IES
UTILITIES INC.

John F. Franz, Jr.
Vice President, Nuclear

August 24, 1994
NG-94-3120

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Response to NRC Generic Letter 94-03: Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors, dated July 25, 1994

References: NRC Generic Letter 94-03

File: A-101b, A-286a, B-11

Dear Mr. Russell:

The NRC Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," requested that IES Utilities Inc. provide within 30 days the following information concerning the Duane Arnold Energy Center (DAEC):

1a) a schedule for inspection of the core shroud,
1b) a safety analysis supporting continued operation of the facility until inspections are performed,
1c) drawing(s) of the core shroud configuration, and
1d) a history of shroud inspections for the DAEC.

This information is provided in the attachments. As requested in the GL, we will provide our inspection plan to the NRC no later than three months prior to performing the core shroud inspections, along with our plans for evaluation and/or repair of the core shroud based on the inspection results. We will provide the results of the inspection to the NRC within 30 days after the inspection is completed.

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This letter contains the following new NRC commitments:

1. Provide core shroud inspection plan to the NRC no later than 3 months prior to inspection. Implement recommendations of BWR Vessel and Internals Project (VIP) as applicable. Include plans for evaluation and/or repair of the core shroud based on the inspection results.
2. Review DAEC operating procedures for possible incorporation of operator guidance on core shroud cracking detection. The review will be completed by September 30, 1994.
3. Inspect the DAEC core shroud during Refueling Outage (RFO) 13.
4. Provide results of core shroud inspection to the NRC within 30 days after the inspection is completed.

Should you have any questions regarding this matter, please contact this office.

This letter is true and accurate to the best of my knowledge and belief.

IES UTILITIES INC.

By John F. Franz
John F. Franz
Vice President, Nuclear

State of Iowa
(County) of Linn

Signed and sworn to before me on this 24th day of August, 1994,

by John F. Franz.

Kathleen M. Furman
Notary Public in and for the State of Iowa

September 28, 1995
Commission Expires

JFF/CJR/pjv

Attachments: 1) IES Utilities Inc.'s Response to GL 94-03
2) Core Shroud Drawings

cc: C. Rushworth
L. Liu
L. Root
R. Pulsifer (NRC-NRR)
J. Martin (Region III)
NRC Resident Office
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IES UTILITIES INC.'S RESPONSE TO GL 94-03

NRC Request 1(a)

Provide a schedule for inspection of the core shroud.

IES Utilities Inc. Response

IES Utilities Inc. plans to inspect the Duane Arnold Energy Center (DAEC) core shroud during Refueling Outage (RFO) 13, currently scheduled to begin on February 23, 1995. We plan to perform ultrasonic examinations of all accessible welds.

We support the BWR Vessel and Internals Project (VIP) Committee which is presently pursuing various issues associated with the core shroud and other reactor vessel internals. We will implement the inspection recommendations of the VIP Committee as applicable to the DAEC during RFO 13. Our inspection plan may be revised, if appropriate, based on the recommendations of the BWR VIP.

NRC Request 1(b)

Provide a safety analysis supporting continued operation of the facility until inspections are conducted.

IES Utilities Inc. Response

IES Utilities Inc. has reviewed several factors to assess the safety of continued operation of the DAEC until the core shroud is inspected during RFO 13. We have concluded that continued operation of the facility presents no undue risk to the public, as discussed below.

Mitigating Factors

The Boiling Water Reactor Owners' Group (BWROG) letter dated August 5, 1994 provided a response to an NRC request for shroud information. This submittal included GENE-523-A107P-0794 Revision 1, "BWR Shroud Cracking Generic Safety Assessment" dated August 1994. This document bounds BWRs and categorizes them into the following seven groups:

1) 304L with lowest conductivity history,
2) 304 with lowest conductivity history,
3) 304L with mid-range conductivity history,
4) 304 with mid-range conductivity history,
5) 304L with highest conductivity history,

6) 304 with highest conductivity history and forged support rings, and
7) 304 with highest conductivity history and welded-plate support rings.

The document states that for "all but one grouping" (number 7), "the likelihood of finding any 360° cracking during the next outage is extremely low." The DAEC shroud is categorized as "304L with mid-range conductivity." Therefore, the likelihood of the DAEC experiencing a 360° indication before the next scheduled refueling outage is extremely low.

The DAEC shroud material is 304L which is less susceptible than 304 to intergranular stress corrosion cracking (IGSCC). Table 2-3 of GENE-523-A107P-0794, Revision 1 supports this conclusion.

The DAEC reactor water chemistry during the first five cycles of operation (1974 through 1980) was the basis for categorizing the plant as "mid-range conductivity." In 1987, the DAEC began operating with hydrogen water chemistry. Also, in 1987, new filter elements with a much smaller pore size were installed in the condensate demineralizer system. As a result of these two actions, DAEC conductivity has consistently been among the lowest in the industry. As an example, average conductivity in 1994, through July, was 0.063 microsiemens/cm. The DAEC has consistently met or exceeded the water chemistry guidelines established by EPRI (EPRI TR 103515) and INPO (Guideline 88-21 Rev. 1). Current industry documents clearly demonstrate that reducing conductivity decreases the crack growth rate under BWR conditions. The typical water purity at DAEC is sufficient to minimize the propagation of IGSCC in BWR materials.

