APAE NO. 49
Supplement No. 1

46

HAZARDS SUMMARY REPORT PREPACKAGED NUCLEAR POWER PLANT FOR AN ICE CAP LOCATION (PM-2A)

ALCO

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT
P. O. BOX 414, SCHENECTADY 1, N. Y.

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ERRATA

(AMENDED AUG. 5, 1960)

APAE No. 49 Supplement No. 1

HAZARDS SUMMARY REPORT
PREPACKAGED NUCLEAR POWER PLANT
FOR AN ICE CAP LOCATION. (PM-2A)

The following changes have been made in the subject report:

Page 2, Section 1.1, fourth line in first paragraph: "reduction is size" should read "reduction in size".

Page 3, Section 1.3:

Revise sentence two, par. 1 to read "The amount of waste disposed of in this manner is estimated to be approximately 7,500 gallons per year."

In the sentence beginning on line 7 of this section, the figure 50 μc/yr should read 50 mc/yr. Delete μ here and replace by m.

710-001

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APAE No. 49
Supplement No. 1
Copy No. 46

HAZARDS SUMMARY REPORT
PREPACKAGED NUCLEAR POWER PLANT
FOR AN ICE CAP LOCATION
(PM-2A)

Editor:
E. M. Reiback

Contract DA-30-347-ENG-284
with U.S. Army Engineer District, Eastern Ocean

Issued: April 15, 1960

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INTRODUCTION

This is a supplement to APAE No. 49, "Hazards Summary Report for a Prepackaged Nuclear Power Plant for an Ice Cap Location (PM-2A)" issued by Alco Products, Inc. on July 1, 1959.

As a consequence of several design changes in the PM-2A Nuclear Power Plant after the preparation of the original Hazards Summary Report, amendments to this report are required to reflect these changes. These amendments are presented here. Further, an errata list has been prepared for APAE No. 49.

In response to questions from the Danish Government, on whose land this reactor will be built, an addendum has been prepared containing additional information which was not in the original Hazards Summary Report.

There is no change in the conclusions of the original report. There will be no hazard to personnel when the Nuclear Power Plant is operating under normal conditions, and there is no major hazard to personnel should the maximum credible accident occur.

1.0 ADDENDUM TO APAE No. 49

1.1 Detector Channels for Startup and Intermediate Power Range

(pages II-8, II-9, and VI-1 to VI-4)

The startup range is protected by one detector channel and the intermediate power range is protected by one detector channel. This design philosophy is inherent in a packaged plant where reduction is size and number of components is a necessity. This reduction has not been made at a sacrifice of plant safety. The instrumentation has been designed to fail safe and two features have been incorporated in the instrumentation system to increase reliability.

1. A lifting mechanism permits the operator to withdraw the startup chamber from the region of high neutron flux above a power level of 4 megawatts. This decreases chance of malfunction due to radiation damage and increases the useful life of the chamber.

2. A checking device has been added to the BF_3 startup channel, in order to check startup channel performance. With this unit, the BF_3 chamber will be checked prior to each startup to insure that the chamber is capable of sensing neutron flux, thus making plant operation more reliable. A neutron source has been attached to the outside of the instrument well approximately five feet above the operating level of the BF_3 chamber. Before startup, the BF_3 chamber is lifted to a point opposite the neutron source. Chamber output is monitored at the console for indication of count rate. After the check, the chamber is lowered into the operating position for startup. This further increases system reliability.

We have maintained 2 out of 3 coincidence in the power range. This insures safety and reliability in the range used most generally in plant operation. To further increase safety and reliability the outputs of each of these power range channels are compared and a variation of $\pm$ 5% is annunciated so that corrective action can be taken.

1.2 Camp Heating, Cooking and Water Supply

Camp cooking and heating will be done electrically. The primary use of the steam produced in the tertiary loop will be

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to melt snow for the camp water supply. However, this steam will be available for other uses in the camp should it be desired. Because this steam is produced in a tertiary loop, it has practically no chance of transferring radioactivity to the uncontrolled environment regardless of its use.

1.3 Waste Disposal (Chapter VI par 9.0)

Liquid waste disposal:

Liquid radioactive waste which contains between 10^{-3} and 10^{-4} $\mu c/cc$ will be disposed of directly in the ice cap. The amount of waste disposed of in this manner will not exceed 7500 gallons per year. This disposal will take place a safe distance from the camp. The amount of activity disposed of in this manner will not exceed 50 μc/year, and the concentration will not exceed 10^{-3} $\mu c/cc$. Any waste in excess of these limits will be transported out of Greenland for disposal.

A record of all amounts of liquid radioactive wastes, including volume, activity level, and total curies deposited in the ice cap will be kept and presented to the Danish Authorities once each year.

Solid Waste Disposal:

The disposal of the spent demineralizer in the ice cap as described on Page VI-16 will not be performed. Instead all solid wastes will be shipped from Greenland for disposal. Extra demineralizer casks will be provided for this purpose.

1.4 Psychological Experiments

All psychological experiments will be conducted in a manner which will not jeopardize the safety of the plant. Such studies will be closely controlled by competent specialists. Participation of the PM-2A crew in these experiments will be contingent upon no adverse effect upon them or the plant. The psychological problems associated with isolation have been considered and guidelines have been established by the Army Surgeon General for selection of stable individuals for PM-2A crews.

1.5 Nitric Acid Formation in the Dry Cap

If air is allowed to remain in the PM-2A dry cap, it is possible for nitric acid to form, under the influence of ionizing radiation. With the proper conditions, most of the oxygen in air can be consumed in the formation of nitrogen dioxide and nitrous oxide. The condition most conducive to the formation of these oxides are a temperature of about 175°C, a pressure of more than 5 atmospheres and a gamma intensity of more than 10^7 R/min.

The conditions for the PM-2A dry cap are not this severe, but sufficient ionizing radiation will be present to cause fixation of some nitrogen and a subsequent reaction with the residual moisture to form nitric acid. This reaction would most probably take place on the underside of the dry well cover which is cooled by the shield tank water.

For the parts of the dry cap constructed completely of low manganese stainless steel, the presence of nitric acid presents no problem. However, the studs and nuts which fasten the reactor vessel cover to the vessel are low alloy steel and carbon steel respectively; both are nickel-cadmium plated. Neither nickel or cadmium possesses useful resistance to corrosion by solutions of oxidizing acids.

