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50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
SHALLER,D.V. Indiana & Michigan Power Co.
RECIP.NAME RECIPIENT AFFILIATION
Region 3, Chicago, Office of the Director

SUBJECT: LER 79-019/01L-0 on 790519:cracks found in 16-inch feedwater elbows adjacent to nozzle welds caused buildup of non-radioactive water in containment sump.16-inch elbows in generators being replaced.Causes being determined.

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0	1	REPORT SOURCE L (6)	0500316 (7)	051979 (8)	060879 (9)
			DOCKET NUMBER	EVENT DATE	REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 SEE ATTACHED LETTER

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0 9	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
	FW (11)	B (12)	A (13)	PIPEXX (14)	D (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
79	—	019	/	01	T	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
A (18)	A (19)	A (20)	A (21)	0336	Y (23)	Y (24)	A (25)	T335 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 SEE ATTACHED LETTER

1 1

1 2

1 3

1 4

1 5	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
	E (28)	100 (29)	N/A	A (31)	OPERATOR OBSERVATION

1 6	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
	Z (33)	Z (34)	N/A	N/A

PERSONNEL EXPOSURES

1 7	NUMBER	TYPE	DESCRIPTION (39)
	000 (37)	(38)	N/A

PERSONNEL INJURIES

1 8	NUMBER	DESCRIPTION (41)
	000 (40)	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

1 9	TYPE	DESCRIPTION
	L (42)	SMALL LEAKS IN FEEDWATER PIPING TO STEAM GENERATORS

PUBLICITY

2 0	ISSUED	DESCRIPTION (45)	NRC USE ONLY
	(44)		

NAME OF PREPARER D.V. SHALLER PHONE: (616)465-5901-Ext 311

JN 11 1979

7906150334

Examination of the elbows has indicated the cracks initiated at the top of the inner surface of the elbow at a machined discontinuity required for the weld-end preparation.

Radiographic examinations of the nozzle/feedwater line weld regions for Unit 2 steam generators Nos. 2 and 3 showed indication of cracks in the same area of the elbows. The same areas in Unit 1 were subsequently radiographed and indications were found in the 16-inch elbows of the feedwater lines to steam generators No. 2, 3 and 4. This discovery of crack indications was promptly reported to Region III. We were not able to detect the presence of indications in the 16-inch elbow to steam generator No. 1 in Unit 1 using radiographic techniques because of the unique conditions of the backing ring. Subsequent liquid penetrant examination indicated cracking in this elbow.

The wall thickness of the 16-inch elbow is approximately 50 percent greater than that of the steam generator nozzle. To compensate for this difference in wall thickness, the weld preparation included machining and beveling a step on the inside of the elbow to match the thickness of the steam generator nozzle at the point of the weld. Stresses concentrated in this thinner section adjacent to the weld and were magnified at the point of discontinuity.

The highest design stress point in the feedwater line inside the containment is at the 90° 14-inch elbows at the bottom of the vertical riser. The 14-inch elbows are in the same vertical plane as the affected 16-inch elbows. Five of the 14-inch elbows were radiographed. No indications were found.

Samples of the cracked areas of the 16-inch elbows were sent to two laboratories for metallurgical examination. Reports from the metallurgical examinations showed that the crack propagation was caused by high cycle fatigue assisted by corrosion. Further details of the metallurgical examination are described in Attachment 2.

After removal of the 16-inch elbows, the inside of the Unit 2 steam generator nozzles were examined by the liquid penetrant method. Light pitting and intermittent linear indications were noted on the nozzle counterbores. Subsequent visual inspection of the Unit 1 nozzles showed similar pitting in the nozzle counterbore region.

We are replacing the 16-inch elbows to all steam generators in Units 1 and 2. These new elbows do not have the sharp discontinuities on the pipe inner surface, and the strength of the elbow in the weld area has been increased. The pits and indications on the steam generator nozzles

are being blended out. The stress concentration at the nozzle counterbore is being reduced with the addition of a 1/2-inch radius fillet.

We are implementing a program to determine the characteristics of the high cycle fatigue. This program includes the installation of strain, temperature, pressure and vibration instrumentation at significant points on the piping system of two feedwater lines in Unit 2. Details of the new 16-inch elbow, the nozzle repairs and instrumentation are in Attachment 5.

The safety implications of this event at Cook Nuclear Plant have been carefully analyzed by our safety review committees. It was concluded that the condition identified did not adversely affect the health and safety of the general public. It was concluded that the return of the units to power with the newly designed elbows will have no adverse effect on the general public. Operation of Unit 2 with the instrumentation program will indicate whether there is any need for further corrective action. A safety evaluation is given in Attachment 6.