In addition, the DAEC has been very aggressive in performing examinations of the reactor pressure vessel internals. Throughout the nuclear industry, core spray spargers and internal piping, access hole covers, jet pump holddown beams, incore dry tubes, and the top guide have exhibited problems; at the DAEC, no IGSCC indications have been reported after many examinations. (Jet pump holddown beams at DAEC were ultrasonically examined in 1981, 1983 and 1985 with no IGSCC indications. All beams were replaced in 1987 as a preventive measure.) These examination results are due in part to the excellent water chemistry that is a standard at the DAEC. While the shroud head bolts have experienced IGSCC, this is attributed to the crevice area of the Ni-Cr-Fe alloy 600 portion of the bolt formed by a 304 stainless

steel sleeve welded to the bolt shaft. This particular condition does not exist in the DAEC core shroud.

Analysis of Core Shroud Failure and Effect on Plant Safety

The DAEC endorses the safety assessment performed in GENE-523-A107P-0794, Revision 1 as applicable to DAEC. This safety assessment concludes that the lift of the shroud as a result of a 360° through-wall crack in combination with a main steam line break or large loss of coolant accident (LOCA) is not a safety concern.

The main steam line break is the postulated worst case because it results in the most severe depressurization. The large LOCA is also postulated to cause a rapid vessel depressurization. The probability of the DAEC experiencing a main steam line break is remote because IES Utilities Inc. examines the DAEC main steam piping system in accordance with ASME Section XI; no indications of safety significance have been found. The recirculation system piping at DAEC has been examined and IGSCC indications were detected in 1985 which were repaired by the application of nine weld overlays. These overlays and other welds in the recirculation piping are examined in accordance with Generic Letter (GL) 88-01; no new IGSCC indications or propagation of existing indications have been found.

Leakage Detection

A review of Table 2-3 of GENE-523-A107P-0794, Revision 1 reveals that each utility that has experienced cracking of the shroud had cracking in the H-3 or H-4 welds. Although these welds did not exhibit the worst cracking, they are predictors of cracks in other welds.

If leakage were to occur in the H-3 or H-4 welds, depending on the location and the amount of leakage, parameters such as reactor power and reactor recirculation loop temperature may provide indication of leakage. The main indication would be a reduction in power level, causing a power-to-flow anomaly. The control room operators routinely monitor power and core flow conditions which would indicate a change in the parameters discussed above.

For lower shroud leakage, there are no detectable symptoms/indicators since the leakage temperature would be very similar to the fluid temperature in the downcomer region. However, as discussed above, the H-3 and H-4 welds are the best predictors for cracking and it is possible to detect leakage at these locations.

The DAEC operating procedures will be reviewed for potential incorporation of operator guidance on the detection of core shroud cracking. This review will be completed by September 30, 1994.

Probability Risk Assessment

IES Utilities Inc. has evaluated the risk associated with core shroud cracking from a probabilistic risk perspective. The initiator frequencies for the DAEC are virtually identical to those considered for Dresden Unit 3 and Quad Cities Unit 1 (Docket Nos. 50-249 and 50-254) and discussed in the NRC Safety Evaluation (SE) dated July 20, 1994. In that SE, the NRC concluded that operation of Dresden and Quad Cities is acceptable for 15 months based on conservative estimates of the risk contribution from shroud cracking.

The NRC concluded that seismic events are not a limiting contributor at the Dresden/Quad Cities plants. They are even less of a contributor for the DAEC because it is a reduced scope seismic plant.

In the Dresden/Quad Cities SE, the NRC concluded that without considering the failure probabilities for cracks and the failure probabilities of mitigating systems, the core damage sequences ranged from 1E-4 to 1E-5 per reactor-year. Consideration of the crack and mitigating system failures would reduce the frequencies of these sequences to much lower values. The DAEC has core damage frequencies of the same magnitude or less than those for the Dresden and Quad Cities plants for which the NRC has authorized continued operation for 15 months.

NRC Request 1(c)

Provide drawings of the core shroud configuration showing details of the core shroud geometry.

IES Utilities Inc. Response

The DAEC has developed inservice inspection (ISI) reactor pressure vessel sketches. The shroud sketches are included as Attachment 2. These sketches include the core shroud configuration giving details of material, location of welds and weld joint design, along with material thickness.

NRC Request 1(d)

Provide the history of shroud inspections for the plant.

