TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

May 6, 1986

Director of Nuclear Reactor Regulation
Attention: Mr. B. Youngblood, Project Director
PWR Project Directorate No. 4
Division of Pressurized Water Reactors (PWR)
Licensing A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

| | | |
|---|---|---|
| In the Matter of | ) | Docket Nos. 50-327 |
| Tennessee Valley Authority | ) | 50-328 |

By the November 18, 1985 letter from E. Adensam to H. G. Parris, NRC transmitted questions regarding the Sequoyah Nuclear Plant (SQN) Inservice Inspection (ISI) Program. An initial response was submitted to NRC by my letter to you dated March 12, 1986. The enclosure provides information requested by question Nos. 7 and 8 of the November 18, 1985 letter.

If you have any additional questions regarding this subject please get in touch with Jerry Wills at FTS 858-2683.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

D. L. Lambert for

R. Gridley, Director
Nuclear Safety and Licensing

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. Carl Stahle
Sequoyah Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814



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ENCLOSURE

ADDITIONAL RESPONSE TO THE NOVEMBER 18, 1985
LETTER FROM E. ADENSAM TO H. G. PARRIS WHICH
REQUESTED INFORMATION REGARDING THE INSERVICE INSPECTION PROGRAM
FOR THE SEQUOYAH NUCLEAR PLANT
DOCKETS 50-327 AND 50-328

NRC ITEM 7

Relief Request No. ISI-4 states that one circumferential shell weld on each steam generator is inaccessible due to upper support brackets and that this weld will be examined on a "best effort" basis. The licensee provided drawings showing the support bracket causing interferences. However, the drawings are not of sufficient detail to ascertain the possible extent of examination. To enable evaluation of this relief request, the licensee should define the extent of the limited examinations that will be performed, including the volume of area, and provide dimensioned drawings with sufficient detail to enable the evaluation of the need for relief due to interference.

TVA RESPONSE TO ITEM 7

Attachment 1 provides a report detailing the physical clearances and estimated limitations to the steam generator ultrasonic examinations. This report also includes drawings which better define the physical dimensions in the steam generator lower cone-transition circumferential shell weld areas.

NRC ITEM 8

Relief Request No. ISI-5 states that the accessible areas of the lower head dollar weld will be examined to the extent possible by a remote ultrasonic method from the vessel inside diameter. The licensee should define the extent of the examination by giving the percentage of the weld to be examined.

TVA RESPONSE TO ITEM 8

Attachment 2 contains copies of Southwest Research Incorporated (SwRI) Supplemental Reports Projects 17-5339 and 17-6037, "Reactor Pressure Vessel Preservice Ultrasonic Examination Limitations at Sequoyah Nuclear Plant Unit 1," and "Unit 2," respectively. These reports detail the limitations encountered in performing an ultrasonic examination of the reactor vessel welds and the lower head dollar weld from the inside diameter.

Attachment 3 provides a health physics survey performed December 12, 1985, and a schematic representation of the reactor vessel lower head area showing the radiation doses encountered in the survey. The doses as shown are in units of millirem per hour (mr/hr) and measured on contact, at a distance of 18 inches, and in the general areas.

Example: window cover = 100/75-50
where: 100 = 100 mr/hr, on contact
75 = 75 mr/hr at 18 inches
50 = 50 mr/hr, general area

Other readings annotated GA are general area readings only. Smear survey locations are shown by circled numbers with the results tabulated in disintegrations per minutes per 100 square centimeters (DPM/100 $cm^2$). It should be noted that these survey readings are taken with the reactor vessel lower insulation in place. Attachment 3 also shows a table of the estimated man-hours involved in performing the examination from the outside diameter.

Based on attachment 3, the expected doses incurred during the examination would be conservatively estimated at 16.4 man-rem. Actual doses encountered during the examinations would have to be determined from readings taken close to the examination area with the insulation removed and would most likely have a factor of several times higher. In addition, these readings are highly dependent upon operational considerations, such as the position of incore instrumentation thimbles and the core power history which could result in locally high radiation doses from "crud-bursts." Also, area surface, particulate and airborne radiation hazards would most certainly increase with the removal of the lower vessel head area insulation. In short, the actual radiation doses incurred when performing the examination could not be easily determined without actually performing much of the work.

Justification for not performing the additional volumetric examination from the outside diameter, as it was performed during preservice examination, lies in the fact that the additional margin of safety obtained would not justify the increased man-rem exposure and would be contrary to any ALARA considerations. Additionally, the preservice examination from the outside diameter revealed no reportable indications, and a limited examination will be performed remotely on the inside diameter of the weld.

