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Volume II

AEC Research and Development Report
UC-81, Reactors - Power
[Special Distribution]

PWR preliminary design for PL-3

Contract No. AT[30-1]-2900
with U. S. Atomic Energy Commission
New York Operations Office

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ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT

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APAE-115
Volume 2
AEC Research and
Development Report
UC-81, Reactors Power
(Special Distribution)

PWR PRELIMINARY DESIGN FOR PL-3

Approved by:
G. E. Humphries, Project Engineer

Issued: February 28, 1962

Contract No. AT(30-1)-2900
with U. S. Atomic Energy Commission
New York Operations Office

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ABSTRACT

The pressurized water reactor preliminary design presented in this volume is the preferred design developed under Phase I of the PL-3 contract. This report presents plant design criteria, summary of plant selection, plant description, reactor and primary system description, thermal and hydraulic analysis, nuclear analysis, control and instrumentation description, shielding description, auxiliary systems, power plant equipment, waste disposal, buildings and tunnels, services, operation and maintenance, logistics, erection, cost information and a training program outline.

ACKNOWLEDGEMENT

This report was prepared by members of the PL-3 staff and was compiled by D. G. Ott, W. R. Pearce, J. W. Niestlie and E. R. Schmidt of Internuclear Company.

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1.0 INTRODUCTION

In October, 1961, Contract AT(30-1)-2900 was signed between Alco Products, Incorporated and the Atomic Energy Commission whereby Alco Products shall perform the research, development, design, construction, installation and testing of a nuclear steam power plant which utilizes a light water cooled and moderated nuclear reactor, capable of being transported by air and installed and operated in a snow tunnel at Byrd Station, Antarctica.

1.1 THE PL-3 PROJECT

Task 1 of Phase I of the contract was devoted to a thorough survey of concepts and designs of nuclear steam power plants which satisfied the general requirements for PL-3. Five concepts were selected for design modifications so that each plant concept met fully the technical requirements for PL-3 as delineated in Section 2.0, "Plant Design Criteria." Two concepts employed BWR to produce steam, one a natural circulation direct cycle reactor system which was a modification of PL-2, and the other a natural circulation indirect cycle reactor system. The remaining three concepts utilized the PWR to produce steam. Each concept was similar to portable nuclear power plants already constructed for service in remote areas (PM-1 and PM-2A). An evaluation procedure was developed to implement the selection of two nuclear power plant concepts which were most promising for PL-3. Based on the criteria established for the evaluation, the direct cycle BWR (PL-2) and a PWR which was a generic derivative of PM-2A were selected as the preferred concepts. It was recommended that both plants undergo further design analysis to realize additional saving in plant size and weight, principally reductions in core size and improvements in refueling and spent fuel shipment techniques, with concomitant reductions in module weights and spent fuel shipping requirements. These recommendations were altered when discussions with the Army Reactors Branch elicited a strong emphasis on extended core lifetime for either concept, with acceptance of the increased system weight implied by the resultant larger reactor core.

The two perferred concepts were subjected to further design analysis to resolve such problems as the selection of vertical or horizontal steam generator for the PWR and the attendant vapor containment provisions and primary system arrangement. The ultimate objective of this work was to achieve preliminary or reference designs for each plant during Task 2. A selection was made of the preferred preliminary design which would be the basis for final design, fabrication, erection, and operation of the PL-3 plant for Byrd Station, Antarctica. The design preferred for this application was the PWR design similar to PM-2A. The arguments and discussions underlying this selection are presented in a companion report, AP Note 408, Volume I, "PL-3 Concept Selection."

1.2 CONTRACT REQUIREMENTS

The arguments supporting the selection of the preferred preliminary design for PL-3 and the description of the two plant designs, developed during Task 2, which vied for preference comprise the contents of three reports. The selection arguments were presented in AP Note 408, Volume I, "PL-3 Concept Selection." The PWR and BWR design descriptions are presented respectively in this document and APAE-115, Volume 1, "BWR Reference Design for PL-3." These documents partially fulfill the contractual requirements for the Phase I portion of the PL-3 contract.

Phase II will be devoted initially to the preparation of a design analysis, plans, drawings and procurement specifications, followed by the manufacture and/or procurement of components, materials and equipment for the preferred PWR plant. The plant items will be tested and assembled finally into modules designed to permit shipment of the entire plant to Antarctica by ship and air transport.

Phase III will consist of transportation supervision, installation and test operation of the plant at Antarctica.

During Phase I, research and development work pertaining to the PL-3 plant design was initiated to demonstrate the feasibility of advanced technological features of the preliminary design. This work will be continued, terminated or modified as required to demonstrate the integrity of the plant design during Phase II.

1.3 SCHEDULE

The performance of work under the PL-3 contract will be accomplished in accordance with the completion time of the principal events listed below:

1. Completion of preliminary design and preparation of development program - February 28, 1962.

2. Delivery of the plant support facilities to the point of embarkation in continental United States - September 15, 1962.

3. Delivery of the plant to the point of embarkation in continental United States - September 15, 1963.

4. Completion of on-site tests - October 15, 1964.

1.4 CONTENT OF REPORT

The data and descriptive material presented in this report form a preliminary design of a pressurized water reactor power plant. This optimized design minimizes system weights and the number of shipping modules and conforms to the requirements, development objectives and criteria described in the Technical Provisions attached to the PL-3 contract.

The preferred PWR design borrows the technologically advantageous features of the PM-1A, PM-2A, and SM-1, while meeting the 20,000 lb module weight limitation. Two core designs were considered during the early stages of the preliminary design. The preferred core is a 7 x 7 full enrichment plate type core design used in SM-1. The alternate design is a 5 x 5 core design with SM-2 plate type fuel elements. This design was developed to reduce the system weight and spent fuel handling. The 5 x 5 design work is included in an appendix to this report in order to present in full the engineering analysis and design performed during Task 2.

2.0 PLANT DESIGN CRITERIA

The technical requirements and objectives for the plant are presented in the contract as "Appendix B - Technical Provisions." Environmental data for the Byrd Station site is also provided in Appendix B. The data and requirements which are significant to the plant design and to the selection of a preferred plant for PL-3 are summarized in these paragraphs.

2.1 GENERAL CRITERIA

2.1.1 Design Requirements

2.1.1.1 The plant shall have a net output of 1,000 KWe at 0.8 pf (3 phase, 480 volt, 60 cycle) plus 1.5 x 10^6 Btu/hr of thermal energy.

2.1.1.2 The reactor shall be light water cooled and moderated, either of the forced circulation pressurized type or natural circulation boiling (direct cycle) type.

2.1.1.3 The plant shall utilize a conventional condensing steam cycle with turbine generator with cycle waste heat dissipated to the air.

2.1.1.4 The plant shall include the foundations and housing and all supporting facilities and equipment to operate and maintain the plant as a self-sufficient unit.

2.1.1.5 The plant shall be capable of being transported in preassembled modules by C-130 aircraft operating under Antarctic conditions.

2.1.1.6 The plant shall be capable of being installed and operated in snow tunnels at Byrd Station, Antarctica.

2.1.1.7 The plant shall be capable of being installed with equipment which can be transported by C-130 aircraft operating under Antarctic conditions.

2.1.1.8 The plant shall be inherently safe to operate and maintain.

2.1.1.9 The reactor plant shall be contained.

2.1.1.10 The plant when installed in the Antarctic shall have a high degree of assurance of satisfactory operation.

2.1.1.11 The plant shall not release radioactivity of a type and magnitude that would adversely affect future scientific studies in the Antarctic.

2.1.1.12 The plant (other than foundations and housing and all supporting facilities) shall be capable of being installed in a single Antarctic construction season.

2.1.1.13 The plant shall be capable of being operated and maintained under normal operating conditions by personnel working an 84-hour week with cumulative radiation dosage less than 1.25 rem/quarter per man.

2.1.2 Research and Development Objectives

2.1.2.1 The plant should be capable of being installed with a minimum amount of effort and with a minimum of construction equipment.

2.1.2.2 The plant must be capable of being safely operated and maintained with a minimum number of personnel and should be capable of being refueled by the normal total crew.

2.1.2.3 The volume and weight of the plant and associated equipment to be transported should be a minimum.

2.1.2.4 The plant should be capable of producing power on a continuous basis.

2.1.2.5 The cost of the plant should be the minimum consistent with the other requirements.

2.1.2.6 The core life should be not less than two years; the fuel cycle costs per unit of energy should be a minimum, and the total weight of spent fuel casks should be a minimum.

2.1.2.7 The plant should require a low inventory of repair parts and operating supplies and should provide for maximum interchangeability of components.

2.1.2.8 The design life of the plant should be at least 20 years.

2.1.2.9 The plant should be simple to install, operate and maintain and should require a minimum of special skills and complex procedures or processes.

2.1.2.10 Insofar as possible, the plant should be of a design that could be utilized at other Antarctic snow tunnel installations with a minimum of modification.

2.1.2.11 The plant should be capable of being relocated with a minimum amount of effort and with the maximum reuse of equipment and facilities.

2.1.2.12 The plant should be of such a design that replacement crews can be trained without requiring additional new training facilities in the United States.

2.2 PLANT SITE AND ENVIRONMENTAL DATA

The environmental conditions at Byrd Station are presented in Table 2.1.

2.3 ADDITIONAL DESIGN REQUIREMENTS

In addition to the general criteria for PL-3, listed in Section 2.1, guidelines for the plant design were included in Appendix B which further defined the design requirements. The guidelines which markedly influenced the preliminary design and selection of the plants developed during Task 2 are listed below:

2.3.1 The plant should be capable of being started from a dead cold condition without external power.

2.3.2 The plant shall have a net output of 1,000 kw at 0.8 pf (3 phase, 480 volt, 60 cycle) and 1.5 x 10^6 Btu/hr of thermal energy. The 1.5 x 10^6 Btu/hr of thermal energy shall be in the form of steam at 100 psi dry and saturated, and the steam system shall be appropriately designed to prevent radioactive contamination to the export steam.

2.3.3 All rotating parts and all electrical instrumentation components should be capable of being removed and replaced without cutting structural members or pressure piping.

2.3.4 All major components shall have a design life of 20 years.

2.3.5 After extended occupied shutdown (greater than 24 hours), the plant shall produce rated power output in not more than 24 hours without damage or permanent deformation to any part of the plant.

2.3.6 Following a scram at power operation, the plant shall be capable of recovery to demand power operation within 20 minutes, provided startup can be safely initiated within 8 minutes after the scram.

TABLE 2.1
ENVIRONMENTAL DATA AT BYRD STATION, ANTARCTICA

Location	80^{o}S, 120^{o}W
Accessibility	
Air	October - February
Sea	No
Other	No
Construction	
Type	Subsurface (Snow Tunnel)
Season	90 days
Equipment	Transportation Limited
Limitations	Peter Snow Plow Capability
Water	
Source	Snow
Quality	Soft and Palatable Requires Minimum Treatment
Availability	Limited
Wind	
Average Velocity	15 - 17 mph
Maximum Velocity	77 mph
Prevailing Direction	NNE
Temperature	
Mean Annual	-18^{o}F
Minimum	-82^{o}F
Maximum	$+34^{o}$F
Average Construction	$+10^{o}$F
Minimum Construction	-35^{o}F
Precipitation	
Annual Snowfall	13 in.
Elevation (above sea level)	5,000 ft

2.3.7 Components of the plant which are capable of producing interference to radio communication equipment shall be equipped with suitable means of suppression.

2.3.8 The plant shall be designed so that the maximum temperature of the surrounding snow will not be greater than $0^{o}F$. In any case, the temperature rise in the snow under the foundations shall not exceed $20^{o}F$.

2.3.9 All foundations should be designed to have uniform loading and provisions to accommodate differential settlement of a maximum of 5 feet between the primary and secondary systems.

2.3.10 The plant shall be designed and constructed in accordance with the latest applicable codes, standards, and practices.

2.3.11 The complete plant should be capable of being operated from a central control room by one man, except that an additional man may be used 5 minutes after a scram, and to check equipment every 2 hours. During startup and shutdown operation, a second man may be required in the control room.

2.3.12 The reactor complex shall be designed to allow for safe routine operation even in the event of simultaneous release to the coolant of one per cent of the activity of the end-of-life fuel elements.

2.3.13 The steam at the turbine inlet under steady state conditions and during instantaneous load changes of 20% of full load steam within range of 10-100% of full load shall not contain moisture in excess of 0.25%.

2.3.14 Steam generators in any indirect cycle reactor system shall be of tube and shell construction with reactor coolant on the tube side. Provisions should be made for removal and replacement of tube bundle assembly utilizing flanged connections. Access shall be provided for plugging tubes.

2.3.15 The reactor shall have negative void and temperature coefficients of reactivity within all expected operating and shutdown conditions.

2.3.16 The fuel shall be enriched uranium clad with either stainless steel or zirconium.

2.3.17 The core life should be as long as possible (greater than 2 years) consistent with keeping the fuel cycle costs per unit of energy and the total weight of spent fuel casks to a minimum.

2.3.18 A soluble poison system shall be provided for emergency shutdown of the reactor.

2.3.19 Sufficient control rod worth shall be provided to make the reactor subcritical at atmospheric pressure and a coolant temperature of $4^{\circ}C$ with any single rod in the fully withdrawn position. This requirement shall apply at any time during core life and shall not require the aid of soluble neutron poisons.

2.3.20 If the reactor pressure vessel is austenitic stainless steel, the reactor shall be designed so that integrated fast neutron dose to the pressure vessel shall not exceed 10^{21} nvt over the design lifetime of the plant at 0.8 plant factor. If the reactor vessel is carbon steel, fast neutron exposure over the life of the plant as described in the preceding sentence shall be such that there will be no restriction of operating and maintenance procedures.

2.3.21 When the turbine generator is in operation, the system shall be automatically load following. If rod motion is required to accomplish this, an automatic control system shall be furnished.

2.3.22 The plant under normal operating conditions, shall not release radioactive wastes above the following concentrations: gaseous, 4×10^{-14} μc/cc; liquid 10^{-8} μc/cc.

2.3.23 Spent fuel storage facilities shall be furnished to store a minimum of two complete cores at maximum reactivity in a subcritical condition, to reject safely the fission product decay heat of the cores, and to contain and control the release from ruptured spent fuel elements of 2% of the fission product inventory of a spent core.

2.3.24 The primary shield shall be designed so that in no event will the dose exceed 100 millirem/hour outside the primary shield 120 minutes following shutdown. The radiation level 3 feet from other plant components shall not exceed 1.2 millirem/hour.

2.3.25 The plant secondary shield shall be designed so that the radiation level outside of the power plant controlled area shall not exceed 0.06 millirem/hour. No detectable radiation attributable to the plant will be allowed in the camp living areas.

2.3.26 Neutron flux attributable to the plant shall not exceed one neutron per square meter per minute at one mile from the plant under any operating condition.

3.0 SUMMARY OF PLANT SELECTION

3.1 TYPES OF PWR PLANT CONSIDERED

Three PWR plant concepts were evaluated during Task 1 of Phase I. One concept was a modification of PM-1; the remaining two were modifications of PM-2A. During this initial portion of the PL-3 contract, each concept was evolved from PM-1 or PM-2A by modifying these existing plants only to the extent necessary to satisfy the requirements for PL-3. Although one or more of these plant concepts would be dropped from consideration as engineering design and analysis progressed, it was expressly intended that advantageous design characteristics of any of these concepts would be included in the design of the plant to be selected ultimately for PL-3.

The principal modification made to the concept derived from PM-1 entailed the design of vapor containment to satisfy PL-3 requirements. PM-1 utilizes vertical enclosures to facilitate back fill earth shielding, but does not provide pressure containment in the event of primary system rupture. Consideration was given also to the problems associated with installation and operation of PL-3 in a snow tunnel and the sizing and packaging of plant components to satisfy weight and size limitations for PL-3.

The two plant concepts derived from PM-2A were designed for snow tunnel installation and total containment of the primary system and incorporated the glycol-coupled heat rejection system used on PM-2A. Extensive revisions of the PM-2A design were required to meet the design criteria for air transportability. These concepts were essentially similar except for their cores. The core for one concept was a 7 x 7 array of SM-1 fuel elements; the other used a 5 x 5 arrangement of SM-2 elements.

3.2 SELECTION OF PREFERRED PWR CONCEPT

Preliminary design work was directed initially toward a plant design which utilized the 5 x 5 core. While the lifetime of this core marginally satisfied the 2 yr core life requirement for PL-3, the smaller core size was expected to yield savings in spent fuel shipping casks and primary system size and weight. Further analysis of this design approach led to the conclusion that the pressure vessel size could not be reduced significantly because of the possibility of radiation damage to the pressure vessel with prolonged exposure. In addition, the advantage of reduced number of spent fuel shipping casks was negated by the increased frequency of shipment characteristic of the shorter core life. The hazards associated with the transportation of men and material to the PL-3 site are great so that there is a premium attached to methods or techniques which minimize shipping to or from Byrd Station.

These considerations led to the rejection of the 5 x 5 core in favor of the 7 x 7 array of SM-1 fuel elements with its extended lifetime. One further advantage of this core is the long and successful history of operating experience with this core in SM-1.

The preliminary design which evolved from the 7 x 7 concept incorporated the vertical steam generator and containment used on PM-1. The concept of high pressure containment, developed for the BWR described in a companion report, was applied to this preliminary design and resulted in two vertical containers for the primary system rather than the four which characterize the PM-1 and PM-3A designs. One container encloses the reactor and the other encloses the steam generator, pressurizer, primary pump and control rod drive motors. The two cylinder containment design permits convenient access to the container enclosing the pump, steam generator and pressurizer through an access hatch at grade level. This simplifies the problems associated with primary system maintenance. The preferred plant with 7 x 7 core design and vertical containment for the PWR preliminary design is described in the body of this report. The alternate 5 x 5 core design is described in an attached appendix.

4.0 PLANT DESCRIPTION

4.1 TUNNEL AND BUILDING ARRANGEMENT

The plant is located in two tunnels parallel to an existing tunnel leading from camp as shown in Dwg. AEL-734. These tunnels are 27 ft in width at the base and all buildings located in these tunnels are 20 ft in width.

The tunnel configuration has been designed with the short plant erection schedule of 45 days as a major consideration. It will be feasible to bring in equipment during the plant erection from all four of the ramps leading into the two plant tunnels.

Three building-enclosed areas are within the two plant tunnels and are connected by necessary personnel and piping interconnecting tunnels. The first of these enclosed areas is the primary system. This is the most remote from the camp and downwind of all of the other plant and camp areas. In the same tunnel as the primary system, although separated by a snow barrier, is the building housing the power conversion system (turbine-generator, condenser, and auxiliary equipment) plus the heat rejection system (air blast coolers and related equipment).

The plant tunnel closest to the main camp tunnel contains waste processing and laboratory equipment, the control console, and other necessary plant utilities. At the end of this tunnel, again downwind of both plant and camp, is the processing area for both liquid and gaseous wastes. Due to the slightly radioactive nature of these wastes, this equipment is located on the lee side in relation to other equipment and is separated from other equipment by a shield barrier. The chemical laboratory, emergency diesels, maintenance and personnel facilities are also located in this tunnel.

An access tunnel connects the plant complex to the camp tunnel and enters the plant complex at the personnel area where necessary monitoring of personnel and equipment is performed.

4.2 REACTOR AND POWER PLANT EQUIPMENT

The primary system of the PL-3 plant consists of a pressurized water reactor and a vertical steam generator. Primary system operating pressure is maintained at 1200 psia by the pressurizer. The primary system coolant is circulated by the primary coolant pump, picking up heat while passing through the reactor core. The heat is then transferred to the secondary system, generating steam at 315 psia, dry and saturated, for delivery to a geared turbine-generator unit. An auxiliary heat exchanger provides low-pressure export steam to the camp system. The turbine exhausts to a surface condenser and a glycol-coupled air blast cooler heat rejection system. A summary of plant data is presented in Table 4.1.

TABLE 4.1
PLANT DATA SUMMARY

Reactor Power	9.3 Mw
Gross Electrical Output	1500 kw
Net Electrical Output	1000 kw
Low Pressure Export Steam	1.5×10^6 Btu/hr
Core Lifetime	2 yr
Fuel Type	Fully enriched, plate type
Fuel Loading	22.48 Kg U-235
Steam Conditions at Turbine	26,250 lbs/hr @ 300 psia
Condenser Vacuum	6 in. Hg
Design Ambient Temperature	34^oF
Number of Planeloads Required:	
Buildings and Foundations	29
Reactor and Primary Systems	24
Power Conversion Equipment and Auxiliaries	22
Construction Equipment and Supplies	4
Emergency Diesel Fuel Oil	30
TOTAL	109

The reactor complex or steam producing equipment is housed in the primary building in two vertical vapor containters. The reactor vessel is in one of the vapor containers, connecting to the other which houses the steam generator, the pressurizer and the primary coolant pump. These containments are designed to sustain a maximum credible accident postulated to release the total energy in the primary system. Nuclear instrumentation and the primary shielding are located within the reactor vapor containment. An overhead bridge crane is provided within the primary building for use during erection, maintenance and fuel handling operations.

The arrangement of the power plant equipment and plant auxiliaries is shown in Dwg. AEL-734. The condenser module, 28 ft x 8 ft is oriented so that the turbine-generator module and turbine auxiliaries module are located alongside the condenser. The lengths of these modules are 13 ft 6 in. and 7 ft respectively. The width of both modules is 6 ft 9 in., and this arrangement allows a 5 ft 3 in. aisle by the power conversion equipment. Additional laydown areas of 15 ft and 8 ft are provided at the ends of the modules.

The condenser module contains a surface condensing unit with an integral deaerating hotwell. Operating vacuum and duty at full load and 34°F ambient temperature are 6 in. Hg and 23.7×10^6 Btu/hr. Two condensate pumps, in addition to module piping, valving, wiring, and instrumentation, are located in the module.

The turbine-generator module has a 1500 KWe turbine-generator with necessary piping and control equipment. This module must be shipped in two packages because of the 20, 000 lb limitation for each shipping module.

Turbine auxiliaries are placed on a separate module. Turbine lubricating components, such as a heat exchanger, purifier, filter and circulating pump are located on this skid. Dual steam jet air ejectors and a combination steam jet and gland seal leak-off condensing unit are also on this module in close proximity to the turbine and condenser modules.

The secondary auxiliaries module is positioned beyond the 15 ft laydown area. The export steam evaporator (1.5×10^6 Btu/hr duty) and subcooler for auxiliary cooling are placed on this skid. The main and auxiliary glycol coolant circulating pumps and two feedwater pumps and their motor control center are also located here. Two small storage tanks (5 gals) and metering pumps are necessary for chemical treatment. A tank for 100 gals of condensate storage, in addition to that provided in the condenser hotwell, is provided.

Plant switchgear along with extra electronic, recording, and other control gear is located on a 15 ft x 8 ft switchgear module. This module is located at the end of the 8 ft laydown area beyond the condenser package.

The condenser coolant is a 60% by weight solution of ethylene glycol. This is cooled by two identical air blast coolers. Air intake is through stacks from the outside, through a plenum chamber to settle out snow, and through louvered openings to the air blast coolers. Four 60 in. dia. fan units per module draw air over straight lengths of tubes containing the coolant. Each module is 30 ft x 8 ft and has an exhaust stack structure directly over it.

Waste processing equipment is contained on a module that is 28 ft x 8 ft. A waste processing evaporator, utilizing some main steam for evaporating liquid wastes, a condenser and a waste heat exchanger are the major items of equipment on this module. Miscellaneous circulating pumps, storage tanks, mixed bed demineralizers and a rotating drum for solid wastes are also required for the processing of plant wastes.

A chemical laboratory is mounted on a module with shipping dimensions of 15 ft x 8 ft. However, this is expanded to 15 ft x 12 ft to allow a work area during plant operation. Standard laboratory equipment, such as a balance, fume hoods, sinks, a desk, files and an emergency shower are conveniently located on this module.

A control panel for plant operation is mounted on a 15 ft x 8 ft module. Other components, such as batteries, battery chargers and two inverters complete this module.

Two 250 kw diesel generating sets are utilized for plant start-up and emergency power for heating and lighting. Their control equipment enables diesels to come "on-line" within 20 sec after turbine shutdown.

The maintenance facility contains machine shop equipment necessary for repair and service of small and medium size plant components, including nuclear instrumentation. Storage bins for spare parts are located within the maintenance area.

The equipment in the personnel facility is used for radiation monitoring of both personnel and equipment. Film badge interpretation, along with office facilities, is also located in this 20 ft x 20 ft enclosed area.

Provisions have also been made for such plant utilities as diesel fuel storage, fire protection, and tunnel ventilation and cooling.

5.0 REACTOR AND PRIMARY SYSTEM

5.1 GENERAL DESCRIPTION

The basic primary system arrangement, shown on Dwg. AEL-762, consists of two vertical vapor containers and surrounding shield and refueling tanks. The reactor vessel container houses the reactor vessel enclosing the reactor, the reactor vessel support ring, the lead primary shield rings, the nuclear instrumentation and a portion of the primary coolant piping. This container is filled with shield water and also serves as the primary shield tank. The shield water in this tank not only assists in shielding during reactor operation but also serves to cool the lead primary shield rings, and to quench the energy of the primary coolant in the event of a system rupture. The column of water above the reactor core provides adequate shielding for personnel during fuel transfer and refueling operations.

The second vertical vapor container houses the vertical steam generator, the pressurizer, the seven control rod drive mechanisms, the primary coolant pump and a portion of the primary coolant piping. This is a dry container and is connected to the wet container with flexible or expansion type pipe couplings utilizing seal partitions or water barriers located at the wet tank side. These water seal partitions are designed for low pressure and are intended to rupture in the event of a maximum credible accident so that pressure buildup and released energy may communicate between vapor containers. These couplings and water seals also provide a means for the primary piping and the control rod drive thimbles to pass from the wet to the dry container.

Access into the dry steam generator container is by way of a manhole that is located near the floor level of the primary building. Entry into the container at this lower level provides safe, easy and quick maintenance of the equipment located in its bottom section, especially the control rod drive mechanisms which may be maintained or replaced without disturbing the rest of the system. A manhole in the top section of the dry vapor container also provides access into the upper section of the container if required.

Additional shielding is provided in the form of peripheral water shield tanks arranged around the vertical reactor container. These tanks extend approximately 8 ft from the core centerline and provide ample cooling water and storage space for the spent fuel shipping casks.

Table 5.1 presents a data summary for the primary system components.

TABLE 5.1

PRIMARY SYSTEM COMPONENTS DATA

Pressure Vessel

Material: Type 304 SS to SA-336 Grade F8	
Pressure	
Design	1350 psia
Operation	1200 psia
Temperature	
Design	550°F
Operation	470°F
Design stress (ASME Code Allowable)	11,900 psi
Inside diameter	40 in.
Wall thickness	2-1/2 in.
Overall length of vessel	11 ft 7 in.
Diameter of opening at top	28 in.

Primary Coolant Pump

Type: Centrifugal canned rotor (1 required)	
Material: Type 304 or 316 SS	
Rated flow of pump	2250 gpm
Operating head of pump	50 ft
Hydraulic horsepower	
Hot	20 hp nom
Cold	25 hp nom

Steam Generator

Type: Vertical; U-tube and shell with the reactor coolant on the tube side.

Materials: Those in contact with primary coolant are of either austenitic stainless steel or nickel-chromium-iron alloy; all others low alloy ferritic steel.

Tube side pressure	
Design	1350 psia
Operating	1200 psia
Shell side pressure	
Design	600 psia
Operating	315 psia
Tube side temperature	
Design	600°F
Operating	470°F
Shell side temperature	
Design	600°F
Operating	422°F

TABLE 5.1 (Cont'd)

Design secondary flow	30,000 lb/hr
Heat transfer area	761 ft^2
Steam quality (20% to 100% load)	99.75%

Pressurizer

Material: Type 304 SS	
Overall length	60 in.
Inside diameter	26-1/2 in.
Wall thickness	1-11/16 in.
Operating pressure	1200 psia
Design pressure	1350 psia
Design temperature	600^{o}F
Steam volume	8.6 ft^3
Water volume	7.3 ft^3
No. of heaters (1 kw each)	18

Primary Coolant Piping

Material: Type 304 or 316 SS	
Outside diameter (Schedule 80)	6.625 in.
Wall thickness	0.432 in.

Control Rod Drives

Type: Rack and Pinion (7 required)	
Total travel	22 in.
Control rod speed (either direction)	3 in/min
Control rod drop time at 500^{o}F (predicted)	0.5 to 0.7 sec

Seal

Type: Mechanical breakdown labyrinth	
Average leakage per seal (predicted)	5.0 to 7.0 lb/hr

Vapor Containment

Type: Vertical (two required)

	Reactor Container	Steam Generator Container
Material:	(Type 304 or 316 SS)	(SA-350-LF-3 and SA-203 Grade E)
Design pressure	165 psia	165 psia
Design temperature	200^{o}F	400^{o}F
Outside diameter	8 ft 4 in.	8 ft 4 in.
Wall thickness	1/2 in.	1/2 in.
Overall height	32 ft 8 in.	32 ft 8 in.

5.2 REACTOR VESSEL AND CORE SUPPORT STRUCTURE

5.2.1 Reactor Vessel

The reactor vessel is shown on Dwg. AES-612. It is a vertical, cylindrical vessel, with a hemispherical bottom head and a flanged opening at the top to permit installation of the core assembly and fuel elements. It will be designed and stamped in accordance with the ASME code for Unfired Pressure Vessels. The design pressure and temperature are 1350 psig at 550^{o}F. The reactor vessel will be fabricated of austenitic stainless steel rings, forged and machined to shape and then welded to form the vessel. The material will be Type 304 stainless steel conforming to the requirements of ASME code materials specification SA-336 Grade F8.

A pair of six inch nozzles, located radially in the reactor wall above the top plane of the core, are provided for attaching the primary piping. The seven control rod drive penetrations are located in the lower portion of the vessel.

The primary coolant enters above the core through a 10 in. nozzle, flows downward along the vessel wall, turns toward the vessel center and flows up through the core. The flow then exits from the vessel by way of the six inch outlet nozzle.

A maximum integrated fast neutron flux of 1 x 10^{20} nvt was used as the design value for sizing the inside surface of the reactor vessel. This value was chosen to assure that the specified maximum allowable value of 1 x 10^{21} is not exceeded. This factor of ten should be sufficient to allow for any possible errors in the flux calculations. Based on this value of 1 x 10^{20} nvt, the vessel inside diameter is 40 in. To meet the ASME code rules for design at 1350 psig and 550^{o}F, a minimum vessel wall thickness of 2-1/2 in. was required for the center cylindrical portion of the vessel. The vessel is 45 in. outside diameter and approximately 11 ft 7 in. overall outside height including the top cover.

The flanged opening in the top of the vessel will be 28 in. inside diameter and the closure will be made by the use of through bolts. The through-bolted design has been selected to eliminate the possibility of the stud threads galling the vessel flange.

The bolt material will be A-286 as permitted in a special ruling under ASME code case 1296; nuts will be either A-286 or SA-194 Grade 8 (Type 304 SS). The closure seal will be made by using either a concentrically mounted pair of spiral wound metal gaskets with a tell-tale bleed between the two, or a single octagonal cross-section metal ring gasket. Gasket selection will be based on overall joint integrity and on gasket seating forces which will permit design of the thinnest and lightest flange.

5.2.2 Core Support Structure

The core support structure which is shown on Dwg. AEL-763 locates and supports the stationary fuel elements and guides the control rods. The structure will be shop-assembled and lowered as a complete unit into the reactor pressure vessel. The components of the structure consist of the top orifice plate, the top grid plate, the support plate, core skirt, the bottom grid plate, the pinion support carrier plate, the pinion bearing supports, and the necessary latches, tie rods and fasteners.

The seven pinion bearing supports, which include the control rod dash-pots and the control rod rack rollers, are supported by the pinion support carrier plate which is located by means of spacers between it and the bottom grid plate. The tie rods hold the entire structure together and transmit all loads to the support plate located at the top of the structure.

The bottom plate locates and orients the lower end of the stationary fuel elements by engaging the elements in large pilot holes. Square holes permit passage of the control rods through the bottom plate. A clearance of approximately 1/8 in. is provided between the control rod assembly and the bottom plate. The pilot holes provide an entrance passage for coolant flow from the lower plenum into the stationary fuel elements. Additional smaller holes in the plate provide entrance passage for coolant around the stationary fuel elements and the control rod tubes.

The support plate is spaced from the bottom plate by means of the core skirt. The skirt which confines the total coolant flow within the active core area is restrained from lateral movement by grooves machined in both the bottom and support plates. The top grid plate rests upon the support plate and is held in place by four latches engaging the tie rods. The orifice plate is mechanically fastened to the top plate to become an integral part of the top grid plate. Square holes permit passage of the control rods through the top plate and provide lateral support to the top of the control rod assembly. A total nominal clearance of approximately 0.030 in. is provided between the top plate and the control rod.

The core support structure is similar to that used in the SM-1, SM-1A and PM-2A plants. It will be fabricated from Type 304 SS annealed, except for dowels and capscrews, which are to be made from Type 302 or 303 SS.

5.3 VAPOR CONTAINMENT

Vapor containment for the PL-3 plant consists of two vertical tanks, one containing the reactor vessel and the other the steam generator, pressurizer and primary pump. The two vessels are connected together by three flexible ducts approximately 2 ft in diameter. The upper two contain the primary piping connecting the reactor and steam generator; the lower one houses the seven control rod drive thimbles.

The reactor containment vessel is 8 ft 4 in. outside diameter and 32 ft 8 in. overall height. The tank has elliptical heads and is made in two flanged sections. The top section is flanged above the shield water level and may be removed to provide access to the reactor for refueling. In addition, this top section provides a void volume for energy containment in the event of a primary pipe rupture within the containment tank. The reactor containment vessel is designed for a pressure of 165 psig at 200°F. Its wall thickness is 1/2 in. and its assembled weight, exclusive of internal fittings and attachments, is estimated to be 22, 100 lbs. Material used in fabrication will be Type 304 or 316 stainless steel to specification SA-336. Nuts and bolts will be A-286 material.

The second vertical vapor container tank is designed to house the steam generator, pressurizer, primary pump, control rod drives and auxiliary equipment. Working platforms at appropriate elevations and support structures for the various contained components will be included. A flanged dome is included at the top of the vessel to facilitate installation or replacement of major components. Normal access to this container is by way of a manhole located near floor level, permitting ready and safe entry for maintenance of lower level equipment. An access manhole is also provided in the top cylindrical section of this container. The steam generator containment vessel is fabricated of 1/2 in. thick, A-203 Grade E (3-1/2% nickel) material. Flanges will be forged of SA-350-LF-3 (3-1/2% nickel) steel. Bolting material will be SA-193-B7 with SA-194-2H alloy steel nuts. Both materials will be quenched and tempered to 180-240 Brinell and will possess a 15 ft lb Charpy keyhole at -20°F. The weight of the vessel, exclusive of fittings and attachments, is estimated to be 22, 100 lb. This vessel has the same diameter and height as the reactor containment vessel, and is designed for 165 psig pressure at 400°F. The design temperatures for these vessels are different because the steam generator container is dry while the reactor container is filled with shield water. It is expected that the quenching effect of the water in case of a rupture in the primary system would drastically reduce the maximum expected temperature in the reactor container.

The pressure loadings in the vapor containers were analyzed[(1)] for the maximum credible accident to determine peak hydrostatic pressure for the design of the vapor containers. The maximum credible accident was defined as a failure of the primary system piping at operating conditions. The analysis predicted that the shield water in the reactor containment vessel will completely quench the steam formed by the loss of coolant attending rupture of the primary piping inside the reactor containment vessel.

Both containment vessels are designed to meet the requirements of the ASME code for Unfired Pressure Vessels and will be code stamped. The same high quality workmanship and inspection required for the reactor vessel will also be utilized for the vapor container vessels. Reinforced penetrations are provided for primary coolant piping, steam lines, feedwater line, blowdown lines, drain lines, wiring, etc.

All vapor container outlet piping penetrations are equipped with electrically operated, fail-safe trip valves located directly outside the vapor container. These valves will close automatically if the pressure in the vapor container rises above a predetermined level. All inlet piping penetrations are equipped with check valves to prevent leakage of primary coolant, liquid or vapor. The check valves are backed-up by manually operated globe valves.

Two containment vessels will be mounted on a common base to minimize the displacement of one relative to the other and to keep the stresses in the interconnecting ducts to a minimum.

5.4 REACTOR CORE

The PL-3 core consists of 38 stationary fuel elements and 7 control rod elements with absorber sections arranged in a 7 x 7 array. Core specifications will be developed from PM-2A Core II and SM-2 Core I specifications, as modified for the specific PL-3 application. The stainless steel plate type fuel elements will be basically of PM-2A Core II design, incorporating improvements developed in the SM-2 program. A core vendor will be selected by competitive bid for fabrication of the two required cores. The vendor will be required to qualify his fabrication procedure before the start of actual core production. The methodfor qualification of procedure will be stated in the PL-3 core specifications. All materials employed will be certified and tested as required by these specifications.

5.4.1 Stationary Fuel Elements

The stationary fuel element is shown in Dwg. R9-13-2075. It is composed of 18 fuel plates brazed into Type 347 stainless steel side plates to form a rectangular assembly. Type 304L stainless steel end boxes, for positioning in the reactor, are welded to both sides of the element.

The fuel core, or meat, of each fuel plate consists of highly enriched spherical UO_2 fuel and a small quantity of ZrB_2 uniformly dispersed in a prealloyed Type 347 stainless steel matrix. This core is prepared by powder metallurgy techniques and is metallurgically clad by roll bonding with Type 347 stainless steel. The cladding prevents exposure of the fuel matrix to primary water. The final clad-core-clad thicknesses are 0.005-0.020-0.005 in.

5.4.2 Control Rod Assembly

The control rod assembly is shown on Dwg. AEL-763, and consists of the following components: cap assembly, tube and yoke weldment, rack, fuel element, and absorber element. This control rod design is basically the same as that used with the SM-1 core.

The fuel and absorber elements are housed in a square tube and held in place by the control rod cap. The rack is bolted and keyed to the bottom of the yoke. The yoke, which is welded to the bottom of the square tube, is machined in such a manner that it provides coolant inlet passage, a piston for deceleration of the control rod assembly at the end of a scram, and a means of attaching the rack. The yoke also has built-in flow deflection vanes which help provide proper flow distribution within the control rod assembly. As previously noted, the control rod cap assembly locks the fuel element and absorber section in the control rod tube. A large straight-through opening provides exit of the primary coolant from the control rod. The yoke and control rod cap were developed and tested under the SM-2 program. The materials of construction, with the exception of the fuel and absorber elements, are generally Type 304 SS and Armco 17-4 PH.

The fuel element portion of the control rod assembly is shown in Dwg. D9-13-1030. It consists of 16 fuel plates, each with an integral Eu_2O_3 flux suppressor dispersed in Type 347 stainless steel incorporated into the top end of the core of each plate. The same powder metallurgy and roll bonding techniques are used in fabricating the control rod elements as are used for the stationary elements. No end boxes are used with the control rod elements. A pin at the outlet end provides a means of handling these elements.

The absorber section of each control rod assembly consists of four composite plates. Each plate contains Eu_2O_3 dispersed in a stainless steel matrix which is clad by roll bonding with Type 347 stainless steel. The composite plates are TIG welded to form a rectangular parallelepiped.

5.5 CONTROL ROD DRIVES

The Alco design of the bottom mounted PL-3 control rod drives will provide:

1. A power source for reliable operation of the control rods at a fixed withdrawal and insertion speed and with accurate position indication and limit control.

2. A means for scramming the control rods by gravity with a fast scram rate within specified time limits.

3. An auxiliary power source during scram to apply an additional downward force in the event of a stuck rod.

4. A controlled, low leakage seal where the drive penetrates the reactor vessel.

5. Complete accessibility to the seals, right angle drive units, clutch assemblies, instrument assemblies and gear motors for ease of maintenance and replacement.

The control rod drive is shown on Dwg. R9-11-1030. It is driven by a commercial 1/8 hp, 1140 rpm, 3 phase, 440 v motor with an internal 2250-to 1 gear reducer producing 1/2 rpm at the output shaft. A universal joint is attached to the drive motor output shaft by a shear key rated at 700 in. -lb The other end of the universal joint is splined to the input shaft of the clutch drive assembly. This ball bearing-mounted input shaft is keyed to an electric clutch for normal up and down movement of the control rod. An over-running cam clutch is provided which engages only in the direction of downward control rod travel and applies additional torque to release a possible stuck rod. It is not anticipated that constant force spring motors will be required for PL-3 drives, as are shown in the reference drawing. The low pressure drop expected across the core will not result in an upward hydraulic force on the control rods which would increase scram time beyond acceptable limits.

Rod position indication is provided by a gear train from the control rod rack through the drive pinion and shaft to the selsyn motor transmitter of the instrument assembly. The instrument assembly includes two cam operated microswitches, geared directly to the pinion shaft, which stop rack and rod travel at the limits of the upper and lower positions.

A splined universal joint connects the clutch drive to a right angle drive unit. This unit is shown on the primary system arrangement Dwg. AEL-762. The right angle drive unit is a commercial, 1:1 ratio, spiral miter gear box which is mounted in the gear train between the clutch assembly and the shaft seal assembly, permitting optimum use of the available space and assuring complete accessibility to the seals and all external components.

The right angle drives are connected to the water seal assembly through splined universal joints. The splines act as a slip joint which will allow thermal expansion between the water seal shaft and the fixed, right angle drive. The water seal assembly is a mechanical breakdown labyrinth type, bolted first to a ground valve seat flange and then through the ground flange to a mounting flange on the reactor vessel. Spiral wound and filled stainless steel gaskets seal both bolted flanges. The ground valve-seat flange accommodates a spherical valve mounted on the water seal assembly shaft which permits seal removal without depressurizing or draining the reactor vessel. The thrust of primary system pressure acting on the seal shaft is carried by an external, conventionally lubricated, ball thrust bearing at the input end of the water seal shaft.

The pressure differential across the seal is approximately system pressure. Cool primary make-up water, at a pressure of 30 to 50 psi above reactor vessel operating pressure, is introduced into the reactor side of the seal to cool the seal and dilute any reactor water which may enter the seal. In this manner crud accumulation in the seal has been kept to a practical minimum in previous installations.

The pinion drive shaft, connected to the water seal assembly shaft, drives the straddle mounted pinion through a loose fitting spline supported by two ball bearings in the same yoke that carries the back-up roller. This arrangement permits accurate location of the pinion and back-up roller with respect to each other and proper tooth alignment and backlash between the rack and pinion are assured. The loose fitting spline acts as a slip joint permitting differential thermal expansion between the pinion shaft and the water seal shaft without introducing additional stresses or bearing loads. A collar at the reactor end of the pinion shaft enables the bearings and pinions to be withdrawn for replacement.

Any gear motor clutch drive and instrument assembly or right angle drive can be removed easily for maintenance without disturbing the rest of the gear train. The instrument assembly is screwed and doweled to the clutch drive assembly, and the clutch drive assembly can be removed from the drive train by pulling a pin, sliding the universal joint off the drive shaft and removing the mounting bolts. Similarly, a gear motor can be removed from the drive train by pulling the same pin, sliding the universal joint off the drive shaft and removing the gear motor mounting bolts.

The right angle drive can be removed from the gear train by removing its mounting bolts and sliding it from the universal joints at both its input and output shafts. To remove a seal assembly, the right angle drive and the connecting shaft would be removed first and the nut which positions the thrust bearing backed-off. The water seal shaft would then slide outward due to primary system pressure, seating the spherical valve. With the seal thus isolated, it can be removed from the mounting flange for maintenance.

All components within the reactor vessel, or in contact with reactor coolant, are of stainless steel or other corrosion resistant material. Those components located outboard from the water seal housing (clutch and drive assemblies and linkages) are primarily carbon steel with a protective finish.

5.6 SHIELDING ARRANGEMENT

The primary system shielding arrangement to provide personnel safety and reduce radiation heating of the snow walls and floor is shown on Dwg. AEL-762. The primary gamma shield surrounds the reactor vessel and consists of a lead cylinder 4 ft 5 in. inside diameter, 2-3/4 in. thick, with 1/4 in. stainless steel canning. A close fitting, 4 in. thick lead slab, with 1/4 in. stainless steel canning, is mounted on a common base with the cylinder and forms the bottom of the vertical right cylinder. The lead cylinder is located within the reactor container which also serves as the inner shield tank. This tank is 8 ft 4 in. outside diameter and provides an additional uninterupted 12 ft column of shield water above the reactor vessel flange.

The inner shield tank is surrounded by the outer peripheral shield. This shield consists of four segmented water tanks and incorporates a stepped joint design to form a single, 16 ft outside diameter shield tank. The centerlines of the inner shield tank and the outer shield are offset to permit storage of spent fuel shipping casks in the two larger tank quadrants of the outer peripheral shield.

The heat generated in the lead cylinder by gamma radiation is removed by natural convection cooling during reactor operation. Cooling coils near the top of the inner shield will remove any excess heat from the shield water to an auxiliary cooling system. It is not anticipated that cooling coils will be required in the outer shield tanks during operation. However, cooling coils may be added at the top of these tanks if thermal studies indicate the necessity for removing decay heat from a spent core being stored in these tanks.

The segmented design of the outer shield tank was necessary to meet the specified shipping package size. In order to reduce the overall number of shipping packages, the five shield tank packages will be utilized as shipping containers for spare parts, relatively small components and installation and operating supplies.

5.7 PRIMARY PIPING

The primary piping includes all piping in the primary loop connecting the reactor vessel, steam generator, and primary pump. It consists of 6 in. schedule 80 pipe connecting the components in series in a closed loop and provides a system of all welded joints except for flanged connections at the vapor container interconnections to permit field assembly. Specially designed weld neck flanges are used with spiral wound gaskets. The piping system design pressure is 1350 psig and the design temperature is 600^{o}F.

The pipe loop extends from the reactor vessel to the steam generator to the primary pump and back to the reactor vessel. Two penetrations are required in each vapor containment tank for the inlet and outlet pipes. The steam generator nozzles are connected to 8 in. x 6 in. reducers for joining the primary pipe.

Primary piping will be Type 304 or Type 316 stainless steel, either forged and bored or centrifugally cast and cold wrought. All heats of material employed will be certified by the vendor. Only the highest quality will be used, and in all instances optional tests such as 100% ultrasonic testing will be employed.

If forged, the piping must conform to SA-336 Types 304 or 316. If centrifugally cast and cold wrought, SA-452 Grades TP-304H or TP-316H are applicable. Additional optional tests will be performed.

Shop welding only will be permitted on primary piping. Field assembly will be with A-286 bolts and nuts, using stainless steel asbestos spiral wound gaskets.

5.8 PRESSURIZER

The pressurizer is designed to maintain the primary coolant system at the operating pressure of 1200 psia. It performs this function by heating contained water, creating steam at the saturation temperature corresponding to the required pressure. The water in the pressurizer is static and system pressure is exerted on the primary loop through a small connecting pipeline. The pressurizer performs an additional function by suppressing the pressure excursions and temperature transients in the primary system resulting from changes in plant load.

A tentative PL-3 pressurizer volume was obtained for preliminary cost and size estimates, by scaling down an existing pressurizer design based on the ratios of the system volumes. A complete pressurizer sizing study will be done when the final primary system design is completed. The differential equations describing the nuclear response of the core, the heat transfer characteristics of the core and steam generator and the transport delay and mixing factors will be derived. The response of the primary system to load changes will then be determined by solution of these equations on an analog computer.

The basic pressurizer design, shown on Dwg. AEL-762, is a vertical cylindrical pressure vessel with top and bottom hemispherical heads. The inside diameter is 26-1/2 in. and the wall thickness is 1-11/16 in. Overall height, not including the support skirt, is 5 ft. The water volume is 7.3 ft^3 and the steam volume is 8.6 ft^3. Total dry weight is approximately 2100 pounds.

The pressurizer is heated by 18 tubular heating elements of 1.0 kw capacity each. The heating elements are enclosed in wells which permit replacement without necessity for draining the pressurizer.

Design and fabrication of the vessel is in accordance with Section VIII of the ASME Unfired Pressure Vessel Code. The pressurizer will be Type 304 stainless steel. As with all other primary components, the highest quality will be specified throughout, both for materials and for fabrication. Forgings will be to specification SA-336 Grade F8, plate to SA-240 Grade S and forged penetration fittings to SA-182 Grade F 304. Gasket material for attachments requiring flanged connections will be spiral wound stainless steel and asbestos or solid dead-soft annealed Type 304 stainless steel octagonal cross-section rings.

5.9 STEAM GENERATOR

The steam generator is the vertical type of U-tube and shell construction with the primary coolant on the tube side. Design and fabrication is in accordance with Section VIII of the ASME Unfired Pressure Vessel Code. The steam generator will be designed to supply 30,000 lb/hr of dry saturated steam at 315 psia and 422^{O}F. A moisture separator of the centrifugal downflow type is used to remove excess moisture from the steam. Steam quality is 99.75%.

The tube bundle and shell assembly is flanged and bolted to the channel section to permit removal for servicing or replacement. Inlet and outlet nozzles on the channel section are provided for connection to the primary piping. The steam generator is supported on lubrite bearings to allow for piping expansion. Access for inspection of tubes and tube plugging is provided by a manhole in the bottom head of the channel. The top section of the shell is flanged to permit access to the interior of the steam drum for servicing the moisture separation equipment, and to permit easier installation of the tube bundle and shell assembly. Less headroom is required with a removable top shell section. To gain access to the steam generator for major servicing, the top section of the vapor container shell must be removed.

The tube bundle and shell assembly is replaced by first disconnecting the steam line and other piping and removing the top shell section. After the tube bundle and shell assembly is unbolted from the channel flange, it can be lifted out of the vapor container with the overhead crane.

The unit has 490 U-tubes of 1/2 in. outside diameter on 3/4 in. triangular pitch; 761 sq ft of heating surface is provided. The steam generator will have a shell constructed of 3-1/2% nickel steel to specifications SA-350 Grade LF-3 or SA-203 Grade E. The inside diameter of the shell is 30 in.; the overall height is approximately 19 ft. The tube sheet will be SA-350 Grade LF-3, weld clad with a minimum of 1/4 in. of Inconel to specification SB-304 for protection against corrosion by the primary coolant. The U-tubes will be Inconel of nuclear quality specifications and will be required to conform to the ASME condenser and heat exchanger tube specification SB-163, including all optional tests. The primary channel will be constructed of SA-203 Grade E clad with Type 304 stainless steel. Bolting materials joining low alloy ferritic steel or clad components shall be in accordance with specification SA-320 Grade L-43 and SA-194 Grade 4. Bolting materials joining austenitic stainless steel components shall be in accordance with the general requirements of specifications SA-193 and SA-194, as further defined in ASME code case 1296 for A-286 material.

5.10 PRIMARY PUMP

The primary pump is a single stage, single suction, zero leakage centrifugal pump. The induction type motor is of canned construction. The motor shaft is vertical, the inlet to the pump casing is vertical, and the outlet is horizontal. The motor housing is flanged to permit removal of the motor and impeller for servicing. The motor is encased in a cooling coil to remove generated heat to an auxiliary cooling water system. The pump is "free floating" and is supported by the primary piping inlet elbow and the outlet piping. Additional pipe support or hangers will be provided to minimize pipe stresses.

The pump is designed to deliver 2250 gpm at a total head of 50 ft of water and at an operating pressure of 1200 psia. Design pressure of the pump is 1350 psia. Approximate weight of the pump is 2150 lb. Maximum diameter of the casing is 25-1/2 in. and the height from the inlet nozzle to the top of the motor housing is 64 in.

The pump will be constructed essentially of Type 304 or Type 316 stainless steel. Both materials appear suitable at this time. An analysis will be made, based on Alco experience and vendor suggestions, to determine which material is best for this particular application. Primary pump components, such as the tube liners and the can around the rotor, may be made of Inconel. Journal bearings, thrust disc inserts and thrust shoes may be of materials such as Inconel X, Stellite 19, Graphitar 14 and ceramics, or combinations thereof, as recommended by the manufacturer for this application. The highest quality and quality control will be required for all materials and fabrication.

5.11 REFERENCE

1. Matthews, F. T., "Pressure Loadings and Quenching in the PL-3 Vapor Containment," Alco Products, Inc., AP Note 411, to be published.

6.0 CORE THERMAL AND HYDRAULIC ANALYSIS

This section describes the thermal and hydraulic analysis performed on the core for the PL-3 reactor. The results and the calculational models employed are described and discussed. A summary of data at design conditions is given in Table 6.1.

6.1 STEADY STATE THERMAL ANALYSIS

6.1.1 Steady State - Analytical Model

The steady state thermal analysis was performed using the STDY-3 computer program.[1] This program performs a steady state, parallel channel thermal analysis, computing the enthalpy rise in the average rectangular water channel of a plate-type fuel element. Two-phase pressure drop calculations in the core are performed using semi-empirical correlations developed at WAPD. These calculations include spatial acceleration, elevation, friction, entrance and exit pressure losses.

Hot channel pressure drop is computed by modifying the nominal channel pressure drop by an input controlled plenum equation to compute the hot channel flow. A table of hot channel flow vs pressure drop is then compiled. This table is interpolated for a pressure drop value within two percent of the modified nominal pressure drop. The flow corresponding to this pressure drop is designated as the hot channel flow. Enthalpy, meat center line temperature, steam quality and departure from nucleate boiling ratio (DNBR) calculations are then performed using this flow.

The DNB correlation used in this analysis is recommended by WAPD[2] for bulk fluid enthalpies below 589 Btu/lb and is given by

$$\frac{\phi_{DNB}}{10^6} = (1.4) \left(\frac{H_j}{1000} \right)^{-0.72} \left(e^{-0.0012\, L/s} \right)$$

where:

ϕ_{DNB} = departure from nucleate boiling heat flux

H_j = fluid enthalpy in the j th axial region

L = length along channel

s = channel gap

The criteria for nucleate boiling in a channel is based upon a comparison between the average film drop and the film drop as calculated from the Jens-Lottes correlation.(3) If the average film drop exceeds or equals the film drop as calculated from the Jens-Lottes correlation, the channel is considered to be in nucleate boiling and the proper pressure drop correlation applied. Since the purpose of this criteria is not to establish the plate surface temperature but to establish the presence of nucleate boiling with its associated increased pressure drop and decreased flow, the analysis is based upon bulk or average parameters.

When calculating the maximum plate surface temperature, the film drops are compared in a similar fashion, but local parameters are inserted into the equations.

With these procedures utilized by the STDY-3 code to define nucleate boiling, it is conceivable to have a condition where some portion of the channel is essentially at a uniform surface temperature with reasonably high superheat and yet not in nucleate boiling. This has been observed and indicates that these elements are marginal with respect to the inception of nucleate boiling.

6.1.2 Power Distribution

The axial power distribution was calculated by the IBM-650 code, VALPROD,(4) using flux weighted nuclear constants. Figure 6.1 shows the normalized power vs. axial position. No flux suppressors were included at the inlet; consequently, a high power spike is present at the bottom of the core. The axial power distribution in the section of the control rod fuel plates below the active core is shown in Fig. 6.2.

TABLE 6.1
THERMAL AND HYDRAULICS - DATA SUMMARY

General	
Core thermal power	9.3 Mw
Primary system pressure	1200 psia
Coolant flow rate	2250 gpm
Number of passes thru core	1
Heat Flux Data	
Initial heat transfer area	560 ft^2
Core average heat flux	56,600 Btu/hr ft^2
Core maximum heat flux	
Spike	297,000 Btu/hr ft^2
Peak axial	182,000 Btu/hr ft^2

TABLE 6.1 (CONT'D)

Max-to-avg core power	
Spike	5.25
Peak axial	3.22
Minimum DNB ratio	6.44

Core Temperature Data

Core inlet temperature	454°F
Core mean temperature	470°F
Core average outlet temperature	487°F
Max plate surface temperature	576°F
Max fuel centerline temperature	597°F

Hydraulic Data

Core coolant flow area	2.089 ft^2
Flow channel thickness	0.133 in.
Core pressure drop	0.8 ft H_2O
Core average coolant velocity	2.5 ft/sec
Core average Reynolds Number	32,900
Total primary system pressure loss	61.0 ft H_2O

The radial power distribution was found using IBM codes and experimental correction factors.[5] The nominal channel average radial peaking factors, $Q(\overline{\Delta T})$, the hot channel average radial peaking factors, $Q(\Delta T)$, and the local radial peaking factors, $Q(\Delta \theta)$, are listed in Table 6.2. Since these nuclear factors are symmetrical about the center element, only one quadrant was analyzed. The numbering system for the elements is shown in Fig. 7.1.

TABLE 6.2
RADIAL PEAKING FACTORS
(T = 470°F, No Xenon)

Element Positions	$Q(\Delta T)$ Hot Channel Average	$Q(\Delta \theta)$ Local Value	$Q(\overline{\Delta T})$ Nominal Channel Average
44	1.58	1.70	1.33
45, 43	1.59	1.72	1.28
46, 42	1.38	1.48	1.08
47, 41	1.46	1.53	0.82
54, 34	1.43	1.65	1.23
55, 53, 35, 33	1.42	1.63	1.15
56, 52, 36, 32	1.25	1.44	0.94

TABLE 6.2 (CONT'D)

Element Positions	$Q(\Delta T)$ Hot Channel Average	$Q(\Delta \theta)$ Local Value	$Q(\overline{\Delta T})$ Nominal Channel Average
57, 51, 37, 31	1.36	1.51	0.74
64, 24	1.22	1.49	1.06
65, 63, 25, 23	1.22	1.46	1.00
66, 62, 26, 22	1.07	1.31	0.84
67, 61, 27, 21	1.22	1.40	0.67
74, 14	0.95	1.26	0.78
75, 73, 15, 13	0.94	1.24	0.69
76, 72, 16, 12	1.15	1.37	0.65

6.1.3 Mechanical Hot Channel Factors

The mechanical hot channel factors employed in this analysis are the factors which account for average and local deviations of meat length, uranium content and clad thickness. They are denoted respectively by $F_{M\Delta T}$ and $F_{M\Delta\theta}$. These factors were calculated using the latest fuel element dimensions and tolerances and the equations found in APAE No. 91[3] and are presented in Table 6.3. The methods for combining these mechanical factors and the nuclear peaking factors for use in the analysis are also given in APAE No. 91.

TABLE 6.3
MECHANICAL HOT CHANNEL FACTORS

	Stationary Element		Control Rods	
	$F_{M\Delta T}$	$F_{M\Delta\theta}$	$F_{M\Delta T}$	$F_{M\Delta\theta}$
Meat length	1.0357	1.0357	1.0299	1.0299
Uranium Content	1.005	1.025	1.005	1.025
Clad thickness	1.007	1.012	1.007	1.012
Composite mechanical factor	1.0481	1.0740	1.0422	1.0680

The hot channel factors, for plate spacing, rippling and flow maldistribution are applied in the STDY-3 code, to the individual dimensions to which they are related. Plate spacing is applied to the average dimension of the hot channel while the rippling ractor is applied to the local dimension of the hot channel. Maldistribution of channel-to-channel flow is included in the code as a plenum factor.

6.1.3.1 Uranium Content Deviation

The fuel plates forming the hot channel are assumed to contain the maximum allowable uranium density per plate. If on the average, the uranium content is higher by a certain percentage than the nominal, the water temperature will be increased since more heat will be transferred to the coolant moving between the two plates. The average hot channel factor describing this effect is the ratio of hot channel maximum fuel loading to the nominal channel loading.

In addition to the total uranium deviation per plate, which affects the total sensible heat in the channel, the possibility exists of having a non-homogeneous mixture which would affect the local film temperature rise. The local hot channel factor which accounts for this effect is the ratio of the maximum local fuel loading to the nominal local fuel loading.

6.1.3.2 Active Core Length Deviation

For a given volume of meat per plate, a reduction in the active fuel length increases the amount of meat per unit length and the meat per unit heat transfer area. It is assumed that the decreased length increases only meat thickness and not active width. This affects both the bulk coolant temperature rise and the film gradient. Both the average and the local hot channel factors are defined as the ratio of the nominal to minimum fuel meat lengths.

6.1.3.3 Clad Thickness Deviation

If the clad thicknesses on each side of a fuel plate are unequal, a greater portion of the total heat generated in the meat will pass out through the thinner clad because of lower thermal resistance. A hot channel, therefore, is defined as being composed of two fuel plates whose inner and outer clad thicknesses are at the minimum and maximum observed values, respectively. Both the average and local hot channel factors are then given by the ratio of the heat transferred through the inner sides to that transferred through cladding of nominal thickness. The average is based on the average clad thickness variation over a fuel plate while the local factor is based on the local thickness variation. The water temperatures and the film coefficients are assumed equal on both sides of each fuel plate.

6.1.4 Additional Factors Included in Thermal Analysis

To allow for uncertainties in the claculated power distribution, a nuclear uncertainty factor of 1.05 was used for the average conditions and a factor of 1.10 for the local conditions.

