BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

June 28, 1984
BECo. 84-095

Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing
Office of Nuclear Regulatory Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 205

License No. DPR-35
Docket No. 50-293

Dear Sir:

On May 2, 1984 a telephone conference took place regarding the Pilgrim Nuclear Power Station Inservice Inspection Program. Participants at this meeting were P. Leech from the NRC; G. Freund and R. Yorg from EG&G; and F. Famulari, T. Ferris, and M. Williams from Boston Edison Company. Eleven items were discussed, in which resolution was reached on NRC Items 1, 2, and 3. This letter provides information for resolution of the remaining items (4-11).

Item 4

NRC Request
Your letter to NRC (answer to NRC Question #2, BECo , #82-296, dated 11/15/84) mentioned that none of the welds in the recirculation pump casing were pressure-retaining. Please provide cross-section drawings of your pumps that show the casing welds.

BECo. Response
The requested reactor recirculation pump cross-section drawings are provided as Attachments 1 and 2 to this letter.

Item 5

NRC Request
Concerning relief request PRR-3, your answers to NRC Question #1 submitted in BECo letter 82-296, stated that some Class 1 valves had been disassembled for maintenance. What were the results of the visual examinations of their internal surfaces? If they have been previously furnished to the NRC, please document by reference.

A047
1/1

BECo. Response
The requested information has been submitted to the NRC in the following Inservice Inspection Final Reports:

1. 1977 Inservice Inspection of Pilgrim Nuclear Power Station, Unit 1-Volume 1.
2. 1980 Inservice Inspection of Pilgrim Nuclear Power Station, Unit 1-Volume 1.
3. 1981 Inservice Inspection of Pilgrim Nuclear Power Station, Unit 1-Volume 3.

Item 6

NRC Request
Please identify any class 1 valves (and their categories) that have been disassembled since November 1982. What were the results of the visual examinations on their internal surfaces? If they have been previously furnished to the NRC, please document by reference.

BECo. Response
No Class I valves have been disassembled from November 1982 through December 10, 1983. Pilgrim Nuclear Power Station commenced the current refueling outage on December 10, 1983. The valves, listed in Attachment 3, are or will be disassembled and a VT-3 visual examination of the internal surfaces will be performed during this outage. The results of the examination will be transmitted as part of the 1984 Inservice Inspection Final Report.

Item 7

NRC Request
Please provide a drawing or sketch of your reactor vessel that shows the locations of the longitudinal and circumferential shell welds. The drawing should indicate which welds are in the beltline (core) region. Also give the percentage of each beltline weld that is accessible, whether internally or externally, to either volumetric or remote visual (VT-1) examination (PRR-4).

BECo Response
A reactor vessel layout drawing delineating the beltline region welds and the portions accessible for ultrasonic examination from the vessel exterior is provided as Attachment 4 to this letter. As shown, ten percent of each longitudinal weld and five percent of each circumferential weld are accessible for examination. Due to the BWR vessel interior configuration, the beltline region welds are not accessible for ultrasonic examination from the vessel interior.

Also, it should be noted that the reactor vessel interior is in the as-welded cladded condition. Thus, a remote visual (VT-1) examination to determine the condition of the underlying beltline region is not a meaningful examination.

Item 8

NRC Request
Item B1.21 and B1.22 of Category B-A (PRR-5) require that the accessible portions up to 100% of one head weld in each direction be examined. It appears that your program meets code requirements for examining head welds. Please state why you believe that relief is needed.

BECo. Response
The reactor vessel bottom head meridional welds are accessible for ultrasonic examination. However, the configuration of the reactor vessel bottom head and vessel support skirt limits the examination of the outer circumferential weld to a one side only examination. See Attachments 5 and 6 for drawings depicting the accessible welds and the restrictions. Approximately sixty percent of the length of the reactor vessel bottom head welds is accessible for examination. Revised Relief Request No. PRR-5 is submitted as Attachment 7.

Item 9

NRC Request
The containment atmospheric control system (PRR-7) would seem to be exempt from code NDE by IWC-1220(b). Please state why you believe that relief is needed.

