Duquesne Light
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Pittsburgh, Pa.
15219

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July 17, 197%

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334
LER 79-14 (Feedwater Line Elbow Cracks)
Response to IE Bulletin 79-13

Gentlemen:

This letter and the accompanying report in the form of three attachments, supplement our Licensee Event Report 79-14 dated June 19, 1979 and the followup report dated July 2, 1979 concerning the small cracks identified in the feedwater line elbows at the Beaver Valley Power Station, Unit No. 1. The report, submitted in accordance with Appendix A, Technical Specification 6.9.1.8.i, details the results of the Duquesne Light Company's investigation into this matter, the corrective act[illegible]s taken and planned, and a safety evaluation of the report.

Radiographs were taken on June 15, 16, and 18, 1979, of the three steam generator nozzle-to-feedwater inlet piping welds.

The radiographs showed cracking to be present in all three inlet pipes. The inlet piping at these locations are 90° elbows. The lines are 16 inches in diameter with an 0.843 inch wall thickness. In each instance, the cracks originated at the shoulder of the counterbore in the piping, or approximately 9/16 inch from the root of the weld.

Mea[illegible]uring in a circumferential and clockwise direction and facing the flow, the specific locations and size of the cracks are as follows. The measurements of 0 and 51 inches are at the top of the pipe; 13 at 90°; 25-26 at the bottom; and 38 at 270°.

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Steam Generator No. 1A	Steam Generator No. 1B	Steam Generator No. 1C
Cracks were present at:	Cracks were present at:	Cracks were present at:
49 1/2" through 0 to 2"	48" through 0 to 2"	48" through 0 to 5"
10 1/2" to 15"	9" to 13"	26" to 30" in a suck-up area of the root pass
39" to 41"	33" to 43"	35" to 43"

A magnetic particle examination of the affected areas did not show any cracking to be present on the outside surface of the piping.

All three elbows have been removed from the feedwater lines and a four inch ring specimen which contains the defect has been obtained from each line. The observed defects appear to be similar to those which have previously been discovered at the DC Cook and HP Robinson Plants. The C Loop ring sample was provided to the Westinghouse Electric Corporation for metallographic examination.

The entire ring sample from the B loop was forwarded to Steen Engineering, Inc. at the request of G. Walton of Region I. The sample from the A loop is being retained at the site for possible future examination.

In accordance with IE Bulletin 79-13, a rad[illegible]phic examination of all welds in the feedwater lines inside the Reactor C[illegible]inment Building has been performed. This examination was performed in acc[illegible]e with ASME Section III requirements as described in Action 1.a of the bulletin. A tabulation of the results of these examinations is provided in Attachment 1.

A total of 13 welds were found to have indications that required further evaluation. We have decided to repair all thirteen (13) welds t[illegible]at exhibited indications, even though these welds would be acceptable by the inspection requirements of ANSI B.31.1 which is the governing code under which the piping was installed.

An instrumentation program for the feedwater lines adjacent to the steam generator nozzles will be developed by the Duquesne Light Company in cooperation with the Westinghouse Electric Corporation. This program will be developed taking cognizance of the adequacy and reliability of the data obtained from similar programs which are underway at other facilities.

The instrumentation required to implement the program will be provided and installed during the fall refueling outage which will commence after approximately six full power weeks of operation after startup have been achieved.

A summary description of the instrumentation program will be provided to the NRC prior to its implementation and informal, periodic reports will be provided during the conduct of the program. A full formal report will be provided upon the completion of the program.

The replacement elbows on the feedwater lines are provided with a 1/2" radius at the counterbore to eliminate the sharp discontinuity on the inner diameter of the line, at which point the cracks on the original elbows were initiated.

Minor pitting and machine groove have been removed from the inside surface of all three (3) steam generator nozzles. It was required to perform a weld build-up on the inside surface of the "C" Steam Generator nozzle to restore the required wall thickness.

A radiographic examination of the root pass of the nozzle welds was conducted during the installation of the elbows. The pre heat temperature was not maintained during the radiography since the temperature was detrimental to the film being used.

The weld repair on the feedwater line elbows was stress relieved at 1150°F for two hours upon completion of the repair. Radiographic and Ultrasonic examination of the nozzle weld will be performed subsequent to the stress relieving operation.

The nozzle weld on the 'C' Steam Generator will be reexamined by RT and UT at the fall refueling outage and similar examinations will be conducted on all three stream generator nozzle welds at the following refueling.

Metallurgical and stress evaluation reports are included as Attachments 2 and 3 to this report.

A research of the operating records discloses that the oxygen content of the feedwater has been maintained at less than 5 ppb except for the following:

a) During station start-up, shutdowns, and large transients.

b) During a three week period in early 1979, the O_2 content was or great as 30 ppbdue to air leakage on the condenser.

Hydrazine is added to the feedwater at th condensate pump discharge for oxygen control and morpholine is added at the same point for pH control. The condenser utilizes stainless steel tubes, and except for two or three cases of mechanical damage caused by loose baffles at points of entry of recirculation lines and steam dumps, the integrity of the main condenser has been excellent. The occurrence of mechanical damage was quickly detected and the unit was removed from the line in a timely manner to effect the necessary repairs. In general, the chemistry control of the feedwater has been maintained in accordance with the recommendations of the Westinghouse Electric Corporation, and the condenser tube integrity has been excellent.

Safety Evaluation

Since no crack had progressed through the walls of the feedwater lines, the existence of these cracks did not represent a threat to the health and safety of the general public.

These cracks could possibly have progressed to the point where the walls of the feedwater lines were penetrated and a leak of the non radioactive feedwater to the containment atmosphere could have occurred. Such a leak would have been detected by increased frequency of operation of the containment sump pump and by an increase in the containment humidity.

Since the plant has been designed to withstand a complete rupture of a main feedwater line, no leak resulting from the propagation of these cracks would have resulted in an event that was greater than that which has been previously analyzed.

The Station Onsite Safety Committee and the Company Offsite Review Committees have reviewed the feedwater elbow cracking condition and have concluded that the matter does not constitute an unreviewed safety question, and that it is safe to operate the unit upon the completion of the repair.

In accordance with the other requirements of IE Bulletin 79-13, an inspection of all feedwater system piping supports has been performed and the operability and conformance to design of these supports has been verified.

Item 4 and 5 of the Bulletin have previously been submitted in accordance with the requirements of the bulletin. This letter and the accompanying attachments fulfill the requirements of Item No. 6.

A complete formal report of the results of the metallurgical and stress evaluations shall be forwarded as soon as these reports, which are being prepared by the Westinghouse Electric Corporation, are available.

