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APAE 115
Volume I

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

BWR reference 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 1
AEC Research and Development Report
UC-81, Reactors, Power
(Special Distribution)

BWR REFERENCE DESIGN FOR PL-3

Approved by:

G. E. Humphries, Project Engineer

Issued: February 28, 1962

Contract AT(30-1)-2900
with U. S. Atomic Energy Commission
New York Operations Office

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ABSTRACT

The natural circulation, direct cycle, boiling water reactor reference design presented in this report is the alternate to the preferred preliminary design developed under Phase I of the PL-3 contract. The 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 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, 1962, 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 discussion 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 preferred 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 justification for this selection is 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 BWR and PWR design descriptions are presented respectively in this document and APAE-115, Volume 2, "PWR Preliminary 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, drawing 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 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 reference design of a natural circulation direct cycle boiling water reactor power plant. Its design conforms to the requirements, development objectives and criteria described in the Technical Provisions attached to the PL-3 contract.

The preferred BWR design is a modification of PL-2. The principal modifications in the system layout evolved from the 20,000 lb weight limit for each shipping module. Appended to this reference design report is a description of an alternate reactor design for this plant which uses full enrichment flat-plate fuel elements. This design was developed to satisfy module weight limitations because initial design studies indicated that the larger rod core used in PL-2 could not be accommodated without exceeding the module weight limitation for the pressure vessel. The flat-plate core design is included to present fully the BWR engineering analysis and design performed during Phase I.

The information provided in this report conforms to the requirements set forth by the AEC Contracting Officer for the alternate to the preferred preliminary design developed during Task 2. The design effort devoted to the BWR was limited by contractual requirements to development of data necessary for selection of a preferred plant. The data needed to present a comprehensive preliminary design were not developed once the preferred PWR was selected.

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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 are 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

1. The plant shall have a net output of 1,000 KWe at 0.8 pf (3 phase, 480 volt, 60 cycle) plus 1.5×10^6 Btu/hr of thermal energy.
2. The reactor shall be light water cooled and moderated, either of the forced circulation pressurized type or natural circulation boiling (direct cycle) type.
3. The plant shall utilize a conventional condensing steam cycle with turbine generator with cycle waste heat dissipated to the air.
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.
5. The plant shall be capable of being transported in preassembled modules by C-130 aircraft operating under Antarctic conditions.
6. The plant shall be capable of being installed and operated in snow tunnels at Byrd Station, Antarctica.
7. The plant shall be capable of being installed with equipment which can be transported by C-130 aircraft operating under Antarctic conditions.
8. The plant shall be inherently safe to operate and maintain.
9. The reactor plant shall be contained.
10. The plant when installed in the Antarctic shall have a high degree of assurance of satisfactory operation.
11. The plant shall not release radioactivity of a type and magnitude that would adversely affect future scientific studies in the Antarctic.

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

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

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

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.

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

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

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

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.

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

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

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

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.

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

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:

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

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

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

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

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.

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

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

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.

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

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.

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

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.

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 percent of the activity of the end-of-life fuel elements.

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

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.

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

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

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.

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

19. Sufficient control rod worth shall be provided to make the reactor subcritical at atmospheric pressure and a coolant temperature of $4^{O}C$ 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.

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.

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.

22. The plant under normal operating conditions, shall not release radioactive wastes above the following concentrations: gaseous, $4 \times 10^{-14} \mu c/cc$; liquid 10^{-8} $\mu c/cc$.

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 percent of the fission product inventory of a spent core.

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 ft. from other plant components shall not exceed 1.2 millirem/hr.

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.

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 BWR PLANT CONSIDERED

The two BWR plants evaluated during Task 1 of Phase I were a direct cycle natural circulation BWR plant similar to PL-2 and an indirect cycle BWR plant. This latter concept was proposed to take advantage of the isolation afforded the primary and secondary systems while retaining the simplicity of the natural circulation BWR. The retention of radioactivity in the primary of an indirect cycle BWR for the PL-3 application would simplify the processing of waste material and ease the design problems associated with the release of radioactivity to the Antarctic environment. Unfortunately, the comparative analysis performed during the evaluation indicated that the indirect BWR inherits many disadvantages of both the direct cycle BWR and the PWR without offering any significant advantages. In particular it does not offer the simplicity and compactness or the improved steam conditions of the direct cycle BWR.

The increased primary system pressure attending the use of the isolating primary-secondary steam generator in the indirect BWR restricted the diameter of the reactor pressure vessel in order to meet the weight limitation of 20,000 pounds for the pressure vessel shipping module. Using the rod-type PL-2 core, the downcomer flow area between the core shroud and the pressure vessel wall was reduced with a corresponding increase in the predicted flow velocity of water in the downcomer. Design conservatism with regard to the effects of steam carryunder predicated a smaller flow velocity, a condition which could be better satisfied by reducing the core diameter. An alternate core design utilizing full enrichment flat plate fuel elements was developed with the characteristically smaller core diameter desired for this situation. Although the indirect BWR concept was rejected from further consideration for PL-3, the smaller core size of the full enrichment plate-type core offered obvious weight advantages which might enhance the design of the direct cycle BWR for PL-3.

Two reactor designs for the direct cycle BWR were sustained during the initial preliminary design period. One design was a modification of PL-2 which retained identically the low enrichment PL-2 rod-type core. The other utilized the full enrichment plate-type fuel elements. These elements are similar to those of the SM-2, the principal changes being the increased element length and larger coolant gaps between plates with concomitant reduction in the number of plates per element. The design selected for the direct cycle BWR employs the rod-type core used originally for PL-2. This design is described in the main body of this report. The alternate plate-type core design is described in an appendix to this report. The differences and advantages of these cores are discussed in the following sections.

3.2 SELECTION OF PREFERRED BWR CONCEPT

3.2.1 Inherent Differences Due to Enrichment

The PL-3 design criteria require a minimum core lifetime of two years, which can be met readily with either a highly enriched core or a low enrichment core.

The differences in fuel volume requirements and U^{235} density within the fuel material at these extreme conditions of enrichment, together with U^{238} resonance considerations, the properties of available fuel materials, and present fuel technology, lead quite naturally to distinctive fuel types - - to thin plates consisting of UO_2 dispersed in stainless steel in the case of full enrichment, or to thick rods of low enrichment UO_2 pellets, as in PL-2. The distinction between rod-type fuel and plate-type fuel in this study implies this difference in enrichment.

The relative advantages of equivalent boiling water reactors using these two fuel concepts may be summarized as follows:

1. greater lifetime potential and reduced fuel cycle costs with low enrichment.

2. smaller core and vessel with full enrichment.

3. less fission product release associated with clad defects in a stainless steel - UO_2 plate than from UO_2 rods.

3.2.2 Evaluation

Comparison of the modified PL-2 design with the plate-type reactor, and evaluation with regard to the inherent advantages enumerated above, indicate the following:

3.2.2.1 Lifetime and Relative Economy

PL-2 is designed for 25.5 megawatt-years, equivalent to a 4-year core lifetime at the PL-3 conditions. The reference plate-type core is predicted to have an endurance of 2.3 years at the same operating conditions. The relative annual fuel cycle costs based on these lifetimes are indicated to be $120,000 for PL-2 and $226,000 for the plate-type core, with fuel use charges neglected.

Fuel cycle costs were also estimated for the PL-2 design loaded for the same lifetime as the plate-type core, using a speculative reduction in enrichment and adjustment for plutonium.[1,2] Under these conditions, the rod-type core appeared to offer little economic advantage.

3.2.2.2 Vessel Size

With sufficient downcomer area provided to assure that no more than about 5 weight percent of the steam will be carried under, the vessel internal diameter is 53-1/2 in. for the plate-type reactor and 62 in. for the rod-type reactor.

The relative weights of carbon steel vessels without their closures are 15,700 pounds for the plate-type core, based on a preliminary design pressure of 750 psi, and 15,050 pounds for the rod-type core with the design pressure relaxed to 700 psi. Both designs meet the packaged shipping weight and size limitations, and they involve the same number of modules.

3.2.2.3 Fission Product Release

Anticipated differences in fission product release from the two fuels were found to have no effect on the required waste disposal facilities. This is because the specified design condition of 1 percent release and the specified limit on airborne activity concentration require that identical equipment, including a krypton collection system, be provided in either case.