BECo. Response
IWC-1220 (b) states the following components are exempt from the inservice examination requirements of IWC-2500:
"(b) Components of systems or portions of systems, other then Residual Heat Removal Systems and Emergency Core Cooling Systems, that are not required to operate above a pressure of 275 psig (19.00 KPa) or above a temperature of 200°F (93°C)."

The containment atmospheric control system, CACS, at PNPS normally operates at a pressure of 1 psig and a temperature of 50°F. However, the CACS maximum operating pressure and temperature is 56 psig and 281°F, respectively.

IWC-1220(b) does not specifically state that the exemption applies to systems that normally do not operate above the specified values. Therefore, since the maximum operating temperature of 281°F exceeds the 200°F temperature required for exemption by IWC-1220(b), Relief Request No. PRR-7 was submitted.

Item 10

NRC Request
Of the code-allowed exemptions listed on pages 4-2 and 4-3 of your program, only one, EX-5, excludes ECCS and RHR systems. Please confirm that none of the other listed exemptions includes ECCS and RHR systems.

BECo. Response
Code-allowed exemptions, EX-3, EX-5 and EX-10, do not apply to the ECC and RHR systems at PNPS. The remaining listed exemptions do apply to the ECC and RHR systems.

Item 11

NRC Request
Concerning your Inservice Inspection Checklist for the 1983/1984 refueling outage (Enclosure A to BECo letter 83-290, dated 12/1/83), the nozzle-to-vessel weld item (B3.90) contains a remark that there is limited access. If the code examination requirements cannot be met on any nozzles, a relief request should be submitted that identifies the nozzles and degree of accessibility. Certain other items in the table (such as those in Category B-A and B-J) should be attributed to existing relief requests.

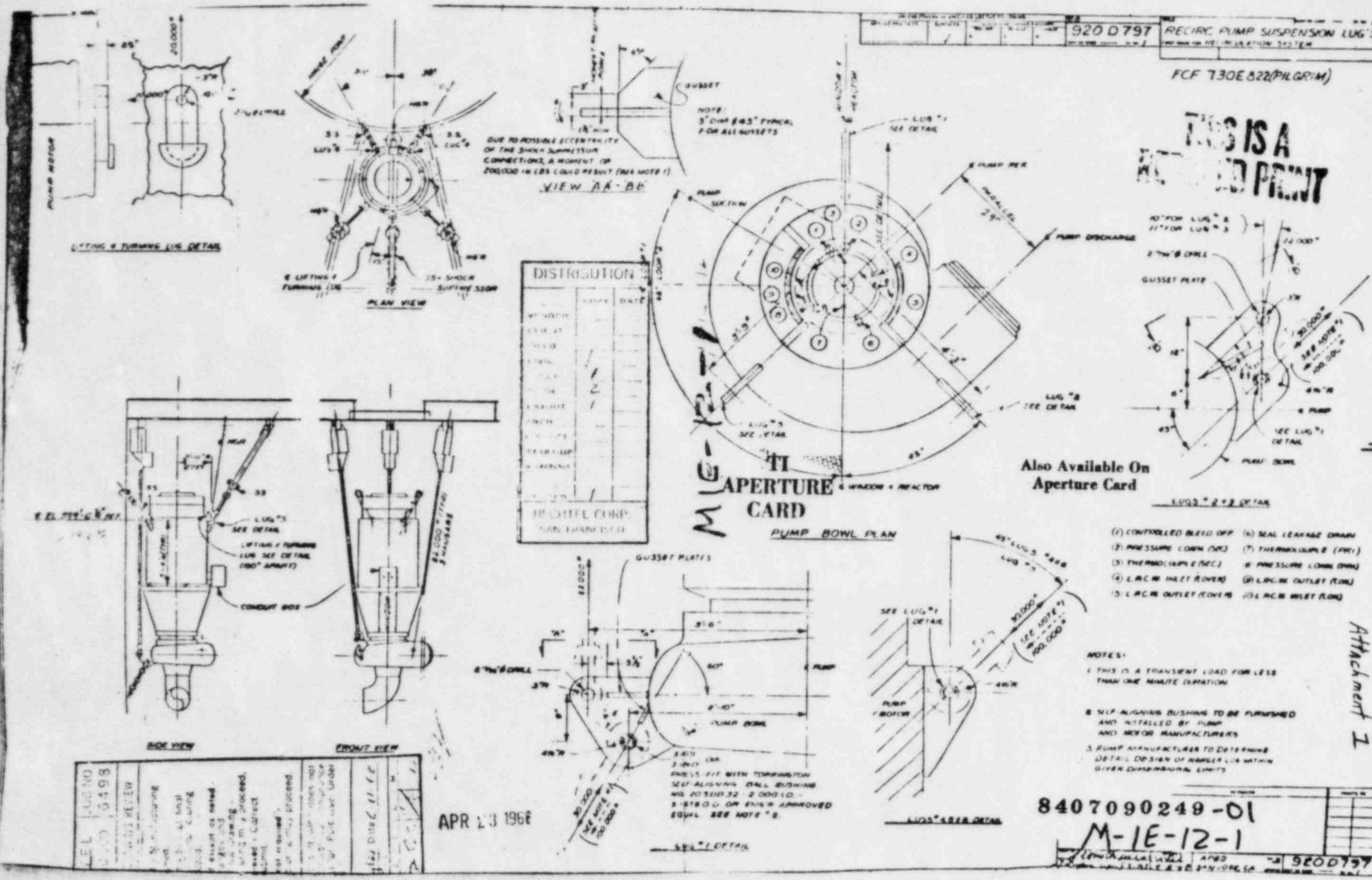
BECo Response
The code examination requirements cannot be met on the twenty-one nozzles located within the biological shield due to the reactor vessel insulation/biological shield configuration. Relief Request PRR-9 has been prepared and is submitted as Attachment 8 to this letter.