To prevent attack on the studs or nuts a simple conical shield or baffle is inserted over the reactor head. This shield prevents any nitric acid formed on the underside of the dry well cover from possibly reaching the studs or nuts. The studs and nuts themselves will be operating well over the boiling point of water and it should not be possible for nitric acid to form directly on their surfaces.

1.6 The Reactor Vessel Cover as a Missile

A study was made of the reactor vessel cover as a probable missile through the vapor container dry well and vapor container cover and the conclusions are stated below.

Vapor Container Missile Penetration:

The reactor cover can become a dangerous missile only if all the vessel studs fail instantaneously. If this happens, the reactor vessel cover can cross the dry well and destroy the vapor container access cover.

However, such a failure is not considered a realistic possibility. If stud failures occur, they will occur in a progressive manner. Substantial leakage will occur before all the studs have failed in the case of a progressive symmetrical stud failure, as well as in the case of progressive non-symmetrical stud failure. This leakage will relieve the pressure on the cover and will eliminate the missile danger.

Calculated results show that in an instantaneous failure of all 18 bolts, the reactor cover would strike the vapor container cover after 0.02 seconds at a velocity of 90 ft/sec and would destroy the container cover. However, if leakage is assumed to start when gasket pressure is zero, there will be 8 bolts remaining when bolt stress has reached a value of 30,000 psia. When sufficient leakage has taken place to reduce the stress below the yield point of the bolts, there will be 4 bolts left to restrain the cover from becoming a dangerous missile.

Despite the revision in the fabrication and assembly of the vapor container (two flanges versus original field welding), information contained in Chapter III, section 1.5 and in Chapter IV still pertains to the design and no elaboration is needed on the revised flanged vapor container.

1.7 Effects of Core Meltdown on Reactor Vessel and Vapor Container

An analysis was performed to determine the behavior of the PM-2A pressure vessel and vapor container after a loss of primary coolant and the subsequent meltdown of the core.

In the analysis of the core meltdown, the water level inside the vessel after the primary system rupture and subsequent blowdown was assumed to remain just sufficiently high to prevent natural circulation of air from cooling the core. For this analysis it was assumed that this water would remain as a heat sink only until the time that the melted core arrives at the bottom of the vessel. This analysis was done assuming no water present after the core reaches the bottom of the vessel. If no water had been present after the blowdown, the effect would have been to delay considerably the time for core meltdown. The analysis was performed for the pressure vessel to determine if and when it would melt.

Three time intervals were assumed based on the preliminary analysis of the problem.

1. For a short time after the melted core has arrived at the bottom of the vessel, the heat removal rate from the puddle is greater than the heat generation rate. Part of the puddle solidifies at first, and as the heat removal rate decreases, this part of the puddle reliquifies. During this time, the lumped puddle average temperature was assumed constant. The vessel wall temperature is steadily increasing. This part of the problem ends when all the core becomes liquid.

2. During the next time interval, both the average puddle temperature and the average vessel wall temperature increase. This section of the problem ends when the average wall temperature reaches the melting temperature.

3. The final time interval involves the melting of the vessel wall. During this time, the puddle temperature continues to increase. The problem ends when the vessel melts through.

The following assumptions were imposed on the problem:

1. Only 60% of the core will melt down.

2. Only 80% of the decay power will be available in the core for heat production, due to the passage of gamma radiation from the molten material without absorbtion.

3. Meltdown of 60% of the core will be completed 330 secs. after loss of coolant.

4. The melted part of the core will arrive in one lump 510 sec. after loss of flow. This additional time delay was imposed because the melting core must first drop onto the pinion carrier plate and then melt thru the perforated baffle before it can flow off the carrier plate and arrive at the bottom of the vessel.

5. After 510 sec., decay heat will be generated only from fission products that are not volitile above 1200°C.

6. A constant radiation heat sink of 500°F was used.

7. Water is no longer present in the vessel during this stage of the problem.

The result of this analysis shows that the vessel will melt through approximately 26.5 minutes after loss of coolant. The puddle temperature at meltout is 3075°F.

The vapor container was then analyzed in a similar manner, for which several problem time intervals were set up for specific conditions of the melted core and container wall.

1. In the first time interval, the puddle will be quickly chilled to the fusion point and the thin container wall will rapidly rise in temperature.

2. In the second time interval, the puddle will solidify at constant temperature and the container will reach the diluted core temperature.

3. In the third time interval, the following assumptions were made.

   a. The container walls are in equilibrium with the water and vapor in the container at 350°F.

   b. The equipment in the vapor container represents a relatively constant temperature radiation heat sink.

   c. Heat is conducted through the container wall and radiated out of this wall to a constant temperature heat sink.

   d. The surface tension of the melted core is low and the puddle will spread out along the bottom of the container, producing relatively large radiation and conduction heat transfer areas.

   e. No heat was removed by the water in the vapor container.

   f. The decay heat per unit volume has been reduced by the dilution of the melted core by the melted vessel wall.

Because of the short time involved for the wall temperature to reach core temperature and because of the large heat loss from the puddle, preliminary calculations indicated that the container wall might not melt or that if it did this would occur after a considerable time. The condition of incipient wall melting was set up with the core having a heat generation rate which would occur at 10^{-4} sec. after loss of coolant. Calculations, based on estimated heat transfer areas, show that initially the heat conduction from the puddle is the dominant factor and reduces the puddle temperature within seconds to the fusion temperature of 2640°F. The container wall temperature rise was calculated at 425°F/sec. During the time when the puddle is solidifying, the container wall temperature will reach the core temperature. Heat loss from the puddle will then be predominantly by radiation.

The radiation loss was calculated at this time, to be 2.5 times the heat generation rate. For the radiation loss to just balance the heat generation rate, all equipment in the container must be at 2470°F with no conduction to the secondary system, and no extended surface benefit obtained from the vapor container around the periphery of the puddle. Radiation losses from the puddle are sufficient to keep the wall temperature from reaching the melting temperature.

The assumption of a large radiation area seems justified, as the surface tension of the puddle should not be large enough to keep it from spreading. This is so even if a thin layer of the puddle immediately solidifies on the container wall.