The fraction of fission heat released directly in the fuel plates was accounted for by power generation factors of 1.00 for average conditions and 0.95 for local conditions.

Deviations from the reference pressure of 1200 psia were accounted for by performing the thermal analysis at 1175 psia.

To account for ripples in the fuel plates caused by compressive stress resulting from the temperature differential between the meat and the side plates, values of rippling[6] were applied to the hot channel dimensions. A ripple ratio of 3.0 was used, resulting in a hot channel spacing of 0.172 in. for both the stationary and control rod elements.

To account for the maldistribution in channel-to-channel flow caused by lower plenum effects, maldistribution factors based on a $\pm$ 12% variation in mass flow for both the stationary and control rod elements were applied to the plenum factors.

6.1.5 Results at Reference Conditions of 9.3 Mw

Table 6.4 gives the steady state results at the reference conditions of 9.3 Mw core power, 470°F mean core temperature and 33°F core temperature rise. The minimum departure from nucleate boiling ratio (DNBR) occurs at the inlet spike of fuel element number 45. The value of 6.44 is considerably above the minimum allowable of 2.0. No nucleate boiling is indicated, and it may be concluded from the above results that the core can be operated safely at the reference design conditions.

6.1.6 Effects of Varying Core Temperature Rise

The effects of varying the core temperature rise for several core powers are shown in Table 6.5. The results are for fuel element 45, the hottest element in the core. Nucleate boiling is observed only at the highest power and core temperature rise. Since the minimum DNBR at each power level increases with increasing core temperature rise (decreasing coolant flow rate) the possibility of reducing the reference design coolant flow rate is indicated.

TABLE 6.4
RESULTS OF THERMAL ANALYSIS AT REFERENCE CONDITIONS

Core Power Level - 9.3 Mw
Average Core Temperature - 470°F
Core Temperature Rise - 33°F

Core Position	Nominal Channel Flow 10^5 lb/hr ft^2	Hot Channel Bulk Outlet Temperature °F	Hot Channel Max. Plate Surface Temperature °F	$(DNBR)_{Min}$ at Inlet Spike	Nucleate Boiling
44	7.015	493.8	575.8	6.45	0
45	6.750	505.6	575.8	6.44	0
46	5.700	496.6	575.4	7.40	0
47	4.325	518.9	576.2	7.22	0
54	6.490	502.4	576.1	6.71	0
55	6.065	505.3	576.1	6.79	0
56	4.960	506.1	575.8	7.67	0
57	3.900	517.6	576.0	7.35	0
64	5.590	502.0	575.7	7.44	0
65	5.275	494.7	575.1	7.51	0
66	4.430	504.1	575.4	8.45	0
67	3.530	516.8	575.7	7.90	0
74	4.115	502.1	575.1	8.81	0
75	3.640	504.2	575.0	8.94	0
76	3.430	515.4	575.6	8.06	0

TABLE 6.5
RESULTS OF THERMAL ANALYSIS FOR ELEMENT 45

Average Core Temperature - 470°F

Core Power Mw	Percent of Coolant Flow at 8 Mw & $\Delta T_{core} = 28^{o}F$	Core Temp Rise ΔT_{core} °F	Nominal Channel Flow 10^5 lb/hr ft^2	Hot Channel Flow 10^5 lb/hr ft^2	Hot Channel Bulk Outlet Temp °F	Hot Channel Max Plate Surface Temp °F	Min. DNBR at Inlet Spike	Nucleate Boiling
8.0	100.0	28	6.750	5.940	498.94	576.00	7.48	0
8.0*	100.0	28	6.750	5.940	470.06	550.06	7.90	0
9.3	100.0	33	6.750	5.940	505.6	575.8	6.44	0
7.4	92.5	28	6.250	5.500	498.90	574.95	8.08	0
7.4	74.0	35	5.000	4.675	503.39	575.79	8.13	0
7.4	57.5	45	3.885	3.846	509.17	575.79	8.19	0
7.4	47.0	55	3.180	3.148	517.53	575.79	8.25	0
8.8	92.5	33.3	6.250	5.500	504.87	576.26	6.83	0
8.8	74.0	41.6	5.000	4.675	508.83	576.26	6.86	0
8.8	57.5	53.5	3.885	3.846	515.62	576.26	6.94	0
8.8	47.0	65.5	3.180	3.148	526.35	576.26	6.99	yes

* 440°F Mean Core Temperature

6.2 TRANSIENT ANALYSIS

The analysis of the thermal transients following a loss of flow accident has been performed by means of the ART-02[7] code to determine if this circumstance represents a hazardous condition.

6.2.1 Method of Analysis

The ART-02 code utilizes a one-dimensional model to predict the behavior of a water-cooled and moderated reactor with plate-type elements during transients which are slower than a prompt excursion. The reactor model considered in the ART code is either a one or two pass core operating initially in a steady-state condition. The reactor is subjected to a variation in flow rate with or without a variation in reactivity, as induced by control rod motion, to simulate a loss of flow with or without scram. With this information, a reactor kinetics calculation is performed to determine the core power as a function of time.

The behavior of a single coolant channel through the core represents the nominal response of each pass. Thermal calculations are performed for one or more additional channels in each pass. The additional channels are treated as hot channels to represent possible extremes in dimensions, pressure drop and heat input. The hot channel is analyzed to determine reactor safety during the loss of flow transient under consideration.

Fog or homogeneous flow is assumed for the hydrodynamic calculation. Though this is somewhat less rigorous than the slip flow model used in ART-04,[8] it was utilized in the absence of good void fraction data.

For analytical purposes, the problem of the loss of primary coolant flow is divided into two parts. The first part begins at the instant of pump failure and extends for about 5 sec, ending when natural circulation becomes the predominant driving force for coolant flow. The second part analyzes the natural circulation flow from decay heat. The ART code is limited because:

1. It analyzes only reactor core and makes no account for the secondary system effects. This representation will be valid up to the time it takes the fluid to complete one cycle of the primary loop.

2. The code, as used, has no provision for the reactivity changes due to steam formation, and as the quality increases, the model becomes much to conservative to formulate a reasonable representation of the actual conditions in the reactor. For this reason, the saturation region pressure data is only programmed up to 40 percent quality and all problems are discontinued when the exit quality exceeds this criterion.

3. Most important, the burnout heat flux, ϕ_{DNB}, is limited to a range of enthalpies less than 1000 Btu/lb and mass velocities greater than 0.1 x 10^6 lb/hr ft^2.

6.2.2 Departure from Nucleate Boiling Correlations

The departure from nucleate boiling correlations used in the ART code are not identical to those used in STDY-3, since during the loss of flow transients, there is considerable variation in mass velocity, (G_i^L). The following correlations with their indicated ranges were used:

$$\left(\frac{\phi_{DNB}}{10^6}\right)_{ji} = 0.325 \left(\frac{H_{ji}}{10^3}\right)^{-2.5} (F_c)_{2i} \text{ ; for } G_i^L \leq 1.6 \times 10^6 \text{ lb/hr ft}^2;$$

$$\left(\frac{\phi_{DNB}}{10^6}\right)_{ji} = 0.240 \left(1 + \frac{G_i^L}{10^7}\right) \left(\frac{H_{ji}}{10^3}\right)^{-2.5} (F_c)_{2i} \text{ ; for } G_i^L > 1.6 \times 10^6 \text{ lb/hr ft}^2;$$

where:

ϕ_{DNB} = departure from nucleate boiling heat flux

G_i^L = local mean mass velocity

$$= \left(\frac{G^L}{G}\right) x\, G_i$$

$\frac{G^L}{G}$ = local mean mass velocity correction factor

H_{ji} = local fluid enthalpy

$(F_c)_{2i}$ = channel length and non-uniform local heat flux correction factor

6.2.3 Power and Flow Distribution

The loss of flow analysis was performed for element 45 at a core power level of 8 Mw. The axial and radial power distributions were the same as for the steady state analysis. For conservatism, it was assumed that the rods were not scrammed so that the core power generation during the transient becomes a function of the temperature coefficient only.

As in the steady state analysis, the flow maldistribution used in the hot channel is based on isothermal measurements which showed a $\pm$ 12% maldistribution. An initial mass flow for the element of 0.675 x 10^6 lb/hr ft^2 was obtained from the steady state analysis.

6.2.4 Hot Channel Factors

The hot channel factors used in this analysis were formulated for average conditions which affect heat generation rates and for local conditions which affect local heat fluxes in the DNB correlations. The numerical values of the individual factors used in the transient analysis are equal to those used in the steady state analysis. However, the input format for the ART code is somewhat different than that for the STDY-3 code. The methods of formulating these factors for the transient analysis are given in APAE 91.[3]

Unlike the previous steady state thermal analyses, there is no hot channel factor for plate spacing deviations caused by rippling. This factor is omitted because the ART code will accept a local mass velocity correction factor as direct input to evaluate variations in the local mass velocity. The variations in plate spacing which causes variations in local mass velocities are fully covered by the local correction factor.

6.2.5 Flow Coastdown

The reactor coolant flow coastdown, the driving force for the transient portion of the code, was represented by a decreasing flow in the nominal channel of the form

$$\frac{G}{G_o} = \left(\frac{1}{1 + bt} \right)^{1.25}$$

where t is the time in seconds. For conservatism, the analysis is based on the most adverse flow coastdown, that of a stuck or frozen pump impeller. A value of b for this case was taken as 2.2 sec^{-1}. This is based on a preliminary pressure loss analysis of the loop.

6.2.6 Allowable DNB Ratio for Transient Operation

The minimum allowable DNBR in a reactor transient was conservatively set at 1.5, compared to the steady state minimum allowable DNBR of 2.0. If the most critical element in the PL-3 does not have a minimum DNBR of less than 1.5 in a loss of flow transient, the entire reactor is deemed safe for the duration of the thermal transient.

6.2.7 Results of Transient Thermal Analysis

The transient behavior of the hottest element in the core, element 45, has been determined for a loss of flow incident without scram. The steady state conditions at the start of the transient were: 1) core power - 8 Mw, 2) inlet temperature - 456^{o}F, 3) system pressure - 1175 psia, and 4) coolant flow rate - 2250 gpm.

The flow coastdown, the power coastdown and the minimum departure from nucleate boiling ratios (DNBR) are given in Figs. 6.3, 6.4 and 6.5 respectively. Because of the enthalpy and mass velocity limitations in the ART code, the plot of DNBR vs time in Fig. 6.5 has been stopped 2.9 sec after initiation of the transient.

The most significant result of the analysis is that the minimum DNBR never goes below the steady state value of 7.48 and is therefore well above the minimum allowable value of 1.5 given in Section 6.2.6. The oscillation in the DNBR occurring about 1.8 sec after the start of the transient is caused by the development of some nucleate boiling in the hot channel. Nucleate boiling during a transient is not considered to violate the condition of operation without nucleate boiling. Because of the wide margin of safety existing during the transient initiated at a core power of 8.0 Mw, it is expected that a transient from the reference condition of 9.3 Mw could be handled by the core with adequate safety.

6.3 PRIMARY SYSTEM PRESSURE DROP

The pressure drop in the primary loop has been calculated on the basis of the $470^{\circ}F$ mean water temperature and 2250 gpm flow rate. The loop consists of three main parts: the loop piping, the reactor vessel, and the steam generator.

The primary loop, shown in Dwg. AEL-762, has a total of 24.5 ft of 6 in., Schedule 80 pipe. This includes five 90° bends, two 25° bends, and one 45° bend. Of these, the 90° bend immediately preceding the pump is not included in the pressure loss calculations because this is an integral part of the pump, and the latter cannot be tested without it. There are no valves in the main pipe line. Based on the flow rate of 2250 gpm, the resulting kinetic head loss is 15.1 ft H_2O, at velocity of 25.4 ft/sec. At the mean temperature of $470^{\circ}F$, the Reynolds Number is 8.8×10^6 for an inside diameter of 5.761 in., and the friction factor is 0.0081. This yields a frictional pressure drop of 5.0 ft; thus, the total loss in the piping is 20.1 ft H_2O.

In this type of reactor and control rod design, the overall core pressure drop is dependent entirely on the control rod flow requirements. For this analysis the pressure loss was obtained from SM-2 data, since an SM-2 type rod inlet is planned for the PL-3 and the cap and exit configuration will have a pressure drop similar to that of the SM-2 rods. From APAE No. 91[3] the pressure drop for an SM-2 type control rod was found to be 2.22 ft of water at 100 gpm flow through each element, varying as the 1.91 power of the flow rate. In the 7 x 7 core each control rod should have a flow of about 60 gpm so that the core pressure drop is about 0.8 ft of water.

In the reactor vessel, shown in Dwg. AEL-763, the frictional pressure drops are in the nozzle and in the core itself. The total straight length in the two nozzles causes a pressure drop of 0.4 ft H_2O. The kinetic pressure loss in the vessel due to inlet nozzle diffusion and discharge was calculated from diffuser performance data, assuming a one velocity head loss at the diffuser discharge. The baffle loss is adjustable and represents an estimate of the resistance required to achieve adequate flow equalization around the vessel. The exit plenum and nozzle loss was taken directly from experience with the SM-2 flow test at the PL-3 flow rate.

The steam generator pressure drop has been selected from values reported for heat exchangers of similar thermal duty and flow rate. Final calculations for the PL-3 configuration have not been performed by the steam generator manufacturer, but it is expected that the assumed pressure drop of 25.5 ft of water is adequate.

The resultant component and total pressure drops are given in Table 6.6 for the reference flow of 2250 gpm. Figure 6.6 shows these results for a range of flow rates. The theoretical pumping work required at the reference conditions of flow and head loss is 26.9 hydraulic horsepower.

TABLE 6.6
PRIMARY SYSTEM PRESSURE LOSSES

Component	Head Loss - ft H_2O
Piping	20.1
Reactor core	0.8
Reactor nozzle (friction loss)	0.4
Reactor vessel (kinetic pressure loss)	
Inlet nozzle diffusion and discharge	7.2
Horizontal flow distributing baffle between core and vessel (not shown)	1.2
Exit plenum and nozzle	0.3
Steam generator	25.5
Total	55.5
Plus 10% contingency	61.0

6.4 REFERENCES

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7. Meyer, J. E., Smith, R. B., Gelbard, H. A., George, D. G., Peferson, W. D., "ART-02 A Program for the Treatment of Reactor Thermal Transients on the IBM-704", WAPD-TM-156, November 1959.

8. Meyer, J. E., Peterson, W. D., "ART-04 - A Modification of the ART Program for the Treatment of Reactor Thermal Transients on the IBM-704," WAPD-TM-202, July 1960.

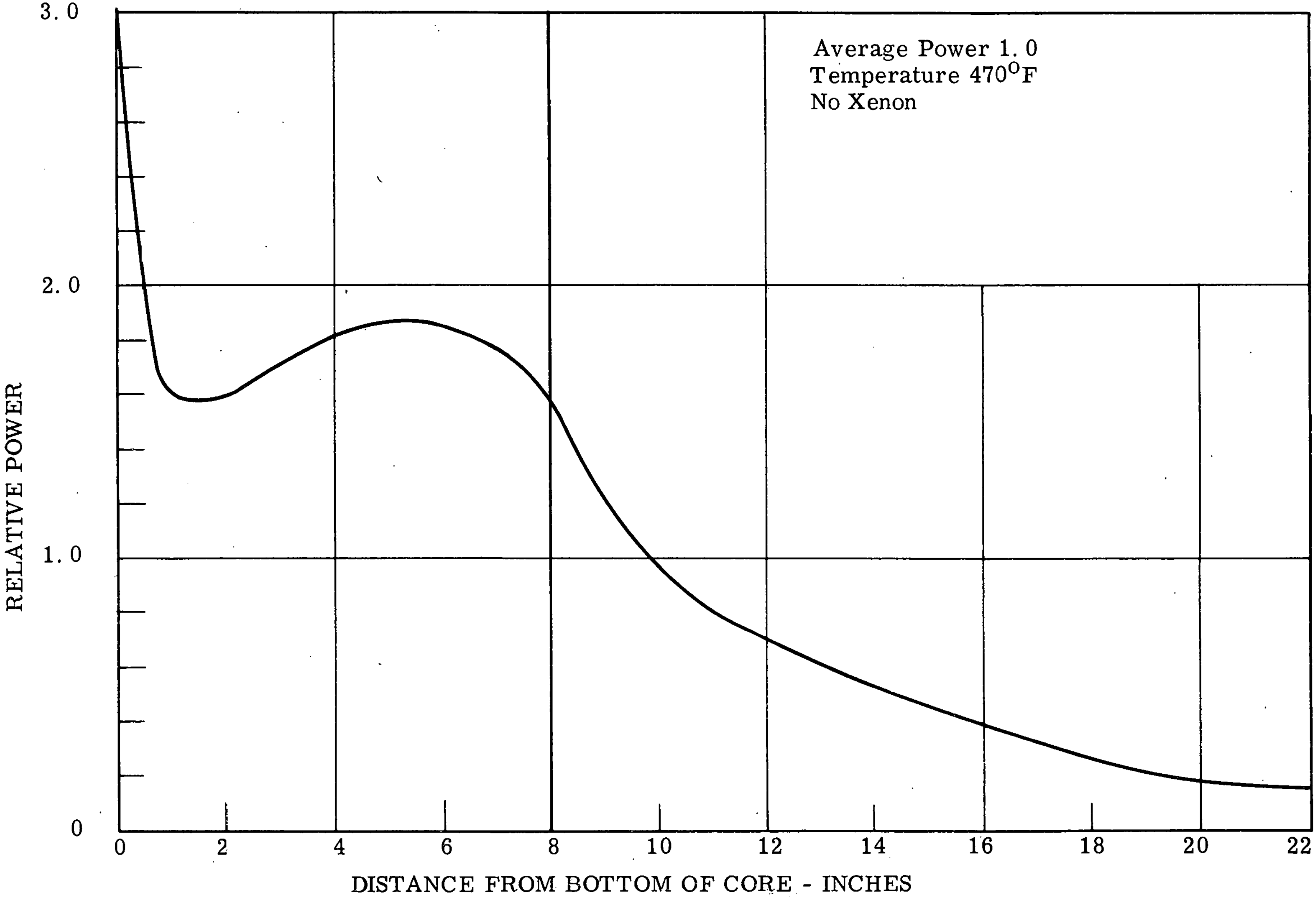

Figure 6. 1. PL-3 Core Axial Power Distribution

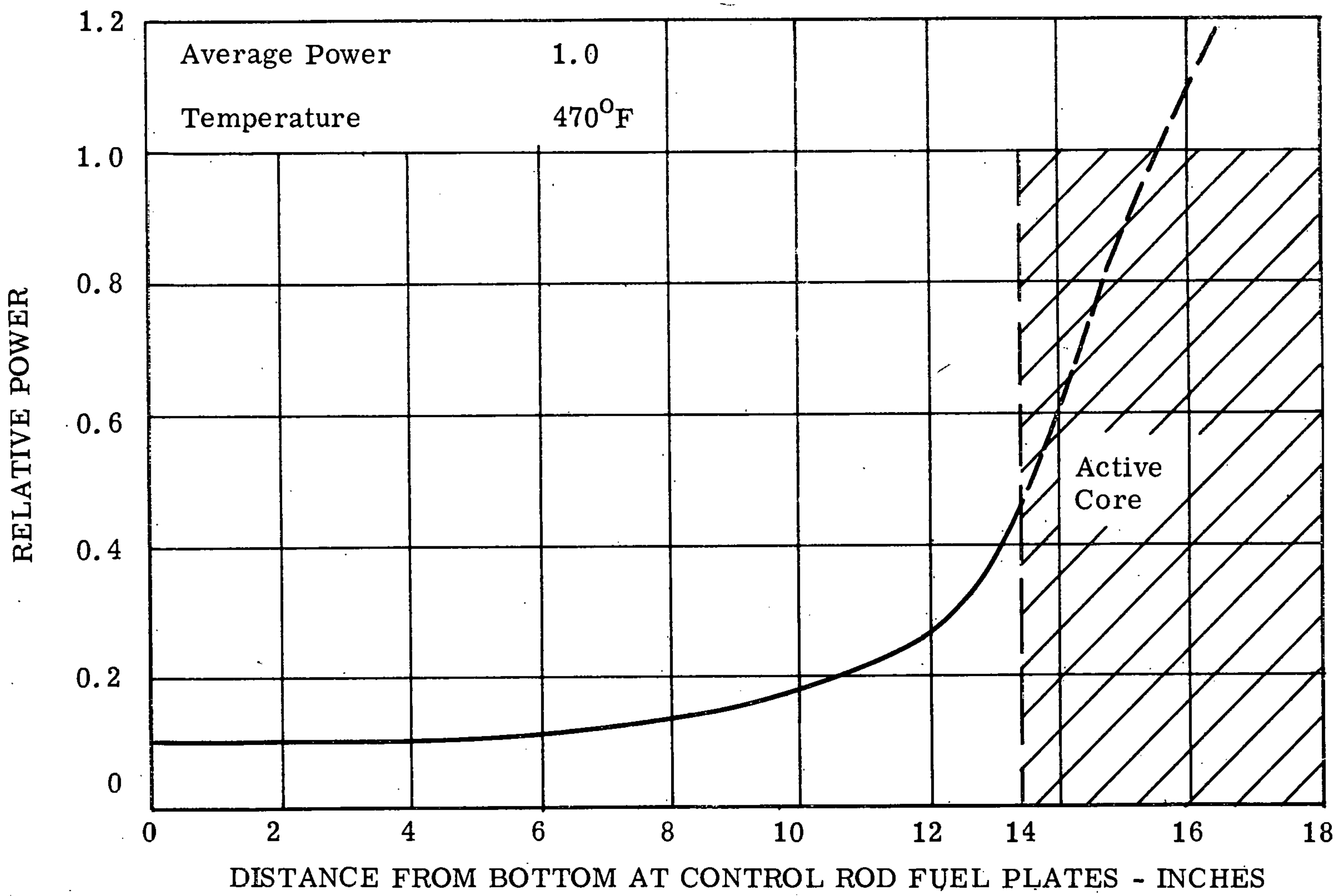

Figure 6.2. Axial Power Distribution in Control Rod Fuel Plates

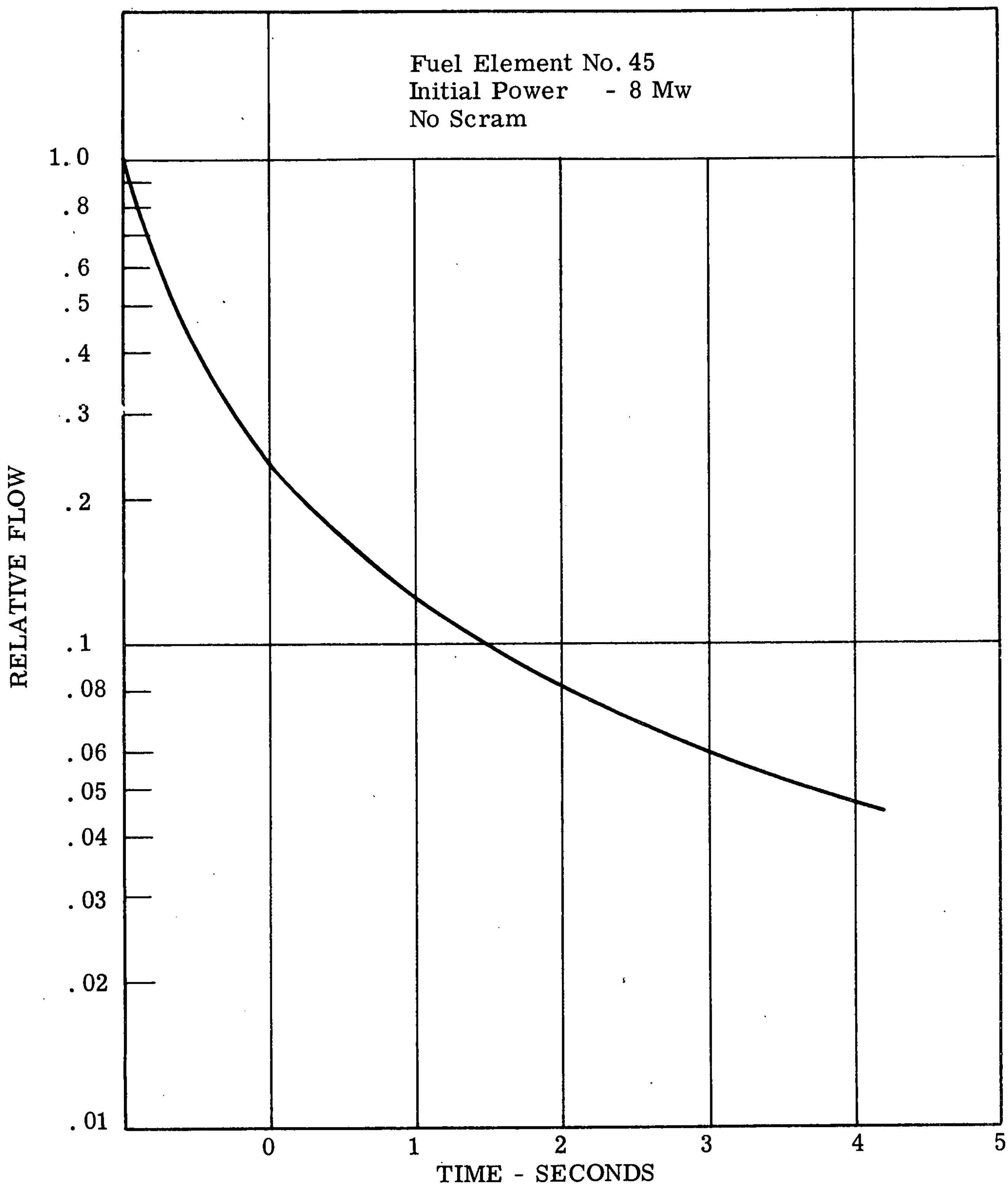

Figure 6.3. Flow Coastdown Following Pump Failure

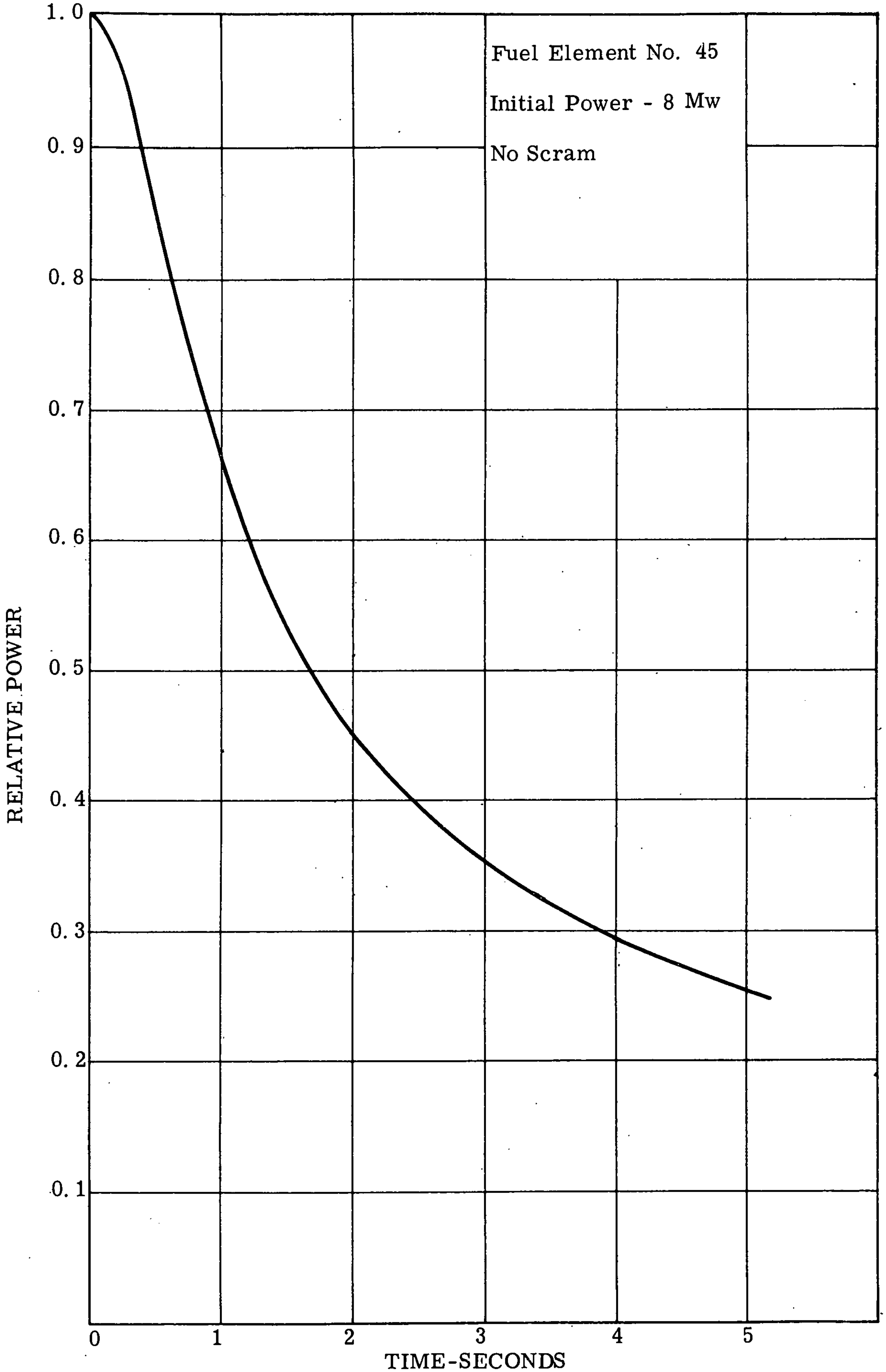

Figure 6.4. Power Coastdown Following Pump Failure

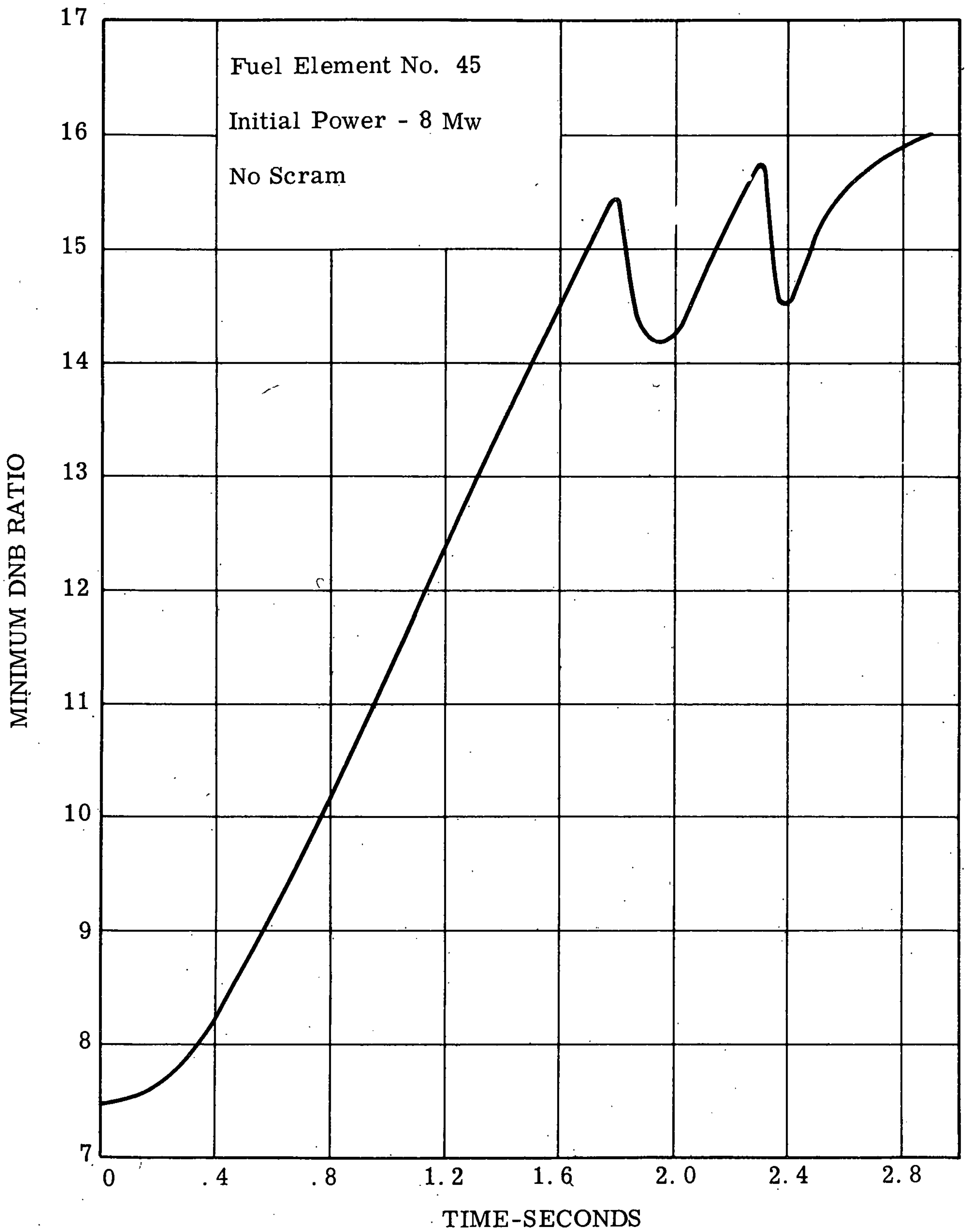

Figure 6.5. Minimum DNB Ratio Following Pump Failure

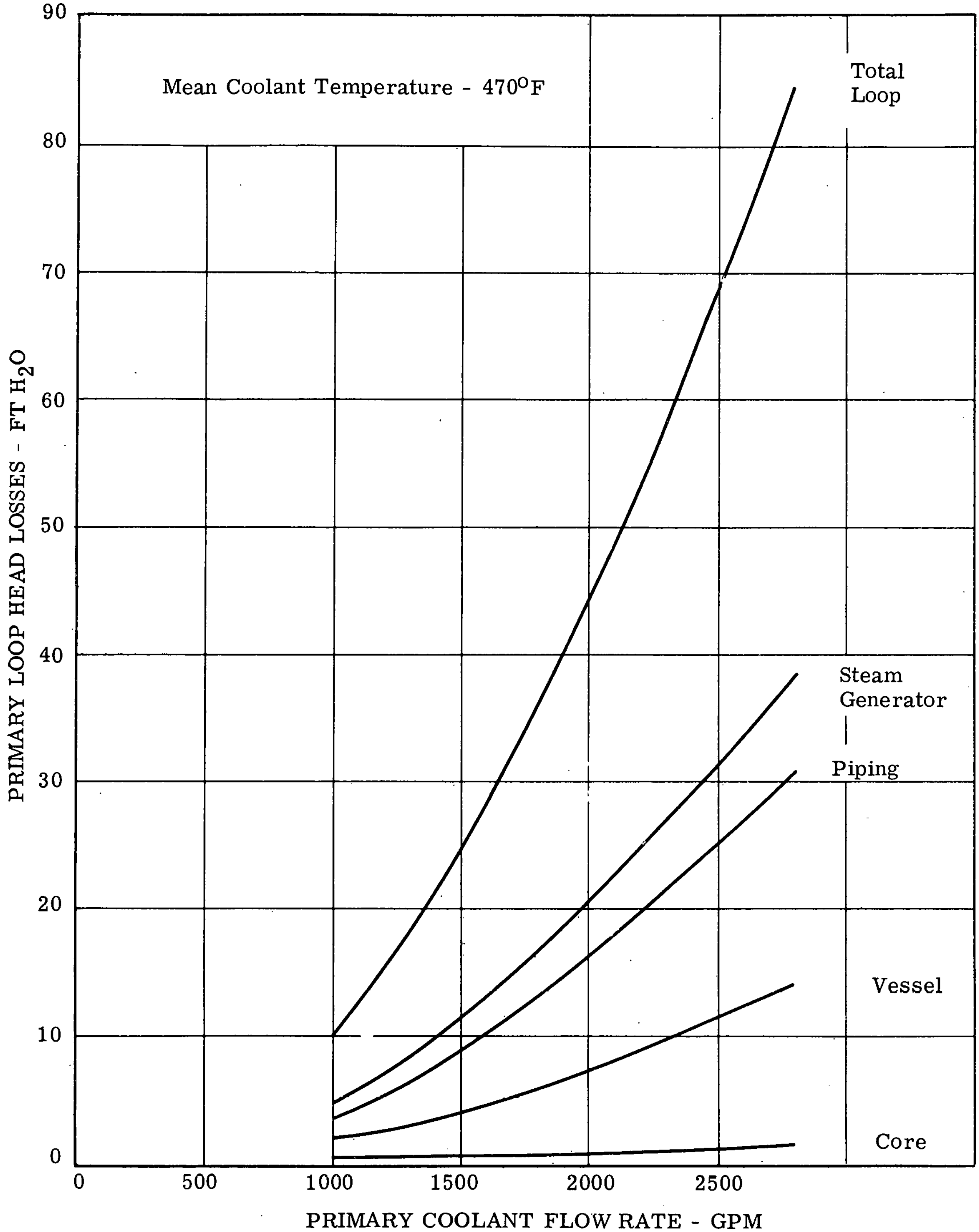

Figure 6. 6. PL-3 Primary System Pressure Losses

7.0 CORE PHYSICS ANALYSIS

This section describes the nuclear analysis performed on the preliminary design PWR core for the PL-3 reactor. The analysis includes reactivity and lifetime calculations, control rod worth, stuck rod data and flux and power distributions. Extrapolations of measurements made on similar cores have also been used to predict certain nuclear characteristics for which direct measurements and/or analyses are not available.

7.1 GENERAL DESCRIPTION OF CORE

7.1.1 Geometric and Material Configuration

The PWR reference design core utilizes Type II fuel elements and absorbers similar to those employed in the SM-1 Core II. A schematic cross-section of this core is shown in Fig. 7.1. The core employs flatplate fuel elements composed of a highly enriched UO_2 - stainless steel matrix and clad with stainless steel. The boron burnable poison is intimately mixed with the UO_2 and stainless steel in the fuel matrix. The fuel plates are 30 mils thick, including the 5 mil cladding on each side of the fuel matrix. Control is accomplished by the use of 7 box-type absorbers with fuel element followers arranged in an open array and operated as a bank. The absorbers are composed of Eu_2O_3 in stainless steel and clad with stainless steel. A complete tabulation of data on the core is presented in Table 7.1.

7.1.2 Calculational Techniques

The calculational models employed for the PL-3 nuclear analysis were essentially the same as those used previously at Alco to successfully predict the nuclear characteristics of other parallel-plate, pressurized water reactors. The MUFT III, IBM-650 code[1] was used to obtain the flux-averaged fast parameters. Thermal parameters were obtained from the P_3 IBM-650 code[2] which calculates average macroscopic thermal cross-sections for the fuel element and the intra-element spatially-dependent thermal flux. The microscopic cross-sections employed were averaged over a hardened Maxwell-Boltzmann distribution.

The CANDLE-2 IBM-704 burnup code[3] was employed for predictions of core lifetime and fuel and poison burnup distributions. Power distributions and local power peaking factors were calculated using the two-dimensional PDQ-2 (x, y) code[4] for the IBM-7090 computer in conjunction with the axial WINDOW-SHADE, IBM-650 code.[5]

TABLE 7.1
CORE DATA

General Description

Configuration: 7 x 7 array of elements, corners missing.	
Fuel: UO_2-SS matrix, fully enriched, SS clad	
Burnable poison: ZrB_2 in SS-UO_2 matrix	
Absorber: EU_2O_3-SS matrix, SS clad	
Structural material: Stainless steel	
Moderator: Water	
Average temperature	470^oF
Pressure	1200 psia
Average power level	7.4 Mw
Core life	2 yrs
Number of fuel elements	
Stationary	38
Movable	7
Number of absorbers	7
Initial loadings	
U-235	22.48 kg
U-238	1.7 kg
B-10	15.75 gm
Total number of cells	45
Cell size	2.9375 in.x2.9375 in.
Active core height	21.75 in.
Effective core diameter	22.2 in.
Core volume	138.0 liters
Initial heat transfer area	560.0 ft^2
Average metal/water volume ratio	0.26

Fuel Element

Type: Rectangular assembly of flat fuel plates brazed into side plates with attached end boxes.	
Meat material: Spherical UO_2 (26%), ZrB_2 and Type 347 SS	
Clad material: Type 347 SS	
Side plate material: Type 347 SS	
End box material: Type 304L SS	
Fuel element active length	
Stationary	21.75 in.
Movable	21.125 in.
Fuel element cross section	
Stationary	2.844 in. x2.863 in.
Movable	2.624 in. x2.619 in.
Element loading U-235	
Stationary	515.16 gm
Movable	417.76 gm

TABLE 7.1 (CONT'D)

Element loading B-10	
Stationary	0.361 gm
Movable	0.293 gm
Number of plates per element	
Stationary	18
Movable	16
Plate width	
Stationary	2.778 in.
Movable	2.558 in.
Plate thickness	0.030 in.
Clad thickness	0.005 in.
Meat width	
Stationary	2.54 in.
Movable	2.32 in.
Meat thickness	0.020 in.
Water gap between plates	0.133 in.

Absorber Element

Type: Box assembly of four flat plates.	
Meat material: EU_2O_3 and stainless steel	
Clad material: Type 347 SS	
Active length	20.81 in.
Active width	2.20 in.
Active thickness	0.096 in.

Physics

Maximum reactivity	
at 470°F	11.3%
at 68°F	17.1%
at 40°F (4°C)	17.3%
Control bank worth at 68°F	
Six rods	19.3% ρ
Seven rods	23.2% ρ
Shutdown margin, one rod fully withdrawn	
at 68°F	2.2% ρ
at 40°F (4°C)	2.0% ρ
Temperature coefficient	
at 470°F	$-3.2 \times 10^{-4}\% \ \Delta\rho/°F$
at 68°F	$-0.6 \times 10^{-4}\% \ \Delta\rho/°F$
at 40°F (4°C)	$-0.5 \times 10^{-4}\% \ \Delta\rho/°F$
Average thermal flux, hot clean core	1.0×10^{13} neuts/cm^2 sec

TABLE 7.1 (CONT'D)

Fuel burnup (U-235) at end of life	
Average	35%
Maximum	73%
Boron burnup at end of life	
Average	95%
Maximum	100%
Max-to-avg core power at 470°F	
Gross radial	1.41
Local radial	1.72
Gross axial	1.87

Energy dependent neutron distributions on the pressure vessel were calculated using the P1MG-2, IBM-704 code[6] and the PDQ-2 (r, z) IBM-7090 code.[4] A complete description of these codes may be found in previous Alco reports.[7,8,9] Experimental data on SM-1 Core I,[10] SM-1 Core II,[11] PM-2A Cores I[12] and II[13], and the SM-1A Core I[11] reactors were used to confirm the validity of these analytical models since the cores are similar in many respects to that of the PL-3 preliminary design.

7.2 REACTIVITY AND LIFETIME

The lifetime of the PL-3 core was calculated using the CANDLE-2, IBM-704 code[3] and extrapolations of the measured SM-1 Core I lifetime.

The CANDLE-2 code will calculate reactivity, critical rod bank positions, U-235 spatial distributions, B-10 spatial distributions, thermal and fast flux spatial distributions, and power distributions, all as a function of core life. The maximum reactivity as well as the most adverse power distribution occurs at the start of core life, thereby simplifying the calculational problems somewhat.

Figure 7.2 shows both the calculated and extrapolated measured reactivity as a function of core energy release. The measured values were obtained from the SM-1 Core I measurements at 440°F. At present, the differences between SM-1 Core I and the PL-3 PWR core are the mean core operating temperature, which is 470°F for PL-3 compared to 440°F for SM-1, and the use of integral Eu_2O_3 flux suppressors at the top of the control rod fuel element instead of the external Haynes combs.[14] The measured SM-1 Core I curve has been corrected to account for two fresh fuel elements which were substituted at 10.5 MWYR. No such change is contemplated in the PL-3 design.

The lifetime of the PL-3 core is estimated to be 15.2 MWYR or 2.05 yrs at 7.4 Mw. The maximum reactivity occurs at the start of life; 17.1% at 68°F, and 11.3% at 470°F, with no xenon.

7.3 CONTROL ROD WORTH AND STUCK ROD CRITERIA

The worth of the bank of seven control rods was extrapolated from measurements on SM-1 Core I; see Fig. 7.3. This measured worth of 23.3% was used to obtain an equivalent control rod bank thermal absorption cross-section with which control rod critical bank positions were then calculated for various operating conditions;(14) see Table 7.2.

A requirement of the PL-3 reactor is that the core can be shutdown with any one control rod stuck full out under any operating condition. This condition will easily be met in the PL-3 PWR core since it has been demonstrated that the SM-1 Core I can be shutdown with almost one and one half control rods stuck out at 68°F.(15) The shutdown margins at 68°F and 4°C are estimated to be 2.2% and 2.0% $\Delta\rho$, respectively.

In the unlikely event that two control rods are stuck full out, the boron injection system will insure complete shutdown of the core. It should be noted that the core reactivity is maximum at the start of life and continually decreases with core burnup thereby improving the shutdown margin throughout core life.

TABLE 7.2
CALCULATED 7 ROD BANK POSITIONS FOR PL-3 OPERATING CONDITIONS

Operating Condition	Control Rod Bank Position (Inches from Bottom of Core)
68°F - Clean	5.3
440°F - Clean	7.6
440°F - Eq. Xenon	8.6
470°F - Clean	8.0
470°F - Eq. Xenon	9.0

7.4 TEMPERATURE COEFFICIENT OF REACTIVITY

Figure 7.4 shows the measured values of the SM-1 Core I temperature coefficient of reactivity as a function of temperature. This curve applies to the PL-3 design as well and shows a negative temperature coefficient over the operating temperature range of the core. An extrapolation of this curve shows that the coefficient should still be negative in the 40°F (4°C) temperature range, thus meeting the PL-3 criterion.

7.5 POWER DISTRIBUTIONS

Power distributions for the open 7 rod array were calculated using the PDQ-2 (x, y) code. Since the mean core temperature was not firmly established at the time these calculations were performed, the calculations used a mean core temperature of 440°F. This will introduce some conservatism into the power peaking calculations since the power peaking decreases with an increase in core temperature.[9] Table 7.3 lists the radial power generation factors for each element. The element numbering system is shown in Fig. 7.1. The axial power distribution is given in Table 7.4. The power distribution at the start of life, without xenon, will be the most adverse since the control rod bank is continually withdrawn as the core burns out.

Although the seven rod bank is further out than the five rod bank in the SM-1 Core I, the axial power peaking in PL-3 is 6% greater than SM-1 Core I. The increased reactivity worth of the 7 rod bank depresses the power in the upper part of the core, causing an increase in the power generation in the lower section of the core.

The power distribution through the center of the core, parallel to and perpendicular to the fuel plates are shown in Figs. 7.5 and 7.6, respectively.

TABLE 7.3
NORMALIZED RADIAL POWER GENERATION FACTORS
(440°F; NO XENON)

Element No.	Average Power in Element	Average Power in Hottest Plate of Element	Maximum Power In Element
44	1.33	1.58	1.70
45	1.28	1.59	1.72
46	1.08	1.38	1.48
47	0.82	1.46	1.53
54	1.23	1.43	1.65
55	1.15	1.42	1.63
56	0.94	1.25	1.44
57	0.74	1.36	1.51
64	1.06	1.22	1.49
65	1.00	1.22	1.46
66	0.84	1.07	1.31
67	0.67	1.22	1.40
74	0.78	0.95	1.26
75	0.69	0.94	1.24
76	0.65	1.15	1.37

TABLE 7.4
NORMALIZED AXIAL POWER DISTRIBUTION
(440°F; NO XENON)

Inches from Bottom of Core	P(z)	Inches from Bottom of Core	P(z)
0	3.05	12	0.70
1	1.60	13	0.61
2	1.59	14	0.52
3	1.72	15	0.45
4	1.82	16	0.38
5	1.87	17	0.32
6	1.85	18	0.27
7	1.77	19	0.22
8	1.61	20	0.17
9	1.25	21	0.14
10	0.95	22	0.15
11	0.81		

Note: Rod Bank Position is 8 in. from bottom of core.

7.6 FAST NEUTRON EXPOSURE OF VESSEL

The integrated fast neutron flux (E > 1 Mev) for the SM-1, 7 x 7 array, operating at 440°F for a 20 yr period at an average 7.4 Mw level (148 MWYR) is shown in Fig. 7.7. The incident flux of fast neutrons with energies greater than 1 Mev on the surface of a 40 in. i.d. pressure vessel is seen to be 1.1×10^{20} neuts per cm^2. The one dimensional, multigroup P1MG-2[6] and the two dimensional, few group PDQ-2 (r, z) codes[4] were employed to determine the integrated fast neutron flux. The calculations were normalized to measurements of fast neutron flux within the SM-1 mockup carried out at the Alco critical facility. These data were increased by a factor of 1.3 to account for a probable error in the measurements. The calculated axial variation in fast flux at the core centerline, core-reflector interface, and at a radial distance of 25 in. from the core centerline is shown in Fig. 7.8.

A comparison of the calculations with measurements using the S(n, p)P threshold detector, as carried out at the Alco critical facility, is shown in Fig. 7.9. The results show good agreement between theory and experiment for fast neutron fluxes above the effective S(n, p)P threshold energy of 2.9 mev.

Neither the mean temperature of the core nor the actual reflector configuration were determined at the time of these calculations. Consequently, the calculations were performed at a mean core temperature of 440°F for an all-water reflector. Increasing the mean core temperature to 470°F and/or the addition of a steel reflector is not expected to significantly change the results of the calculation.

7.7 REFERENCES

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14. Williamson, T. G., Editor, "Reactor Analysis APPR-1 Core II," APAE No. 32, Issued July 15, 1958.

	12	13	14	15	16	
21	22	23 X	24	25 X	26	27
31	32	33	34	35	36	37
41	42 X	43	44 X	45	46 X	47
51	52	53	54	55	56	57
61	62	63 X	64	65 X	66	67
	72	73	74	75	76	

X Fuel-absorber control elements.

Figure 7.1. PL-3 Core Array with Type 2 Fuel Elements

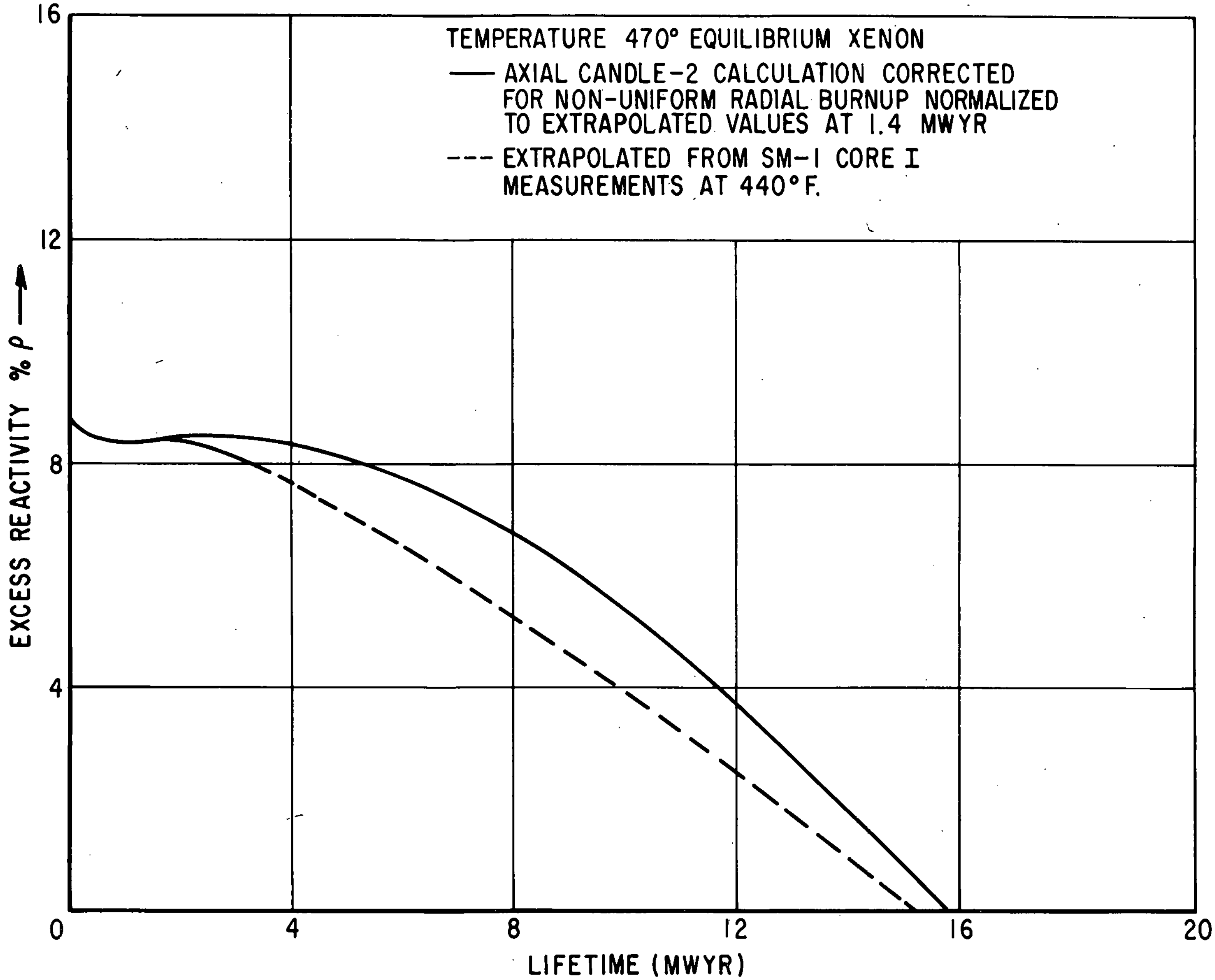

FIGURE 7.2 PL-3 CORE EXCESS REACTIVITY VS. LIFETIME

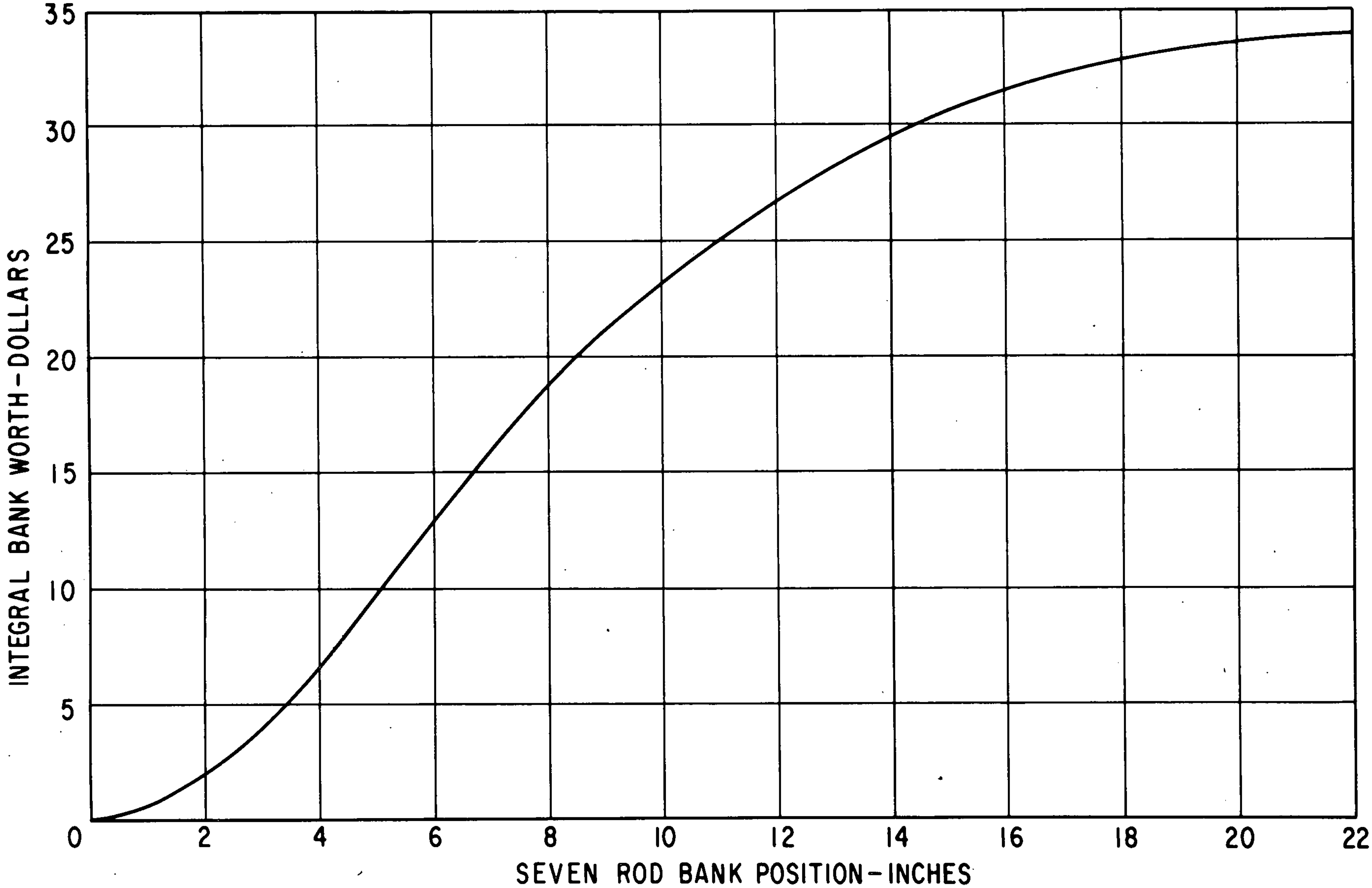

Figure 7.3 SM-1 Core I Seven Rod Bank Integral Worth

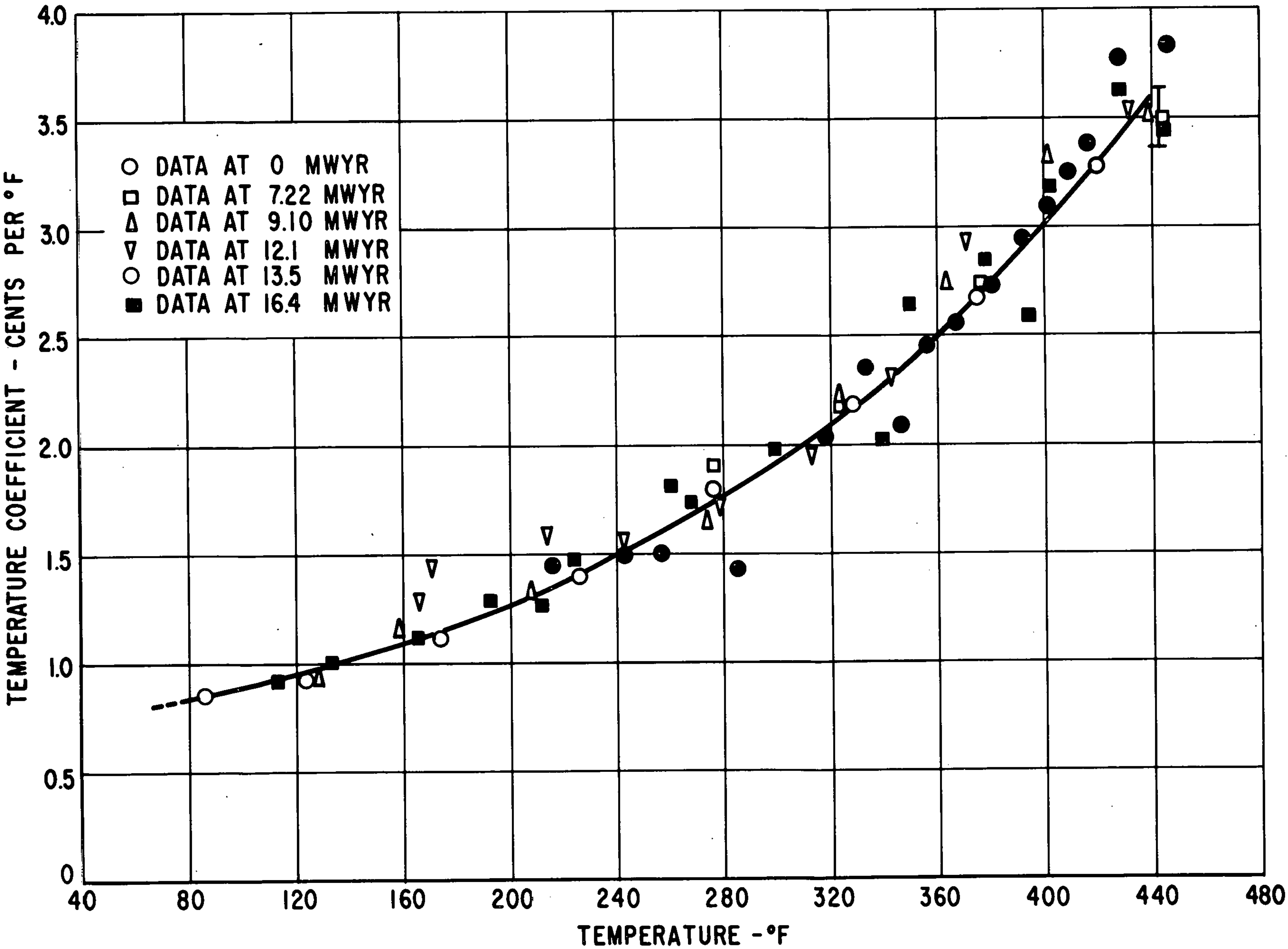

Figure 7.4 - SM-1 Core I Temperature Coefficient vs. Temperature

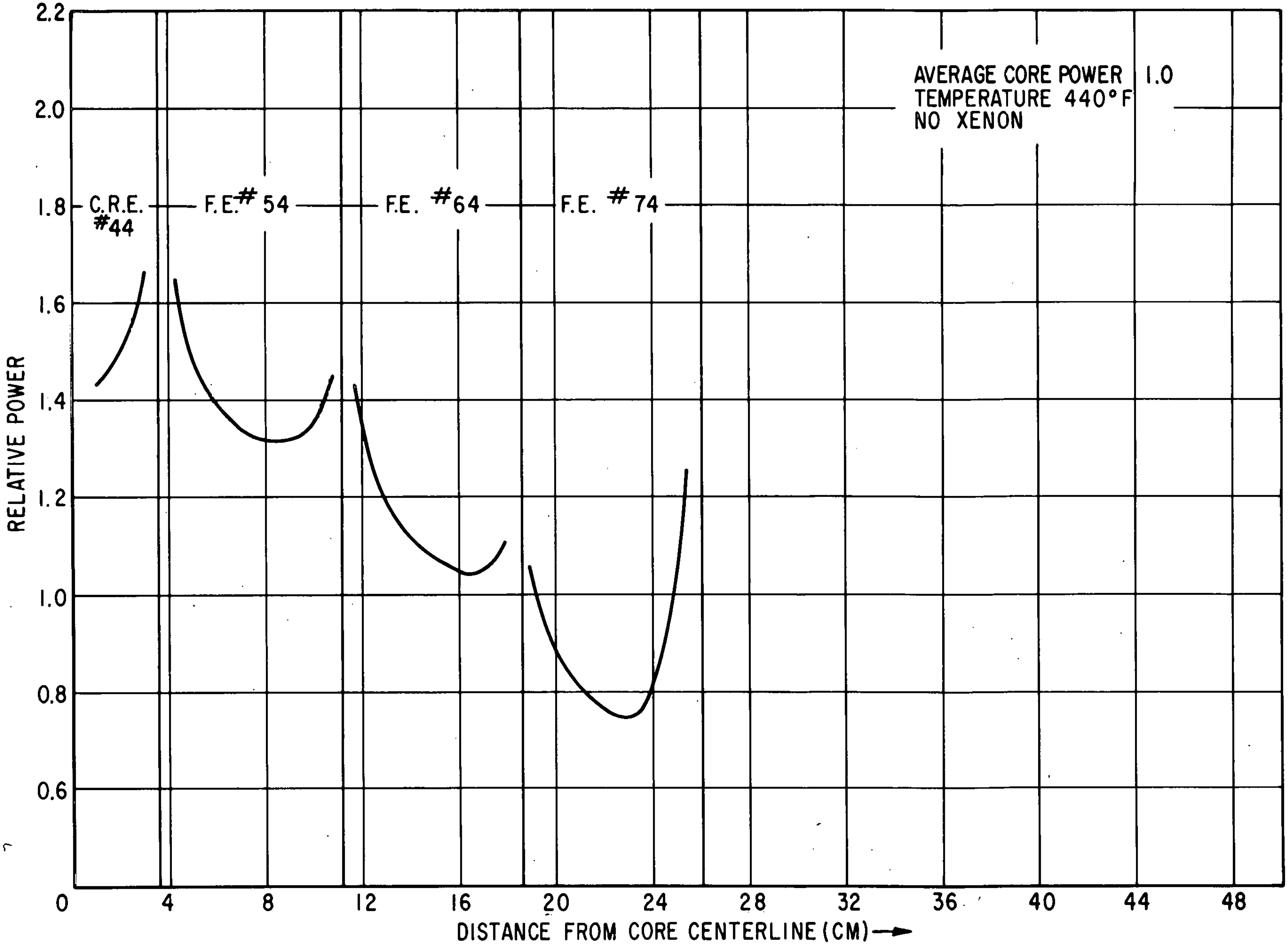

FIGURE 7.5 PL-3 CORE POWER DISTRIBUTION (CENTRAL X,Y PLANE PARALLEL TO FUEL PLATES)

