 07-07-79. Prior to welding to the reducer, the nozzle weld preparation was radiographed with acceptable results.

2.2 Reducer

This new reducer was machined and inspected the same as the "A" reducer.

2.3 Auxiliary Feedwater Connection

Visual examination of the three-inch branch connection weld from the inside of the pipe revealed a hole in the weld approximately one-quarter inch deep. This connection was cut off completely and the weld redone. Dye penetrant checks of the three-inch pipe end prior to rewelding revealed several indications so the pipe was shortened approximately 1¼". A radius was ground on the inside edge of the penetration in the main feedwater piping.

593 306

Structural reinforcement (a Weldolet) is being used for this connection to meet the requirements of B31.1.

2.4 Elbow to Pipe Weld

The outside surface of the elbow to pipe weld was ground flat, polished and blended into the pipe in order to facilitate radiography. No outer diameter indications were observed. The radiographs revealed several linear indications. The inside of the weld was then ground, polished and dye penetrant examined. A three to four-inch long indication near the top and a five to six-inch long indication near the bottom of the pipe on the pipe side of the weld were ground out. Also observed and ground out were two spots of porosity. Minimum wall thickness was maintained and no repair welding was required.

593 307

APPENDIX C

Wisconsin Electric Power Company
Point Beach Nuclear Plant

INTERIM REPORT OF THE METALLURGICAL EVALUATION OF UNIT 2 "A" MAIN FEEDWATER PIPING REDUCER

July 1979

593 308

INTERIM REPORT

METALLURGICAL ANALYSIS OF POINT BEACH 18-IN. TO 16-IN. REDUCER

1.0 Introduction

On July 5, Southwest Research Institute was supplied one 18-in. to 16-in. reducer, removed from the A steam generator of the Point Beach Unit 2 Generating Station. The reducer was removed by cutting through the 18-in. reducer to nozzle and the 16-in. reducer to pipe welds. In general, the cut passed through the weld crown at the O.D. and in the heat-affected zone of the root pass at the I.D. on the reducer side of the weld. The weld fusion line at the I.D. was not present over most of the circumference.

2.0 Specimen Preparation

To date, only the 18-in. nozzle to reducer weld has been examined. Radiographs performed at Point Beach by Superior Industrial X-Ray Corp. and interpreted by Mr. S. Wenk of SwRI showed a crack-like indication from station marker #10 to #16, 63° - 100°. Two transverse linear indications, each 1/4 in. long, were also detected at station markers #42 and #43, ~265° running from the weld crown into the reducer base metal.

Before removing any material from the reducer, ultrasonic inspection was performed to more accurately position the defect causing the RT indications. This examination confirmed the existence of a flaw which was clearly present from stations #10 to #21 (62° - 130°) and #23 to #25 (144° - 157°). The flaw was positioned approximately 3/8 in. from the end of the reducer, that is, in the vicinity of the transition from the counter-bore to full reducer I.D. Only the section from stations #8 to #26 was examined using ultrasonics.

After location of the flaw giving rise to the RT indications, a ring 1-1/2 in. to 2 in. wide was cut from the 18-in. end of the reducer. No lubricant was employed during this or any subsequent cutting operation. The ring was then cut in half, through stations #0 and #28, 0° and 180°. Each half was examined visually using a stereoscopic microscope at magnifications up to 50X.

No crack could be unambiguously identified at the counter-bore transition. In the zone from station #10 to #20 the oxide was unusually rough and porous in appearance. Over most of the circumference the oxide at the counter-bore transition was not noticeably different from that in other regions. Randomly distributed pit-like areas were present, but were not grouped linearly as at stations #10 - #20. Cracks could be positively identified at the root pass fusion line at stations #37 - #37.5, #42 - #45, and #45.5 and #49 (~230° and 260° - 310°).

Following this examination, specimens for metallographic polishing were removed at station #10, #15, #22, #28, and #43 (63°, 94°, 138°, 180°, and 270°). Sections 1/2 in. wide were removed adjacent to stations #10,

593 309

#15, and #43 for fractographic examination. These sections were notched in the region of the counter-bore transition, cooled in liquid nitrogen, and broken open. This procedure causes the remaining sound material to cleave, thereby facilitating identification of the subcritical crack.

