TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
500C Chestnut Street Tower II

FEB 14 1979

Director of Nuclear Reactor Regulation
Attention: Mr. S. A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Varga:

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

Enclosed are response to the Materials Engineering Branch (MEB) questions on preservice and inservice inspection transmitted by S. A. Varga's letter to N. B. Hughes dated December 8, 1978.

The response to MEB item 1, Requests for Relief from Testing Requirements, is contained in Enclosure 1 and will be incorporated into the Sequoyah Nuclear Plant (SNP) Final Safety Analysis Report (FSAR) as a revision to question 5.9 and as question 5.29.

Enclosure 2 is the response to MEB item 2, Compliance with 10 CFR Part 50, Appendixes G and H, which will be incorporated into the SNP FSAR as the response to question 5.30.

MEB item 3, Steam Generator Inservice Inspection, is addressed in Enclosure 3 and will be incorporated into the SNP FSAR as the response to question 5.31.

This material will be incorporated into the SNP FSAR by Amendment 60.

Very truly yours,

J E Gilleland

J. E. Gilleland
Assistant Manager of Power

Enclosure

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An Equal Opportunity Employer

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ENCLOSURE 1

Request for Relief from Certain Testing Requirements
(MEB Item 1 of December 8, 1978, Letter from S. A. Varga to N. B. Hughes)

5.29 <u>Specific Areas Identified in the Review which may Require Relief</u>

Based on our review of TVA's preservice and inservice inspection program for Sequoyah Nuclear Plant Unit Nos. 1 and 2, we have identified the following items that may require relief:

1. The NRC's position is that the inspection of like type of nozzels for the reactor vessel, Item B1.4, Category B-D be evenly distributed over the inspection interval to meet the requirements of ASME Section XI. Since your program indicates this requirement will not be met, relief is required for this item.

2. Your program indicates that the control rod drive housing will not be examined per the requirements of Item B1.18, Category B-O; therefore, a request for relief is required. To support your relief request, discuss the maximum crack opening (largest size line break) in which the reactor can be shutdown and cooled by the reactor coolant makeup system only. Compare the normal cool-down capacity with the maximum opening created if the CRDM housing weld failed.

3. Clarify the extent of volumetric examination that can be achieved for the cast stainless steel RCS piping. If the complete volume is not examined for the Category B-J welds, relief from the requirement is needed. Describe the volume that will be examined.

4. State if there is any bolting two inches in diameter or less (Category B-G-2) associated with the reactor vessel. If there is, include it in the program.

5. In Table A of the ISI program for Class A components, 2-1/2 inch diameter welds listed under "Safety Injection system, circumferential and socket welds" are scheduled for PT examination. Clarify if these welds are socket welds. If the welds are not socket welds, relief is required from the volumetric examination requirements of Item B4.5, Category B-J.

6. In Table A of the ISI program for Class A components, 11-inch diameter stainless steel and 6-inch diameter Inconel piping welds are scheduled for PT examination. State why these welds are not being examined in conformance with the requirements of Item B4.5, Category B-J. Request relief for these items is necessary if the above stated requirements cannot be met.

7. State if the reactor coolant pump casings are welded. If the pump casings are welded, include the examination of the welds in the preservice/ISI program.

8. Since the inspection of valve bodies per the requirements of Category B-M-2 will not be fulfilled by your program, relief from the requirement is needed. For preservice inspection, indicate if the visual inspections of the valve bodies have been performed by the manufacturer and if the test records are available.

9. Section XI requires that an independent third party be utilized as the inspection agency. In your program you state the TVA will perform the independent third party inspection function itself since it is an agency of the Federal government. TVA should provide the technical and/or legal justification that allows relief or exemption from having the third party inspection performed by an independent agency, other than TVA.

10. Provide details regarding the system testing that will be performed. Will the hydrostatic and leak testing meet the requirements of ASME Section XI?

11. In your ISI program, you state the program will essentially comply with ASME Section XI, 1974 Edition through Summer 1975 Addenda. However, for the examination of piping you proposed to use Appendix III of Section XI from a later code edition. Since the approved code, 1974 edition including Summer 1975 Addenda requires ultransonic testing to Appendix I or Article 5 of ASME Section V, relief from the requirements of the code is needed.

12. Table B of the licensee's ISI submission dated April 19, 1978, includes ASME Class 2 piping, Categories C-F and C-G. The total sample of the piping is listed but not the sample to be tested or the test schedule for this interval. This information must be supplied to demonstrate compliance with code requirements.

13. Clarify the program statement that integrally welded supports and support components, Classes A, B, and C will be examined as prescribed by ASME Section XI but the examinations are not included in Tables A and B.

<u>Response</u>

1. Section XI does not designate that examination of like-type nozzels for the reactor vessel, Item B1.4, Category B-D, be distributed evenly over the inspection interval. Informal discussions with the ASME Section XI Code Committee revealed that the intent of Section XI is that examination of nozzles, rather than like-type nozzles, be distributed over the inspection interval in accordance with IWB-2400. This position is further clarified in the 1977 Edition, Summer 1978 Addenda, of Section XI, Table IWB-2500-1, Items B3.10 and B3.90.

2. The Sequoyah inservice inspection program will be revised to include examination of the pressure-retaining welds in the peripheral control rod drive housings, Item B1.18, Category B-O.

3. The extent of volumetric examination for the cast stainless-steel RCS piping includes the weld metal and base metal for one wall thickness beyond the edge of the weld in accordance with Item B4.5, Category B-J. Ultrasonic (UT) examinations of each weld are performed parallel to the weld from two directions and perpendicular to the weld from two directions.

4. There is no bolting two inches in diameter or less (Category B-G-2) associated with the reactor vessel.

5. The 2-1/2-inch diameter welds listed in Table A of the ISI program under "Safety Injection System, Circumferential and Socket Welds" are scheduled for UT examination. The schedule for liquid penetrant (PT) examination was made in error.

6. The 11-inch diameter stainless-steel piping welds in Table A of the ISI program (upper-head injection system) were listed in error and will be removed from the program. The 6-inch diameter Inconel piping welds listed in Table A are scheduled for UT examination. The schedule for PT examination was made in error.

7. The reactor coolant pump casings are not welded.

8. At least one valve in each group of similar valves, as identified in Examination Category B-M-2, will be visually examined for the preservice inspection. The ISI program will be revised to reflect these examinations. See Q5.9, Exception 18.

9. See Q5.9, Exception 1.

10. The hydrostatic pressure testing program is in the course of preparation. Any requests for relief from Section XI requirements will be submitted pending completion of the program.