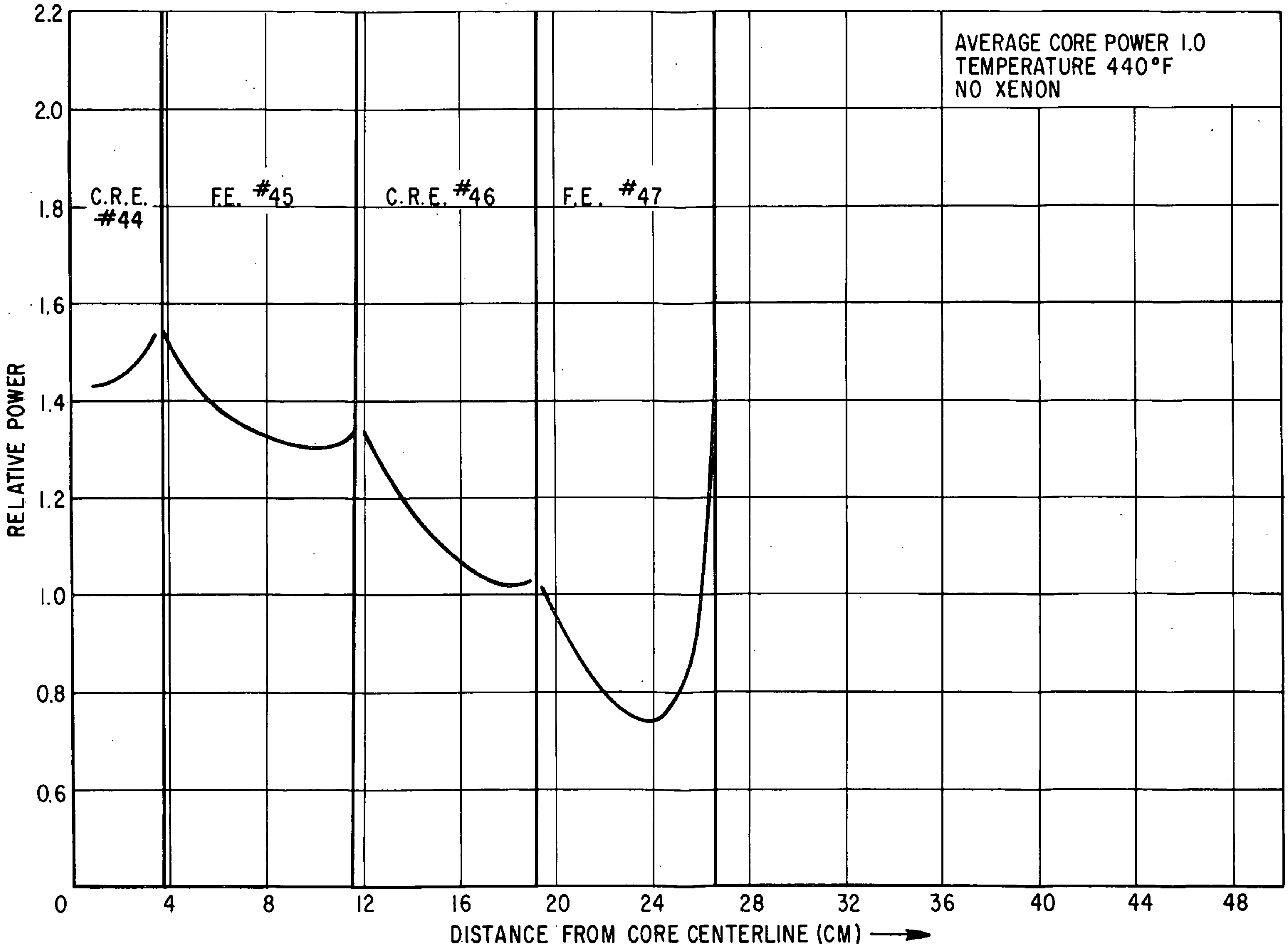

FIGURE 7.6 PL-3 CORE POWER DISTRIBUTION (CENTRAL X,Y PLANE PERPENDICULAR TO FUEL PLATES)

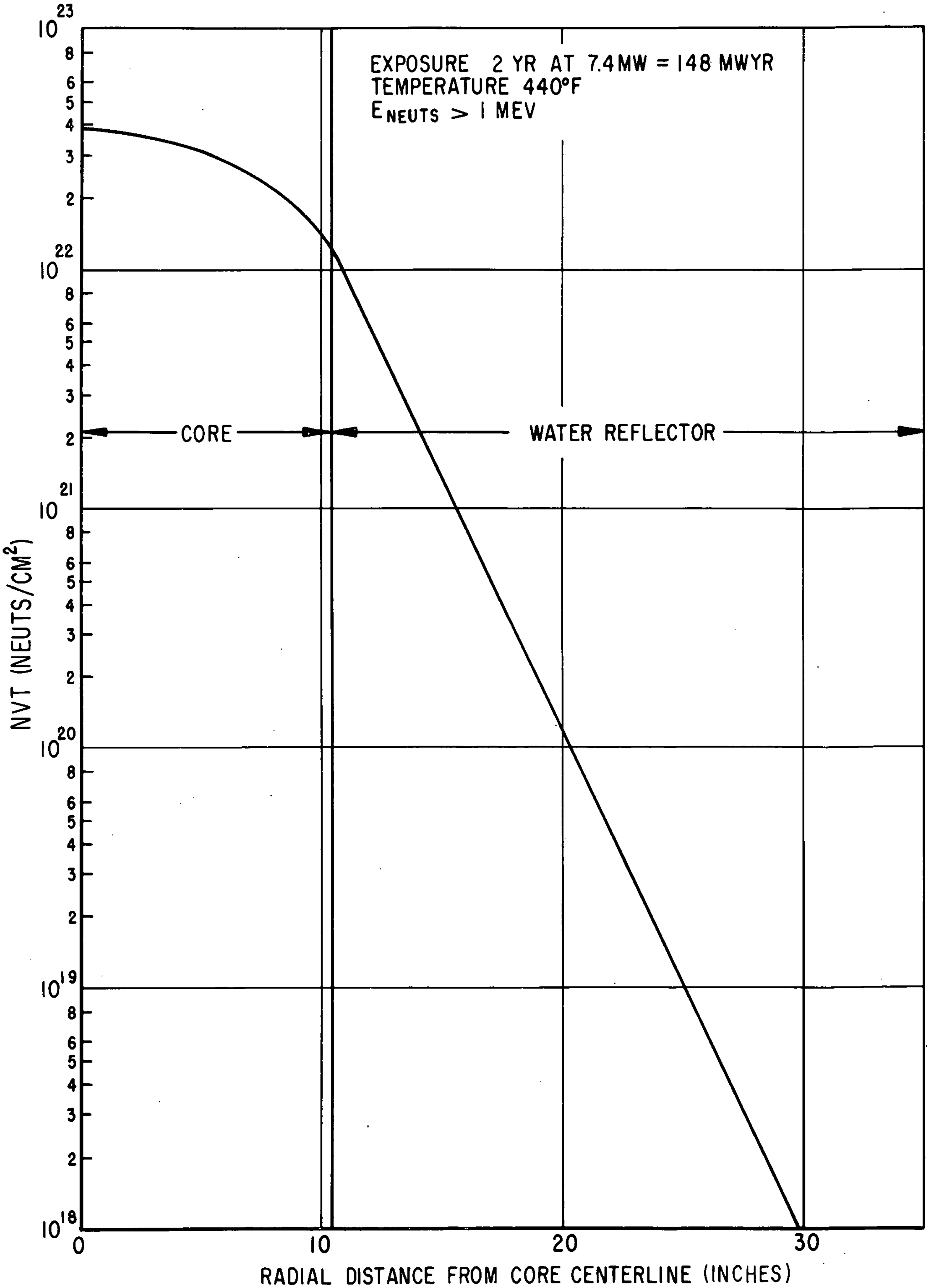

FIGURE 7.7 PL-3 CORE INTEGRATED FAST NEUTRON FLUX RADIAL DISTRIBUTION

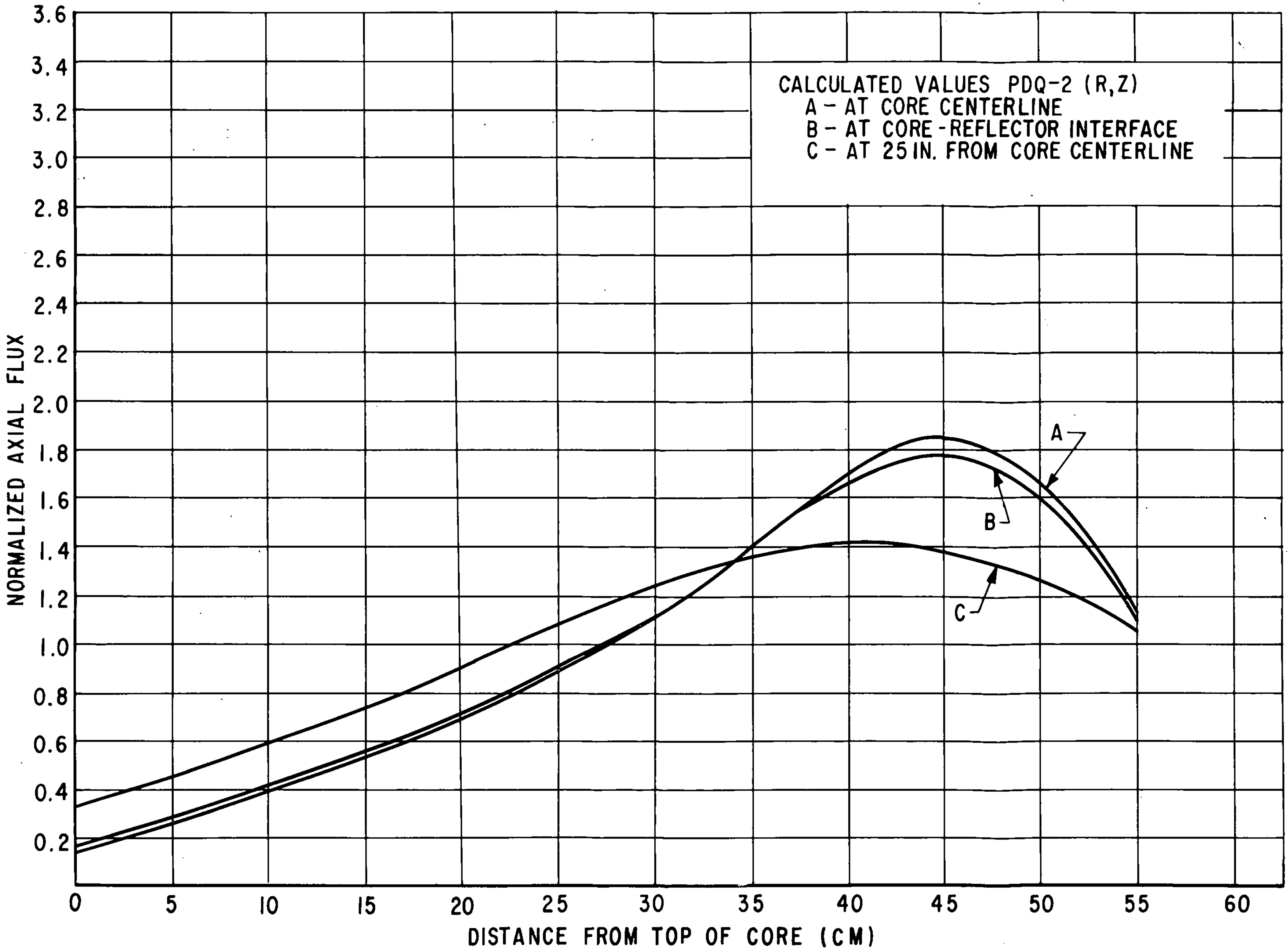

FIGURE 7.8 PL-3 CORE FAST NEUTRON FLUX AXIAL DISTRIBUTIONS

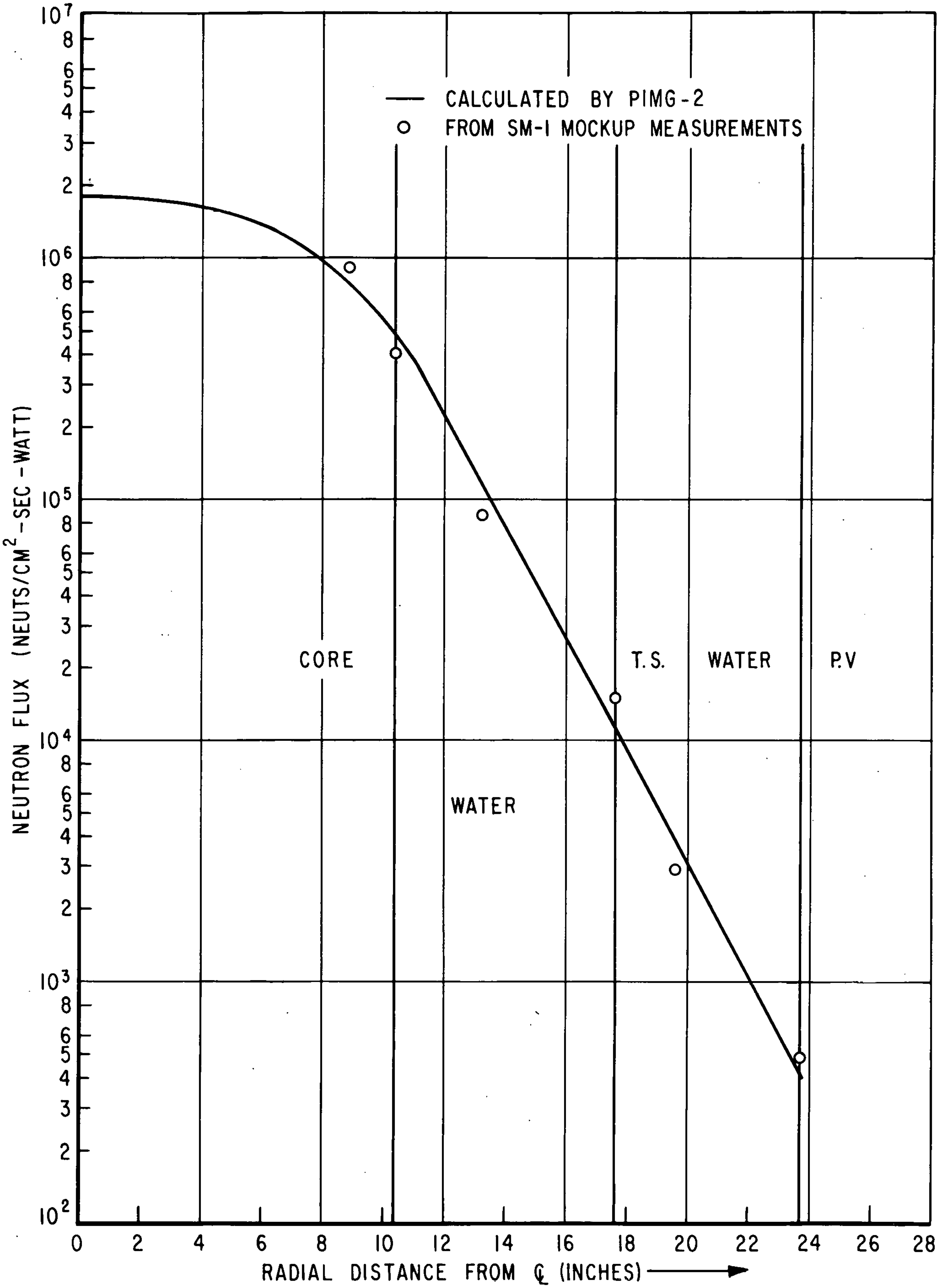

FIGURE 7.9 PL-3 CORE FAST NEUTRON FLUX RADIAL DISTRIBUTION

8.0 CONTROL AND INSTRUMENTATION

8.1 PLANT SYSTEMS CONTROL

The inherent load-following characteristics of the PWR can be relied upon during normal plant operation. Seven control rods are provided for safety, start-up and shutdown, and for the shim adjustments of reactivity required to compensate for the long term effects associated with fuel burnup and fission product poisoning. The reactor control rod drives may be operated individually or in groups by selector and control switches on the control console.

The pressurizer maintains primary system pressure by control of the fluid temperature within the pressurizer, using electric heater units. These heaters may be operated manually, but normally an automatic control will regulate the system pressure within a narrow range about 1200 psia, energizing the heaters on low pressure and shutting them off on high pressure. Load changes, reflected in the expansion and contraction of the volume of primary system water, are absorbed in the pressurizer. The pressurizer level is controlled by rate-regulation of the primary system blowdown. Makeup rate is set by water chemistry requirements.

An automatic feedwater control system, utilizing a three-element controller, will correlate feedwater flow rate, steam flow rate, and steam generator water level to actuate a feedwater control valve and maintain proper water level in the steam generator.

Secondary system blowdown is through a closed purification system to the condenser hotwell. Only minor losses of secondary water inventory are anticipated; manual control of makeup will be adequate to maintain the level in the condenser hotwell within acceptable limits.

Conventional controls for the turbine-generator unit will be provided.

An instrumented flow diagram, AEL-750, is provided for reference.

8.2 NUCLEAR INSTRUMENTATION

The nuclear instrumentation, as shown in the block diagram Dwg. AEL-730, indicates the nuclear behavior of the plant and maintains safe limits of reactor operation. It accomplishes this by measuring and indicating neutron flux from reactor startup to approximately 150% full power. The nuclear instrumentation essentially covers 10 to 11 decades of neutron flux. These decades are divided into three ranges: 1) the source range, 2) the intermediate range, and 3) the power range.

From the neutron flux measurements, and the rate of change of neutron flux measurements, control signals are derived to perform the required reactor operations and to maintain safe normal conditions. Duplicate nuclear instrumentation channels are used for safety and reliability. Two out of three coincidence circuitry is used in the power range to ensure that a false signal from one power range channel will not cause a scram.

The nuclear instrumentation system has seven analog channels; two for source range, two for intermediate range, and three for the power range. The instrumentation system monitors reactor neutron flux and computes reactor period from source level (2.5×10^{-1}nv) to 150% of rated reactor power (2.5×10^{10}nv).

8.2.1 Source Range Channels

The two source range channels monitor neutron flux and compute reactor period over a range of at least 2.5×10^{-1} to 2.5×10^{4} nv. These channels provide analog signals of reactor neutron flux and reactor period for recording and indicating, and for scram logic circuitry. They furnish signals of 0.5 watts each and indicate: 1) neutron flux as log of count rate over a neutron flux range corresponding to 2.5×10^{-1} to 2.5×10^{4} nv, 2) period from -30 sec to +30 sec, and 3) detector power supply voltage on the front panel. The source range channels also furnish analog signals to the scram logic circuitry for neutron flux level, e.g., rod withdrawal prevention below a minimum count rate.

Each source range channel is supplied with a self-contained test calibrate unit. This unit provides at least three check points equally spaced throughout the range of the channel. The test calibrate unit is used with panel mounted meters to align instrumentation and to verify proper channel operation.

A scaler and mechanical register, capable of counting all pulses within the channel range, are provided. The scaler has a scaling factor adjustable in factors of 10 or less over the instrument range. The mechanical register produces an audible "click" upon registering a count.

8.2.2 Intermediate Range Channel

The two intermediate range channels monitor reactor neutron flux and compute reactor period over a range of at least 2.5×10^{2} nv to 2.5×10^{10} nv, for recording and indicating and for scram logic circuitry. The intermediate range channels furnish signals of 0.5 watt each and indicate 1) neutron flux on a log scale as percent of rated reactor power, 2) period from -30 sec to +30 sec, and 3) detector power supply voltage.

The channels also furnish analog signals to the scram logic circuitry for:

1. Neutron flux for zero power scram.

2. Neutron flux for bypass of source range and for cutting off source range detector voltage.

3. Period for rod withdrawal prevention on a 10 second period.

4. Period for scram on duration period curve.

Each intermediate range channel is supplied with a self-contained test calibrate unit. This unit provides at least three check points throughout the range of the channel and is employed with panel mounted meters to align instrumentation and to verify proper channel operation.

8.2.3 Power Range Channels

The three power range channels monitor neutron flux over a range of 1.25×10^8 nv to 2.5×10^{10} nv. These channels provide analog signals of reactor neutron flux for recording and indicating and for scram logic circuitry. The power channels furnish signals of 0.5 watt each for remote recording of neutron flux and indicate neutron flux on a linear scale and detector power supply voltage on the panel front. Analog signals to the scram logic circuitry are provided for high power scram.

The power range instrumentation is supplied with a self-contained unit to monitor the operation of each channel's circuitry and continuity of the channel, chamber and cable. Maloperation and/or failure is audibly indicated.

Each power range channel is supplied with a self-contained test calibrate unit. This unit provides at least three check points throughout the range of the channel and is employed with panel mounted meters to align instrumentation and to verify proper channel operation.

8.2.4 Logic Circuitry

The scram and trip logic circuitry receives analog signals from the various parameter sensors to perform the proper logic operations satisfying plant safety requirements.

The startup logic circuitry receives analog signals from the startup channels for rod withdrawal permission above a minimum count rate.

The intermediate range logic circuitry receives analog signals from the intermediate range channels to:

1. Bypass low level protection from startup channels.

2. Cut off BF_3 detector high voltage and insert a shorting resistor across BF_3 terminals.

3. Scram on zero power.

4. Stop rod withdrawal on short period.

5. Scram on duration period curve.

The power range logic circuitry receives analog signals from the power range channels for high power level reactor scram.

Process scram logic circuitry receives analog signals to:

1. Scram on primary system high pressure.

2. Scram on primary system low pressure.

3. Scram on primary system high temperature.

4. Scram on primary system low flow.

5. Scram if an attempt is made to start up in the off or zero positions of the rod transfer lock switch.

6. Scram manually.

7. Prevent rod withdrawal when primary coolant pump is not operating.

The process inhibit circuitry allows the removal of one process channel to permit emergency repairs and testing without causing a scram, and the test inhibit circuits allow testing of one instrumentation channel at a time without causing false scrams.

8.2.5 Shielding of Detectors

To accomplish a reactor startup within 8 min after a scram during high power operation, the effects of high gamma radiation in comparison to low neutron flux must be overcome. A solution is provided by shielding the detector tubes against gamma radiation but not to such an extent that neutron sensitivity is greatly affected. Lead is used as a gamma shield and paraffin is inserted between the lead and the detector to thermalize fast neutrons and increase neutron sensitivity during high backgrounds.

8.3 CONTROL CONSOLE

The majority of all plant instrumentation and control functions are located at the control console on the control skid. The control console utilizes a "U" shaped geometry with the top panel faces inclined toward the operator. A 45°

sloping panel is used for control levers and switches; all controls are located in proper relation to the display they affect. The control console includes the following pertinent design features:

1. One man control is assured because all controls necessary during plant operation are located on the console.

2. Only instruments essential to operator surveillance are located on the front of the console.

3. Where practical meters will utilize rotating bases so that for normal indication a vertical position reading occurs.

4. Nuclear, primary, secondary and electrical displays are grouped from left to right respectively around the control console vertical panels.

5. Instrumentation and controls are arranged so that optimum viewing distance is obtained.

The nuclear instrumentation and control display includes meters, indicator lights and control switches affecting the plant nuclear characteristics. Included are meters for source range, intermediate range and power range nuclear indications and the necessary parameter selector switches; instrumentation for rod height indication and switches for rod drive motor control; and indicators for nuclear parameter scrams. Primary instrumentation and control panels include meters, indicator lights and controls affecting the plant primary system. Included are pressure and/or temperature indications for the reactor vessel, the pressurizer, the primary coolant pump and the primary side of the steam generator. For the pressurizer, level indicators and heater indicators are provided. Control and protective devices are installed for these parameters. Indications are given for high or low pressure, high temperature, low flow and manual scram conditions. Also included as part of the primary display and control panel are meters and controls which indicate and control parameters within the primary vapor container and shield tanks. These include temperatures, water levels, flows, pressures, valve positions and so forth. The secondary system display and control panels include instrumentation for steam temperature, steam flow, steam pressure, steam generator water level, makeup rate, valve positions, turbine speed, glycol system temperature and flow, auxiliary cooling system parameters, lube oil system parameters, and auxiliary steam system parameters. Controls for these parameters are provided where applicable. Electrical instrumentation and control panels include instrumentation measuring generator voltage and current and exciter voltage and current. Controls for electrical system circuit breakers, load transfer equipment, and generator loading are provided where applicable.

9.0 SHIELDING AND RADIATION LEVELS

9.1 OPERATING RADIATION LIMITATIONS

9.1.1 Operating Neutron Flux Limitations

The technical requirements for PL-3 limit the neutron flux at one mile from the PL-3 to one neutron per square meter per minute. No experiments or scientific observations with which higher levels of neutron fluxes might interfere are at this time in evidence.

The initial approach taken to interpretation of this criterion was to integrate the fast neutron leakage from the primary shield and utilize bomb blast data to evaluate its transmission to one mile.(1) Neutron leakage through the top of the primary shield may be neglected since the water cover required for fuel handling is considerably greater than that required on the side of the primary shield for neutron shielding. For preliminary calculation, neutron leakage through the bottom of the primary shield was neglected. The analytical model is thus reduced to an infinite right circular cylinder surrounding the reactor core.

The neutron leakage through the side of this cylinder approximates a cosine distribution of the form:

$$\phi = \frac{\phi_o}{2} \left[1 + \cos \frac{\pi}{2} \left(\frac{x}{x_1} \right) \right]$$

Where:

ϕ_o = fast neutron leakage flux in reactor midplane (n/cm^2-sec)

x = vertical separation from reactor midplane (cm)

x_1 = $\sqrt{\left(\frac{\ln 2}{\Sigma}\right)^2 + \frac{2 (\ln 2) R_o}{\Sigma}} = \sqrt{47.1 + 13.7 R_o}$

R_o = primary shield radius (cm), and

Σ = effective cost neutron removal cross section for water ($0.101\ cm^{-1}$).

The total leakage is found from the following integral:

$$\phi \text{ total} = 2\pi R_o \phi_o \int_0^{2x_1} \left[1 + \text{Cos} \frac{\pi}{2} \left(\frac{x}{x_1} \right) \right] dx$$

$$= 2\pi R_o \phi_o \quad 2x_1 \quad = \quad 4 \quad R_o \quad \phi_o \sqrt{47.1 + 13.7 R_o} \text{ neutrons/sec}$$

In an infinite air medium, the flux at one mile, ϕ_1, is

$$\phi_1 = \frac{\phi_{total}}{4\pi D^2} \quad \exp\left(- \frac{D}{2.21 \times 10^2} \right) \quad n/m^2\text{-sec}$$

Where:

2.21×10^2 meters = fast neutron relaxation length in air (bomb blast data)[1]

$D = 1 \text{ mile} = 1.61 \times 10^3$ meters

$$\phi_1 = 1.6 \times 10^{-8} \; R_o \phi_o \sqrt{47.1 + 13.7 R_o} \quad n/m^2\text{-min}$$

From this relationship, a plot of neutron flux at one mile vs. primary shield radius has been prepared, Fig. 9.1. From this figure it can be seen that a 7.5 ft radius shield will give 40% of the limiting neutron flux from shield penetration. Neutron streaming through radial streaming paths and shield penetrations can then be allowed to contribute the remaining 60% of the total primary shield neutron leakage.

It was assumed that 10^4 neutrons/cm^2-sec are equivalent to 1 Rem/hr and account was taken in the calculations above, of the reduced water density in the reactor vessel. The fast neutron dose rate was taken from the BSR pure water data.[2] The corresponding thermal neutron flux is about 150% of the fast flux. The thermal neutrons leaking from the shield surface are of no importance in this calculation since they are captured long before they reach one mile. However, they are contributors to dose in the camp due to their interactions in air.

The following conservative and non-conservative factors should be explored in subsequent phases of design where the need for refinement of the calculations is evidenced:

1. Losses due to scatterings in the tunnel - It is estimated that only about 25% of the neutrons leaking from the primary shield will escape from the tunnel. No tunnel losses were considered in the above model; the assumption used was conservative.

2. Increased shielding effectiveness of steel - Steel is as effective as about 1.6 times as much water for shielding fast neutrons provided sufficient hydrogenous material follows it. The magnitude of this effect will depend upon the amount of steel in the final configuration; steel effectiveness was conservatively assumed equal to that of water in the above model.

3. Snow scattering outside the reactor tunnel - The bomb blast radiation data are applicable for an infinite air medium. Consideration of the snow as a second scattering medium would probably account for about a factor of two in greater neutron flux at one mile. The assumption used was non-conservative.

4. Effect of annular voids within the primary shield - The insulation void around the reactor vessel and other non-radial voids within the primary shield do not contribute to the effectiveness of the shielding. No account was made of these voids in the above model; a non-conservative assumption.

5. Dose-flux conversion factor - The factor used corresponds approximately to 0.8 Mev. An evaluation based upon an analysis of the neutron spectrum should be done.

6. Effects of shield penetrations and radial voids in the primary shielding - These will have to be considered on an individual basis and shielded to permit leakage compatible with direct penetration. An effort should be made to achieve optimum distribution between neutrons streaming through shield penetrations and those neutrons penetrating directly through the primary shielding.

9.1.2 Operating Gamma Dose Limitations

Operating gamma dose rates are restricted by design requirements of the PL-3 contract. Limitations imposed are:

1. 0.06 mrem/hr outside power plant controlled area.

2. Integrated exposure of 1.25 rem/quarter for each man, based upon 84 hr work week.

3. No detectable radiation in camp living quarters.

An empirical relationship for gamma dose from air reactions in this type geometry was obtained from FZK-122.[3]

$$D_{100} = 3.5 \times 10^{-9}\, D_{sg} + 5.2 \times 10^{-10}\, D_{sn} + 2.3 \times 10^{-11}\, \phi_{sn}$$

where,

D_{100}	=	air scatter gamma dose-rate at 100 ft (mrem/hr)
D_{sg}	=	integrated gamma leakage on snow surface over the reactor tunnel (mrem-cm^2/hr.)
D_{sn}	=	integrated fast neutron leakage on snow surface over the reactor tunnel (mrem-cm^2/hr.)
ϕ_{sn}	=	integrated thermal neutron flux on snow surface over the reactor tunnel (n/sec).

From Section 9.1.1, D_{sn} and ϕ_{sn} were found to be 3.2×10^7 and 4.8×10^8 respectively. The integrated operating gamma leakage from the primary shield is about 1.0×10^{10} mrem-cm^2/hr if it is designed to meet shutdown limits on the shield surface. All gammas leaking from the sides of the primary shield must undergo at least one scattering of 90^o or more to escape through the top of the tunnel. This will reduce the gamma dose by 0.04, due to the scattering, to give an integrated gamma dose of 2.5×10^8 mrem-cm^2/hr. Only about 25% will escape from the tunnel on one scattering. Using these figures in the above formulation gives an air scatter gamma dose rate at 100 ft of 0.25 mrem/hr on the snow surface at the distance of the lateral tunnel. Similar calculations were performed for 500 and 1000 ft separation and the results are plotted in Fig. 9.2. Attentuation of neutrons and gamma radiation in the reactor tunnel roof are ommitted in this calculattion as are neutron losses due to scattering within the tunnel.

This calculation is sufficient to demonstrate that the air scattered dose is within controllable limits. Detailed calculations performed after the reactor shielding, tunnel layouts, and plant buildings have advanced in their designs would indicate any additional shielding which might be required either in the form of snow cover over the camp tunnels or in the form of additional shielding on the reactor. The present calculations indicate a requirement for a controlled area on the snow surface for about 300 ft surrounding the reactor.

Factors in this calculation which need additional study are:

1. Coefficients in the empirical air scatter formulation - The coefficient of $\emptyset_{sn}$ has some unusual characteristics which lead to questioning of some of the values given in FZK-122.

2. Thermal neutron capture by steel. - Thermal neutron captures in the outer steel shield tank wall and in the tunnel arches were neglected in this calculation.

3. Thermalization of fast neutrons by tunnel wall scattering. - If the fast neutrons, which are responsible for the gamma dose from inelastic scattering of neutrons in air, are thermalized before escaping into the air, they should be included in $\emptyset_{sn}$. They then contribute to the air capture gamma dose.

4. Neutron escape from the snow tunnels - The same neutron fluxes obtained in the previous section were used; hence, the same uncertainties exist with respect to the flux values as in the previous section.

5. Gamma scattering from the reactor tunnel - A more detailed study of the gamma scattering from the reactor tunnel must be made as the design of the primary shielding, reactor building, and tunnel configurations progresses. This study should consider energy degradation on scattering, gamma leakage distribution from the primary shield, attenuation by the building walls and tunnel arches, and contributions from multiple scatterings.

9.2 SHUTDOWN RADIATION REQUIREMENTS

The primary reactor shield leakage after shutdown is limited by the technical criteria. This requirement is interpreted to limit radiation leakage from the primary shield to 100 mr/hr at 2 hrs after shutdown from an extended full power operation. Since ROC[4] code library decks were available for 2.4 hrs shutdown time, and since a dose reduction of about 20% occurs between 2.0 and 2.4 hrs shutdown time, an equivilant preliminary design target was taken as 80 mr/hr at 2.4 hrs after shutdown.

The primary shield optimization study indicates that about 3 in. of lead shielding must be added within the 7.5 ft radius primary shield to meet this criterion. Fig. 9.3 gives the gamma dose rate at 2.4 hrs after shutdown outside the primary shield as a function of lead shielding thickness. The machine calculations have indicated that about 37% more weight of steel is required to replace the lead for gamma shielding. This and other factors contributed to the selection of lead for the PL-3 gamma shielding material.

The total operating gamma dose for the shield which resulted from the shutdown criterion is about 12 r/hr. The shutdown dose is predominantly from the reactor core, while the operating dose is predominantly due to hydrogen captures in the shield.

9.3 CONSEQUENCES OF FUEL ELEMENT FAILURE

One of the basic design criteria for PL-3 has been that the reactor tunnel is not accessible during operation. Therefore, the fission products release by a fuel element failure are shielded so long as they are contained within the primary system in the reactor tunnel. The primary purification system is located outside the reactor tunnel and would be a large radiation source in the event it is used to clean up a substantial fission product release. For example, if a release in the amount of 1% of the fission products occurs, about 10 in. of lead shielding would be required on the demineralizer. Shielding would also be required on blowdown piping, valves, etc.

Study of the fuel integrity for the PL-3 reference fuel has indicated no mechanism which can be credibly postulated to account for release of as much as 10^{-3}% of the end-of-life fission products. In conclusion, it appears that the purification system should be designed to accommodate routine fission product release of about 10^{-4}% of the end-of-life inventory. Care should be taken in the design that releases as high as 10^{-2}% do not represent an untenable hazard or render any function of the plant inoperative without chance of recovery.

9.4 REFERENCES

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3. Wheeler, D. M., Bostick, L. M., "Military Field Expedient Shielding Experiment," General Dynamics (Convair Div.) FZK-122, October, 1960.

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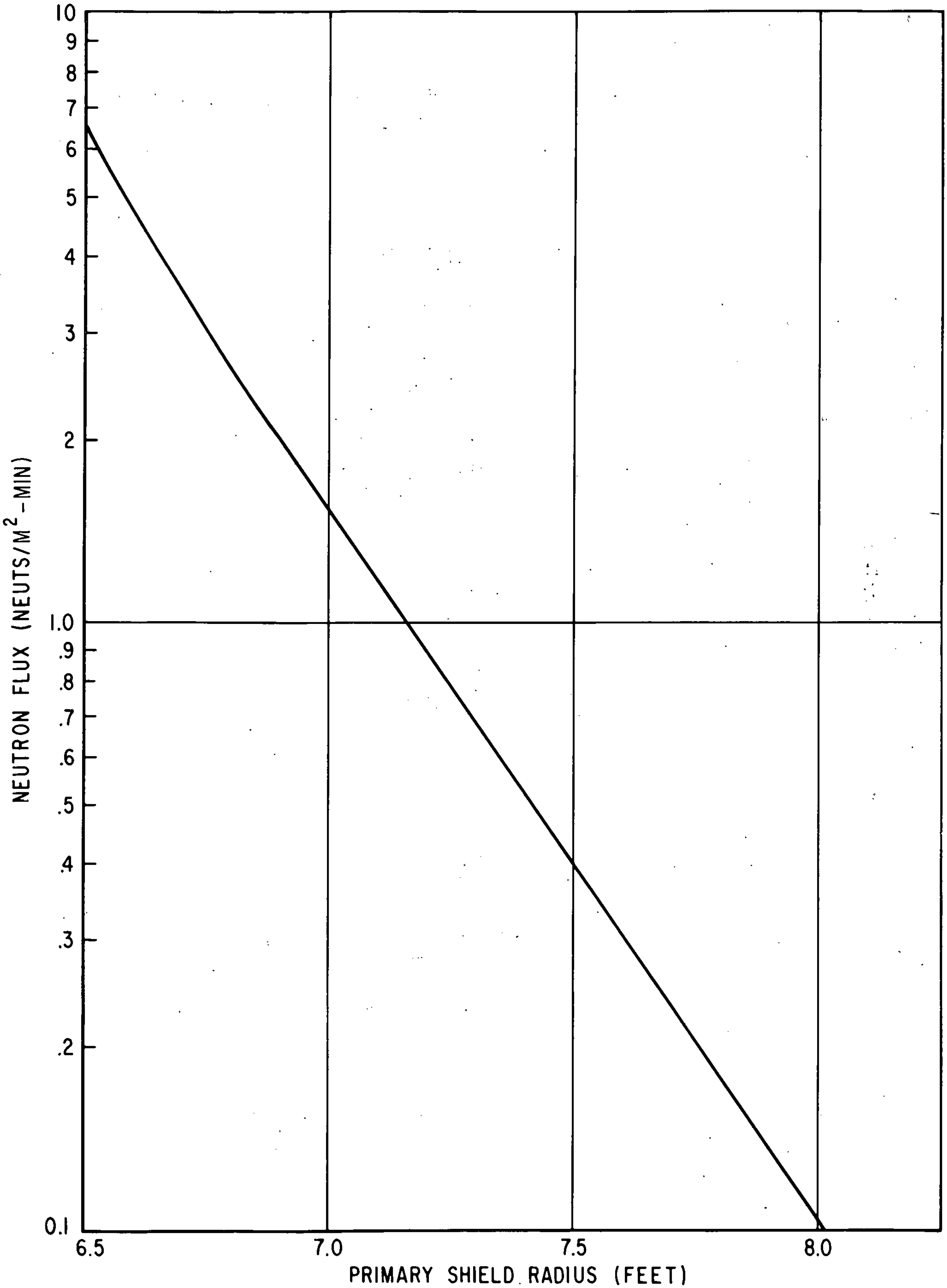


FIGURE 9.1 NEUTRON FLUX AT ONE MILE FROM DIRECT SHIELD PENETRATION NEUTRONS VS. PRIMARY SHIELD RADIUS

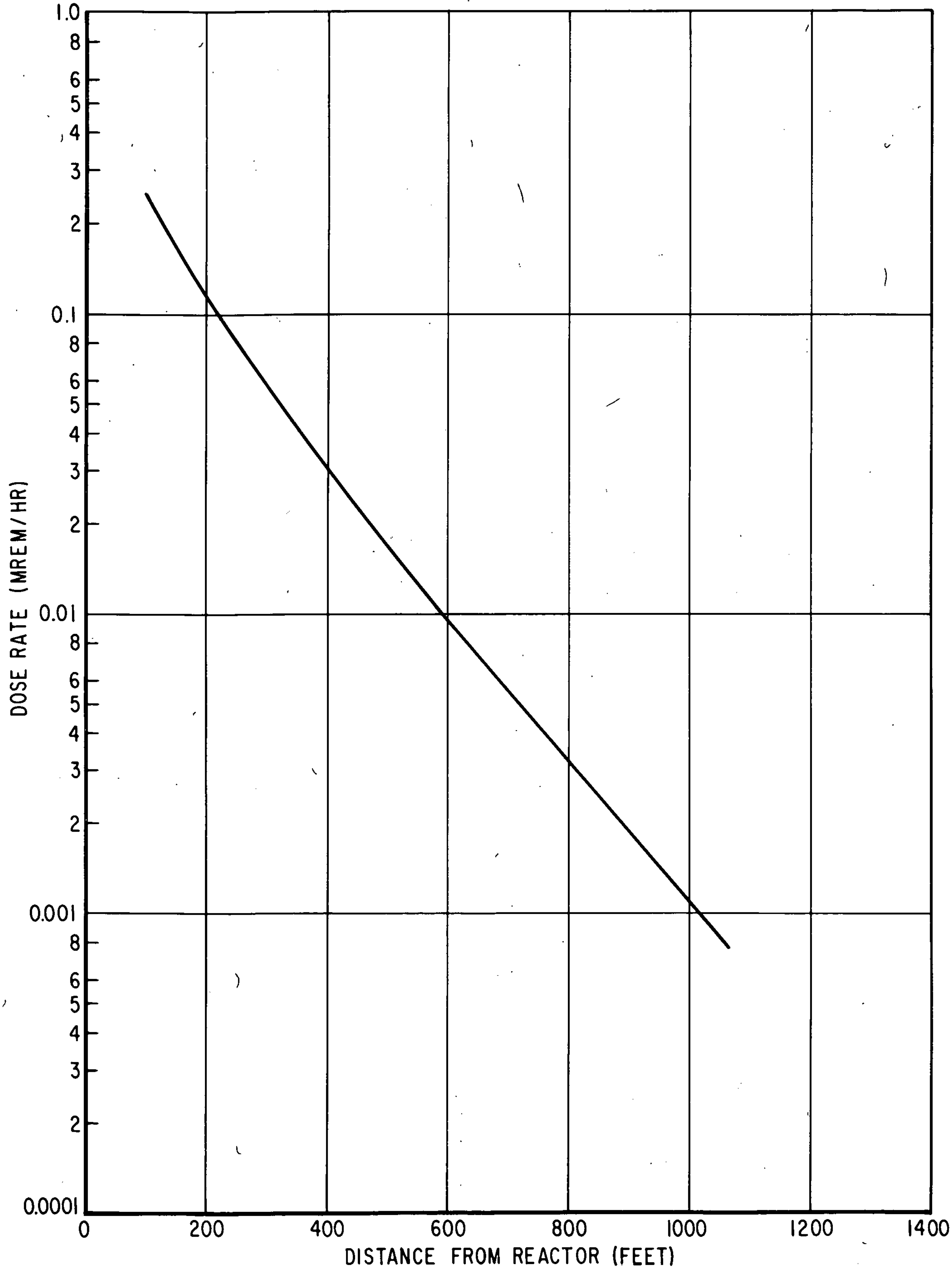


FIGURE 9.2 AIR SCATTER GAMMA DOSE RATE ON SNOW SURFACE

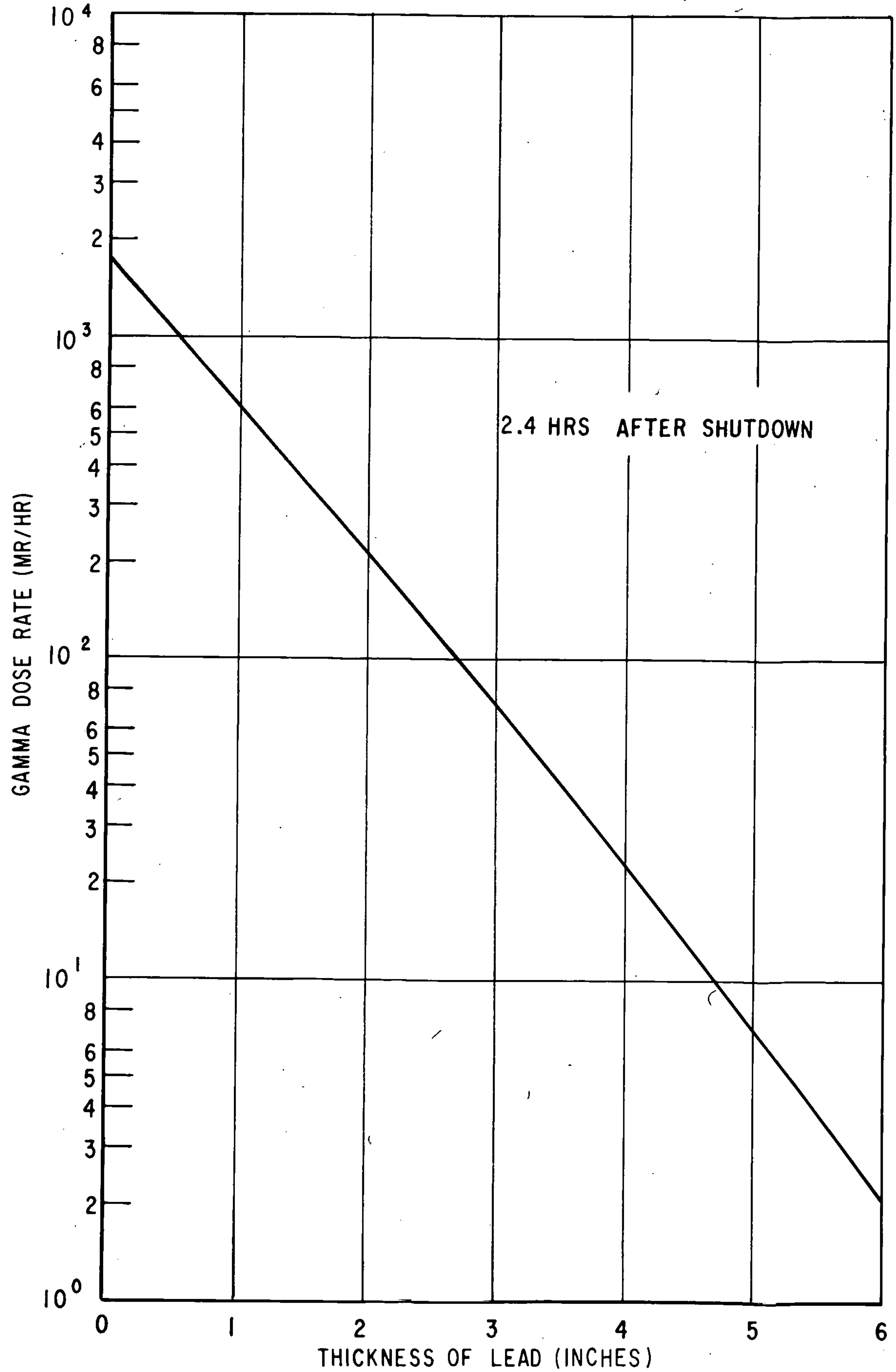


FIGURE 9.3 SHUTDOWN DOSE RATE VS. LEAD THICKNESS

10.0 PRESSURIZED WATER REACTOR AUXILIARY SYSTEMS

10.1 PRIMARY COOLANT PURIFICATION SYSTEM

A low pressure, 25 psig system has been selected for PL-3 application in preference to a high pressure (reactor pressure) system. The low pressure system has several advantages.

1. Most of the components are designed for a 125 psig design, pressure.

2. Maintenance and replacement of components is less hazardous then with high pressure systems.

3. More latitude is allowed in placement of purification system components.

The only significant disadvantage is the requirement for continuous operation of a high pressure pump. If high pressure sealing water is required for primary system components, this disadvantage virtually disappears.

Clean-up of the primary system or reactor water is accomplished by continually removing a small flow from the reactor, purifying it, and then returning it to the reactor system. This flow, or "blowdown," is on the order of 0.5 gpm. It is basically a high purity water contaminated with activity and corrosion products. A two step cation mixed bed arrangement has been selected for PL-3. The small cation bed, preceding the mixed bed, serves several functions.

1. The major portion of the activity is either cationic or particulate and will be retained on the cation bed either by filtration or ion exchange.

2. The volume of high activity is concentrated providing a reduction in shielding requirements.

3. A cation mixed bed combination obtains a greater removal of activity than could be accomplished with either a cation or mixed bed used alone.

One possibility that must be considered is the use of pH control. This could effect not only ion exchange resin selection, but the overall purification system design. However, pH control is not contemplated at the present time.

Operation of the system as shown in AEL 710 is completely automatic. Water is removed from the reactor, cooled below 120°F and then reduced in pressure to 25 psig. This pressure is sufficient to provide flow through the ion exchangers to a makeup storage tank. The tank provides a supply of high purity water for the primary system and is sized to hold the water removed from the primary system due to expansion during heatup of the reactor. It also represents an adequate source of high purity seal water. Water is returned from the tank to the reactor by a positive displacement pump.

An automatic pressure control valve maintains downstream pressure at 25 psig. Safety features include:

1. A temperature control valve set to close if the blowdown exceeds $130^{\circ}F$.

2. A pressure relief valve to dump the blowdown to one of the pools if the pressure at the ion exchanger inlet exceeds to 100 psig.

3. High and low pressure alarms.

4. A blowdown isolation valve upstream of the cooler to close at 75 psig.

5. High and low flow alarms.

From the makeup tank, a pump raises the pressure to 50 psi above primary system pressure. The major portion of the return flow passes through a back pressure valve and returns to the primary system. Sufficient flow is diverted to the control rod drive seals. The seal outleakage may return directly to the makeup tank or may go to an intermediate seal drain tank.

Blowdown flow control, which is from the pressurizer level indicator, adjusts to maintain a constant level in the pressurizer. There are several advantages of this control method:

1. Expansion and contraction of primary coolant due to temperature change is automatically compensated.

2. If makeup is lost, blowdown stops to prevent the loss of primary coolant.

Since the primary system of a PWR is a closed cycle, chemical treatment is feasible. Hydrogen is added to suppress the decomposition of water and remove oxygen. One other possible means of control is the addition of lithium hydroxide to raise the pH.

10.2 DECAY HEAT REMOVAL

Decay heat will be removed from the reactor by allowing water to boil in the secondary side of the steam generator. The vertical steam generator establishes natural circulation in the primary system. The primary circulation pump is not required to be in operation for decay heat removal.

To compensate for steam removed from the steam generator, a small make-up water flow must be provided. Two methods are being investigated to accomplish this. The first would be to discharge the steam to the main condenser and return condensate to the steam generator. The second is to provide a small condenser wherein the steam is condensed and flows back to the steam generator by gravity. This arrangement would provide a closed cycle and be advantageous in the event of a generator tube rupture. The small condenser would be cooled with either shield water or air.

The temperature of the reactor and cooldown rate is controlled by regulating the boiling rate of the secondary water. Figure 10.1 shows the shutdown heat removal requirement for cool down and decay heat removal. As can be seen, this requirement drops off rapidly. A boiling rate of 500 lb/hr would be more than satisfactory.

10.3 REFUELING AND FUEL TRANSFER SYSTEM

10.3.1 Refueling Objectives

The pressurized water reactor must be refueled at regular scheduled intervals. This change should be accomplished as quickly as possible, with operating personnel adequately protected against radiation exposure. A refueling system for Byrd Station should incorporate the following features:

1. Adequate protection of personnel from radiation.
2. Functional and structural integrity of apparatus subjected to adverse climatic conditions.
3. Minimum number of units to adequately meet requirements.
4. Ease of operation with a minimum of personnel.
5. Minimum time to accomplish entire operation.
6. Minimum weight consistent with reliability.
7. Simplicity of design to minimize cost and maintenance.

10.3.2 Removal of Fuel From Reactor

The optimum transfer system utilizes the fuel transfer chute method. A typical application of this system is shown in Drawing AEL-762; a conceptual chute design is shown in Fig. 10.2. The transfer chute is simple and economical, requiring little or no maintenance. In operation, it is a direct and rapid method of transferring fuel and requires only a small operating crew. An overhead crane is required only for the reactor vessel and vapor container disassembly and reassembly. The transfer chute is welded directly to the vapor container and containment integrity is maintained by use of a seal plug in the chute. The seal plug seats against a tapered shoulder on the inside of the chute, and a flexible seal is expanded against the tube wall.

The following procedure is followed when transferring spent fuel elements after preparation of the transfer tools:

1. Loosen, remove and store vapor container cover nuts.

2. Remove and store vapor container cover.

3. Fill dry cap (if used), unbolt, remove and store.

4. Loosen, remove and store reactor cover nuts.

5. Install stud caps.

6. Remove and store reactor vessel cover.

7. Remove and store core cover.

8. Remove and store control rod caps.

9. Loosen, remove and store fuel chute plug or plugs.

10. Position auxiliary shield on top of cask.

11. Using hand tool, select element, lift from core, place in chute and lower into hopper.

12. Disengage transfer tool from element and remove from chute.

13. Using fuel handling tool, unlatch hopper and rotate to vertical position.

14. Lift element from hopper and place in basket of shipping cask.

15. Rotate hopper into alignment with chute and latch.

16. Repeat steps 11 through 15 until all fuel is transferred. The auxiliary shield must be moved from cask to cask as one cask is filled and another is prepared for loading.

17. Load new core in reactor using transfer tools.

18. Reassemble reactor by reversing steps 1 through 9.

10.3.3 Storage of Spent Fuel in Peripheral Shield Tanks

The transferred fuel will be stored in shipping casks placed around the vapor container in the shield tanks. The fuel will be kept in the shield tanks for not less than 90 days after removal from the reactor. During this decay period, the cask will be cooled by natural convection. The lid of the cask will be supported on a pedestal above the cask, allowing circulation of the shield tank water and reducing radiation scattering.

10.3.4 Shipping Casks For Spent Fuel Elements

The shipping casks for the spent fuel elements of the PL-3 reactor will be designed in accordance with AEC, ICC, and ICPP regulations regarding containers for shipment of radioactive materials. Preliminary shielding calculations indicate that little advantage is gained in shielding thickness by allowing cool-down for one year rather than ninety days. The concepts are, therefore, based on radiation levels ninety days after shutdown of the reactor.

The reference core for the PL-3 reactor is a 7 x 7 Type 2 core. This core consists of 45 fuel elements, 7 of which are control rod followers with absorbers. Shielding calculations indicate that the entire core can be shipped in 2 casks, based on a maximum allowable gross shipping weight of 27,500 pounds, and in 4 casks, based on a maximum allowable gross shipping weight of 20,000 pounds. A tabulation of this shipping cask data is presented in Table 10.1. All shipping casks will be skid mounted for shipment. A conceptual design of this skid is presented in Fig. 10.3. This skid will be sized to fit the selected cask and will be constructed of T-1 steel with appropriate tiedown between cask and skid.

10.4 SOLUBLE POISON SYSTEM

Injection of a soluble neutron absorber will be accomplished by providing a small pressure tank containing boric acid crystals through which the make-up pump discharge can be diverted. This water flow will dissolve the boric acid and carry it into the reactor to accomplish the required shut-down.