3.0 Results

The structure of the reducer in all sections was normal for mild steel, Figure 1. No microstructural abnormalities, such as islands of martensite, weld porosity, or lack of fusion, were evident in the weld or heat-affected zone.

Cracking was found in all the sections taken. The depth of the largest crack and the number of cracks in each section are given in Table I. In all cases, multiple cracking was observed, Figures 2 through 4. Although cracks were observed in all areas of the counter-bore, from the weld fusion line to the original reducer I.D., cracking with depths greater than 0.015 in. was restricted to the weld fusion line and counter-bore transition region. Of the two, the counter-bore transition zone contained the deeper cracks, Figure 3.

Many of the cracks nucleated at grooves created during the coarse machining operation used to counter-bore the reducer. The mouths of most of the deeper cracks were wide and oxide filled, creating a pit-like appearance. However, several small cracks, and one larger crack in Figure 3, were not associated with large pits. This indicates that pits form after cracks have nucleated, rather than being a precursor of cracking. The presence of wide pits at the mouths of larger cracks, and the absence of such pits at small cracks may be interpreted as evidence that the larger cracks have been present for a considerable length of time. The extent of oxidation along crack walls, which is considerable even in the vicinity of crack tips, Figures 2, 3, and 7, also suggests that the crack growth rate is small. The operating conditions experienced by the reducer, 450°F in pure water, do not normally cause rapid oxidation, and thus a considerable length of time would be required to cause the degree of oxidation observed.

Figure 5 shows the fracture surface of specimens cut at stations #10 - 10.5 and #43 - 43.5. The black zone near the edge of the sample is the oxidized inservice crack, the silver area, laboratory cleavage fracture. Both samples were broken in the vicinity of counter-bore transition zone. It is of interest to note that one specimen was notched and impacted at room temperature. In this case, the inservice crack blunted considerably, but no crack extension occurred.

Fractography has only been performed on specimens in the as-fractured condition. No cleaning or oxide removal has yet been attempted. Figure 6 shows a typical region near the crack tip. The surface is intergranular in appearance, with grain size comparable to that of the base metal, Figures 1 and 7. This suggests that the mode of cracking is intergranular. This observation should not be interpreted as definitive since the possibility remains that the grains imaged are of the thick oxide often present near the crack tip, Figure 7, rather than of base metal. This point will be clarified

593 310

TABLE I

NUMBER AND DEPTH OF CRACKS

Section	Maximum Depth (in.)	Number of Cracks Deeper than 0.02 in.
#10	0.018	8
#15	0.043	13
#22	0.014	4
#28	0.008	3
#43	0.042	7

593 311

during the continuing metallurgical analysis. Energy dispersive x-ray analysis was performed on all fracture surfaces. Iron was the only major constituent of the spectrum, with very small copper and sulphur peaks. With extended count times, peaks due to phosphorus, zinc, nickel, and titanium could also be resolved. These results are consistent with normal operating conditions, since Admiralty Brass condenser tubes and Inconel-600 steam generator tubes are present in the system. No indication of contamination of the system by chlorides or sodium hydroxide could be found.

It is postulated that cracking is the result of corrosion fatigue. The morphology of cracking is similar to that sometimes observed in conventional steam plants subjected to fatigue loadings.(1) Such failure could be intergranular at low cyclic stresses and transgranular at higher stresses, similar to the behavior observed for sensitized austenitic stainless steels in pure water environments.(2)

The origin of the cyclic stress is not immediately apparent. The circumferential orientation of the cracks indicates that the primary stress is axial, rather than a hoop stress. Further analysis is necessary to determine if cracking is symmetrical about the vertical axis. Such a finding would indicate that reverse bending in the horizontal plane could be the primary cause of failure. The preliminary findings reported here suggest that this is unlikely.

Thermal stresses, caused by cold water eddy currents, can generate sufficient cyclic stress to cause corrosion fatigue. The stress in such a situation is biaxial in nature. Unless an axial bias is imposed, thermal yclic stress is expected to cause both axial and circumferential cracking. Further analysis is required to substantiate whether axial cracks are present.

4.0 Conclusions

Linear indications in radiographs of the 18-in. reducer to nozzle weld area were caused by the presence of oxide-filled cracks up to 0.043 in. deep.

Multiple crack nucleation occurred at stress concentrators, such as the weld fusion line, machining grooves, and the counter-bore transition zone. All cracking was restricted to the reducer internal diameter. No O.D. cracks were observed.