Radiation dosage from power plant equipment resulting from fuel failure should be greater from rod-type fuel because of the greater release expected. However, the dosage would be similar in either case if it is postulated that 1 percent of the accumulated fission products escape.

3.2.3 BWR Selection

Compared on the basis of a minimum core life of two years, the plate-type core was regarded as marginally preferable. The reduction in pressure vessel module weight attending the use of the plate-type core was critically important initially because the pressure vessel module weight for the rod-type core exceeded the 20,000 pound shipping weight limit. The lower dose rates anticipated from clad failure of the plate-type fuel reinforced the arguments favoring development of the plate-type core design for the BWR.

Subsequent interpretation of the core lifetime criterion strongly emphasized the desirability of extended lifetime. The development of a pressure vessel design for the rod-type core which satisfied the shipping weight criterion erased the one definitive objection to this design for PL-3. The longer core life also eases the logistics problem attending shipment of spent fuel from a remote site.

Thus the rod-type reactor design was preferred for the reference BWR plant described in this report. The core is identical to the PL-2 design, the time schedule and scope of the Phase I study precluding the detailed reactor analysis necessary to investigate and confirm any improvements in the design of this portion of the plant.

3.3 REFERENCES

1. Pearce, W. R., "Choice of Full vs. Slight Enrichment for the Direct Cycle BWR," Internuclear, M-WRP-61-107, December, 1961.

2. Pearce, W. R., "Added Fuel Cost Data Applicable to Choice of Full vs. Slight Enrichment for PL-3 BWR", Internuclear, M-WRP-62-101, January, 1962.

4.0 PLANT DESCRIPTION

4.1 TUNNEL AND BUILDING ARRANGEMENT

The plant is located in two tunnels parallel to an existing tunnel leading from the camp as shown in Dwg. AEL-738. These tunnels are 27 ft wide at the base and all buildings located in these tunnels are 20 ft wide.

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.

Within the two plant tunnels are three building-enclosed areas 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 auxiliaries equipment) and 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. Other equipment located in this tunnel include the chemical laboratory, emergency diesels, maintenance and personnel facilities.

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 reactor power plant consists of a direct-cycle natural circulation boiling water reactor, a steam turbine-generator plant which uses directly the steam generated in the reactor, a glycol-to-air heat rejection system, and the attendant auxiliary and service systems. A summary of plant data is presented in Table 4.1.

TABLE 4.1
PLANT DATA SUMMARY

Reactor Thermal Power	8.1 Mw
Gross Electrical Output	1500 Kw
Net Electrical Output	1000 Kw
Low Pressure Export Steam Flow	1.5 x 10^6 Btu/ hr
Core Lifetime	4 years
Fuel Type	UO_2 Pellets in rod
Fuel Enrichment	4.8 w/o
Fuel Loading	1140 kg U
Reactor Pressure	615 psia
Steam Flow	25,520 lb/hr
Feedwater Temperature	154°F
Condenser Vacuum	6 in. Hg
Design Ambient Temperature	34°F
Number of Planeloads Required:	
Buildings and Foundations	28
Reactor Plant	24
Power Conversion Equipment and Auxiliaries	22
Construction Equipment and Supplies	4
Emergency Diesel Fuel Oil	30
Total	108

The reactor complex is housed in a primary building approximately 20 ft wide x 30 ft long x 40 ft high. This building houses only the reactor complex or steam producing equipment of the plant. It is located within one of the two parallel plant tunnels next to and in line with the power conversion equipment, but separated from it by a snow wall barrier as shown in the plant layout, Dwg. AEL-738.

The reactor complex is housed in a single vapor container. The vapor container for this system is designed to sustain a static pressure equal to the reactor operating pressure. Nuclear instrumentation and shielding are located around the outside of the vapor container. Inside the primary building there is an overhead bridge crane that is used during the erection of the reactor complex and during fuel handling operations.

The arrangement of the power plant equipment and plant auxiliaries is also shown in Dwg. 738. 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 18.7×10^6 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 eKw 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.

Beyond the 15 ft laydown area is a secondary auxiliaries module. The export steam evaporator (1.5×10^6 Btu/hr duty) and subcooler for auxiliary cooling are placed on this skid. Pumping components are a main and an auxiliary glycol coolant circulating pump, and two feedwater pumps. Two small storage tanks (5 gals) and metering pumps are necessary for chemical treatment. A tank for 100 gallons of condensate storage is provided. This storage is additional to that provided in the condenser hotwell. A motor control center is provided for pump motors.

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. diameter 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 seconds 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 reactor complex shown on Dwg. AEL-740 consists of one vertical vapor container assembly which houses the entire reactor system. The reactor complex is 15 ft dia x 27 ft 11 in. high and is contained in a building 20 ft wide x 30 ft long x 40 ft high. An overhead bridge crane in the reactor building is utilized for the assembly of the reactor complex and for subsequent fuel handling operations.

The reactor vessel is supported in the lower section of the flanged containment vessel. The upper containment vessel is provided with ports utilizing hinged covers for access to the control rod drive power units. The containment vessel design is based on low volume and high pressure to reduce the overall height requirements of the reactor complex installation.

The primary shielding is located external to the containment vessel. It consists of an 8 ft dia water shield tank which houses lead shielding outside the lower containment vessel. Four peripheral type water shield tanks arranged around the 8 ft dia shield tank results in a symmetrical shield 15 ft in. diameter. Four water shield tanks around the upper containment vessel and polyethylene plugs at the control rod drive access ports complete the primary shield complex.

Access for refueling the reactor necessitates the removal of the upper shielding and the upper containment vessel. A fuel transfer cask is utilized to remove the spent fuel from the reactor vessel. The spent fuel is then transferred to shipping casks which are located and stored in the four peripheral lower shield water tanks.

All shipping modules are within the space and weight requirements for air transportability to Byrd Station.

This design utilizes a PL-2 slightly enriched core in a low alloy ferritic steel vessel clad with 1/4 in. stainless steel overlay. The reactor data are summarized in Table 5.1.

5.2 REACTOR VESSEL AND CORE SUPPORT STRUCTURE

5.2.1 Reactor Vessel

The reactor vessel is of a cylindrical shell design. The inside diameter is 62 in. at the core mid-plane, and is reduced at the top closure flange to 51-1/2 in. A hemispherical bottom head is utilized to reduce vessel weight.

TABLE 5.1
REACTOR DATA SUMMARY

General

Thermal Power	8.1 Mw
Steam Flow Rate	25,500 lb/hr
Operating Pressure	615 psia
Feedwater Temperature	$154^{o}F$
Core Lifetime	4 yr
Fuel Type	UO_2 pellets in rods
Fuel Enrichment	4.8 w/o
Fuel Loading	1140 Kg U
Power Density in Coolant	21.5 Kw/liter

Core Description

Active Fuel Length	38.3 in.
Equivalent Core Diameter	36.3 in.
Number of Fuel Assemblies	24
Fuel rods/assembly	59
Poison rods/assembly	3
Assembly Spacing	6.319 in. center to center
Rod Spacing	.732 in. center to center
Rod Diameter	.466 inch
Cladding	.020 in. Type 347 SS

Fuel

Pellet Diameter	.420 in.
Enrichment	4.8 w/o
Fuel Loading	1140 KgU
Initial B^{10} Loading	99.5 gm
Average Fuel Burnup	8200 MWD/MT

Control Rods

Number	9
Shape	Cruciform
Length	53.77 in.
Span	11.44 in.
Thickness	.250 in.
Material	Ag-Cd-In

Thermal and Hydraulic Data

Riser Height	34.5 in.
Coolant Channel Equivalent Diameter	0.867 in.
Core Flow Area	4.16 ft^2
Assumed Downcomer Carryunder	5%
Average Void Fraction	.160
Average Exit Void Fraction in Hottest Region	.307
Downcomer Velocity	1.03 ft/sec
Core Flow Rate	729 lb/sec
Total Steam Generated	7.48 lb/sec
Average Heat Flux	50,140 Btu/hr ft^2
Maximum Heat Flux	232,700 Btu/hr ft^2
Max. Fuel Centerline Temperature	$3080^{o}F$