TABLE 10.1
PL-3 SHIPPING CASK DATA SHEET
PWR - 7 x 7 Type 2 Core (Reference Core)
38 Stationary Elements; 7 Control Rod Followers; 7 Absorbers

Cask Design	1/2 Core	1/4 Core
No. of Casks per core	2	4
Elements per cask	22 or 23	11 or 12
Absorbers per cask	4 or 3	2 or 1
Element length, in.	31.625	31.625
Absorber length, in.	23.0	23.0
Lead Equivalent, in.	10.85	10.0
Bare Weight per Cask, lb.	25,350	18,650
Loaded Weight per Cask, lb.	27,050	20,150
Basket Opening, in. sq.	3	3

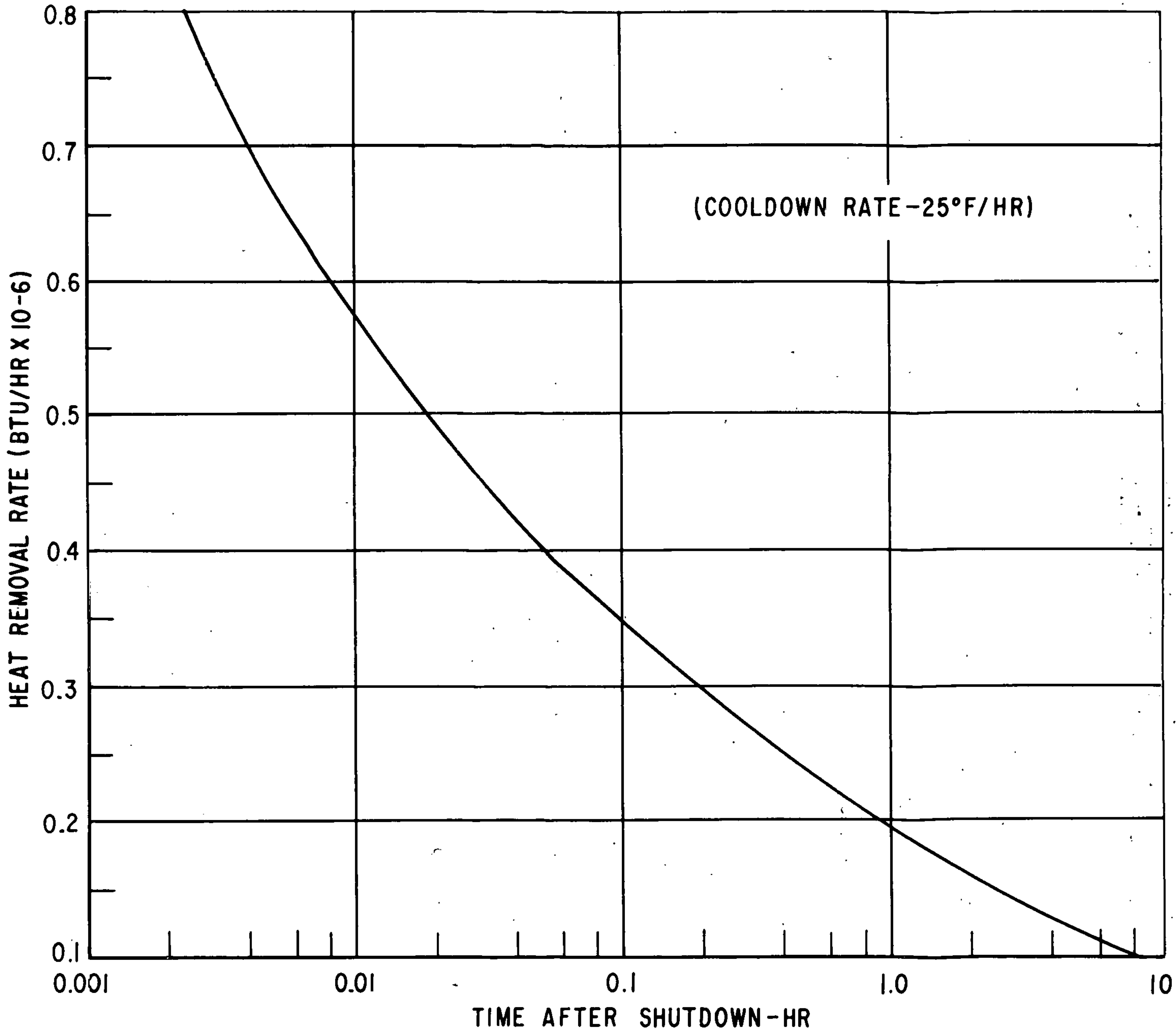


FIGURE 10.1 SHUTDOWN HEAT REMOVAL REQUIREMENTS

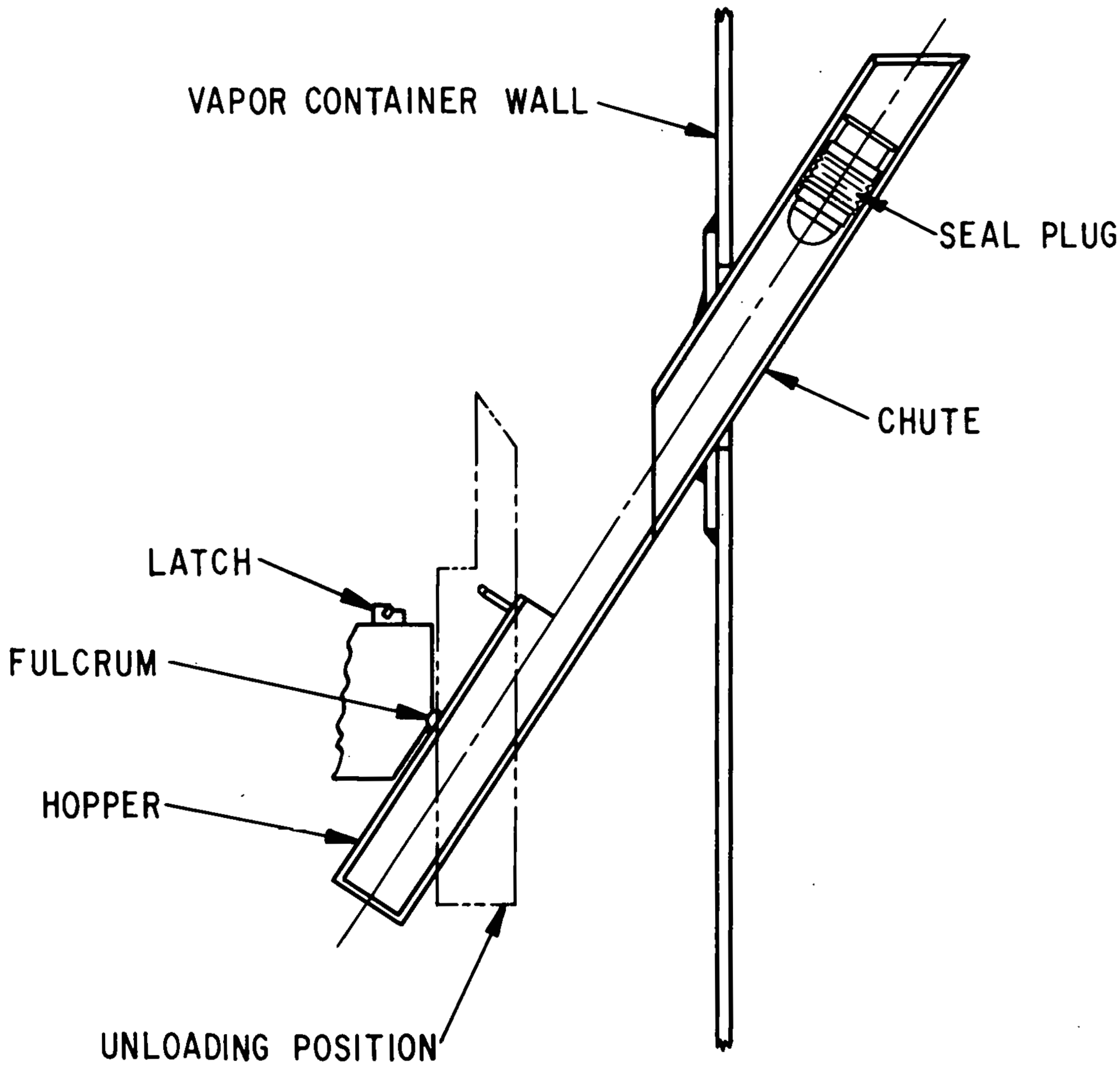


FIGURE 10.2 SPENT FUEL CHUTE

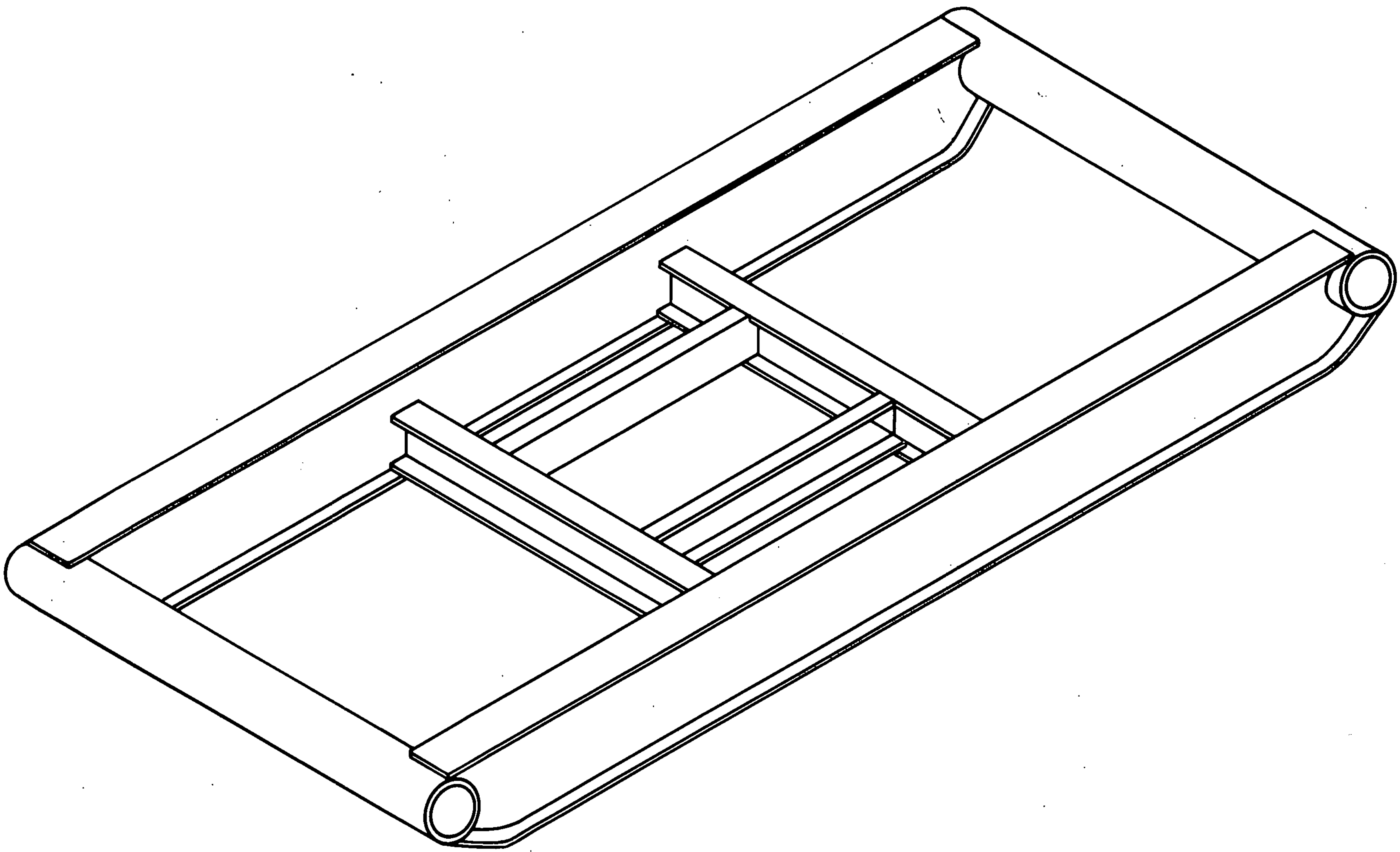


FIGURE 10.3 SPENT FUEL SHIPPING CASK SKID

11.0 POWER PLANT EQUIPMENT

11.1 STEAM CYCLE AND HEAT BALANCE

The steam system of the PL-3 consists of the main steam piping, a line separator, the steam turbine and the condenser. The energy from the reactor is utilized in the generation of steam in the steam generator. The steam passes through the main steam piping to the turbine inlet throttle valve. A steam separator is located just ahead of this valve in the main steam line. The separator removes the entrained moisture from the steam. The steam is then expanded through the turbine and is condensed in the surface condenser. The cool fluid used in the condenser is an ethylene glycol solution which is circulated between the surface condenser tubes and the air blast coolers where it is cooled by the cold Antarctic air. The air-ejection system removes the non-condensables from the condenser insuring proper exhaust pressure. The lube oil purification system keeps the turbine generator lubricating oil clean and useable. The condensate is then pumped from the hotwell to the boiler feed pump and returned to the steam generator. The heat balance, Fig. 11.1, shows the state of the steam and condensate at the various points in the cycle. Dwg. AEL-736 shows the components, connections and valves used in the PL-3 system. Table 11.1 presents a data summary for the secondary system components.

TABLE 11.1
SECONDARY SYSTEM COMPONENTS DATA

Turbine-Generator

Type: Multi- [illegible], non-extraction, condensing.	
Inlet steam pressure at full load	300 psia
Turbine speed	10,000 rpm
Generator speed	1200 rpm
Exhaust pressure	6 in. Hg
Rating (0.8 power factor)	1500 kw
Voltage	480 volts
Frequency	60 cps
Phase	3
Excitation	static

Condenser

Type: Horizontal, multiple pass.	
Duty	23.7×10^6 Btu/hr
Operating pressure	6 in. Hg
Inlet glycol temperature	80°F

TABLE 11.1 (CONT'D)

Air Blast Coolers	
Type: Continuous plate fin, forced convection.	
Duty	24.5×10^6 Btu/hr
Glycol circulation flow	1200 gpm
Tube outside diameter	5/8 in.
Condensate Pump	
Type: Centrifugal	
Head	110 ft H_2O
Flow	60 gpm
Feedwater Pump	
Type: Centrifugal	
Head	800 ft H_2O
Flow	60 gpm
Export Steam System	
Type: Submerged tube	
Evaporator duty	1.5×10^6 Btu/hr
Steam pressure (from steam source)	300 psia
Steam pressure (to camp system)	100 psig

11.2 TURBINE-GENERATOR AND LUBE OIL COOLING SYSTEM

The turbine-generator unit to be used in the PL-3 plant utilizes a non-extraction multi-stage condensing steam turbine connected through a geared speed reduction system to a salient pole A. C. generator. The turbine will operate at 10,000 rpm and the generator at 1200 rpm.

The turbine design will provide for operation over a range of exhaust pressures from 8 in. Hg absolute to 2 in. Hg absolute. The turbine-generator is designed to produce 1500 kwe at 6 in. Hg absolute using 300 psia inlet steam. Provisions have been made for moisture removal from the steam to minimize erosion problems in the last stages of the turbine. This moisture will be collected at the low points of the turbine exhaust casing and drained to a tank. From the tank, the moisture will be pumped to the main condenser by an eductor. The eductor derives its motive fluid from the condensate pump.

The turbine lube oil system will accomplish continuous filtering, purification, and temperature control. A centrifuge is used for filtering and purification and an oil-to-glycol heat exchanger for oil temperature control. Oil temperature is held constant at the bearing header.

The electric generator to be used in PL-3 is a six pole (salient), 1200 rpm, air cooled, open type machine. The generator will produce 480 volt, 3 phase, 60 cycle electric power. Other features of the machine include static excitation and automatic voltage regulation.

11.3 CONDENSER

The main surface condenser is of shell and tube construction utilizing multiple pass arrangement, suitable construction materials, and adequate deaeration of the condensate. Removal of non-condensables will be accomplished by a twin stage, two element ejector capable of removing the non-condensables to 2 in. Hg absolute. A feature of the condenser is the construction of a double tube sheet to guard against glycol leakage into the condenser and ultimate contamination of the steam generator.

For an ambient air temperature of +34°F and a 1500 kw gross output of the turbine generator, the condenser will have a duty of approximately 23.7 x 10^6 Btu/hr.

11.4 HEAT REJECTION SYSTEM

Waste heat from the main condenser and plant auxiliaries will be rejected to the atmosphere by forced air cooled heat exchangers using 60% solution of glycol by weight in water as the transfer fluid. The solution will be circulated through the main condenser, turbine lube oil coolers, and the condensate sub-cooler. The heat rejection duty at full load and at +34°F ambient air temperature will be approximately 24.5 x 10^6 Btu/hr and the flow through the air blast coolers will be approximately 1200 gpm at a mean temperature of about 105°F. The air blast coolers are continuous plate fin cores with 5/8 in. tubing arranged to achieve optimum exchanger effectiveness. Air will be forced over the cores at approximately 800 ft/min face velocity with a total air requirement of approximately 500,000 cfm at peak demand. Four fans for each cooler will be provided. The glycol system can function with practically no operator attention during all plant conditions. It offers inherent reliability from freezing.

11.5 CONDENSATE FEED SYSTEM

The condensate feed system utilizes two stages of pumping from the condenser hotwell to the steam generator. The first stage of pumping is accomplished by the condensate pump. This pump has sufficient head capacity to overcome the pressure drop associated with the cooling circuit and also provide adequate NPSH for the second stage of pumping. The boiler feed pump delivers feedwater directly to the steam generator through the feedwater control valve. Thus the two stages of pumping are accomplished by two pumps in series. This system of pumping has had successful operation on the PM-2A design and has also been utilized successfully in many advanced commercial power plant designs. Deaeration is accomplished in the main condenser hot well so that a deaerating heater or deaerator tank is not required. The condensate from the hot well must be subcooled to about 100°F for use as a coolant in the plant auxiliaries which include the primary and secondary blowdown coolers, the spent fuel tank, and the upper and lower shield tanks. The subcooling is accomplished in a water to glycol heat exchanger. After passing through the cooling circuit the condensate passes through the air ejector condensers and then to the feedwater pump suction. The condensate pump will have a capacity of approximately 60 gpm with a TDH of about 110 feet. The boiler feedwater pump will also have a capacity of 60 gpm but will have a TDH of about 800 ft. The feedwater pump will also be capable of 1400 ft of TDH at 10% flow capacity. These pumps are centrifugal units with orificed recirculation provided to protect against high temperatures during shutoff conditions.

11.6 EXPORT HEAT SYSTEM

Camp process steam is produced in a shell and U-tube type evaporator located on the secondary auxiliaries module in the plant. The supply water is furnished at an inlet temperature of 40°F. A small suction tank and feed pump is required in the inlet pipe to the shell side of the evaporator. A branch line off the main steam line, with saturated steam up to 300 psia in pressure, is used to evaporate supply water. The evaporator is designed for a transfer rate of 1.5×10^6 Btu/hr and furnishes saturated steam at 100 psig for camp process purposes. Condensate is drained to the condenser hotwell.

11.7 PIPING AND VALVES

Piping and valves have been chosen in accordance with their application; the parameters considered include system fluids, system pressures and system temperatures. The table below lists the ranges of these parameters for some of the secondary systems.

SECONDARY SYSTEM PARAMETERS

System Name	Fluid	Pressure	Temperature
Main Steam	Steam	Atm. to 550 psi	Amb. to 480°F
Export Steam	Steam	Atm. to 150 psi	Amb. to 365°F
Exhaust Steam	Steam	2 in. Hg to 15 psi	Amb. to 480°F
Condensate	Water	2 in. Hg to 40 psi	Amb. to 480°F
Feedwater	Water	Atm. to 600 psi	Amb. to 200°F
Primary Blowdown	Water	Atm. to 2000 psi	Amb. to 200°F
Glycol System	Glycol	Atm. to 100 psi	-90°F to 190°F
Lube Oil System	Oil	Atm. to 150 psi	Amb. to 300°F
Waste Disposal	Wastes	Atm. to 50 psi	Amb. to 190°F

Lines and valves are coded according to the system in which they are applied. The line and valves are numbered serially with a different group of numbers assigned to each major system.

11.8 EMERGENCY POWER

Emergency power is supplied by two diesel generator units; the rating of each unit is 250 kw. The generators are connected to the plant auxiliary bus through their own protective circuit breakers. If the plant auxiliary bus goes dead, drop-out relays will automatically start the diesels and bring them on-line to re-energize the bus. Battery power is used to start the diesel units. A one-line electrical diagram is shown in Dwg. AEL-743.

A source of preferred power for the nuclear instrument rack, the laboratory, radiation monitors, the control console, the BF_3 lifting mechanism and all DC power requirements is provided through a battery bank on the control skid. The battery is composed of ninety-five nickel cadmium cells and supplies 115 volts DC for DC power requirements and for two rotary 2 kw inverters to meet AC requirements. Two static battery chargers operated in parallel normally keep the batteries charged. The battery chargers are supplied with 440 volts AC, 60 cycles from the auxiliary bus when the batteries are being charged.

11.9 STARTUP AND SHUTDOWN POWER REQUIREMENTS

Auxiliary power requirements during periods when the plant is shutdown are supplied by the emergency diesel generator units. Similarly, during plant startup, power must be supplied to the plant auxiliary bus until the plant becomes self-sustaining. The various auxiliary power requirements are tabulated in Table 11.2.

11.10 SECONDARY SYSTEM CHEMICAL TREATMENT

The secondary system is treated much the same as any other moderate pressure boiler. Morpholine is added for pH and corrosion control. Either hydrazine or sodium sulfite is used for oxygen removal. Hydrazine is being given careful consideration, since it would considerably reduce secondary blowdown requirements. Oxygen control is simplified by providing adequate deaeration of condensate.

TABLE 11.2
AUXILIARY POWER REQUIREMENTS

Item	Power Required (kw) During Shutdown	Power Required (kw) During Startup
Primary Coolant Pump	-	35
Pressurizer Heaters	-	20
Glycol Pump	-	36
Auxiliary Glycol Pump	2	
Heat Rejection Fans	15	15
Feedwater Pumps	-	10
Auxiliary Oil Pump	-	3
Centrifuge Oil Pump	1	1
Condensate Pump	5	5
Waste Process System	3	3
General Ventilation	14	14
Special Ventilation	3	3
General Heating	84	84
Special Heating	50	50
Tunnel Ventilation	30	30
Lighting	30	30
Exciter	-	8
Evaporator Pump	-	
Miscellaneous Power	15	15
Total Power Required	252	362

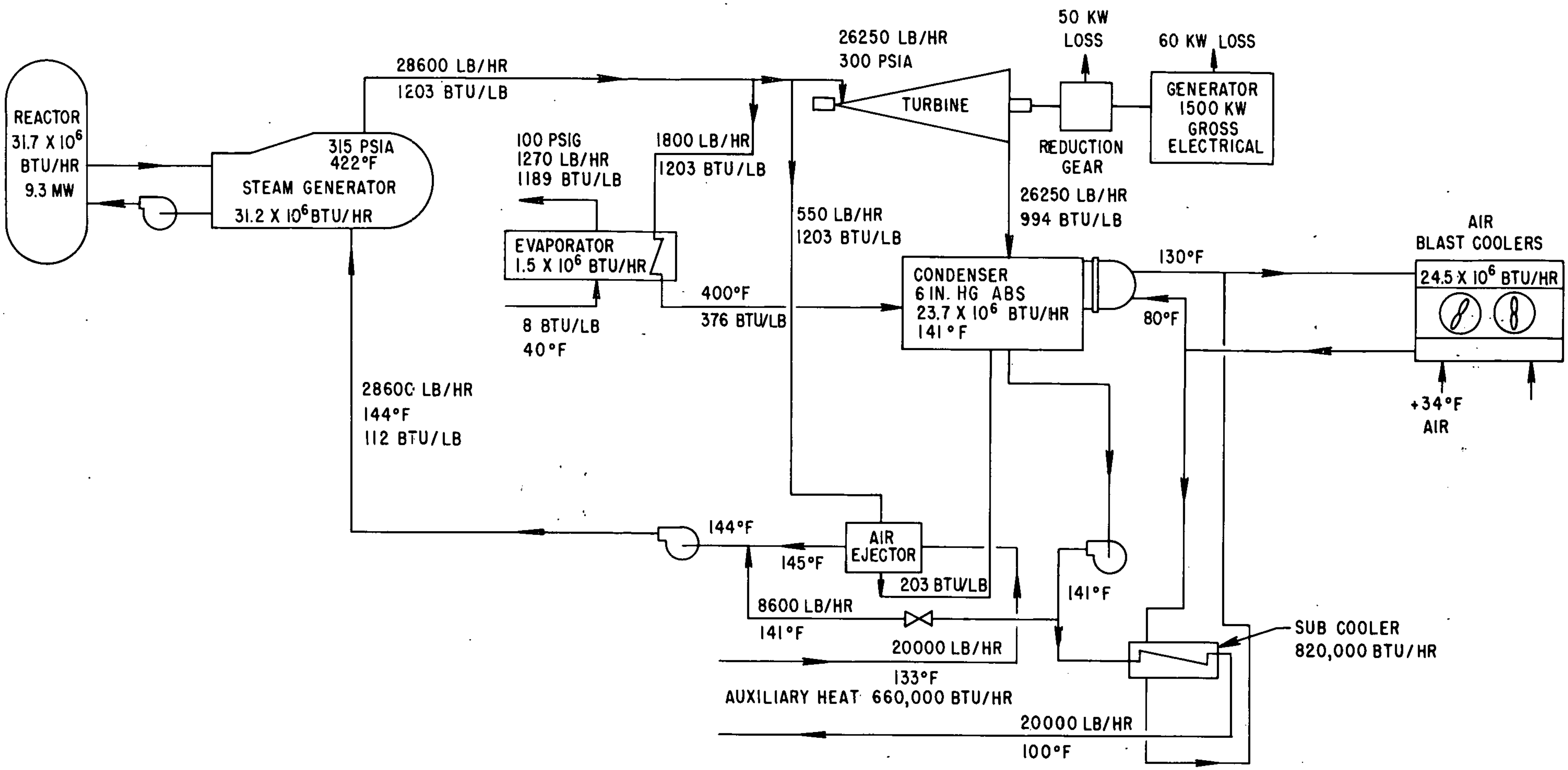


FIGURE II.1 PL-3 HEAT BALANCE

12.0 RADIOACTIVE WASTE DISPOSAL

12.1 GASEOUS WASTE DISPOSAL

The PL-3 specifications provide that any gaseous waste discharged from the plant shall not exceed 4×10^{-14} μc/cc. A further specification effecting gas activity states that the plant shall be designed to operate after the simultaneous release to the reactor coolant of 1% of the fission product inventory of the core after 2 yr of full power operation.

For gaseous waste system design purposes it was assumed that:

1. One percent of the fission gas inventory of a 2 yr core is released to the coolant.

2. The reactor continues to operate and release 1% of the fission gas being produced.

The above design assumptions do not imply that such circumstances are possible. In reality, such a large release would require an accident such as a core meltdown. Continued operation of the reactor is then virtually impossible.

Several problem areas are as follows:

1. Any gaseous activity present in the pools might be released to the surroundings. Equipment will be provided to control any such release.

2. Gaseous activity will be present in any leakage from the primary system and must be controlled.

3. Sampling and automatic analysis equipment will be designed to prevent the release of gaseous activity.

4. Any other potential sources such as tank vents must be controlled.

Air in-leakage is eliminated through operation of the system at greater than atmospheric pressure. In addition, radiolytic decomposition is eliminated through the practice of maintaining a hydrogen overpressure in the primary system. This hydrogen, on the order of 30-40 cc/liter of water, has the effect of causing recombination of the hydrogen and oxygen product of radiolytic decomposition. The non-condensables in the primary system resolve themselves to: 1) fission gas, 2) induced activity gas and 3) hydrogen added to suppress decomposition.

At primary system operating pressure, all of the gas will remain in solution. Venting should not be required during normal operation. However, when the pressure is lowered for removing the reactor vessel head, some gas could be released from the coolant. Postulating a considerable number of cladding failures coupled with the need for refueling, it would be highly desirable to remove any active gas in the primary system before attempting to remove the reactor head. For PL-3, where allowable limits are on the order of $4 \times 10^{-14} \mu c/cc$, this removal of active gas before removing the reactor head becomes quite important.

One concept of a gaseous waste collection and disposal system is shown in AES-596. Operation is as follows:

1. The top of the reactor vessel is vented to a condenser. Initially, a little water will be blown down until a steam dome is formed. The primary water is allowed to boil slowly at a rate slightly above that required to remove decay heat. The active gas will be stripped from the reactor and carried to the condenser with the steam.

2. The steam is condensed and returned to the reactor.

3. A vacuum pump removes active gases from the condenser and transfers them to a surge tank.

4. From the surge tank, the gas is compressed into bottles for off-site shipment and disposal. If the activity is sufficiently low, the collected gas might be slowly released for dilution and discharge to the atmosphere through the cooler discharge stack.

Such a system would be operated whenever removing the reactor head was anticipated. There would be little need for accurately determining fission gas content of the primary system. The major gas collected would be hydrogen with a total quantity less than 4 scf per cycle.

12.2 LIQUID WASTE DISPOSAL

The PL-3 specifications provide that any liquid waste discharged from the plant shall not exceed $10^{-8} \mu c/ml$. Evaporation is the only practical method of processing liquid waste to meet PL-3 specifications. Discharge of processed waste from the plant will be kept at a minimum by using this processed waste or "reclaim water" as primary system and pool makeup. The system schematic is shown in Dwg. AEL-710.

The various wastes are collected in a storage tank. The size of this tank will be determined by primary system decontamination waste volumes. Here the waste can be mixed, sampled, and any required adjustment in pH made. All waste sent to the system will be processed by evaporation. The evaporator condensate may be passed through a small demineralizer to guard against carryover and then stored for reuse or discharge.

The activity and other solids in the evaporator feed concentrate in the evaporator bottom. When a predetermined quantity of sludge has been accumulated, it is sent directly to a disposal drum which will be precharged with a dry concrete mixture. By going directly to the disposal drum, the problems associated with storing and pumping a highly active sludge which may solidify are eliminated. After adding the evaporator bottoms, the contents of the drum are mixed by rolling, then allowed to stand while the concrete sets. The contents are self-shielded; however, additional shielding will be provided if required.

The preliminary liquid waste system is designed to process any waste which may reasonably be expected from the PL-3 plant and to produce a high quality reclaim or processed water which is suitable for makeup or discharge. The activity and solids which are removed from the waste will be solidified so as to be acceptable for off-site shipment and disposal.

All of the radioactive water processing equipment is included on a single skid employing the same design approach as shown in AEL-747. This arrangement will provide easy access to components for operation and maintenance. The skid layout also makes use of the required water storage capacity to provide additional shielding of the components.

12.3 SOLID WASTE DISPOSAL

Any solid waste will be packaged in containers suitable for off-site shipment and disposal. It may prove desirable to place small contaminated objects in the liquid waste drums where they would then be immobilized in cement.

13.0 BUILDINGS AND TUNNELS

13.1 TUNNEL LAYOUT

Snow tunnels will be of the same general type of construction as that of the remainder of the station. Some modifications, however have been introduced as a result of prior experience both at Byrd Station and at Camp Century. These modifications are principally intended to reduce arch settlement.

The layout of the plant and tunnel system is indicated on Dwg. AEL-734. The principal units of the plant will be located in parallel tunnels with smaller connecting tunnels for personnel access and piping. The small tunnels will enter the larger tunnels at the ends to avoid the arch support of the main tunnels.

Access to the larger tunnels for equipment installation will be through ramps from the snow surface to the tunnel floor with a maximum slope of 10 percent. After construction is completed, the ramps will be allowed to fill with snow, except at the primary building and the waste processing building, since no further usage is anticipated. The ramps to the primary building and the waste processing building will be used periodically for fuel supply and waste removal respectively.

The construction of the tunnels will be a cut-and-cover operation utilizing the Peter Snow Plow. Except for the primary building, where considerable depth is involved, the normal procedure will be to excavate a ditch to the proper depth, with a shelf on each side to support corrugated metal arches. A typical cross-section is shown on Dwg. 7385-SK-S-11. The metal arches will rest on continuous timber footings. Snow will be backfilled over the arches to a depth of approximately 3 ft at the arch crown. The metal arches for the primary building will be constructed of sections similar to those of the shallower tunnels, but with a higher arch rise.

Bulkheads will be of timber construction and will be located at ends of the wide tunnel sections. In general, construction will be similar to those now in existence at the site. Openings will be provided where needed to conform with access requirements.

Personnel escape hatches will also be similar to those now installed at the site and will consist of vertical ladders in 5 ft diameter circular corrugated metal enclosures with a hatch above the snow surface.

13.2 BUILDINGS AND FOUNDATIONS

13.2.1 Building Superstructure

All building superstructures will be of panelized construction utilizing stressed-skin plywood panels. All components will be designed to be quickly erected and disassembled with a minimum of tools and erection equipment. Connection materials will be chosen with an emphasis on re-erection; nails, rivets or field welding will not be used for field assembly.

The primary building and that portion of the secondary building which houses the air blast coolers will have metal frames. Such framing, although primarily installed to support such equipment as the primary building crane and the cooler stacks, will support the wall panels as well. The remaining buildings will be frameless and will utilize the bending resistance of the wall and roof panels, suitably connected, to provide structural stability.

The design and fabrication of all panels will conform to the specification of the Douglas Fir Plywood Association.

Wall panels for all buildings will consist of 1/4 in. thick plywood sheets, pressure glued to both sides of framing members 3-1/2 in. deep. The framing at the panel edges will be suitable for engagement with contiguous panels. For all buildings, except the primary building, the wall panels will contain 1 in. x 3-1/2 in. interior studs on approximately 1 ft centers. Interior studs will not be necessary for the primary building wall panels, although some interior members may be required for the purpose of receiving bolts or lag bolts to attach the panels to the building frame. Wall panels for the primary building will be, in general, 4 ft x 8 ft. Wall panels for the other buildings will be 4 ft wide, running the height of the wall.

Roof panels for all buildings except the primary building will consist of 3/8 in. plywood sheets pressure glued to both sides of framing members 5 in. deep. One inch thick interior joists at 12 in. spacing will run in the direction of span, otherwise construction will be similar to that described above for wall panels. In general, the panels will span the entire width of the building. The panel size will be 4 ft wide and 20 ft 8 in. long. An exception will be in the air blast cooler room where the stacks will interrupt the span. Roof panels in this location will be supported by metal framing and will be sized to accommodate the stack penetration.

Roof panels for the primary building will be similar to the wall panels for that building.

Plywood for all panels will be DFPA Grade C-C Exterior Plugged. Framing for panels will be Douglas Fir, Construction Grade. Wood and plywood may be impregnated for fire resistance.

In all panels, the recess between the plywood faces will be filled with urethane foam insulation. A continuous sheet vapor barrier will be adhered to the plywood face forming the interior wall or ceiling. This barrier will be either sheet aluminum or aluminum foil reinforced with plastic coating. All panel surfaces including the vapor barrier will be shop painted with fire resistant paint except those areas of the vapor barrier to which pressure-sensitive tape will be applied.

Joints between panels will be gasketed with soft, closed-cell, elastomeric material. Compression of the gasket and tight fit of the panels, one to another, shall in general be accomplished by the use of cam or wedge type locking devices and/or bolts.

After erection of panels, all joints in the vapor seal will be closed by the application of impervious pressure-sensitive type protected by gasketed batten strips. Taped areas will be painted to match the remainder of the panels after the buildings are heated.

Exterior doors will be of light-weight construction of a type normally designed for cold storage buildings. The normal size of door opening will be 2 ft 6 in. wide by 6 ft 6 in. high. Doors will open outward and will be equipped with quick opening latches with a push bar on the interior side. Somewhat larger doors may be provided should mechanical design development indicate that larger items must frequently be moved through the openings.

Easily removable wall panels will be provided in the ends of all buildings for normal access for equipment which is too large to be handled through personnel doors.

All exterior doors will be "air locked" to reduce the flow of warm air from the buildings when doors are opened. This will be accomplished by means of uninsulated vestibules, approximately 4 ft x 4 ft, 8 ft high, located at the exterior doors inside the buildings. Double swinging doors with gravity closing hinges will separate the vestibule from the building.

Metal framing is required to support the discharge stacks for the air blast coolers. Since the stacks will protrude above the surface, they will be exposed to lateral wind loads. The framing will be designed to resist these loads as well as the weight of the stack and a portion of the building superstructure loads. Depending on design development this framing may be either inside or outside of the building.

13.2.2 Foundations

All buildings will be supported on spread-footing type foundations. Foundations will be proportioned and located so that snow-bearing pressures resulting from long-term loads will be approximately equal under foundations at the same elevation.

The primary system tunnel will be deeper than the tunnels housing the other plant components. Therefore, the primary building foundations will bear on snow which has been naturally compacted to a greater density and load-carrying capacity than that in the shallower tunnels. The primary building foundations will be proportioned so that the snow-bearing pressures under maximum loading will not exceed 1000 lb/ft^2.

A reduced maximum allowable snow bearing pressure, not to exceed the pre-construction weight of overlying snow at the level of the foundation-bearing surface, will be used to proportion the foundations in the shallow tunnels. The foundations at different levels will be sized so that the snow-bearing pressures under protracted loading will vary in proportion to the preconstruction loads of overlying snow. By accounting for the increase in natural densification of deeper snow, it is expected that the annual change in vertical separation between foundations at two levels will not exceed the natural contraction of the snow mass.

With the passage of time, it is probable that the floor of the tunnel cross section will change from a level to a crowned surface in which the outer edges will be depressed from their original elevation relative to the building floor, and the centerline will be raised. Where practicable, foundations will be located near the quarter points of the tunnel width, where the change in relative elevation is minimal.

To allow free circulation of tunnel air under the buildings or other potential heat sources, and in accordance with contract requirements, floor systems will be elevated to maintain at least 2 ft 6 in. between the tunnel floor snow surface and the lowest point on buildings or drain pipes. Foundations and jacking equipment will utilize this under-floor space.

Wood-framed, stressed-skin plywood panels will be used for the foundation-bearing surfaces. Wood panels are relatively light in weight, easy to erect, and provide a snow-bearing surface with desirable heat transfer properties. The bottom of the panels will be located at least 6 in. below the tunnel floor snow surface so as to bear on snow undisturbed by construction traffic, and to provide further insurance against heat transmission to the snow-bearing surface.

Metal beams and trusses will be used to elevate the buildings and to distribute loads to the snow-bearing panels. Bearing blocks and sills, of wood or other materials with good insulating properties, will be used as required to break any direct path which might conduct heat to the snow-bearing surface. Building floor panels will be similar to wall and roof panels as modified to suit heavier loads.

The primary building will be suspended at all times from jacks located under the building. To facilitate the remote level measurement and control of jacking, only four jacks will be used. The building frame will be used as a stiff box truss to deliver building and primary equipment loads to the foundations. A lined pit, stiffened to resist the lateral pressure of the snow and surcharge loads, will be provided at each jack screw rod.

U. S. Steel T-1 steel, or equal, will be used for the framing members of the primary building superstructure and substructure. To take advantage of the cold temperature ductility properties of this material in the unwelded condition, shop and field welding will be prohibited. In cases where shop welding is unavoidable in the detailed design, U. S. Steel HY-80 steel, or equivalent, will be used.

Drawing 7385-SK-S-9 illustrates a preliminary design of the foundations and supertsturcure of the primary building for the pressurized water reactor with the primary equipment enclosed in two vertical containment vessels.

The secondary and service buildings will be of light-weight construction, and will ordinarily be supported directly by their foundations. Jacking stations along the length of the buildings will be used, as required for local leveling or to raise the entire building until shims or cribbing members can be inserted to bear the load. To avoid buckling of the building walls or other damage from the settlement of foundations, a continuous truss of triangular cross section will be provided along each side of the building to deliver loads to the snow-bearing panels. Floor support beams will cantilever beyond the top chord of the truss on each side.

Aluminum will be used generally for the secondary and service building substructure framing. A preliminary design of a portion of the maintenance and storage area of the entrance building is illustrated on Dwg. 7385-SK-S-11.

13.2.3 Jacking and Level Indication

The proposed primary building jacking system consists of four Limitorque HM-4X valve operators installed on trusses supported by the foundation with the building hung from their screw stems. The jacks will rest on spherical lubrite bearings designed to compensate for five degrees of tilt of the footings. A system of sliding guides, also lubrite equipped, will hold the jacks in line with the building and relieve any bending moments on the jack screw. Lubrite bearings are bronze with the bearing surface drilled in a suitable pattern and the holes filled with a solid lubricant.

The jacks will be electric motor operated and equipped with torque and limit switches to prevent operation either under overload or when the jacks have reached the limit of travel. Operation of the jacks will be from a control panel containing the jack controls and level indication, with a diagram of the primary building containing lights to show the jack and manometer selection. The jacks will have a selector switch allowing operation of one jack at a time and a spring-return control switch to raise or lower the building. A lock on the selector switch will prevent unauthorized operation. A jack range indicator will be provided to show the amount of jack movement available at any time.

The secondary building will be jacked by portable jacks and shims placed as needed. The jacks will be similar to Blackhawk portable power jacks, which have a lever-action pump and a ram connected by about 10 ft of hose, allowing the operator to jack from a point not under the building for better control of the operation.

The level indication will consist of a methanol filled manometer system to indicate the building level and four pendulum type tilt indicators to detect uneven settlement of the primary building footings.

The primary building manometer system will have five capacitance type level indicators, one at each corner of the primary building and one as reference on one of the secondary buildings. These will be provided with a selector switch to connect each manometer to a single-panel indicator and, at the same time, light an indicating light on the panel diagram to show the location of the manometer being read. Thus, by checking correspondence of the jack selector and manometer lights, the operator can be sure that the proper jack is in operation.

The system for the secondary buildings will be provided with flat glass, reflex type gauges for local reading, but valves will also be provided so that the remote reference indicator in the primary building system can be used to check the fluid level in the secondary building system. Solenoid valves, operated from the control panel, will be provided to isolate these systems and to connect the reference manometer to either.

The manometer systems will be closed with a fluid line and an equalizing line connecting the points. The system will use copper tubing where it can be protected from physical damage; elsewhere, copper pipe will be used.

A storage tank and two pumps will be provided to contain a reserve supply of fluid and to fill and empty the system. One pump will be a rotary pump of approximately 3 gpm capacity for transfer service. The other will be a remote head diaphragm pump, adjustable from 0 to 0.1 gpm for adjusting the level. The pumps and associated valves will be operated from the control panel and the trim pump and its rate control will be in the building. With this arrangement, all filling operations can be performed from inside the building. Hose connections will be provided so the transfer pump may also be used to pump fluid between shipping drums and the storage tank.

The foundation tilt indicators will consist of a pendulum and ring contact mounted to detect deviations of 5 degrees from vertical in any direction. Contact between the pendulum and the ring will complete the electrical circuit to light a tilt indicator light on the panel diagram.

13.3 CRANE ARRANGEMENT

A study was undertaken to determine the most practical and economical type of crane installation for the primary building. Of necessity, the primary building will be a metal frame structure due to its height, thus lending itself readily and economically to support of a bridge-type crane. A gantry-type crane would necessitate a wider, and consequently heavier, building because of the size of the primary equipment and the space required for the gantry legs on either side. Other disadvantages of a gantry crane are that it would not service as great an area as a bridge-type crane, would represent a greater hazard to personnel, and would limit the use of walls for supporting piping, ducts and equipment.

A 10-ton bridge-type crane and a 2-ton auxiliary for use in refueling are specified for the primary building. The main hoist may be electrically operated if final design requires long vertical lifts. The auxiliary hoist will be hand operated for accuracy in refueling operations. The crane will be fabricated principally of aluminum because of erection weight and climatic requirements; such construction has become relatively common.

The secondary building will be of relatively light panel construction and will be unable to support a bridge crane. The hoisting requirements for this building can be adequately served by a 5-ton, manually operated gantry-type crane. The crane will have rubber-tired wheels for use in and around limited spaces and equipment mounted on skids without need for tracks imbedded in the floor. It will be designed for quick and easy disassembly and reassembly by two men. Disassembled parts will be sized to permit passage through a man door and will be easily transportable by two men. These specifications enable use of the crane for maintenance in all other buildings of the facility not having permanent hoist installations. The hoisting mechanism will be a chain hoist of 5-ton capacity and standard construction, but smaller hoists to be furnished in the maintenance building can also be used with this gantry. No major problems are anticipated in the design of hoisting facilities for secondary equipment. Hoisting equipment is manufactured in a variety of standard units and designs today which can be adapted for special applications. If final design indicates that the special secondary building gantry cannot serve a particular skid, e. q. the turbine-generator, an adaptation can readily be made to suit existing conditions. One of the gantry's end legs may have to be removed, for instance, and a special "A" frame provided which can be attached to the turbine-generator skid at the time of overhaul for removal of turbine-generator rotor. In such a compact installation, only final design can dictate specific maintenance needs.

13.4 TUNNEL COOLING AND VENTILATING

The tunnels in the plant complex must be maintained at a constant low temperature in order to limit the temperature rise in the surrounding snow. Since the buildings within the tunnels are maintained at a higher temperature than the ambient snow temperature, there is a constant heat flow tending to increase the temperature of the snow. This problem can be alleviated by circulating cold air through the tunnels throughout the year. During the Antarctic winter months, air at temperatures as low as $-80^{o}F$ will be drawn from the atmosphere, circulated through the tunnels and discharged to the atmosphere. During the summer months when temperatures range as high as +34°F, cold air will be drawn from air wells sunk in the tunnel floor, circulated through the tunnel, and discharged to the atmosphere.

The air well installation will consist of a well casing of about 18 in. diameter, inserted 20 to 25 ft into a drilled hole in the snow. The depth of the uncased hole should be about another 25 ft, for a total of 50 ft beneath the tunnel floor. The depth of this uncased drilling is not considered to be critical; however, sufficient uncased area should be provided to permit easy access of air across the snow boundary. An axial flow fan, discharging upward, will be mounted on the top of the air well casing with a discharge duct mounted on the discharge flange of the fan. This discharge duct will terminate in a diffuser so that optimum direction of air flow will be provided.

The location, number, and size of the air wells will be dependent on final tunnel configuration. The total air well capacity is a function of building area, building heat loss, snow temperature, and maximum permissable tunnel temperature.

The present tunnel arrangement requires several air wells, each with a capacity of 3500 cfm.

The air discharge from the air wells will be vented from the tunnel through ducted fans. These fans will also provide tunnel cooling during the winter months when the air wells are not in service. During this period, air will be drawn into the tunnel through inlet vents and discharged through the ducted fans. The amount of air circulated will be the minimum required to maintain tunnel temperature within safe limits.

It is intended to provide each of the main tunnels with at least one exhaust fan. With this arrangement each fan will be sized to handle 7000 cfm.

In as much as the ducted fans will be sized primarily to exhaust the air well discharge, it can be concluded that the amount of outside air that can be circulated will be approximately equal to the total air well capacity, so that whenever ambient air temperature is lower than snow temperature, outside air can be used for tunnel cooling.

13.5 ELECTRICAL SYSTEMS

Cabling to equipment will be multiconductor, 600 volt, arctic butyl or polyethelene insulated, PVC jacketed cables with a ground wire. Cable trays will be used, where applicable; feeders and control cables leaving a tray will be enclosed in conduit. Fluorescent lighting circuits will utilize trolley bus duct system so as to provide maximum flexibility. Other lighting cable circuits and receptacles will be enclosed in conduit. Any motor cable circuits which may be exposed to mechanical injury will also be enclosed in conduit.

Miscellaneous electrical equipment will be grounded, using the ground wire provided in each multiconductor cable. This ground wire will be connected to the ground bus of the distribution panel, which will be connected in turn to the main ground cable. In addition, all metallic building panels, large motors, electrical enclosures, cable trays, conduit, pipes, building steel, and equipment skids will be grounded individually by a bare copper conductor "Cadwelded" to the main ground cable loop. The main ground cable will extend throughout the tunnel complex establishing a low resistance ground path to the neutral of the generator. The ground cable will be extended to any additional standby power source and feeder distribution points to eliminate a ground return through the snow fields.

The one-line diagram, Dwg. 7385-SK-E-1, illustrates the power and lighting circuits.

Power panels will be 3-phase, 277/480 volt and will contain breakers for miscellaneous 3-phase power feeders. Power requirements for normal lighting and receptacles will amount to 30 kva; an additional 15 kva will be required for miscellaneous single phase motors, laboratory equipment and control relays. The service required for building heating, humidity control and pipe line heating will amount to 135 kw; tunnel supply air and ventilation requirements are an additional 65 hp. Thermostatically controlled electric heating cables and full covering of insulation will be provided for water pipe heating, where required. An additional thermostat will be provided for low temperature alarm. A static type annunciator will be used to monitor pipe heating systems and primary building ventilation fans. A malfunction in the system will sound plant alarm. Power panels, lighting panels, and starter groups will be premounted on free standing plywood panels to keep field installation and wall mounting to a minimum. If wall space is at a premium, motor control centers will be centrally located.

Miscellaneous lighting and receptacle panels will be 3 phase, 120/208 volt and will contain breakers for incandescent lighting circuits and miscellaneous 120 volt equipment. Receptacles for 120 volt circuits will be of the grounded type.

Motors will be totally enclosed and, where applicable, fan-cooled type and motor strip heaters will be specified. Magnetic starters will be combination type, complete with circuit breakers, thermal overloading protection and control stations mounted in covers.

Wherever possible, equipment terminals will be the plug-and-receptacle type of connector. Where this is not feasible, standard terminations with Stakon pressure connectors and tape will be used.

During building erection, a minimum of 50 kw of diesel electric power will be required for construction tools, temporary lighting and electric heating of the erected buildings.

13.6 BUILDING HEATING AND VENTILATING

A study of various types of heating was conducted. Electric heating is a dry type of heat in a climate of extremely low humidity and was selected in spite of low efficiency for the following reasons:

1. Electric heating in this particularly cold climate will be much more reliable since it presents no major problems due to freezing, as would be encountered in a wet type or boiler system. Start-up problems, initial equipment costs, and maintenance will be less. The need for a separate exhaust and intake stack is eliminated.

2. Heat will be required during building erection of the first season. Heat will also be required during the second season of plant erection. A mobile diesel electric generating unit is the logical heating choice for these periods.

3. A standby diesel electric generator will be provided for plant start-up. During shutdown periods, this capacity may be available for heating.

4. Use of a second type of heating system would multiply maintenance and spare parts problems.

The heating system to be provided will consist of a standard air conditioning unit, capable of being shop fabricated and tested stateside. This system will incorporate control and preheating of ventilation air, filtering, heating by either electricity, steam, hot water or combination, humidification and distribution control. Steam or hot water heating coils, for areas where steam is available and/or recovery of waste heat proves economical during plant operation, can be readily installed in such units very economically without danger of freeze-up if piping serving these coils is located entirely within the building served. Thus the heating system will take advantage of the electric power available during construction and shutdown periods and the most economical and available power during plant operation. It will be a single combination system involving no extra spare parts.

Radiant heaters may be used in specific areas such as the control console, where an operator must sit for prolonged periods and where additional heat is required beyond that provided by the building heating system.

A typical arrangement of the heating and ventilating system is shown on Dwg. 7385-SK-M-2. Ventilation air as required will be taken from the tunnel. It will be preheated immediately upon entering the air conditioning unit to control static electricity build-up. A ventilation air control damper (D_1) will automatically regulate the supply of ventilating air. A self-closing damper (D_5) and a similar damper (D_6) in the exhaust will be fitted where required to close on a pressure build-up within the building to contain and permit control of any contaminated air in the event of a release of radioactivity within that building. Ventilation air will mix with return air entering the return air grille. Return air will be automatically controlled by the return air control damper (D_2). This air mixture will then pass through suitable filters to remove any dirt or dust. It will then pass through the air conditioner main electric heating unit (E_2) and heating coil, if the latter is fitted for use of steam or for waste heat recovery. The air will be heated as it passes through these units to a temperature slightly lower than that required for heating individual rooms at the various supply air terminals. The air will then be drawn over a pan-type humidifier and heated by an electric immersion unit (E_4) as required, automatically controlled by room humidistat (H_1). Air leaving the supply fan (F_1) will be discharged to branch outlets fitted with room supply air heaters (E_3) controlled by individual room thermostats (T_1). Supply air diffusers located near the floor will sweep the floor and far wall with a curtain of warm air to maintain comfortable conditions.

The quantity of exhaust air will be slightly greater than the fresh air supply to maintain infiltration into the building. An axial fan (F_2) and ductwork will take relatively cold air from near the floor and exhaust it directly to the atmosphere, not into the tunnel.

The temperature control system will be automatic. On start-up, with the supply fan (F_1) running, the room temperature below the setting of a thermostat (T_1) and with the air condition air stream temperature below the setting of a bulb-type thermostat (T_2), a program motor will be modulated by the latter signal for maximum heating by the electric heaters (E_2 and E_3). The ventilation air intake control damper (D_1) will be closed, electric preheater (E_1) will be off, the exhaust air control damper (D_4) will be closed and the exhaust fan (F_2) will be off. Balancing dampers (D_3) will have been set to balance the system and then locked in position. As the air-to-supply fan (F_1) approaches the setting of (T_2), the program motor will energize electric heaters (E_2) as required to maintain the (T_2) setting. Room thermostat (T_1) will energize electric heaters (E_3) to maintain space setting, ventilation air intake control damper (D_1) will open, preheater (E_1) will be turned on, exhaust fan (F_2) will start and exhaust air control damper (D_4) will open. When the supply fan (F_1) is off, exhaust fan (F_2) will be off, and exhaust, supply and return air dampers will be closed. Supply fan (F_1), exhaust fan (F_2), fresh air intake control damper (D_1), and exhaust air control damper (D_4) will be wired to the relay cabinet of fire protection system to shut-down upon actuation of the fire protection system.

During normal plant operation, units fitted with steam or hot water coils will be manually shifted over to de-energize electric heaters (E_2). Automatic controls will then modulate hot water or steam coil control valves.

The primary building ventilation will be sized for six air changes per hour during normal plant operation and will exhaust to the atmosphere and not to the tunnel. During shut-down for maintenance, refueling or other requirements, heating and ventilation of this building will be similar to the system used for the remainder of the complex. It will be possible to monitor the primary building exhaust.

13.7 LIGHTING

General over-all plant lighting will be 120 volt fluorescent. Lighting for tunnels, escape hatches, air-blast cooler areas, waste processing building, upper reactor level, storage building and miscellaneous fixtures will be 120 volt incandescent type. The approximate operating illumination intensities given in Table 13.1 will be established.

TABLE 13.1
OPERATING ILLUMINATION INTENSITIES

Location	Foot-Candles
Air-Blast Coolers	10
Decontamination	10
Purification Area	30
Reactor Lower	30
Reactor Upper	10
Waste Processing Building	15
Maintenance Building	40
Personnel Building	50
Heat Exchanger and Condenser Area	30
Generator and Switchgear Area	30
Control Room Area	50
Diesel Room	30
Laboratories	40
Storage Areas	10
Tunnel Lighting	10

All areas will have portable, lightweight, automatic, reelites each equipped with a 100 watt hand lamp for supplementing fixed area lighting during periodic inspections and maintenance of equipment.

Automatic emergency lighting is provided for the possible loss of normal power. These units will consist of nickel cadmium batteries, two 6 volt lamps, a voltmeter, a trickle charger with neon lamp and automatic relay. They will be located strategically throughout the building and tunnel complex. The units for tunnel illumination will be located within the buildings, and the lamp heads will be mounted on the outside walls of the buildings.

13.8 FIRE PROTECTION

A low-pressure, centralized-storage carbon dioxide fire protection system is contemplated for the entire PL-3 complex. High-pressure carbon dioxide and water sprinkler systems were also considered. Installation of the fire protection system during the first season, along with erection of the buildings, is recommended for the following reasons:

1. The system can be installed so that it will not interfere with installation and erection of plant equipment the following season when time is a premium.

2. Fire protection will be provided for the unoccupied and idle buildings during the first year.

3. Fire protection will be provided during the entire plant installation and erection period the second year.

4. Early installation will afford sufficient time for a complete test of the system to insure adequate and satisfactory operation, to check the density of carbon dioxide discharge in the building and to set satisfactory timing sequences of the various portions of the system.

Use of a water or liquid sprinkler system was eliminated from consideration because of the following reasons:

1. Such systems are not suitable for electrical fires.

2. Systems containing water present many obvious problems of operation and maintenance at the low temperature encountered in PL-3 application.

3. Cleanup at these low temperatures would be extremely difficult.

A CO_2 system has been selected because:

1. Such a system can be used for all types of hazards involved.

2. There is no difficult cleanup necessary after use, nor is there damage to equipment.

Two types of CO_2 systems are available, low-pressure and high-pressure. The high-pressure CO_2 system utilizes bottles or cylinders at approximately 2,300 psig pressure for CO_2 storage. Liquid CO_2 is stored in a centrally located unit in the low-pressure system. This unit and the dry ice converter would be enclosed in separately insulated aluminum housings for outdoor installation.

The low-pressure CO_2 system is favored over the high-pressure system because:

1. No floor space in buildings is required, since the storage unit can be located in an unheated tunnel area.

2. The entire storage capacity is available at all times for the protection of all hazards covered.

3. Maintenance is less, particularly because only one storage unit has to be recharged.

4. Greater protection is afforded because a higher percentage of carbon dioxide snow is discharged which cools combustibles and fires to sub-ignition temperatures. In addition, a fire can occur in the same area repeatedly, and protection is still available.

5. A nitrogen expellant is not required with a low-pressure system.

A schematic diagram, Dwg. 7385-SK-M-1, shows the essential parts of the low-pressure system. The storage unit (1) is a self-contained unit with integral electric heater and refrigeration unit, including necessary controls to maintain the carbon dioxide in a liquid state at $0^{o}F$, and approximately 300 psig. Normally the electric heater will maintain the internal temperature of the storage unit at $0^{o}F$. Should the tunnel temperature rise above $0^{o}F$, the refrigeration unit will maintain the correct internal temperature. If at this time the refrigeration unit should fail, a built-in relief valve protects the storage tank of the unit against excessive pressure. A normally pressurized manifold, including two normally locked-open shut-off valves (3), each serving only one tunnel, are provided on the unit for maintenance purposes. From this manifold, headers will be extended to the various buildings and fire hazards of the complex. Dwg. 7385-SK-M-1 does not indicate every building or hazard. Its purpose is to show typical arrangements for the types of hazards requiring protection. An entire building with no specific potential hazard, such as the waste processing building, will be protected by total flooding using a master valve (4) as a master-selector valve. Buildings with several specific hazards, such as the secondary building with the turbine-generator, the switchgear, and/or other equipment will be protected by spot flooding, using one master valve (4) and a separate selector valve (5) for each hazard.

The storage unit capacity is selected on the basis of the maximum hazard to be protected. In a normal installation where recharging of the storage unit is possible immediately after use, sufficient charge must remain to adequately cover a second discharge into the maximum hazard. Realizing that this storage unit can be recharged only once a year, and after consultation with the manufacturers, a storage unit will be selected of standard capacity to insure four discharges covering the maximum hazard.

Actuation of master valve (4) and selector valve (5) by the pilot control cabinet (6) can be accomplished by manual operation of push button (7) located at each building exit, or automatically by heat-actuated devices (8) located in the vicinity of potential fire hazards. Actuation causes alarm (9) to sound, warning personnel to vacate the area in the vicinity of discharge, and operates relay cabinet (10) to shut down ventilation fans, exhaust fans, dampers and to operate an annunciation system. Manual-coded stations will give audible and visual identification of fire locations. After a preset interval, carbon dioxide will be discharged through nozzles (11) to cool and extinguish the fire. A control panel (12) will house all auxiliary relays, code transmitter, alarm contacts and the normal and standby power supplies for the complete operation and supervision of the system.

Recharging of the low-pressure system will be by use of dry ice and a dry ice converter (2). Facilities are presently available for air-lifting or air dropping, if necessary, dry ice in 10 in. cubes weighing 50 lb each. These cubes will be placed in the dry ice converter (2), which consists of a suitable tank to which heat will be applied for converting the dry ice into liquid carbon dioxide. The converter has facilities for pumping the liquid CO_2 into the storage unit (1).

13.9 PLUMBING AND PIPING SYSTEMS

This section covers the hot and cold water service, sewage drains and vents, and the radioactive or contaminated waste systems. Water for the plant will be furnished and piped to the plant complex from the camp supply.

One electric hot-water heater will be adequate to heat water for use at the plant complex. Generally, copper tubing, wrought copper fittings and brass valves will be used for the hot and cold water systems. Nitrogen-charged shock absorbers will be used to suppress water hammer as required. Electrical tracing will be employed where necessary to prevent freezing and all lines will have adequate insulation, covered on the outside with an aluminum skin.

Generally, plumbing drains and vents will be of copper tubing and brass fittings, since cast iron is not suitable for service at extremely low temperatures. These materials are suitable for the temperatures involved and are conventionally used in normal climates. All sewage drain lines will be electrically traced to prevent freezing and will be adequately insulated and covered on the outside with an aluminum skin. Due to difficulties in assuring sufficient pitch of the drain system, final design will probably dictate the collection of sewage in a localized tank. Duplex sewage ejectors will then pump this sewage into the camp system. Since the sewage system will be pumped, the mains can be run within the buildings whenever possible rather than outside in the below-freezing temperatures. Pitch of the system will be a minimum of 1/8 in. per foot, with no pockets.

To simplify construction problems at the site, the fixtures and equipment required for the toilet area will be prefabricated and mounted on a skid as indicated on Dwg. 7385-SK-M-3 such that final installation will simply entail connection of electric power supply and the following four bolted piping connections:

a. Cold water supply
b. Hot water supply to laboratory area
c. Vent
d. Sewage drain

The entire unit will be fabricated of stainless steel, including base, water closets, lavatory, sewage tank, hot-water heater, floors and walls. The hot-water heater and sewage ejectors will be completely wired including starters, relays, float switches and junction boxes. All available unused space on the skid unit will be used for mop, pail, toilet paper, towel, soap and other associated supply storage.

The decontamination area will also be prefabricated. It will contain an emergency shower, a laboratory type cabinet sink, and industrial type clothes washer-extractor combination unit and a clothes dryer. Fabrication will be entirely of stainless steel and advantage will be taken of unused space on the skid unit to provide storage for accessories required for this facility, such as mop, pail, towel, soap and clothes storage.