Very truly yours,

C N Dunn

C. N. Dunn
Vice President, Operations

Attachments
CC: United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Operating Reactors Inspection
Washington, DC 20555

CC: D. L. Wigginton
Licensing Project Manager, Beaver Valley Unit No. 1
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, DC 20555

ATTACHMENT #1

BEAVER VALLEY POWER STATION
RESPONSE TO IE BULLETIN 79-13

1 (A)	WELD #	DATE	REJECT	INDICATION	ACCEPT
1 (A)	1				
1 (A)	2	7-7-79			X
1 (A)	3	7-6-79			X
1 (A)	4	7-6-79			X
1 (A)	5	7-5-79			X
1 (A)	6	7-3-79			X
1 (A)	7	7-2-79			X
1 (A)	8	7-2-79	X	INCOMP PENET.	
1 (A)	9	7-1-79			X
1 (A)	10	7-1-79	X	SLAG, INCOMP FUSION	
1 (A)	11	6-28-79	X	INCOMP PENET	
1 (A)	12	6-27-79	X	CRACKS	
1 (A)	13	6-21-79			X
1 (A)	14	6-21-79			X
1 (A)	15	6-19-79	X	CRACKS	
1 (A)	FW-9A	7-7-79	X	INCOMP PENET	
1 (A)	FW-9C	7-7-79			X
1 (A)	FW-9B	7-8-79			X
1 (A)	FW-8A	7-8-79	X	INCOMP PENET	
1 (A)	FW-8B	7-8-79	X	INCOMP PENET	
1 (A)	16" BAND	7-6-79			X
1 (A)	AFW-1	7-6-79			X

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2 (B)	WELD #	DATE	REJECT	INDICATION	ACCEPT
2 (B)	1				
2 (B)	2	7-8-79			X
2 (B)	3	7-6-79			X
2 (B)	4	7-6-79	X	INCOMP PENET	
2 (B)	5	7-5-79			X
2 (B)	6	7-5-79			X
2 (B)	7	7-5-79			X
2 (B)	8	6-29-79			X
2 (B)	9	6-29-79			X
2 (B)	10	6-30-79			X
2 (B)	11	6-19-79			X
2 (B)	12	6-21-79			X
2 (B)	13	6-21-79			X
2 (B)	14	6-18-79	X	CRACKS	
2 (B)	AFW	7-6-79			X
2 (B)	FW5-A	7-9-79			X
2 (B)	16" BAND	7-6-79			X

ATTACHMENT #1

3 (C)	WELD #	DATE	REJECT	INDICATION	ACCEPT
3 (C)	1				
3 (C)	2	7-7-79			X
3 (C)	3	7-6-79			X
3 (C)	4	7-6-79	X	INCOMP PENET CRACK	
3 (C)	5	7-5-79			X
3 (C)	6	7-4-79	X	CRACKS, INCOMP FUSION	
3 (C)	7	7-5-79			X
3 (C)	8	7-4-79			X
3 (C)	9	6-22-79			X
3 (C)	10	6-22-79			X
3 (C)	11	6-22-79	X	CONNECTED POROSITY	
3 (C)	12	6-22-79			X
3 (C)	13	6-22-79			X
3 (C)	14	6-22-79			X
3 (C)	15	6-22-79			X
3 (C)	16	6-18-79	X	CRACKS	
3 (C)	FW9-A	7-7-79			X
3 (C)	FW-8A	7-7-79			X
3 (C)	FW-7D	7-7-79			X
3 (C)	FW-7A	7-7-79			X
3 (C)	FW-7B	7-8-79	X	INCOMP PENET, INCOMP FUSION	
3 (C)	FW-7C	7-8-79			X
3 (C)	AFW-1	7-3-79			X
3 (C)	FW-5A	7-9-79	X	INCOMP PENET, INCOMP FUSION	
3 (C)	16" BAND	7-6-79			X

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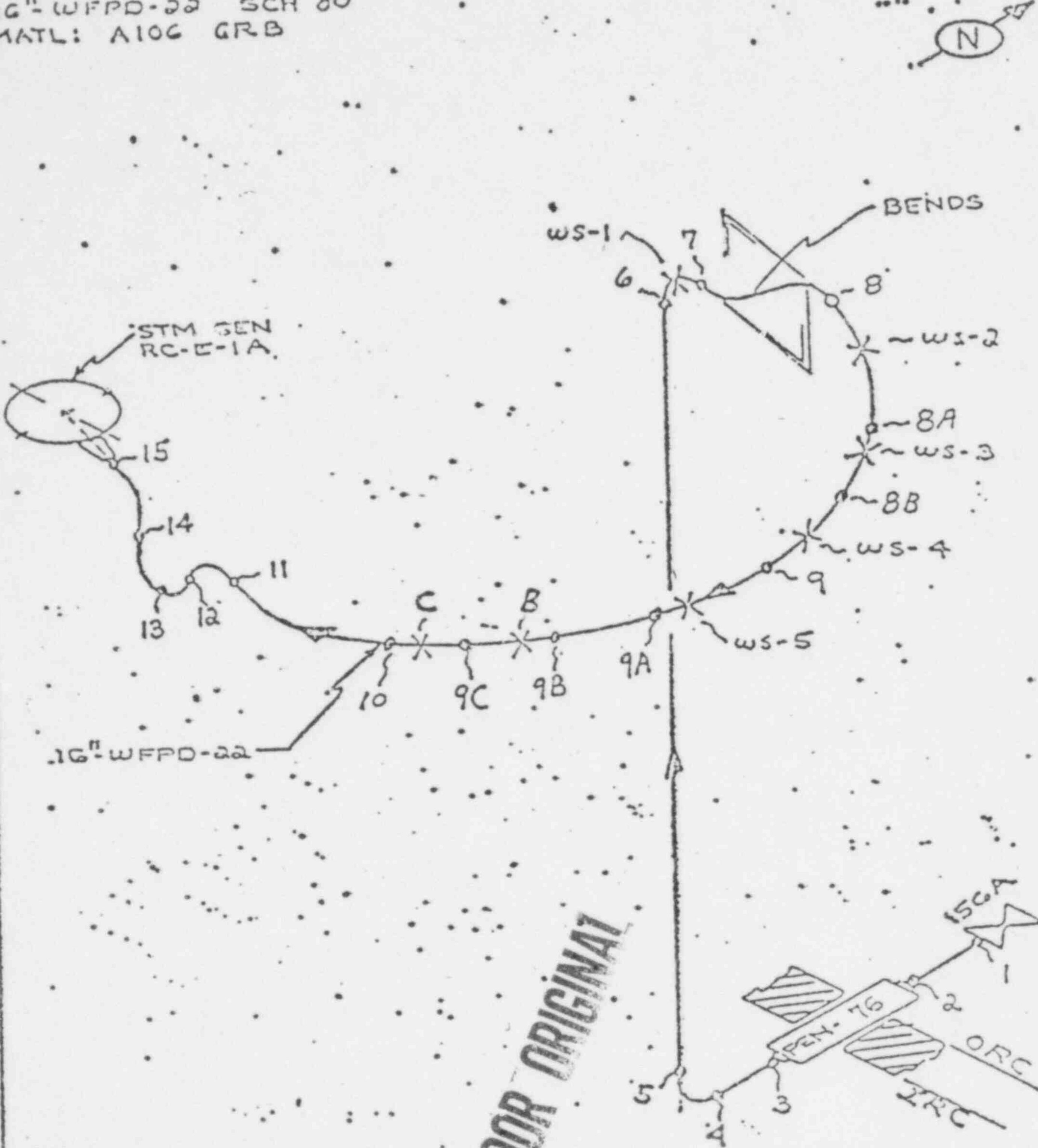
FEEDWATER