IES Utilities Inc. Response

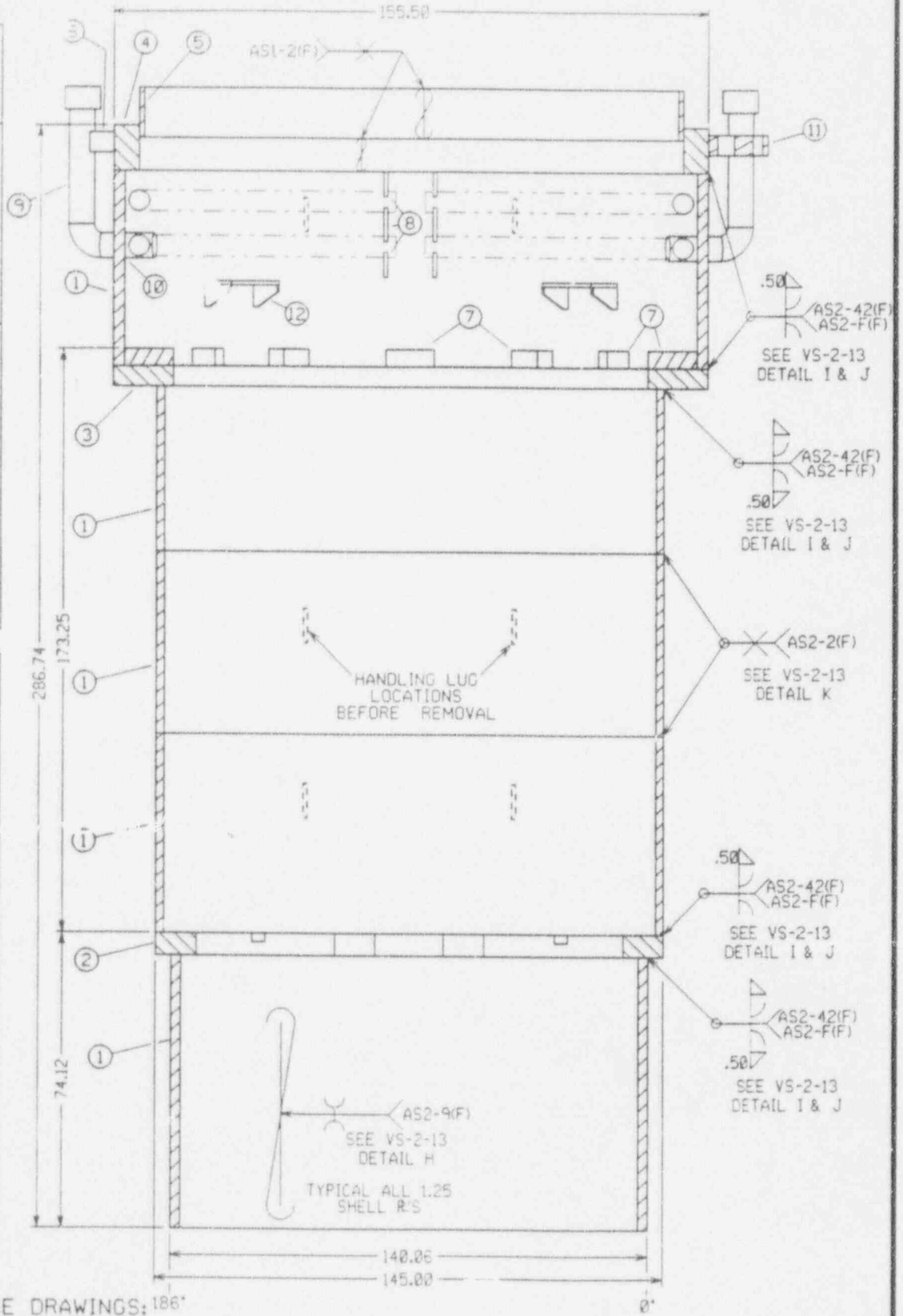
The DAEC has performed inspections in accordance with General Electric RICSIL 054 and SIL 572 Rev 1, as follows:

RICSIL 054

During RFO 11 (1990), the inside diameter (ID) surfaces of welds V5, V6, V7 (accessible portions) and H5 (between 0° and 180°) were inspected. The visual examination was performed utilizing the one-mil wire standard for resolution. No crack-like indications were seen.

SIL 572 Rev 1

During RFO 12 (1993), the ID surfaces of welds H3, H4, H5, V3, V4, V5, V6, V7, and V8 (accessible portions) were examined. The visual examination was performed utilizing the one-mil wire standard for resolution. The Level III examiner who reviewed the DAEC inspection results on videotape was the same Level III examiner who detected the initial cracking at Brunswick. No crack-like indications were seen.



NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	WALL PLATE (1.25")	304L A240	
2	CORE PLATE SUPPORT RING	(SEE VS-2-2)	
3	TOP GUIDE SUPPORT RING	(SEE VS-2-3)	
4	SHROUD CLOSURE FLANGE	(SEE VS-2-4)	
5	FLANGE EXTENSION PLATE	(SEE VS-2-4)	
6	LUG	(SEE VS-2-5)	
7	BLOCK	(SEE VS-2-3)	
8	BRACKET	(SEE VS-2-7)	

NO.	DISCRIPTION	BILL OF MATL.	PROCED.
9	CONNECTOR	(SEE VS-2-9)	
10	SHROUD PENETRATION	(SEE VS-2-7)	
11	PIN GUIDE	(SEE VS-2-6)	
12	TOP GUIDE LATCH PLATE	(SEE VS-2-8)	

REFERENCE DRAWINGS:
APED-B11-18<1>
APED-B11-3084-6

NO.	DATE	REVISION	DFTR	CHK'D	ENGR	VER
1	8-15-94	DRAFTED FOR VESSEL INSPECTION PROGRAM	JM	[illegible]		