The contaminated (radioactive) waste piping and fittings will be of stainless steel. There will be no pockets in the system, which will drain by gravity under the buildings for discharge into hot waste storage tanks. Electrical tracing and insulation will be employed to provide protection against freezing. Final design may dictate collection and pumping of contaminated waste, in a fashion similar to the sewage, if gravity drainage proves inadequate.

Electrical tracing of piping has been selected because of the many problems associated with steam tracing at freezing temperatures, especially at temperatures as low as -80°F. Both systems have demonstrated extensive and satisfactory performance histories.

14.0 SERVICES

14.1 HEALTH PHYSICS FACILITIES

The health physics facilities will contain the necessary equipment to analyze prepared water samples and air samples for radioactivity and to develop and interpret neutron and beta-gamma sensitive personnel monitoring films.

14.1.1 Sample Preparation

The preparation of high-activity samples will be accomplished in the chemistry laboratory. This facility cannot be used to prepare low activity samples due to the possibility of cross-contamination. Therefore, the health physics facility will contain the equipment necessary to prepare low-activity samples of counting.

14.1.2 Counting Room

The counting room of the PL-3 is located within the reactor plant complex, but at the maximum distance possible from all systems containing radioactivity. The room will contain Geiger-Mueller and scintillation detectors and scalers for alpha, beta and beta-gamma analysis of system samples, air samples and swipes.

The room also contains the necessary equipment for interpretation of the personnel monitoring films. The densitometer is provided for interpretation of the beta-gamma sensitive films, and a projection microscope for counting neutron tracks.

14.1.3 Personnel Monitoring Dark Room

This room is equipped with the necessary facilities to develop the personnel monitoring films. It contains both hot and cold water, a sink equipped with photographic developing tanks, storage space and a bench. The room is painted with a flat black paint and provided with a light tight door to minimize light reflections which could cause fogging of the monitoring films.

14.2 DECONTAMINATION AND LABORATORY FACILITES

In addition to chemical instrumentation, it will be necessary to provide laboratory facilities for back-up purposes and for the routine analysis not handled by installed instrumentation. The single chemistry laboratory shown in Dwgs. AEL-729 and AEL-739 is proposed for both radioactive and non-radioactive chemistry. As it is recognized that this laboratory will not be suitable for low activity counting equipment, the health physics laboratory will be used for these purposes.

Facilities for decontamination of small parts and a washer-dryer combination are also included with the laboratory equipment. Any chemical recording instruments will be located in the laboratory. Where necessary, alarms and readouts will be located at the main plant console.

The chemistry laboratory is located in close proximity to the liquid waste disposal and purification systems to minimize the distance that radioactive samples will have to be carried. The laboratory collapses for shipment, providing a compact package. Upon installation, the two parts are separated and the area between is used as a work space.

14.3 MAINTENANCE FACILITIES

Provisions for small and medium size component repair and for nuclear and standard instrumentation and control equipment maintenance are included in a machine shop located between the control console modules and personnel facility. The machine shop contains, in addition to the standard machine shop hand tools and working stock, such components as a hydraulic press, hoists, grinding equipment, power saw, lathe, compressor, oscilloscope, autotransformer, cutting and welding outfit, and a brazing and soldering set. Space is also allocated in this enclosed area, for the storage of 2 years' supply of plant component spare parts.

Typical examples of maintenance that could be performed in this area are service and repair to pumps, heat exchangers, and piping and valving. Major components such as turbine rotors and condenser tube bundles must be serviced in the laydown areas adjacent to their modules.

An additional maintenance area, located between the waste processing module and the chemical laboratory module, will be designated as a "hot" component repair center. A duplication of some of the main machine shop tools, such as a drill press, lathe, grinding equipment, welding and cutting gear will be required to perform maintenance on items where some precaution is required due to exposure to radiation.

14.4 PERSONNEL FACILITIES

Entry from the main camp to the plant complex must be made through the enclosed personnel area. This area is approximately 20 ft x 20 ft and provides office space, consisting of desks and filing cabinets for both the plant Health Physicist and Superintendent. A sanitary fixture and sink is located adjacent to this office space within the personnel area.

This area also houses the health physics analysis equipment. The instrumentation required for radioactive sample analysis, along with film badge interpretation equipment, will be utilized by the plant's health physicist. A portion of the building, near the entrance from the camp, is used for personnel monitoring purposes.

14.5 TUNNEL UTILITIES & CONNECTIONS TO CAMP SYSTEMS

The principal connections between the camp area and the plant complex are for: (1) Supply water from the camp for plant utilities and process steam evaporator; (2) Export steam supply to the camp from an evaporator in the plant; (3) Electrical supply, terminating at the perimeter of the plant complex for connection to the camp system; and (4) Sanitary sewage, also terminated at the plant perimeter for the connection of the drain line from the plant complex with the camp's disposal area.

The access and piping tunnel connecting the camp tunnel and the plant complex enters the plant complex at the personnel building where necessary monitoring of personnel and equipment is performed.

15.0 OPERATION AND MAINTENANCE

The operational work load is controlled by the requirements for the reactor operation, chemical and waste disposal systems operation, and refueling. The normal reactor operation is essentially automatic, relying on the inherent load-following characteristics of the PWR. Only one man is required at the console to monitor the plant operations. A second man is required to periodically check plant equipment and serve as a relief for the control room operator.

The above two shift positions are filled by the one man per shift designated Shift Supervisor, and the one man per shift designated Control Room and Equipment Operator. The Shift Supervisor can serve, as desired, in the capacity of either control room or equipment operator.

Chemistry and health physics work will be handled by the process control technician. The continuous processing systems are designed for automatic operation. Periodic sampling and routine maintenance will be accomplished by shift operating personnel under the direction of the process control technician. He will also be required to direct the processing and disposal of radioactive waste which will be an intermittant operation on a batch basis.

Refueling will be scheduled every two years, utilizing the entire crew for three 12-hr days.

Pressurizer heaters, the coolant pump and steam generator have proven reliable in similar service and require only a minimum of maintenance. Spare parts for this equipment will be held in stock.

Standard power plant equipment and auxiliaries are specified for the secondary system. Normally scheduled maintenance following the manufacturers recommendations results in minimum downtime and manpower requirements. All secondary systems components are accessible for maintenance during plant operation.

Instrumentation will be modular, allowing rapid changeout and economy of maintenance operations. Self-checking circuitry will be incorporated as an instrumentation design feature and will assist in both preventative maintenance and in reducing downtime.

Routine maintenance on the primary and secondary systems and components, as well as instruments and controls, will be performed by instrument technicians and mechanical and electrical maintenance specialists, assisted by the other men on duty. In the event of an emergency, any or all of the plant crew would be "on call". Major overhaul and maintenance work would be accomplished with the help of the entire crew. In the event an "on-call" specialist is unavailable for duty, one of the shift supervisors or equipment operators will be qualified to assume these duties.

Based upon the foregoing, the minimum crew requirements and duties are outlined below:

Officer in Charge	- One required; assumes overall responsibility for the plant and crew.
Plant Superintendent	- One required; second in command and responsible for plant operation, maintenance and crew training.
Non-Commissioned Officer in Charge	- One required; qualified as a shift supervisor and in a specialty. Assists the plant superintendent and handles administrative duties and personnel.
Shift Supervisor	- Three required; responsible for plant operation and maintenance during regularly scheduled shifts and qualified in a specialty.
Control Room and Equipment Operator	- Three required; qualified to operate control room and plant equipment under direction of shift supervisor. In training for shift supervisor and qualified in a specialty.
Instrumentation - Technician	- One required; qualified to operate and maintain instruments and controls. Capable of acting as control room and equipment operator. Normally on day work but "on call" for emergencies.
Mechanical Maintenance	- Two required; qualified to install and maintain mechanical equipment and as control room and equipment operator. Normally on day work but "on call" for emergencies.
Electricians	- One required; qualified to install and maintain all plant electrical equipment. Also qualified as a control room and equipment operator.

Health Physics and Process Control Technician	-	One required; qualified to monitor and control radiation hazards, enforce safety regulations and control plant water chemistry. Also qualified as a control room and equipment operator.
Clerk	-	One required; assists NCOIC in general office duties and qualified as a typist.

16.0 LOGISTICS AND ERECTION

16.1 PLANT SHIPPING REQUIREMENTS

The equipment and material required for the complete PL-3 power plant including buildings and foundations can be packaged into a total of approximately 75 shipping modules, each one a single aircraft load. The total consists of 29 modules of foundations, buildings and support facilities, 24 modules for reactor and primary system and 22 modules for power conversion equipment and auxiliaries. A list of the shipping modules and their weights and cubages is given in Table 16.1.

The modules will be transported from the port of embarkation to McMurdo Sound by ship in two loads. They will arrive at McMurdo on about 15 December, 1963, and 15 December, 1964, the starting dates for the two construction periods. The foundations, buildings and support facilities will be shipped for the first period and will be escorted by two sub-contractor personnel. The remainder of the plant, consisting mainly of the reactor and primary system and the power conversion equipment and auxiliaries, will be shipped for the second period. Two contractor personnel will escort this shipment.

The plant will be airlifted from McMurdo Sound to Byrd Station by C-130-B Hercules aircraft. Twenty nine plane loads will be required during the first (1962-63) construction period and 46 will be required during the second (1963-64) period. Because of the large difference in the number of flights required during the two construction periods an effort will be made to procure and ship during 1962 some of the spare parts and spare equipment and operating supplies. The above totals do not include the shipping requirements for the emergency diesel fuel oil. Based on 10% outages over two years, 73,000 gallons of oil will be necessary and will require thirty C-130 flights during the two construction seasons.

The delivery schedules of material and equipment to Byrd Station will be optimized to minimize handling and storage at the construction site. Two or three flights per day will be required during most of the two construction periods to expedite erection and to insure startup of the plant on schedule.

16.2 CONSTRUCTION SCHEDULE, MANPOWER AND EQUIPMENT

The PL-3 plant will be constructed in two phases; 1) erection of the foundations, buildings and support facilities, and 2) installation of the reactor and other plant equipment. This work will be spread over the two construction seasons, 1962-63 and 1963-64. The plant construction will start on approximately December 15 each season and continue for about 60 days. Although some preliminary work

TABLE 16.1
SHIPPING MODULES

Contents	Number	Weight-lb	Cubage-ft^3
Foundations, Buildings and Support Facilities			
Foundations and Jacking	11	168,400	15,500
Buildings and Superstructure	11	224,200	20,363
Support Facilities	7	109,200	10,645
Sub-total	29	501,800 lb	46,508 ft^3
Reactor and Primary System			
Reactor Vessel	1	18,500	960
Lower Vapor Container (Reactor) Shielding and Vessel Support	1	19,500	1,730
Upper Vapor Container (Reactor) Reactor Vessel Head and Core Structure	1	17,400	650
Lower Vapor Container (Equip't) and Structure	1	19,500	1,730
Upper Vapor Container (Equip't) and Pressurizer	1	17,500	650
Steam Generator, Primary Pump and Piping	1	19,700	960
Spent Fuel Storage Tanks and Lead Shielding	2	39,700	2,940
Shield Water Tanks and Lead Shielding	2	38,500	2,940
Lead and Polyethelene Shielding	5	100,000	4,800
Fuel Shipping Casks (27,050 lb each)	4	108,200	1,150
Spare Parts and Equipment	2	40,000	1,720
Miscellaneous Structure and Equipment	3	60,000	2,880
Sub-total	24	498,500 lb	23,110 ft^3

TABLE 16.1 (CONT'D)

Contents	Number	Weight-lb	Cubage-ft^3
Power Conversion Equipment			
Turbine	1	15, 450	686
Generator	1	15, 000	210
Turbine Auxiliaries	1	16, 700	728
Condenser	1	20, 000	1, 792
Air Blast Coolers	2	39, 000	3, 840
Secondary Auxiliaries	1	19, 500	960
Control Console	1	13, 600	960
Switchgear	1	13, 100	960
Waste Processing	1	20, 000	1, 792
Chemical Laboratory and Maintenance and Decontamination Equipment	1	11, 800	1, 760
Auxiliary Power (2-250 kw units)	2	40, 000	864
Piping	2	31, 000	3, 840
Auxiliary Electrical	2	37, 450	3, 810
Secondary System Supplies	1	20, 000	400
Lead Shielding for Waste Disposal System	2	30, 500	640
Spare Parts, Tools and Misc.	1	20, 000	1, 280
Spare Components	1	20, 000	1, 280
Sub-total	22	383, 100 lb	25, 802 ft^3

may begin at Byrd Station prior to December 15, actual plant construction in each phase must await the arrival of materials and equipment by ship at McMurdo Sound and subsequent airlift to Byrd Station

The construction of the PL-3 snow tunnels will begin early in the 1962-63 season, so that when the first plane load of foundation material arrives erection of the plant may begin at once. Work on the foundations, buildings and support facilities will proceed through the first construction season and be essentially completed at the end of the season. Final work in this phase may be completed in the next season before the arrival of the reactor and plant equipment.

The second construction phase will begin with the arrival of the reactor and plant systems about December 15, 1963. As the modules arrive by air the equipment will be moved into the tunnels, erected on their foundations and interconnected. By the end of January 1964 the plant erection and installation will be completed and component checkout and system integrity and continuity tests started. All non-nuclear test will be completed, the reactor fueled and zero power operation attained by February 15.

The construction crew requirements can be sub-divided into the manpower needed for foundation, building and support facility erection during the first season and the personnel needed for reactor and plant installation during the second season. Table 16.2 lists the construction crew manpower requirement by crafts for the two seasons. The list does not consider tunnel excavation and arch erection.

TABLE 16.2
CONSTRUCTION CREW MANPOWER REQUIREMENTS

Craft	Number of Men	Months Required
1962-1963 Construction Season		
Survey Crew	3	1.5
Construction Crew Supervisors	4	3.0
Carpenters	10	3.0
Steelworkers	12	3.0
Laborers	16	3.0
Electricians	8	3.0
Plumbers	4	1.0
Millwrights	2	2.0
Equipment Operators	5	3.0
Sheetmetal Workers	2	2.0
Riggers	2	3.0
Sub-total	68	

TABLE 16.2 (CONT'D)

Craft	Number of Men	Months Required
1963-1964 Construction Season		
Survey Crew	3	2.5
Construction Crew Supervisors	4	2.5
Carpenters	4	2.5
Steelworkers	4	2.5
Laborers	16	2.5
Utility Men	10	2.5
Electricians	8	2.5
Pipefitters	8	2.5
Millwrights	4	2.5
Equipment Operators	5	2.0
Sheetmetal Workers	2	2.0
Riggers	2	2.0
Sub-total	70	

In addition to these personnel there will be contractor and sub-contractor personnel required. During the first season there will be two prime contractors and two foundations and buildings sub-contractor representatives at the site. Contractor and sub-contractor personnel required during the second season are listed in Table 16.3.

TABLE 16.3
1963-64 SEASON CONTRACTOR AND SUB-CONTRACTOR PERSONNEL REQUIREMENTS

	Number of Men	Months Required
Prime Contractor		
Site Manager	1	3.0
Construction Engineer	1	3.0
Electrical Engineer	1	1.0
Instrument Technician	1	1.5
Nuclear Engineer	1	1.0
PL-3 Operating Supt.	1	*
Shift Supervisor	3	*
Health Physicst	1	*
Chemist	1	*
Sub-Contractor		
Foundation and Buildings	2	3.0
Turbine	1	1.0
Instrumentation	2	2.0

* These men will arrive for startup and will remain for the first operating season.

There will be about $40, 000 worth of construction and rigging equipment and supplies required for the building and plant erection. This will include such items as power supplies, winches, hoists, hydraulic jacks, electric and gas welding equipment, saws, timber cribbing, and miscellaneous hand tools. This material will require about four aircraft flights from McMurdo Sound. The heavy construction equipment which should be available at Byrd Station is listed in Table 16. 4.

TABLE 16. 4
HEAVY CONSTRUCTION EQUIPMENT

Item	Number of Items
D-8 Tractor	1
Traxcavator with Lift Jacks	1
3-Ton Mobile Crane	1
10-Ton Sleds	6
20-Ton Sleds	2

17.0 COST INFORMATION

17.1 CAPITAL COSTS INCLUDING SPARE PARTS INVENTORY

Table 17.1 summarizes the cost of the major plant components and systems and presents the estimated cost of the equipment comprising the complete plant.

TABLE 17.1
CAPITAL COSTS

Item	Cost
Capital Equipment	
Reactor Vessel	$ 140,000
Steam Generator	90,000
Pressurizer	34,000
Primary Pump	65,000
Vapor Container (2)	200,000
Core Support Structure	20,000
Control Rod Drives	70,000
Shield and Refueling Tanks	50,000
Shipping Casks	65,000
Air Blast Coolers (2)	90,000
Turbine-Generator	235,000
Condenser	55,000
Secondary Auxiliaries	43,000
Turbine Auxiliaries	35,000
Waste Processing Equipment	50,000
Chemical Laboratory	27,000
Maintenance Facility	55,000
Diesel Generators (2)	78,000
Switchgear	43,000
Control Console	252,000
Radiation Monitoring System	33,000
*Reactor Core	330,000
**Miscellaneous Items	314,000
	$ 2,374,000
Support Facilities	696,000
	$ 3,070,000

* First core procurement and fabrication costs.

** Included in category are interconnecting piping and wiring health physics and personnel equipment, diesel fuel storage and miscellaneous primary equipment.

Spare Components

Primary Pump Motor & Rotating Assembly	$ 60,000
Steam Generator Tube Bundle	60,000
Control Rod Drive	10,000
Control Rod Basket & Rack	5,000
Nuclear Instrumentation	15,000
Second Reactor Core	226,000
	$ 376,000

Spare Parts

Fuel Oil - 10% Supply for 18 Months	$ 10,000
Primary Coolant Pump Spares	20,000
Reactor Vessel Stud Nuts and Bolts	6.000
Trip and Control Valves	10,000
Process Instrumentation	35,000
Nuclear Instrumentation	20,000
Control Console	15,000
Turbine and Generators	100,000
Diesel Generator	16,000
	$ 232,000

17.2 FUEL CYCLE COSTS

Fuel cycle cost were computed for the reference 7 x 7 Type 2 PWR core based on the following assumptions:

1. 2.1 yr core life at 0.8 plant factor(7.4 Mw; 15.54 (MWYRS)

2. Fabrication costs are based on the average of the charge for two complete cores and includes the following: fuel use charges, fuel losses, UF_6 to UO_2 conversion costs, scrap and reprocessing costs and technical liaison.

3. Current AEC price schedule for enriched uranium.

4. Chemical reprocessing plant charges @ $17,600 per day, $32 per Kg U for conversion to UF_6 and 1.3% losses.

5. Shipping charges for fresh fuel Schenectady to P.O.E. only.

6. Two spent fuel shipping casks @ 27,050 lb/cask, $1,200/ton shipping charge.

7. A use charge of 4-3/4% is applied only to the two full cores plus 10% spare elements for one core required within the reactor complex at all times.

The fuel cycle costs in dollars per year may be converted to mills per net electrical killowatt-hour by multiplying by 1.426×10^{-4}. This assumes 1.0 Mwe output at 0.8 plant factor, and neglects any credit for the export steam generated.

TABLE 17.2
SUMMARY OF FUEL CYCLE COSTS

Cost Item	2.1 Yr Core	$/Core	$/Yr
1. Fuel Burnup Cost			
Kg U Initial	24.17		
Kg U Final	17.55		
Initial Enrichment	0.93		
Final Enrichment	0.88		
$/kg U Initial	12,720		
$/kg U Final	12,000		
Initial U Value	$ 307,442		
Final U Value	$ 210,600		
Burnup Costs		$ 96,842	$ 46,115
2. Core Fabrication Cost		$ 262,500	$ 125,000
3. Reprocessing Costs			
Conversion to UF_6	$ 770		
Plant Costs	$ 52,800		
Reprocess Losses	$ 2,420		
Total Reprocessing Costs		$ 55,990	$ 26,661
4. Shipping Costs			
Fresh Fuel	$ 1,000		
Spent Fuel	$ 32,460		
Total Shipping Costs		$ 33,460	$ 15,933
FUEL CYCLE COSTS WITHOUT USE CHARGE		$ 448,792	$ 213,709
5. Fuel Use Charges			
In Reactor Core	$ 25,090		
Spare Core	$ 30,619		
10% Spare, 1 Core	$ 3,062		
Total Use Charges		$ 58,771	$ 27,986
TOTAL FUEL CYCLE COSTS		$ 507,563	$ 241,695

18.0 TRAINING PROGRAM

The scope of the training program for the PL-3 program includes the training of the plant operating crew and the supervisory members of the building construction group and the preparation of a training manual.

18.1 OPERATOR TRAINING

The PL-3 operator's training course is designed for personnel who have completed the academic and operations phases of the SM-1 Operator's Training Program. The schedule is based on the training requirements for a total of twenty government personnel. Fourteen members of this group will comprise the PL-3 operating crew. The other six trainees will receive parallel instruction to serve as reserve and to take part in the stateside and on-site assembly and initial operation and testing of the PL-3 plant.

All twenty trainees will be made available for training on January 1, 1963; the schedule is based on the assumption that all the operator trainees will be trained together and no presentation of make-up classes will be required.

It is further assumed that all the PL-3 operator trainees have received six months' operational experience at the SM-1, although it is not expected that they have as yet operated together as a crew. This qualification on the SM-1 reactor in both reactor operation and technical specialties eliminates the need for further preparatory training at any other reactor facilities and greatly simplifies training and qualification on the PL-3 reactor. However, emphasis must be given to the coordination of the group for function as an integrated crew.

The PL-3 operator training will begin with a brief period of operation at the SM-1 for the entire PL-3 crew, under Alco supervision. All trainees will then receive orientation training and instruction in the PL-3 plant systems, plant procedures, assembly, disassembly, site testing and plant building construction, including instruction by vendor's personnel. During stateside testing and operation, all trainees will receive practical training both in their specialty and in reactor operation. Detailed classroom and laboratory training will be given in each specialty prior to the practical training. The duration of this program, as outlined in Table 18.1 and Figure 18.1, will be approximately eight months.

TABLE 18.1
PL-3 OPERATOR'S TRAINING PROGRAM

Session	Training Description	Location	Instruction Hours
1	Integrated Crew Training	SM-1	40
2	Advance Orientation	SM-1	40
3	Test Site Orientation	Sch'dy	40
4	Plant Systems	Sch'dy	40
5	Plant Procedures	Sch'dy	40
6	Specialty Training	Sch'dy	
6A	Health Physics Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
6B	Chemistry Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
6C	Electrical Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
6D	Instrument Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
6E	Mechanical Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
	Review of all Specialties (10-15 hrs/each)		40
7	Assembly, Disassembly and On-Site Testing	Sch'dy	40
8	Plant Operation and Testing	Sch'dy	160
9	Plant Buildings	Sch'dy (J & M)	20
10	Vendor Training	Sch'dy (Vendors)	40

18.1.1 Session 1, Integrated Crew Training

Intensive one-week operation of the SM-1 will be scheduled for the PL-3 operating crew to function as an integrated crew. This session will be held on a three-shift basis, around-the-clock for six days, during which time the crew will take the SM-1 through planned shutdowns, cold startups, emergency shutdown, full-power runs, and dry-runs of maximum credible accident procedures. Army crew members will report to their military supervisors during this session; the sessions will be monitored by Alco instructors. Each trainee will participate in this session for a minimum of 40 hours.

18.1.2 Session 2, Advance Orientation

Preliminary orientation on the PL-3 will be given at the SM-1 facility immediately following Session 1. The objective of Session 2 will be familiarization of the trainees with the PL-3 plant by comparisons to the SM-1 plant. Conference discussions will be held to compare the operation of PL-3 with SM-1. Prints, photographs, and cut-away drawings will be used as training aids. This session will involve 40 hr of classroom lectures and discussions.

18.1.3 Session 3, Test Site Orientation

Prior to Session 3, the trainees will be given approximately one week in which to re-locate to Alco's Schenectady Plant. Session 3 will resume the training program with 40 hours of indoctrination at the stateside test facility for the PL-3, consisting or orientation and possibly using such training aids as a full-size mockup of the PL-3. The trainees will learn the physical layout of the plant and components prior to final erection. This session will be held eight hours per day for five days.

18.1.4 Session 4, Plant Systems

This session will consist of ten four-hour lectures held over a two-week period and will cover descriptions and design philosophy of all plant systems. Emphasis will be given to system differences from SM-1. Trainees will sketch all systems under the supervision of Alco instructors. Plant drawings and manufacturer's literature will be used as training aids.

18.1.5 Session 5, Plant Procedures

This session will follow Session 4 and will also consist of ten four-hour lectures over a two-week period. These conference-type training sessions will take place both in classroom and in the plant. Preliminary operating procedures and manufacturers' literature will be used as training aids. Trainees will dry-run operating procedures in the plant until they have proven their proficiency and familiarity.

18.1.6 Session 6, Specialty Training

Specialty training will be given to all operating trainees in their selected specialty. It is assumed that each man would attend only one specialty training session and the overall specialty review, with the exception of the process control technician specialists, who would attend both Session 6A, Health Physics, and 6B, Chemistry Procedures.

It is further assumed that each man has already qualified in his specialty on the SM-1 prior to joining the PL-3 training program. In addition to the operator's specialty training, he will receive a 15 to 20 hour review in all the other specialty subjects to enable him to fully understand all phases of the PL-3 operation. It is not the intent of this review to qualify the trainee to work unassisted in a specialty other than his own. However, in an emergency, any operator should be able to give limited assistance in any specialty under the direct supervision of a qualified man. This should be very advantageous at the remote Byrd Station location. The specialty review training will be taught by the qualified specialists and monitored by Alco instructors, who will not participate except to give guidance where necessary.

Session 6A, Health Physics Specialty

Ten four-hour lectures will be given to process control technician specialists over a two-week period, covering health physics procedures, plant health physics equipment, and the special techniques to be used at PL-3. All health physics equipment will be checked out by these specialists during stateside testing.

Session 6B, Chemistry Specialty

Forty hours of classroom lectures and laboratory work will be given to familiarize the process control technicians with those techniques necessary for control of water chemistry of both primary and secondary systems. Ten four-hour sessions will be held over a period of two weeks. The plant chemistry equipment will be used for training and the trainees will be required to handle water chemistry chores under Alco supervision during the stateside testing period. Radiochemistry procedures will be taught in the Alco laboratory. Session 6B will be divided into 20 hours of lectures and 20 hours of laboratory testing.

Session 6C, Electrical Specialty

Training in the electrical specialty will consist of ten four-hour lectures and conference meetings held over a two-week period covering all phases of electrical work. Electrical testing during stateside plant tests will provide additional practical training.

Session 6D, Instrument Specialty

Instrument specialists will receive their specialty training in 40 hours of lectures and on-the-job instrument training during plant testing. Session 6D lectures will cover the theory of operation of all plant instrumenation and control systems, with comparisons being made to the SM-1 systems. Manufacturers' literature will be the principal training aids. A minimum of 40 hours of instrument specialty training will be given in the plant on an informal basis during the testing period. This training will consist largely of lectures by the Alco instrument technician and engineer covering the operation and maintenance of instrument components and systems.

Session 6E, Mechanical Specialty

Forty hours of lectures and demonstrations will be given to mechanical specialist trainees covering mechanical maintenance of plant equipment. This training will include operation and maintenance of special handling tools, control rod drives, rotating plant equipment, etc. Plant equipment and mockups will be used during this session to provide practical training following classroom meetings. Practical training will be gained throughout plant startup and testing.

18.1.7 Session 7, Assembly, Disassembly and On-Site Testing

All operator trainees will attend 40 hours of conference-type training sessions covering procedures developed for assembly, disassembly, and on-site testing of the PL-3 plant. Preliminary assembly and disassembly instructions and on-site test procedures will be used as training aids. In addition, all crew members will witness the initial assembly and disassembly of the plant at Alco's Schenectady plant.

18.1.8 Session 8, Plant Operation and Testing

All operators will be assigned to shift crews for the purpose of receiving operating experience during initial stateside plant testing and operation with a non-nuclear heat source. A minimum of 160 hours of this on-the job training will be received as an integrated military unit within Alco's PL-3 operations group.

18.1.9 Session 9, Plant Buildings

Representatives of Jackson and Moreland will give 20 hours of classroom lectures covering construction of PL-3 plant buildings. This will be held at Alco's stateside test facility, and will prepare the operating crew for their participation in the Byrd Station PL-3 assembly.

18.1.10 Session 10, Vendor Training

One to two days of instruction will be given by technical representatives of vendors supplying major equipment such as the turbine-generator set, nuclear instrumentation, process instrumentation, and area radiation monitoring at Alco's stateside facility. This training will be given at opportune times during initial checkout of the equipment by the vendors' representatives.

18.2 CONSTRUCTION SUPERVISION TRAINING

The PL-3 training program also provides for the training of four supervisory personnel from the military organization responsible for initial erection of the plant buildings and installation of plant components. The training program for these key construction personnel is given in Table 18.2.

TABLE 18.2
PL-3 CONSTRUCTION SUPERVISOR'S TRAINING PROGRAM

Session	Training Description	Location	Instruction Hours
1	Plant familiarization and orientation	Sch'dy	20
2	Disassembly, Packing and Assembly	Sch'dy	40
3	General Building Construction	Boston (J & M)	40
4	Detailed Construction Methods	Plant (Vendor)	80

18.2.1 Session 1, Plant Familiarization & Orientation

The construction supervisory group will be given a general orientation session consisting of 20 hours of classroom lectures to familiarize them with the plant layout and principles of operation. This session will be directed to providing broad familiarization to serve as a sound basis for the detailed erection training to follow. Use will be made of all available training aids such as plant layout drawings, flow sheets, scale models, etc.

18.2.2 Session 2, Disassembly, Packing and Assembly

Session 1 will be followed by 40 hours of lectures and group discussions of the details of disassembly, packing, and assembly of the plant. The sequence of erection of the plant and buildings for the most expeditious scheduling of site testing will be emphasized.

18.2.3 Session 3, General Building Construction

Session 2 will be followed by 40 hours of lectures and group discussion by Jackson and Moreland at their Boston Office, covering erection of the PL-3 buildings.

18.2.4 Session 4, Detailed Construction Methods

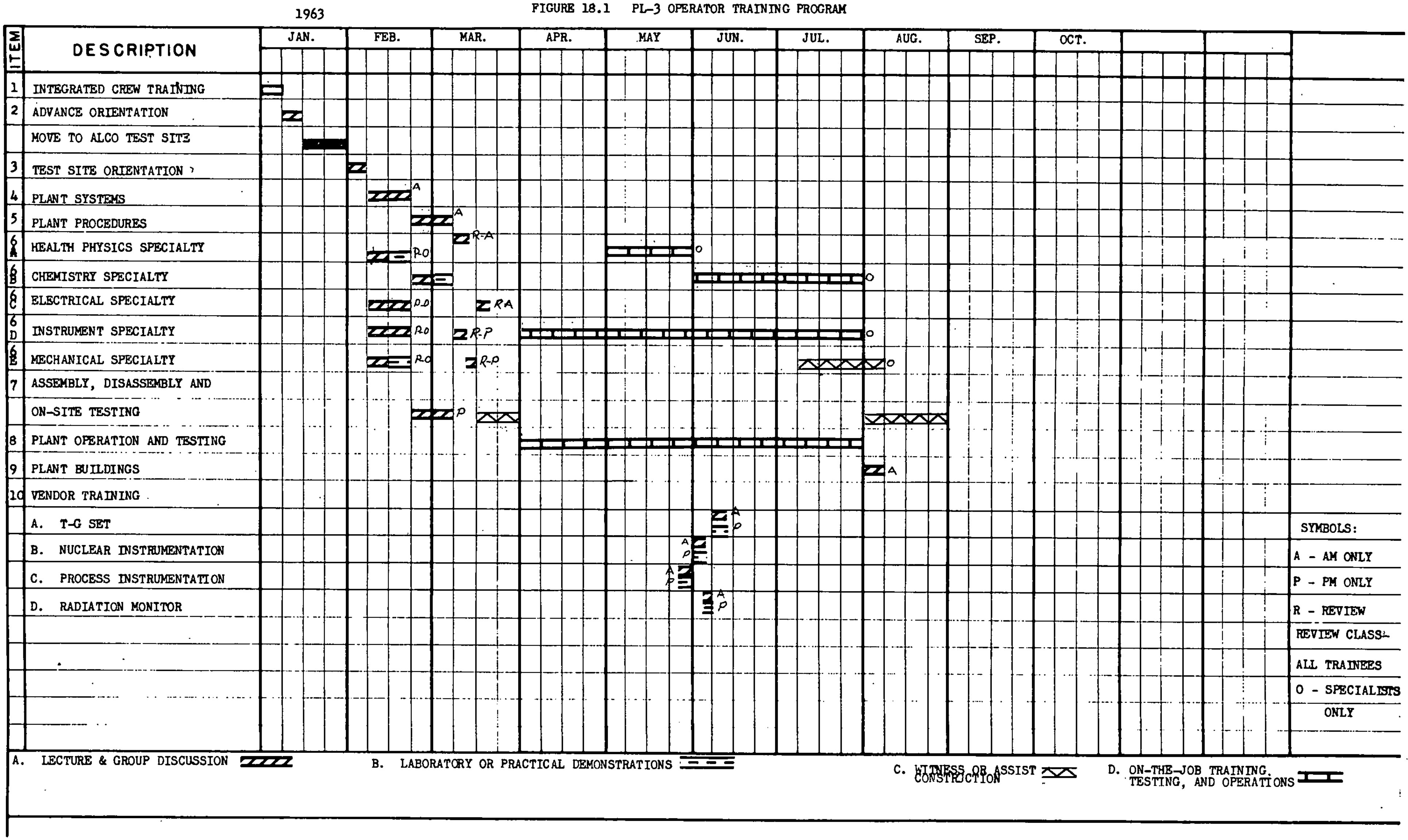
Eighty hours of conferences covering details of building construction will be held at the building vendor's facilities. All necessary information, instructions, and procedures to enable the construction group to effectively direct their work at the jobsite will be provided.

18.3 OPERATOR TRAINING MANUAL

An operator training manual will be developed which will contain all lesson plans and photo reductions of all sketches used in the training course. This manual will be bound in loose leaf folders and copies presented to all trainees for use as a study aid. Each trainee will keep his lecture notes and laboratory work notes in his manual which will be used, in part, to grade his performance during the training period.

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT
SCHENECTADY NEW YORK
PROJECT SCHEDULE

FIGURE 18.1 PL-3 OPERATOR TRAINING PROGRAM



APPENDIX A

DESIGN INFORMATION FOR ALTERNATE PWR CONCEPT

5 X 5 CORE - TYPE III FUEL ELEMENTS

1.0 INTRODUCTION

During Task 1 of the PL-3 contract, considerable attention was given to a small size core employing Type III[1] fuel elements. These elements are similar to the SM-2 type elements,[2] scheduled to be used in replacement cores for the SM-1, SM-1A and PM-2A reactors.[1][3] The SM-1, SM-1A and PM-2A reactors presently employ Type II fuel elements[1] as does the reference PL-3 design. Use of the Type III elements, which contain approximately 60 percent more fuel than the Type II elements, offers the advantage of either a considerable increase in core life for a given core size, or a significant reduction in core size for a given core lifetime. During Task 1, the potential advantages of using a minimum sized core seemed quite attractive from the standpoints of reducing the weight of the primary system and various mechanical aspects. A smaller pressure vessel could be utilized, resulting in a reduction of the weight of lead required for primary shielding. Only 4 or 5 control rods are needed, minimizing the number of vessel penetrations. In addition, problems associated with the storage and shipping of fuel elements would be less for a smaller core.

It was determined that a 5 x 5 array containing 25 elements, including 5, or possibly only 4, control rod fuel elements could meet the required core life of 2 years. Accordingly, a considerable amount of physics, thermal and hydraulic analysis was performed for this core. In addition, critical mockup experiments were performed for this core under Task 3.

Unfortunately, the requirement to survey many types of cores during Task 1 did not permit time to exploit fully all the potential non-nuclear advantages of the small sized core. From a nuclear standpoint, the core suffers in comparison to some of the larger cores sizes studied during Task 1,[4] which present greater potential in lifetime. During the course of Task 1, the thermal power level requirements were increased, reducing the already marginal lifetime of this core. In addition, preliminary studies indicated that an increase of the primary system operating temperature would be necessary and this would further decrease core life. These effects could be compensated to some degree by an increase in fuel loading or by using a thick stainless steel reflector; however, these modifications would require additional development effort.

The comparison of reactor concepts performed at the end of Task 1,[5] indicated no significant advantages of the 5 x 5 concept relative to the other designs considered. For this reason, in addition to its status of development as compared to the 45 element Type II core, the PL-3 preferred concept, the 5 x 5 core was dropped from further consideration for PL-3 application.

This appendix summarizes the physics, thermal and hydraulic analyses and experiments performed during Tasks 1 and 3 upon the 5 x 5 core containing 20 stationary fuel elements and 5 control rod fuel elements.

The calculations for this core were performed during an early phase of the evaluation program, before the thermal power requirements were determined. They were based on an operating thermal power level of 8.0 Mw, a core mean coolant temperature of 440°F, and the same coolant flow as in the 7 x 7 core, 2250 gpm.

Studies of the heat balance and power requirements for the 7 x 7 core, performed subsequent to termination of work on the 5 x 5 core, indicate a design thermal power of 9.3 Mw and a core mean coolant temperature of 470°F.

Detailed results are presented only for the initial assumptions for the 5 x 5 core, but the effects of increased power and temperature are indicated. In particular, nucleate boiling would occur at the indicated design power level of 9.3 Mw if the flow rate is unchanged. However, the occurrence of nucleate boiling does not represent a hazardous condition, nor is it precluded by the design criteria.

Calculations of core lifetime and fuel cycle costs are based on an average power level of 7.4 Mw, corresponding to an 0.8 plant factor opperation of the plant which, at design power, requires 9.3 Mw thermal power.

2.0 DESCRIPTION OF CORE AND VESSEL

During the study of the 5 x 5 Type III core, various combinations of core, reflector, and vessel configurations were investigated; a typical layout is shown in Dwg. AEL-711. This particular drawing shows a 5 x 5 core containing 4 control rods. A five rod core would be nearly identical except that the central stationary fuel element would be replaced by a control rod fuel element as in Fig. A.1. The drawing shows the core surrounded by a laminated steel reflector having a total thickness of 7-7/16 inches.

The reactor vessel shown is of low alloy ferrite steel and has an inside spherical radius of 25 inches. The control rod actuators are of the bottom-entry type, similar to those for the SM-1, SM-1A, and PM-2A reactors.

The reactor vessel primary coolant nozzles are both located below the core, with the inlet nozzle being located in the lower cylindrical portion of the vessel. The coolant flow from the inlet passes upward through the core, turns 180° at the top and passes between the outer reflector rings and the vessel wall, leaving through the exit nozzle.

Use of a laminated steel reflector is advantageous from the nuclear standpoint, since it reduces the critical mass of the core. However, the reflector must be relatively thick in order to be effective, resulting in mechanical design problems and limitations upon the minimum inside vessel dimensions. Therefore, since it would be desirable to eliminate the laminated steel reflector from the mechanical design standpoint, the nuclear, thermal and hydraulic analyses (described in Sections 4.0 and 5.0, respectively) were based primarily upon a pure water reflector.

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3.0 CRITICAL MOCKUP EXPERIMENTS

Critical mockup experiments were performed upon a 5 x 5 array of Type III fuel elements. These experiments were limited to the determination of certain important quantities needed to confirm the nuclear characteristics of the small-sized core. The measurements included control rod calibrations, stuck rod configurations, reflector reactivity worth, temperature coefficient and substitution effects.

3.1 EXPERIMENTAL ASSEMBLY

Descriptions of the core support assembly, the stationary and movable fuel elements, and the control rods employed, are described in detail in APAE-100.[6]

The core support assembly consists of a three-tiered stainless steel table located over the center of the reactor tank floor at the Critical Facility. Structural support and alignment of the assembly are assured by tie rods and spacers.

Reactor control is maintained by insertion and withdrawal of control rod assemblies which contain both fuel and box-type boron absorbers in a stainless steel basket. The control rod assemblies are driven by overhead drives and dropped by gravity upon a scram signal.

The mockup Type III stationary fuel elements contained 18 stainless steel UO_2 matrix fuel plates, each loaded with 46.3 gm U-235. Control rod fuel elements contained 16 similar fuel plates, each loaded with 42.2 gm U-235. The individual fuel plates are composed of 26.5 w/o uranium oxide held in a 0.030 in. thick matrix of stainless steel and clad with 0.005 in. of stainless steel.

The two cores selected for the experiment consisted of a 5 x 5 lattice of mockup fuel elements with either 4 or 5 control rod assemblies in an open array as shown in Fig. A.1. The fuel cell was a square, 2.94 in. on a side. The core size was 14.7 in. square, giving an equivalent core diameter of 16.6 inches. The core metal/water volume ratio was approximately 0.352 and the active fuel height was 22 inches.

3.2 EXPERIMENTAL MEASUREMENTS

All reactivity measurements were taken by measuring the difference in control rod position to achieve criticality between some reference condition and the condition under investigation. The reactivity worth of moving the single rod or bank was established by a series of control rod calibrations obtained using the period method at various control rod or bank locations. The total worth of a reactivity measurement was obtained from the difference in the critical positions of a calibrated control rod or bank.

3.2.1 Control Rod Calibrations

The initial core loading was 20 kg of U-235 and approximately 33.6 grams of B-10 in the form of uniformly distributed mylar tape impregnated with boron. The boron was removed in approximately 6 gram increments, in order to drive the control rod bank and obtain the differential worth curves for both the 4 and 5 rod systems. All control rod calibrations were made by period measurements. After a critical position was established, the rod or rods to be calibrated were raised above the critical position to obtain about a 30 sec reactor period. The period is related to the reactivity change divided by the change in rod positions, which yields the differential rod worth at a point midway between the critical position and the period position.

The differential worth of the total control rod bank was measured as a function of position for the 4 and 5 control rod bank configuration with the initial core loading of 20 kg of U-235 and 33.6 gms of B-10. These results are presented in Figs. A.2 and A.3. Measurements were also taken for the critical bank position as a function of gms of B-10 in the core and these results are presented in Fig. A.4 and A.5 for the 4 and 5 rod bank, respectively.

The differential rod worth curve for control rod A is presented in Fig. A.6. These measurements were performed for the core with a loading of 9.526 gms of B-10, surrounded by a 2.575 in. thick steel reflector.

3.2.2 Temperature Coefficient

Immersion heaters and a mixer were used in the reactor tank to increase the moderator temperature from room temperature, 68°F, to 115°F. The change in reactivity associated with the change in temperature was determined by the change in critical position of a calibrated control rod.

The average core temperature is plotted as a function of the critical position of calibrated control rod A in Fig. A.7. These measurements were made with a B-10 loading of 9.526 grams and a 2.575 in. thick stainless steel reflector. Table A.1 shows the temperature coefficient of this core configuration for temperatures ranging from 61°F to 115°F.

3.2.3 Reflector Worth

The relative worth of 3 solid stainless steel reflectors, 1/2, 1, and 2.5 in. thick, with an essentially infinite water reflector, were evaluated by observing the change in reactivity for the four configurations using a calibrated control rod bank.

Table A.2 shows the worth of the stainless steel reflectors relative to an essentially infinite water reflector. It can be seen that replacing the water by a solid 2.575 in. stainless steel reflector achieved a positive reactivity gain of 110.82 cents. However, the thinner stainless steel reflectors decreased the core reactivity compared to the water reflected core.

TABLE A.1
TEMPERATURE COEFFICIENT MEASUREMENTS
5 x 5 CORE PL-3 CRITICAL MOCKUP - 5 ROD BANK

2.575 In. Stainless Steel Reflector - 9.526 Grams B-10

Temperature °F	Reactivity Cents/°F
61.3	0.000
70.0	-0.261
79.6	-0.414
81.8	-0.259
83.5	-0.261
86.9	-0.536
89.1	-0.500
91.8	-0.614
93.2	-0.833
95.0	-1.02
97.8	-0.892
100.5	-1.15
102.4	-0.939
104.3	-1.07
107.2	-0.856
110.3	-0.978
113.0	-0.970
115.0	-0.928

TABLE A.2
REFLECTOR WORTH MEASUREMENTS
5 x 5 CORE PL-3 CRITICAL MOCKUP

Reflector	Reactivity Change Cents
Water	0
2.575 in. SS	+110.82
1.030 in. SS	-125.97
0.515 in. SS	-183.25

3.2.4 Substitution Effect

A four rod critical bank position was established with the center fuel element position occupied by a control rod fuel element. A stationary fuel element was then substituted for the central control rod fuel element and a four rod critical bank position was again established. The difference in reactivity attributed to the two different calibrated bank positions determined the substitution effect.

The reactivity of the center stationary fuel element, position No. 33, is greater than the central control rod fuel element by 141.46 cents. This measurement was made with a core loaded with 9.526 gms of B-10 and an essentially infinite water reflector.

3.2.5 Stuck Rod Measurements

Criticality was determined for various rod configurations with a 9.526 gm B-10 core loading, an essentially infinite water reflector and a moderator temperature of 68°F. The results of these experiments are shown in Table A.3. These experiments were repeated with all of the boron removed and the results are contained in Table A.4.

TABLE A.3
STUCK ROD MEASUREMENTS - 9.526 GM B-10
5 x 5 CORE PL-3 CRITICAL MOCKUP

Configuration Number	Rod(s) Fully Inserted	Rod(s) Fully Withdrawn	Critical Rod(s)	Critical Position
1	ABCD		E	Subcritical
2	ABDF		C	Subcritical
3	ABD	F	C	Subcritical
4	ABC	F	D	Subcritical
5	ACD	F	B	Subcritical
6	AD	CE	B	5.251 in.
7	AB	CE	D	3.873 in.
8	ABD	C	E	Subcritical
9	E	C	ABD	7.115 in.
10	E		ABCD	7.742 in.
11	C		ABDE	9.477 in.
12*	DE	A	B	1.240 in.
13*		A	BDE	0.340 in.

* Stationary element in center core position No. 33.

TABLE A. 4
STUCK ROD MEASUREMENTS - NO B-10
5 x 5 CORE PL-3 CRITICAL MOCKUP

Configuration Number	Rod(s) Fully Inserted	Rod(s) Fully Withdrawn	Critical Rod(s)	Critical Position
1	ABCD		E	Subcritical
2	ABDE		C	Subcritical
3	ABD	E	C	6. 990 in.
4	ABC	E	D	7. 258 in.
5	ACD	E	B	12. 463 in.
6	AD	CE	B	Supercritical
7	AB	CE	D	Supercritical
8	ABD	C	E	10. 570 in.
9	E	C	ABD	5. 911 in.
10	E		ABCD	5. 745 in.
11	C		ABDE	6. 710 in.

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4.0 NUCLEAR ANALYSIS

The nuclear analysis of the 5 x 5 Type 3 element core was based upon the calculational techniques used successfully on other parallel plate PWR cores. The methods of analysis are described with regard to the 7 x 7 core in Section 7.0 of the text. The analysis included the following items:

1. Core reactivity and lifetime as a function of boron loading.

2. Control rod worth and effects of a stuck rod.

3. Power distributions and power peaking factors.

4. Integrated fast flux on the pressure vessel.

The geometrical and nuclear data for the 5 x 5 core are presented in Tables A.5 and A.6.

4.1 CORE REACTIVITY AND LIFETIME

A series of calculations were performed to determine the initial core reactivity and the change in reactivity vs core burnup, as a function of the initial boron loading.

The reactivity variation with burnup is indicated in Fig. A.8 for initial loadings of 0, 10, and 15 gms of B-10. In each case, the maximum reactivity encountered over life occurs at beginning of life. The average worth of the boron at beginning of life is 0.44% $\Delta\rho$ per gram of B-10. The difference in core lifetime between cores containing no boron and 15 gms of B-10 is 1.2 MWYR.

The core with a loading of 10 gms B-10 was selected as the reference 5 x 5 core. The lifetime of this core operating at 440°F and 7.4 Mw average power is about 16 MWYR. Since only survey type calculations were performed, various factors which can affect the core lifetime must be considered. The factors which will tend to increase core lifetime are:

1. Stationary center element in place of central control rod fuel element, i.e., operate with a 4 rod bank instead of 5 rods.

2. Increased UO_2 weight percent in meat.

3. Reduced average power level from 7.4 Mw.

4. Use of a stainless steel reflector with thickness greater than 2 inches.

TABLE A.5
5 x 5 CORE DESCRIPTION

General Description

Configuration: 5 x 5 Array of Type 3 fuel elements.

Fuel: UO_2-SS matrix, fully enriched, SS clad.

Burnable poison: ZrB_2 in SS-UO_2 matrix.

Absorber: Eu_2O_3-SS matrix, SS clad.

Structural material: Stainless steel.

Moderator: Water

Number of fuel elements:	
Stationary	20
Movable	5
Number of absorbers	5
Initial loadings:	
U-235	19.95 Kg
U-238	1.50 Kg
B-10	10.0 gm
Total number of cells	25
Element loading U-235:	
Stationary	833.4 gm
Movable	655.58 gm
Element loading B-10:	
Stationary	0.41 gm
Movable	0.36 gm
Number of plates per element:	
Stationary	18
Movable	16
Plate thickness	0.040 in.
Clad thickness	0.005 in.
Meat thickness	0.030 in.

TABLE A.5 (CONT'D)

Meat width:	
Stationary	2.65 in.
Movable	2.40 in.
Water gap between plates	0.123 in.
Cell size	2.9375 in. x 2.9375 in.
Active core height	22.0 in.
Effective core diameter	16.6 in.
Core volume	77.7 liters
Initial heat transfer area	319.6 ft^2
Average metal/water volume ratio	0.36

Fuel Element

Type: Rectangular assembly of flat fuel plates welded into side plates with attached end boxes.

Meat material: Spherical UO_2 (25%), ZrB_2 and SS.

Clad material: Type 347 SS.

Fuel element active length:	
Stationary	22.0 in.
Movable	21.5 in.
Fuel element cross section:	
Stationary	2.874 in. x 2.863 in.
Movable	2.656 in. x 2.619 in.

Absorber Element

Type: Box assembly of four flat plates.

Meat material: Eu_2O_3 (32.74%) and SS.

Clad material: Stainless steel.

Active length	20.81 in.
Active width	2.20 in.
Active thickness	0.096 in.

TABLE A. 6
5 x 5 CORE NUCLEAR DATA

Average power level	7.4 Mw
Average moderator temperature	440°F
Average moderator pressure	1200 psia
Core life	2 yr.
Maximum reactivity:	
At 68°F (clean)	14.9%
At 440°F (clean)	11.9%
At 440°F (eq. xenon)	10.0%
At 440°F (max. xenon)	9.5%
Control bank worth at 68°F:	
5 rods	32.7% ρ
4 rods	27.3% ρ
Shutdown margin, one rod fully withdrawn:	
At 68°F	12.4% ρ
Critical control bank position:	
At 68°F (clean)	6.2 in.
At 440°F (clean)	8.5 in.
At 440°F (eq. xenon)	9.7 in.
At 440°F (max xenon)	10.0 in.
Temperature coefficient:	
At 440°F	$-3.1 \times 10^{-4}\% \Delta\rho/^{\circ}F$
At 68°F	$-0.1 \times 10^{-4}\% \Delta\rho/^{\circ}F$
Average thermal flux, hot clean core	1.1×10^{13} neuts/cm^2 sec
Fuel burnup (U-235) at end of life:	
Average	~37%
Maximum	~90%
Boron burnup at end of life:	
Average	~95%
Maximum	~100%
Max-to-Avg core power at 440°F:	
Gross radial	1.54
Local radial	1.95
Gross axial	2.17

Factors which will tend to reduce the core lifetime are more probable and of greater magnitude. These include:

1. Flux suppressors at the bottom of the stationary fuel elements.
2. Use of an all water reflector with a half inch stainless steel skirt around the core.
3. Increased mean core operating temperature from 440°F to 470°F.
4. Increased average power level from 7.4 Mw.
5. Reduced active plate width from 2.65 inches.
6. Reduced weight percent of UO_2 in the matrix.

On the basis of these preliminary calculations, it seems probable that a core lifetime of at least 14.8 MWYR, based on 2 yr operation at 7.4 Mw average power, is attainable.

4.2 CONTROL ROD WORTH

The worth of the 4 and 5 control rod bank was calculated using the PDQ-2 code with control rod cross-sections derived from blackness theory.[7]

The five rod bank is calculated to be worth 32.65% $\Delta\rho$. The four rod bank is worth 27.30% $\Delta\rho$, which means that the central control rod is worth 5.35% $\Delta\rho$. Since the maximum core reactivity is about 20% $\Delta\rho$ at 68°F with no boron, there should be a considerable shutdown margin even with the central rod stuck full out. This calculation was verified in the critical experiment program, where rod C was fully withdrawn and rods A, B and D were kept full in. Rod E had to be withdrawn 10.57 in. to achieve criticality even with no boron in the core. Therefore, it can be concluded that the core can be safely shutdown in the event of a rod stuck full out.

4.3 POWER DISTRIBUTIONS

Due to the absence of a fission foil mapping of the core in the experimental program, and due to certain inaccuracies in the analytical model in predicting local power peaking at the corners of the fuel elements,[7] only a preliminary estimate of the power distributions can be obtained.

Table A.7 shows the relative radial power generation for each typical fuel element and the local power peaking factors within the elements, and Table A.8 lists the axial power distribution, both for the case of the 5 rod bank. The large worth of the control rods causes the power in the upper section of the core to be greatly depressed, resulting in a high power peaking factor for the core. However, as in previous analyses,[7] the calculations tend to overestimate this axial peaking.

Radial and axial power distributions for the case of a 4 rod bank are given in Tables A. 9 and A. 10 respectively.

4.4 FAST NEUTRON EXPOSURE OF VESSEL

The procedures outlined in Section 7. 6 were used to calculate the integrated fast neutron fluxes external to the 5 x 5 core. The results of the calculations are shown in Fig. A. 9 which presents the integrated fast neutron flux for neutrons with energies greater than 1 Mev in the axial plane of maximum exposure. The results shown are from a multigroup P1MG-2 calculation modified by an axial peak-to-average factor calculated by the PDQ-2 (r, z) code. The accumulated nvt at the surface of the 50 in. inside diameter pressure vessel is 2.8×10^{18} neut/cm^2 for 20 yr operation at 0. 8 plant factor.

TABLE A. 7
5 x 5 CORE RADIAL POWER DISTRIBUTION - 5 ROD BANK
(T = 440°F, No Xenon)

Element	No.	Q ($\overline{\Delta T}$) Avg. Power in Element	Q (ΔT) Avg. Power in Hottest Plate of Ele.	Q ($\Delta\theta$) Max. Power in Element
33 CR	1	1.35	1.69	1.89
34	2	1.22	1.73	1.92
35	2	0.76	1.23	1.42
43	2	1.25	1.57	1.91
44 CR	4	1.11	1.54	1.84
45	4	0.74	1.25	1.48
53	2	0.84	0.96	1.38
54	4	0.78	0.96	1.38
55	4	0.48	0.79	1.12

TABLE A. 8
5 x 5 CORE - 5 ROD BANK
NORMALIZED AXIAL POWER DISTRIBUTION
(T = 440°F, No Xenon)

Inches from Bottom of Core	P (Z)	Inches from Bottom of Core	P (Z)
		11	0. 65
0	4. 64	12	0. 52
1	1. 87	13	0. 41
2	1. 90	14	0. 33
3	2. 05	15	0. 26
4	2. 11	16	0. 20
5	2. 10	17	0. 16
6	2. 00	18	0. 12
7	1. 81	19	0. 10
8	1. 52	20	0. 07
9	1. 06	21	0. 05
10	0. 82	22	0. 06

NOTE: 5 rod bank critical at 8. 5 in. from bottom of core.

TABLE A. 9
5 x 5 CORE RADIAL POWER DISTRIBUTION - 4 ROD BANK
(T = 440°F, No Xenon)

Element	No.	Q ($\overline{\Delta T}$) Avg. Power In Element	Q (Δ T) Avg. Power In Hot-test Plate of Element	Q ($\Delta\ \theta$) Max. Power In Element
33	1	1. 34	1. 54	1. 76
34	2	1. 22	1. 54	1. 76
35	2	0. 78	1. 24	1. 42
43	2	1. 27	1. 55	1. 84
44	4	1. 16	1. 54	1. 80
45	4	0. 75	1. 26	1. 48
53	2	0. 84	0. 96	1. 38
54	4	0. 77	0. 96	1. 38
55	4	0. 53	0. 77	1. 12

TABLE A.10
5 x 5 CORE - 4 ROD BANK
NORMALIZED AXIAL POWER DISTRIBUTION
(T = 440°F, No Xenon)

Inches from Bottom of Core	P (Z)	Inches from Bottom of Core	P (Z)
0	4.05	12	0.64
1	1.70	13	0.55
2	1.75	14	0.47
3	1.90	15	0.40
4	1.98	16	0.33
5	1.99	17	0.28
6	1.92	18	0.23
7	1.76	19	0.19
8	1.34	20	0.15
9	1.02	21	0.11
10	0.87	22	0.12
11	0.75		

NOTE: 4 rod bank critical at 7.7 in. from bottom of core.

5.0 THERMAL AND HYDRAULIC ANALYSIS

Thermal and hydraulic analyses performed for the 5 x 5 Type III core were based upon the same analytical techniques described for the 7 x 7 core in Section 6.0 of this report. The following sections describe the analyses performed and the results obtained. A summary of thermal and hydraulic data for the 5 x 5 core, both for 4 and 5 rod bank operation, is given in Table A. 11. The table is based on the initially assumed design power level of 8.0 Mw.

TABLE A. 11

THERMAL AND HYDRAULIC - DATA SUMMARY

5 x 5 CORE - TYPE 3 FUEL ELEMENTS

	Core with 5 rod bank	Core with 4 rod bank
General		
Core thermal power, Mw	8.0	8.0
Primary system pressure, psia	1,200	1,200
Coolant flow rate, gpm	2,250	2,250
Number of passes thru core	1	1
Heat Flux Data		
Initial heat transfer area, ft^2	319.6	326.5
Core average heat flux, Btu/hr ft^2	85,423	83,631
Core max. heat flux, Btu/hr ft^2		
Spike	761,970	623,050
Peak axial	347,670	306,090
Max.-to-avg. core power, Btu/hr ft^2		
Spike	8.92	7.45
Peak axial	4.07	3.66
Minimum DNB ratio	2.49	2.91
Core Temperature Data		
Core inlet temperature, ^{o}F	456	456
Core mean temperature, ^{o}F	470	470
Core average outlet temperature, ^{o}F	484	484
Max. plate surface temperature, ^{o}F	579	578
Max. fuel centerline temperature, ^{o}F	641	629
Hydraulic Data		
Core coolant flow area, ft^2	1.214	1.220
Flow channel thickness, in.	.123	.123
Core average coolant velocity, ft/sec	5.6	5.6

5.1 STEADY STATE ANALYSIS

The steady state thermal analysis was performed using the STDY-3 thermal analysis code. A core having both 5 control rods and 4 rods was investigated for a core power of 8 Mw and for a core temperature rise of 28 °F.

5.1.1 Power Distribution

The core is symmetrical with either rod configuration; therefore only one quadrant has been analyzed. The radial power distributions are given in Tables A.7 and A.9; the axial power distributions are shown in Figures A.10 and A.11 and described in Tables A.8 and A.10.

5.1.2 Mechanical Hot Channel Factors

Section 6.0 gives the formulation of the mechanical hot channel factors. The following values are used in analyzing the Type III element. These values apply to both rod configurations.

Factor	Average Condition	Local Condition
Plate length	1.0233	1.0233
Homogeniety	1.005	1.025
Clad thickness	1.007	1.012

5.1.3 Fuel Plate Spacing

Because the 40 mil thick fuel plate element specified for the 5 x 5 core is welded, close plate spacing tolerances are specified. In addition the thick plate will have a small ripple amplitude. Table A.12 shows a comparison of plate spacing factors as used in the steady state analysis of the 5 x 5 and the 7 x 7 core.

TABLE A.12

PLATE SPACING FACTORS

Plate thickness mils	Spacing - in. min.	avg.	max.	Tolerance
30	.128	.133	.172	± .008 avg. ± .013 local
40	.119	.123	.136	± .004 avg. ± .008 local

5.1.4 Average Core Heat Flux

The following table gives a comparison of the heat fluxes and power ratios for the 5 x 5 cores and the reference 7 x 7 core, both assuming an 8.0 Mw power level.