LOOP #1

16"-WFPD-22 SCH 80
MATL: A106 GRB

DLW-2-2210
REF DWG SWF-62



RESPONSE TO IE BULLETIN 79-13

ATTACHMENT #1

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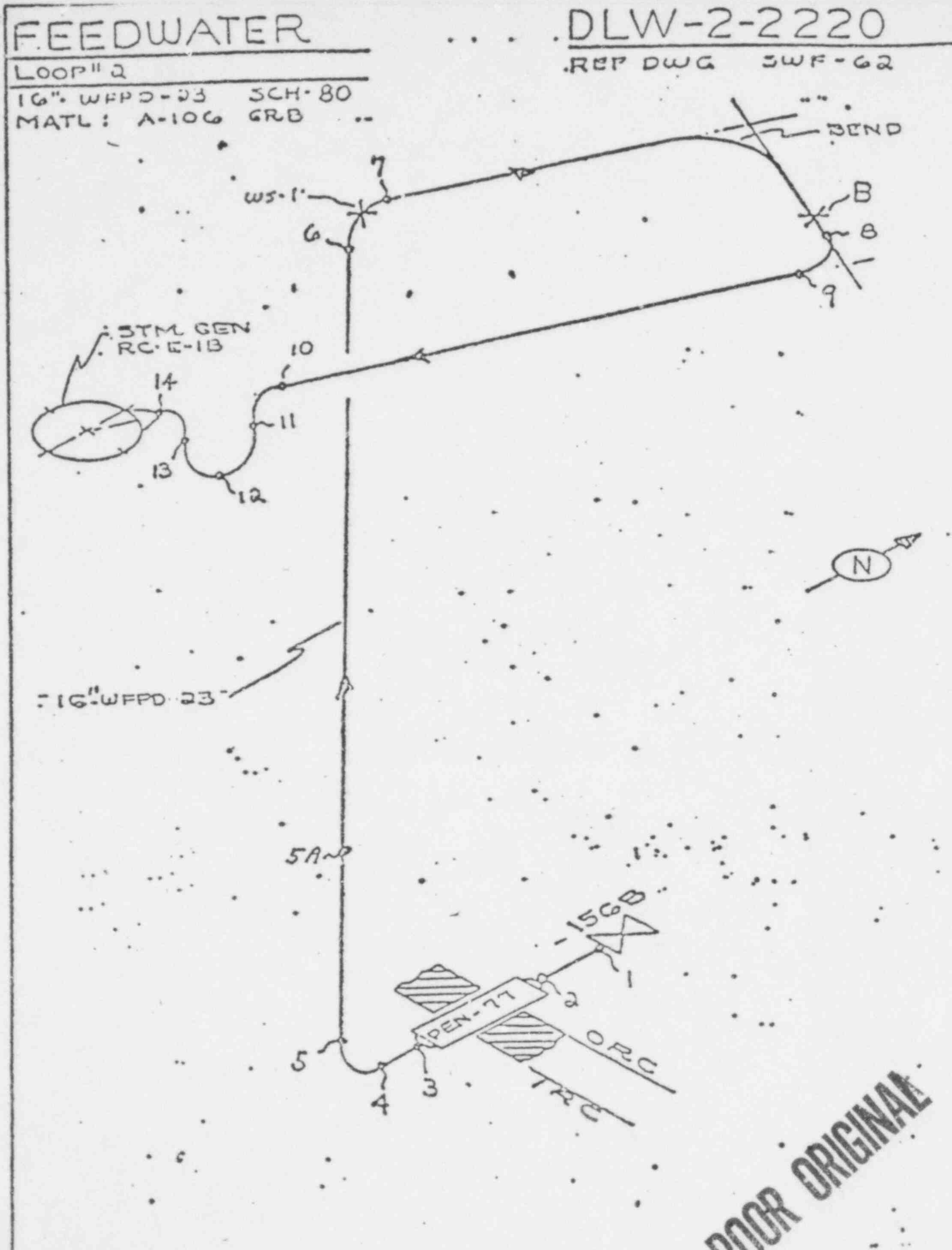