Nuclear Data

Metal/Water Ratio, Rods In	0.595
Metal/Water Ratio, Rods Out	0.465
Stainless Steel/UO_2 Ratio, Rods Out	0.55
Reactivity in Voids	5.1% $\Delta k/k$
K_{eff}, Initial, No Xe, 16% Avg. Voids	1.051
Shutdown K_{eff}, Initial, No Xe, $4^{o}C$, Center Rod Out	0.982
Temperature Coefficient at Beginning of Life, $4^{o}C$, Rods Out	$-2.42 \times 10^{-5}\ \Delta\rho/^{o}F$

Reactor Vessel and Head

Design Pressure and Temperature	700 psi, $500^{o}F$
Maximum Inside Diameter	62 in.
Vessel Wall Thickness	1-3/8 in.
Head Thickness	2-1/2 in.
Internal Height to Flange	15 ft
Material:	
Forgings	SA 350, Grade LF-3
Plate Stock	SA 203, Grade E
Cladding	1/4 in. 304 SS

Containment Vessel

Design Pressure and Temperature	600 psi, $600^{o}F$
Maximum Inside Diameter	71 in.
Wall Thickness	1-1/4 in.
Overall Height	24 ft, 4 in.

The internal height of the vessel is 15 ft without the vessel closure. Flanged nozzle penetrations are provided for the steam, feedwater and purification lines. All penetrations in the vessel are above the reactor core, thereby minimizing the possibility of core meltdown in the event of a pipe rupture.

A conventional bolted flange vessel closure, utilizing a 2:1 elliptical head, provides penetrations for nine control rod drive mechanisms, three water level control instruments and an access port for handling the capsule neutron source.

The steam dryer is mounted integrally in the vessel closure. The separated moisture collects at the bottom of the dryer and then drains back into the vessel. The top of the closure utilizes concrete pellets canned to the level of the flanged penetrations for neutron shielding.

The reactor vessel and head will be fabricated from SA-350 Grade LF-3 (3-1/2% nickel steel) forgings and SA-203 Grade E (3-1/2% nickel steel) plate stock, in accordance with the latest editions of Section II and Section VIII of the ASME Boiler & Pressure Vessel Code, and all other applicable code cases. The vessel will be code stamped. Design pressure and temperature of the vessel are 700 psig and 500°F respectively. The operating pressure is 600 psig at 489°F.

All material in contact with primary water will be weld clad with 1/4 in. (minimum) Type 304 stainless steel. All penetrations such as for the primary piping will be SA-350-LF-3 clad with Type 304. Downcomers will be Type 304. The gasket mating surfaces will be clad with Type 304 and the octagonal cross section gasket will be Type 304. Studs will be Type 403 stainless steel with Type 304 nuts.

Materials supplied will be thoroughly inspected and tested. The forgings will be supplied with an intentional end projection that may be removed for testing, or a second forging from the same heat will be supplied. This material will be tested for chemical composition, room and elevated temperature tensile strength, yield strength, percent elongation, and percent reduction in area. A 45° Charpy "V" impact value of 30 ft lb @ -90°F will be required on all forgings, plate material and weld metal. In addition, end quench data, grain size, and thermal critical temperatures will be determined. All welding will be qualified and tested in accordance with Alco nuclear welding specifications.

The specification for the reactor vessel material including all welding, will be essentially as detailed in APAE No. 107, "Effect of Radiation Damage on SM-1, SM-1A and PM-2A Reactor Vessels."

5.2.2 Core Support Structure

The core support structure is an independently fabricated structure installed and bolted in place within the reactor vessel. It provides a means of accurately locating and supporting the stationary fuel elements and control rods inside the reactor vessel. It is constructed of Type 304 stainless steel and mounted in the reactor vessel so as to direct and distribute the primary cooling water through and around the stationary fuel elements and control rods. Rigid quality control will be maintained throughout to insure the use of satisfactory materials and approved fabrication techniques.

5.3 VAPOR CONTAINMENT VESSEL

A low volume, high design pressure vessel is utilized for containment of the reactor vessel and associated components. This vapor containment vessel will consist of two sections, an upper and a lower, both constructed of either SA-350 Grade LF-3 or SA-203 Grade D or E, in accordance with the ASME Code for Unfired Pressure Vessels. The design temperature and pressure will be 600°F and 600 psi respectively. Cladding of the containment vessel with an austenitic stainless steel will not be required.

The lower containment vessel (71 in. I. D. x 15 ft high) houses and supports the reactor vessel. The upper containment vessel (64 in. I. D. x 9 ft 4 in. high) houses the reactor vessel cover and the control rod drive mechanisms. Hinged covers are provided for maintenance access to the power units of the drive mechanisms. A man hole employing a hinged cover is provided for access into the vapor container for the purpose of servicing the rod drive seals, electrical and piping connections. The upper and lower containment vessel design pressures are the same as the operating pressure of the reactor system, 600 psi. Lower containment vessel penetrations employ coded expansion joints to maintain containment integrity and allow for thermal expansion of reactor vessel piping within the containment vessel. Motor-operated isolation valves are closed in the event of an accident to maintain containment integrity at the steam and purification line penetrations. A check valve in the feedwater line is located near the expansion joint and serves the same purpose as the steam and purification line isolation valves.

The vapor container gaskets will be spiral wound stainless steel and asbestos. The bolting material will be AISI 4140 with Type 304 stainless steel nuts. The same high quality fabrication and inspection techniques required for the reactor vessel will be applied to the vapor container.

5.4 REACTOR CORE

The PL-3 direct cycle BWR employs a PL-2 slightly enriched rod type core as designed by Combustion Engineering. This core consists of 24 fuel assemblies and 9 cruciform control rods. (See Dwg. PL-2-J-2258).

Core specifications for this PL-3 direct cycle BWR will be developed from the existing PL-2 specifications. A core vendor will be selected by competitive bid for the fabrication of the two required cores. The vendor will be required to qualify his fabrication procedure before the start of actual core production. The method for qualification of procedures will be stated in the PL-3 core specifications. All materials employed will be certified and tested as required by these specifications. Proper supervisory and production control will also be required during core fabrication to insure rigid adherence to the approved fabrication procedures.

5.4.1 Fuel Assemblies

Each fuel assembly consists of 59 fuel elements or rods, 3 poison elements, 2 end plates, ferrules, spacers and appropriate end fixtures for lifting and guiding the assembly.

The fuel elements (rods) which make up a PL-2 fuel assembly employ a column of pressed and sintered low enrichment ceramic grade UO_2 fuel pellets hermetically sealed in a low cobalt Type 347 stainless steel tube with welded end plugs. The poison elements are 1% boron stainless steel rods which are also contained in low cobalt Type 347 stainless steel tubes, filled with helium and sealed in the same manner as the fuel tubes. The fuel assembly is formed by brazing these components into an integral unit.

The brazed fuel assemblies are suspended from and laterally positioned by sixteen stainless steel stanchions. These stanchions are rigidly held in place by an upper and lower grid.

Around the core is a flow shroud which is attached to the stanchions. Above the upper grid is a riser section. The flow shroud, the outer sections of the upper grid, and the riser section serve to separate the downcomer region from the core in the reactor, thereby defining the natural circulation coolant flow path.

The fuel assemblies are held down by their own weight (approximately 200 lbs each). This is over ten times the maximum upward hydraulic force which exists during operation.

A neutron startup source is positioned in the downcomer region at the mid-plane of the core. The source support structure and locking device are such that the source can be installed and removed through a port in the closure head.

5.4.2 Control Rods

Nine cruciform shaped control rods are employed in the PL-3 core. These cruciform control rods are composed of an alloy of silver, cadmium and indium, nickel plated for corrosion resistance, and sheathed with Type 347 stainless steel for mechanical strength and scuffing resistance. The length and width of the control rods are such that when the blade is fully inserted, the poison material completely overlaps the entire active fuel section.

The control rods, which are suspended from the control rod drive mechanism, are laterally positioned and guided by the upper grid and stanchions.