TABLE A. 13

COMPARISON OF CORE HEAT FLUX

Core	Power Mw	Heat Transfer Area ft^2	Average Core Heat Flux Btu/hr ft^2	Max. to Avg. Power Ratio Spike	Max. to Avg. Power Ratio Peak Axial
7 x 7 reference	8	560.1	48,753	5.25	3.22
5 x 5 5 rod bank	8	319.6	85,423	8.92	4.07
5 x 5 4 rod bank	8	326.5	83,631	7.45	3.66

5.1.5 Results of Steady State Analysis-Core with 5 Rods

The steady state thermal analysis of this core was made for average core temperatures of 440°F and 470°F and a core temperature rise of 28°F. Table A. 14 shows a summary of the results of the analysis.

The minimum DNB ratios quoted are for the inlet spike. Again a 2.0 minimum DNB ratio criterion for steady state operation is used. Element 44 exhibits the lowest DNB ratio.

No nucleate boiling is indicated for the core operating at 440°F, but at 470°F almost half the elements have some nucleate boiling. Nucleate boiling is exhibited at the design power of 9.3 Mw for 440°F operation.

5.1.6 Results of Steady State Analysis-Core with 4 Rods

Table A. 15 shows the results of the steady state analysis of the 5 x 5 core with 4 rods at 8 Mw core power and at 470°F core mean temperature. Again element 44 has the lowest DNB ratio.

The analysis indicates no nucleate boiling present in the core. This is to be expected because of the more uniform power distribution with this rod configuration.

5.1.7 Effect of Increasing Design Power Level to 9.3 Mw

The design power level is now expected to be about 9.3 Mw, as for the 7 x 7 core, rather than the 8 Mw assumed for initial calculations on the 5 x 5 core. The minimum DNB ratio that will occur at this new power level can be obtained from values calculated at 8 Mw by direct ratio:

$$\text{DNB ratio} = \frac{\text{DNB heat flux}}{\text{Local heat flux}}$$

$$\text{Local heat flux} \propto \text{Core power}$$

The minimum DNB ratios at 9.3 Mw are therefore:

Core	Element	T mean	DNB Ratio
5 x 5- 5 rods	44	440°F	2.26
5 x 5- 5 rods	44	470°F	2.14
5 x 5- 4 rods	44	470°F	2.50

At 9.3 Mw thermal power the minimum DNB ratios for element 44 are above the minimum criterion for thermal safety of 2.0

Nucleate boiling will be developed at 9.3 Mw in the 5 x 5 core with either rod configuration.

5.2 TRANSIENT LOSS OF FLOW ANALYSIS-CORE WITH 5 RODS

A loss of flow analysis was done for element 34 using the ART-02 transient analysis code. The analysis was done at a power level of 8 Mw and a core mean temperature of 470°F. For conservatism, no rod scram action was assumed.

The flow coastdown is given by:

$$\frac{G}{G_0} = \left(\frac{1}{1 + bt}\right)^{1.25}$$

where b was taken as 2.2 sec^{-1}.

Figure A.12 shows a plot of flow and power coastdown as a function of time after loss of flow.

Figure A.13 shows a plot of minimum DNB ratio during the loss of flow accident. The curve is stopped at 3 sec at which time the hot channel flow becomes less than 0.1×10^6 lb/hr ft^2, the value of the lower limit for the burnout correlations. At .05 seconds a minimum DNB ratio of 2.74 is obtained. This is only slightly lower than the steady state value of 2.75.

5.3 CONCLUSIONS

At 8 Mw the 5 x 5 core will operate without nucleate boiling at 470 °F, with either 5 rods or 4 rods. At a design power of 9.3 Mw both core configurations would have some nucleate boiling.

Both cores will be safe during a loss of flow transient without scramming the control rods.

TABLE A.14
STEADY STATE THERMAL ANALYSIS
5 x 5 CORE-5 ROD BANK

POWER-8 Mw: AVERAGE CORE TEMPERATURE-440 DEG. F

Element	G_{NC}	G_{HC}	To_{NC}	To_{HC}	$TsMAX_{HC}$	$TmMAX_{HC}$	DNB Ratio Min.	Nucleate Boiling
33	1.404	1.236	452.93	466.74	559.71	597.53	3.785	No
34	1.268	1.116	454.37	474.52	578.98	640.85	2.910	No
35	.79	.6952	454.46	481.33	577.85	623.46	3.950	No
43	1.23	1.082	455.92	471.56	578.65	640.23	2.930	No
44	1.154	1.016	452.99	471.17	578.59	637.85	2.620	No
45	.769	.6767	454.51	483.63	577.90	625.54	3.780	No
53	.874	.7691	454.30	465.11	577.07	621.52	4.060	No
54	.811	.7137	454.42	468.09	577.07	621.52	4.060	No
55	.499	.4666	454.17	478.97	576.52	612.54	5.000	No
POWER-8 Mw: AVERAGE CORE TEMPERATURE - 470°F								
33	1.365	1.201	483.06	496.83	578.90	638.16	2.490	No
34	1.234	1.086	484.47	504.52	578.98	640.85	2.750	Yes
35	.7690	.6767	484.56	511.67	577.85	623.46	3.720	Yes
43	1.264	1.112	484.45	499.23	578.65	640.23	2.775	No
44	1.123	.9882	483.10	501.20	578.59	637.85	2.485	Yes
45	.7480	.6582	484.62	513.59	577.90	625.54	3.570	Yes
53	.8500	.7480	484.42	495.19	577.07	621.52	3.840	No
54	.7890	.6943	484.53	498.15	577.07	621.52	3.840	No
55	.4860	.4544	484.25	508.91	576.52	612.54	4.725	No

G_{NC} - Nominal Channel Flow-10^6 lb/hr ft^2

G_{HC} - Hot Channel Flow-10^6 lb/hr ft^2

To_{NC} - Nominal Channel Bulk Outlet Temperature-°F

To_{HC} - Hot Channel Bulk Outlet Temperature-°F

$TsMAX_{HC}$ - Hot Channel Max. Plate Surface Temp-°F

$TmMAX_{HC}$ - Hot Channel Max. Temperature-°F

TABLE A.15
STEADY STATE THERMAL ANALYSIS
5 x 5 CORE-4 ROD BANK

POWER 8 Mw: CORE MEAN TEMPERATURE - 470 DEG. F

Element	G_{NC}	G_{HC}	To_{NC}	To_{HC}	$TsMAX_{HC}$	$TmMAX_{HC}$	DNB Ratio Min.	Nucleate Boiling
33	1.327	1.168	484.47	495.58	578.11	627.09	3.490	No
34	1.208	1.063	484.40	499.38	578.11	627.09	3.490	No
35	.7720	.6794	484.46	510.19	577.42	617.11	4.300	No
43	1.258	1.107	484.33	497.94	578.13	629.37	3.330	No
44	1.149	1.011	479.75	494.06	578.11	628.35	2.910	No
45	.7430	.6538	484.49	513.03	577.47	618.66	4.140	No
53	.8310	.6856	484.38	497.68	576.65	615.08	4.440	No
54	.7620	.6706	484.49	498.59	576.65	615.08	4.440	No
55	.5250	.4620	484.58	505.71	576.05	607.20	5.475	No

6.0 FUEL CYCLE COSTS

The fuel cycle costs reported in Table A.16 for the 5 x 5 core, using Type III fuel elements, were based on the following assumptions:

1. 2 yr core life at 0.8 plant factor (7.4 Mw average; 14.8 MWYR).
2. Fabrication costs are based on the average of the charge for two complete cores and include the following: fuel use charges, fuel losses, UF_6 to UO_2 conversion costs, scrap and reprocessing costs and technical liaison.
3. Current AEC price schedule for enriched uranium.
4. Chemical reprocessing plant charges @ \$17,600 per day, \$32 per Kg U for conversion to UF_6 and 1.3% losses.
5. Shipping charges for fresh fuel, Schenectady to P.O.E. only.
6. Two spent fuel shipping casks @ 22,600 lb/cask, \$1,200/ton shipping charge.
7. A use charge of 4 3/4% is applied only to the two full cores plus 10% spare elements for one core required within the reactor complex at all times.

The fuel cycle costs in dollars per year may be converted to mills per net electrical killowatt-hour by multiplying by 1.426×10^{-4}. This assumes 1.0 Mw electrical output at 0.8 plant factor, and neglects any credit for the export steam generated.

TABLE A.16
SUMMARY OF FUEL CYCLE COSTS
5x 5 CORE - TYPE 3 FUEL ELEMENTS

Cost Item	2.0 Yr Core	$/Core	$/Yr
1. Fuel Burnup Cost			
Kg U Initial	$ 21.45		
Kg U Final	14.91		
Initial Enrichment	0.93		
Final Enrichment	0.84		
$/kg U Initial	12,720		
$/kg U Final	11,300		
Initial U Value	272,844		
Final U Value	168,483		
Burnup Cost		$ 104,361	$ 52,180
2. Core Fabrication Cost		210,000	105,000
3. Reprocessing Costs			
Conversion to UF_6	686		
Plant Costs	52,800		
Reprocessing Losses	2,287		
Total Reprocessing Costs		55,773	27,886
4. Shipping Costs			
Fresh Fuel	1,000		
Spent Fuel	27,120		
Total Shipping Costs		28,120	14,060
FUEL CYCLE COSTS WITHOUT USE CHARGE		398,254	199,127
5. Fuel Use Charges			
In reactor core	19,894		
Spare core	24,556		
10% spares for 1 core	2,456		
Total Use Charges		46,906	23,453
TOTAL FUEL CYCLE COSTS		$ 445,160	$ 222,580

7.0 REFERENCES

1. Gallagher, J.D., "Ten Year Core Development Program for AEC-Army Rectors," AP Note 355, June 5, 1961.

2. Hoover, H.L., Project Engineer, "SM-2 Core and Vessel Design Analysis," APAE No. 69, Volume 3, March 8, 1961.

3. Dixon, M.H., Project Engineer, "Work Program; Engineering Support and Development of Army Pressurized Water Reactor Power Plants," AP Note 378, September 6, 1961.

4. Humphries, G.E., Project Engineer, "PL-3 Conceptual Design Evaluation, Task 1 Phase I Report," AP Note 400, January 19, 1962.

5. Humphries, G.E. "PL-3 Concept Selection," AP Note 408, Volume I, January 31, 1962.

6. Dixon, M. H. Project Engineer, "Nuclear Measurements for Type 3 Replacement Cores for SM-1, SM-1A and PM-2A, CE-3," APAE No. 100, January 11, 1962.

7. Fried, B.E., et al, "Flux and Power Distributions for the SM-2 Reference and Critical Experiment Cores," APAE Memo 286, June 30, 1961.

Control Rods

5 Rod Bank A, B, C, D, E
4 Rod Bank A, B, D, E

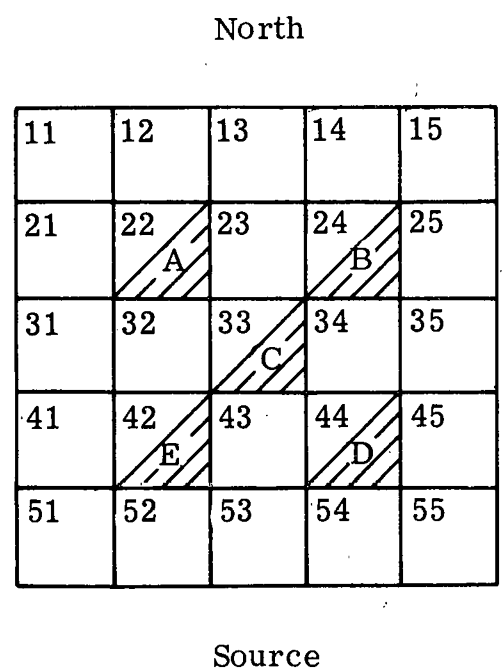


Figure A. 1. 5 x 5 Core PL-3 Critical Mockup

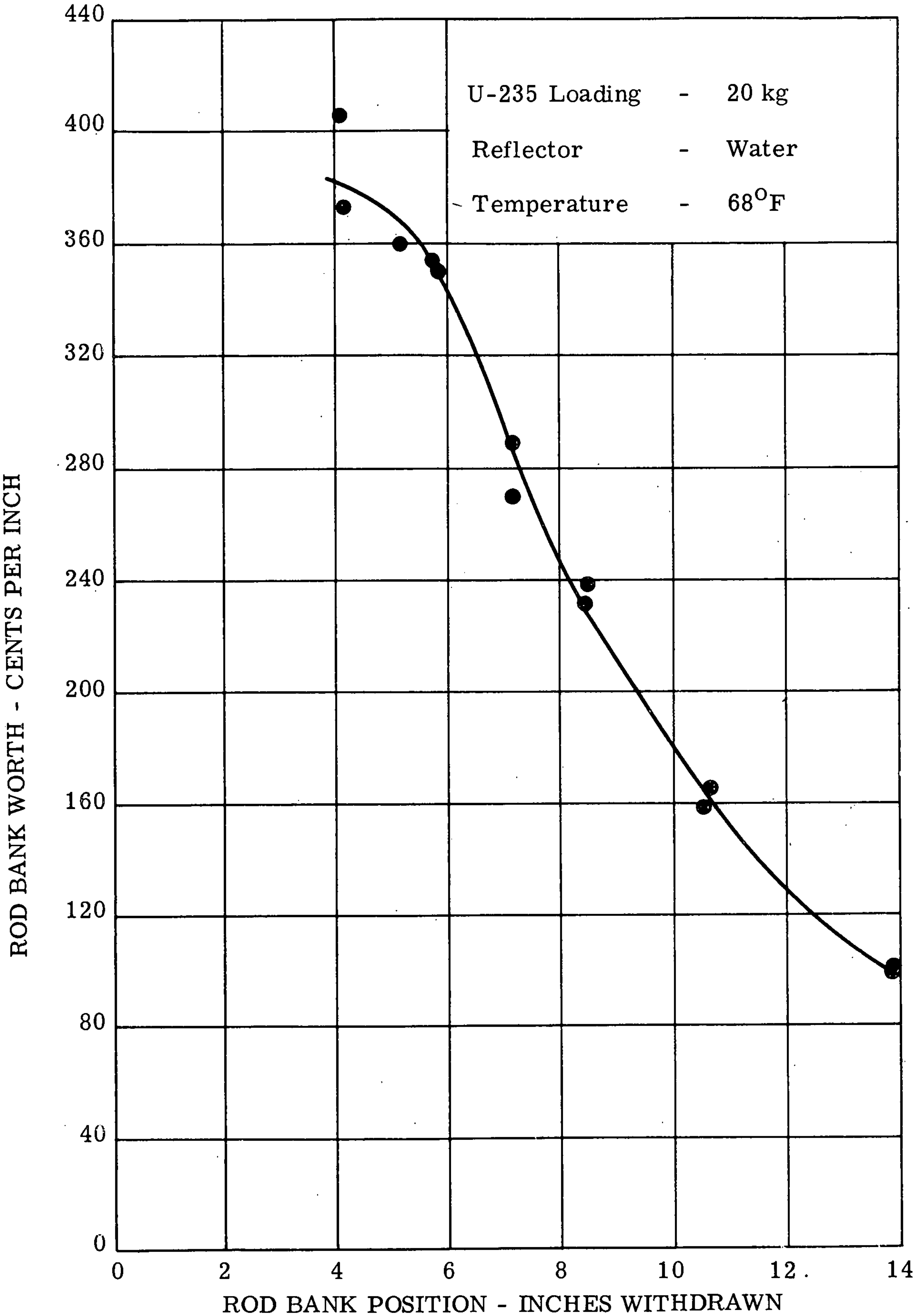


Figure A. 2. 5 x 5 Core PL-3 Critical Mockup - 4 Rod Bank Worth Vs Position

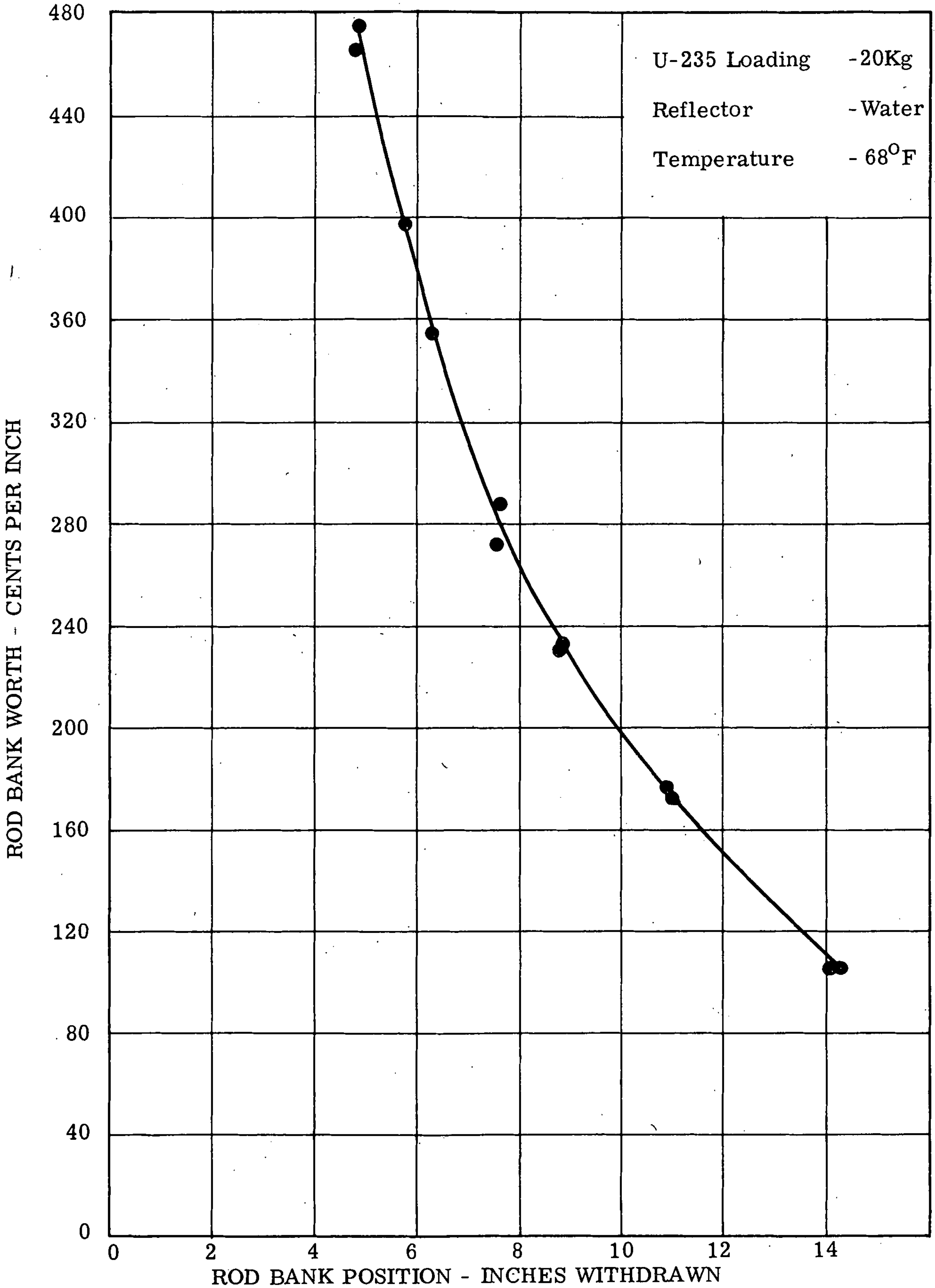


Figure A. 3. 5 x 5 Core PL-3 Critical Mockup - 5 Rod Bank Worth Vs Position

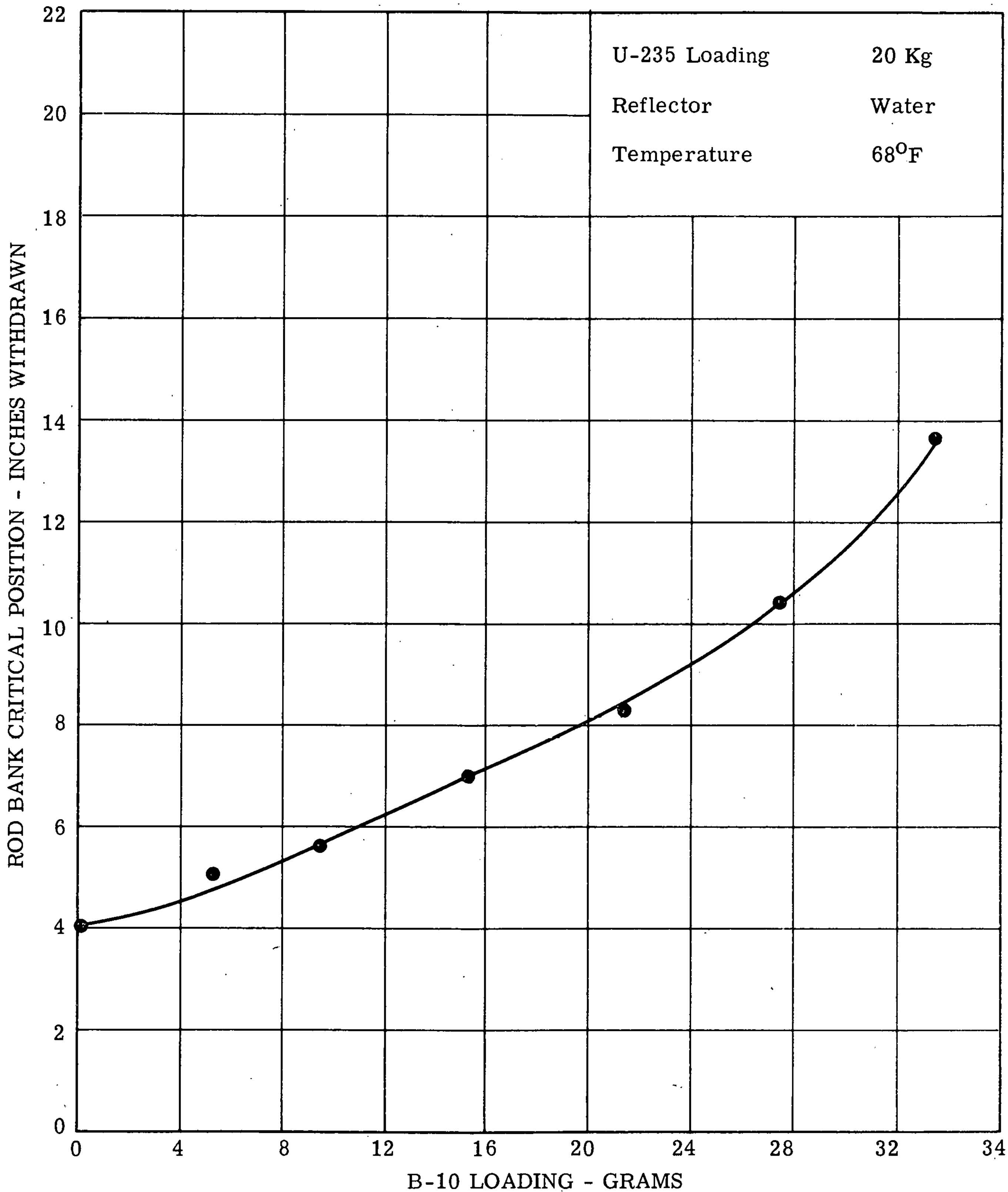


Figure A.4. 5 x 5 Core PL-3 Critical Mockup - 4 Rod Bank Critical Position Vs Boron Loading

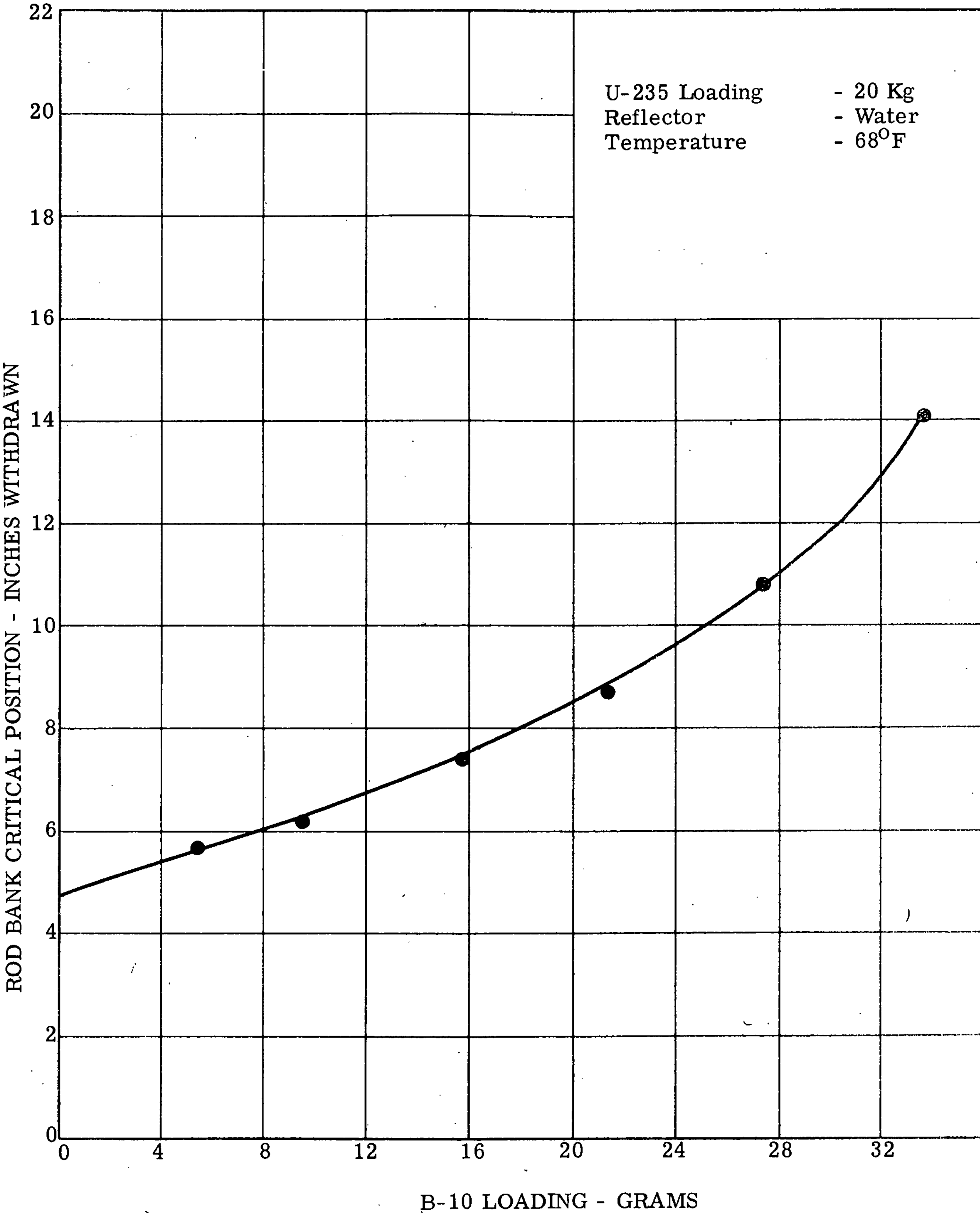


Figure A. 5. 5 x 5 Core PL-3 Critical Mockup - 5 Rod Bank Critical Position Vs Boron Loading

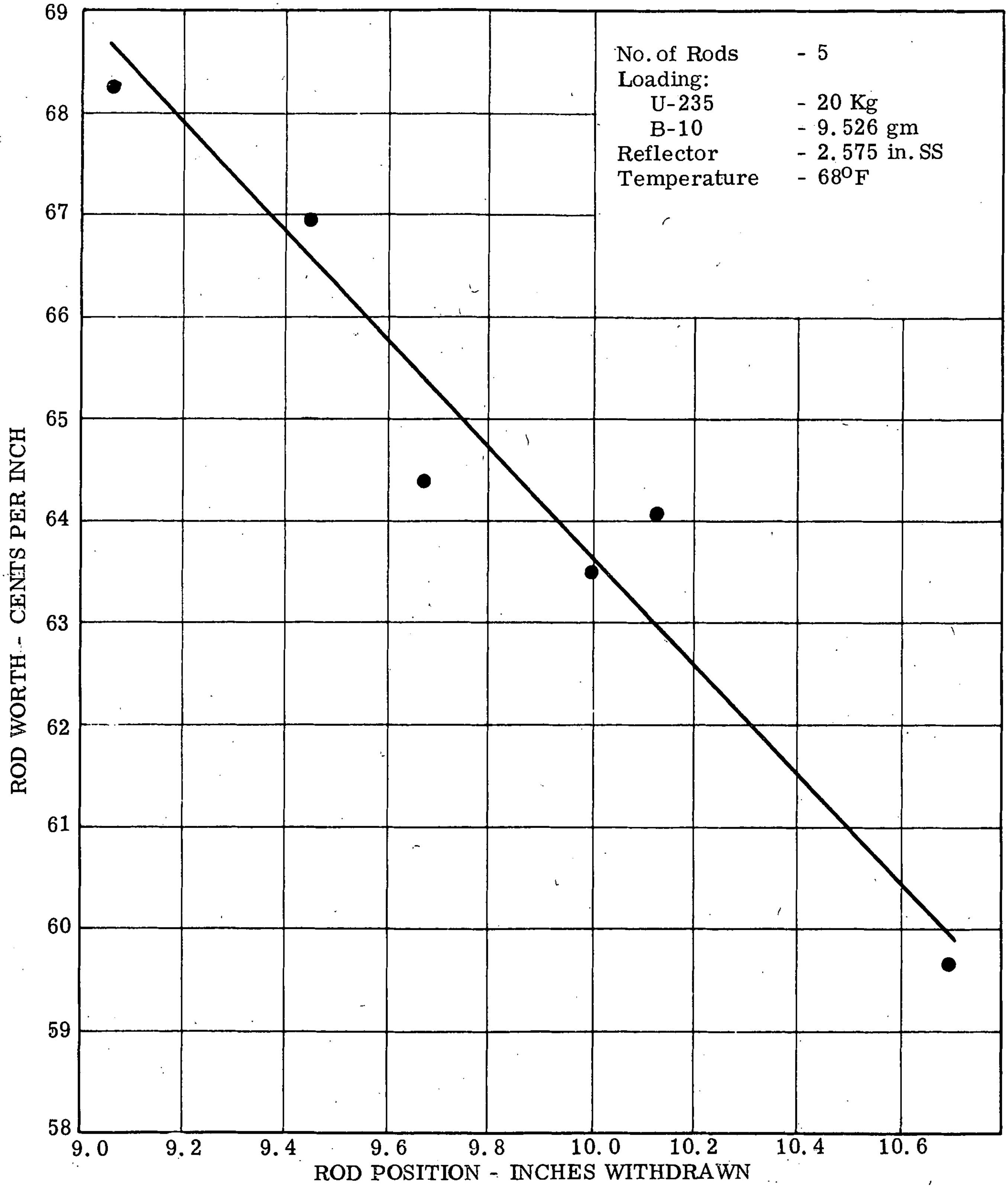


Figure A. 6. 5 x 5 Core PL-3 Critical Mockup - Control Rod A Worth Vs Position

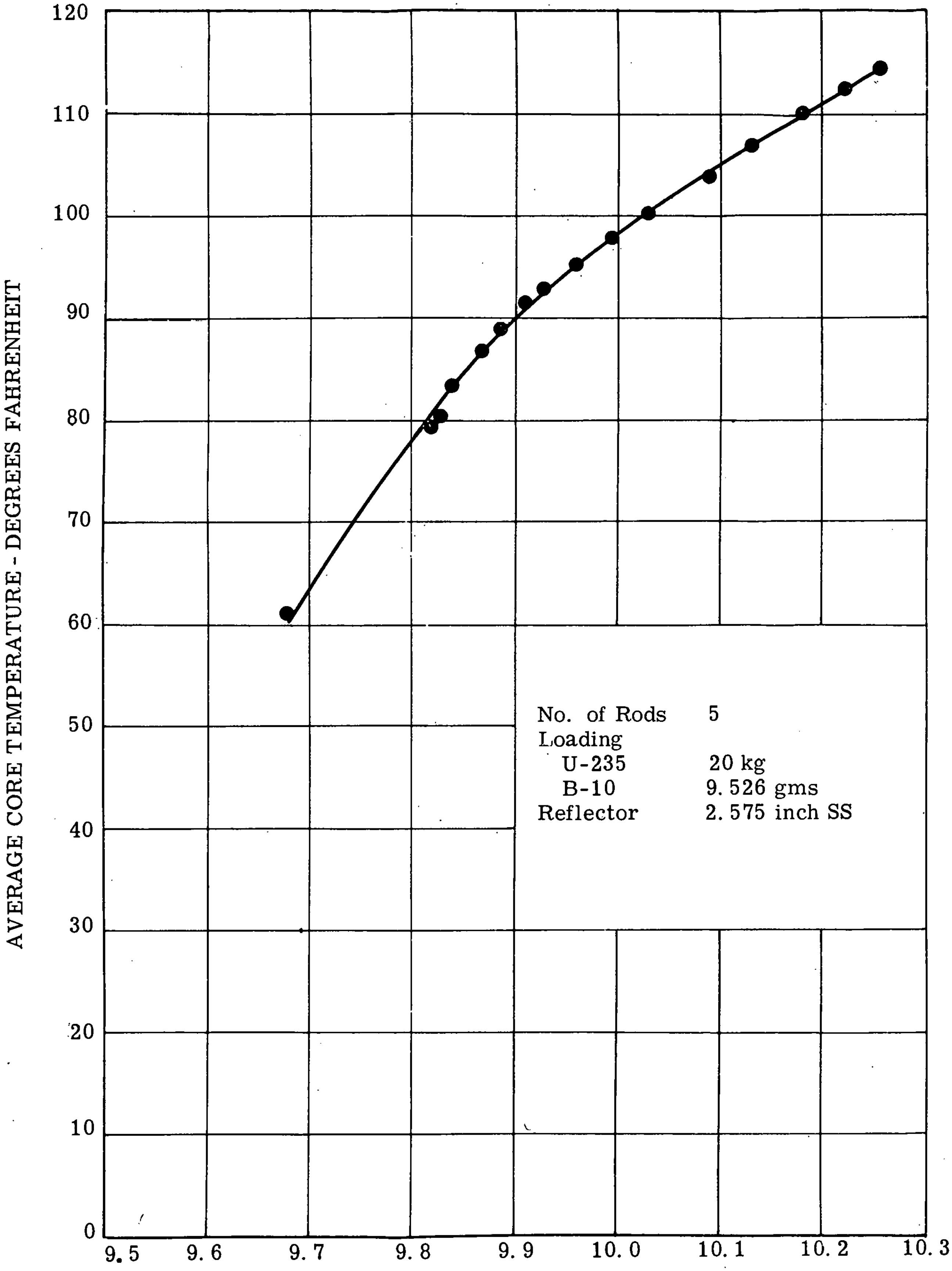


Figure A. 7. 5 x 5 Core PL-3 Critical Mockup - Average Core Temperature Vs Control Rod A Position

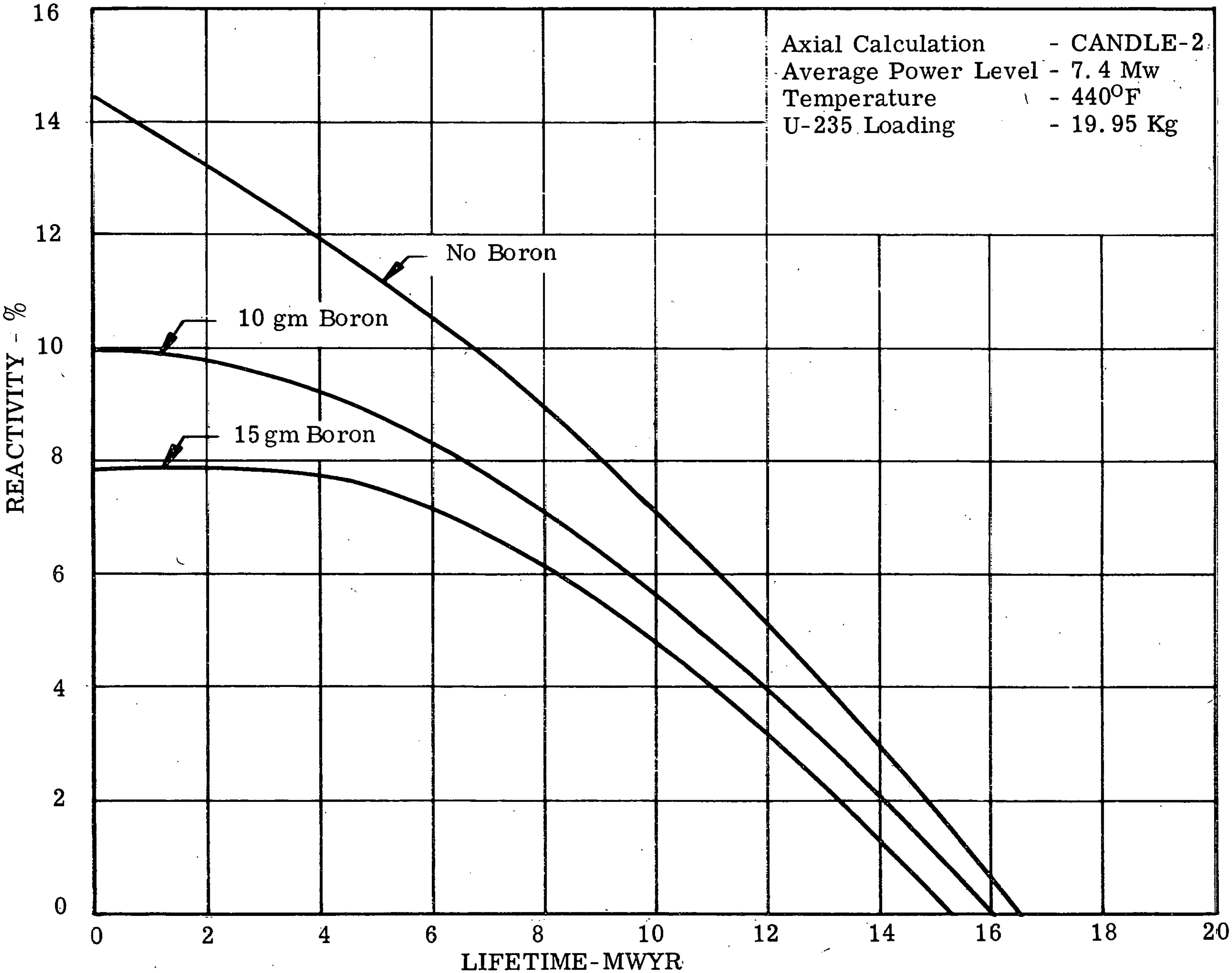


Figure A. 8. 5 x 5 Core Reactivity Vs Lifetime

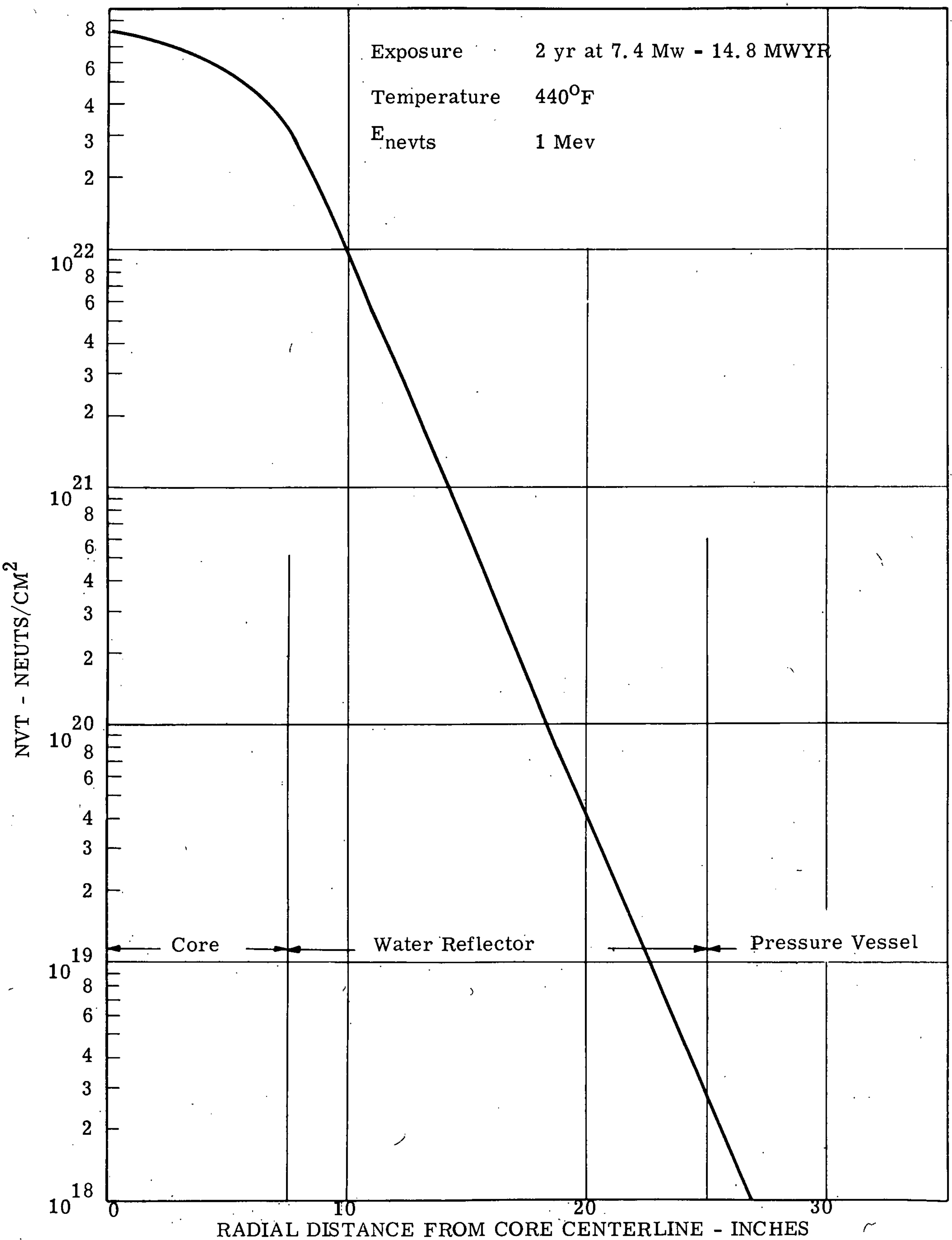


Figure A.9. 5 x 5 Core Integrated Fast Neutron Flux Radial Distribution

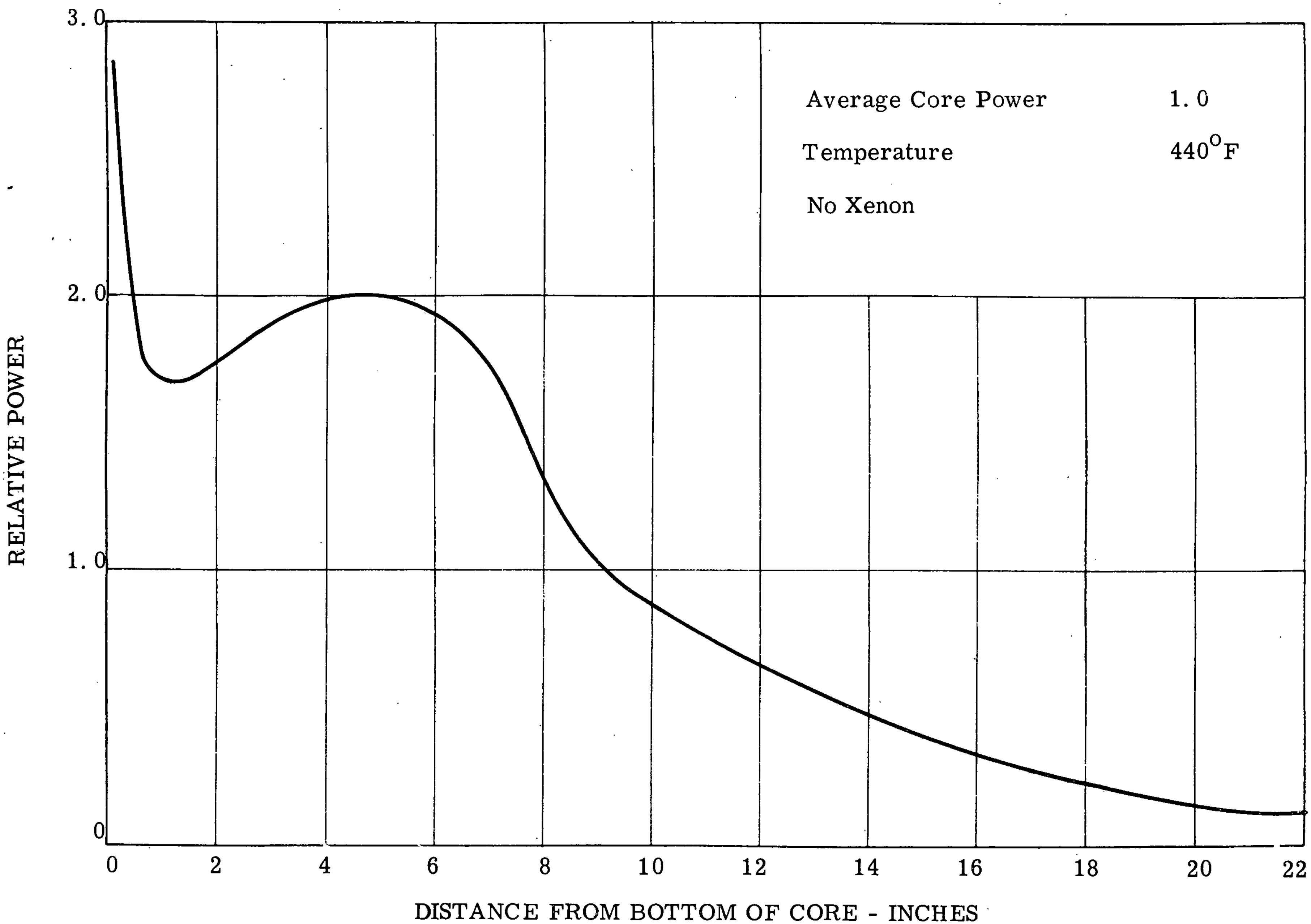


Figure A.10. 5 x 5 Core Axial Power Distribution - 4 Control Rod Bank

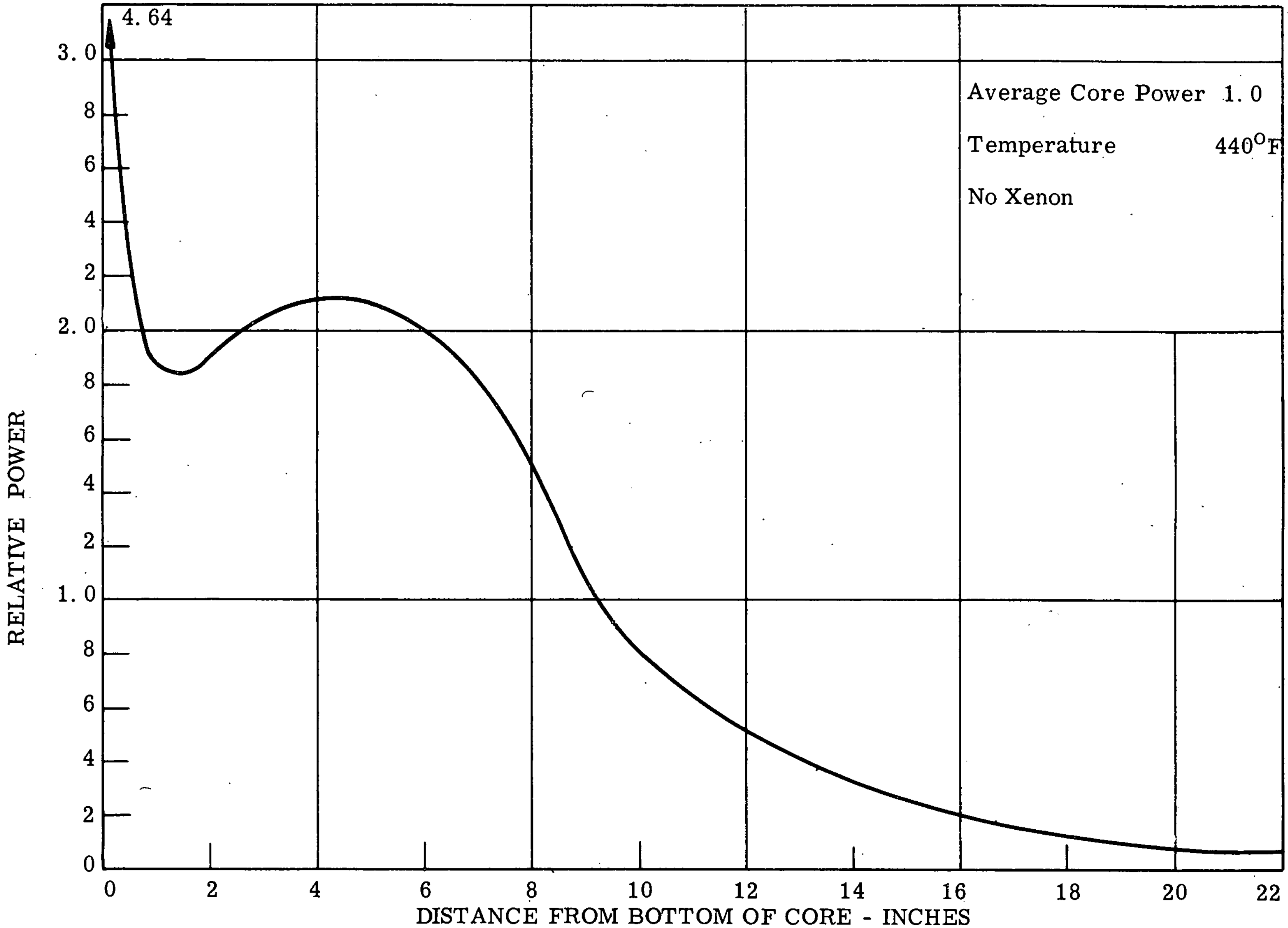


Figure A.11. 5 x 5 Core Axial Power Distribution - 5 Control Rod Bank

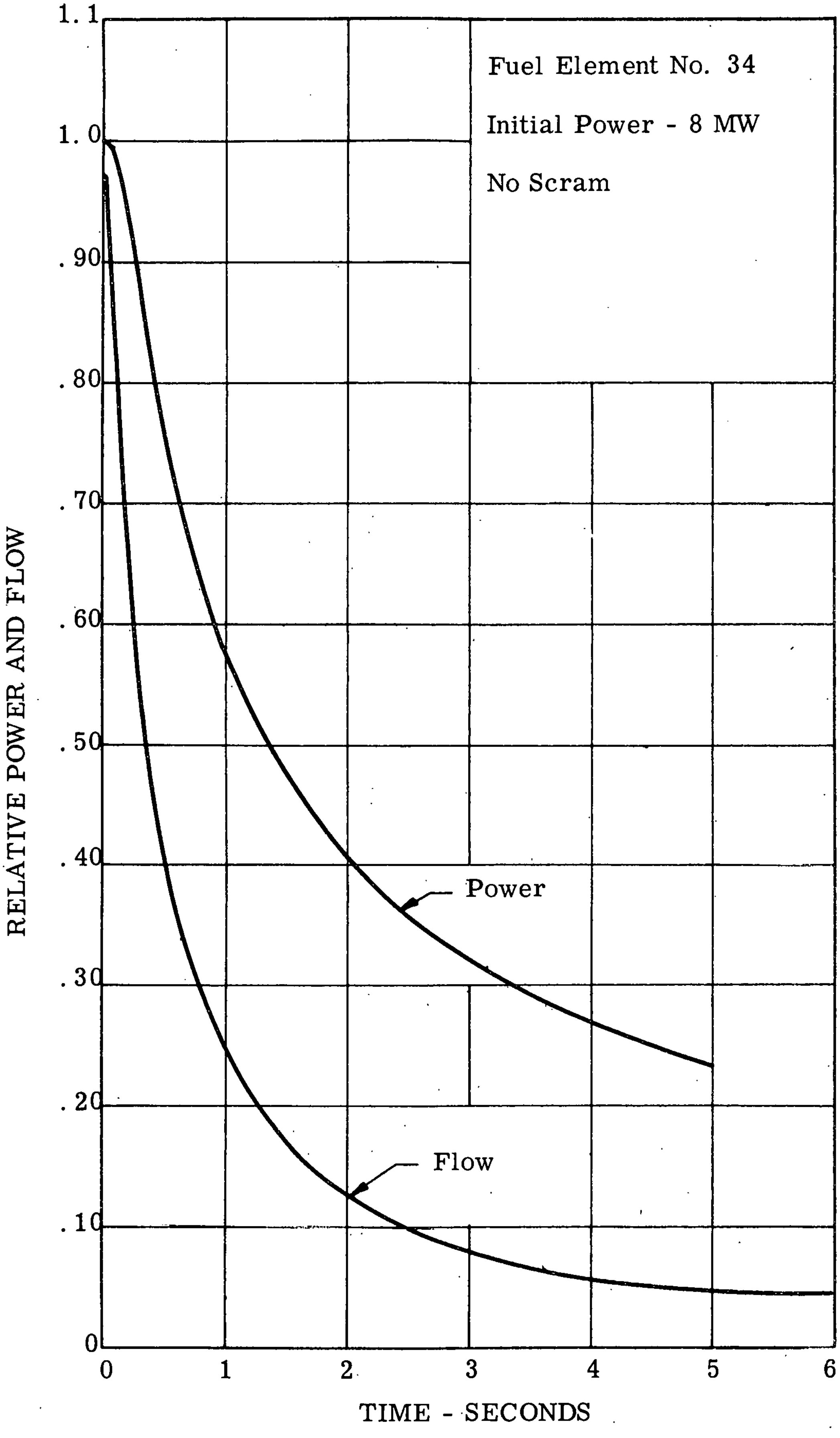


Figure A.12. 5 x 5 Core Power and Flow Coastdown Following Pump Failure

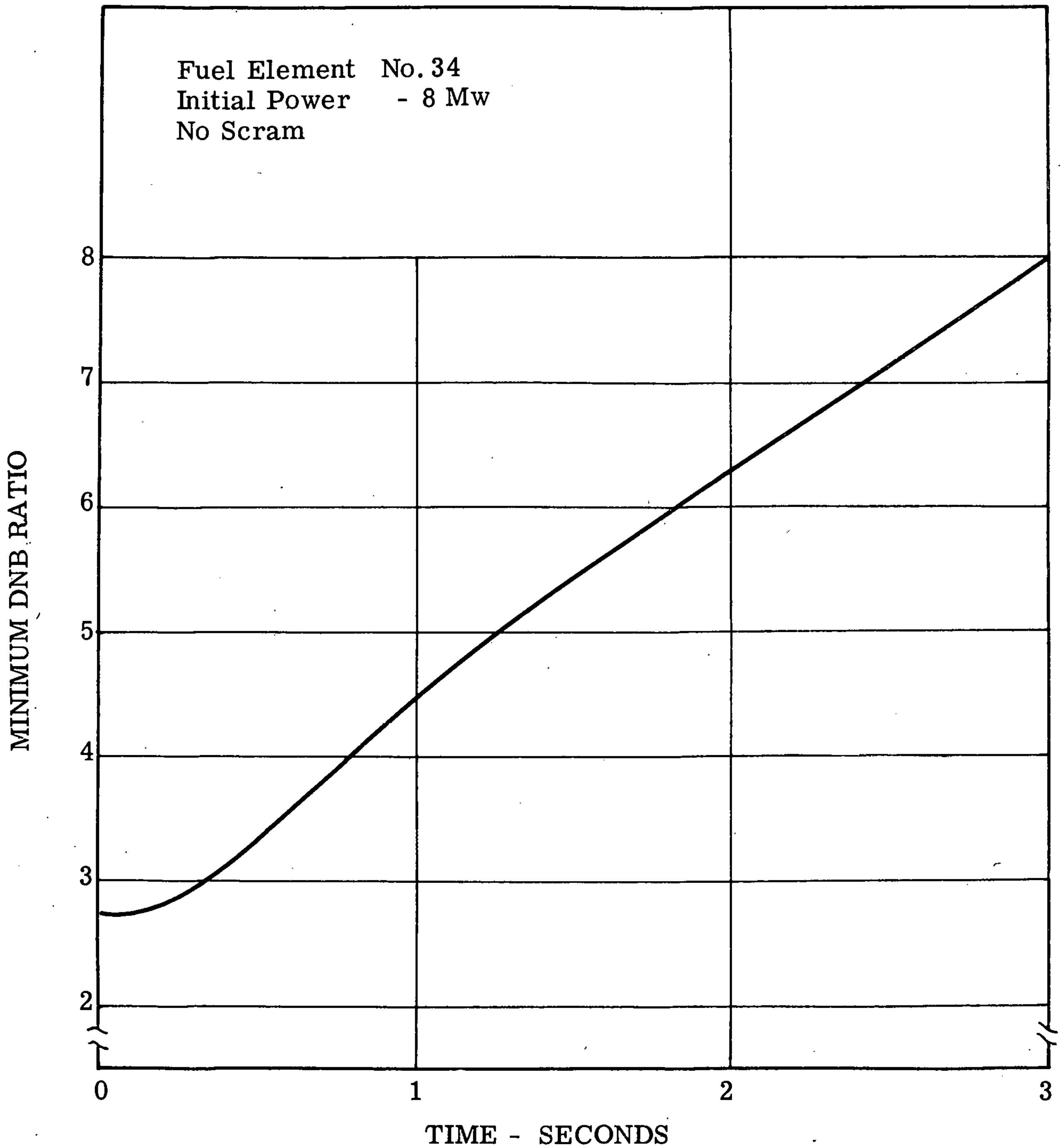
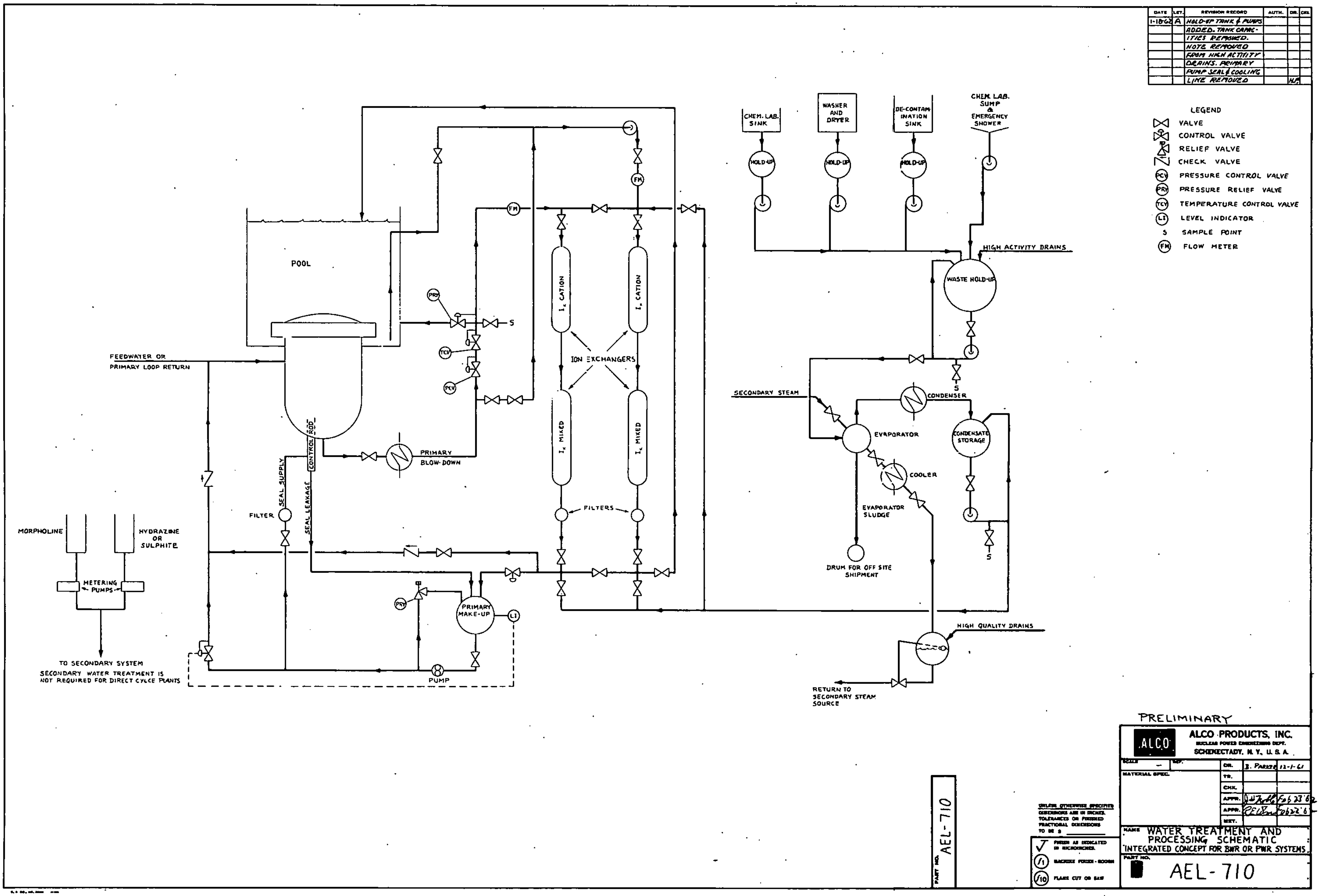