RESPONSE TO IE BULLETIN 79-13

ATTACHMENT #1

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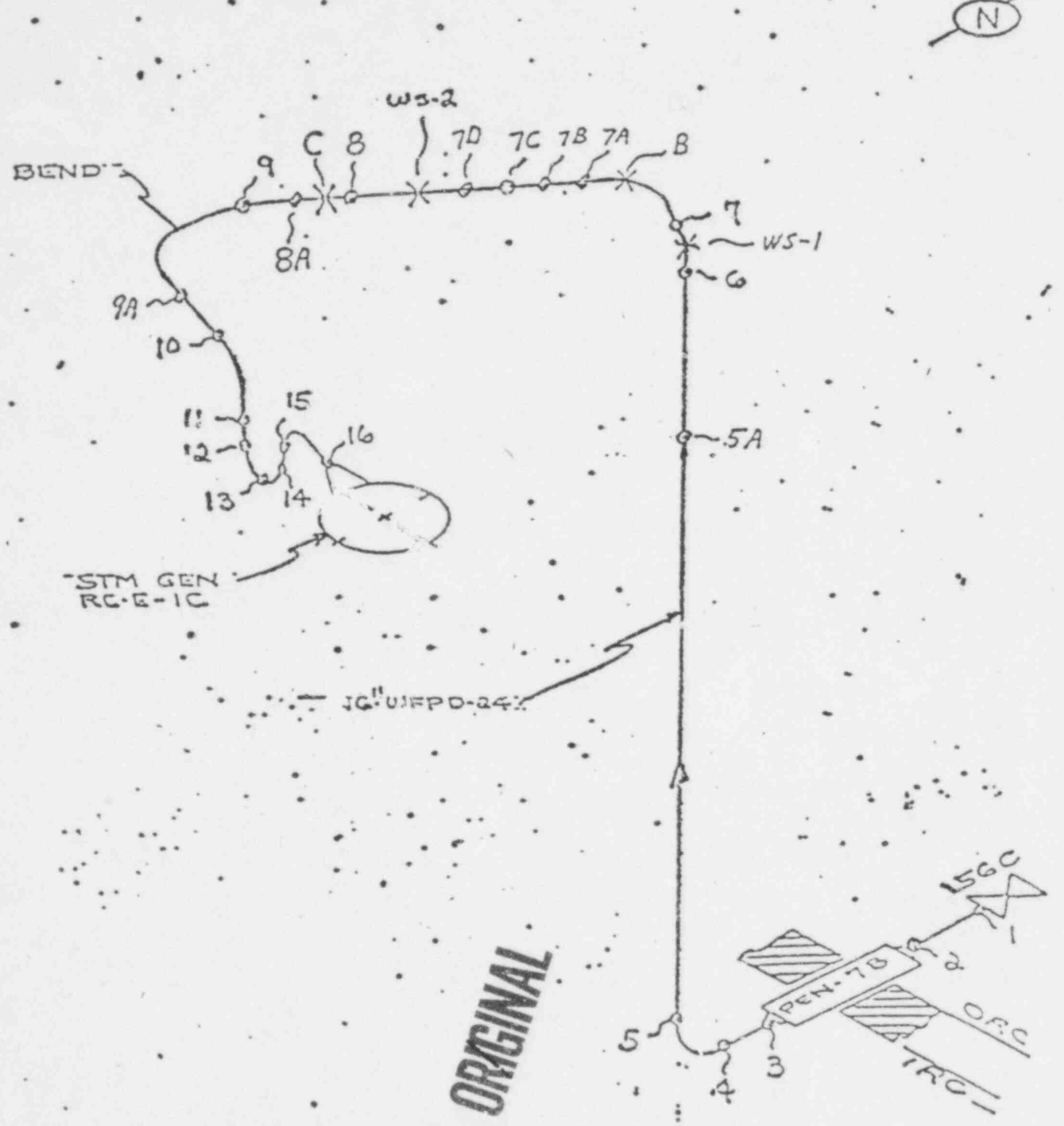
FEEDWATER

DI-W 22230

LOOP #3

REF. DWG SWF-62

16"-WFPD-24 SCH. 80
MATL: A-106 GR B



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RESPONSE TO IE BULLETIN 79-13

ATTACHMENT #1

BEAVER VALLEY UNIT #1

LOOP C - METALLURGICAL EVALUATION

A complete ring section, identified as being from Loop C - Beaver Valley Unit 1, was examined at Westinghouse R&D Center by PT, UT, Standard Metallography and both Scanning and Transmission Electron Microscopy. In addition, the corrosion products on the fracture surface was analyzed by EDAX (Energy Dispersion Analysis by X-Ray). The ring section was examined ultrasonically from the cut face through the weldment. The deepest penetration was estimated at the 9:00 position, to be 0.40 inches (0.25 inches from OD). Metallography confirmed that the deepest crack penetration was approximately 0.40 inches.

The deepest crack penetration was at a section change in the feedwater elbow where the Schedule 80 fitting had been counterbored to match the Schedule 60 nozzle. There was only one major crack extending from the counterbore, as was the case with D. C. Cook Units 1 and 2, and H. B. Robinson Unit 2. The depth of cracks at other locations around the periphery were 0.08 inches at 0°, 0.04 inches at 62°, and 0.02 inches at 248°. Cracks were not detected during metallography evaluation at 124° and 186°. The microscopic tests revealed beachmarks. Although they were extremely hard to identify as straitions, preliminary assessment indicates straition spacing of the order of 1 to 3 micro-inches. Preliminary conclusion is that the probable cause was "HIGH CYCLE CORROSION ASSISTED FATIGUE."

EDAX of the as-received fracture surfaces indicated the presence of iron, copper, aluminum, silicon, and sodium.

Chemical analysis of the reducer material showed that it was within specification for A106 Grade B steel.

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STRESS ANALYSIS OF THE DLW FEEDWATER PIPING

Stress analysis was performed on the DLW feedwater line configuration in an effort to determine the mechanism causing the observed cracking. This analysis was broken into three parts:

(1) Structural analysis of the feedwater line including the effects of thermal, deadweight and pressure.

(2) 2D finite element fatigue analysis of the feedwater nozzle/elbow configuration.

(3) Frequency analyses of the feedwater line.

The structural analysis was performed using a 3D finite element model of the feedwater line with anchors included at the steam generator (SG) and containment penetration and the vertical and horizontal thermal growth of the SG applied at the feedwater nozzle. The WESTDYN7 computer code was used for the analysis. The geometry consists of the feedwater nozzle with a 16" Schedule 60 end prep which is connected to a 16" Schedule 80 short radius elbow and then to three additional S.R. elbows to form a loop seal. The pipe runs horizontally from the loop seal, varying from ~ 35' (Loop B) to ~ 120' (Loop A), to a vertical drop of ~ 21' then horizontally ~ 8' to the containment penetration. Supports consist of deadweight spring hangers, several vertical rigid supports and horizontal snubbers.

Two thermal conditions were run. The first with the SG at ~ 550°F and the feedwater line at ~ 450°F representing normal operation. The second with the SG at ~ 550°F and the feedwater line cold representing the hot shutdown condition. The analyses results show a maximum thermal stress of approximately 12 ksi at the nozzle to pipe junction. The maximum deadweight and pressure stresses

were 2.6 ksi and 4.7 ksi respectively. These stresses are well below code allowable values. The attached sketch shows the basic geometry and the stresses in the feedwater piping.