Oxide-filled pits are associated with deeper cracks. All cracks were difficult to detect visually because of this thick oxide.

Cracks were shallow, less than 0.043 in., and appeared to have been present for a lengthy period.

The brittle to ductile transition temperature of the reducer material appears to be below room temperature.

593 312

The most probable cause of cracking is corrosion fatigue.

5.0 References

1. Barer, R. D., "Boiler and Turbine Component Failures," Metallography in Failure Analysis, Plenum Press, New York, 1977.

2. Shoji, T., Ise, T., Takahashi, H., and Susuki, M., Corrosion, Vol. 34, p 366, 9/78.

593 313

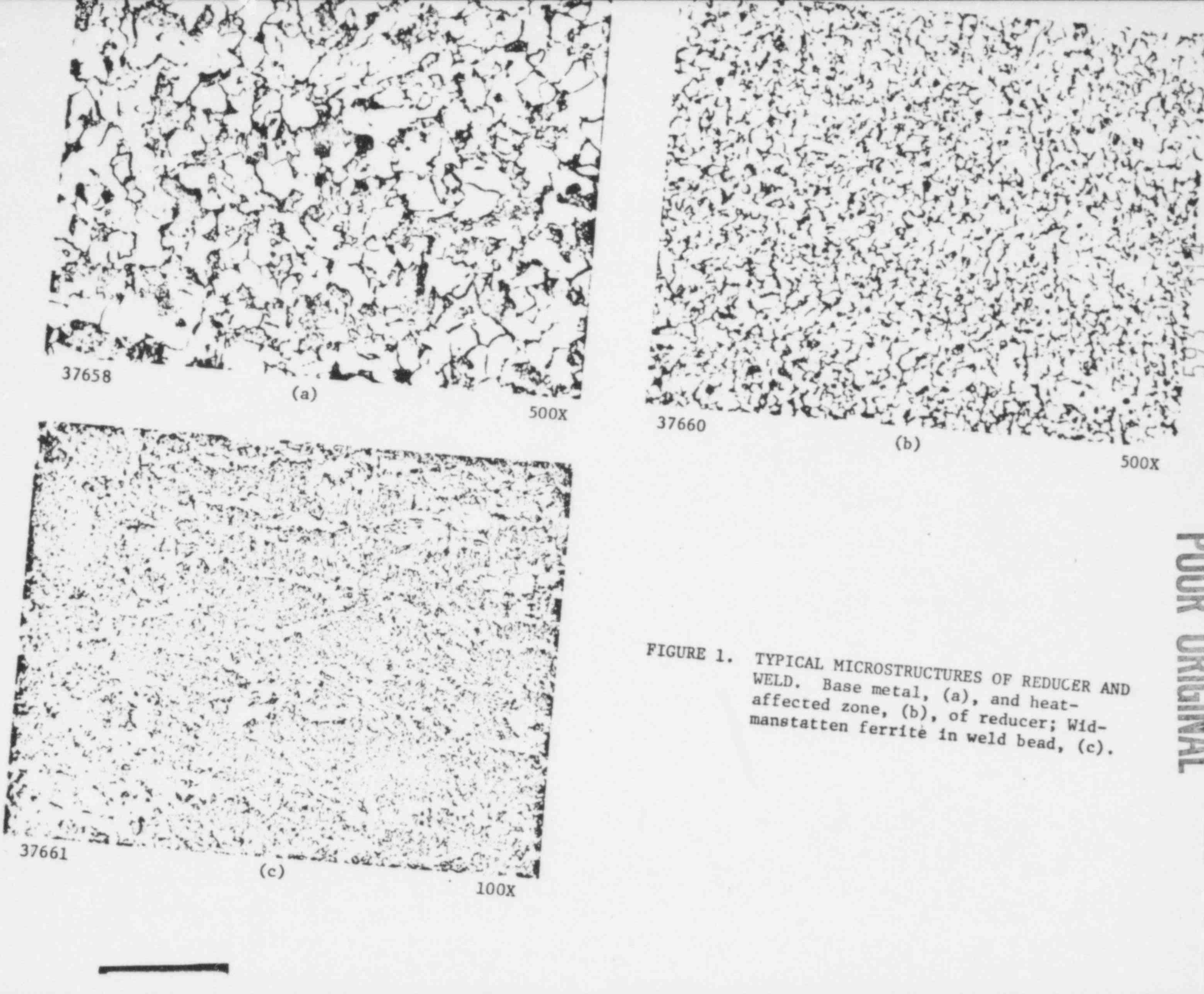
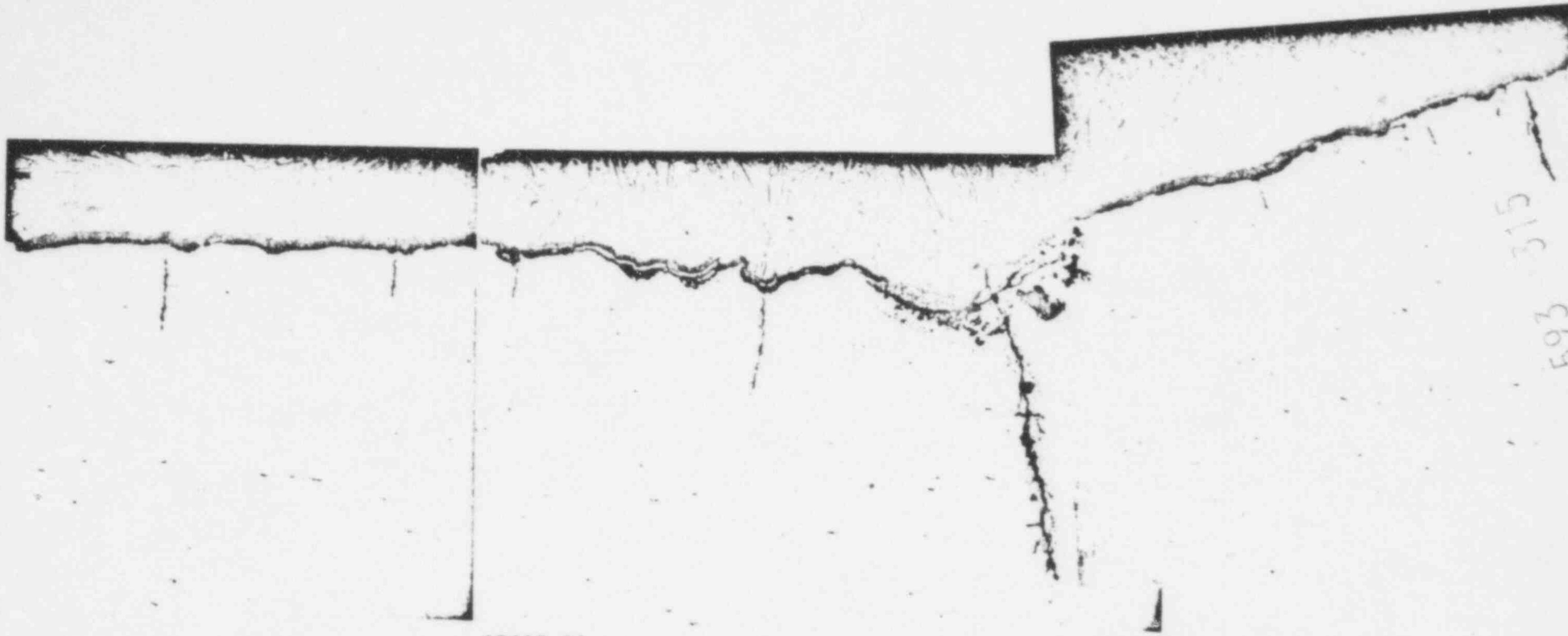