5.5 CONTROL ROD DRIVES

The top mounted control rod drive mechanism consists of a rack and pinion actuator mounted within a pressure housing, which in turn is mounted on the reactor vessel head. Direct mechanical drive through the pressure housing is made possible by use of a controlled leakage labyrinth breakdown seal. All actuator assemblies are interchangeable with the exception of the center actuator, which has a longer extension shaft. The power units, each consisting of a clutch, instrument assembly, and drive motor, are mounted horizontally on a retractable support. During normal operation, the magnetic clutch is energized to drive the control rod blade or to hold it in a fixed position. Scram is initiated by de-energizing the magnetic clutch. An over-running clutch is installed so as to drive in the direction of rod insertion only in the event of a stuck rod. Continuous position indication is provided by a synchro transmitter driven through a set of reduction gears by the clutch output shaft. A set of limit switches, driven by the same shaft, is used to provide both signals and motor shutoff at the extremes of rod travel. The power units can be serviced through access ports in the upper containment vessel. An access port in the upper containment vessel is provided for access to the water seals for servicing. An alternate means for access to the water seals would necessitate the removal of the upper containment vessel. However, past performance records on the labyrinth-type water seals used on other reactors indicate very little required maintenance.

Control rod drive materials will be essentially the same as currently used in the SM-1, SM-1A and PM-2A reactors.

All materials will be reviewed to assure that the best materials, in light of latest technological advances, are employed in these units. Close quality control will be maintained during material procurement and fabrication.

5.6 SHIELDING ARRANGEMENT

The compact reactor package, consisting of the reactor vessel and the lower containment vessel is installed in a water shield tank 8 ft OD and approximately 13 ft high. Additional gamma shielding is provided by a 6 in. thick cylindrical slab of canned lead beneath the lower containment vessel to prevent excessive radiation heating in the snow, and by a cylindrical annulus of canned lead surrounding the lower containment vessel with a thickness of 2-1/2 in. at core mid-plane reducing to a thickness of 1 in. at the upper section. This will permit access around the primary shield at 2 hr after reactor shutdown. Four peripheral water shield tanks of tongue and groove design surround the 8 ft dia shield tank to form a symmetrical shield tank 15 ft dia x 15 ft high.

Polyethylene slabs in the void formed by the top of the 8 ft dia shield tank, the containment vessel and the four peripheral shield tanks complete the neutron shielding up to the level of the operators platform, which is located adjacent to the vapor container parting flanges.

Four removable upper shield tanks are utilized to complete the neutron shielding around the reactor complex. These are required to meet the neutron leakage limitation on the plant. Two of the four tanks provide a cylindrical shield tank 12 ft OD x 82 in. ID x 47 in. high. The other two provide a hemispheral shield tank 12 ft 10 in. high OD x 68 in. ID x 85 in. high. Removable polyethylene plugs are utilized at the rod drive containment housings and the access port.

One notable advantage of this shielding concept is that the containment vessel is completely surrounded by shielding. In the event of an accident, the entire containment vessel, filled with fission product vapors, is shielded and the resulting dose rates in the reactor building are reduced.

The four peripheral lower shield tanks may be used as shipping containers for smaller plant components, thereby reducing the required number of plane loads.

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6.0 THERMAL AND HYDRAULIC ANALYSIS

6.1 INTRODUCTION

The thermal-hydraulic behavior of a boiling water reactor is intimately coupled to the nuclear behavior through the interrelation of power, voids, and reactivity. Complete analysis of the steady-state reactor involves iteration of the nuclear and thermal hydraulic effects. With natural circulation, the thermal-hydraulic analysis itself is an iterative process because the steam generated in the reactor core provides the driving head for the flow of coolant and the flow of coolant affects the steam void content. For a given power distribution the total flow, the flow distribution in the parallel paths, and the void distribution depend on the hydraulic characteristics of the system, and can be determined only by iterative calculation.

The PL-2 design studies included the complete iteration process between nuclear and thermal-hydraulic calculations, with an approximation of separability of the radial and axial power distributions. The analysis undertaken for PL-3 conditions includes only the thermal-hydraulic solution, using the axial and radial power distributions presented in the PL-2 report, and again assumes spatial separability of the power distribution. Despite a reduction in power level, the use of improved slip ratio data, and a difference in carryunder, the average steam void fraction at beginning of life is determined to be identical to that predicted for PL-2.

The natural circulation flow path is composed of core, riser, upper plenum, downcomer and lower plenum. The design considered here differs from the original PL-2 design only in that the reactor vessel has been enlarged to minimize carryunder in the downcomer. A schematic drawing of the convective flow path and a list of the flow areas assumed for this analysis are given in Fig. 6.1.

A vessel internal diameter of 62 in. was selected to obtain a downcomer velocity less than the threshold velocity above which carryunder has been observed to increase rapidly. Below this threshold the weight percent of steam carried under is essentially constant at about 5 percent.[1] A reactor could be designed to operate with the increased carryunder which occurs above the threshold but this would imply increased exit and average void fractions. It is desirable to minimize both of these items from the standpoint of stability and lifetime. The present design effectively avoids the problems of carryunder by providing a large downcomer area, and the resulting penalty in pressure vessel weight is not sufficient to cause the vessel to exceed the allowable weight.

6.2 METHOD OF ANALYSIS

The thermal-hydraulic performance of the core is determined by performing a static pressure balance around the natural convection loop and an energy balance across the core. The pressure balance equates the friction and acceleration pressure losses around the loop to the pressure gain resulting from the different densities in the core, riser and downcomer. The pressure balance provides the relations among the coolant flow rate, void distribution and loop geometry; the energy balance provides the relations among the core power, feedwater conditions and steam flow rate. These relations are solved by an iterative procedure to determine the reactor thermal-hydraulic performance.

In this design study, the STREAC code[2] was used to perform the thermal and hydraulic calculations. STREAC is a steady-state thermal and hydraulic computer code for one-pass natural circulation boiling reactors. The input consists of geometric, hydraulic and thermal descriptions of the reactor, a single axial power distribution and a set of radial multipliers to account for the radial power distribution. The program calculates steam void distribution, inlet velocity, pressure loss and water density distribution for each radial region and for the entire core. Information is also obtained on the behavior of velocity, void fraction and pressure loss as the power is increased in the hottest box.

6.3 DESIGN CONDITIONS AND ASSUMPTIONS

The thermal and hydraulic performance of the PL-2 core was determined for the PL-3 operating conditions at the beginning of life. The analysis differs from that made for the original PL-2 design both in the area of operating conditions and in certain assumptions, correlations, procedures and design restrictions which were used.

With the downcomer area available in the 62-in. vessel, the downcomer velocity is less than the threshold value which, at 600 psi, is reported[(1)] to be 1.2-1.3 ft/sec. Below this threshold the carryunder, which is defined as the percentage of the total steam generated in the core carried into the downcomer, should be about 5 percent. In the analysis the effect of 5 percent carryunder was simulated by increasing appropriately the steam flow and the feedwater enthalpy.

Essential to the determination of the local steam void fraction in two-phase flow is the slip ratio, or the ratio of local steam velocity to liquid velocity. The present analysis employed the latest ANL correlation of slip ratio, which included a dependence on channel equivalent diameter as well as quality, superficial velocity, and pressure.

The single-phase expansion and contraction losses were determined from conventional correlations which give the velocity head loss in terms of the area ratio across the expansion or contraction. To account for additional losses due to turbulence in the lower plenum, one half a velocity head loss was added, distributed equally between the downcomer exit and the core inlet. Two-phase contraction losses were based on the conventional loss coefficient increased by the square of the ratio of the core exit to the core inlet liquid velocity, as programmed into STREAC. Two-phase expansion losses were determined by an equation derived from a theoretical momentum balance across the expansion [3] and modified for use in STREAC. All contraction and expansion coefficients were adjusted to give the static pressure loss (or gain) across the change in area rather than only the total pressure change.

Two-phase friction loss in the core channels was calculated using two-phase friction multipliers from the Martinelli-Nelson correlation, [4] modified by an approximate mass flow correction normalized to 0.6×10^6 lb/hr ft^2.

The axial and radial power distributions used were those presented in the PL-2 design report for the beginning of life conditions.