The second analysis performed was a detailed 2D finite element fatigue analysis of the most severe thermal transient, which occurred during hot shutdown, in the region of the feedwater nozzle to short radius elbow junction. The analysis used the WECAN computer code and the rules of ASME Section III, NB-3200. The model used constant strain quadralateral elements with a minimum of 8 nodes through the wall and ran from the SG shell to 10" beyond the nozzle to elbow weld. The transient analyzed consisted of a ramp change in temperature from ~ 550°F to 60°F in 9 seconds followed followed by a period of constant 60°F operation, flow velocity .38 ft/sec. This represents the injection of auxiliary feedwater into the feedwater nozzle/elbow junction, which has been heated by the SG during the hot standby condition. When the auxiliary feedwater is terminated a step change in temperature is assumed from 60°F to ~550°F. This represents the conservative assumption of a leaking check valve in the feedwater system, which allows water to flow from the SG to the feedwater line and assumes no mixing of auxiliary feedwater with water in the SG. The maximum peak stress range obtained from this transient was 60 ksi which is then multiplied by a conservative factor of 1.7 to account for the detailed affect of the "Notch" at the elbow counter bore. This peak stress range of 102 ksi yields an allowable 5000 cycles using the ASME Section III S/N curves. The design transients given in the SG E-Spec has shown acceptable values of usage factors for the feedwater nozzle. Correspondingly, analysis of the nozzle to elbow junction will have an acceptable value of usage factor since the thermal transient stresses are lower at this junction than in the nozzle.

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The final analysis performed was a frequency analysis of the feedwater line. These frequencies were calculated without the stiffness effects of snubbers since snubbers could be inactive for the low magnitude of vibration expected during normal plant operation. The attached table summarizes the line frequencies less than 10 Hz. These frequencies are in the same range as those found for the SG in the reactor coolant loop of other Westinghouse plants. W testing of other plants has shown that the SG vibrates in its fundamental modes due to flow in the reactor coolant loop. This yields the possibility the feedwater line could be in resonance with the SG and could cause high enough stress in the nozzle to elbow junction to cause the observed cracks.

SUMMARY OF FEEDWATER LINE FREQUENCIES < 10 HZ

LINE	FREQUENCIES (HZ)
Loop A	1.6, 1.8, 3.1, 3.6, 5.1, 9.0
Loop B	2.7, 4.5, 5.0
Loop C	1.7, 2.2, 3.2, 4.7, 6.5, 8.3

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ATTACHMENT #3

MECHANICS & MATERIALS

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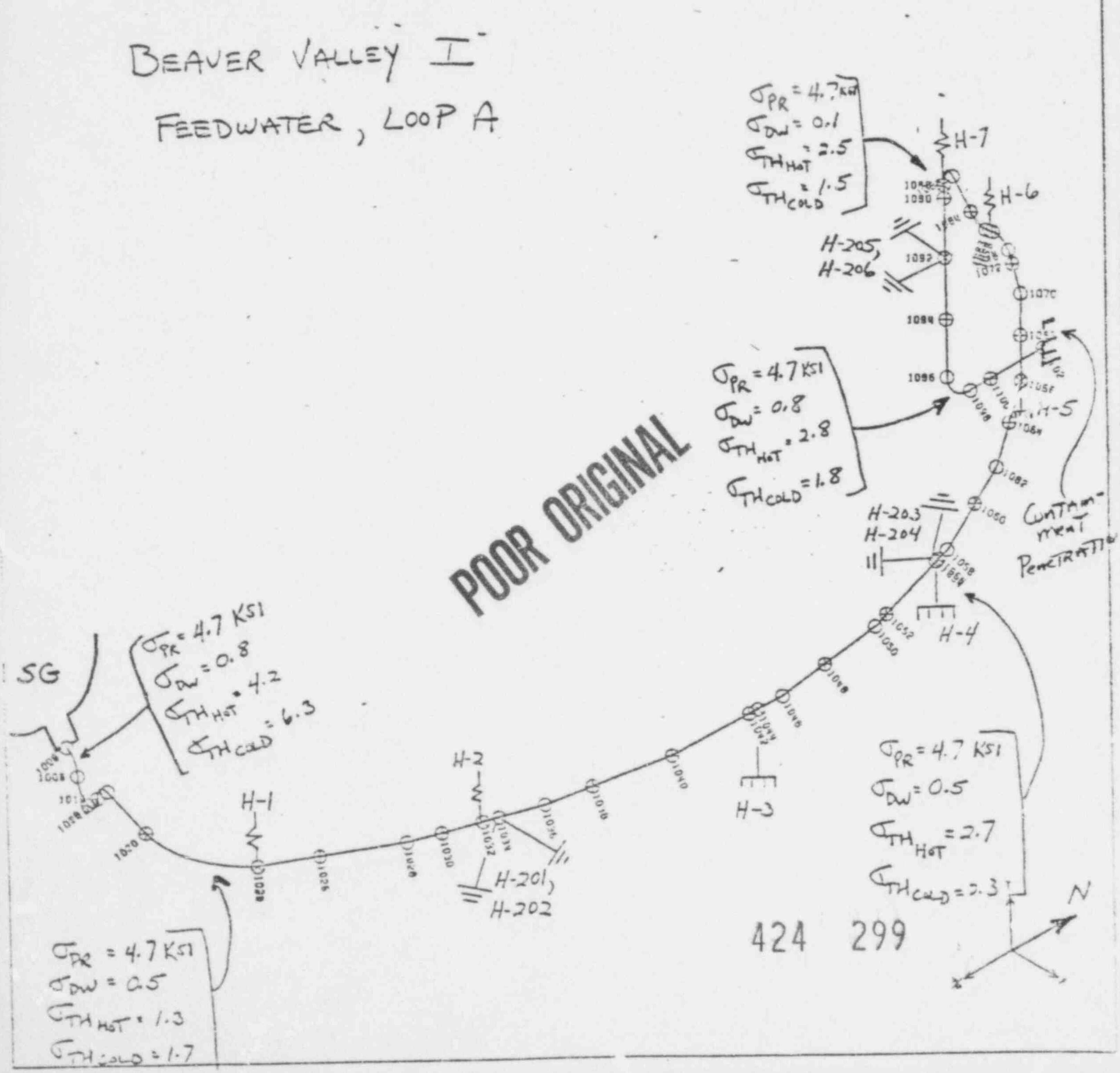