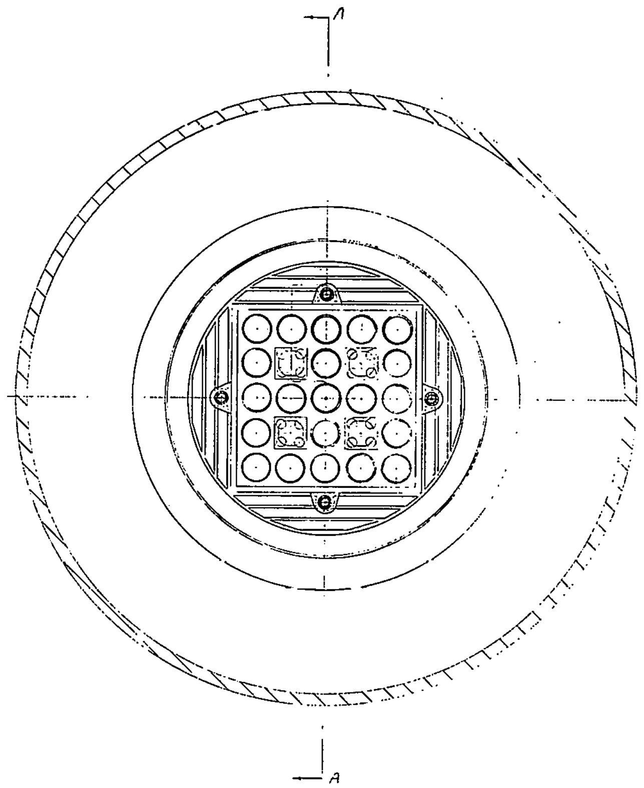
Figure A. 13. 5 x 5 Core Minimum DNB Ratio Following Pump Failure

APPENDIX B

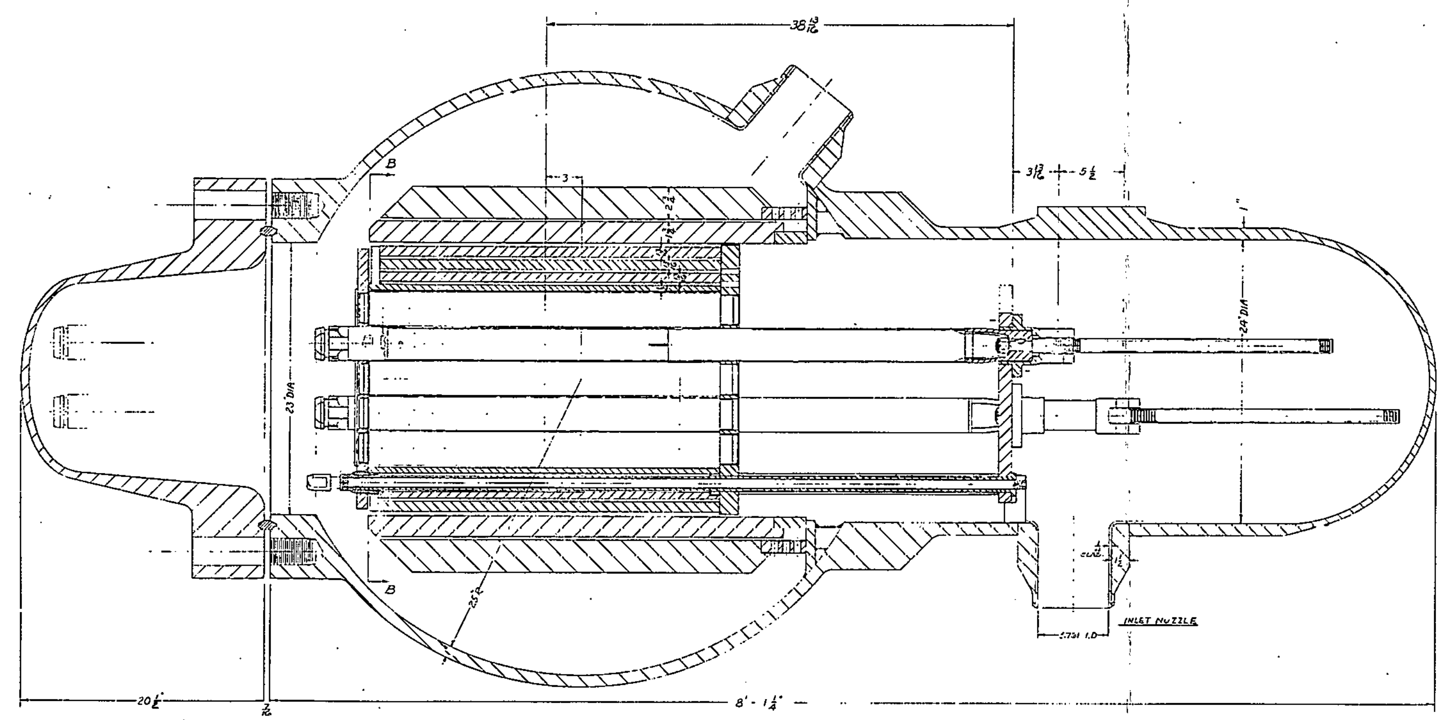
DRAWINGS



DATE
LET.
REVISION RECORD
AUTH.
DR.
CHK.
1-18-62
A
HOLD-UP TANK & PUMPS
ADDED. TANK CAPAC-
ITIES REMOVED.
NOTE REMOVED
FROM HIGH ACTIVITY
DRAINS. PRIMARY
PUMP SEAL & COOLING
LINE REMOVED
LEGEND
VALVE
CONTROL VALVE
RELIEF VALVE
CHECK VALVE
PCV PRESSURE CONTROL VALVE
PRV PRESSURE RELIEF VALVE
TCV TEMPERATURE CONTROL VALVE
LI LEVEL INDICATOR
S SAMPLE POINT
FM FLOW METER
POOL
FEEDWATER OR
PRIMARY LOOP RETURN
CONTROL ROD
SEAL SUPPLY
SEAL LEAKAGE
FILTER
PRIMARY
BLOW-DOWN
PRIMARY
MAKE-UP
PUMP
MORPHOLINE
HYDRAZINE
OR
SULPHITE
METERING
PUMPS
TO SECONDARY SYSTEM
SECONDARY WATER TREATMENT IS
NOT REQUIRED FOR DIRECT CYCLE PLANTS
CATION
MIXED
ION EXCHANGERS
FILTERS
CHEM. LAB.
SINK
WASHER
AND
DRYER
DE-CONTAM
INATION
SINK
CHEM. LAB.
SUMP
&
EMERGENCY
SHOWER
HOLD-UP
HIGH ACTIVITY DRAINS
WASTE HOLD-UP
SECONDARY STEAM
CONDENSER
EVAPORATOR
CONDENSATE
STORAGE
COOLER
EVAPORATOR
SLUDGE
DRUM FOR OFF SITE
SHIPMENT
HIGH QUALITY DRAINS
RETURN TO
SECONDARY STEAM
SOURCE
PART NO.
AEL-710
PRELIMINARY
ALCO
ALCO PRODUCTS, INC.
SCHENECTADY, N. Y., U. S. A.
MATERIAL SPEC.
DR.
J. PARISE
12-1-61
TR.
CHK.
APPR.
Feb 23 '62
APPR.
NAME
WATER TREATMENT AND
PROCESSING SCHEMATIC
INTEGRATED CONCEPT FOR BWR OR PWR SYSTEMS
PART NO.
AEL-710



SECTION BB



SECTION AA

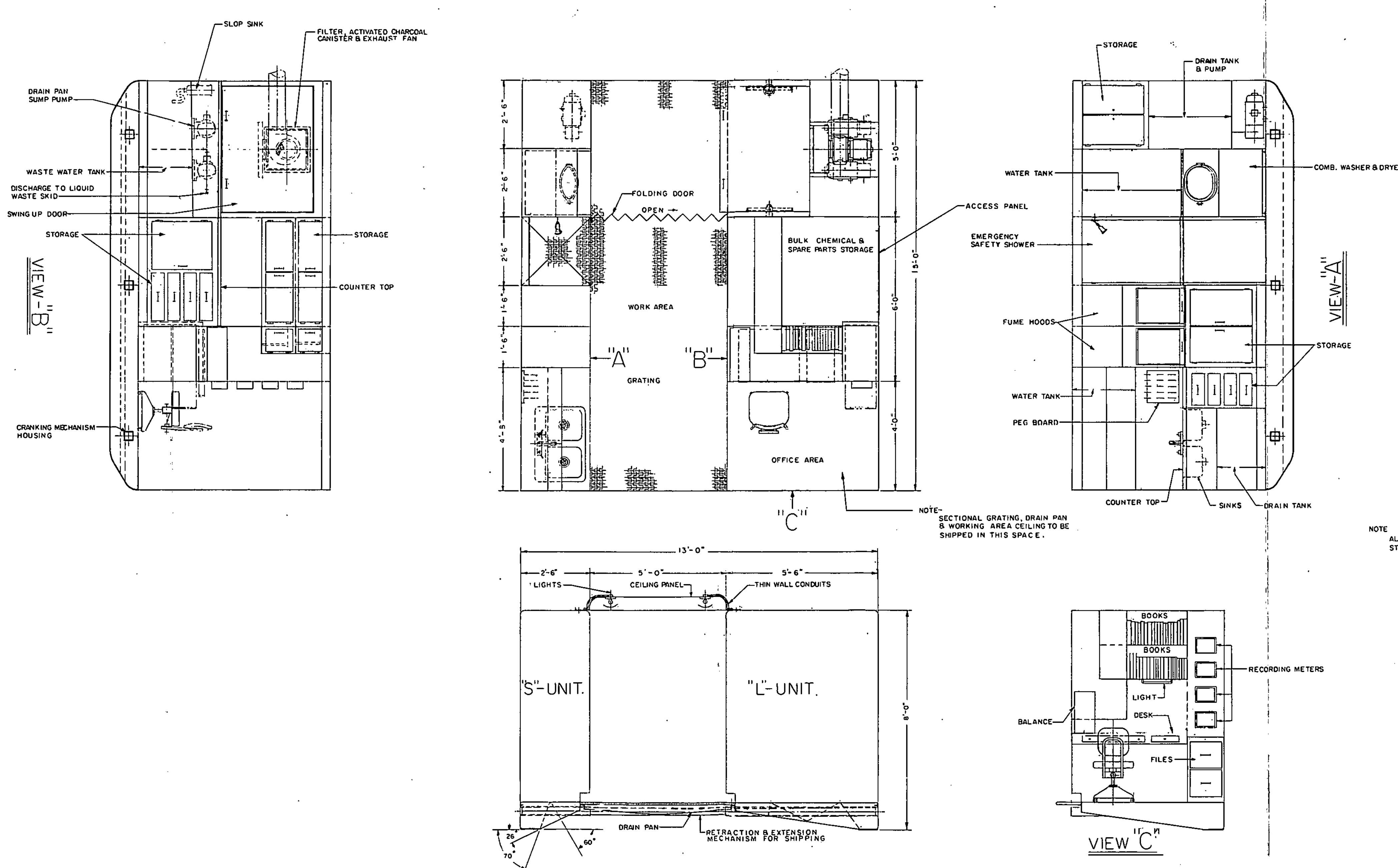
MATERIAL: CARBON STEEL
DESIGN PRESSURE: 1350 PSI

LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL- PL 3	STARTED	FINISHED	SCALE- 1/4
REFERENCES-	DRAWN BY- J. [illegible]		
	DATE-	APPROVED-	
AUTHORITY-	DATE-	APPROVED- [illegible]	
	DATE-	APPROVED-	

TITLE- REACTOR VESSEL CONCEPT A2

LAYOUT NO.- AEL 711



NOTE
ALL STRUCTURE TO BE FABRICATED FROM STAINLESS STEEL TYPE-304(ANNEALED)

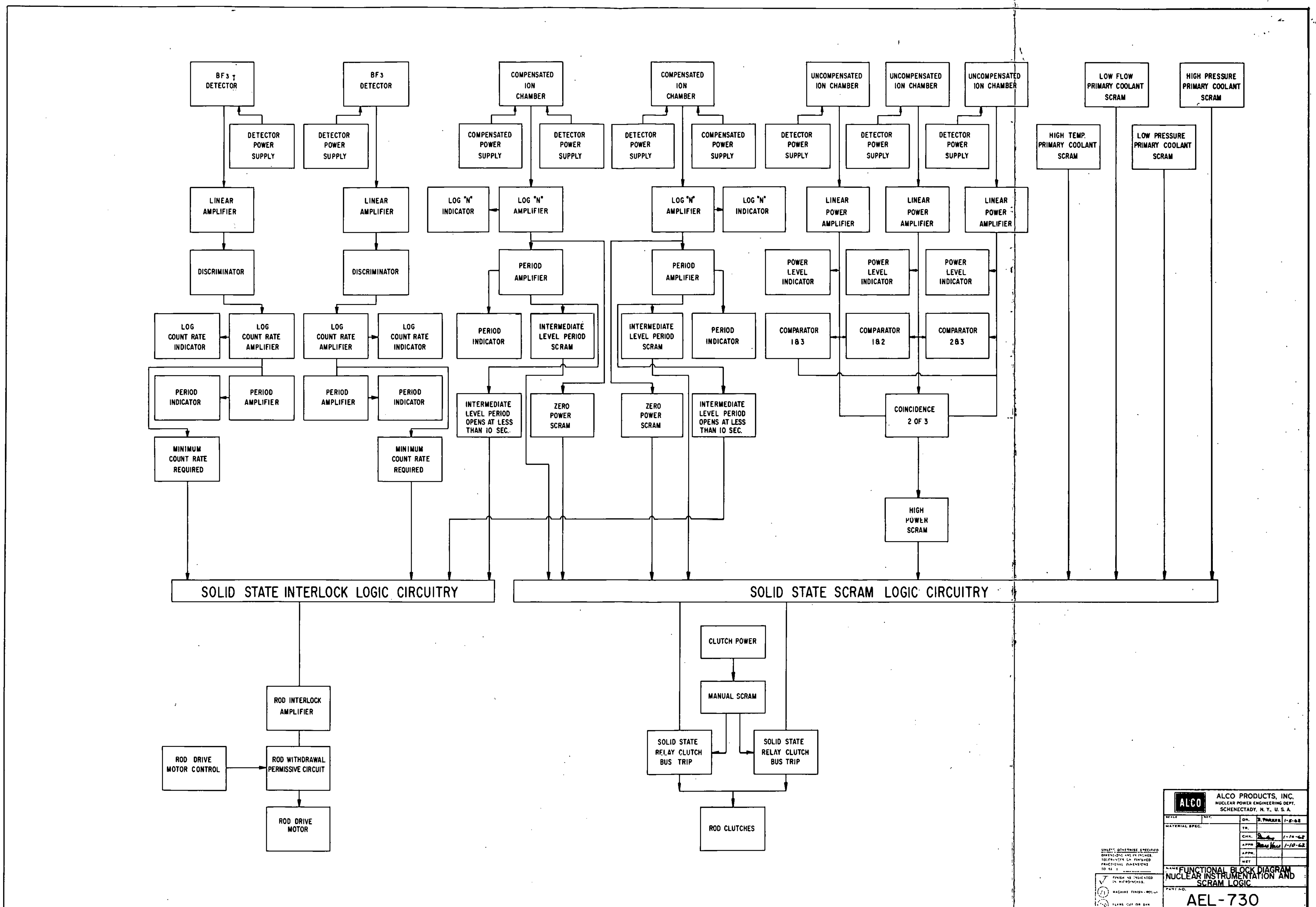
PRELI[illegible]

LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

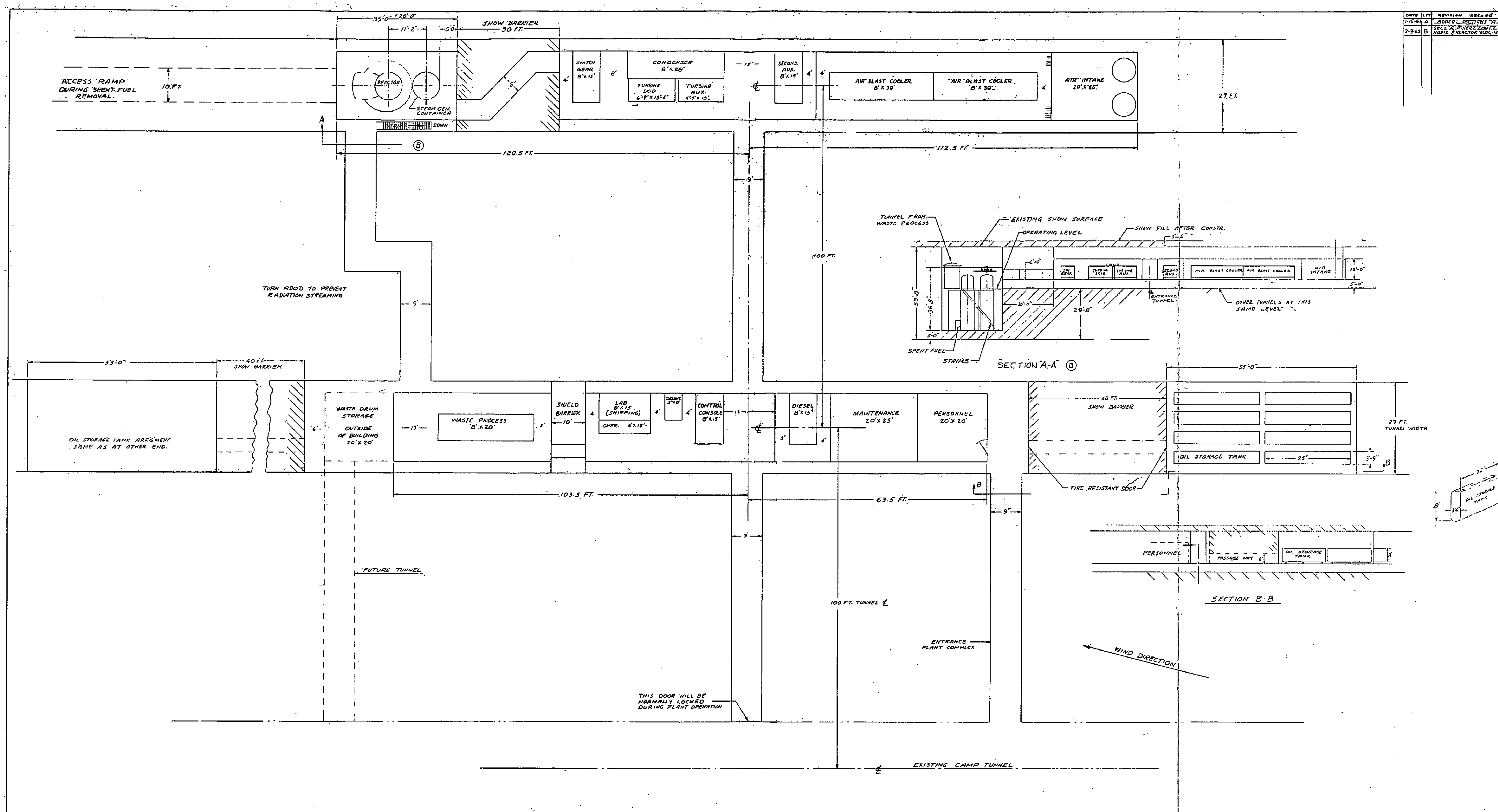
MODEL- PL-3	STARTED 12-20-61	FINISHED 1-3-62	SCALE- 3/4"=1 FT.
REFERENCES- AEL-739	DRAWN BY J. L. Boyd		
	DATE- 1-3-62	APPROVED- W.F. SCAUTERWAY	
AUTHORITY-	DATE- 2-19-62	APPROVED- G.E.H.	
	DATE-	APPROVED-	

TITLE-
CHEMICAL LABORATORY SKID

LAYOUT NO.- AEL-729



BF3 DETECTOR
DETECTOR POWER SUPPLY
LINEAR AMPLIFIER
DISCRIMINATOR
LOG COUNT RATE INDICATOR
LOG COUNT RATE AMPLIFIER
PERIOD INDICATOR
PERIOD AMPLIFIER
MINIMUM COUNT RATE REQUIRED
BF3 DETECTOR
DETECTOR POWER SUPPLY
LINEAR AMPLIFIER
DISCRIMINATOR
LOG COUNT RATE AMPLIFIER
LOG COUNT RATE INDICATOR
PERIOD AMPLIFIER
PERIOD INDICATOR
MINIMUM COUNT RATE REQUIRED
COMPENSATED ION CHAMBER
COMPENSATED POWER SUPPLY
DETECTOR POWER SUPPLY
LOG "N" INDICATOR
LOG "N" AMPLIFIER
PERIOD AMPLIFIER
PERIOD INDICATOR
INTERMEDIATE LEVEL PERIOD SCRAM
INTERMEDIATE LEVEL PERIOD OPENS AT LESS THAN 10 SEC.
ZERO POWER SCRAM
COMPENSATED ION CHAMBER
DETECTOR POWER SUPPLY
COMPENSATED POWER SUPPLY
LOG "N" AMPLIFIER
LOG "N" INDICATOR
PERIOD AMPLIFIER
INTERMEDIATE LEVEL PERIOD SCRAM
PERIOD INDICATOR
ZERO POWER SCRAM
INTERMEDIATE LEVEL PERIOD OPENS AT LESS THAN 10 SEC.
UNCOMPENSATED ION CHAMBER
UNCOMPENSATED ION CHAMBER
UNCOMPENSATED ION CHAMBER
DETECTOR POWER SUPPLY
DETECTOR POWER SUPPLY
DETECTOR POWER SUPPLY
LINEAR POWER AMPLIFIER
LINEAR POWER AMPLIFIER
LINEAR POWER AMPLIFIER
POWER LEVEL INDICATOR
POWER LEVEL INDICATOR
POWER LEVEL INDICATOR
COMPARATOR 1&3
COMPARATOR 1&2
COMPARATOR 2&3
COINCIDENCE 2 OF 3
HIGH POWER SCRAM
LOW FLOW PRIMARY COOLANT SCRAM
HIGH PRESSURE PRIMARY COOLANT SCRAM
HIGH TEMP. PRIMARY COOLANT SCRAM
LOW PRESSURE PRIMARY COOLANT SCRAM
SOLID STATE INTERLOCK LOGIC CIRCUITRY
SOLID STATE SCRAM LOGIC CIRCUITRY
ROD INTERLOCK AMPLIFIER
ROD DRIVE MOTOR CONTROL
ROD WITHDRAWAL PERMISSIVE CIRCUIT
ROD DRIVE MOTOR
CLUTCH POWER
MANUAL SCRAM
SOLID STATE RELAY CLUTCH BUS TRIP
SOLID STATE RELAY CLUTCH BUS TRIP
ROD CLUTCHES
ALCO
ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N.Y., U.S.A.
FUNCTIONAL BLOCK DIAGRAM NUCLEAR INSTRUMENTATION AND SCRAM LOGIC
AEL-730



DATE	LET	REVISION RECORD			
1-15-62	A	ADDED SECTIONS "A-A" & "B-B"			
2-9-62	B	SECT. "A-A" VERT. CONT'S WERE HORIZ. & REACTOR BLDG. WAS 70'x20'			

LAYOUT

MODEL PL-3

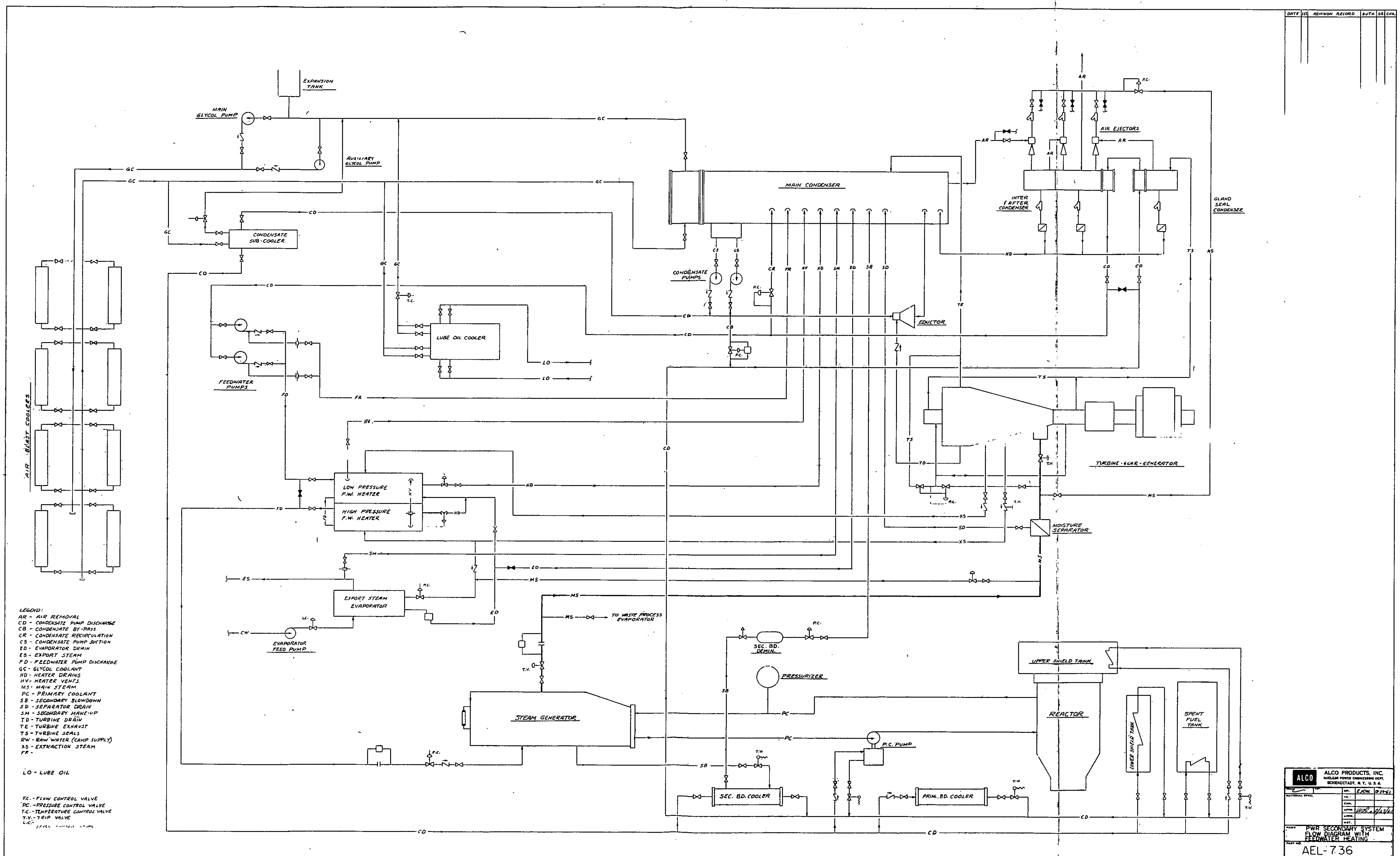
1-10-62 1-10-62 1/8" = 1 FT.

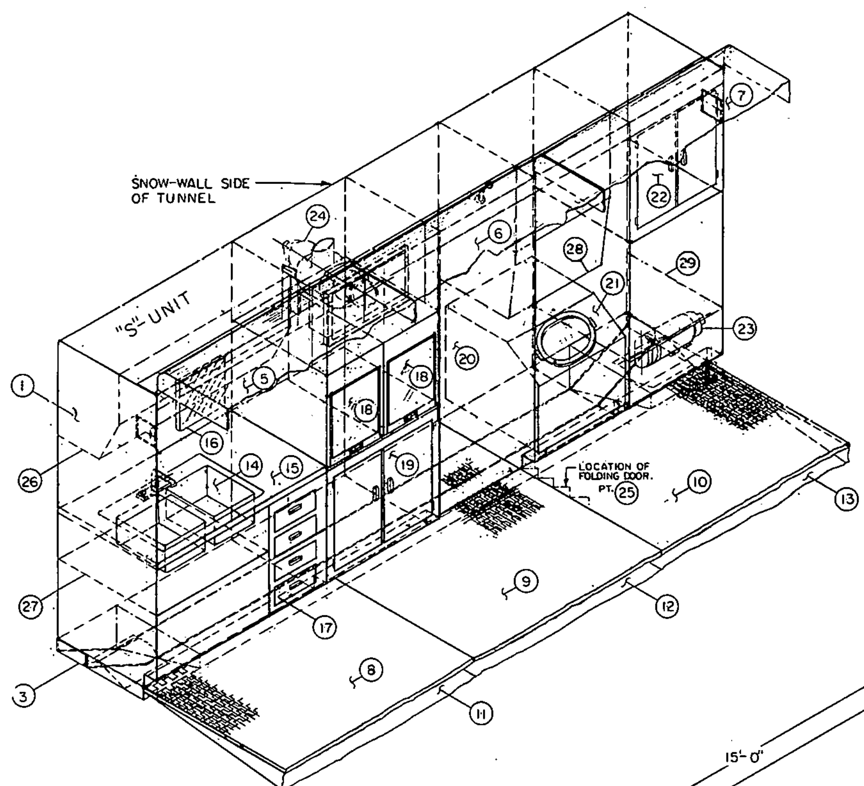
E.H. WILLIAMS

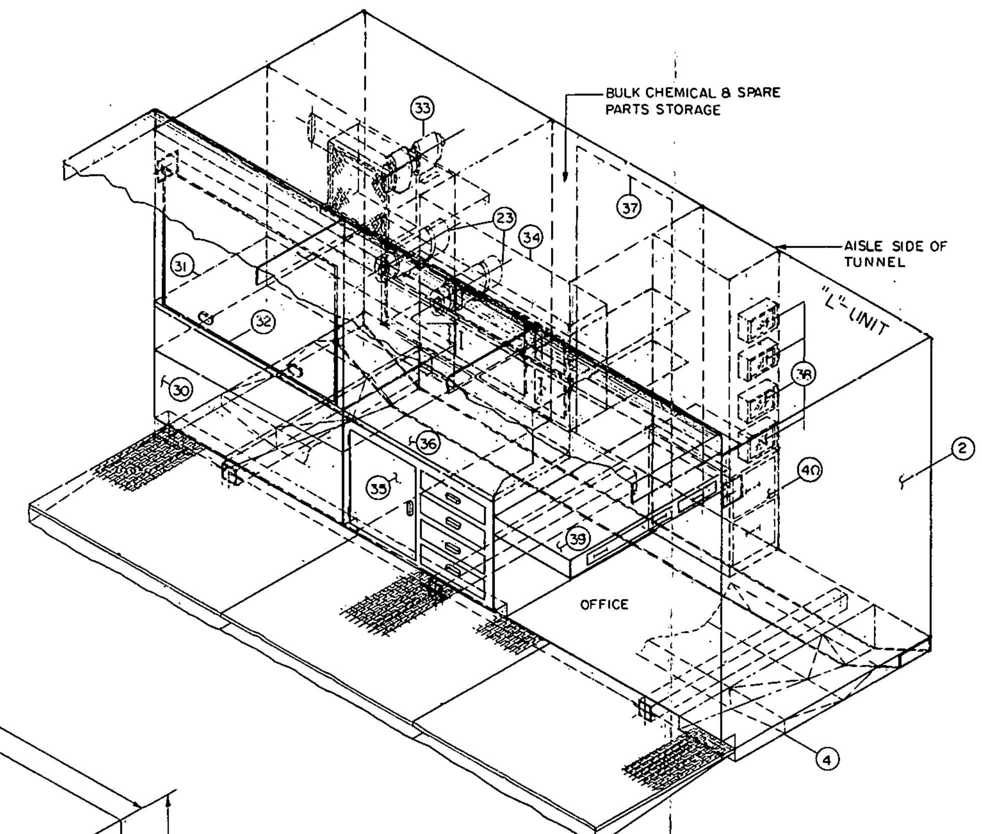
AUTHORITY 2/23/62

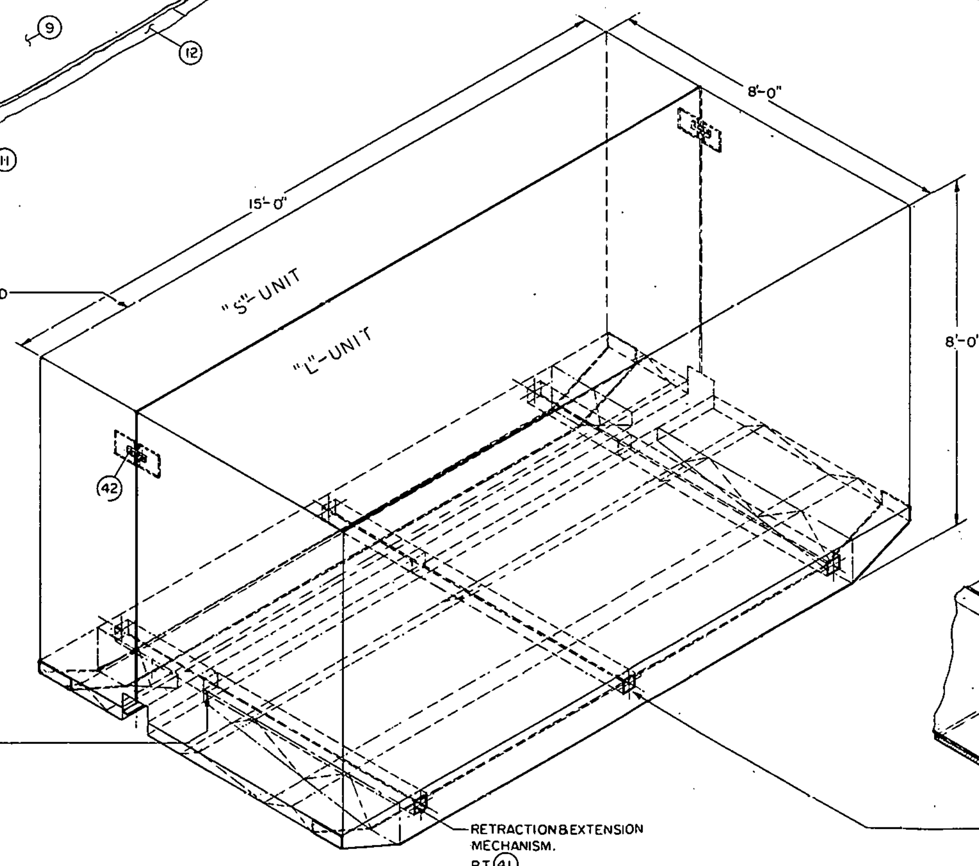
TITLE - PLANT PLAN ARRANGEMENT. BASED ON PWR CONCEPT

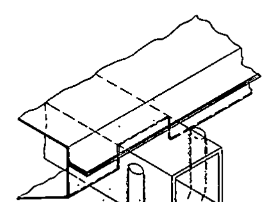
LAYOUT NO. - AEL-734


EXPANSION TANK
MAIN GLYCOL PUMP
AUXILIARY GLYCOL PUMP
CONDENSATE SUB-COOLER
MAIN CONDENSER
AIR EJECTORS
INTER & AFTER CONDENSER
GLAND SEAL CONDENSER
CONDENSATE PUMPS
EDUCTOR
LUBE OIL COOLER
FEEDWATER PUMPS
AIR BLAST COOLERS
LOW PRESSURE F.W. HEATER
HIGH PRESSURE F.W. HEATER
TURBINE-GEAR-GENERATOR
MOISTURE SEPARATOR
EXPORT STEAM EVAPORATOR
EVAPORATOR FEED PUMP
TO WASTE PROCESS EVAPORATOR
SEC. BD. DEMIN.
PRESSURIZER
UPPER SHIELD TANK
REACTOR
STEAM GENERATOR
P.C. PUMP
SEC. BD. COOLER
PRIM. BD. COOLER
LOWER SHIELD TANK
SPENT FUEL TANK
LEGEND:
AR - AIR REMOVAL
CD - CONDENSATE PUMP DISCHARGE
CB - CONDENSATE BY-PASS
CR - CONDENSATE RECIRCULATION
CS - CONDENSATE PUMP SUCTION
ED - EVAPORATOR DRAIN
ES - EXPORT STEAM
FD - FEEDWATER PUMP DISCHARGE
GC - GLYCOL COOLANT
HD - HEATER DRAINS
HV - HEATER VENTS.
MS - MAIN STEAM
PC - PRIMARY COOLANT
SB - SECONDARY BLOWDOWN
SD - SEPARATOR DRAIN
SM - SECONDARY MAKE-UP
TD - TURBINE DRAIN
TE - TURBINE EXHAUST
TS - TURBINE SEALS
RW - RAW WATER (CAMP SUPPLY)
XS - EXTRACTION STEAM
FR -
LO - LUBE OIL
F.C. - FLOW CONTROL VALVE
P.C. - PRESSURE CONTROL VALVE
T.C. - TEMPERATURE CONTROL VALVE
T.V. - TRIP VALVE
L.C. - LEVEL CONTROL VALVE
DATE
REVISION RECORD
AUTH.
ALCO
ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N. Y. U. S. A.
MATERIAL SPEC.
DR. EHW 12-29-61
TR.
CHK.
APPR.
APPR.
MET.
NAME PWR SECONDARY SYSTEM FLOW DIAGRAM WITH FEEDWATER HEATING
PART NO. AEL-736





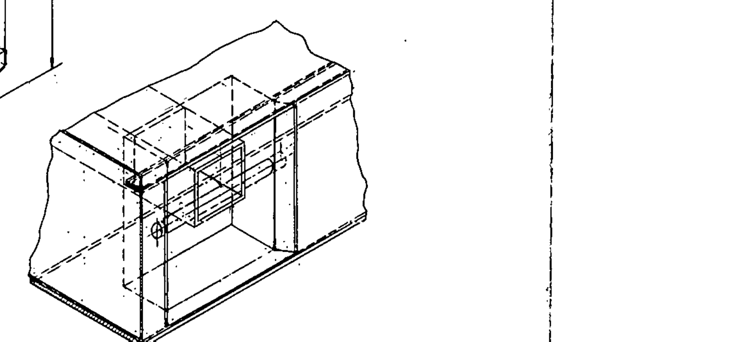




TIE DOWN FITTING (2-PLACES) "S"-UNIT
SCALE- 1/4 SIZE

RETRACTION & EXTENSION MECHANISM. PT 41

GROSS WEIGHT 8,518#



LIFTING & TIE DOWN FITTING (6-PLACES)
SCALE- 1/4 SIZE

NOTE–
ALL STRUCTURE TO BE FABRICATED FROM STAINLESS STEEL TYPE-304 (ANNEALED)

PART NO	QUAN	MATERIAL OR DRAWING NO.	DESCRIPTION
42	2		LOCKING BAR (SHIPPING)
41	3	AEL 753	RETRACTION & EXTENSION MECHANISM
40	1		FILE CABINET
39	1		DESK
38	4		RECORDING METER
37	1		ACCESS PANEL
36	1		COUNTER TOP
35	1		CABINET
34	1		CABINET
33	1		FILTER, ACTIV. CHARCOAL CANISTER & EXH. FAN
32	1		SAFETY DOOR
31	1		SLOP SINK
30	1		WASTE-WATER TANK
29	1		DRAIN TANK
28	1		WATER TANK
27	1		DRAIN TANK
26	1		WATER TANK
25	1		FOLDING DOOR & FITTINGS
24	1		FILTER, ACTIV. CHARCOAL CANISTER & EXH. FAN
23	3		MOTOR DRIVEN PUMP
22	1		CABINET
21	1	COMM.	WASHER & DRYER COMB.
20	1		SHOWER STALL & FITTINGS
19	1		CABINET
18	2		FUME HOOD
17	1		CABINET
16	1		PEG BOARD
15	1		COUNTER TOP
14	1		DOUBLE SINK & FITTINGS
13	1		SECTIONAL DRAIN PAN
12	1		SECTIONAL DRAIN PAN
11	1		SECTIONAL DRAIN PAN
10	1		SECTIONAL GRATING
9	1		SECTIONAL GRATING
8	1		SECTIONAL GRATING
7	1		CEILING PANEL
6	1		CEILING PANEL
5	1		CEILING PANEL
4	1	AEL-756	LOWER STRUCTURE "L" UNIT
3	1	AEL-758	LOWER STRUCTURE "S" UNIT
2	1		UPPER STRUCTURE "L" UNIT
1	1		UPPER STRUCTURE "S" UNIT

PRELIMINARY

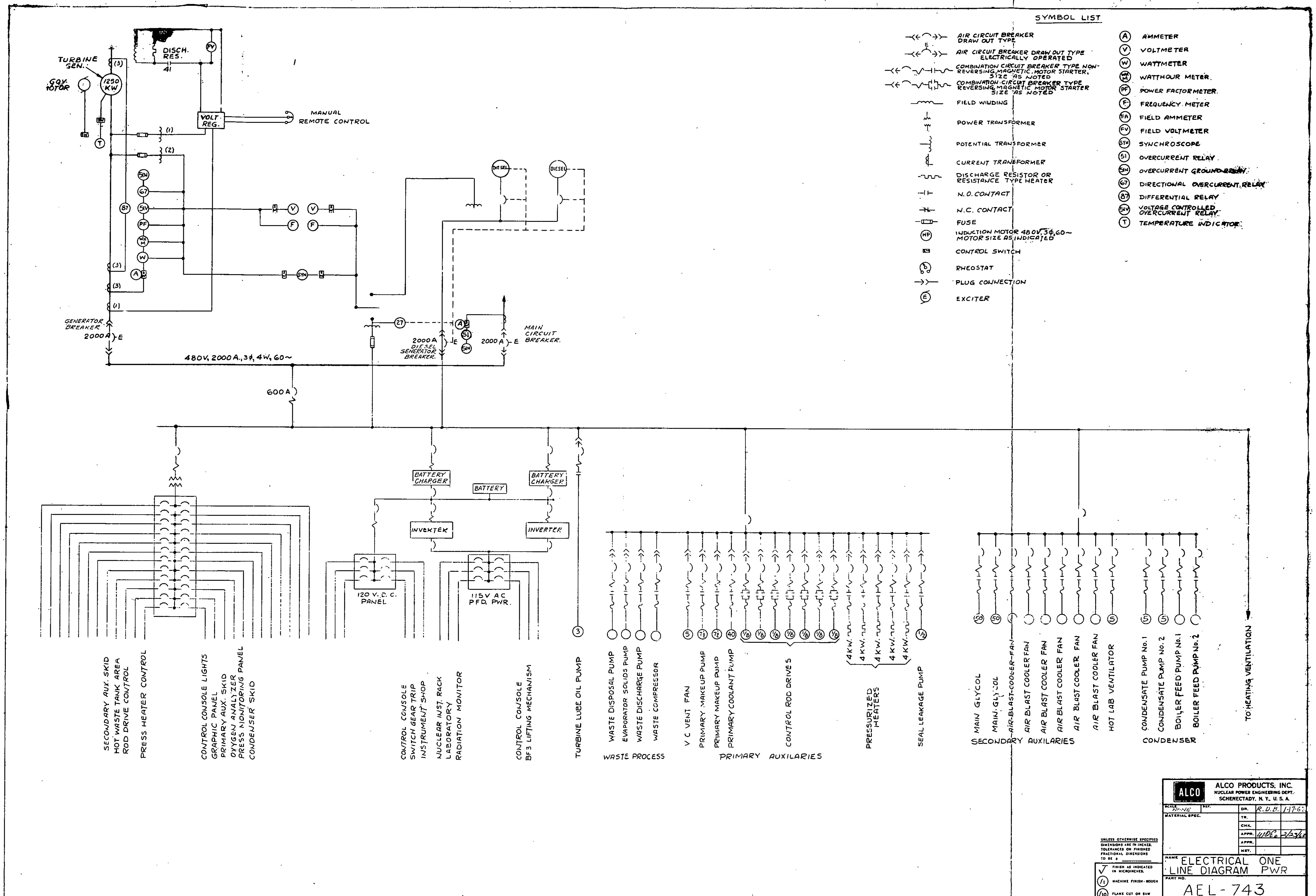
LAYOUT

ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

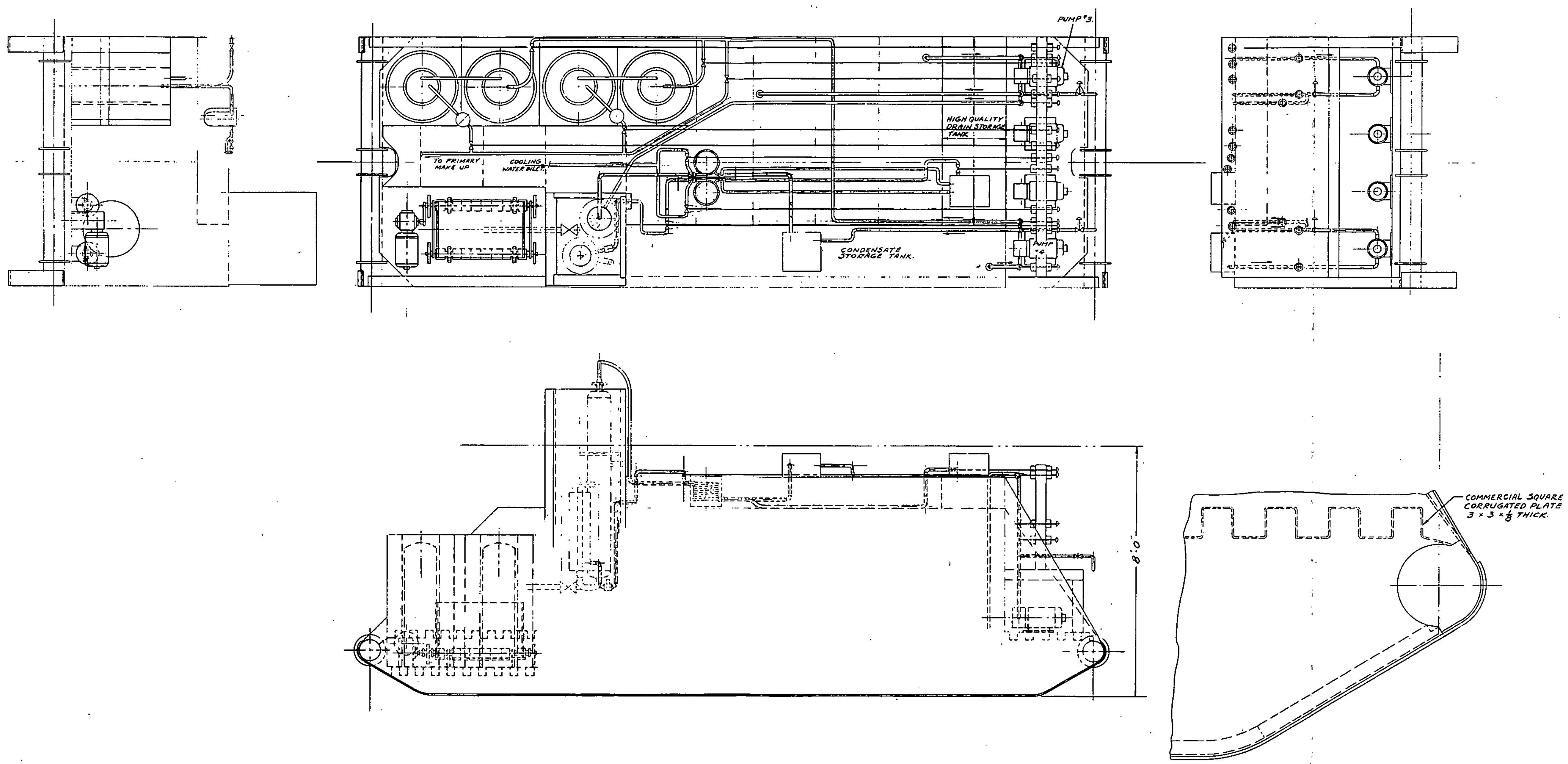
MODEL- PL-3	STARTED 1-3-62	FINISHED 1-18-62	SCALE- 3/4" = 1-FT
REFERENCES- AEL-729	DRAWN BY J. L. Boyd		
	DATE- 1-19-62	APPROVED- W. SCANTLEBURY	
AUTHORITY-	DATE- 2-27-62	APPROVED- G.E.H.	
	DATE-	APPROVED-	

TITLE-
CHEMICAL LABORATORY SKID. (ISOMETRIC)

LAYOUT NO.- AEL-739



SYMBOL LIST
AIR CIRCUIT BREAKER DRAW OUT TYPE
AIR CIRCUIT BREAKER DRAW OUT TYPE ELECTRICALLY OPERATED
COMBINATION CIRCUIT BREAKER TYPE NON-REVERSING MAGNETIC MOTOR STARTER, SIZE AS NOTED
COMBINATION CIRCUIT BREAKER TYPE REVERSING MAGNETIC MOTOR STARTER SIZE AS NOTED
FIELD WINDING
POWER TRANSFORMER
POTENTIAL TRANSFORMER
CURRENT TRANSFORMER
DISCHARGE RESISTOR OR RESISTANCE TYPE HEATER
N.O. CONTACT
N.C. CONTACT
FUSE
INDUCTION MOTOR 480V, 3φ, 60~ MOTOR SIZE AS INDICATED
CONTROL SWITCH
RHEOSTAT
PLUG CONNECTION
EXCITER
AMMETER
VOLTMETER
WATTMETER
WATTHOUR METER
POWER FACTOR METER
FREQUENCY METER
FIELD AMMETER
FIELD VOLTMETER
SYNCHROSCOPE
OVERCURRENT RELAY
OVERCURRENT GROUND RELAY
DIRECTIONAL OVERCURRENT RELAY
DIFFERENTIAL RELAY
VOLTAGE CONTROLLED OVERCURRENT RELAY
TEMPERATURE INDICATOR
TURBINE GEN.
1250 KW
GOV. MOTOR
DISCH. RES.
41
VOLT. REG.
MANUAL
REMOTE CONTROL
DIESEL
DIESEL
GENERATOR BREAKER
2000A
2000A DIESEL GENERATOR BREAKER
2000A
MAIN CIRCUIT BREAKER
480V, 2000A., 3φ, 4W, 60~
600A
BATTERY CHARGER
BATTERY
BATTERY CHARGER
INVERTER
INVERTER
120 V.D.C. PANEL
115V AC PFD. PWR.
SECONDARY AUX. SKID
HOT WASTE TANK AREA
ROD DRIVE CONTROL
PRESS HEATER CONTROL
CONTROL CONSOLE LIGHTS
GRAPHIC PANEL
PRIMARY AUX. SKID
OXYGEN ANALYZER
PRESS MONITORING PANEL
CONDENSER SKID
CONTROL CONSOLE
SWITCH GEAR TRIP
INSTRUMENT SHOP
NUCLEAR INST. RACK
LABORATORY
RADIATION MONITOR
CONTROL CONSOLE
BF3 LIFTING MECHANISM
TURBINE LUBE OIL PUMP
WASTE DISPOSAL PUMP
EVAPORATOR SOLIDS PUMP
WASTE DISCHARGE PUMP
WASTE COMPRESSOR
WASTE PROCESS
V C VENT FAN
PRIMARY MAKEUP PUMP
PRIMARY MAKEUP PUMP
PRIMARY COOLANT PUMP
CONTROL ROD DRIVES
4 KW.
4 KW.
4 KW.
4 KW.
4 KW.
PRESSURIZED HEATERS
SEAL LEAKAGE PUMP
PRIMARY AUXILARIES
MAIN GLYCOL
MAIN GLYCOL
AIR BLAST COOLER FAN
AIR BLAST COOLER FAN
AIR BLAST COOLER FAN
AIR BLAST COOLER FAN
AIR BLAST COOLER FAN
AIR BLAST COOLER FAN
HOT LAB VENTILATOR
SECONDARY AUXILARIES
CONDENSATE PUMP No. 1
CONDENSATE PUMP No. 2
BOILER FEED PUMP No. 1
BOILER FEED PUMP No. 2
CONDENSER
TO HEATING VENTILATION
ALCO
ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N.Y. U.S.A.
NONE
R.D.B.
1-17-61
ELECTRICAL ONE LINE DIAGRAM PWR
AEL-743

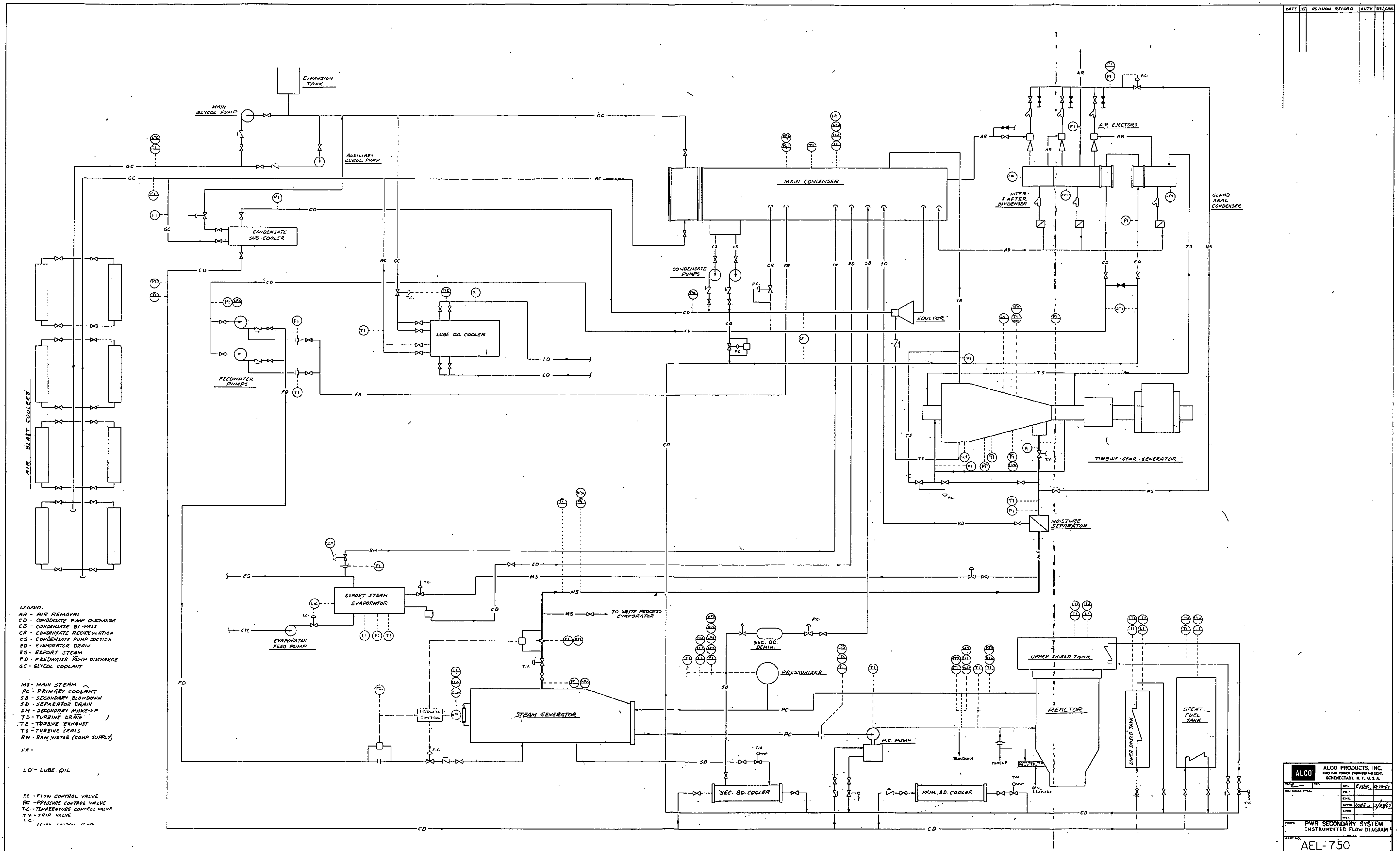


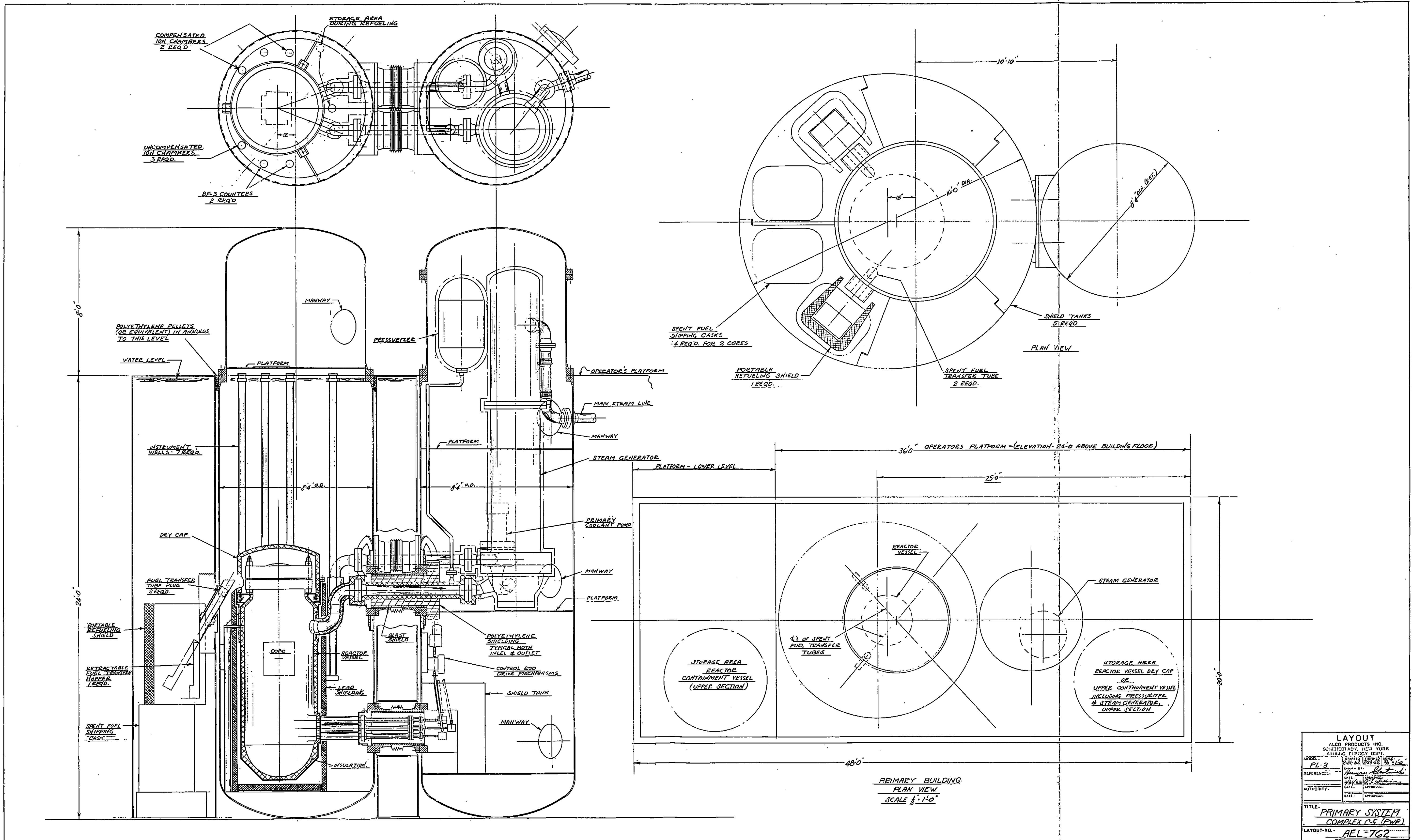
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL- PL-3	STARTED 1-18-62	FINISHED	SCALE- 3/4"-1'-0"
REFERENCES-	DRAWN BY- MUNAFO		
	DATE- 2/23/62	APPROVED- WBEnright Jr	
AUTHORITY-	DATE-	APPROVED-	
	DATE-	APPROVED-	