11. References to Appendix III of Section XI will be withdrawn. UT testing will be performed in accordance with Appendix 1 of Section XI or Article 5 of Section V.

12. The sample of Class B piping welds to be examined, Categories C-F and C-G, will be included in the ISI program.

13. Installation of all Class A, B, and C integrally welded supports and support components has not been finished pending completion of construction of various components. The examinations of the integrally welded supports and support components will be included in Tables A and B of the ISI program after completion of support installation.

TABLE Q5.9-3

EXCEPTION 1

I. Components

All components.

II. Exception Taken

The inspector will be an employee of TVA and will not be qualified in accordance with IWA-2130.

III. Justification

Present TVA policy is to provide its own inspection services. TVA is a Federal agency and, as other Federal agencies, acts as its own inspector and is not subject to State or other non-Federal inspectors.

IV. Testing in Lieu of Section XI Requirements

TVA will provide its own independent review of the Section XI program through its central office staff in Chattanooga.

TABLE Q5.9-3
(Continued)

EXCEPTION 16

I. Components

Various class 1 and 2 piping systems.

II. Exception Taken

Calibration standards will not include the calibration reflector required by paragraph T-533.2(a) of Article 5 of ASME Section V.

III. Justification

Paragraph T-533.2(a) of Article 5 of ASME Section V requires that the basic calibration block include a basic calibration hole drilled parallel to the contact surface. However, paragraph T-533.2(b) permits the use of other calibration reflectors provided equivalent responses to that from the basic calibration hole are demonstrated. In 1973 TVA and Nuclear Services Corporation performed an evaluation of notches versus side-drilled holes in basic calibration blocks. A 5 percent notch calibration reference reflector was used to determine the primary ultrasonic reference level on pipe having a wall thickness less than 1 inch. It is demonstrated in figure Q5.9-1 that calibration with the 5 percent notch is on the average 3.5 dB more sensitive than calibration using a 3/32-inch side-drilled hole.

We conclude that calibration standards fabricated with notches provide a conservative examination.

IV. Testing in Lieu of Section XI Requirements

TVA at its discretion may use 5 percent notches in place of side-drilled holes as permitted by paragraph T-533.2(b) of Article 5 of ASME Section V.

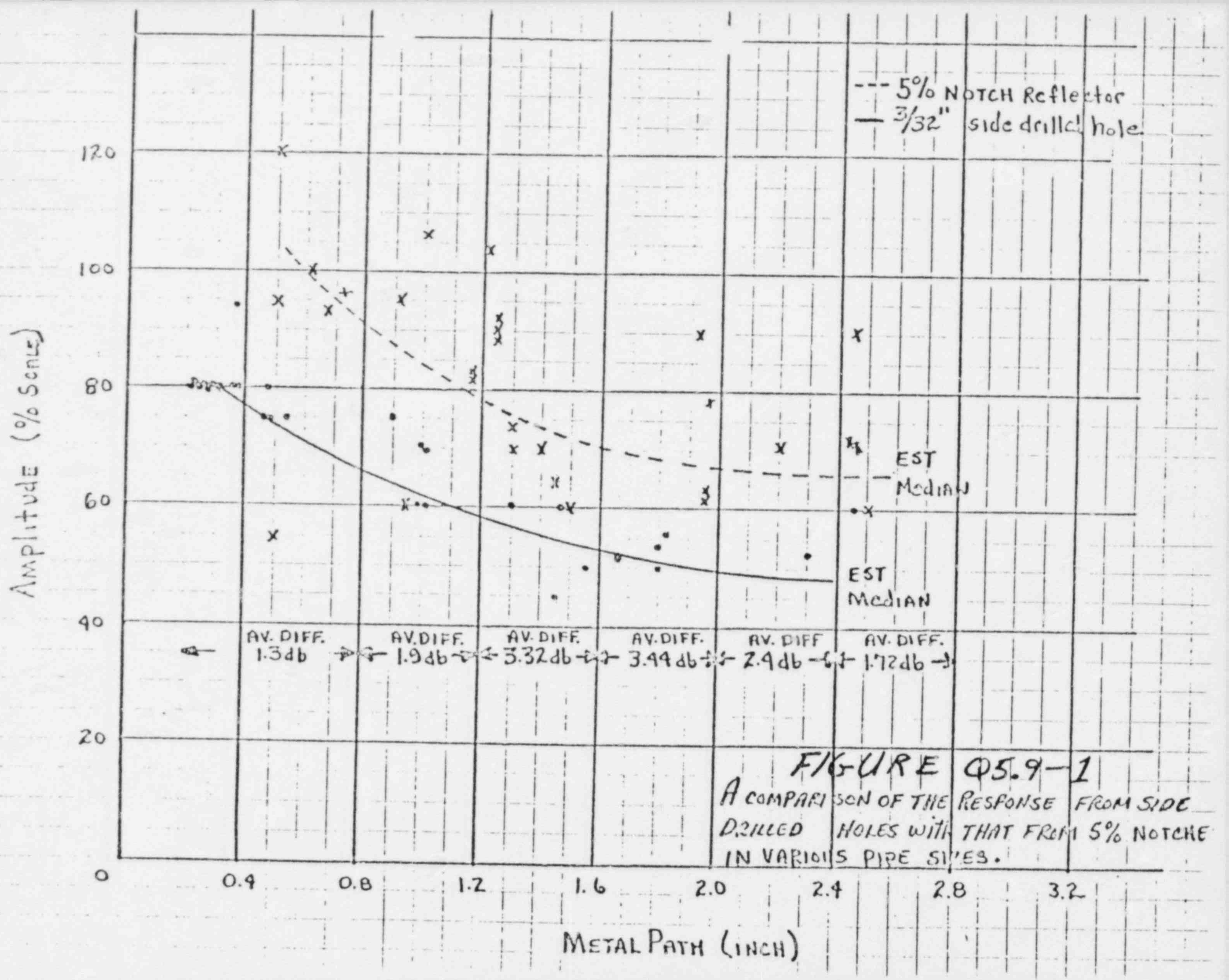


FIGURE Q5.9-1

A COMPARISON OF THE RESPONSE FROM SIDE DRILLED HOLES WITH THAT FROM 5% NOTCHE IN VARIOUS PIPE SIZES.

TABLE Q5.9-3
(Continued)

EXCEPTION 17

I. Components

Reactor coolant pump (RCP).

II. Exception Taken

All the RCPs will not be disassembled solely for visual examination of pump internal pressure boundary surfaces.

III. Justification

In absence of required maintenance, disassembly of a reactor coolant pump solely to perform a visual examination of internal surfaces is impractical. This would represent an unnecessary employee exposure to high radiation and contamination areas and an excessive expense to TVA.