6.4 RESULTS AT REFERENCE CONDITIONS

The results of the thermal and hydraulic analysis at the reference conditions of 8.1 Mw and 154°F feedwater temperature are given in Table 6.1 and 6.2. Table 6.1 summarizes the geometrical parameters of interest and presents the detailed results for the entire core. Table 6.2 gives the values of several thermal and hydraulic performance parameters for each type box or radial region (as determined by the radial power multiplier) and for the average core.

The average steam void fraction of 0.160 agrees with that determined for the original PL-2 design despite the differences in power level, assumptions and correlations. Two major differences were the use in the present analysis of improved slip ratio data and the assumption of 5 percent rather than 15 percent carryunder. The increase in voids due to the lower slip ratios from the latest ANL correlation was approximately compensated by the decrease in voids resulting from the lower value of carryunder. Both analyses assume that any steam carried under is completely quenched at the feedwater ring.

Subcooled voids, which are included in the average void fraction, were approximated by drawing a line tangent to the void fraction versus axial position curve from the point where heating starts. This approximation increases the average void fraction by 0.013 and is thought to be conservative. Figure 6.2 shows the axial void distribution for the four radial regions.

The calculated downcomer velocity of 1.03 ft/sec is significantly below the carryunder threshold of 1.2 to 1.3 ft/sec, indicating that the 62 in. vessel size might be reduced slightly.

TABLE 6.1
THERMAL AND HYDRAULIC DATA

Reactor Power	8.1 Mw
Power Density in Coolant	21.5 Kw/liter
Feedwater Temperature	154°F
Effective Feedwater Temperature (with carryunder)	208°F
Assumed Downcomer Carryunder	5%
Core Flow Area	4.16 ft^2
Core Hydraulic Equivalent Diameter	0.867 in.
Heat Transfer Area	551 ft^2
Riser Height	34.5 in.
Downcomer Flow Area	14.2 ft^2
Total Coolant Flow Rate	729 lb/sec
Total Steam Flow Rate	7.48 lb/sec
Average Exit Quality	1.03%
Core Inlet Subcooling	2.5°F
Downcomer Velocity	1.03 ft/sec
Average Steam Void Fraction	0.160
Exit Steam Void Fraction in Hottest Region	0.307
Stability Multiplier for Hottest Box	1.77
Average Heat Flux	50,140 Btu/hr ft^2
Maximum Heat Flux (4.64 max/avg)	232,700 Btu/hr ft^2
Maximum Fuel Centerline Temperature	3080°F

TABLE 6.2
THERMAL AND HYDRAULIC DATA FOR RADIAL REGIONS

Radial Region	1	2	3	4	Average Core
Number of Boxes	4	8	4	8	24
Radial Power Multiplier	1.62	1.18	0.82	0.62	1.00
Inlet Velocity, ft/sec	3.98	3.76	3.43	3.13	3.53
Fraction Bulk Boiling Length	.866	.837	.799	.768	.812
Exit Quality, %	1.66	1.18	0.80	0.59	1.03
Exit Steam Void Fraction	.307	.251	.194	.157	.220
Avg. Steam Void Fraction	.229	.184	.139	.111	.160

The reactor is designed to operate very much below the burnout heat flux and fuel centerline temperature restrictions. Few measurements of burnout heat flux are available at 600 psi and 1 to 2 percent exit quality but all indications are that the burnout point is considerably above 1 x 10^6 Btu/hr ft^2. The maximum heat flux for this design, based on the PL-2 maximum to average factor of 4.64, is about 233,000 Btu/hr ft^2 and the burnout ratio is at least 4.3. The fuel centerline temperature was calculated at the maximum heat flux and found to be 3080°F, which is considerably below the limiting temperature of 5000°F for similar conditions of fuel burnup.

6.5 STABILITY CONSIDERATIONS

Two sources of instability must be considered in the design of boiling water reactors: hydraulic instability analogous to that observed in electrically heated experiments and coupled nuclear-hydraulic instability. Nuclear-hydraulic stability depends on the characteristics of the closed loop described by the nuclear, hydraulic and thermal parameters for the overall system; it is considered in Section 7.3.

Hydraulic instability can be avoided by designing within the conditions which are observed to cause unstable flow in experiments operated at similar conditions. The best index of stability appears to be the exit steam void fraction, and in this regard the core appears to be designed very conservatively. Experimental work on electrically heated convective loops indicates that stable conditions can be maintained up to exit voids of 65 - 75 percent. Exit voids of approximately 35 percent have been measured on EBWR operating at a power of 42.7 Mw. This was not at the exit from the hottest region of the core, and the reactor subsequently operated stably at power levels up to 62 Mw; therefore exit voids considerably greater than 35 percent have been obtained.

The average exit void fraction from the hottest radial region of the reference reactor is calculated to be about 0.31. The maximum exit void fraction will be somewhat greater due to local power peaking near the control rod water gap. The maximum local exit void fraction is difficult to calculate in the rod-type core but it is estimated to be not more than 35 percent.

Another index of hydraulic stability is given by the STREAC code. This is the factor by which the power in the hottest box would have to be increased in order to cause decreasing coolant flow in that box with all conditions external to the hot box remaining constant. This stability multiplier is calculated to be 1.77 for the reference conditions.

6.6 VARIATIONS OF DESIGN CONDITIONS AND ASSUMPTIONS

In order to evaluate the uncertainties in the thermal and hydraulic design, the relative effects of variations in the major design conditions and assumptions were investigated. The major sources of uncertainity are the largest pressure loss terms, which may be confirmed experimentally in order to predict more accurately the carryunder and the core void fraction and, consequently, the core lifetime.

The relative importance of the assumptions and correlations affecting the pressure loss around the convection loop is evident from Table 6.3 which gives the percentage of the total loss occurring across various portions of the loop. The values were obtained for the average reactor solution at the reference conditions.

TABLE 6.3
COMPONENTS OF TOTAL PRESSURE LOSS

Core Inlet Loss	29.6%
Subcooled Boiling Friction Loss	2.9
Bulk Boiling Friction Loss	20.6
Boiling Acceleration Loss	10.4
Core Exit Loss	29.9
Riser Friction Loss	1.0
Downcomer Loss	5.6
Total Loss (54.5 psf)	100.0%

The pressure gain is split almost equally between the core and the riser -- 49% and 51%, respectively.

The pressure losses due to the expansions and contractions at the core inlet and exit are obviously the most important loss terms in the pressure balance. These loss terms were obtained from the conventional correlations and theoretical relations for simple expansions and contractions. Because of the complex flow path in the reactor, verification of these terms should be made by experimental means. The two next most important losses are the bulk boiling friction and the boiling acceleration losses. The acceleration term is from a theoretical relation, but since this relation is used to account for acceleration losses in determining the bulk boiling friction correlation from experimental measurements, the relation can be considered as accurate. The bulk boiling friction loss depends on a two-phase friction multiplier. These multipliers have been determined experimentally and may be somewhat in error but even a 50 percent error causes only a 10 percent error in the total pressure loss. The remainder of the pressure loss terms make up only 10 percent of the total and need not be considered further.

The effects of varying some of the input design condition correlations and assumptions on the three performance parameters, average void fraction, exit void fraction, and downcomer velocity, are shown in Table 6.4.

TABLE 6.4
EFFECT OF PARAMETER VARIATIONS

	Average Void Fraction	Exit Void Fraction in Hottest Region	Downcomer Velocity, ft/sec
Reference Conditions	.160	.307	1.03
Inlet Loss Coefficient (1.60)			
1.35	.158	.305	1.05
1.85	.161	.310	1.01
Exit Single Phase Loss Coefficient (.99)			
0.75	.157	.302	1.06
1.25	.163	.313	0.99
Two-Phase Friction Multiplier			
10% decrease	.159	.306	1.04
10% increase	.160	.309	1.02
Carryunder (5%)			
0%	.154	.303	1.02
20%	.179	.324	1.07
Slip Ratio			
Old ANL correlation, 10-15% greater than latest correlation	.150	.281	1.01
Power (8.1 Mw)			
8.91 Mw	.168	.322	1.04

For the variations considered, the most significant parameters affecting void fraction are the carryunder, slip ratio and power. The uncertainties in carryunder and slip ratio have been minimized by use of the best information available. The use of the latest ANL slip ratio correlation increases the average void fraction by 0.010 while the reduction in carryunder from 15 to 5 percent decreases the value by 0.012. Other small variations between the original and the present analysis cancel completely any differences in the final average void fraction values. The entrance and exit loss terms, although comprising a major portion of the total pressure loss, do not have a large effect on the void fractions.