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MECHANICS & MATERIALS

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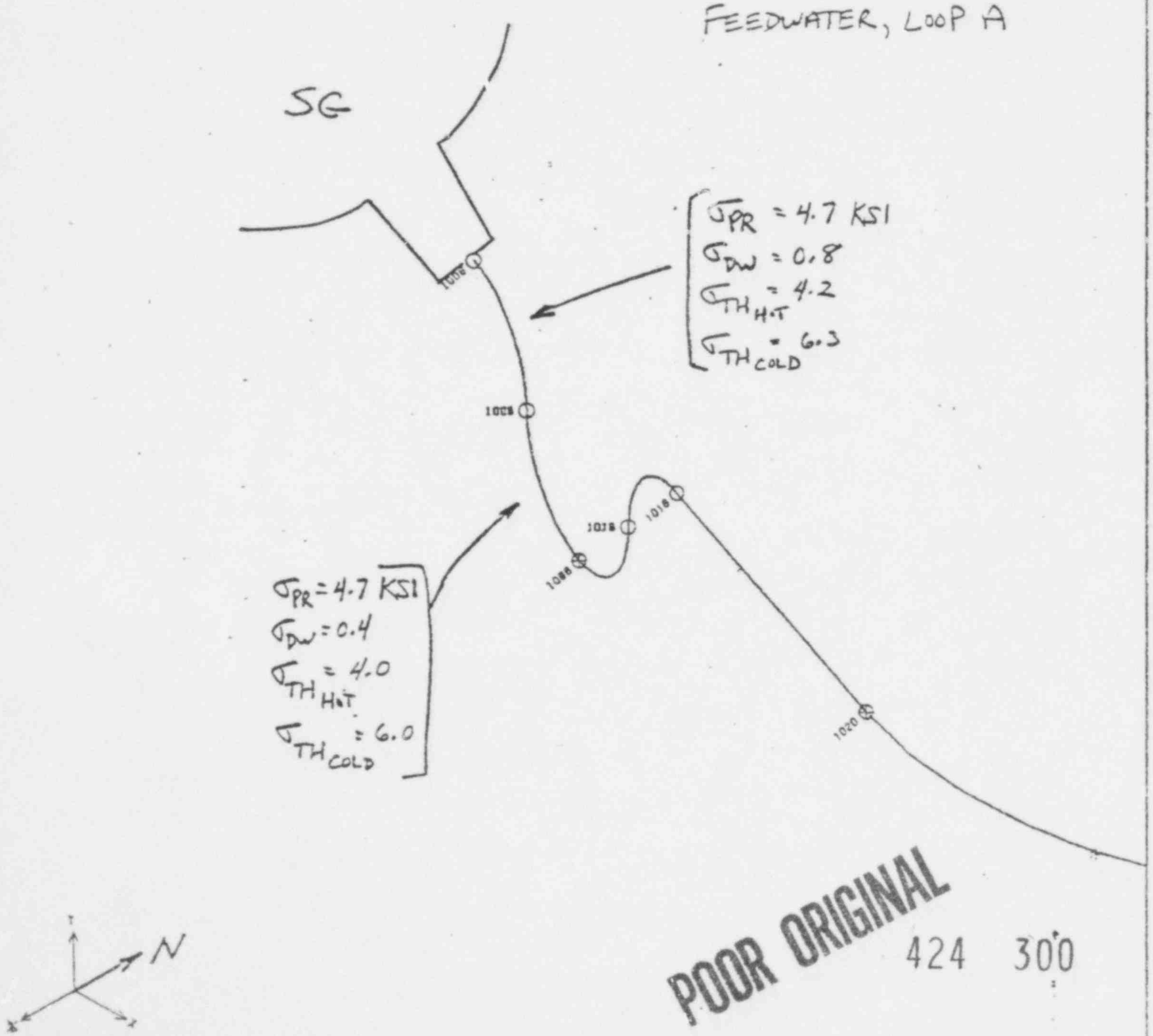
BEAVER VALLEY I
FEEDWATER, LOOP A



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MECHANICS & MATERIALS

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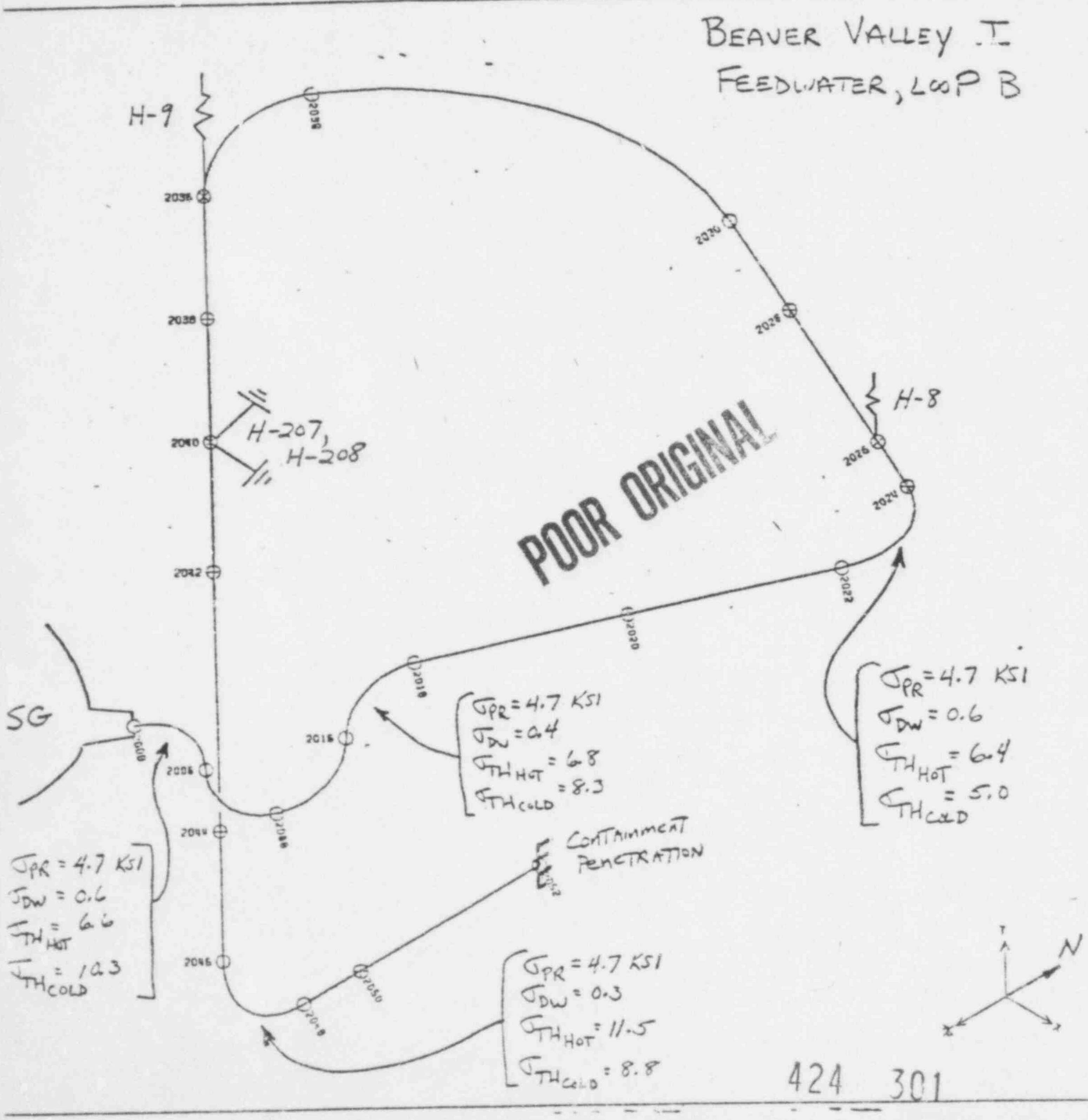