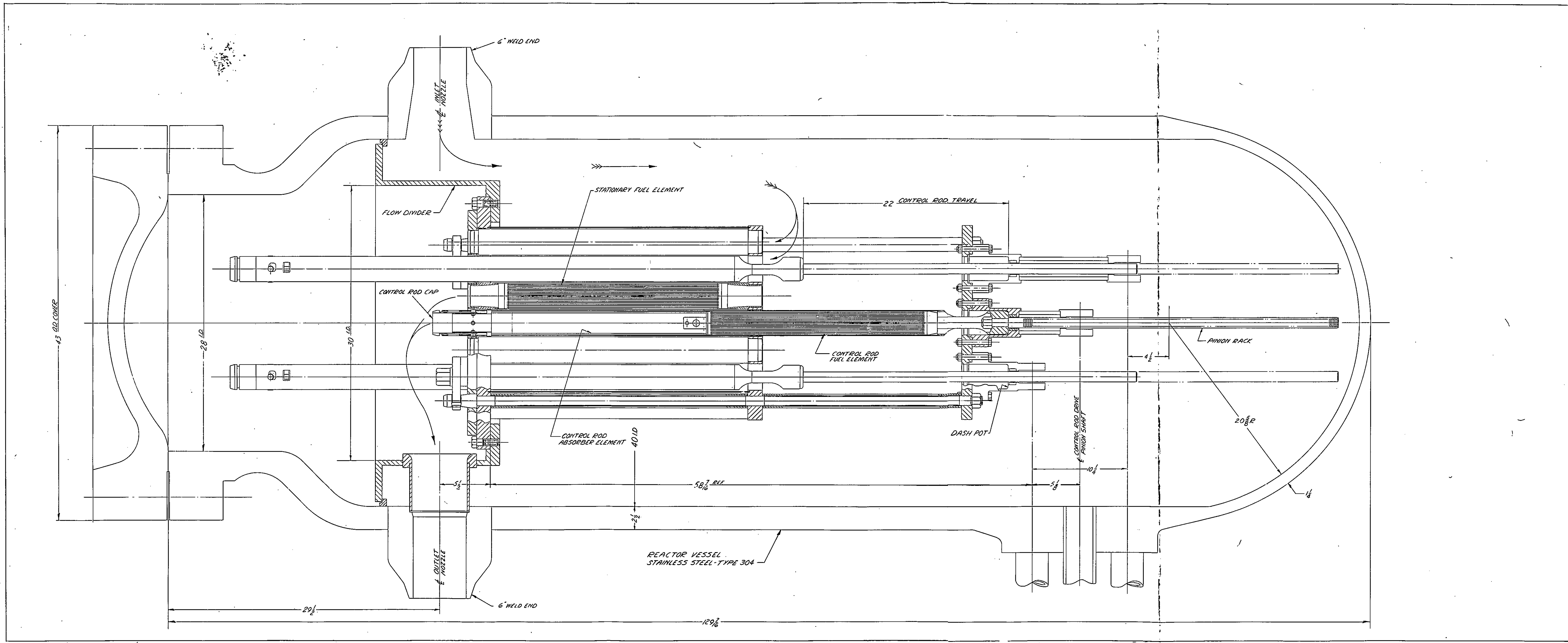
TITLE- WASTE PROCESSING SKID

LAYOUT NO.- AEL-747


EXPANSION TANK
MAIN GLYCOL PUMP
AUXILIARY GLYCOL PUMP
AIR BLAST COOLERS
CONDENSATE SUB-COOLER
MAIN CONDENSER
CONDENSATE PUMPS
EDUCTOR
AIR EJECTORS
INTER- & AFTER CONDENSER
GLAND SEAL CONDENSER
FEEDWATER PUMPS
LUBE OIL COOLER
TURBINE-GEAR-GENERATOR
MOISTURE SEPARATOR
EXPORT STEAM EVAPORATOR
EVAPORATOR FEED PUMP
TO WASTE PROCESS EVAPORATOR
SEC. BD. DEMIN.
PRESSURIZER
STEAM GENERATOR
UPPER SHIELD TANK
REACTOR
LOWER SHIELD TANK
SPENT FUEL TANK
P.C. PUMP
SEC. BD. COOLER
PRIM. BD. COOLER
BLOWDOWN
MAKEUP
SEAL LEAKAGE
LEGEND:
AR - AIR REMOVAL
CD - CONDENSATE PUMP DISCHARGE
CB - CONDENSATE BY-PASS
CR - CONDENSATE RECIRCULATION
CS - CONDENSATE PUMP SUCTION
ED - EVAPORATOR DRAIN
ES - EXPORT STEAM
FD - FEEDWATER PUMP DISCHARGE
GC - GLYCOL COOLANT
MS - MAIN STEAM
PC - PRIMARY COOLANT
SB - SECONDARY BLOWDOWN
SD - SEPARATOR DRAIN
SM - SECONDARY MAKE-UP
TD - TURBINE DRAIN
TE - TURBINE EXHAUST
TS - TURBINE SEALS
RW - RAW WATER (CAMP SUPPLY)
FR -
LO - LUBE OIL
F.C. - FLOW CONTROL VALVE
P.C. - PRESSURE CONTROL VALVE
T.C. - TEMPERATURE CONTROL VALVE
T.V. - TRIP VALVE
L.C. - LEVEL CONTROL VALVE
DATE | LTR. | REVISION RECORD | AUTH. | DR. | CHK.
ALCO
ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N. Y., U. S. A.
DR. E.H.W. 12-29-61
APPD. 2/23/62
PWR SECONDARY SYSTEM
INSTRUMENTED FLOW DIAGRAM
PART NO. AEL-750



COMPENSATED ION CHAMBERS 2 REQ'D
STORAGE AREA DURING REFUELING
UNCOMPENSATED ION CHAMBERS 3 REQ'D.
BF-3 COUNTERS 2 REQ'D
POLYETHYLENE PELLETS (OR EQUIVALENT) IN ANNULUS TO THIS LEVEL
WATER LEVEL
PLATFORM
MANWAY
PRESSURIZER
OPERATOR'S PLATFORM
MAIN STEAM LINE
INSTRUMENT WELLS - 7 REQ'D.
STEAM GENERATOR
PRIMARY COOLANT PUMP
DRY CAP
FUEL TRANSFER TUBE PLUG 2 REQ'D
PORTABLE REFUELING SHIELD
RETRACTABLE FUEL TRANSFER HOPPER 1 REQ'D.
SPENT FUEL SHIPPING CASK
CORE
REACTOR VESSEL
LEAD SHIELDING
BLAST SHIELD
POLYETHYLENE SHIELDING TYPICAL BOTH INLET & OUTLET
CONTROL ROD DRIVE MECHANISMS
SHIELD TANK
INSULATION
10'-10"
16'-0" DIA.
8'-4" DIA. (REF.)
SPENT FUEL SHIPPING CASKS 4 REQ'D. FOR 2 CORES
PORTABLE REFUELING SHIELD 1 REQ'D.
SPENT FUEL TRANSFER TUBE 2 REQ'D.
SHIELD TANKS 5 REQ'D.
PLAN VIEW
36'0" OPERATORS PLATFORM - (ELEVATION 24'-0 ABOVE BUILDING FLOOR)
PLATFORM - LOWER LEVEL
25'0"
REACTOR VESSEL
STEAM GENERATOR
℄'S OF SPENT FUEL TRANSFER TUBES
STORAGE AREA REACTOR CONTAINMENT VESSEL (UPPER SECTION)
STORAGE AREA REACTOR VESSEL DRY CAP OR UPPER CONTAINMENT VESSEL INCLUDING PRESSURIZER & STEAM GENERATOR, UPPER SECTION
20'0"
48'0"
PRIMARY BUILDING PLAN VIEW SCALE 1/8" = 1'-0"
8'0"
24'0"
8'4" O.D.
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.
MODEL - PL-3
STARTED 2-21-62
FINISHED 2-23-62
SCALE - 3/4" = 1'-0"
REFERENCES -
DRAWN BY - Herman Glostwicki
DATE - 3/24/62
APPROVED - R. E. Williams
AUTHORITY -
DATE -
APPROVED -
TITLE - PRIMARY SYSTEM COMPLEX C-5 (PWR)
LAYOUT-NO. - AEL-762

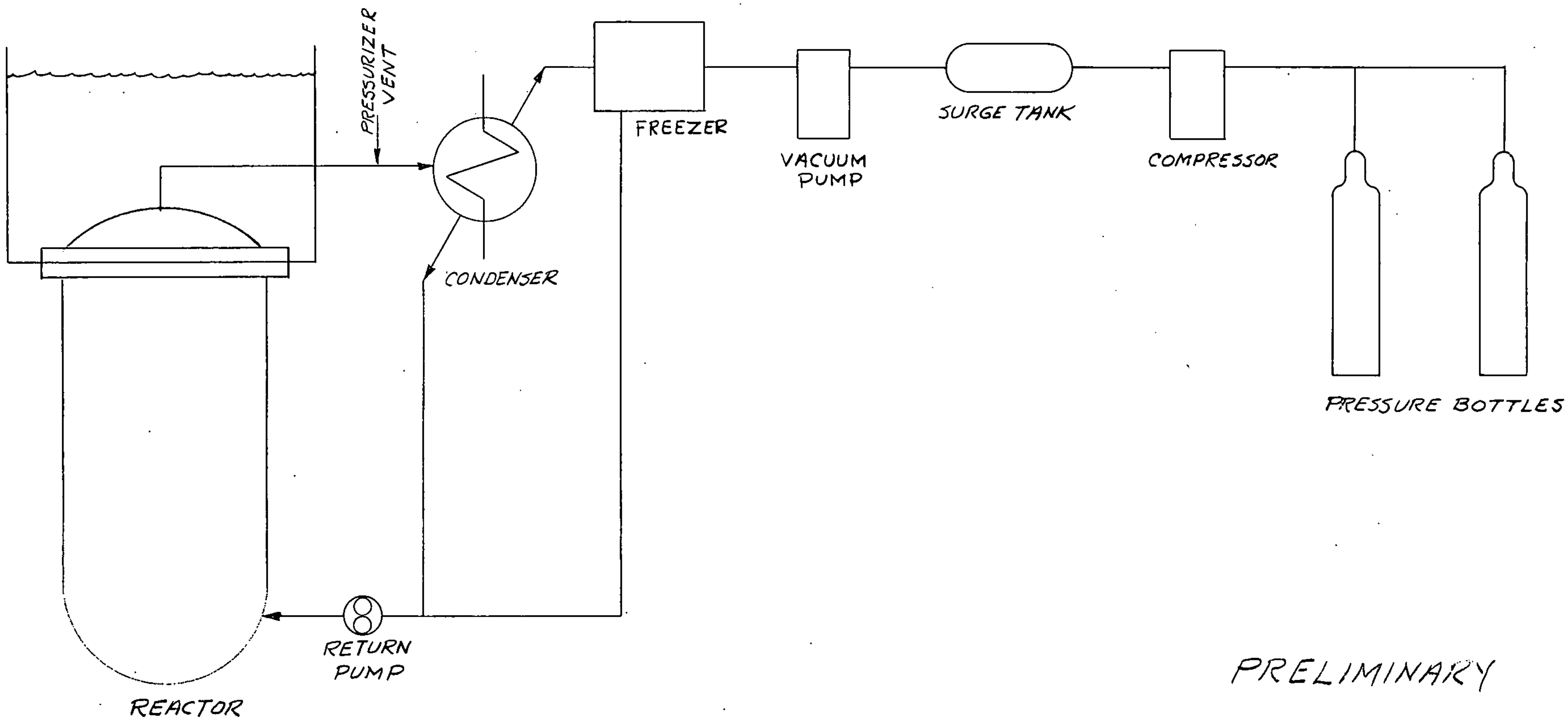


LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL - PL-3
STARTED 2-14-62 FINISHED 2-20-62 SCALE - HALF
DRAWN BY - S. J. Lee
REFERENCES -
DATE - 2-26-62 APPROVED -
AUTHORITY -
DATE - APPROVED -
DATE - APPROVED -

TITLE - REACTOR VESSEL & CORE VERTICAL SECTION

LAYOUT NO. - AEL-763



PRELIMINARY

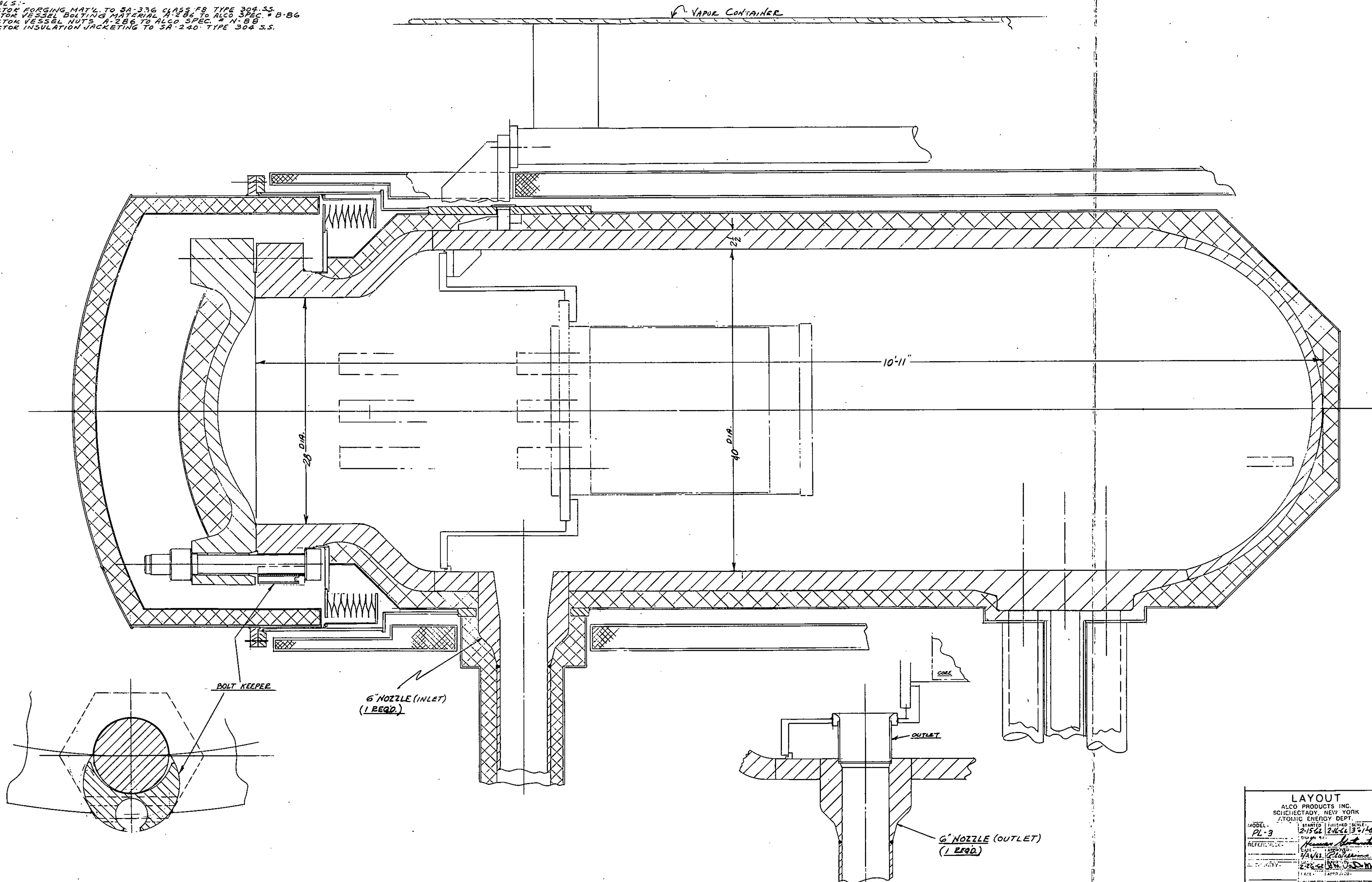
PWR GASEOUS WASTE DISPOSAL SYSTEM

AES-596

WEIGHTS

REACTOR VESSEL, INCLUDING DRIVE ROD MOUNTING PAD & TUBES & PRIMARY PIPING WELDED TO VESSEL	– 14,540#
REACTOR VESSEL INSULATION & INSULATION JACKET INCLUDING DRIVE ROD TUBES & PRIMARY PIPING	– 1900#
REACTOR SUPPORT LUGS & RING	– 1015#
BELLOWS & SEAL ASSEMBLY	– 680#
PRE-ASSEMBLED SHIPPING WGT. (LESS SKID & PACKAGING)	– 18,135#
REACTOR CLOSURE INCLUDING INSULATION & JACKET	– 1950#
DRY CAP & INSULATION	– 1110#
CLOSURE NUTS & BOLTS	450#

MATERIALS:-
REACTOR FORGING MAT'L. TO SA-336 CLASS F8 TYPE 304 S.S.
REACTOR VESSEL BOLTING MATERIAL A-286 TO ALCO SPEC. # B-86
REACTOR VESSEL NUTS A-286 TO ALCO SPEC. # N-88
REACTOR INSULATION JACKETING TO SA-240 TYPE 304 S.S.



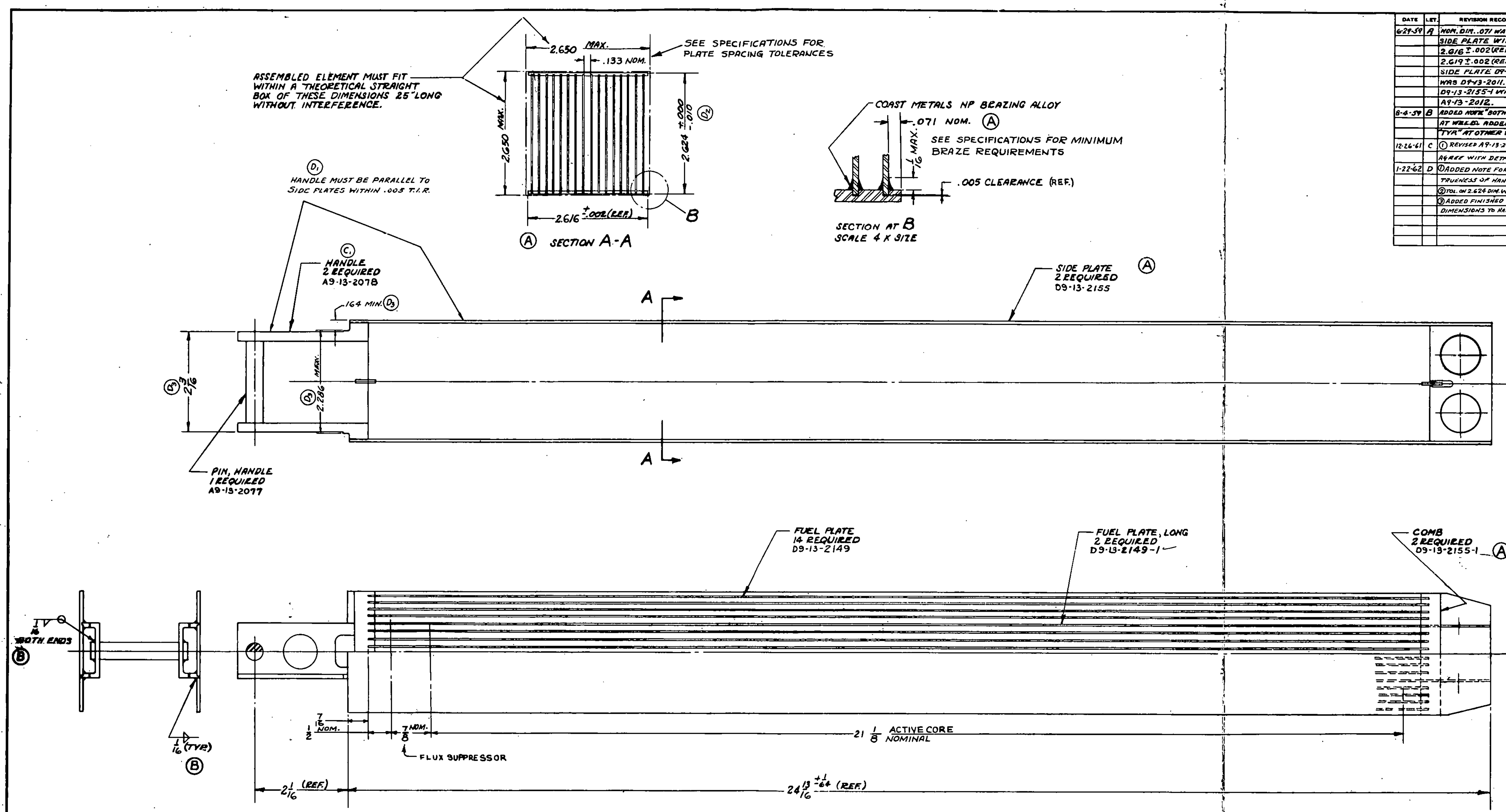
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL- PL-3	STARTED 2-15-62	FINISHED 2-16-62	SCALE- 3"=1'-0
REFERENCE-	DRAWN BY- Herman Gostanski		
	DATE- 2/26/62	APPROVED- R. L. Williams	
AUTHORITY-	DATE- 2-26-62	APPROVED- S. M. VanDeMark	
	DATE-	APPROVED-	

TITLE- REACTOR VESSEL

LAYOUT NO.— AES-612

DATE	LET.	REVISION RECORD	AUTH.	DR.	CHK.
6-29-59	A	NOM. DIM. .071 WAS .073.			
		SIDE PLATE WIDTH			
		2.616 ± .002 (REF) WAS			
		2.619 ± .002 (REF).			
		SIDE PLATE D9-13-2155			
		WAS D9-13-2011. COMB			
		D9-13-2155-1 WAS			
		A9-13-2012.	[illegible]	[illegible]	[illegible]
8-4-59	B	ADDED NOTE "BOTH ENDS"			
		AT WELD. ADDED NOTE			
		"TYP." AT OTHER WELD.	[illegible]	[illegible]	[illegible]
12-26-61	C	(1) REVISED A9-13-2078 TO			
		AGREE WITH DETAIL (ECO 0380)	J.P.C.	BP	CGD
1-22-62	D	(1) ADDED NOTE FOR (ECO-0389)			
		TRUENESS OF HANDLE			
		(2) TOL. ON 2.624 DIM. WAS ±.000 -.022			
		(3) ADDED FINISHED			
		DIMENSIONS TO HANDLE	J.P.C.	ROB	[illegible]



METHOD OF ASSEMBLY

1- HELI-ARC HANDLES TO SIDE PLATES AS SHOWN.
2- ASSEMBLE SIDE PLATES (WITH HANDLES), FUEL PLATES AND COMBS.
3- FURNACE BRAZE FUEL PLATES TO SIDE PLATES AND COMBS TO FUEL PLATES.
4- HELI-ARC PIN TO HANDLES.
5- BREAK SHARP EDGES AND REMOVE ALL BURRS.

PART NO. D9-13-1030

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES. TOLERANCES ON FINISHED FRACTIONAL DIMENSIONS TO BE ± 1/64

√ FINISH AS INDICATED IN MICROINCHES.
(/1) MACHINE FINISH - ROUGH
(/10) FLAME CUT OR SAW

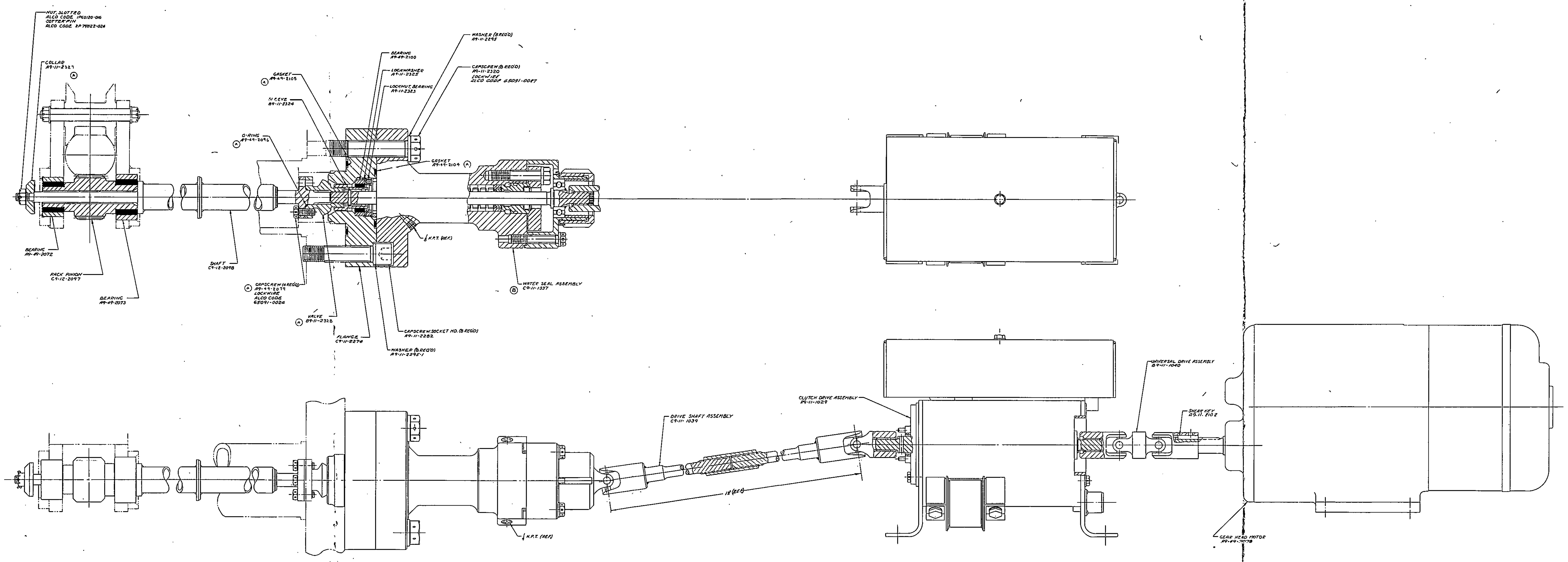
ALCO

ALCO PRODUCTS, INC.
ATOMIC ENERGY DEPT.
SCHENECTADY, N. Y., U. S. A.

SCALE FULL	REF.	DR.	E Bayer 6-17-59
MATERIAL SPEC.		TR.	
		CHK.	[illegible] 6-17-59
		APPR.	[illegible] 6-17-59
		APPR.	[illegible] 1/5/62
		MET.	[illegible] 6-18-59

NAME FUEL ELEMENT (CONTROL ROD)

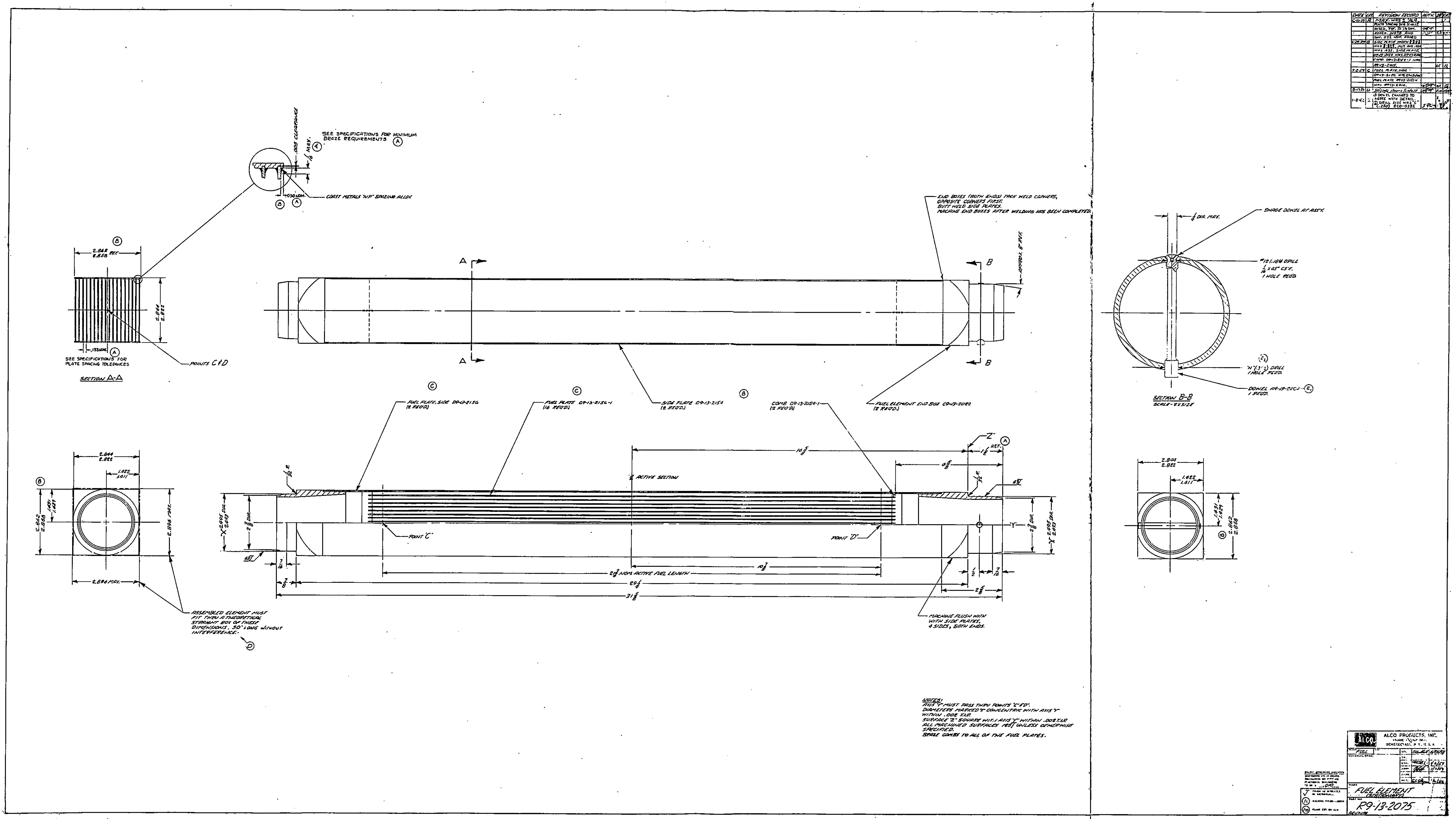
PART NO. D9-13-1030



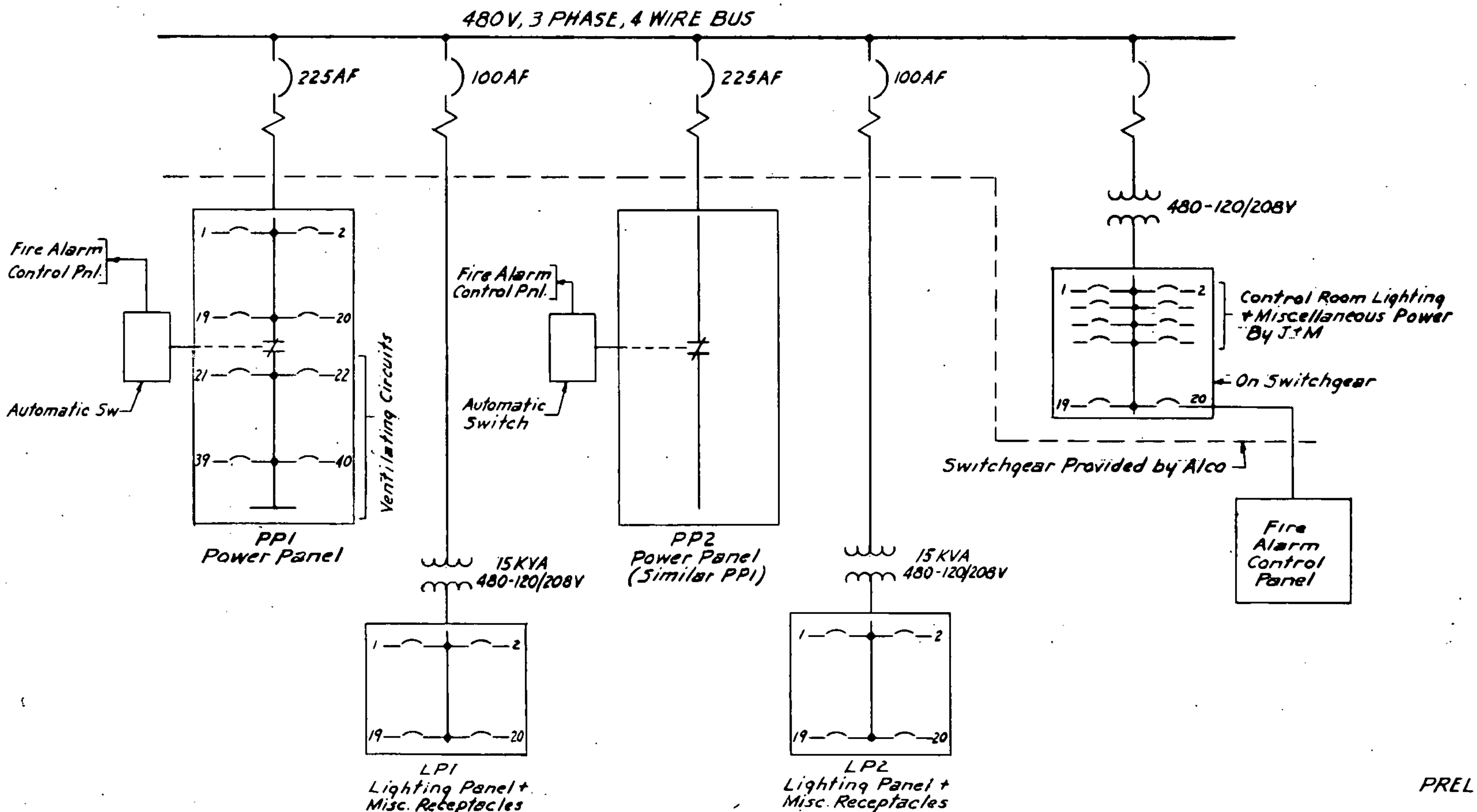
ALCO PRODUCTS, INC.
ATOMIC ENERGY DEPT.
SCHENECTADY, N. Y., U. S. A.

CONTROL ROD DRIVE
SHT 2

R9-11-1030



SEE SPECIFICATIONS FOR MINIMUM BRAZE REQUIREMENTS
COAST METALS "NP" BRAZING ALLOY
SEE SPECIFICATIONS FOR PLATE SPACING TOLERANCES
POINTS C & D
SECTION A-A
END BOXES (BOTH ENDS) TACK WELD CORNERS, OPPOSITE CORNERS FIRST. BUTT WELD SIDE PLATES. MACHINE END BOXES AFTER WELDING HAS BEEN COMPLETED.
SWAGE DOWEL AT ASSY.
SECTION B-B
FUEL PLATE, SIDE
FUEL PLATE
SIDE PLATE
COMB
FUEL ELEMENT END BOX
ACTIVE SECTION
POINT "C"
POINT "D"
ACTIVE FUEL LENGTH
ASSEMBLED ELEMENT MUST FIT THRU A THEORETICAL STRAIGHT BOX OF THESE DIMENSIONS, 30" LONG WITHOUT INTERFERENCE.
MACHINE FLUSH WITH WITH SIDE PLATES, 4 SIDES, BOTH ENDS.
NOTES:
AXIS "Y" MUST PASS THRU POINTS "C" & "D".
DIAMETERS MARKED "X" CONCENTRIC WITH AXIS "Y" WITHIN .002 T.I.R.
SURFACE "Z" SQUARE WITH AXIS "Y" WITHIN .002 T.I.R.
ALL MACHINED SURFACES 125 UNLESS OTHERWISE SPECIFIED.
BRAZE COMBS TO ALL OF THE FUEL PLATES.
ALCO PRODUCTS, INC.
SCHENECTADY, N.Y., U.S.A.
SCALE FULL
MATERIAL SPEC.
FUEL ELEMENT (STATIONARY)
R9-13-2075



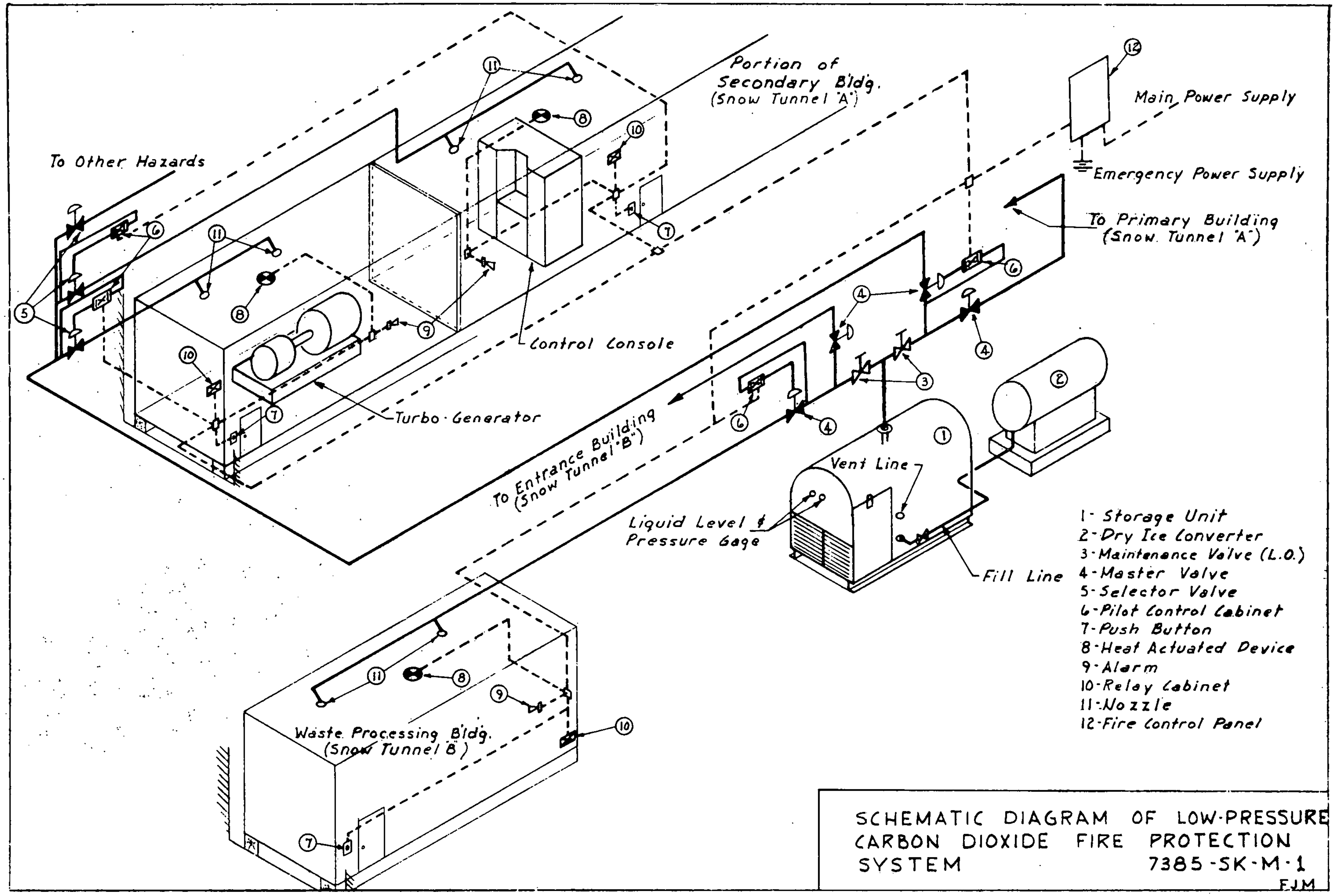
PRELIMINARY

NO	DESCRIPTION	CHKD.	APPD.	DATE
REVISIONS				

JACKSON & MORELAND Inc., ENGINEERS

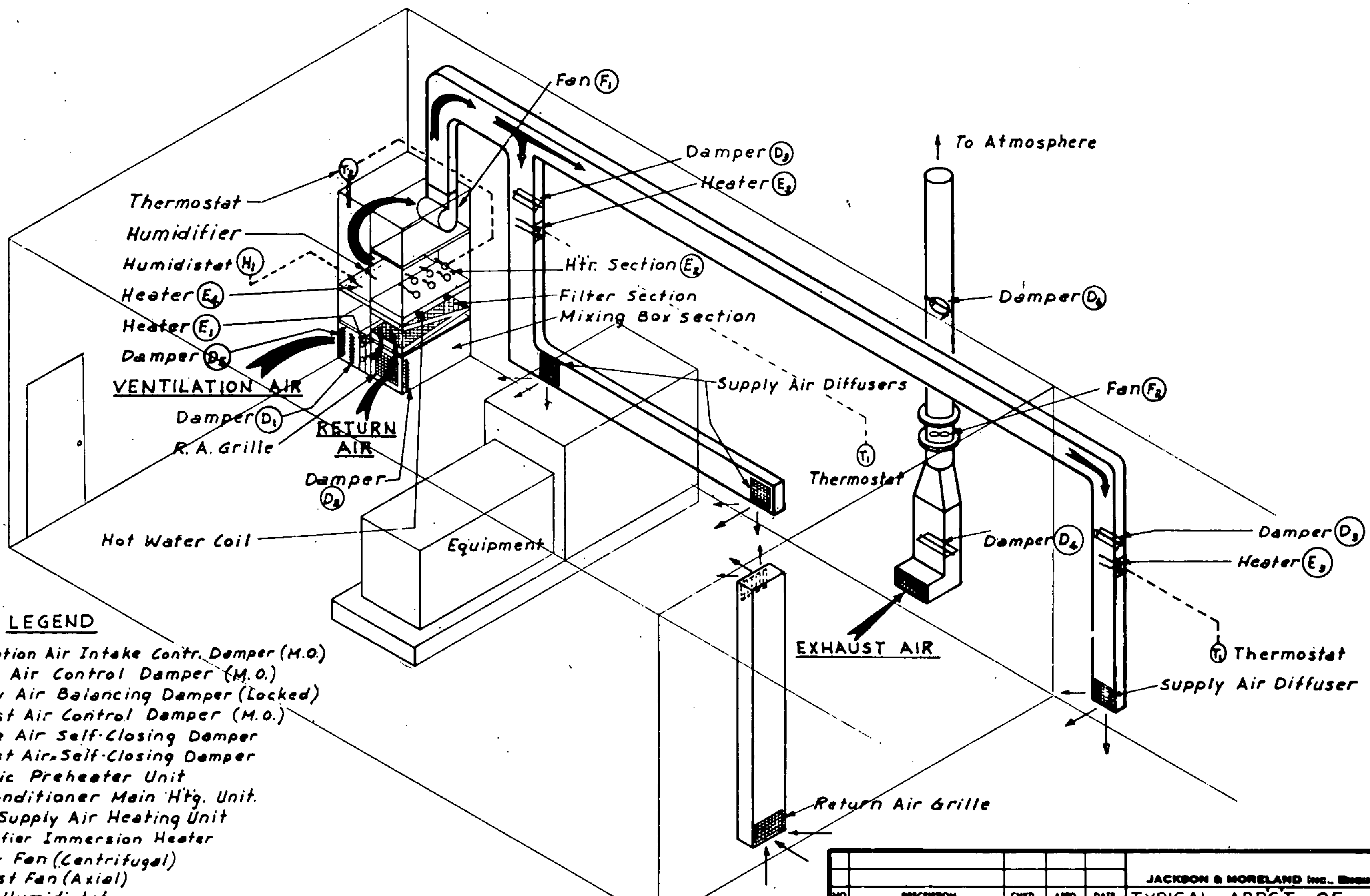
ONE LINE DIAGRAM
LIGHTING & MISC. POWER

SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	NO.
	DS				1-10-62	7385-2	7385-SK-E-1



SCHEMATIC DIAGRAM OF LOW-PRESSURE CARBON DIOXIDE FIRE PROTECTION SYSTEM 7385-SK-M-1

FJM



LEGEND

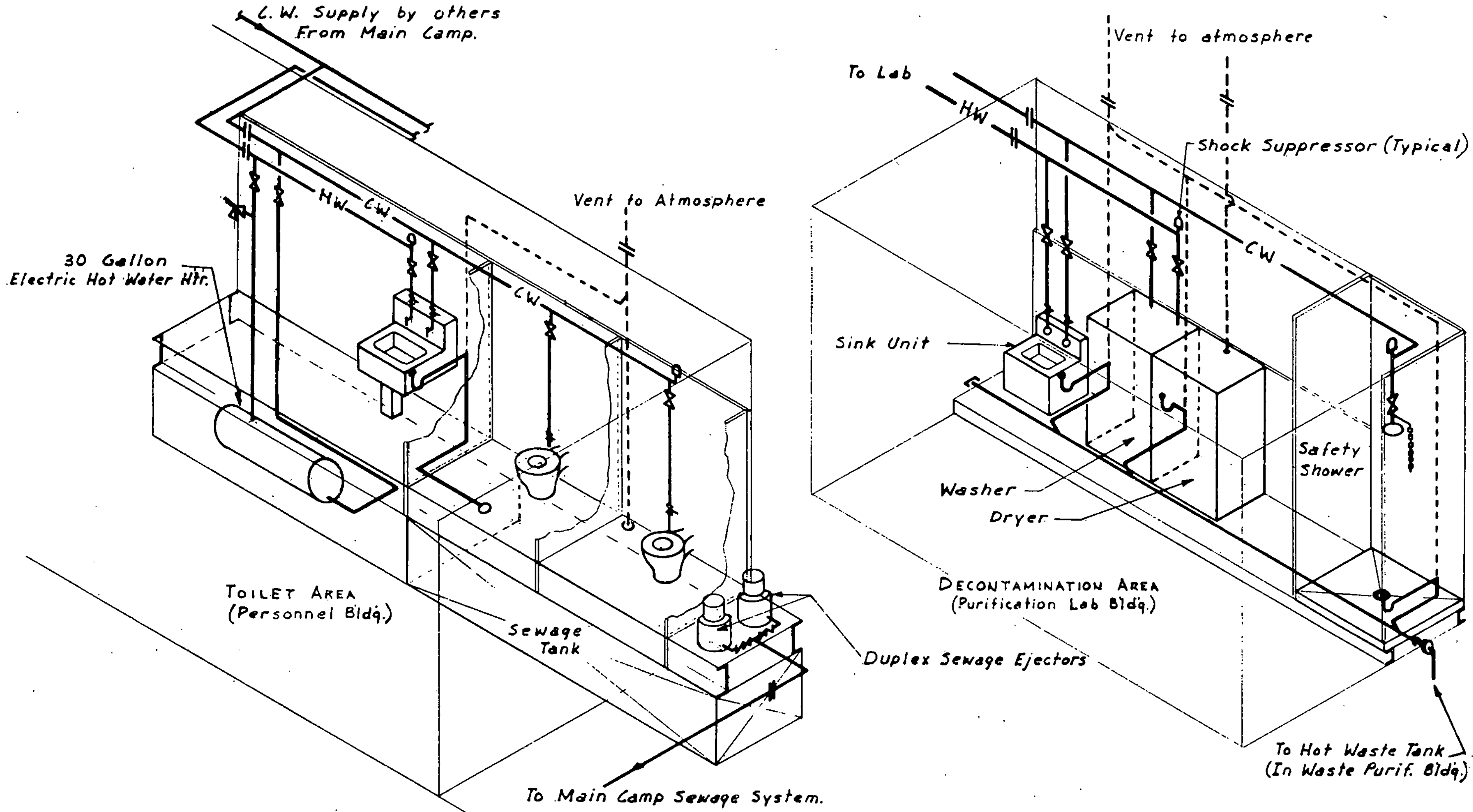
- (D1) Ventilation Air Intake Contr. Damper (M.O.)
- (D2) Return Air Control Damper (M.O.)
- (D3) Supply Air Balancing Damper (Locked)
- (D4) Exhaust Air Control Damper (M.O.)
- (D5) Intake Air Self-Closing Damper
- (D6) Exhaust Air Self-Closing Damper
- (E1) Electric Preheater Unit
- (E2) Air Conditioner Main Htg. Unit.
- (E3) Room Supply Air Heating Unit
- (E4) Humidifier Immersion Heater
- (F1) Supply Fan (Centrifugal)
- (F2) Exhaust Fan (Axial)
- (H1) Room Humidistat
- (T1) Room Thermostat
- (T2) Bulb-Type Duct Thermostat

NO	DESCRIPTION	CHKD.	APPD.	DATE
REVISIONS				

JACKSON & MORELAND Inc., Engineers

TYPICAL ARRGT OF HEATING & VENTILAT'G SYS

SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	NO.
NONE	FJM					7385	7385-SK-M-2

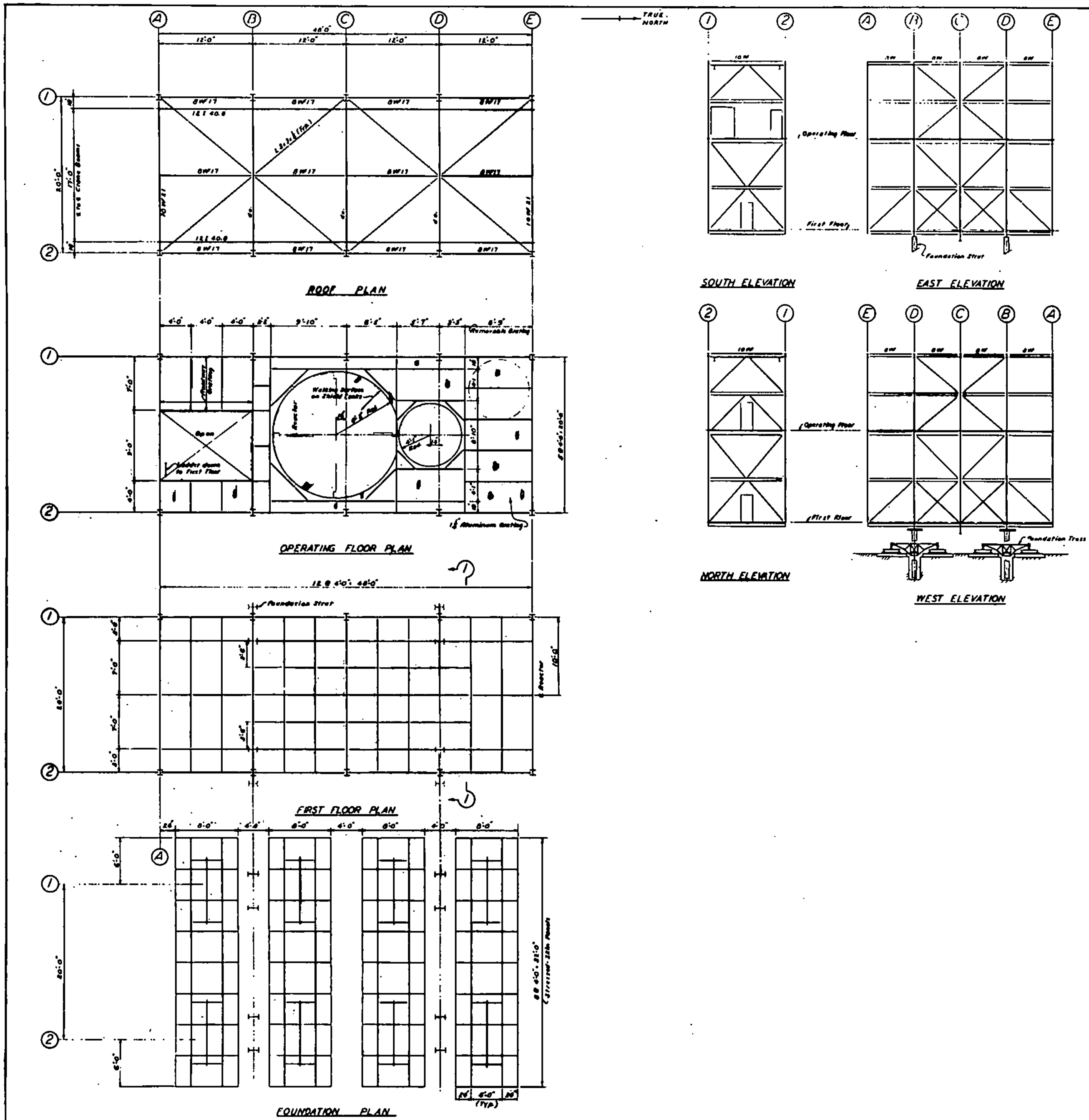


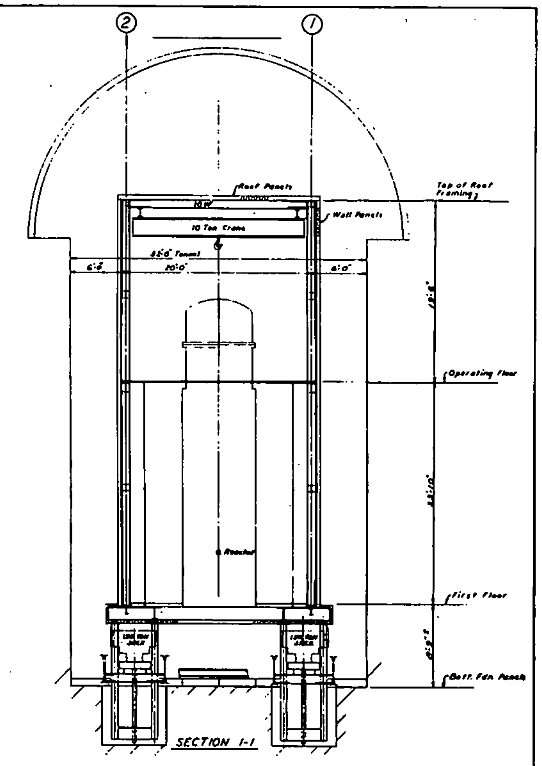
NOTES:
1. All fixtures · st. steel · marine type.
2. Toilet area skid · st. steel W/st. st. facing.
3. Decontamination skid · st. steel W/st. st. facing.

SCHEMATIC ARRANGEMENT OF PLUMBING & DECONTAMINATION SYSTEMS.

7385 · SK · M · 3

FJM





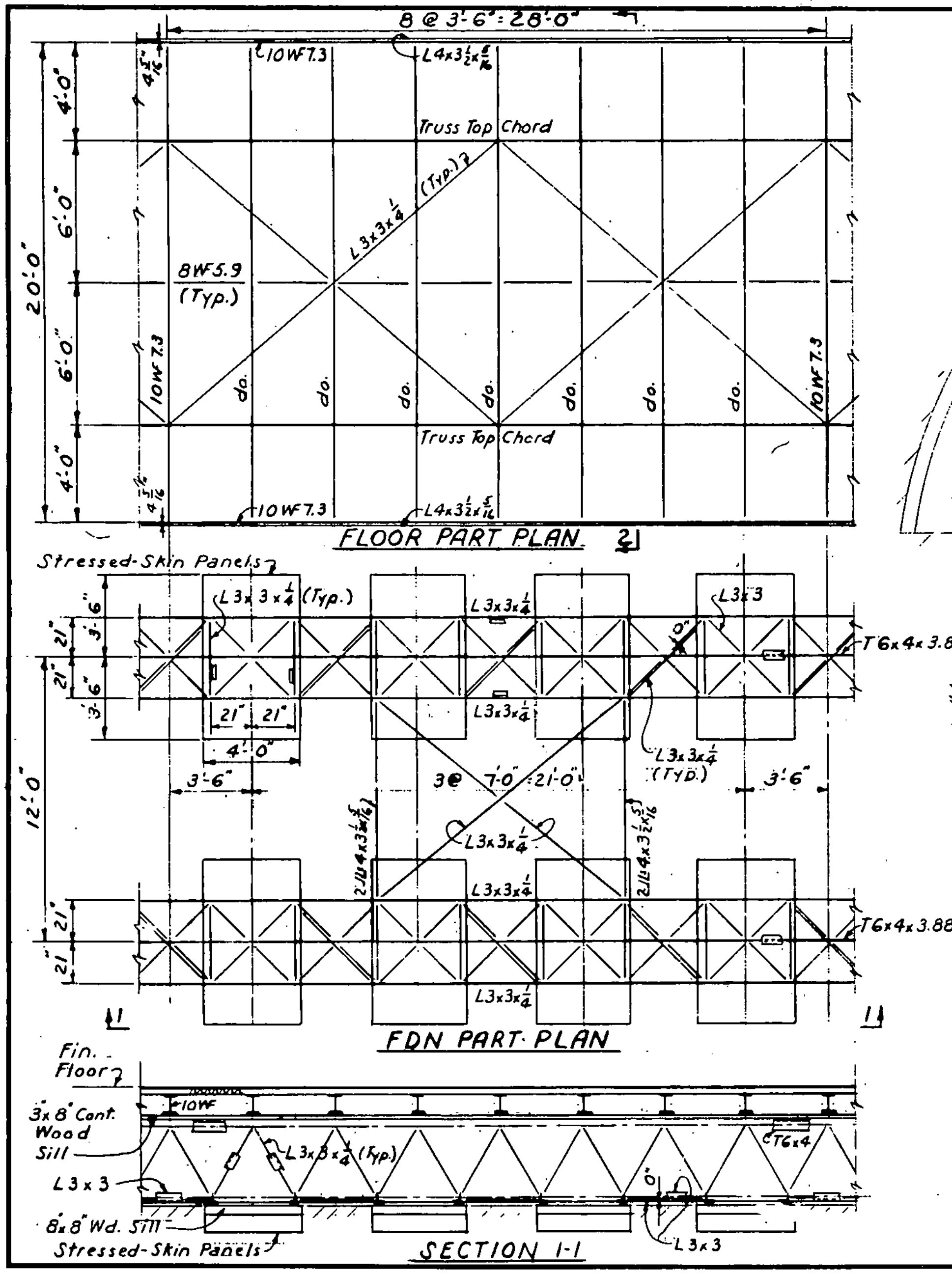
NOTES:
1. This drawing illustrates an arrangement of Framing and Foundations for the Pressurized Water Reactor Primary Building.
2. All Framing is U.S.S.-T1 steel, or equal.
3. Assumed Primary Equipment Weight (including water): 700^K
4. Design Live Loads
Roof: 5 p.s.f. + 10 p.s.f. pipe load.
Operating Floor: 100 p.s.f. + 2^K concentrated load (random location)
First Floor: 100 p.s.f. + 2^K concentrated load (random location)

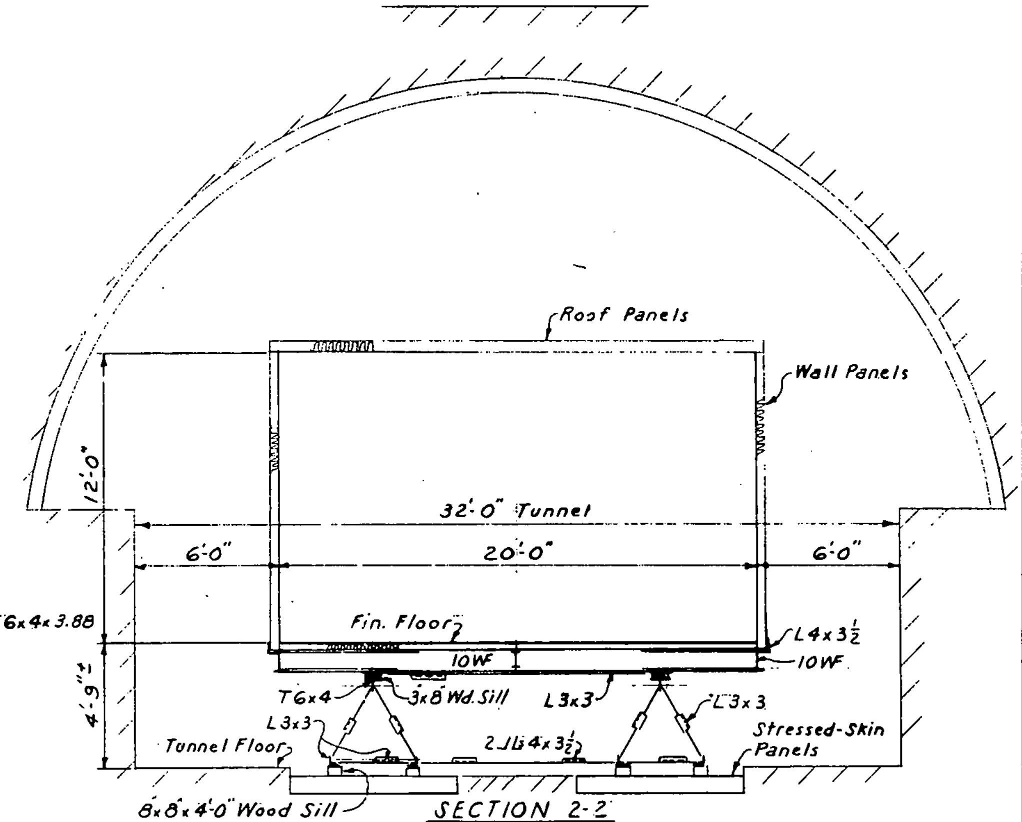
2/16/62

ALCO PRODUCTS INC
JACKSON & MORELAND, INC., ENGINEERS BOSTON — NEW YORK

PL 3
PRESSURIZED WATER REACTOR PRIMARY BUILDING

7385 | 3/16" = 1 FOOT | 7385-SK-S-9





NOTES:

1. This drawing illustrates a preliminary design of a portion of the Maintenance and Storage Area of the Entrance Building.
2. All Framing is 6061-T6 Aluminum.
3. Design Live Loads:
   Roof: 5 p.s.f. +10 p.s.f pipe load.
   Floor: 200 p.s.f. 2 Kip concentrated load (random locations)

NO	DESCRIPTION	CHKD.	APPD.	DATE
	REVISIONS			

ALCO PRODUCTS INC.
JACKSON & MORELAND INC., ENGINEERS
PL 3
ENTRANCE BUILDING

SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	NO.
3/16" = 1'-0"					2-13-62	7385	7385-SK-S-11