The inclusion of ISI-5 as a request for relief is in accordance with a verbal agreement reached with NRC representatives on documentation of preservice examination results in discussions about the Watts Bar Nuclear Plant Preservice Inspection Program. As required in IWA-2232, ultrasonic examination of class 1 vessel welds in ferritic material greater than two inches in thickness shall be conducted in accordance with Article 4 of Section V. Paragraph T-441 of Article 4 states merely that the limitation to the examination must be indicated in the report. It is our position that this limited inside diameter examination meets the requirement of the code.

ATTACHMENT 1

Sequoyah Nuclear Plant

Units 1 and 2

Ultrasonic Scan Limitations

For Steam Generator Welds

SGW-D1, D2, D3, and D4

1.0 Purposes

The purpose of this report is to describe the ultrasonic examination limitations described in SI-114.1 and SI-114.2 "In-Service Inspection Program for Tennessee Valley Authority Sequoyah Nuclear Plants 1 and 2," respectively for welds SGW-D1, D2, D3, and D4. The basis for relief is described in Appendix E, Request for Relief, ISI-4.

2.0 References

2.1 SI-114.1 and SI-114.2 "Inservice Inspection Program for Tennessee Valley Authority Sequoyah Nuclear Plants 1 and 2," respectively.

2.2 Drawing CH-M-2345-B, Sequoyah Nuclear Plant, units 1 and 2, Steam Generator.

2.3 Figures 1 and 2.

3.0 Description of Examination Limitations

SGW-D1 was examined during the unit 1, cycle 3 outage in accordance with TVA procedure N-UT-19. Procedure N-UT-19 is written to comply with the requirements for examination, evaluation, and recording of results in accordance with the 1977 edition, summer 1978 Addenda of Section XI of the ASME Boiler and Pressure Vessel code. Procedure N-UT-19 requires a maximum calibration of a 5/8 nodal response from the 3/4 t hole when visible. In the event the 5/8 node response is not visible, the DAC curve is to be extrapolated by extending the 3/4 t hole response to 0 percent full screen height at the tenth screen marker. A minimum calibration of one-half node is utilized to perform manual examination of vessel welds greater than 2 inches in thickness.

Steam generator welds SGW-D1, D2, D3, and D4 are partially inaccessible for ultrasonic examination because of a permanent support ring located at the edge of the weld. The steam generator weld allows access for ultrasonic examination from the cone side only.

The support ring is held in place by wedging (1/2" x 6" x 12") steel pads between the support ring and the steam generator. These pads are spaced at approximate 13 inch intervals 360 degrees around the support ring. Approximately 42 percent of the weld had a greater limitation because of steel pads resting on the edge of the weld (see Figure 1). The remaining 58 percent of the weld was not blocked by steel pads; however, the clearance between the vessel and support ring precluded achieving 100 percent of the required volume from one side (see Figure 2). It was determined that at least 68 percent of the examination volume was examined despite support ring interference.