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MECHANICS & MATERIALS

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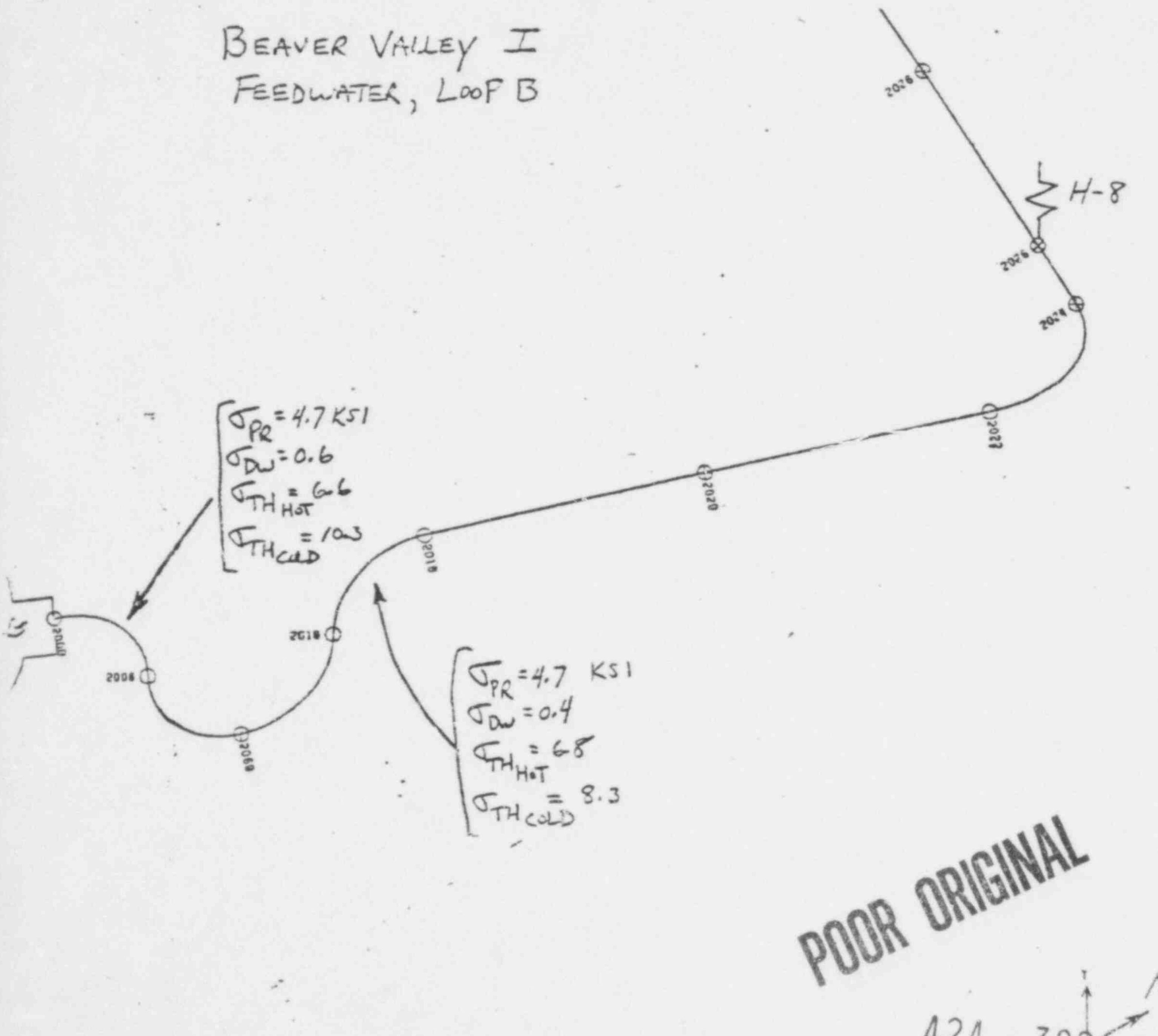
BEAVER VALLEY I
FEEDWATER, LOOP B