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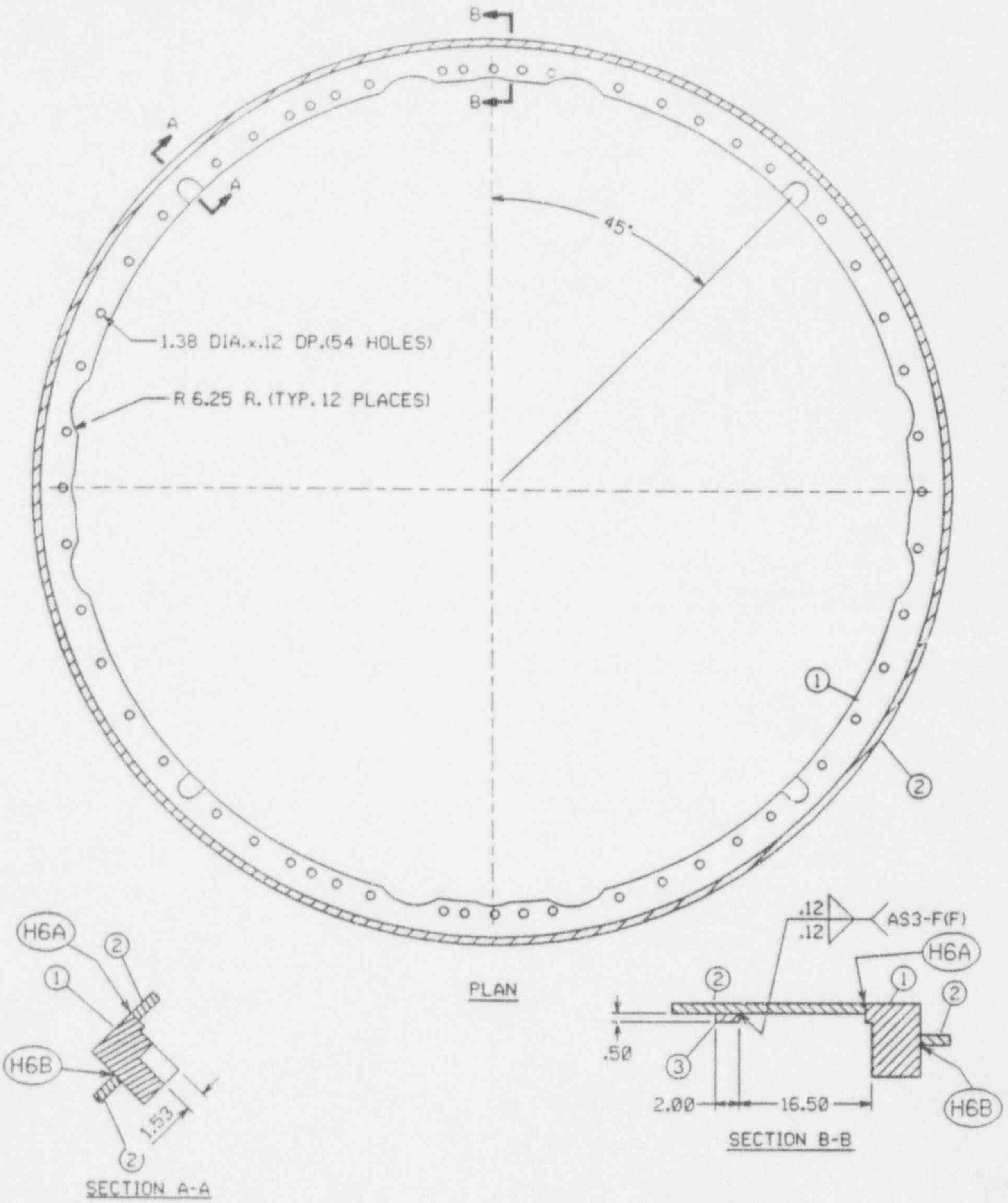
Inservice Inspection Program
Reactor Pressure Vessel Sketch

SHROUD
SECTION

DWG. NO. VS-2-01<1>

REV. 1



NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	FLANGE	304L A240	
2	WALL PLATE	304L A240	
3	BENCH MARK BLOCK	(SEE VS-2-9)	

REFERENCE DRAWINGS:
APED-B11-18<1>, APED-B11-18<2>,
APED-B11-3084-6, APED-B11-3084-7<2>

NO.	DATE	REVISION	DRFTR	CHK'D	ENGR	VER.
1	9/1/94	DRAFTED FOR VESSEL INSPECTION PROGRAM	JM			

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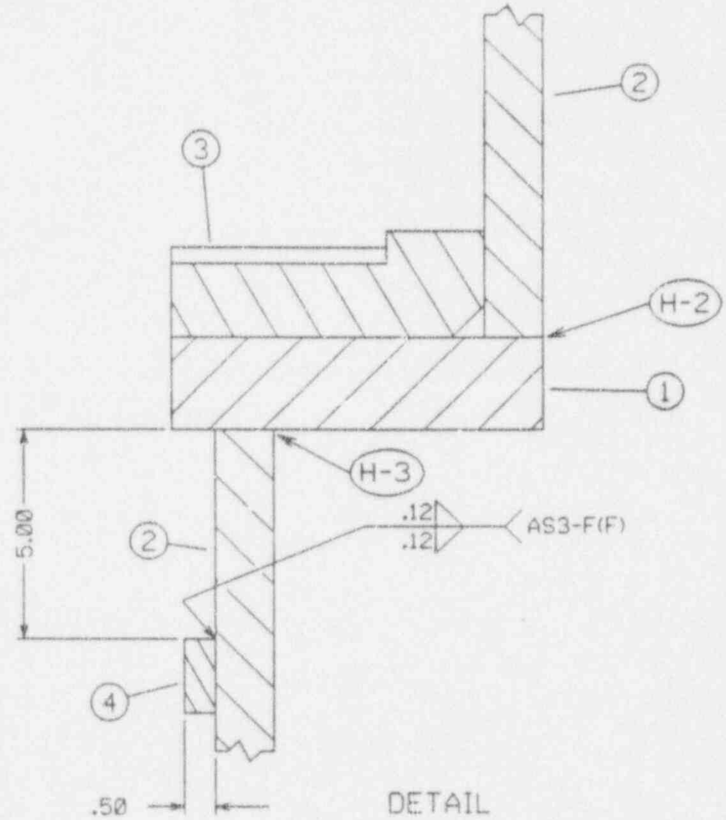
Inservice Inspection Program
Reactor Pressure Vessel Sketch

CORE PLATE
SUPPORT RING

DWG. NO. VS-02-02

REV. 1

NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	FLANGE	304L A240	
2	WALL PLATE	304L A240	
3	BLOCK	304L A240	
4	BENCH MARK BLOCK	(SEE VS-2-9)	