We expect to begin startup of Unit 2 by June 13, 1979 and Unit 1 by June 20, 1979.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

JED kb
Attachments

cc: R. C. Callen
G. Charnoff
R. S. Hunter
R. W. Jurgensen
N. C. Moseley, NRC
D. V. Shaller, Bridgman

bc: S. J. Milioti/J. I. Castresana/K. J. Vehstedt
R. F. Hering/S. H. Steinhart/J. A. Kobyra
A. S. Grimes/ J. J. Markowsky
H. N. Scherer, Jr.
T. W. Baker
R. F. Dodd
R. F. Kroeger
J. F. Stietzel - Bridgman
B. A. Svensson/E. A. Smarrella - Bridgman
J. G. Del Percio/J. F. Etzweiler - For CCL Update
W. W. Lowe
DC-N-6015.4
AEP:NRC:00216 Packet
D. Wigginton - NRC
NRC Site Inspector - Bridgman

ATTACHMENTS TO AEP:NRC:00216

ATTACHMENT NO.	TOPIC/DESCRIPTION
1.0	General Information
2.0.	Metallurgical Evaluation of Main Feedwater Line Cracks
3.0	Stress Analyses of Main Feedwater Lines Inside Containment
4.0	Feedwater Chemistry
5.0	Corrective Actions
6.0	Safety Evaluation

1.0 GENERAL INFORMATION

The cracks in the 16-inch feedwater elbows initiated from the inner wall outside of the heat affected zone of the weld at the discontinuity of the weld end preparation counterbore. Through-wall cracks existed in the 16-inch elbows to Steam Generators Nos. 1 and 4 of Unit 2. The most severe through-wall crack occurred in S.G. No. 1 and was 3½ inches in length on the outer surface. Cracks in the other elbows were smaller.

After the feedwater elbows were removed, liquid penetrant examination of the steam generator nozzles was performed. Results of this examination have shown light pitting and intermittent linear indications along the circumference of the counterbore discontinuity on all eight steam generator nozzles. Figure 1 depicts the steam generator nozzle and feedwater elbow in elevation. Figure 2 is a full scale section of the nozzle to the elbow weld.

The feedwater piping between each Steam Generator and its containment penetration, the two anchor points of this line, was visually inspected in detail for interferences. The seismic hydraulic snubbers were checked for piston displacement and found functional. The constant force hanger was also inspected. No interferences were noted. Minor interferences noted by crushed insulation were found at the Crane Wall sleeve and at one of the pipe whip restraints. Pipe stress analyses were performed with simulated restraints at these locations. The resulting stresses did not increase significantly and were well within the allowable limits.

The highest design stress point in six of the feedwater piping systems was identified at the 90° elbows at the bottom of its vertical riser. The elbows in the same plane as the affected elbow in five of these six lines were radiographed and found acceptable. The other two lines have twin 45° elbows at this location and have lower stress levels.

Three specimens, one taken at the elbow crack to Steam Generators Nos. 1 and 3 of Unit 2 and Steam Generator No. 3 of Unit 1 as well as a complete ring from the elbow to Steam Generator No. 3 of Unit 2, were shipped to Westinghouse for detailed metallurgical examination. Splits of the specimens from elbows from Unit 2 Steam Generator No. 1 and Unit 1 Steam Generator No. 3 were sent to Chicago Service Laboratories for preliminary metallurgical examination. These results are addressed in Attachment 2.

2.0 METALLURGICAL EVALUATION OF MAIN FEEDWATER LINE CRACKS

Specimens were removed from the 16-inch elbows for metallurgical examination from Unit 2 steam generators (SG) 1 and 3, and Unit 1 SG 3. Specimens were from the 12 o'clock position where radiographic examination indicated cracking originated. Half of the specimens from Unit 2 SG 1 and Unit 1 SG 3 were examined by Chicago Service Laboratory. The other halves and the entire sample from Unit 2 SG. 3 were examined by Westinghouse.

Chicago Service and Westinghouse laboratories reported that the primary crack was at the change in section where the counterbore had been machined on the inside surface as part of the weld end preparation. Both laboratories reported several additional cracks on the inside surface near the primary crack. Both reported oxide in the cracks and pitting corrosion. The primary crack, and adjacent smaller cracks, were not in the weld or heat affected zone, but were about 1/2 inch away at the inside surface.

Westinghouse reported the main crack had multiple origins, was wide due to corrosion and blunted at the tip. Two cracks, one which was 1/2 inch deep, and one which was less than .040 inches deep, were opened for examination by Westinghouse. Electron microscope examination at the crack tips revealed fatigue striations. Visual examination of the fracture surface markings of the longer crack showed typical fatigue beach marks.

Presence of fatigue striations, beach marks and corrosion product in the crack indicated crack progression was by high cycle corrosion assisted fatigue. Fracture surface markings close to the inside surface, where the crack originated, were obscured by corrosion and positive identification of the crack initiation mechanism was not possible.

Westinghouse received, in addition to the specimens for metallurgical examination, a four inch ring section from the 16-inch elbow from Unit 2 SG 3. Ultrasonic examination from the end showed four cracks, one from 11 o'clock to 1 o'clock and almost through the wall, a second crack a quarter through the wall from 2 o'clock to 4 o'clock, a third a quarter through the wall from 8 o'clock to 10 o'clock and a barely discernable crack around 6 o'clock. Subsequent inspection after sectioning showed that the second and third cracks were much smaller than the ultrasonic examination indicated.

The 16-inch feedwater elbow is schedule 80 SA234, WPB steel. Chemical and metallurgical examination by Chicago Service Laboratory showed the material to be acceptable. The steam generator nozzle, to which the elbow is welded, is schedule 60 SA508, Cl2 steel.

2.0 Metallurgical Evaluation of Main Feedwater Line Cracks (Cont'd)

Radiography detected cracks in all 16-inch elbows on Units 1 and 2 with the exception of Unit 1 SG 1. Cracks were more extensive and deeper on Unit 2 than Unit 1. Cracking, as detected by radiography, was in the upper half, and ranged from cracks that progressed more than half way around to those that were 3 or 4 inches in length. In each case cracks were evident at the 12 o'clock position.