The downcomer velocity is primarily dependent on the pressure loss terms, the most important component being the core exit loss which is also probably the most uncertain, again indicating the need for experimental verification of two-phase loss coefficients.

6.7 WATER LEVEL AND STEAM SEPARATION

The height of the reactor vessel above the top of the core is determined by two considerations: the depth of water required above the core during refueling, and the distance required between the riser and the steam dryer to accommodate water level variations and to ensure adequate steam separation during operation. The steam dryer is within the vessel head at about the same elevation as the vessel flange. The refueling water level is also at the flange, with no water provided above the vessel during fuel transfer.

Ten feet of water must be provided over the core during refueling, and the corresponding vessel height requirement is greater than that required for water level variations and steam separation.

The maximum indicated water level is the sum of: (1) the minimum riser submergence at zero power to ensure natural circulation, (2) the amount of water displaced by the steam voids in the core and the riser, and (3) the total water level control band. The values used in the PL-2 design are assumed for item (1) and (3): 6 in. minimum riser submergence and 23 in. total control band between high and low level scram points. The water level rise due to voids in the core and riser is 5 in., based on the area at the 52 in. diameter portion of the vessel.

The actual height of the water-steam interface includes also the swell height caused by the steam voids in the water above the riser. Estimates based on direct interpretation of EBWR data [5, 6] indicate the following: at 8.1 Mw, and with the control band centered at 22-1/2 in. above the riser, the level with swell may vary from 20.5 in. to 50 in. as the indicated level varies between the scram points of 11 and 34 in. above the riser. Work at ANL on downcomer carryunder[1] indicates that the riser submergence at full power conditions should be at least half the riser diameter. The 20.5 in. more than meets this requirement.

The 14 in. steam separation or disengaging distance provided in PL-2 should be adequate to reduce the moisture entering the steam dryers so as to meet the PL-3 moisture specifications at the turbine. With this 14 in. height added above the interface, the height requirement above the fuel is 100 in. (36 in. total riser + 50 in. max water + 14 in. disengagement), relative to the 10 ft which must be provided for fuel transfer. A substantial margin of 20 in. remains for uncertainties in swell height and possible modifications in the control band.

6.8 REFERENCES

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REGION	AREA-FT^2
1	5.07
2	3.68
3	4.52
4	3.95
5	4.16
6	2.88
7	5.69
8	6.14
9	20.84
10	14.19
11	12.49
12	14.19
13	13.32
14	14.19
15	13.32
16	20.97