Time required for this major task of disassembly, examination, and reassembly would consume at least three weeks of 24-hours per day work. Radiation dose rates of the pump exterior will average 100-300 mrem/hour, and pump internal dose rates will average 10-20 rem/hour.

The benefit received from this major effort is minimal considering employee exposure, potential damage to safety related equipment, and cost in dollars.

In addition, the two units at Sequoyah Nuclear Plant will operate under similar conditions. Therefore, we feel that if a pump from one of the units is disassembled for maintenance during a 10-year interval, the visual examination performed will be representative of the pump condition for each unit. This would avoid unnecessary employee exposure to the high radiation dose rates noted above. We conclude that if one pump is disassembled for maintenance during the 10-year interval, the visual examination performed satisfies examination category B-L-2 requirements for both units. Disassembly of the pump solely for visual examination is impractical.

IV. Testing in Lieu of Section XI Requirements

The internal surfaces of the reactor coolant pump casing will be visually examined whenever the surfaces are made accessible when a pump is disassembled for maintenance purposes. If during the 10-year interval a pump from either unit is not disassembled for maintenance, a pump from one unit shall be examined from the exterior. This shall be accomplished by ultrasonic thickness measurements of the pump casing.

TABLE Q5.9-3
(Continued)

EXCEPTION 18

I. Components

Class 1 valves exceeding 4-inch nominal pipe size.

II. Exception Taken

All of these valves will not be disassembled solely for visual examination of valve internal pressure boundary surfaces.

III. Justification

During routine maintenance, visual examinations of valve body internal pressure boundary surfaces are performed and documented under existing plant administrative procedures. Most Class 1 valves, particularly containment isolation valves, are disassembled frequently for maintennace. In addition, the two units at Sequoyah Nuclear Plant will operate under similar conditions. If a valve from one of the units is disassembled for maintenance within a 10-year interval, we feel that the visual examination performed would be representative of both units and would be sufficient to satisfy the examination requirements for both units for that particular valve classification as defined in examination category B-M-2.

We conclude that if one valve in each group of valves of the same constructional design and manufacturer that perform similar function is disassembled from either unit during the 10-year interval, the visual examination performed satisfies examination category B-M-2 requirements for both units.

IV. Testing in Lieu of Section XI Requirements

If a valve from a particular classification has not been disassembled as the end of the inspection interval approaches, a case-by-case study will be made to determine the practicality of disassembling a valve from one of the units solely for visual examination (determine if draining the vessel would be required, etc.). If necessary, a request for relief will be issued at that time.

TABLE Q5.9-3
(Continued)

EXCEPTION 19

I. Components

Class 1 and 2 pressure-retaining welds in piping.

II. Execption Taken

Volumetric examination of longitudinal, circumferential, and pipe branch connection welds (examination categories B-J, C-F, and C-G) will not be done for all welds from both sides.

III. Justification

In some cases it will be impractical to inspect all welds from both sides, i.e., nonremovable hanger interference or valve and pump casings adjoining the welds. These welds will be noted on the ultrasonic examination data sheets.

IV. Testing in Lieu of Section XI Requirements

In addition to the visual examination performed during system leakage and hydrostatic pressure tests, a "best effort" ultrasonic examination will be performed. Where there is interference or problems from one direction only, consideration will be given to performing an angle beam examination for 2T + (greater than two wall thicknesses from one edge of weld) from the accessible side of the weld in order to examine the entire weld cross section.

TABLE Q5.9-3
(Continued)

EXCEPTION 20

I. Components

Steam generators (four per unit).

II. Exception Taken

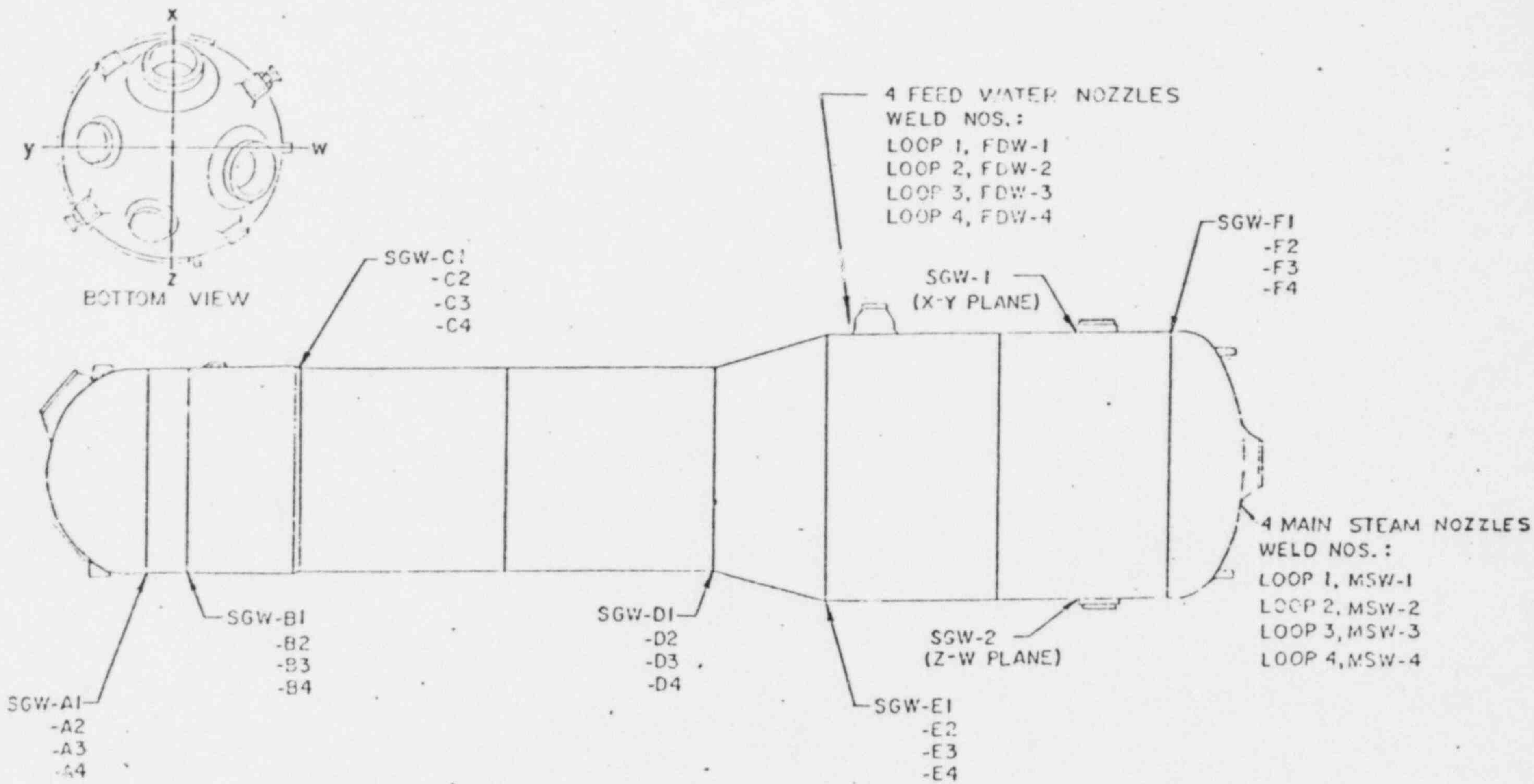
Volumetric examination of circumferential shell welds (examination category C-A) will not be completed for one weld.