The following conclusions are reached:

1. The core will melt through the reactor vessel 26 1/2 minutes after loss of coolant.

2. Based on the predominantly conservative assumptions, it is concluded that the vapor container is safe from meltout.

1.8 Contamination Due to Leakage

The skid mounted system will furnish steam to melt snow for the water supply for Camp Century. Heat is transferred from the primary coolant to produce the secondary steam. Part of the secondary steam will be used in a heat exchanger to produce steam for melting snow for the camp water supply. It is of interest to evaluate the potential radiation hazard produced by simultaneous leakage between primary and secondary loops, and the secondary and tertiary loops under operating conditions and in the case of a fuel rupture.

As a worst case it is assumed that a leak of up to 7.5 gallons/hr. from the primary system could occur without operating personnel being cognizant of the leak for approximately 24 hours. This may happen because under normal operating conditions, water from the oxygen analyzers is continuously discharged to the radioactive waste storage tank. This loss is compensated by makeup water added to the primary makeup tank.

Detection of a minor leak from the primary to the secondary system is readily accomplished by the radiation monitoring instrument in the hot well of the condenser. The radiation monitoring system is an integral part of the skid-mounted plant and alerts operators that a potential hazardous condition is developing. However, it is assumed that this instrument is inoperative for a period of time.

Thus, based on the following assumptions the radioactive concentration in the base water supply was calculated.

1. The primary system contains a maximum of 19.6 μc/cc of activity from corrosion products.

2. The leakage rate into the secondary system is 7.5 gals/hr.

3. 1% of any activity leaking to the secondary is carried over with the steam (based on figures on BWR plants and steam separator efficiency).

4. A leak rate of 7.5 gals/hr from the secondary to the tertiary loop occurs simultaneously with the leak in the primary steam generator.

5. 1% of any activity in the secondary system is carried over with the steam to the tertiary loop.

The resulting concentration of radioactivity in the camp water supply at 32°F is 3×10^{-8} μc/cc. The maximum permissible concentrations of the most restrictive corrosion product (FE 59) is 6×10^{-4} μc/cc.*

If it is also assumed that a 1 1/2 inch section of a fuel plate is ruptured and 1% of the fission products in the section are released to the primary water, then there will be an increase in the activity level of the primary water to 86.5 μc/cc of mixed fission products and corrosion products. The resultant concentration in the drinking water based on the assumed leakage rates would be 1.3×10^{-7} μc/cc. Maximum permissible concentration for the limiting fission product (Sr 90) is 10^{-6} μc/cc.*

Since without benefit of radiation monitoring instrumentation a leak of this size would be discovered in at least 24 hours due to the loss of water and appropriate corrective action taken, these radiation hazards are negligible.

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* This NPC (Ref. NBS Handbook 69) is based on human consumption for 168 hrs/week for 50 years.

1.9 Reporting of Major Nuclear Accidents and Activity Levels

As requested, any major accident that happens to the reactor will be promptly reported to the Danish authorities. The results of pre operation environmental activity level studies will be reported, and later environmental activity levels will be reported to the appropriate Danish authorities once each year.

015

2.0 AMENDMENTS TO APAE No. 49

Several changes were made in the design of the PM-2A plant after the first hazards report, APAE No.49, was prepared. The principal design changes which affect the hazards report are the addition of a vapor container, elimination of the use of steam from a tertiary loop for space heating, and elimination of the local disposal of solid radioactive waste. The changes in the report caused by these and other modifications are shown below.

Page I-1 line 8 -
Delete "area heating" and replace by "snow melting".

Page I-1 line 13 from bottom -
Delete "1615" and replace by "1560".

Page I-1, lines 11-12 from bottom -
Delete "for space heating" and replace by "in a tertiary loop to melt snow for camp water supply and for other possible uses".

Page I-2 paragraph 4.1 -
Wherever 8.0 in Hg appears, add "abs." Change 1930 to 1980; change 1610 to 1560; change 315 to 420.

Page 1-3, lines 13-14
Delete these lines. Replace by:
B-10 26.169 gms.

B_4C 4.935 gms.

Page 1-3, line 5 from bottom. Replace 543.06 with 542.34

Page 1-3, line 4 from bottom.
Delete this line. Replace by:
B-10 .72815 gm.

Page 1-3, line 2 from bottom. Replace 422.72 with 427.20

Page 1-3, bottom line.
Delete this line. Replace by:
B-10 0.5735 gm.

Page 1-7, paragraph 4.4 -
Primary pressure psia - change 1970 to 1925, 1900 to 1880, 1580 to 1575 and 1660 to 1635.

Page I-8, line 1 from bottom -
Change "304 stainless steel" to "inconel".

Page I-9, line 4 -
Operating pressure - shell side - change 465 to 480.

Page I-9, line 13 -
Heat Transfer Area - change 893 ft^2 to 897 ft^2.

Page I-9, line 2 from bottom -
Pressurizer - length - change 82-1/4" to 83-3/4".

Page 1-10, line 2 -
Wall Thickness - Change 1-15/16" to 2-1/16".

Page I-11, line 12 -
Primary Make-Up System - After "mixed bed" resin add "demineralizer".

Page I-11, line 16 -
After "overall dimension" add "of primary make-up tank".

Page I-11, lines 17-24 -
Change "primary coolant pump feed duplex" to - "primary make-up pump". Max. capacity is 104 gph. Number installed is one, not two. Change "Control rod seal water return pumps" to "seal leakage pump". Change discharge pressure from 5 psia to 81.4 psig.

Page I-13, line 4 -
Change 465 psia to 480 psia.

Page I-13, line 7-8 -
Change extraction pressure to a high of 100 and a low of 41.

Page I-13, line 9 -
Change speed from 6000 rpm to 7450 rpm.

Page I-13, line 5 from bottom -
Condenser - change steam flow from 37,700#/hr to 29,750#/hr.

Page I-13, line 1 from bottom -
Change tubes from 7/8 in-16 BWG to 3/4 in-17 Bwg.

Page I-14, line 3-4 -
Feedwater Heater - make plural

Page I-14, lines 6-7 -
Under "Operating Pressure." - Change 800 psig to 600 psia and 350 psig to 92 psia.

Page I-14, lines 9-10 -
Change psig to psia and change 125 to 110.

Page I-14, line 11 -
Change low to 37,500. Note: the low and high here refer to the low and high pressure heaters. Design Temperatures are: High 350°F and Low 300°F.

Page I-15, line 6 -
Condenser circulating pumps -
Change head from 65 ft. to 70 ft.

Page I-15, line 8 -
Change overall length from 36'-10" to 36'-0-3/8"

Page III-4, lines 16 and 17 -
Interchange "top" and "bottom" on the two lines.