3.0 STRESS ANALYSES OF MAIN FEEDWATER LINES INSIDE CONTAINMENT

Stress analysis was performed on the 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 stress analysis of the original feedwater nozzle/elbow configuration and the "as modified" elbow.

(3) Dynamic analyses of the feedwater line and steam generator.

The structural analysis was performed by Westinghouse 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 steam generator applied at the feedwater nozzle. The geometry consisted of the feedwater nozzle, which is connected to a 16" schedule 80 90° elbow followed by a reducer to a 14" schedule 80 pipe, and a 24 ft. vertical run. This pipe run was followed by four pipe segments running to the containment penetration. The supports for the line include a constant force hanger on the first horizontal segment, two locations of hydraulic snubbers attached to the pipe and several pipe whip restraints which do not touch the pipe.

Two thermal conditions were run. The first with the steam generator at 547°F and the feedwater line at 450°F representing normal operation. The second with the steam generator at 547°F and the feedwater line cold representing the hot shutdown condition. The analyses results show a maximum thermal stress of approximately 10 ksi at the first and second elbows from the steam generator. The maximum deadweight and pressure stresses were 1 ksi and 4 ksi, respectively. These stresses are well below code allowable values.

The second analysis performed was a detailed 2D finite element stress analysis of the most severe thermal transient, which occurred during hot shutdown, in the region of the feedwater nozzle to elbow junction. The analysis used the Franklin Institute Computer Codes FEETEMP and FEEAAS. The model used quadralaterial and triangular elements with a minimum of 8 nodes through the wall in the area of the failure and ran from the steam generator shell to 12" beyond the nozzle to elbow weld. The transient analyzed consisted of a ramp change in temperature from 540°F to 40°F in 9 seconds followed by a period of constant 40°F operation with a flow velocity of 0.38 ft/sec. This represents the injection of auxiliary feedwater into the feedwater nozzle/elbow junction, which has been heated by the steam generator during the hot standby condition.

3.0 Stress Analysis (Cont'd.)

The maximum stress obtained from this transient was 27.1 ksi which is then multiplied by a factor of 1.7 to account for the detailed effect of the discontinuity at the elbow counterbore. This peak stress of 46.1 ksi yields an approximate 7000 allowable cycles using the ASME Section III Figure I-9.1 S/N curves. The maximum number of cycles of the transient which could be seen is approximately 2000 assuming the auxiliary feedwater is turned on every 4 hours for an entire year. Donald C. Cook Unit 1 is presently in shutdown 189 and Unit 2 in shutdown 42. We have estimated 400 auxiliary feedwater injection cycles on Unit 2 for the operating year.

A similar detailed 2D finite element stress analysis was performed for the new elbow geometry during a similar severe thermal transient, i.e., the one which occurs during hot shutdown, in the region of the feedwater nozzle to elbow junction. The 2D model was similar except it provided for the thicker pipe wall adjacent to the weld and the modification to the counterbore. The maximum stress obtained from this transient for the modified elbow was 24.3 ksi which then multiplied by a factor of 1.2 to account for the detailed effect of the ½ inch radius fillet at the elbow counterbore. This peak stress of 29.2 ksi yields an allowable 28,000 cycles using the ASME Section III S/N curves. The stresses within the nozzle will likewise be reduced due to the inclusion of a ½ inch fillet to the nozzle counterbore. This reduction in stresses of approximately 29% at the nozzle counterbore will also significantly increase the cycle life of the nozzle.

The final analysis performed was a dynamic analysis of the feedwater line to determine frequencies and mode shapes. The lowest frequencies found were between 5.7 and 10.3 Hz. These frequencies are close to those found for the steam generator in the reactor coolant loop analysis. Westinghouse testing of other plants has shown that the steam generator vibrates in its fundamental modes due to flow in the reactor coolant loop. The possibility therefore exists that the feedwater line could be in resonance with the steam generator. This affect increases the possibility of additional vibratory cycles at the nozzle to elbow junction and might explain the propagation of the observed cracks. Final resolution of this possibility must await results of feedwater line instrumentation to determine if resonant vibration exists. This is necessary due to the degree of uncertainty in determining the feedwater line and steam generator frequencies and the closeness of these frequencies that must be demonstrated to show resonance.

4.0 FEEDWATER CHEMISTRY

It is suspected that dissolved oxygen may have been a contributing factor in the failures and cracking of the feedwater lines on Cook Units 1 and 2.

A thorough review of dissolved oxygen data for both the Cook Units has shown no extremely high dissolved oxygen levels at the feedwater inlets to the steam generators for any extensive period. Most of the time the dissolved oxygen values have been below 10 ppb. Rarely have they exceeded 30 ppb except during unit startups and similar cycle operations.

During unit startups, and hot stand-by, the auxiliary feedwater pump(s) which take suction directly from the condensate storage tank are used to supply makeup water to the steam generators. The condensate storage tank dissolved oxygen levels have generally ranged from 1 to 3 ppm.

While there is no concensus on the minimum concentration of dissolved oxygen that may contribute to this type of failure, AEP intends to reduce and/or maintain dissolved oxygen to a minimum.