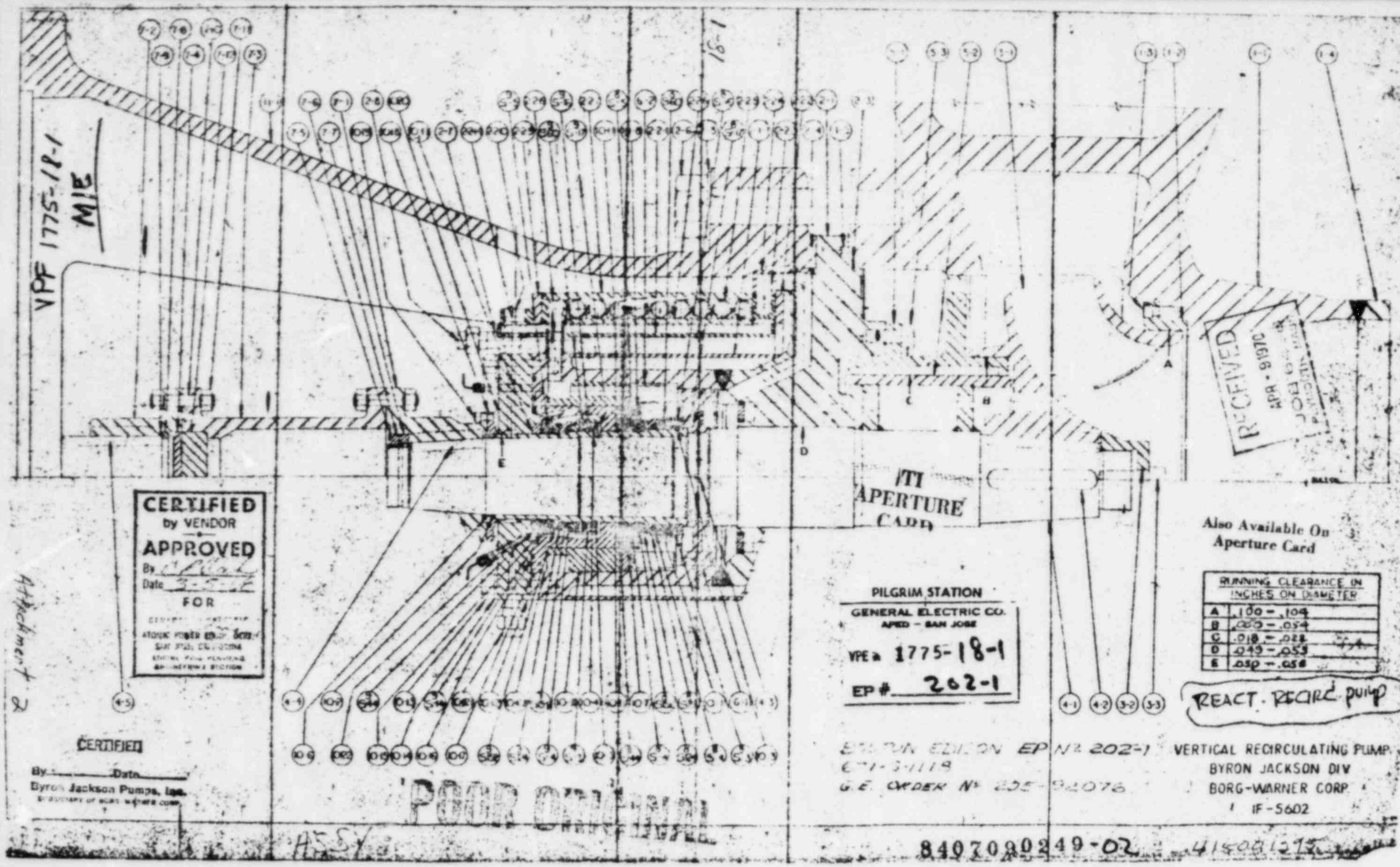
If after reviewing this information you have any further questions, please contact us.