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ATTACHMENT #3

MECHANICS & MATERIALS

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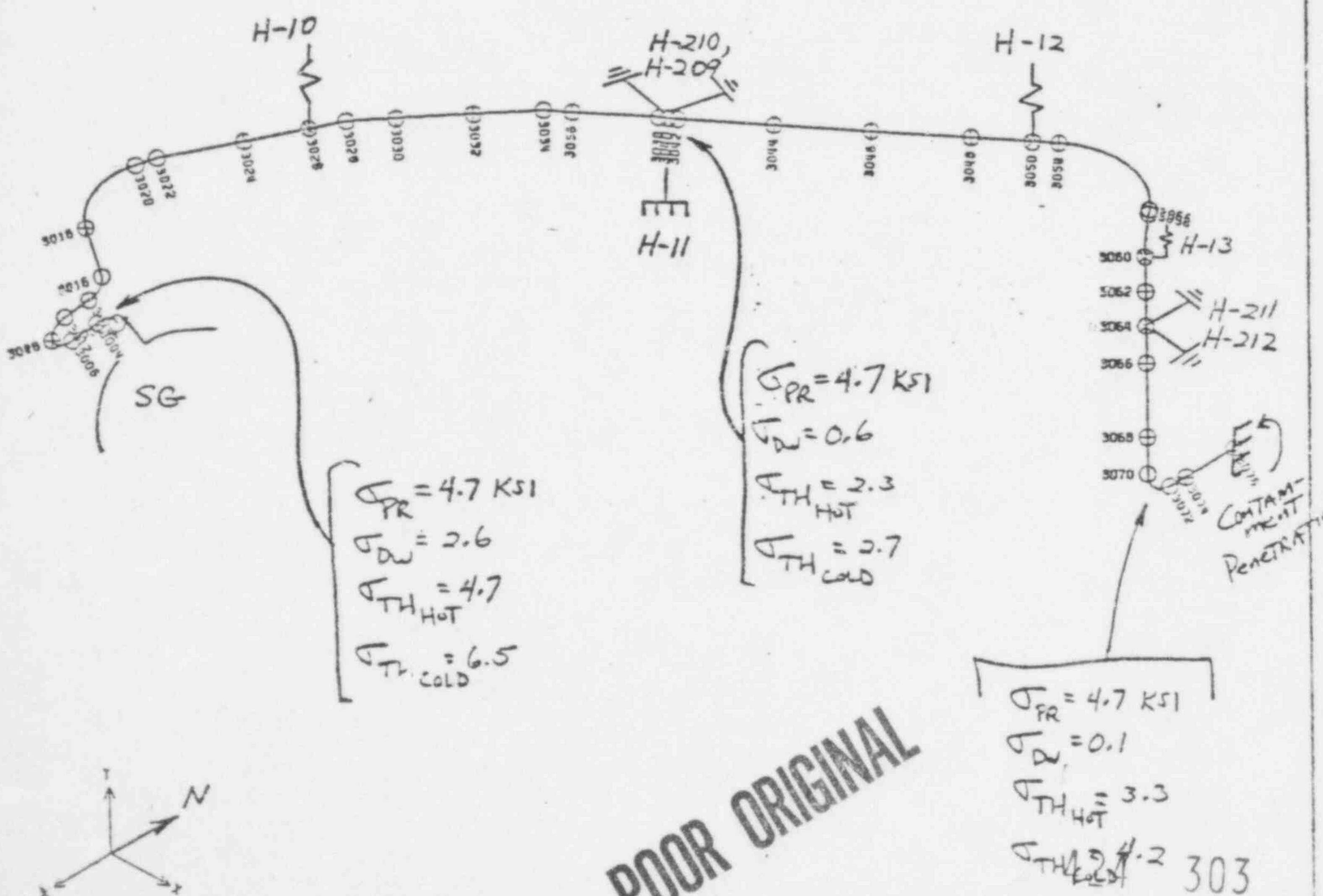
FILE USE COUNT 9　　CHECKED BY: D. R. Elb

FILE NUMBER:　　CHECKED BY:

BEAVER VALLEY I -
FEEDWATER, LOOP C



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MECHANICS & MATERIALS

FILE: BVFWLPC DATE: 7/ 5/79 PLOT 3

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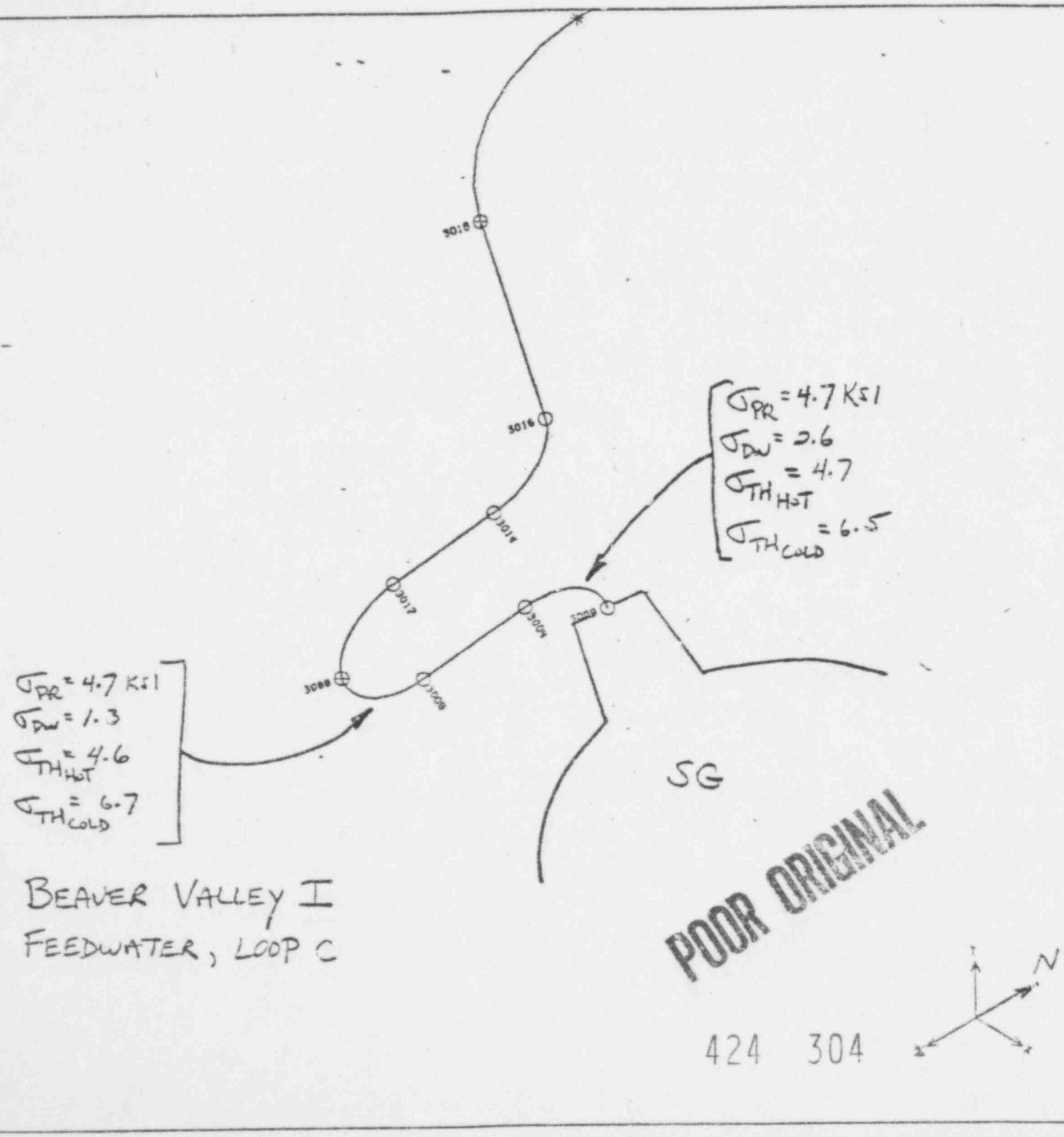
FILE NUMBER: CHECKED BY:



BEAVER VALLEY I
FEEDWATER, LOOP C

POOR ORIGINAL

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