DETAIL

REFERENCE DRAWINGS:
APED-B11-18<1>, APED-B11-18<2>
APED-B11-3084-6, APED-B11-3084-7<2>

NO.	DATE	REVISION	DRFTR	CHK'D	ENGR	VER.
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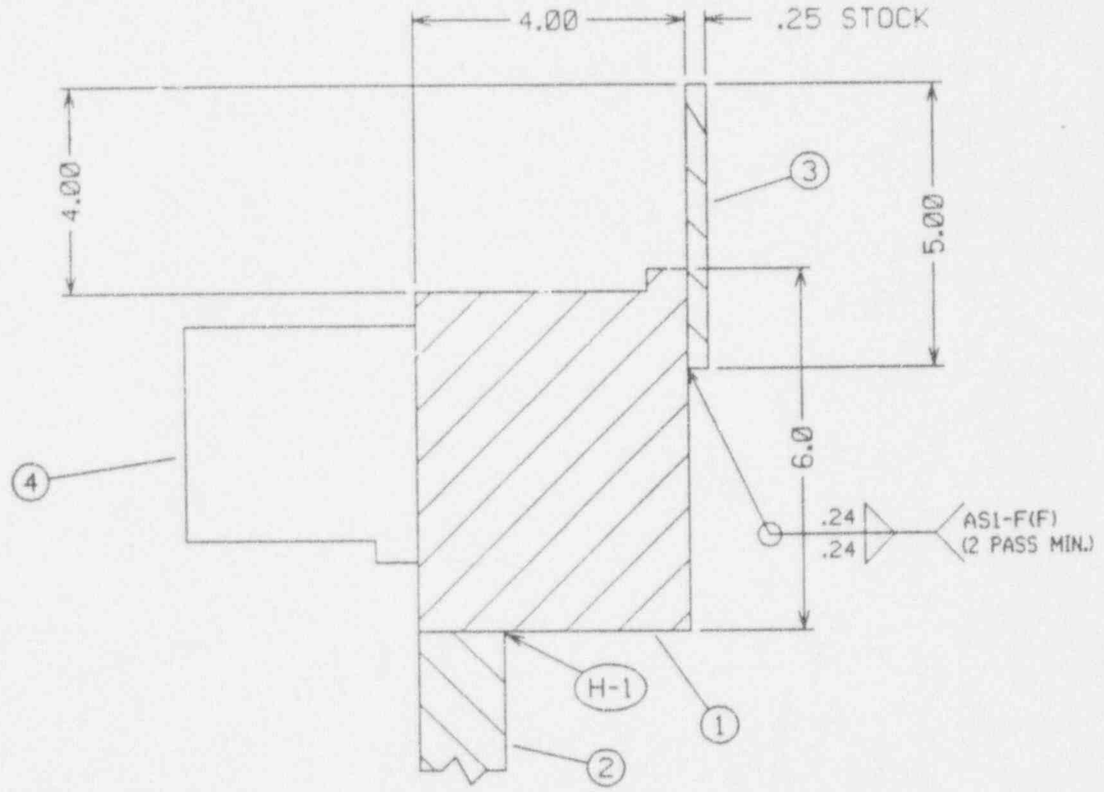
IES:
Inservice Inspection Program
Reactor Pressure Vessel Sketch

TOP GUIDE
SUPPORT RING

DWG. NO. VS-2-03

REV. 1



SECTION

NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	FLANGE	304L A240	
2	WALL PLATE	304L A240	
3	FLANGE EXTENSION PLATE	304L A240	
4	LUG	(SEE VS-2-5)	

REFERENCE DRAWINGS:
APED-B11-18(1), APED-B11-3084-6

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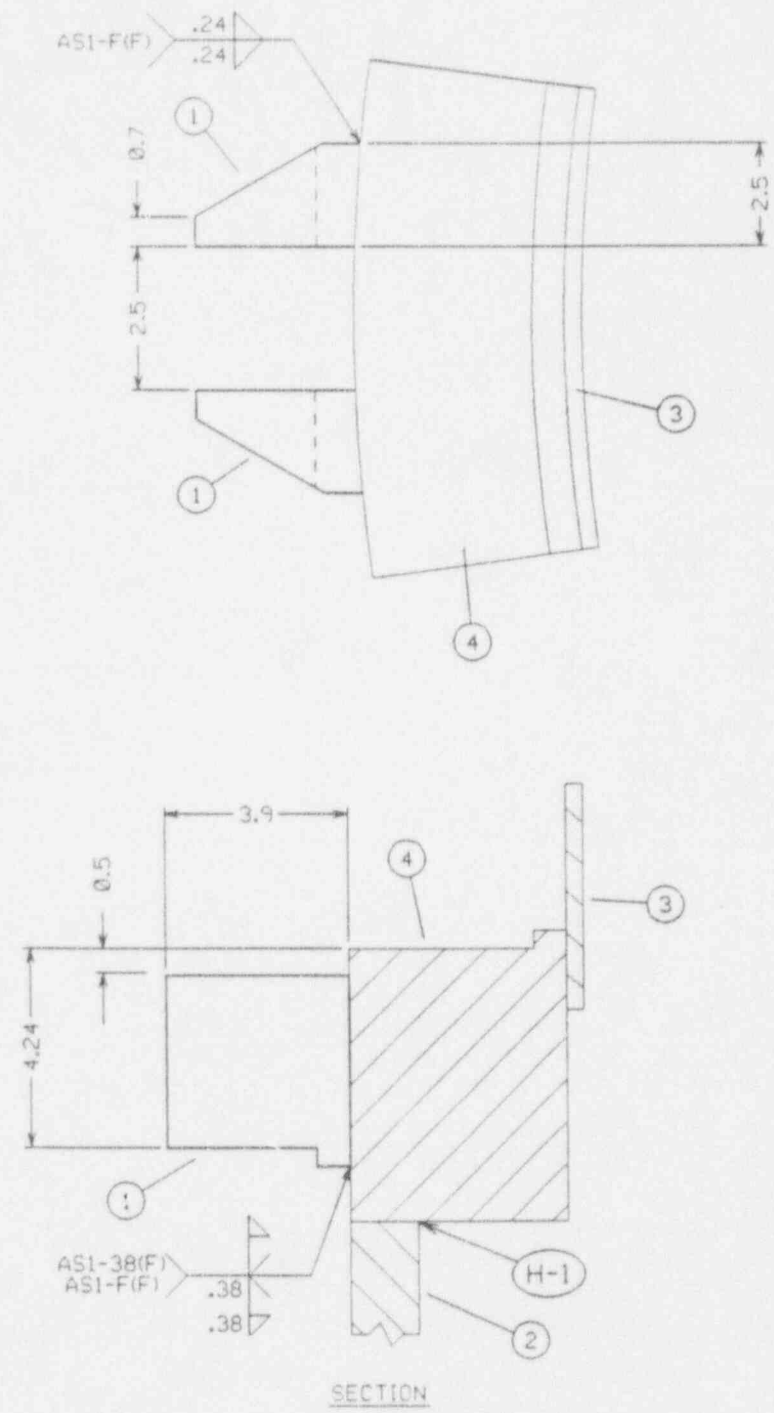
Reactor Pressure Vessel Sketch