Page III-4, bottom line -
Change 37,300#/hr to 37,700#/hr.

Page III-5, line 1 -
Change 4-inch pipe to 6-inch pipe.

Page III-5, line 14 -
Change to read "Twin safety valves... serve to protect..."

Page III-5, line 16 -
After "8 inches Hg" add "abs."

Page III-5, line 4 from bottom -
Change "a pair of" to "three"

Page III-6, line 11 -
Change 1800 RPM to 1200 RPM

Page VI-11, line 9 from bottom -
Delete "for space heating"

Page VI-11, line 4 from bottom -
Delete "space heating" and replace by "melting snow for a water supply and other possible uses"

Page VI-6, line 7 -
Delete "is fabricated on the site and". Add after sentence ending on line 11, "The container is to allow zero leakage when filled with 1 psi air and 1 psi helium and tested with a Consolidated Engineering helium leak detector".

Chapter IX, section 3 & 4 -
The plant will be built with a vapor container, so the calculations used in this section for no vapor containment are applicable only in the event the vapor container is ruptured.

The following changes in the hazards report are a consequence of the post publication addition of a vapor container to the plant.

Page IX-1 -
Title. Delete "Without the Presence of a Vapor Container" and replace by "In the Event of a Catastrophe which Ruptures the Vapor Container".

Page IX-3, line 5 -
After "cloud" insert "through a break in the vapor container"

Page IX-11, line 15 from bottom -
Delete "Without Vapor Containment" and replace by "Due to a Catastrophe"

Page IX-12, line 5 -
Delete "deleted" and replace by "ruptured"

3.0 ERRATA LIST FOR APAE No.49

Page I-8, line 4 -
Delete "3/16" and replace by "1/8"

Page I-8, line 8 -
Delete "28-3/8" and replace by "28"

Page I-10, line 17-18 -
Delete "1/6 HP" and replace by "1/8 HP". Delete "1750 RPM" and replace by "1080 RPM"

Page I-10, line 1 from bottom -
Delete "planetary" and replace by "reduction gearing"

Page I-11, line 5 -
Delete "440°C" and replace by "17-4 PH"

Page I-12, line 11-14 -
Replace 69 1/2 with 70; 64 1/2 with 65; 61 with 61 1/2; 59 with 59 1/2

Page II-2, lines 3 and 4 from bottom -
Delete "Below the lower" and replace by "Above the upper"

Page II-2, line 1 and 2 from bottom -
Delete "equalize the pressure drop" and replace by "provide the proper balance of flow"

Page II-5, line 11 from bottom -
Delete "brake and"

Page II-6, line 9 -
Delete "20 lbs." and replace with "27 lbs."

Page IV-2, line 14 from bottom -
Delete "decays" and replace by "develops"

Page V-6
Delete table V-3 and replace with the following table.

TABLE V-3

DESCRIPTION OF REACTOR SHIELD - AXIAL

Description	Material	Distance from Center of Core to Outer Surface, Inches	Thickness Inches
Core		11	11
Reflector	Water	45.6875	34.6875
Clad	S.S.	46.8125	0.125
P.V. Cover	Steel	48.1875	1.375
Insulation	Glass Wool	50.1875	2.00
Wall	Steel	50.375	0.1875
Neutron and Gamma Shield	Shield Water	178.375	128.0

Page VI-1, line 11 -
Add "per second" after the word "counts"

Page VI-12, line 14 from bottom -
Delete "Fe 56" and replace by "Fe 59"

Page VII-1, line 5 from bottom -
Delete "minimum" and replace by "maximum"

Page IX-1, lines 7 and 8 -
Delete "northwest" and replace by "northeast"
Delete "east" and replace by "west".

Page IX-9, line 13 -
Delete Mr/hr and replace by mr/hr.

4.0 FIGURES

The various design changes which were made in this plant since APAE No.49 was published are also reflected in the following drawing changes:

Old Figure No.	New Figure No.	Title
III-2	III-2A	Vapor Container
None	III-3	Vapor Container Head Assembly
None	III-4	Internal Water Tank
None	IV-3	Shielding between Primary Shield Tank and Vapor Container Head
V-1	V-1A	Plant Layout
V-2	V-2A	Spent Fuel Handling System
VI-5	VI-5	Decay Heat Cooling

The piping and instrument diagram (Fig. III-1, on Page III-7 of APAE No. 49) has not been reproduced for this report. Two changes have been made in this diagram which affect potential hazards.