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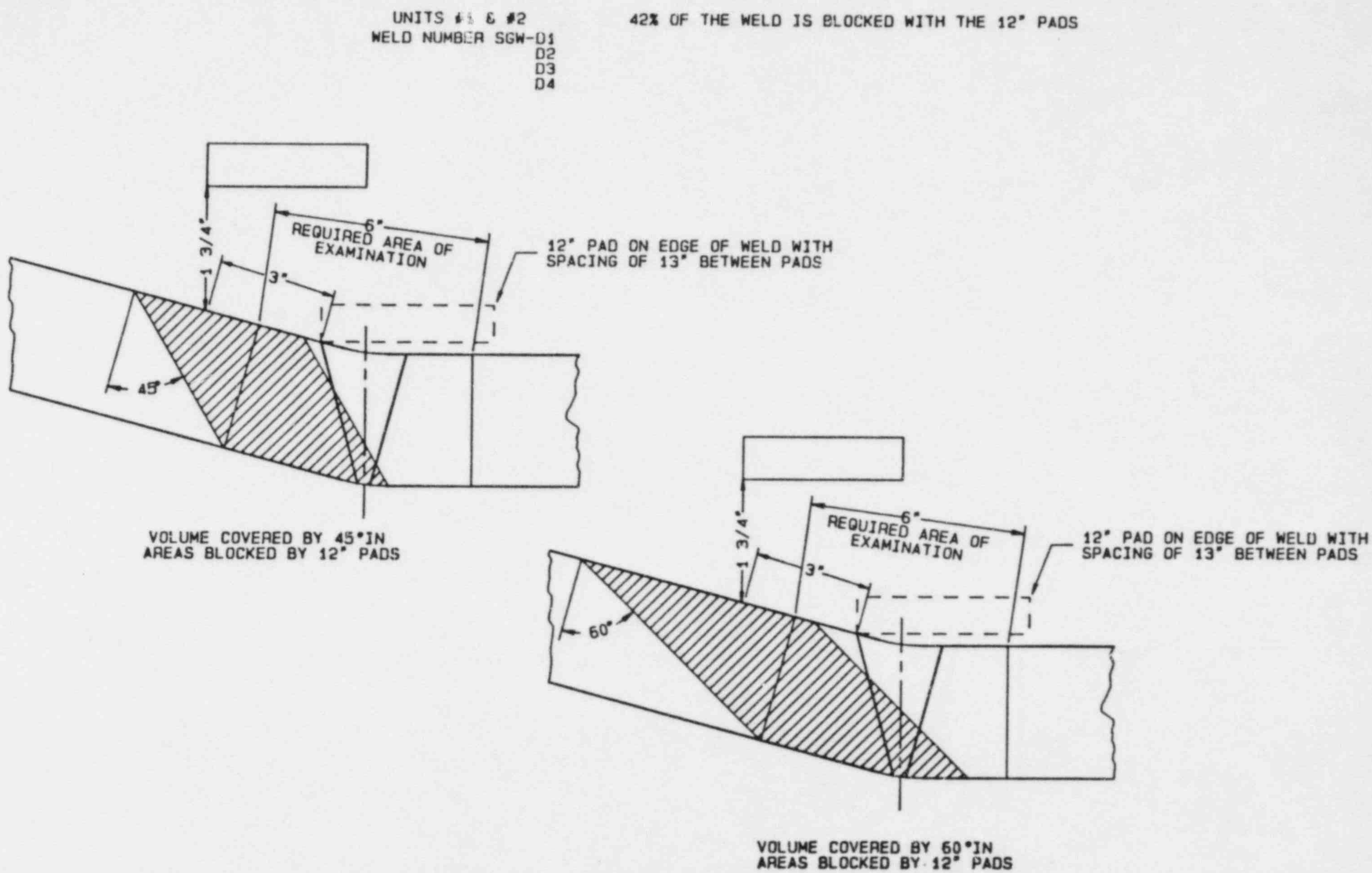
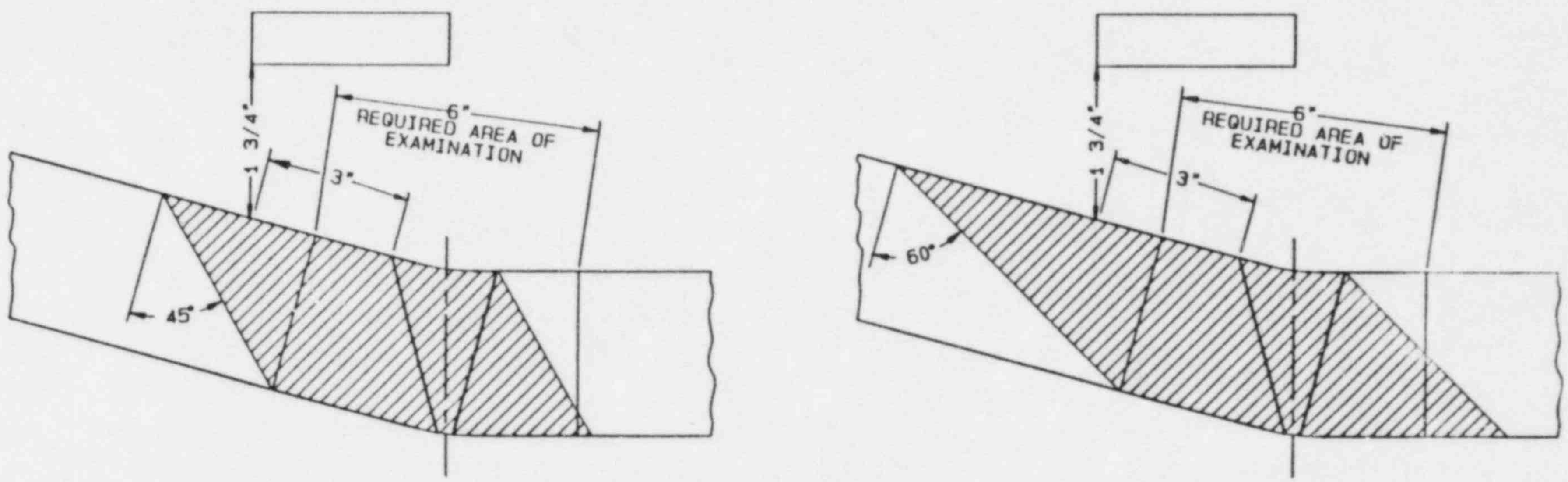

UNITS #1 & #2
WELD NUMBER SGW-D1
D2
D3
D4
42% OF THE WELD IS BLOCKED WITH THE 12" PADS
1 3/4"
6"
REQUIRED AREA OF EXAMINATION
3"
12" PAD ON EDGE OF WELD WITH SPACING OF 13" BETWEEN PADS
45°
VOLUME COVERED BY 45°IN AREAS BLOCKED BY 12" PADS
1 3/4"
6"
REQUIRED AREA OF EXAMINATION
3"
12" PAD ON EDGE OF WELD WITH SPACING OF 13" BETWEEN PADS
60°
VOLUME COVERED BY 60°IN AREAS BLOCKED BY 12" PADS