5.0 CORRECTIVE ACTIONS

Eight new Schedule 80, 16", A234WPB, carbon steel feedwater line elbows were purchased for installation as replacements. Specific requirements were set forth to machine the elbow's weld end preparation to eliminate the previously existing discontinuity caused by the counterbore and to have a 125 RMS finish on transition. The transition from schedule 60 ID to schedule 80 ID is being made in a long ramp-like fashion with a 1/2" radius fillet as shown in Figures 3, 4 and 5. In addition, weld material has been added to the elbow's outside diameter in the vicinity of the counterbore to return the elbow to the original schedule 80 thickness. These design modifications will reduce the stresses in the area adjacent to the nozzle to elbow weld. The modified weld connection is designed to be made without backing rings.

The Steam Generator nozzles were inspected by radiography and magnetic particle techniques with negative results. After the elbows were removed on Unit 2, the inside surface of the nozzles were inspected using liquid penetrant. Light pitting with intermittant linear indications were found on the corresponding discontinuity of the nozzle counterbore. Further visual inspection at the second nozzle transition, where the thermal sleeve begins, also showed minor pitting. Westinghouse recommended grinding out the pits and the linear indications at the counterbore discontinuity. Minor pits at the thermal sleeve discontinuity are acceptable. The following criteria has been set forth by Westinghouse for the grinding of the feedwater nozzles without weld repair:

(1) 0.025" for the entire circumference.
(2) 0.050" for local areas.
(3) Depths within the limits of (1) and (2) above will be blended to approximately 1/2" radius at bottom of repair and tapered to a 4:1 slope.

Each steam generator nozzle counterbore transition from the weld is being prepared similar to the elbow transition, that is, a 1/2" radius fillet.

Test instrumentation will be installed on auxiliary and main feedwater piping and the associated steam generator feedwater nozzles for loops number one and three of Unit 2. The purpose of installing the test instrumentation is to collect information on steady state and transient conditions during normal plant operation from cold shutdown to 100% power for two steam generators and their associated feedwater piping. Information to be collected consists of motions, strains, pressures and temperatures.

Following is a brief summary of the number and location of the various types of test instrumentation located on each of the two instrumented steam generator-feedwater loops. Loops one and three will be instrumented identically with

5.0 Corrective Actions (Cont'd.)

respect to type, quantity and location of test instrumentation.

Strain Gages: Approximately twelve (12) strain gages will be utilized to obtain bending, torsional and axial stresses on the elbow and bending stresses on the nozzle.

Accelerometers: Approximately thirteen (13) accelerometers will be utilized to monitor possible oscillations of the steam generator and associated main feedwater piping within the containment.

Thermocouples: Approximately twenty-four (24) thermocouples will be used to measure; circumferential temperature distribution on the nozzle and associated elbow, axial temperature distribution along the elbow, and main feedwater and auxiliary feedwater temperatures.

Pressure Transducers: Approximately four (4) pressure transducers will be utilized to monitor possible water pressure oscillations within the elbow, and main feedwater and auxiliary feedwater piping.

Displacement Transducers: Approximately ten (10) displacement transducers will be used to measure main feedwater pipe displacement within the containment.

The test instrumentation described above will be located on two loops in order to have confirmation of monitored information. The information obtained from all test instrumentation will be recorded outside the containment on either magnetic tape or strip chart recorders.

A detailed program for data collection and evaluation is being developed. This will include checking the hot and cold positions of the feedwater piping inside containment as Unit 2 is taken from cold shutdown through hot standby. An evaluation of our findings will be submitted to the NRC before the end of the Unit 2 refueling outage.

The piping lines to Steam Generators 2-1 and 2-3 were chosen to be instrumented. Since all eight elbows developed cracks in the same location and the piping lines themselves are virtually identical, one line is sufficient to obtain meaningful data. Two lines are instrumented for redundancy and verification. Unit 2 was chosen because it will restart first.