1. A piping loop seal was installed between the hot waste tank and vent pipe as a safety precaution.

2. A by-pass system with an additional relief valve around the primary make up pump was included in the design.

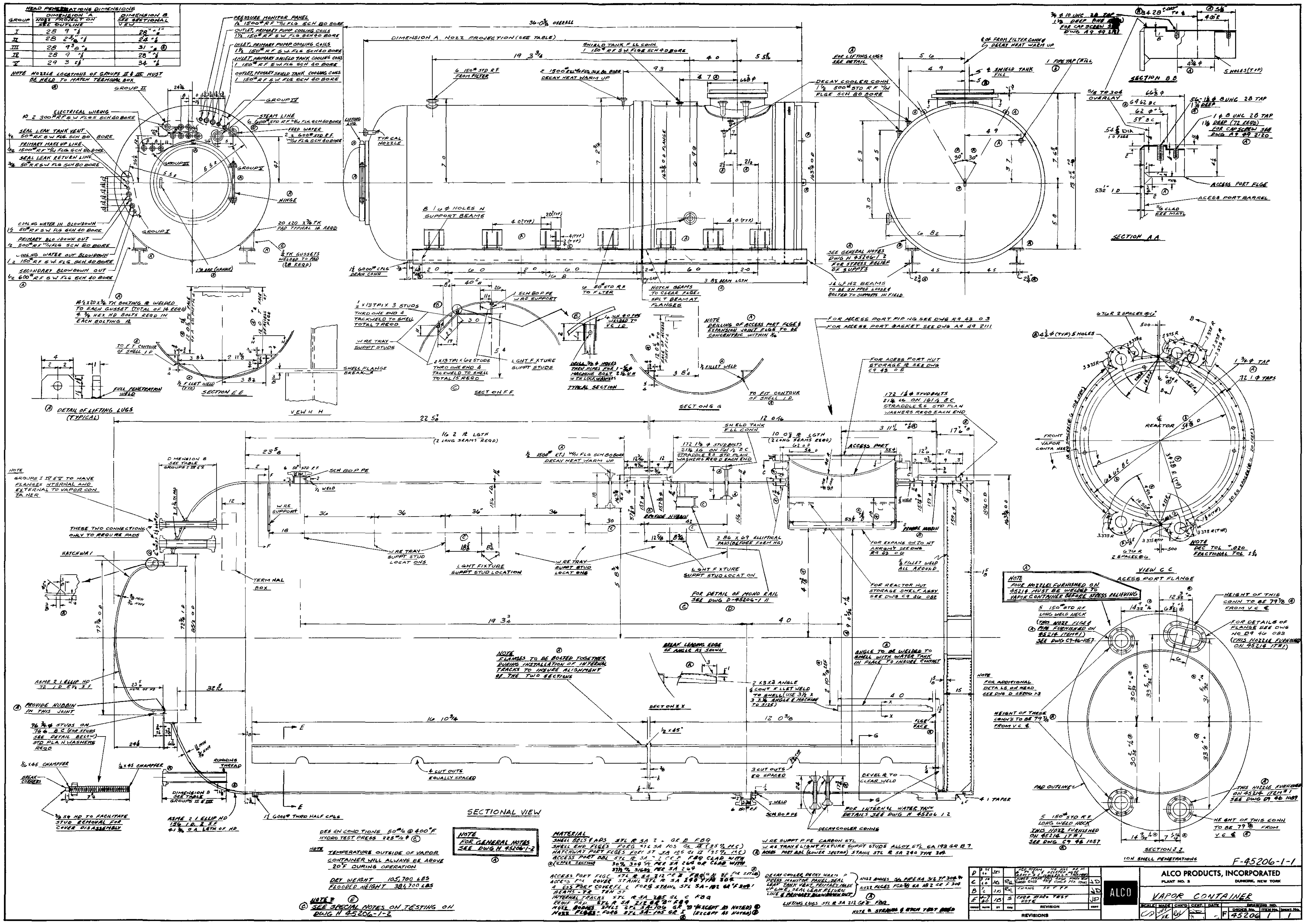


Fig. III-2A - Vapor Container

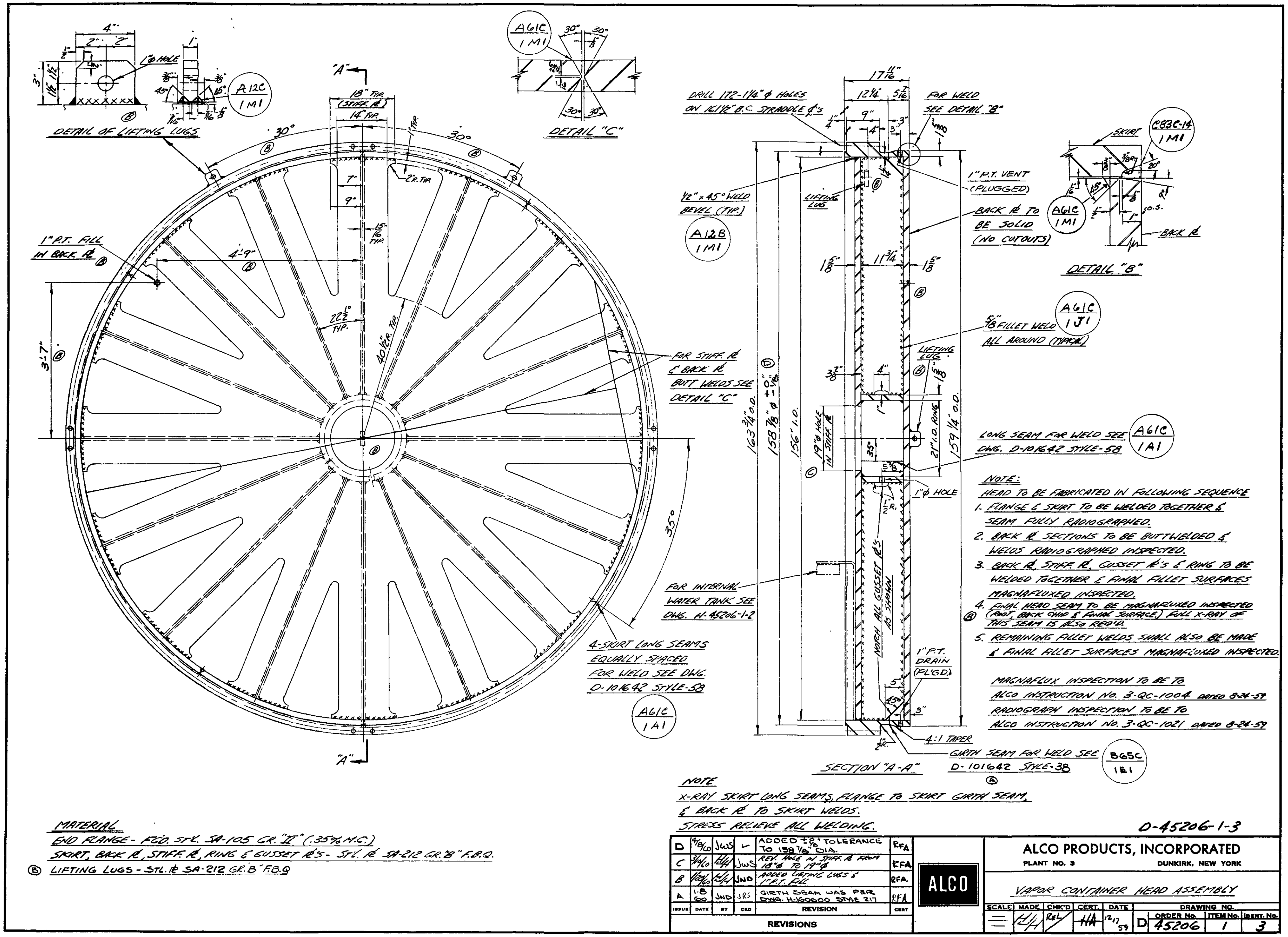


Fig. III-3 - Vapor Container Head Assembly

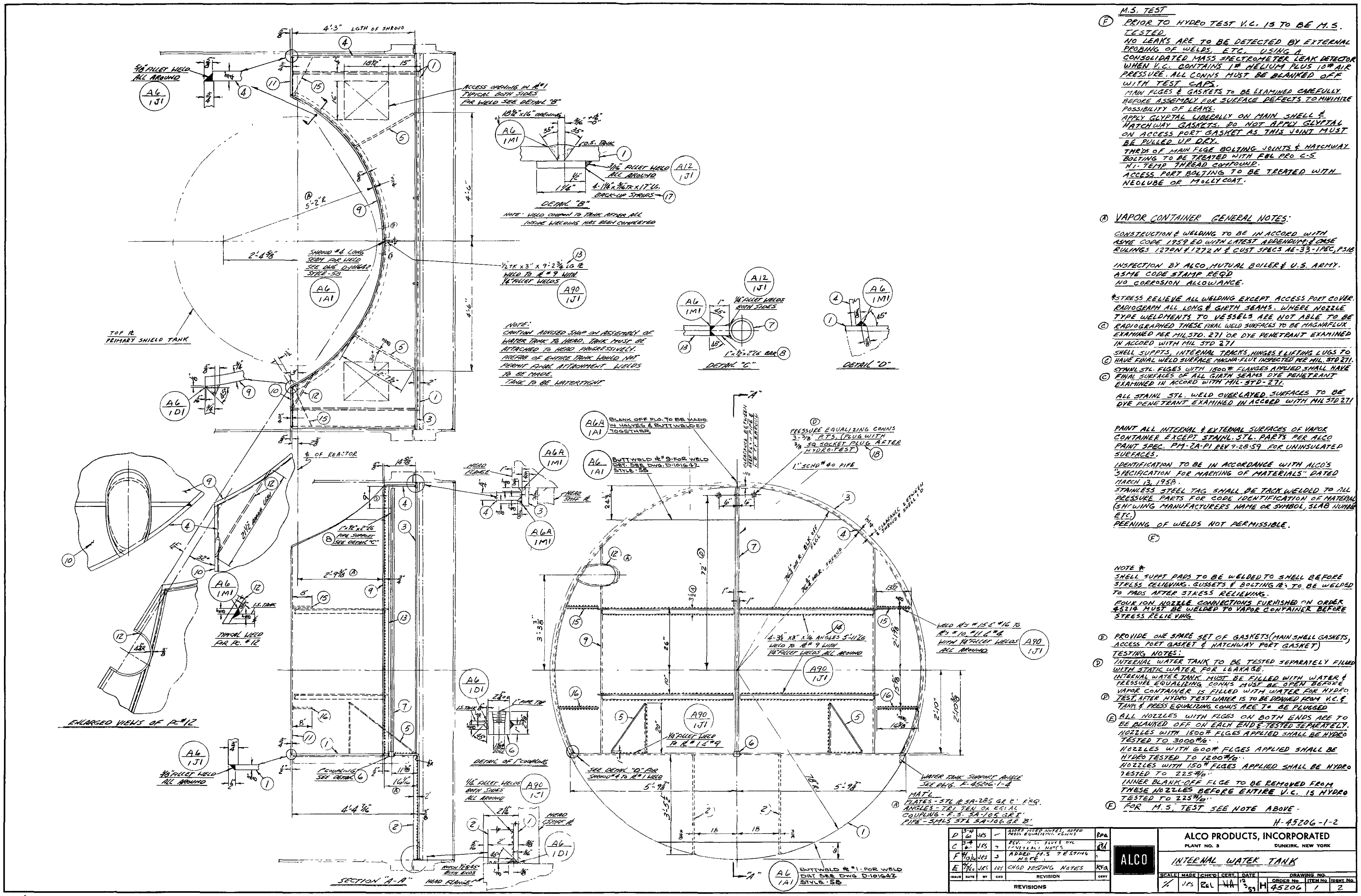
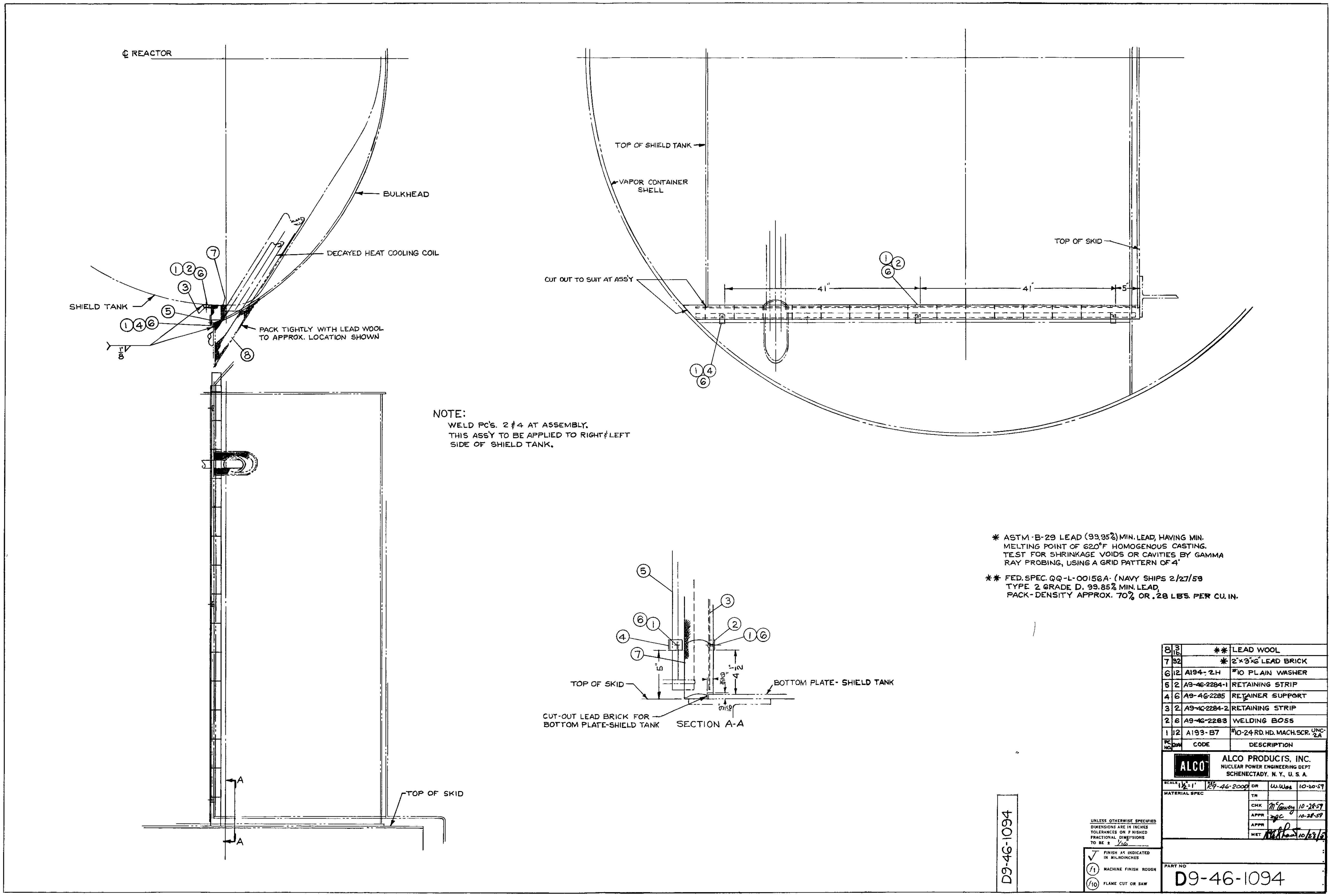