FIGURE 1. TYPICAL MICROSTRUCTURES OF REDUCER AND WELD. Base metal, (a), and heat-affected zone, (b), of reducer; Widmanstatten ferrite in weld bead, (c).

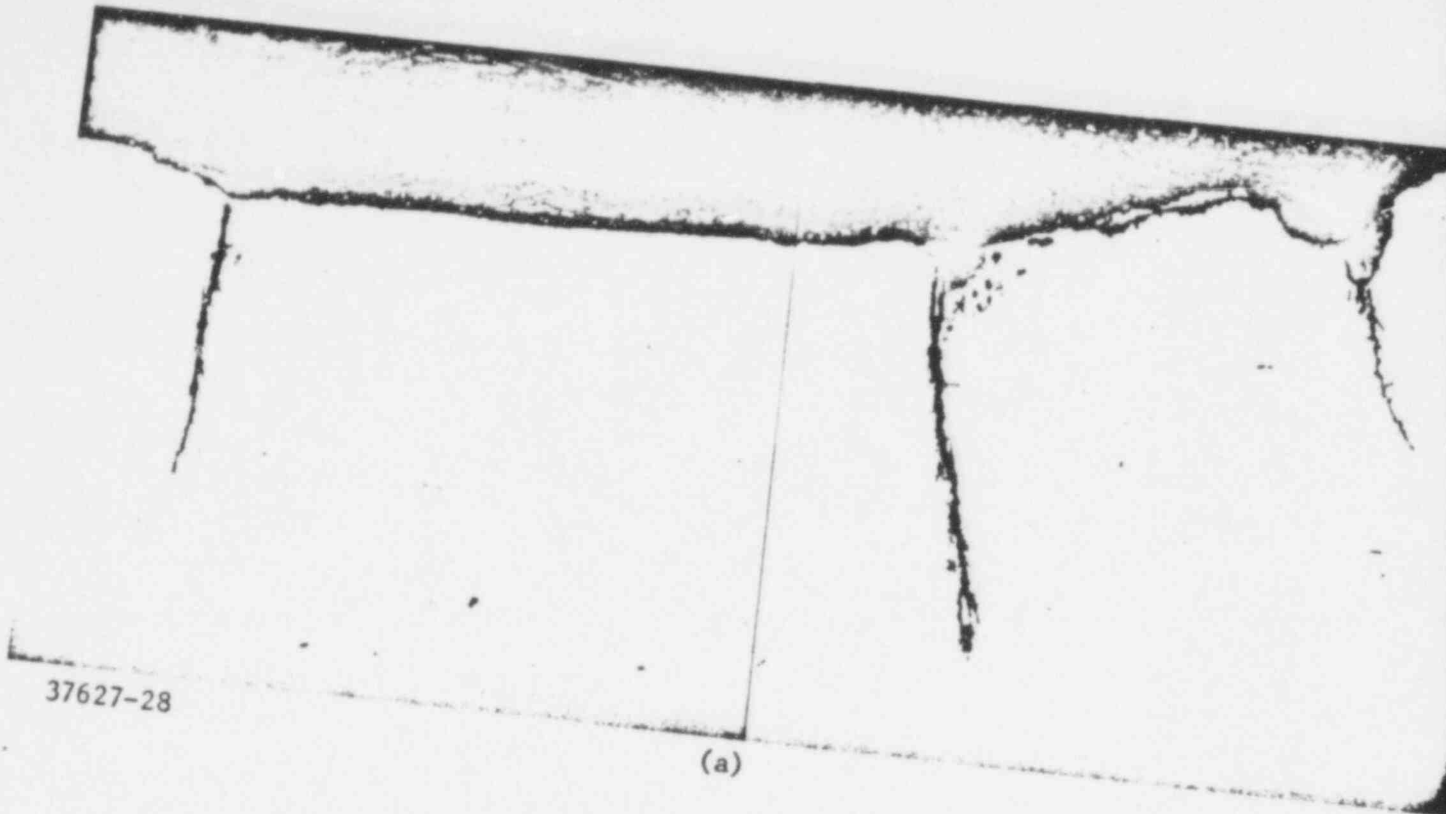


37622-24

50X

593 315

FIGURE 2. LONGITUDINAL SECTION AT STATION #15



37627-28

(a)

50X

37626

(b)