SHROUD
CLOSURE FLANGE

DWG. NO. VS-2-04

REV. 1

NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	LUG	304L A240	
2	WALL PLATE	(SEE VS-2-4)	
3	FLANGE EXTENSION PLATE	(SEE VS-2-4)	
4	FLANGE	(SEE VS-2-4)	



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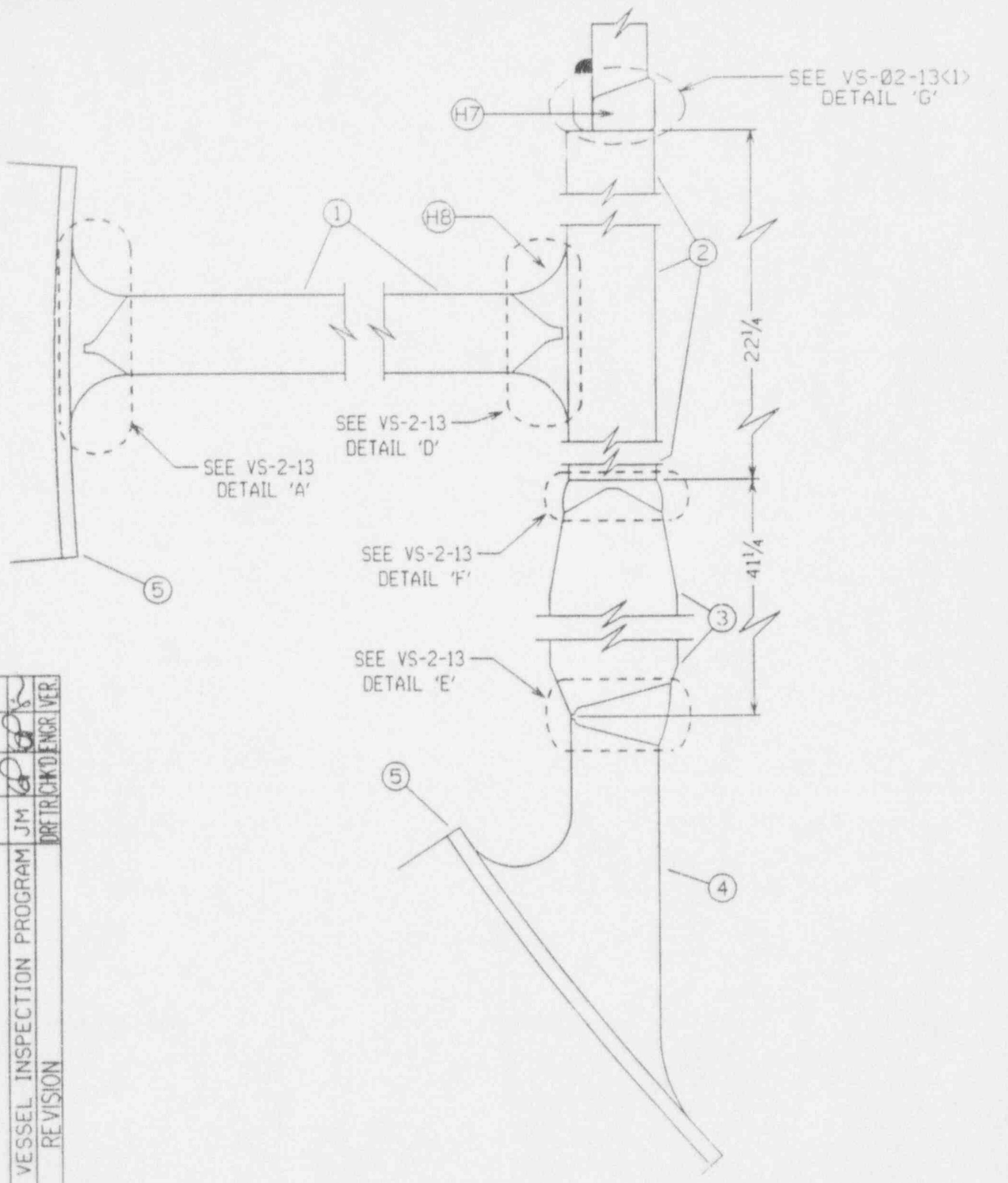
Reactor Pressure Vessel Sketch