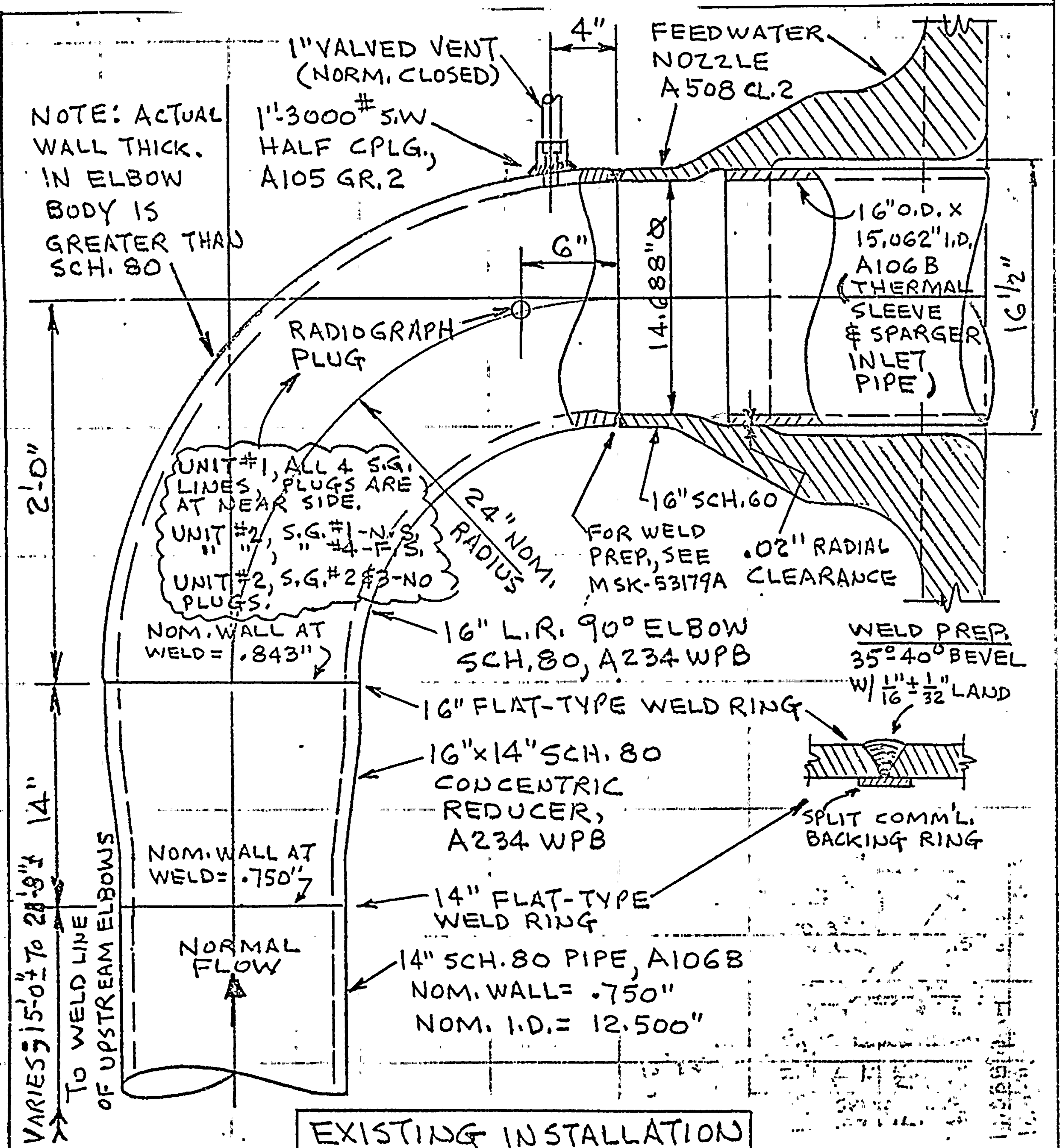
6.0 SAFETY EVALUATION

The cracks in the main feedwater elbows were detected by a normal leak detection procedure. The prompt shutdown of Unit 2 and the corrective actions being taken in both units eliminate the possibility of the subject event becoming a source of concern for the health and safety of the general public.

AEP initiated a thorough review of our safety analysis, specifically those sections relating to feedwater line ruptures, including the sudden circumferential severance of the feedwater elbow and the status of the Unit at the time of the incident. The emergency backup systems were fully capable of mitigating the consequences of a feedwater line break without adversely affecting the health and safety of the general public. Emergency operating procedures were available at the plant to cope with a feedwater line rupture. During the time of the feedwater line leakage and throughout the Unit 2 shutdown, all required auxiliary feedwater pumps and ECCS equipment were operable to mitigate the consequences of a feedwater line rupture.

Our offsite (AEPSC NSDRC) and onsite (PNSRC) committees have reviewed the feedwater elbow cracking event and have concluded that the "as found" feedwater system did not constitute an unreviewed question within the bounds of the safety analysis, and that it is safe to operate the Donald C. Cook Nuclear Plant with the installed modifications.