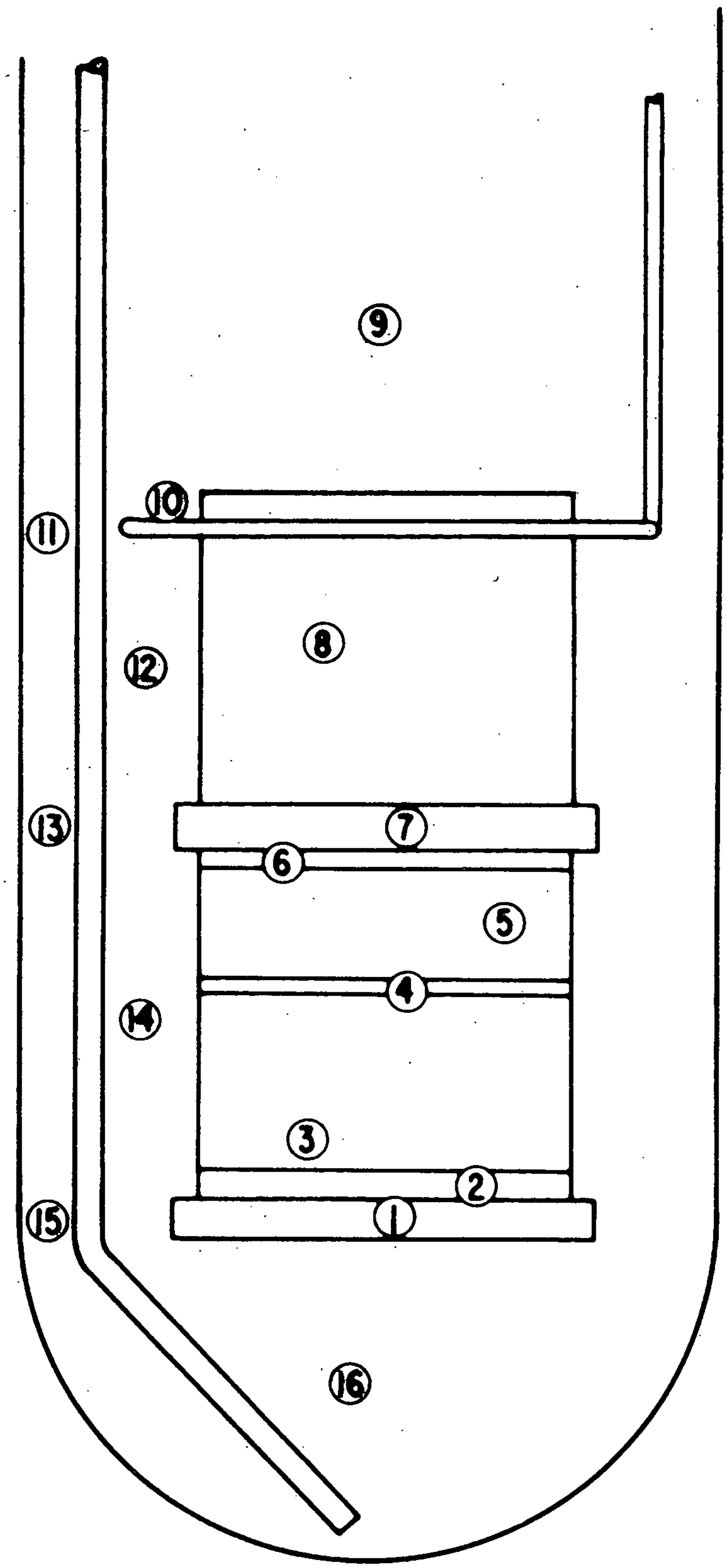

FIGURE 6.1 CONVECTIVE FLOW AREAS

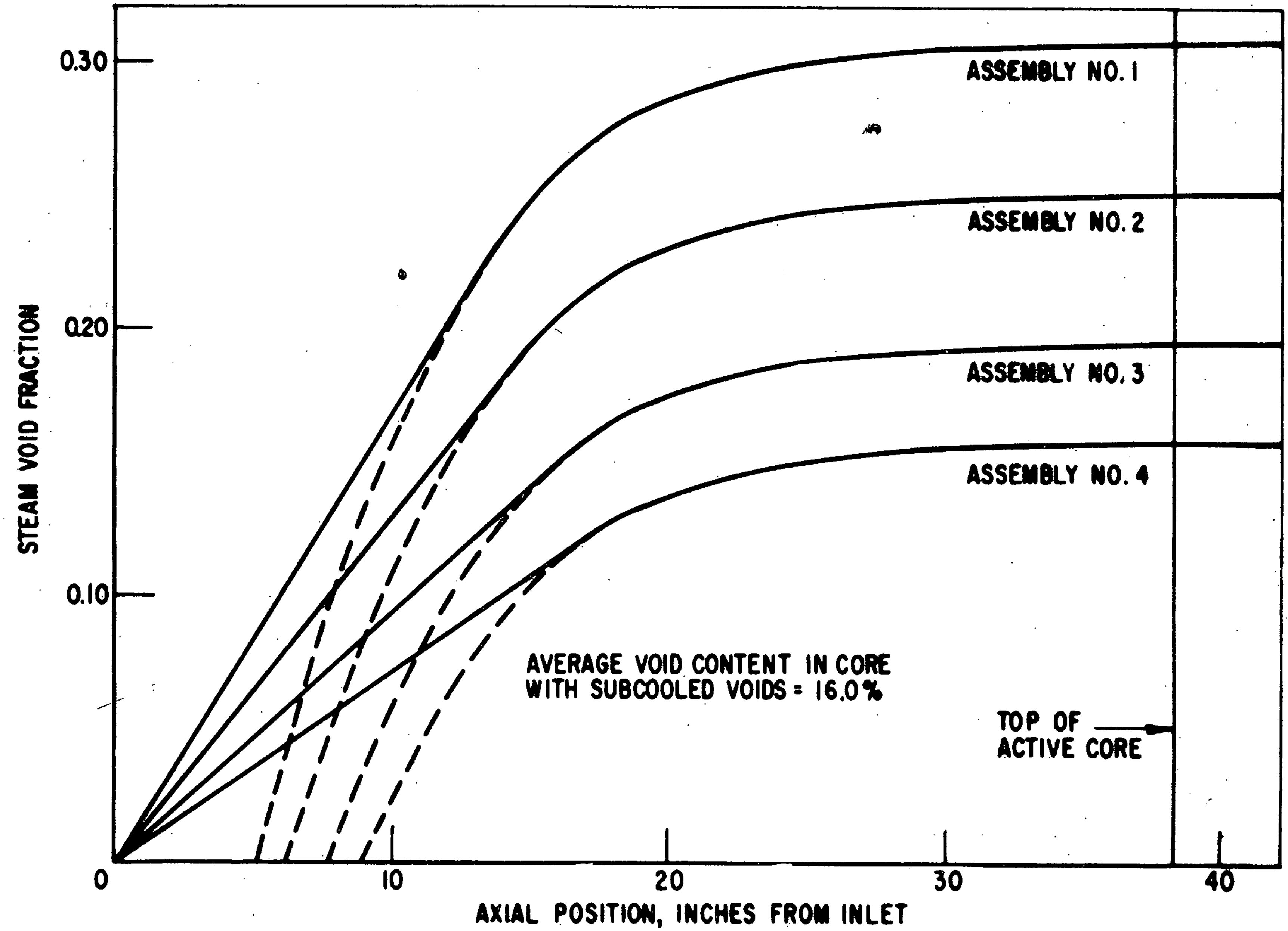

FIGURE 6.2 VOID DISTRIBUTION IN TYPICAL FUEL ASSEMBLIES AT BEGINNING OF LIFE

7.0 NUCLEAR ANALYSIS

The availability of the results of nuclear calculations performed during the PL-2 design study obviated the need for another detailed nuclear analysis; however additional calculations were made in several areas to extend the available design information to PL-3 specifications. The fast neutron exposure was determined for the modified reactor vessel. Also, the temperature coefficient at 4°C at beginning of life was calculated. The balance of information presented here was taken directly from the PL-2 Final Design Report. (1)

7.1 SUMMARY OF NUCLEAR CHARACTERISTICS

The power level requirement for PL-3 application is 8.1 Mw. The refueling mode is total core replacement with an expected core lifetime of 4 yr, based on a plant factor of 0.8. This exceeds the 2 yr minimum core life required by PL-3 specifications. The average fuel burnup is 8200 MWD/MT. The design employs supplementary control in the form of discrete burnable poison rods. These rods, consisting of 1 w/o natural boron in stainless steel and located in the fuel assemblies, provide a 1.8 percent shutdown margin at 4°C at beginning of life with the central cruciform control rod stuck out.

The beginning of life excess reactivity at full power with equilibrium xenon is 4%. This is used up over the four-year life by fuel depletion, fission products, etc. It is expected that critical experiments now underway at Combustion Engineering on a core of PL-2 geometry and composition will indicate that a considerable reduction in boron concentration can be achieved, and that the resulting greater effective burnup of the boron during life will provide a longer core lifetime. Physical removal of the boron rod shims during life would be an additional means for increasing reactivity lifetime.

A summary of significant nuclear characteristics is presented in Tables 7.1, 7.2 and 7.3. The data were taken directly from the PL-2 Design Report and they all apply to the core composition at beginning of life.

TABLE 7.1
EFFECTIVE CORE MULTIPLICATION FOR VARIOUS OPERATING CONDITIONS
(Beginning of Life)

	K_{eff} Rods Out	K_{eff} Rods In	K_{eff} Control Rod Stuck Out
Cold, 4°C			0.982
Cold, 68°F	1.141	0.907	0.978
Hot, Zero Power, 489°F	1.094	.814	
Hot, No Xenon, 18% Vapor	1.004	.757	
Hot, No Xenon, 30% Vapor	1.000	.713	

TABLE 7.2
REACTIVITY DEFECTS WITH TEMPERATURE AND VAPOR

(Beginning of Life)

	Δk	Reactivity $\Delta k/(k_1 k_2)$
Temperature Defect 68^oF to 489^oF		
Rods Out	.047	3.8%
Rods at Hot Zero Power Critical Position	.064	6.0%
Rods In	.093	12.6%
Vapor Defect at 489^oF 0% to 16% Average Vapor		
Rods Out	.044	3.8%
Rods at Hot, 8.5 Mw, No Xenon Critical Position	.049*	5.1%
Rods In	.050	8.0%

* Obtained for 18% uniform average vapor and interpolated down to 16% average vapor.

TABLE 7.3
REACTIVITY COEFFICIENTS

(Beginning of Life)

Temperature Coefficient at 4^oC* (Rods Out)	$-2.42 \times 10^{-5}\ \Delta\rho/^oF$
Avg. Temperature Coefficient (68^oF to 489^oF)	$-1.4 \times 10^{-4}\ \Delta\rho/^oF$
Avg. Void Coefficient (0% to 16% avg. vapor)	$-3.2 \times 10^{-3}\ \Delta\rho/^oF$
Avg. Doppler Coefficient (489^oF to 1000^oF avg. fuel temperature)	$-4.3 \times 10^{-6}\ \Delta\rho/^oF$

* Independent Calculation.

7.2 REACTIVITY AND LIFETIME

The lifetime of the PL-2 core is given in the PL-2 Design Report as 25.5 MWYR. This corresponds to about 4 yr at the present PL-3 design conditions. The beginning-of-life average void fraction has been independently calculated for the PL-3 reference conditions and has been found to agree closely with that used in the PL-2 burnup studies.

The methods used to calculate reactivity for PL-2 are reported to predict a higher reactivity by about 1.0 to 1.5 percent when used to predict critical experiments already performed on the BONUS and Yankee reactors. If the PL-2 critical experiments verify the 1.0 to 1.5 percent bias in reactivity, it may be possible to reduce the boron concentration in the poison rods by as much as 50 percent and still retain adequate shutdown margins with resulting extension of the core life. Not only will the end of life reactivity be increased by the reactivity increase at beginning of life, but a further increase can be realized since a much larger fraction of the effective boron will be depleted due to the lower flux depression factors encountered in the poison rods with lower boron content.

The lifetime reported for the PL-2 core is the result of a comprehensive burnup study. This study is described in detail in the PL-2 design report. It consisted of axial burnup calculations which accounted for depletion effects in the self-shielding factors of the fuel and poison rods and in the transverse flux shapes within a typical fuel assembly. The study also included the effects of decreasing average core void content, resulting from rod withdrawal and shifting of the axial power distribution with burnup. The radial power distribution was, for simplicity, held constant over life. This is reported to cause a reactivity overestimate of less than 1 percent at end of life. The study indicated that the average void fraction in the core decreases from 16.0 to 13.2 percent at end of life. This results primarily from the shift in power to the upper part of the core as the rods are withdrawn. The decrease in average void content corresponds to a reactivity recovery of about 0.8%.

Tables 7.4, 7.5 and 7.6, taken directly from the PL-2 Design Report, indicate the changes in fuel composition and reactivity effects determined from the burnup study.

7.3 STABILITY CONSIDERATIONS

The nuclear-hydraulic stability of a boiling water reactor power plant depends on the characteristics of the closed loop described by the nuclear, hydraulic and thermal parameters for the system. The dynamic behavior of boiling water reactors is today well understood, but methods are not available with which one can predict limitations of stable operation for a reactor which has not been built.