SHROUD
CLOSURE FLANGE

DWG. NO. VS-2-05

REV. 1



NO.	DISCRIPTION	BILL OF MATL.	PROCED.
1	LEDGE SEGMENT PLATE	(SEE VS-2-11)	
2	CYLINDER PLATE	SB-168	
3	SUPPORT LEG	(SEE VS-2-12)	
4	SUPPORT STUB	(SEE VS-1-36)	
5	VESSEL	SA-533 CLASS 1 GR. B	

REFERENCE DRAWINGS:
APED-B11-2655-204

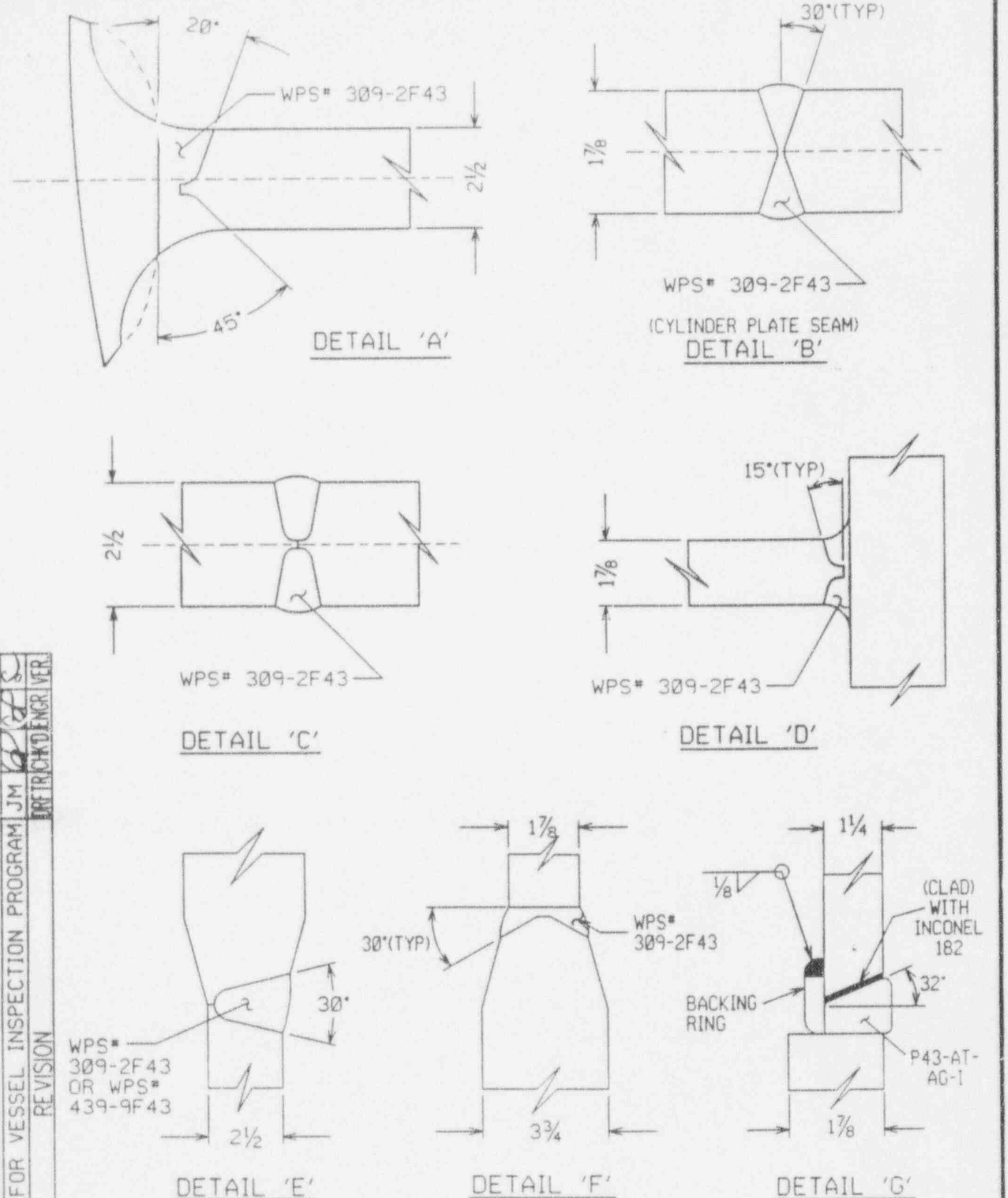
NO.	DATE	REVISION	DRFTR	CHK'D	ENGR	VER.
1	8-4-94	DRAFTED FOR VESSEL INSPECTION PROGRAM	JM			

IES:
Inservice Inspection Program
Reactor Pressure Vessel Sketch

SHROUD SUPPORT
ASSEMBLY

DWG. NO. VS-02-10

REV. 1



REFERENCE DRAWINGS:
APED-B11-2655-204
APED-B11-2655-205

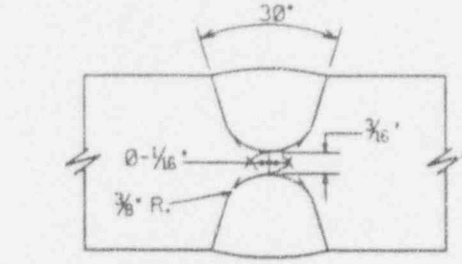
NO.	DATE	REVISION	DRFTR	CHK'D	ENGR	VER.
1	8-15-94	DRAFTED FOR VESSEL INSPECTION PROGRAM	JM			