III. Justification

One circumferential shell weld on each generator is inaccessible due to the upper steam generator support brackets (weld nos. SGW-D1, SGW-D2, SGW-D3, and SGW-D4). See figure Q5.9-2 for weld location. Also, figures Q5.9-3 and Q5.9-4 show arrangements of the support brackets. One weld on one generator will be examined on a "best effort" basis for the baseline inspection and during the four inservice inspection intervals in accordance with IWC-2411.

IV. Testing in Lieu of Section XI Requirements

None.



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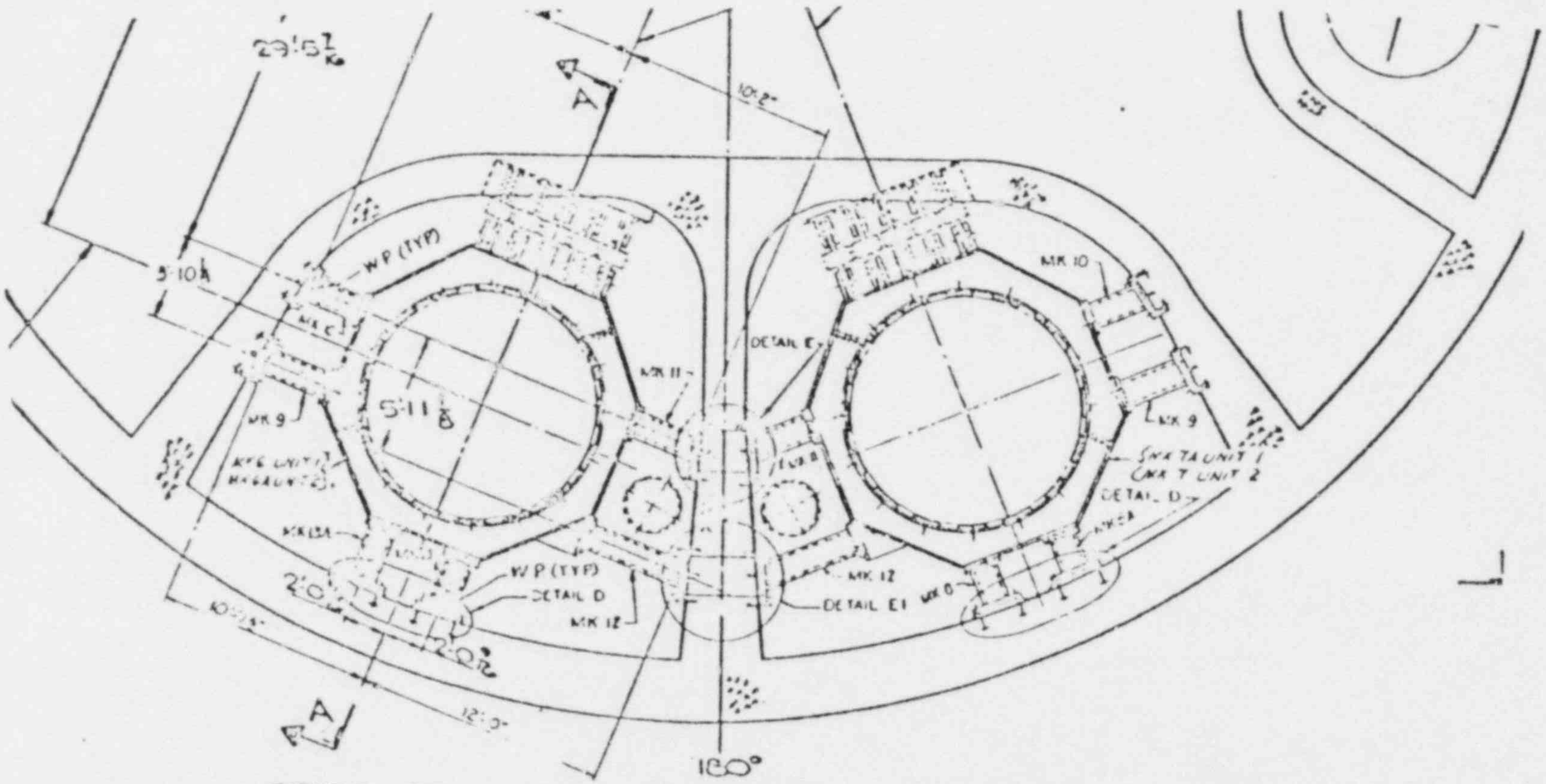
STEAM GENERATOR