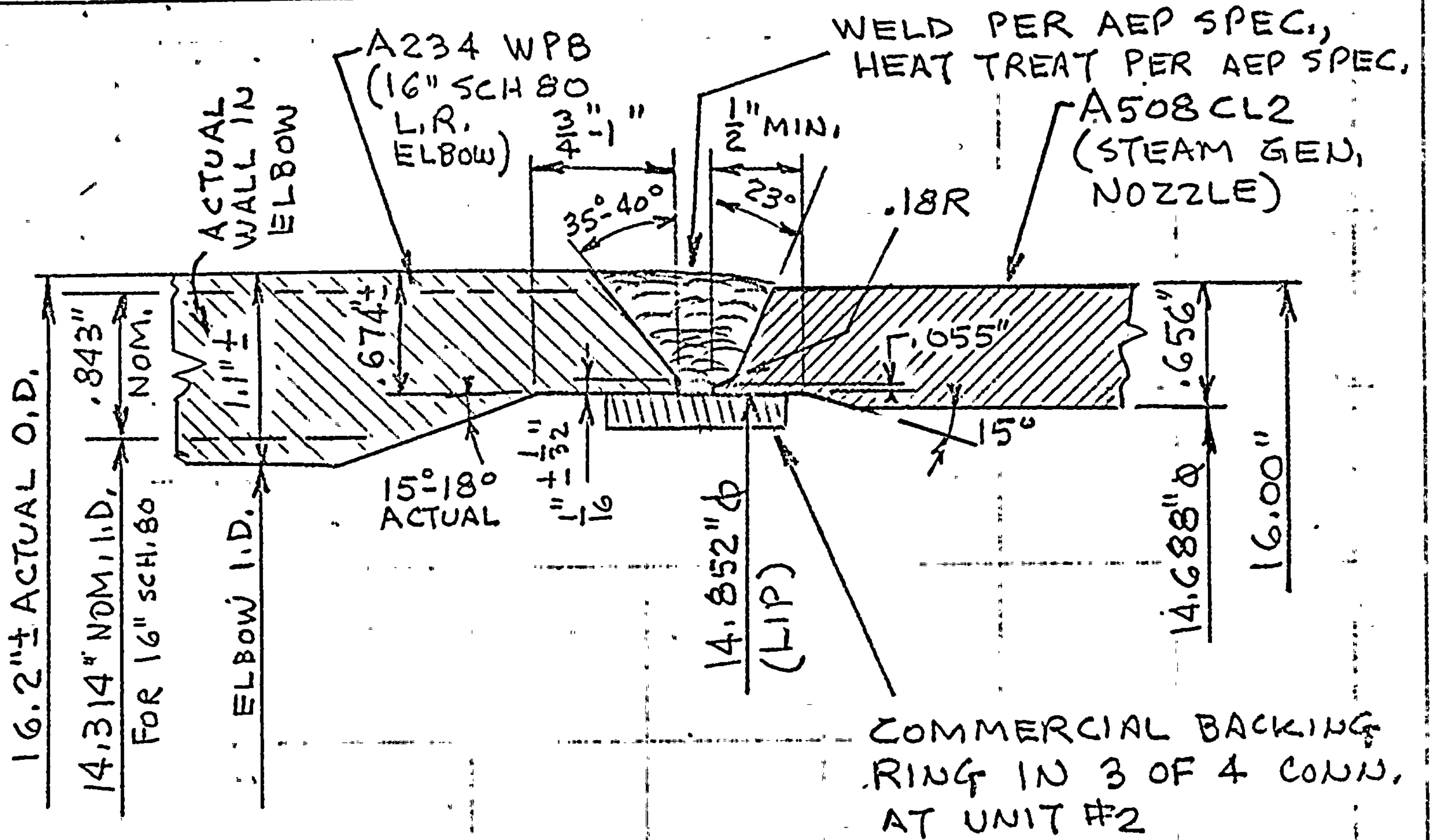
AEP:NRC:00216 FIGURE 1



ARRGT. OF FEEDWATER CONN. TO STEAM GEN. NOZZLE

PIPE SYSTEM FEEDWATER		DR. SH	PLANT D.C. COOK PLANT — SH. 1 OF 2
REF. DWGS.	REV. 2 6/4/79	CK. NB	MSK-53179 REV. № 2
		DATE 5/31/79	

AEP: NRC:00216 FIGURE 2



FULL SCALE SECTION OF EXISTING WELD BETWEEN 16" SCH. 80 L.R. ELBOW IN AEP FEEDWATER LINE AND 16" SCH 60 WELD CONNECTION OF W STEAM GENERATOR

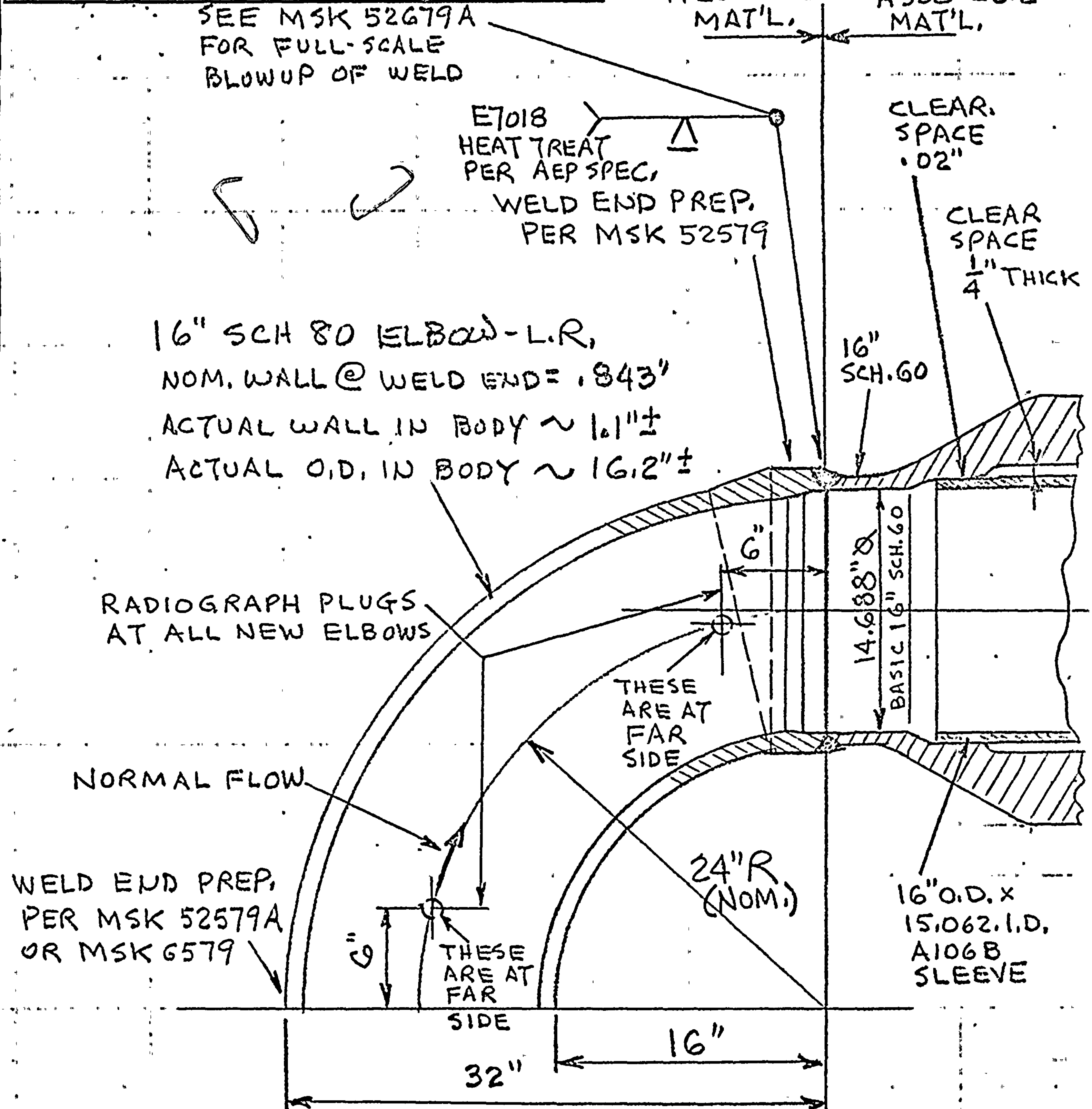
$t_{min.}$ PER ANSI B31.1 = .563" FOR A234 WPB STL.