Very truly yours,

W D Harrington

Attachments


920 D 797
RECIRC PUMP SUSPENSION LUG'S
FCF 730E 822(PILGRIM)
THIS IS A
PRINT
LIFTING & TURNING LUG DETAIL
PUMP MOTOR
VIEW AA-BB
PLAN VIEW
NOTE: 3" DIM & 45° TYPICAL FOR ALL GUSSETS
GUSSET
DUE TO POSSIBLE ECCENTRICITY OF THE SHOCK SUPPRESSOR CONNECTIONS, A MOMENT OF 200,000 IN LBS COULD RESULT (SEE NOTE 1)
PUMP SUCTION
PUMP DISCHARGE
PARALLEL
GUSSET PLATE
LUG #2 SEE DETAIL
LUG #1 SEE DETAIL
LUG #3 SEE DETAIL
SEE LUG #1 DETAIL
PUMP BOWL
DISTRIBUTION
BECHTEL CORP.
SAN FRANCISCO
TI
APERTURE
CARD
M-1E-12-1
Also Available On
Aperture Card
PUMP BOWL PLAN
LUGS # 2 & 3 DETAIL
(1) CONTROLLED BLEED OFF
(2) PRESSURE CONN (SEC)
(3) THERMOCOUPLE (SEC)
(4) L.R.C.W. INLET (COVER)
(5) L.R.C.W. OUTLET (COVER)
(6) SEAL LEAKAGE DRAIN
(7) THERMOCOUPLE (PRI)
(8) PRESSURE CONN (PRI)
(9) L.R.C.W. OUTLET (LOW)
(10) L.R.C.W. INLET (LOW)
CONDUIT BOX
SIDE VIEW
FRONT VIEW
GUSSET PLATES
PUMP BOWL
PRESS FIT WITH TORRINGTON SELF ALIGNING BALL BUSHING NO. 20 SDD 32 - 2.000 I.D. - 3.5/8 O.D. OR EQUIV APPROVED EQUAL. SEE NOTE # 2.
LUG #1 DETAIL
LUGS # 4 & 5 DETAIL
PUMP & MOTOR
NOTES:
1. THIS IS A TRANSIENT LOAD FOR LESS THAN ONE MINUTE DURATION.
2. SELF ALIGNING BUSHING TO BE FURNISHED AND INSTALLED BY PUMP AND MOTOR MANUFACTURERS.
3. PUMP MANUFACTURER TO DETERMINE DETAIL DESIGN OF HANGER LUG WITHIN GIVEN DIMENSIONAL LIMITS.
Attachment 1
APR 23 1968
8407090249-01
M-1E-12-1
920D797