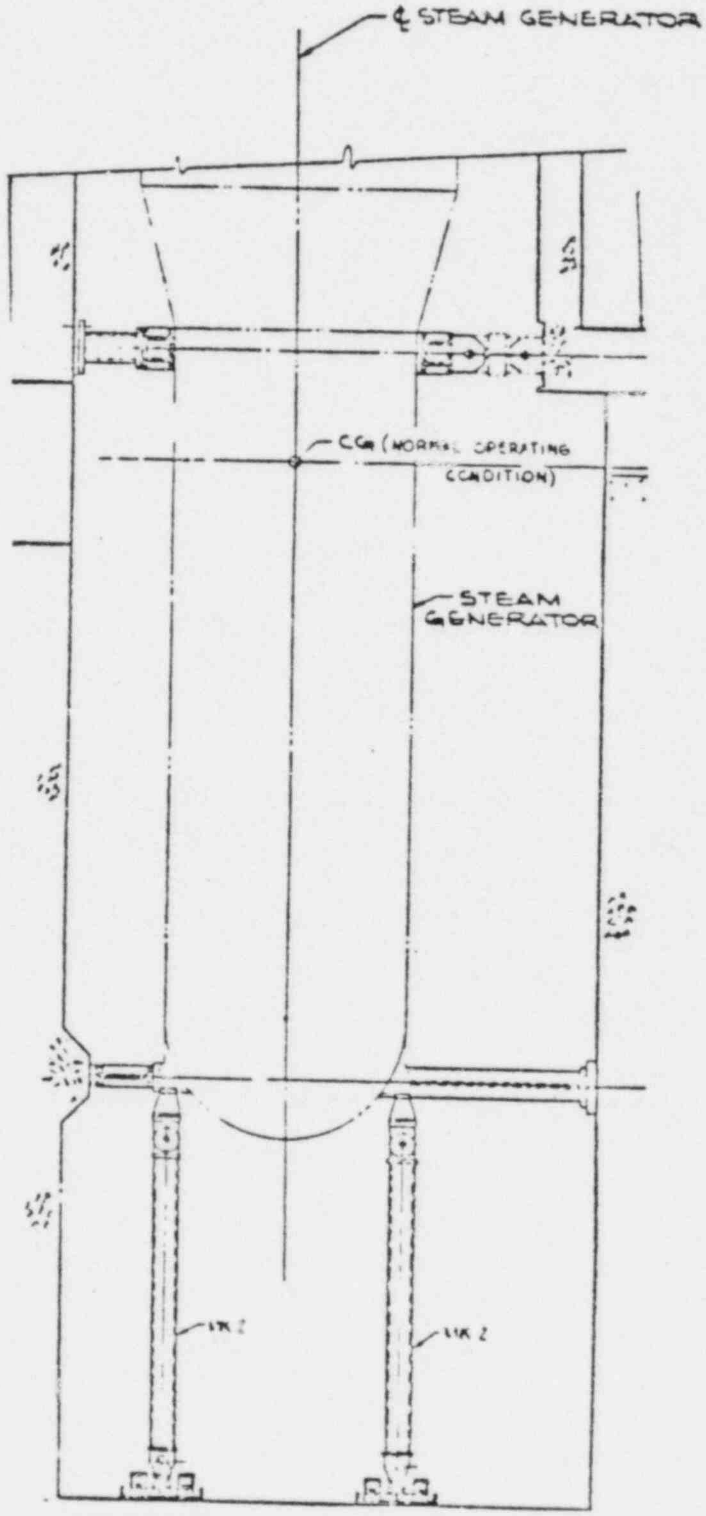
Figure Q5.9-2

CH-M23458



UPPER STEAM GENERATOR SUPPORT ARRANGEMENT

Figure Q5.9-3



SECTION-A-A

Figure Q5.9-4

TABLE Q5.9-3
(Continued)

EXCEPTION 21

I. Components

Reactor pressure vessel.

II. Exception Taken

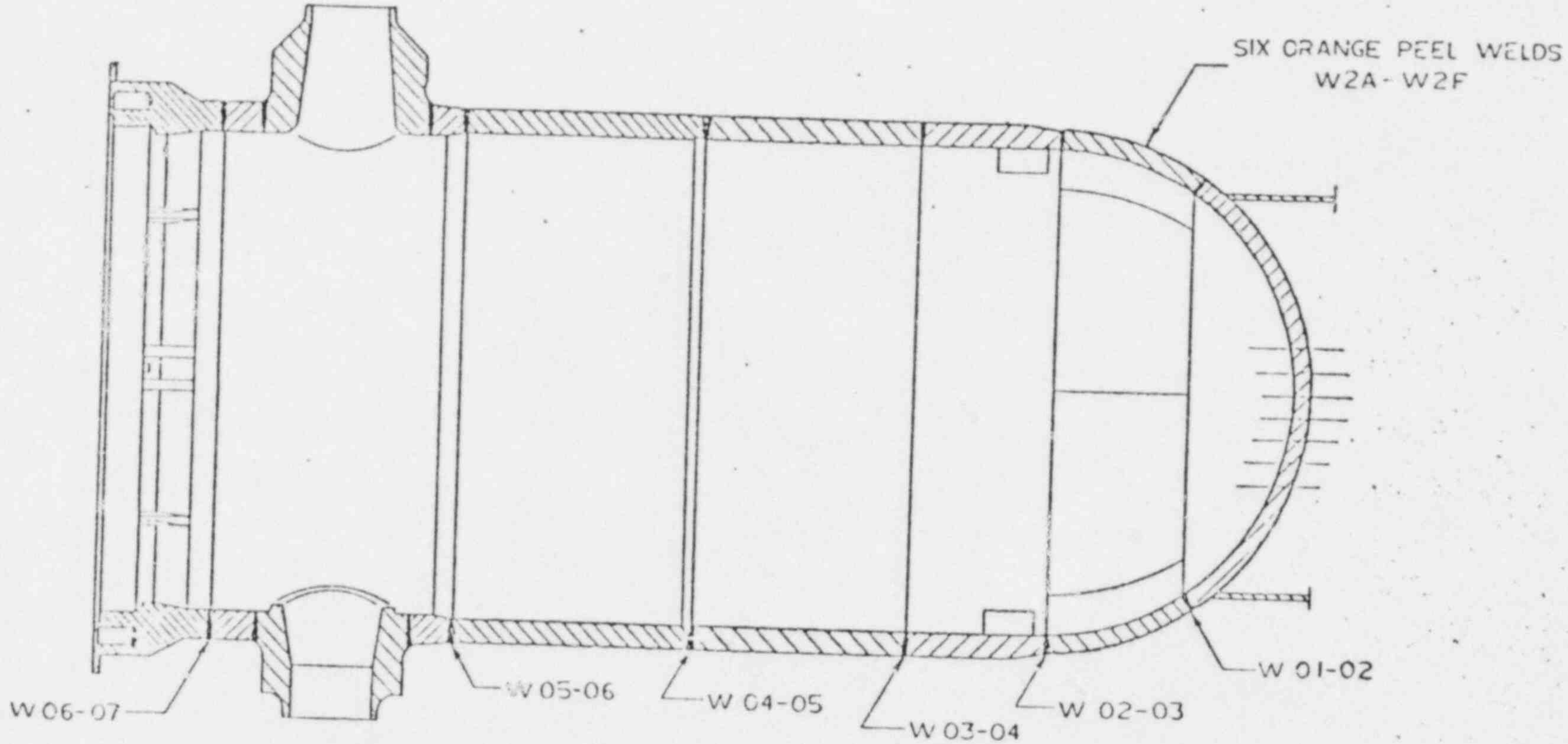
One-hundred percent preservice baseline volumetric examination of lower head dollar weld (examination category B-B) will not be made under conditions and with equipment and techniques equivalent to those expected to be employed during inservice inspection.