$t_{min.}$ PER ANSI B31.1 = .425" FOR A508 CL.2 STL.

t_{WELD} = .674"±

PIPE SYSTEM FEEDWATER		DR. SH	PLANT D.C. COOK PLANT — SH. 2 OF 2
REF. DWGS.	REV. △1 6/4/79	CK. NB	MSK-53179A — REV. NO. △1
		DATE 5/31/79	

AEP:NRC:00216 FIGURE 3



D.C. COOK NUCLEAR PLANT

CROSS-SECTION AT FEEDWATER PIPE CONNECTION TO STEAM GENERATOR AS MODIFIED

SHEET 1 OF 2

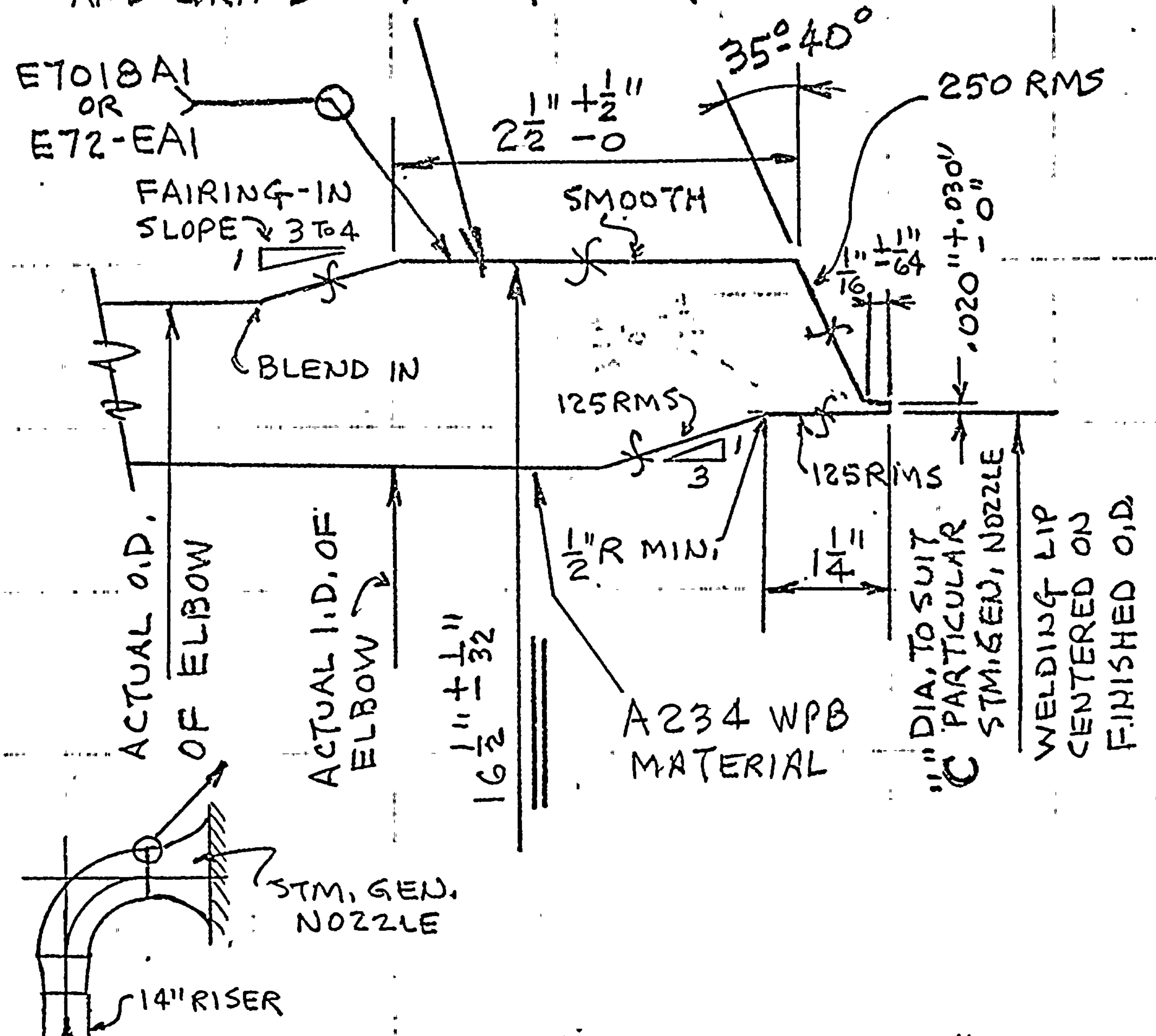
MSK 52679

REV. 1 6/4/79

AEP S. CORP
MECH. DESIGN
SH 5/26/79

AEP: NRC: 00216 FIGURE 4

FABRICATOR TO BUILD UP WITH WELD MATERIAL AND GRIND TO SUIT PRIOR TO WELD PREPARATION,