VPF 1775-18-1
MIE
CERTIFIED
by VENDOR
APPROVED
By
Date
FOR
ATOMIC POWER
Attachment 2
CERTIFIED
By
Date
Byron Jackson Pumps, Inc.
POOR ORIGINAL
APERTURE CARD
Also Available On
Aperture Card
PILGRIM STATION
GENERAL ELECTRIC CO.
APED - SAN JOSE
VPE # 1775-18-1
EP # 202-1
RUNNING CLEARANCE IN
INCHES ON DIAMETER
A .100 - .104
B .050 - .054
C .018 - .022
D .049 - .053
E .050 - .054
REACT. RECIRC pump
RECEIVED
APR - 9 1970
EP Nº 202-1
G.E. ORDER Nº 205-94076
VERTICAL RECIRCULATING PUMP
BYRON JACKSON DIV
BORG-WARNER CORP.
IF-5602
8407090249-02

Attachment 3

VALVES SCHEDULES FOR DISASSEMBLY DURING RFO #6
AS OF MAY 18, 1984

GROUP	SYSTEM	VALVE NO.
1	HPCI	2301-4
3	Main Steam	203-1A
		203-1B
		203-1C
		203-1D
		203-2A
		203-2B
		203-2C
		203-2D
9	RHR	1001-29A
		1001-29B
		1001-50
13	Feedwater	6-58A
		6-58B
		6-62A
		6-62-B
19	Main Steam	203-3A
		203-3B
		203-3C
		203-3D
20	Main Steam	203-4A
		203-4B
		203-4C
		203-4D