III. Justification

TVA will employ automated remote inspection devices to examine most of the reactor vessel welds. These examinations will be conducted from the vessel inside diameter. However, the lower head weld on each reactor pressure vessel is partially inaccessible for examination from the vessel inside diameter due to instrumentation tubes which penetrate the lower head (weld no. W01-02--see figures Q5.9-5 and Q5.9-6). Portions of the weld can be examined from one side (as permitted by 1-5121 of Section XI) and will include 100 percent of the examination volume in accordance with IWB-3511.1 of Section XI. These portions of the weld will be reexamined during the inservice intervals in accordance with examination category B-B of Table IWB-2500.

IV. Testing in Lieu of Section XI Requirements

A 100 percent baseline examination of the weld will be conducted from the vessel outside diameter. This will be acommplished by performance of a manual ultrasonic examination. A remote ultrasonic examination will be conducted from the vessel inside diameter on all accessible areas of the weld.



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REACTOR VESSEL SEAM WELDS
SEQUOYAH NUCLEAR PLANT

Figure Q5.9-5

NTS RGS 1-10-79 2343-B

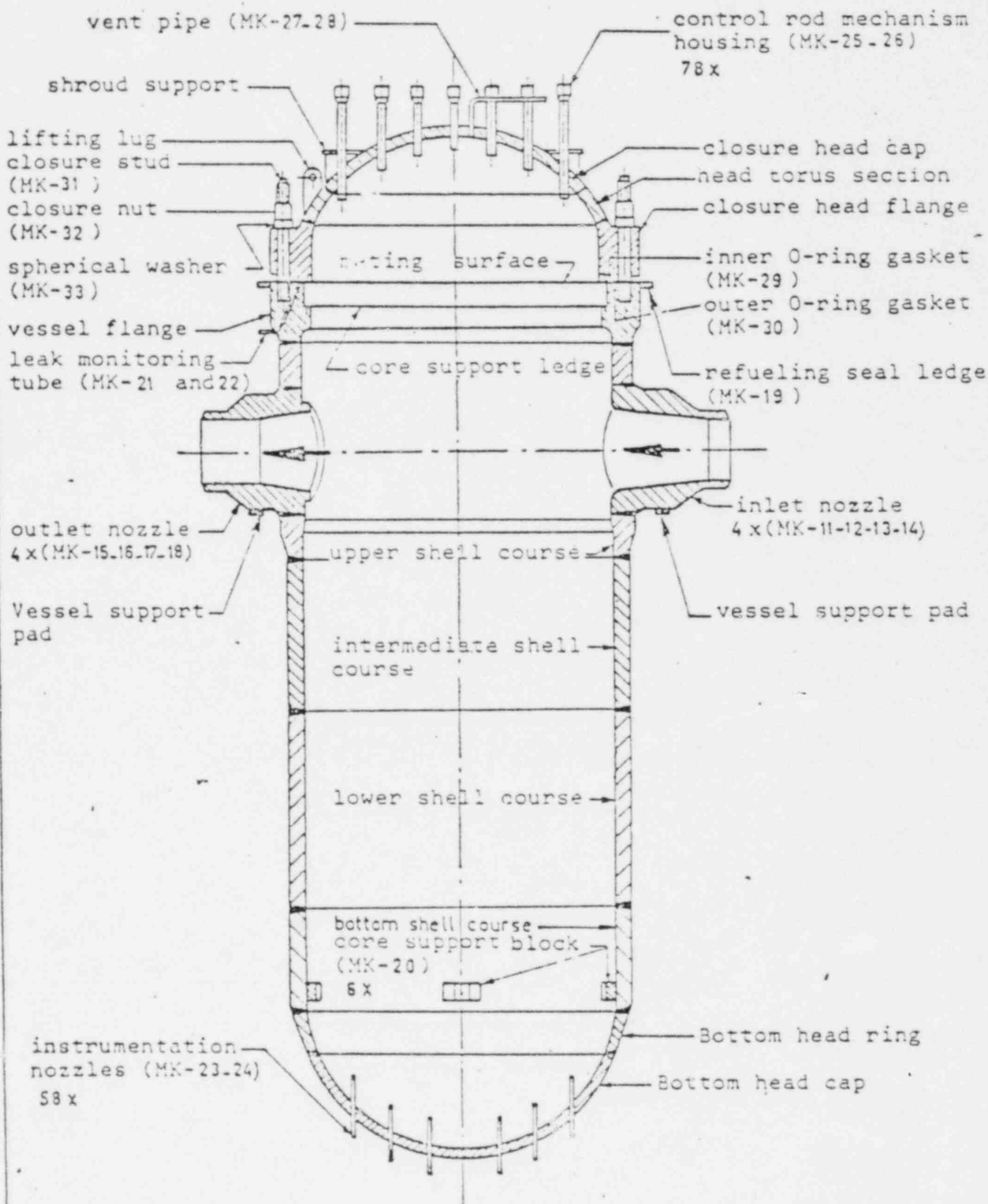

vent pipe (MK-27-28)
control rod mechanism housing (MK-25-26)
78 x
shroud support
lifting lug
closure stud (MK-31)
closure nut (MK-32)
spherical washer (MK-33)
vessel flange
leak monitoring tube (MK-21 and 22)
closure head cap
head torus section
closure head flange
surface
inner O-ring gasket (MK-29)
outer O-ring gasket (MK-30)
core support ledge
refueling seal ledge (MK-19)
inlet nozzle
4 x (MK-11-12-13-14)
outlet nozzle
4 x (MK-15-16-17-18)
upper shell course
Vessel support pad
vessel support pad
intermediate shell course
lower shell course
bottom shell course
core support block (MK-20)
6 x
Bottom head ring
instrumentation nozzles (MK-23-24)
58 x
Bottom head cap

Figure Q5.9-6

TABLE Q9.5-3
(Continued)

EXCEPTION 22

I. Components

Steam generator nozzle.