50X

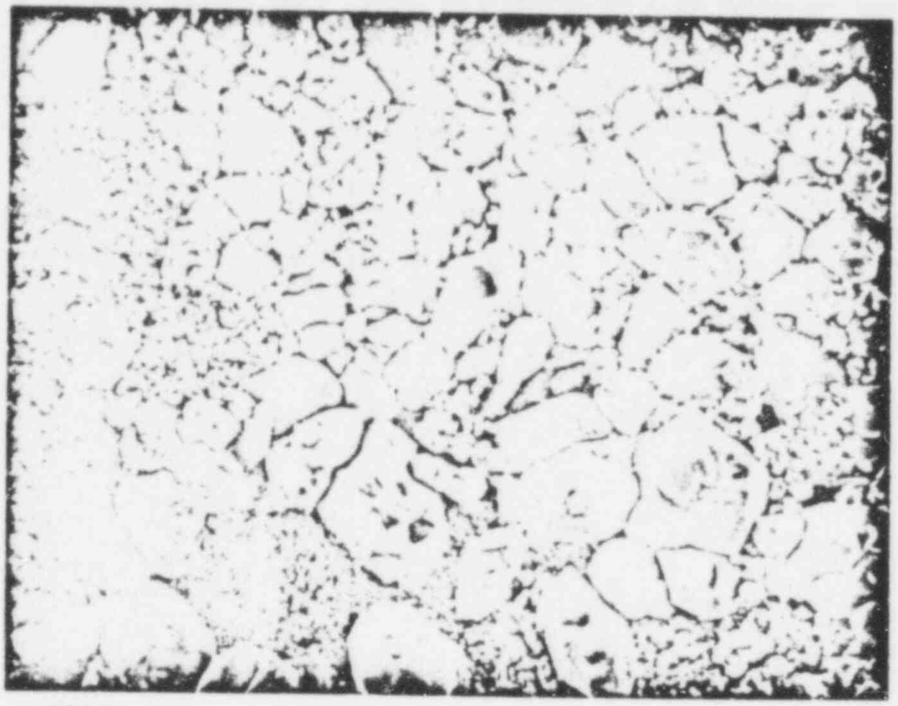
FIGURE 3. LONGITUDINAL SECTION AT STATION #43. Cracking occurred at the counter-bore transition zone, (a), and at the weld fusion line, (b).

593 316

FIGURE 4. LONGITUDINAL SECTION AT STATION #28

POOR ORIGINAL

593 317



1948S 500X

FIGURE 6. TYPICAL FRACTOGRAPH OF CRACK TIP OF #15 SEGMENT

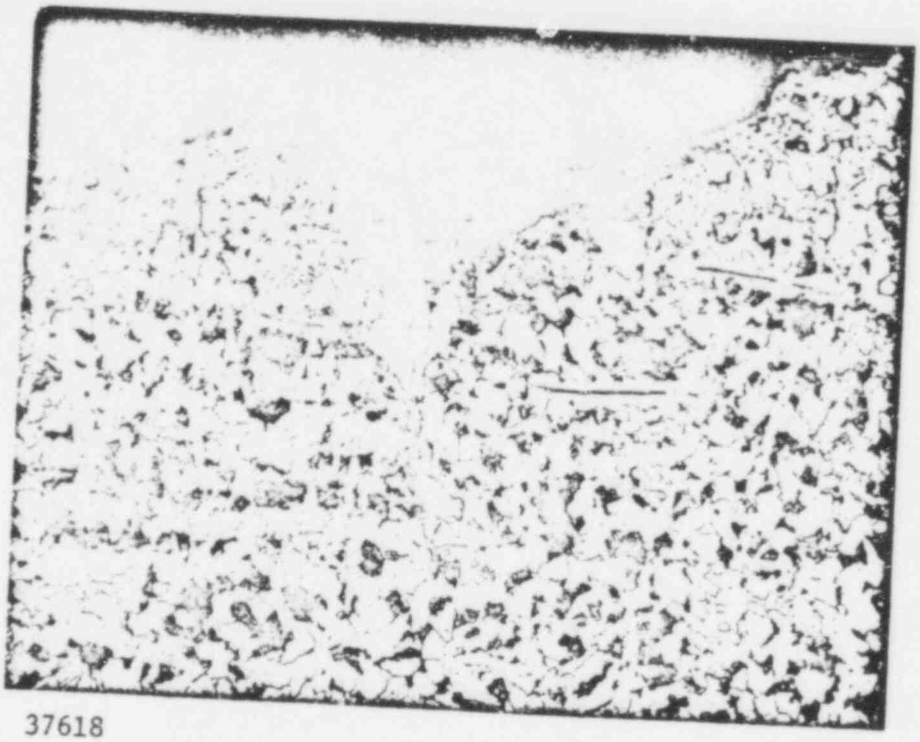
593 318

37620 (a) 4X

37619 (b) 4X

FIGURE 5. SAMPLES FROM STATION #15, (a), AND #43, (b), READY FOR FRACTOGRAPHIC ANALYSIS

593 319



37618

250X

FIGURE 7. HIGHER MAGNIFICATION OF CRACK SHOWN IN FIGURE 4. Etchant, 2% Nital. Note the extent of oxidation at the crack walls.

593 320