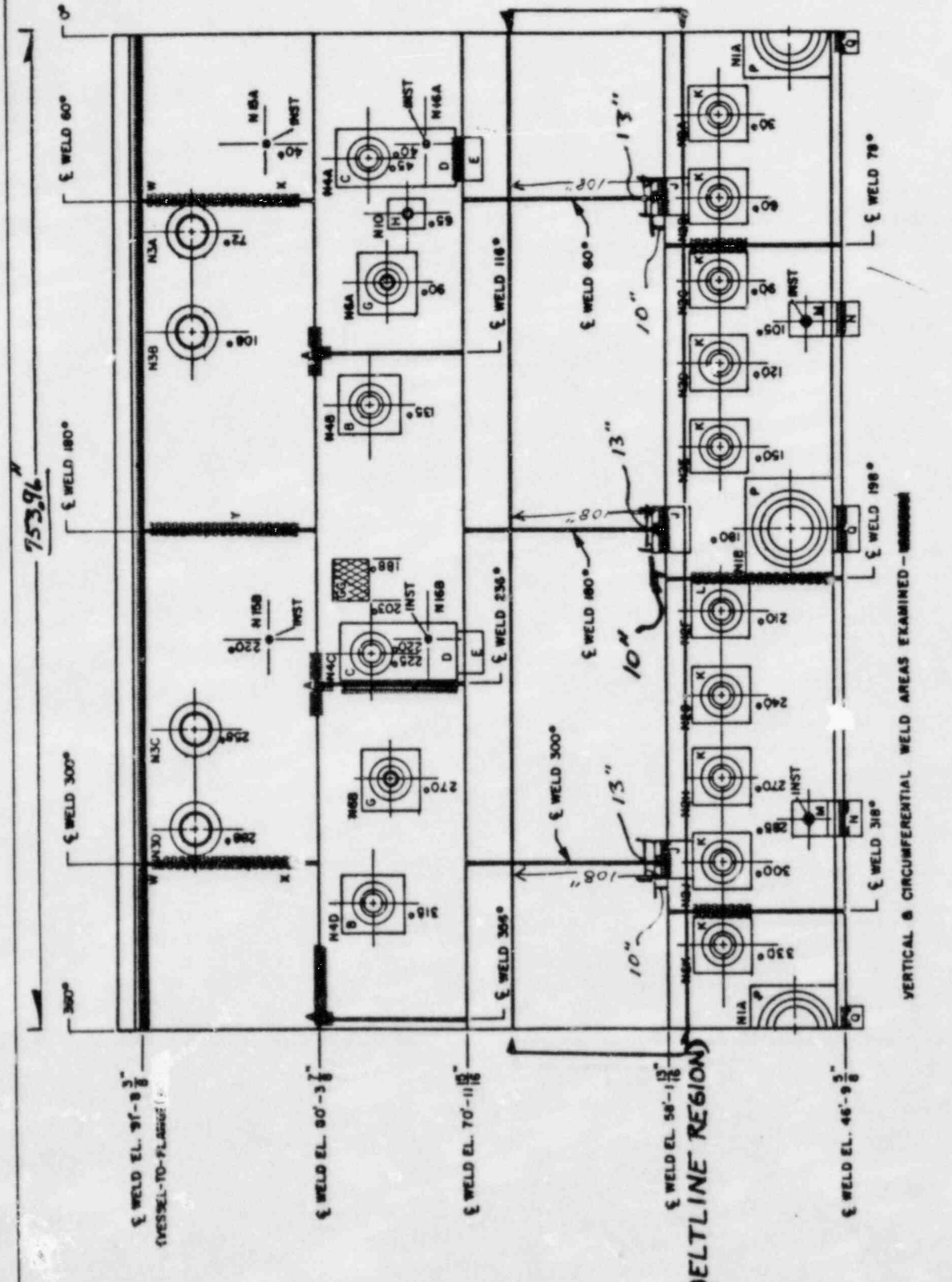


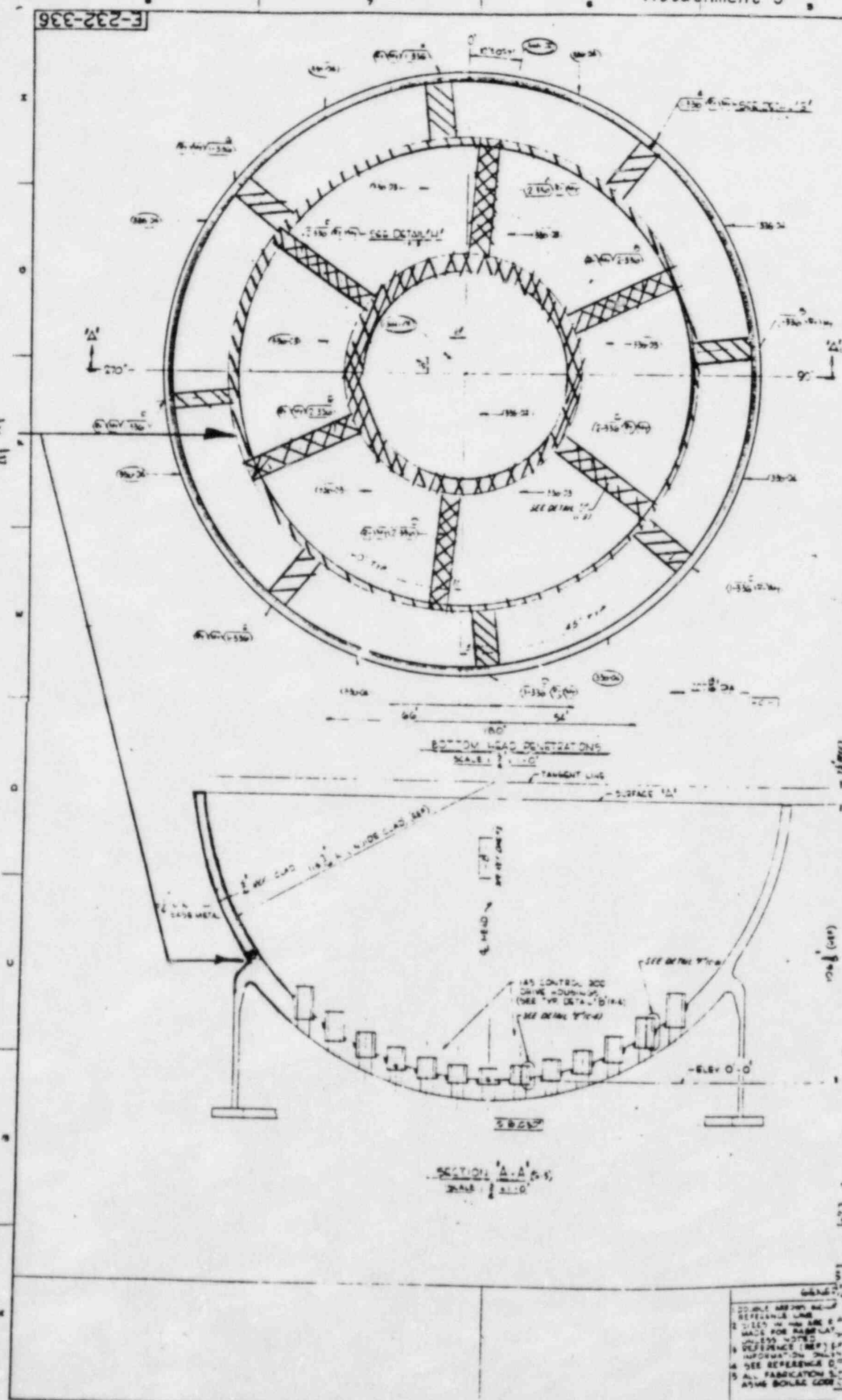
FIGURE B-1. REACTOR PRESSURE VESSEL

Attachment 5

WE CAN ONLY PERFORM A ONE SIDED EXAM DUE TO THE VESSEL SKIRT

Inaccessible Welds

Accessible Welds



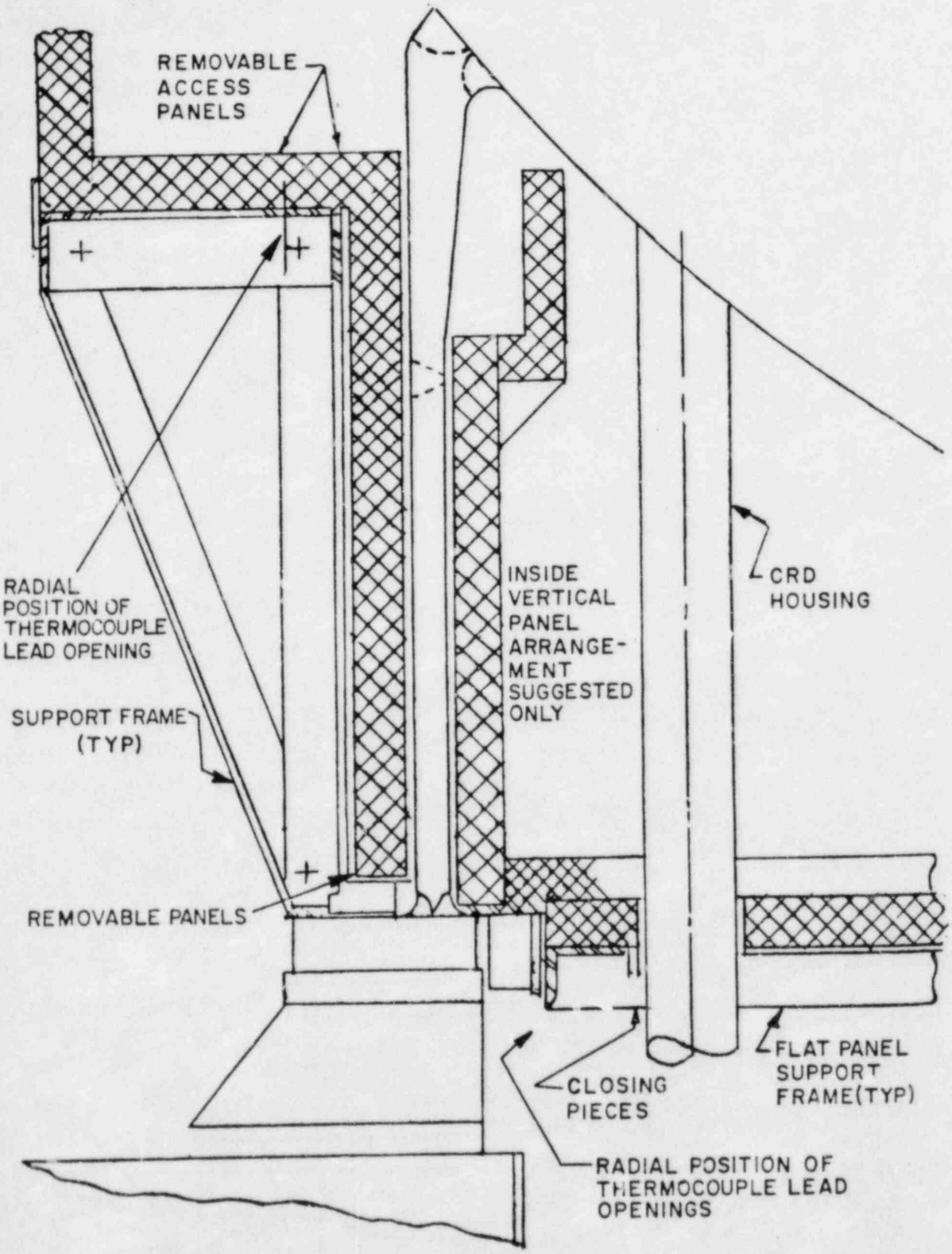


FIGURE 4.2-4
REACTOR VESSEL-SKIRT INSULATION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

Attachment 7

RELIEF REQUEST NO PRR-5 Rev. 1

I. IDENTIFICATION OF COMPONENTS AND IMPRACTICAL CODE REQUIREMENTS.

The reactor pressure vessel bottom head contains seventeen circumferential and meridional welds.

Section XI of the ASME Boiler and Pressure Vessel Code, 1980 Edition through the Winter 1980 Addenda requires a volumetric examination of 100 percent of the length of one meridional head weld and one circumferential head weld each inspection interval (Code Category B-A).