II. Exception Taken

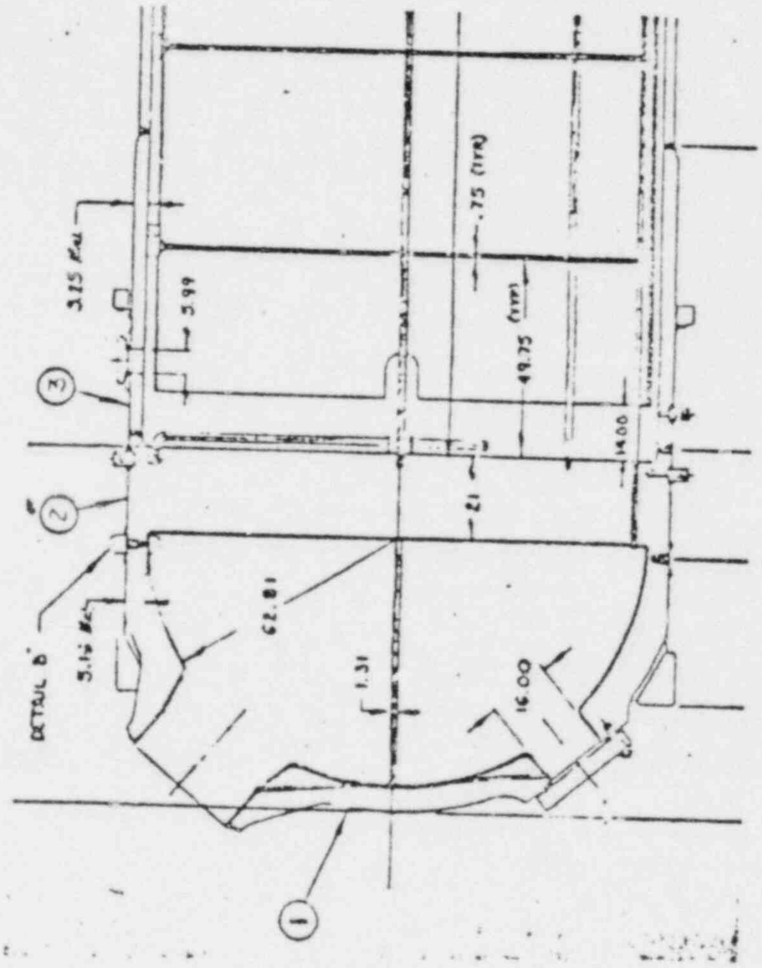
Volumetric examination of nozzle inside radius section on the primary side (examination category B-D) will not be made.

III. Justification

Each steam generator consists of two integrally cast nozzles and two integrally cast manways. The present capability of ultrasonic testing is not sufficient to examine cast material of this thickness and achieve meaningful results. (See figure Q5.9-7.)

IV. Testing in Lieu of Section XI Requirements

None.



Steam generator primary nozzles and manways

Figure Q5.9-7

TABLE Q9.5-3
(Continued)

EXCEPTION 23

I. Components

Reactor coolant loop pipe welds.

II. Exception Taken

Volumetric examination of circumferential welds (examination category B-J) will not be done for all welds.

III. Justification

Two circumferential welds in the reactor coolant loop piping (RC-2351 and RC-3151, loops 3 and 4) are located inside the reactor vessel shield wall and are inaccessible for baseline and inservice examination (see figures Q5.9-8 and Q5.9-9). Both welds have undergone shop radiographic examinations. Since the baseline inspection serves as a reference to future inservice inspections and both welds will be inaccessible for inservice inspections, the shop radiographic examination coupled with the Section III hydrostatic test will provide adequate proof of integrity of the system welds. Inservice system leakage and hydrostatic testing will prove weld integrity during the life of the plant.

IV. Testing in Lieu of Section XI Requirements

None.