Figure 1

UNITS #1 & #2
WELD NUMBER SGW-D1
D2
D3
D4

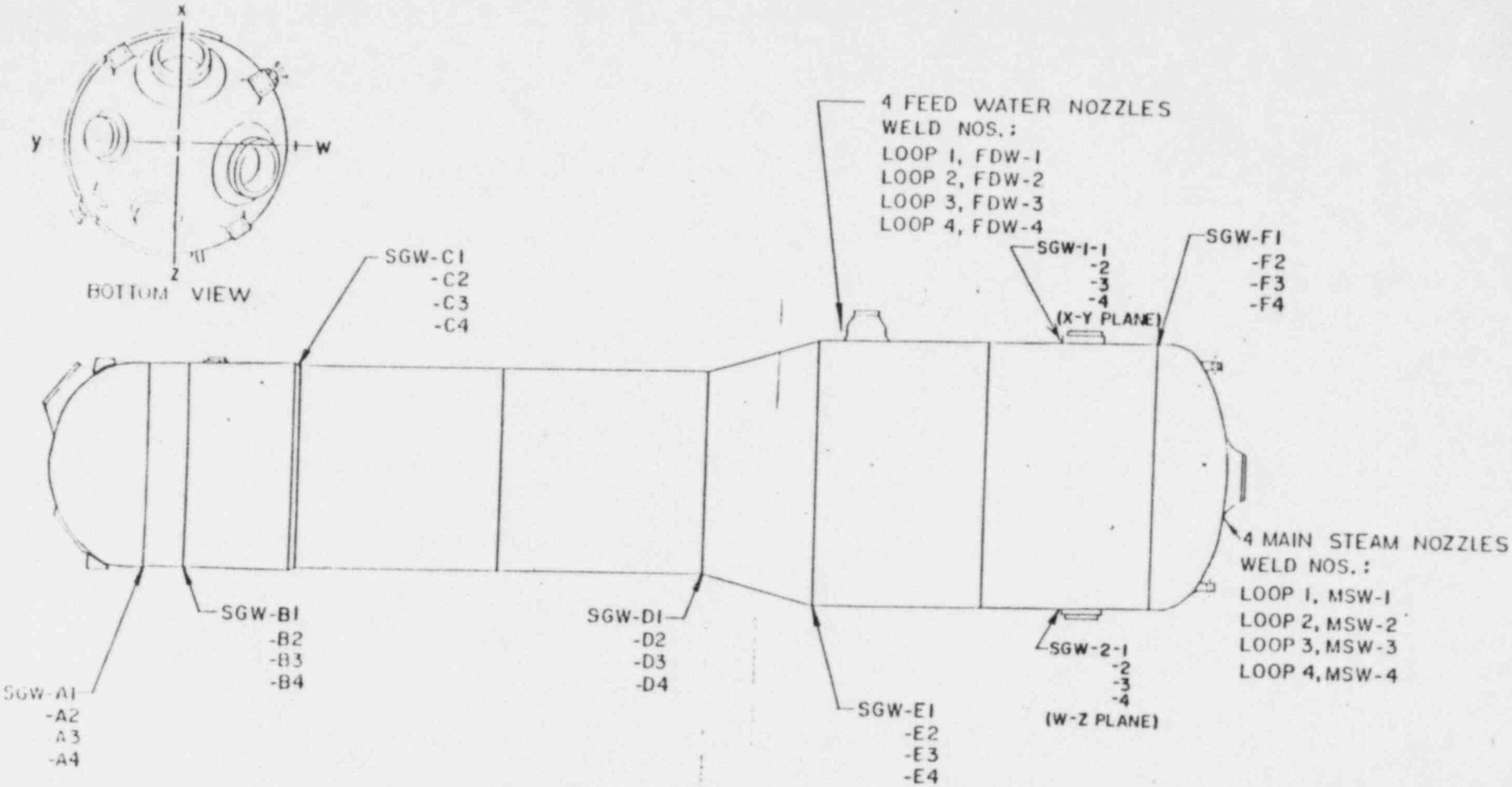
58% OF THE WELD IS NOT BLOCKED BY THE 12" PADS




VOLUME COVERED BY 45° IN AREAS NOT BLOCKED BY 12" PADS

VOLUME COVERED BY 60° IN AREAS NOT BLOCKED BY 12" PADS

Figure 2




TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

UNIT #1 & #2

STEAM GENERATOR

DWN EBC

CH-M-2345B

ATTACHMENT 2