Relief is requested from the above mentioned Code requirements on the basis of inaccessibility.

II. BASIS FOR RELIEF

As discussed in Relief Request PRR-4, accessibility for examination of these welds was not considered in the plant design. The configuration of the vessel support skirt attachment weld and the reactor vessel skirt insulation limits the examination of the outer circumferential weld to a one side only examination.

III. ALTERNATE PROVISIONS

Boston Edison will perform a volumetric examination of the reactor vessel outer circumferential weld from one side only.

Attachment 8

RELIEF REQUEST NO. PRR-9

I. IDENTIFICATION OF COMPONENTS AND IMPRACTICAL CODE REQUIREMENTS

The reactor vessel is designed with 21 nozzle-to-vessel welds located within the biological shield weld.

The ASME Boiler & Pressure Vessel Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components", 1980 Edition, Winter 1980 Addenda, Table IWB-2500-1, Category B-D, Item Number B3.90 requires that a volumetric examination of each nozzle-to-vessel weld be conducted. These examinations are required to be completed each inspection interval.

II. BASIS FOR RELIEF

Relief is requested from the above mentioned code requirements on the basis that 100% accessibility is not permitted due to the reactor vessel insulation/biological shield configuration. Accessability for the examination of all nozzle-to-vessel welds was not provided for in the original plant design which occurred prior to the issuance of Section XI requirements.

III. ALTERNATE PROVISIONS

The reactor nozzle-to-vessel welds will be volumetrically examined to the extent practicable.