IES:
Inservice Inspection Program
Reactor Pressure Vessel Sketch

SHROUD SUPPORT WELDS

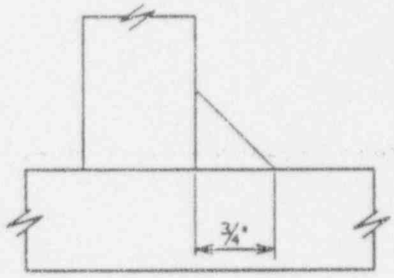
DWG. NO. VS-Ø2-13<1>

REV. 1

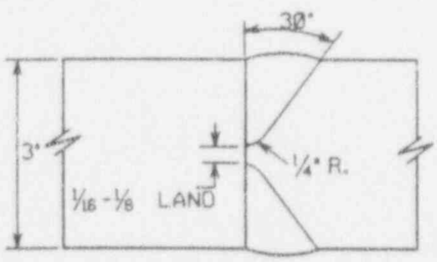


BACK-GOUGE PRIOR TO WELDING OPP. SIDE

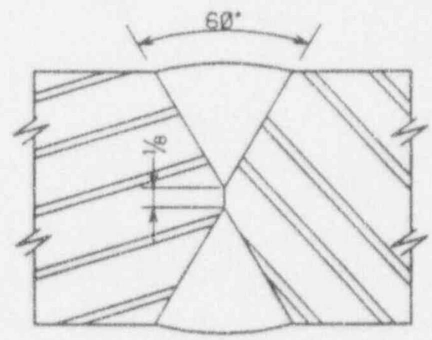
DETAIL H



DETAIL I



DETAIL J



DETAIL K

REFERENCE DRAWINGS:
APED-B11-2655-204
APED-B11-2655-205

NO.	DATE	REVISION	DRFTR.	CHK'D.	ENGR.	VER.
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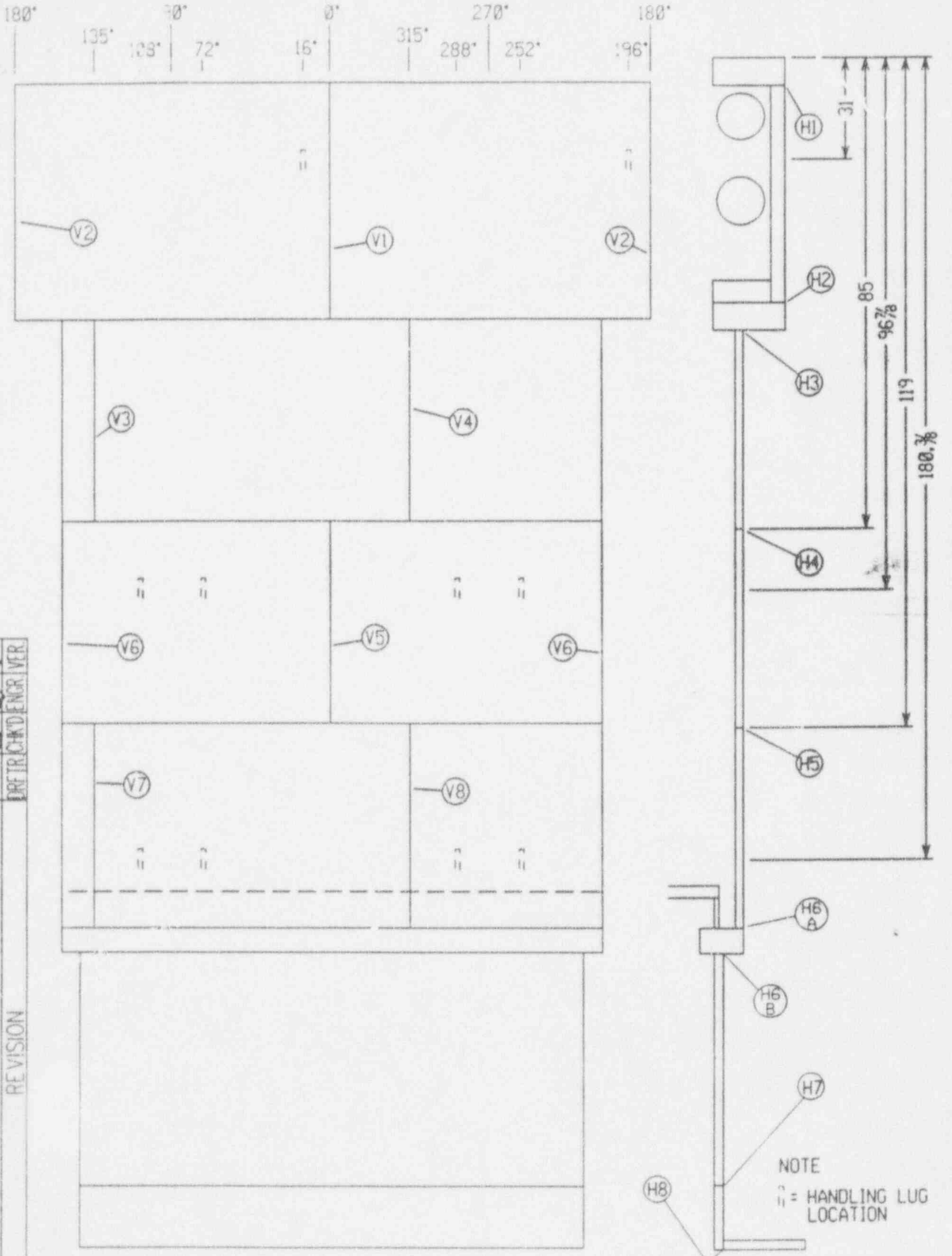
Inservice Inspection Program
Reactor Pressure Vessel Sketch

SHROUD SUPPORT
WELDS

DWG. NO. VS-02-13<2>

REV. 1



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IES:
Inservice Inspection Program
Reactor Pressure Vessel Sketch

SHROUD WELDS

DWG. NO. VS-2-14

REV. 1