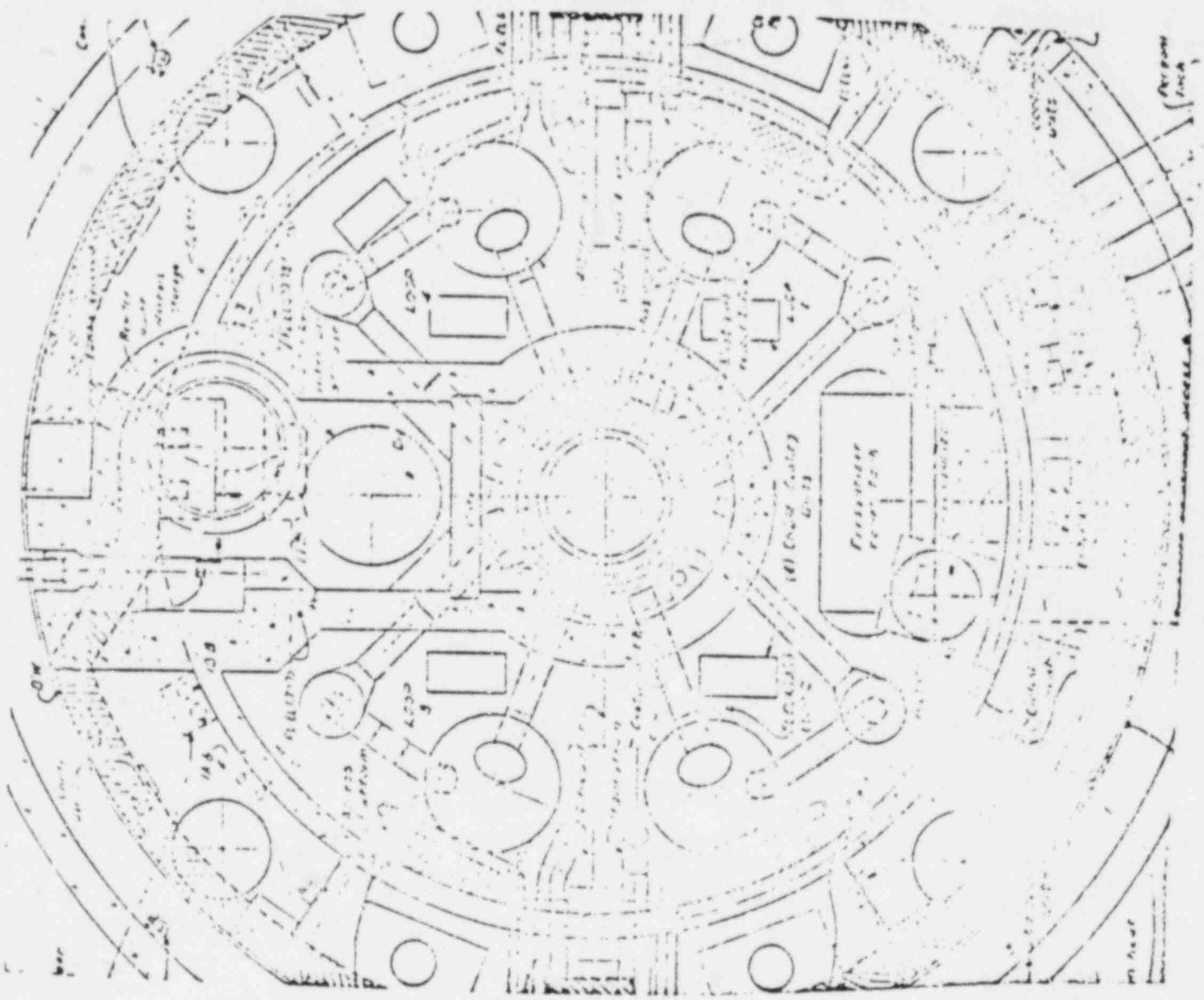


Figure Q5.9-8

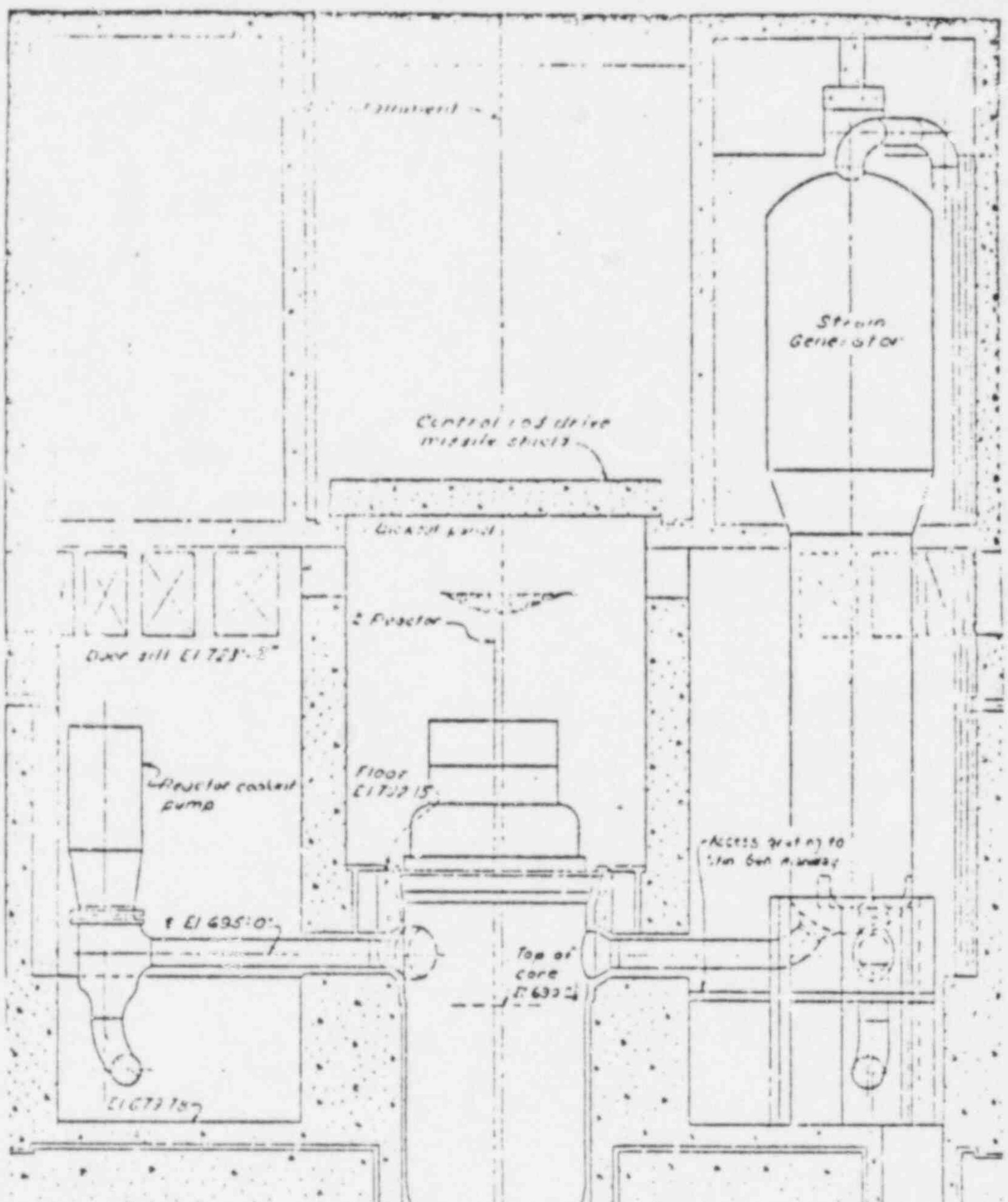


Figure Q5.9-9

ENCLOSURE 2

(MEB Item 2 of December 8, 1978, Letter from S. A. Varga to N. B. Hughes)

5.30 Compliance with 10 CFR Part 50, Appendices G and H

1. State if the ferritic materials in the pressurizer and steam generators that are part of the reactor coolant pressure boundary meet the exact requirements of 10 CFR Part 50, Appendix G. Provide the fracture toughness data available for these materials. State the ASME Code Edition used for these components.

2. State if all material for bolting and other fasteners greater than 1-inch diameter in the reactor coolant pressure boundary meet the exact fracture toughness requirements of 10 CFR Part 50, paragraph IV.A.4. Provide the fracture toughness data available for all bolting greater than 1-inch diameter used for the reactor coolant pressure boundary. Indicate the ASME Code Edition used for these components. The mechanical properties which have been submitted for the closure head bolting for Units Nos. 1 and 2 need not be provided again.

3. State if the requirements of 10 CFR Part 50, Appendix G, paragraphs IIIB.3., 4, and 5 were met for the fracture toughness tests which were performed. Clarify if the testing was performed by an organization with a quality assurance program in conformance with 10 CFR Part 50, Appendix B.

4. Although Westinghouse Topical Report WCAP-7924 has been accepted by the NRC, the method for determining RT_{NDT} shift was not accepted. Provide information demonstrating that the pressure-temperature limits for reactor vessel heat-up and cool down will be constructed using the prediction for temperature shift contained in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Response

It has been determined that retrieval of the requested information required Westinghouse Electric Corporation support; therefore, the information will be forwarded to NRC by March 1, 1979.

ENCLOSURE 3

(MEB Item 3 of December 8, 1978, Letter from S. A. Varga to N. B. Hughes)

5.31 Steam Generator Inservice Inspection

Confirm that the preservice and inservice inspection of steam generator tubing will be conducted in accordance with Regulatory Guide 1.83 Revision 1. If any of these examination requirements cannot be met, a complete technical justification to support your conclusions must be provided.

Response

Preservice and inservice inspection of steam generator tubing will be conducted in accordance with Regulatory Guide 1.83, Revision 1. These requirements will be included in the ISI program.