Fig. III-4 - Internal Water Tank



PC. NO.	QUAN	CODE	DESCRIPTION
8	3 lb.	**	LEAD WOOL
7	32	*	2"×3"×6" LEAD BRICK
6	12	A194-2H	#10 PLAIN WASHER
5	2	A9-46-2284-1	RETAINING STRIP
4	6	A9-46-2285	RETAINER SUPPORT
3	2	A9-46-2284-2	RETAINING STRIP
2	6	A9-46-2283	WELDING BOSS
1	12	A193-B7	#10-24 RD. HD. MACH. SCR. UNC-2A

Fig. IV-3 - Shielding Between Primary Shield Tank and Vapor Container Head

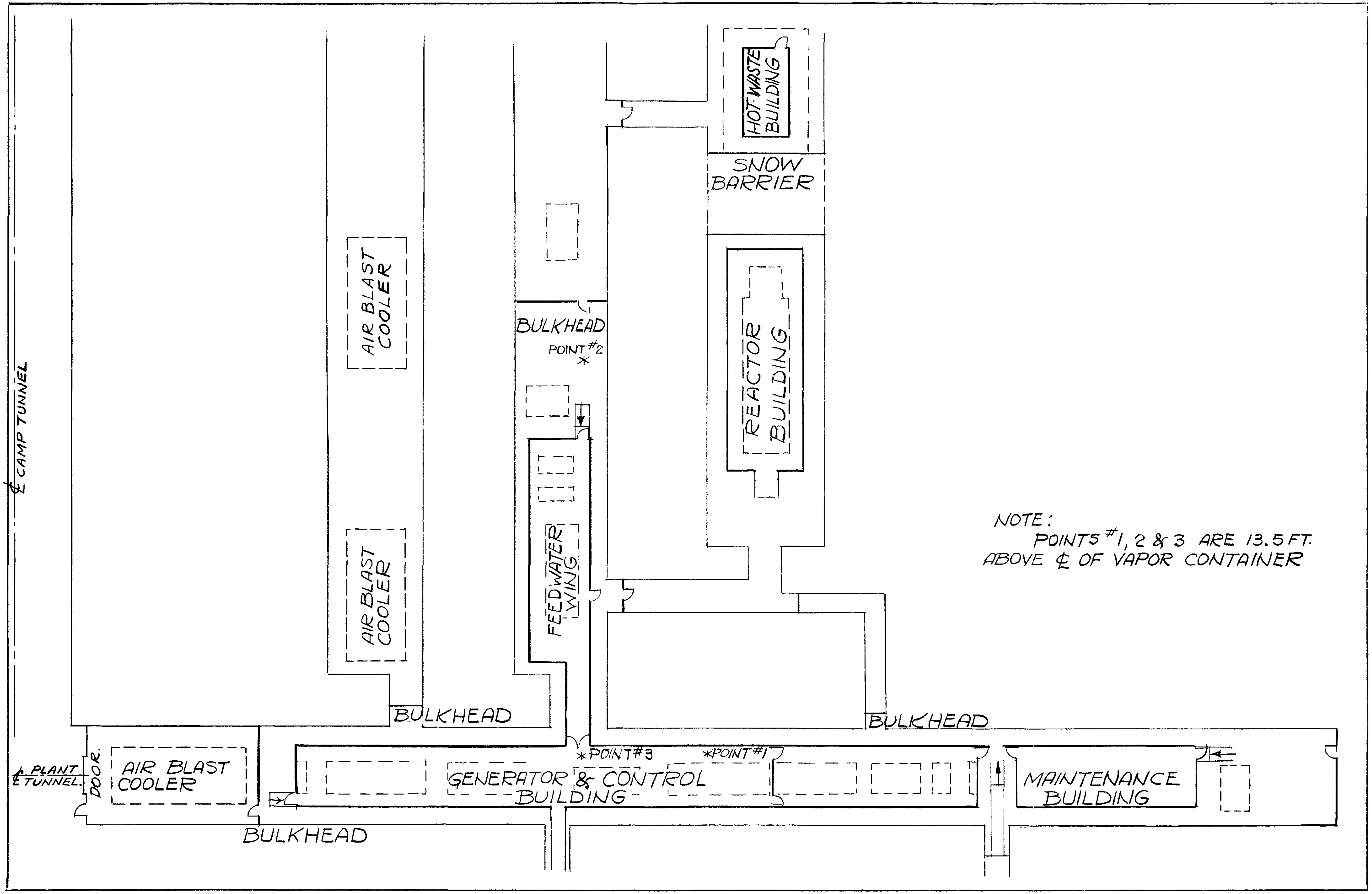


Fig. V-1A - Plant Layout

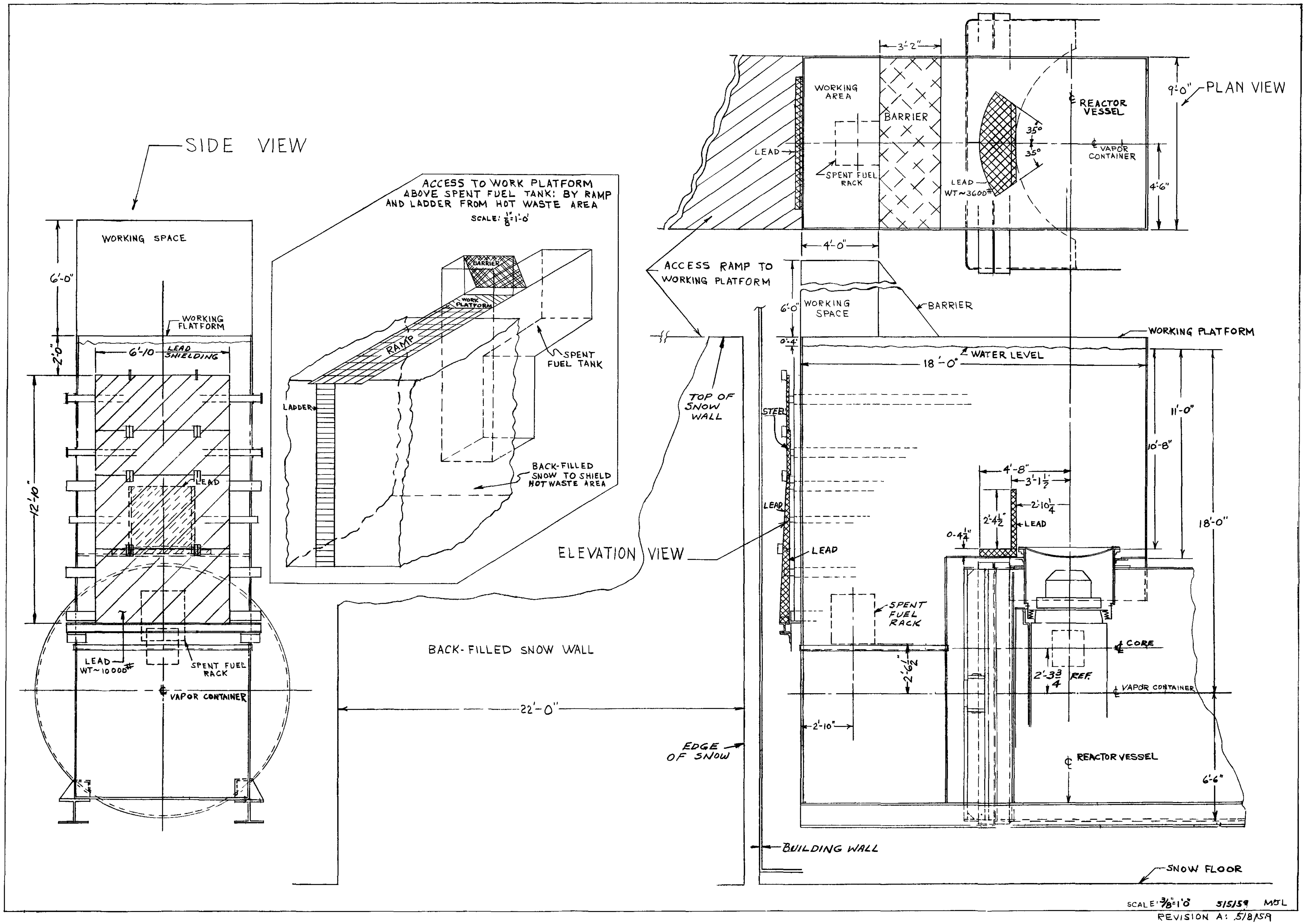
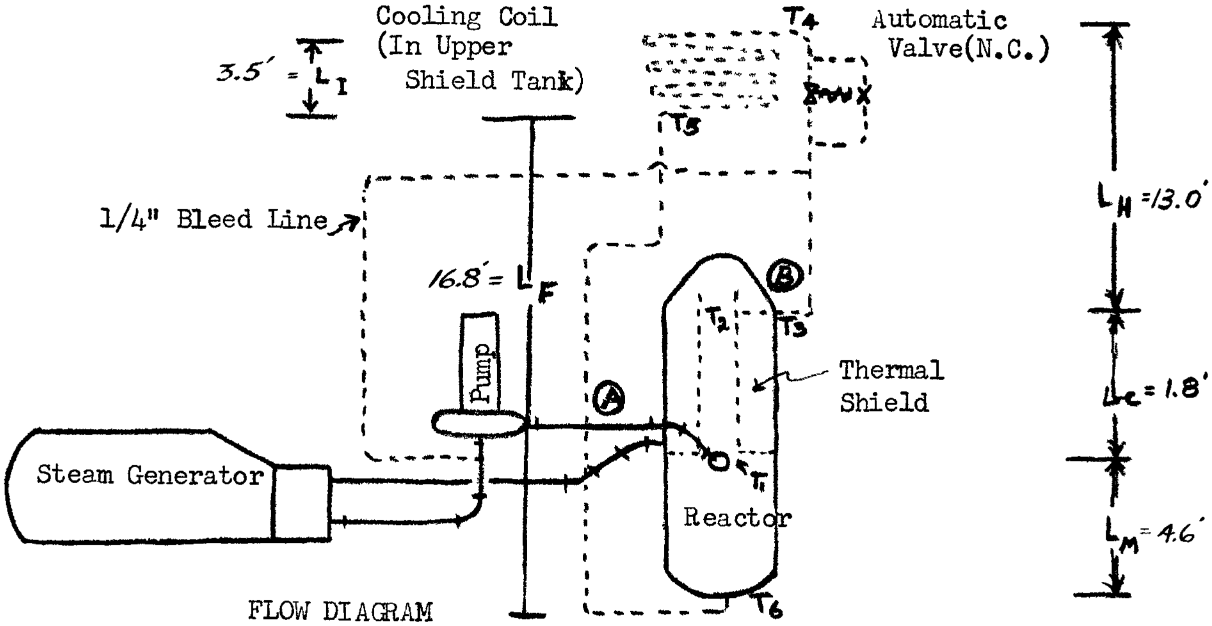


Fig. V-2A - Spent Fuel Handling System



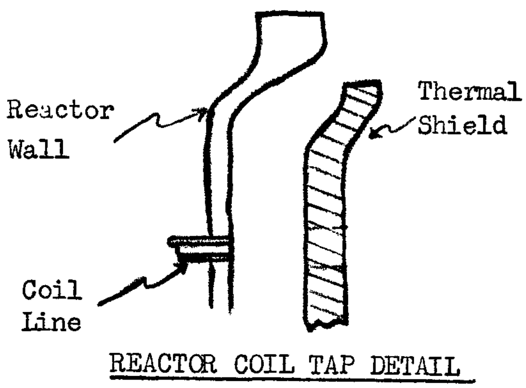


Fig. VI-5 - Decay Heat Cooling

710-029