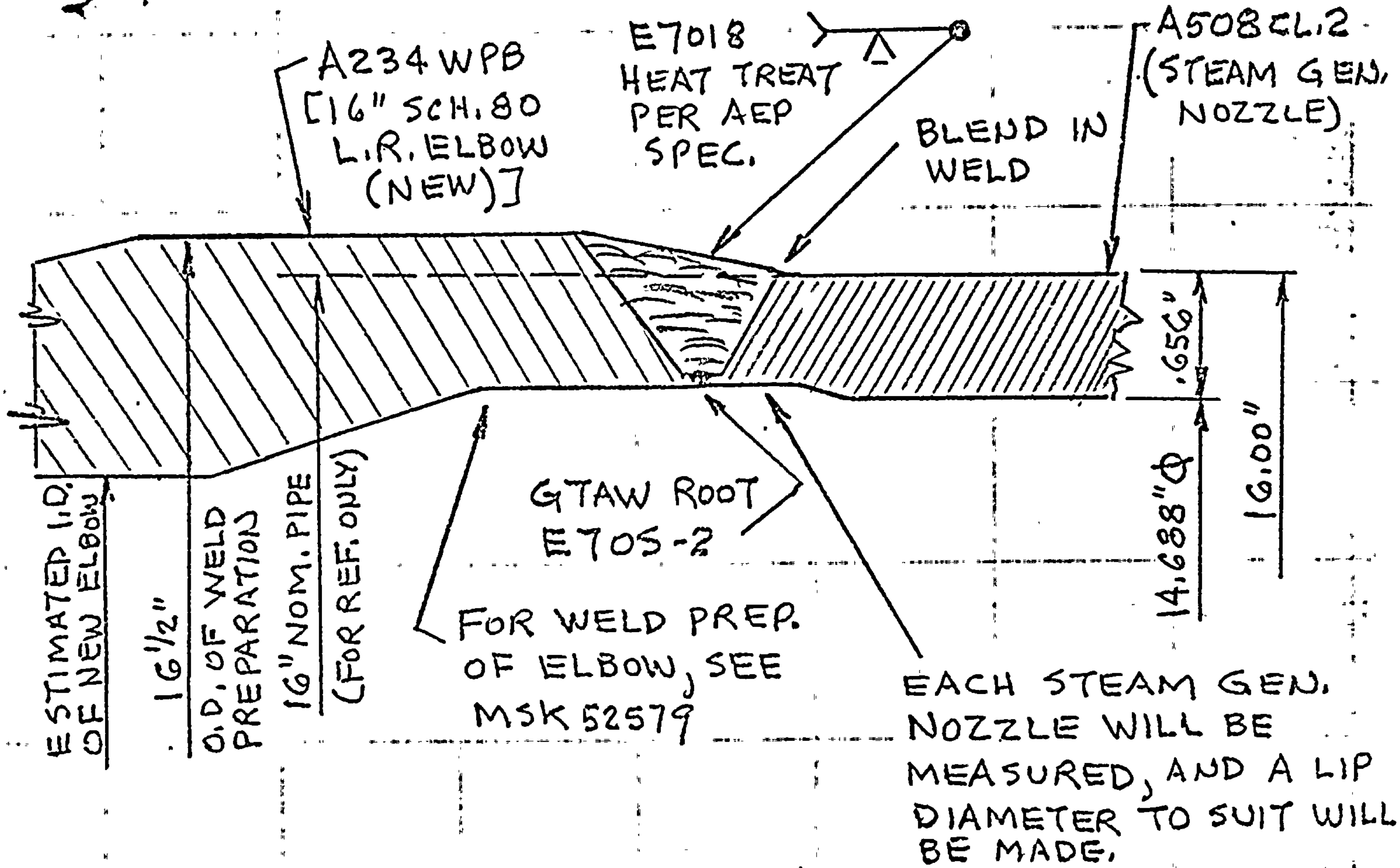


SPECIAL WELD PREP. FOR 16" L.R. ELBOW (NOMINAL SCH. 80) AT MATING CONNECTION TO 16" SCH. 60 STEAM GENERATOR NOZZLE; NEW INSTALLATION

PIPE SYSTEM FEEDWATER	DR. SU	PLANT D.C. COOK	SH. 1 OF 1
REF. DWGS. / REV. 3 6/4/79	CK. DP / DATE 5/25/79	MSK-52579	REV. No 3

AEP: NRC: 00216

FIGURE 5



FULL SCALE SECTION OF NEW WELD BETWEEN NEW 16" SCH. 80 L.R. ELBOW IN AEP FEEDWATER LINE AND 16" SCH. 60 WELD CONNECTION OF (W) STEAM GENERATOR

PIPE SYSTEM	FEEDWATER	DR. SU	PLANT D.C. COOK PLANT · SH. 2 OF 2
REF. DWGS.	REV. △1 6/4/79	CK. NB	MSK-52679A REV. № △1
		DATE 5/31/79	