TABLE 7.4
FUEL AND BORON CONTENT AT BEGINNING AND END OF LIFE

	Beginning of Life Total Kg	End of Life Total Kg	End of Life Grams/Kg of U-238
U-235	54.7	44.1	42.3
Pu-239	-	2.41	2.3
Pu-240	-	.22	.2
Pu-241	-	.08	.1
Total Pu	-	2.71*	2.6
Boron-10	.0995	.0468	

* The data imply an average conversion ratio over life of .26 grams of plutonium formed per gram of uranium-235 destroyed, not taking credit for the plutonium burned as fuel.

TABLE 7.5
POWER FROM VARIOUS FISSIONABLE ISOTOPES IN CORE

	Beginning of Life	End of Life
Uranium 235	95.6%	84.7%
Uranium 238 (fast fission)	4.4%	4.6%
Plutonium 239		10.3%
Plutonium 241		0.4%

TABLE 7.6
COMPARISON OF REACTIVITY VALUES AT BEGINNING AND END OF LIFE (RODS OUT)

	% Reactivity	
	Beginning of Life	End of Life
Vapor	3.8	2.7
Stainless Steel	12.8	13.3
Boron-10	8.0	5.8
Xenon	1.2*	1.3
Samarium	0.8*	0.9
Stable Fission Products	0	1.8

* Saturation occurs early in life. The value quoted for xenon was taken at 50 hours; the value for samarium was taken at 2,000 hours.

An approximate index of coupled nuclear-hydraulic stability is the reactivity in voids. (A supplementary index of the stability would be the void coefficient at design voids; however reliable values of this parameter are not available for the reactors which have operated, nor is this number indicated for PL-2.) Safe design requires that a reactor be designed for no more reactivity in voids than has been indicated to be acceptable in reactors which have operated at similar pressure and which have a similar fuel time constant.

The reactivity in voids for this core with the rods at the critical position corresponding to the hot condition at beginning of life, without xenon, is 5.1 percent Δk. This value will decrease as the fuel is consumed and as the rods are withdrawn, and also as the average void fraction decreases with burnup.

This reactivity in voids is based on an initial average void fraction of 16%, which is identical to the void fraction determined independently in this study, based on different power conditions and correlations for PL-3 application. The void coefficient has not been checked, but it is assumed to be correct. Therefore, the reactivity in voids in the rod-type core is assumed equal to that indicated in the PL-2 Design Report.

Of the reactors which have operated, BORAX IV most resembles PL-2. It has operated stably at 322 psig with 6.9 percent ΔK_{eff} in voids.[2] [3] PL-2 may be expected to operate stably at even greater reactivity in voids because of its higher operating pressure and its longer fuel time constant.

7.4 ABILITY TO MEET PL-3 NUCLEAR SAFETY CRITERIA

The PL-3 specifications require that the reactor have negative void and temperature coefficients of reactivity within all expected operating and shutdown conditions. Also, it is required that sufficient control rod worth be provided at all times during life to make the reactor subcritical at atmospheric pressure and a coolant temperature of 4^oC with any single rod in the fully withdrawn position, without the aid of soluble poisons.

7.4.1 Stuck Rod Requirement

The most reactive condition for this core is at beginning of life at 4^oC. The reactor in this condition with the most reactive rod stuck out of the core is reported in the PL-2 design report to have an effective multiplication constant of 0.982. This gives a shutdown reactivity margin of 1.8 percent Δk. The stuck rod requirement is met even when uncertainties in the calculational methods are accounted for (1.0 to 1.5 percent Δk).

7.4.2 Reactivity Coefficients

The beginning of life temperature coefficient at 4^oC has been independently calculated with all rods out and found to be $-2.42 \times 10^{-5} \Delta \rho/^oF$. With rods inserted to the just critical position, as will actually be the case, this temperature coefficient will be even more negative.

The worst condition to be encountered from the standpoint of void and temperature coefficients will be at end of life, at 4^oC, when the rods are furthest out of the core and the maximum plutonium content is present. In this condition the effective core size will be at its maximum, for 4^oC, and hence the negative leakage effects will be at minimum. Also, with the maximum plutonium content present, the maximum positive contribution will result from the interaction between the low lying plutonium resonances and the thermal spectrum changes due to void and temperature changes. The calculation for this worst condition has not been performed. However, it is thought that the coefficients will not be positive, due to the leakage effects that will be encountered for the just critical effective core size at end of life. The amount of rod insertion required to overcome the end of life temperature and vapor defects will correspond to about 6 percent ΔK_{eff}.

7.5 POWER DISTRIBUTIONS

The overall peaking factor as reported in the PL-2 Design Report changes from 4.64 at beginning of life to 2.98 at end of life. Most of this change is due to the axial peaking factor changes which come about primarily through the withdrawal of the control rods over life. The maximum local burnup of uranium is 65 percent, corresponding to 26,000 MWD/MT burnup. The average core burnup is 8,200 MWD/MT, which gives a burnup peaking of factor 3.2.

7.6 FAST NEUTRON EXPOSURE OF REACTOR VESSEL

Calculations using the one-dimensional, multigroup PIMG-2 code[4] and the two-dimensional, few-group PDQ (r, z) code[5] were employed to determine the integrated fast neutron flux above 1 Mev at the vessel. The calculations were normalized to measurements of fast neutron fluxes carried out for the SM-1 mockup at the Alco critical facility, under the Army-supported AE-90 program. The results of the calculations, after normalization to experiment, were increased by a factor of 1.3 to account for the reported ± 30 percent probable error in the SM-1 measurements.

The results of the calculations indicate that the maximum vessel exposure, over a 20 yr life at an average power level of 6.4 Mw, is 1.7×10^{18} nvt.

An exposure of 1.0×10^{19} or less conforms to an interpretation of the design criterion applied to carbon steel, that "there will be no restriction of operating and maintenance procedures." Thus the nvt requirement is satisfied by a wide margin.

7.7 REFERENCES

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8.0 CONTROL AND INSTRUMENTATION

8.1 REACTOR CONTROL SYSTEM

8.1.1 General Description

Essential to the design of a boiling water reactor is a negative void coefficient which, at constant pressure, causes the reactor to be self-regulating in the sense that a stable power corresponds to each rod position and inlet coolant condition. In a natural circulation direct-cycle system, neither the flow nor temperature of the recirculated water is controlled, hence adjustments in reactor power require repositioning of the control rods.

A negative void coefficient in a direct cycle reactor necessarily implies a positive pressure coefficient, which causes the reactor to respond to pressure changes associated with load variation in a direction opposite to load demand. Proper power demand response can be obtained with a control system which holds reactor pressure constant.

The constant pressure control system can be based on rod movement in response to deviation of pressure from its nominal value. Operating experience has demonstrated successfully that direct cycle BWR systems can be operated stably by manual control in this manner, even during extreme changes in demand.

Alternatively, a reactor of this type could be operated by methods which maintain reactor pressure and which avoid continuous direct matching of load by dissipating a variable fraction of the energy output. One method was considered which used an electrical compensation system with a variable load bank on the generator output. A comparable or supplementary system would use variable steam bypass to the condenser and/or steam accumulation to maintain constant system pressure through constant total steam demand on the reactor. Such methods were rejected on the basis that they do not conform strictly to the design criterion regarding load-following capability.

The PL-3 would be controlled by automatic movement of rods in response to a signal derived primarily from pressure deviation relative to a set point. As in PL-2, a bypass to the condenser would be provided for operational flexibility. The bypass valve, however, would also open automatically whenever the pressure exceeded the set point by a predetermined amount. This pressure relief function is added to reduce the probability of high pressure scram occurring during the pressure rise accompanying a severe load reduction. The high pressure scram setting must be consistent with the 700 psi design pressure of the vessel.

The water level in the reactor vessel is controlled by regulating the feedwater flow to avoid excessive carryover of moisture to the turbine and to insure adequate riser submergence for recirculation. The water level control system should respond smoothly and rapidly to mismatch between steam flow and feedwater flow, as well as to level variation. Changes in feedwater flow will be reflected almost immediately in the sub-cooling at the core inlet with consequent changes in void content and reactivity. Rapid corrective action by the water level control system will enhance the load-following capability of the overall control system. The effects of changes in feedwater temperature with load will be minor because of the large recirculation ratio and because of the delay and attenuation in transit through the loop.

8.1.2 Control Rod Operation

Manual positioning of the nine control rods by the operator will be enabled by an operating switch and selector on the console. The eight peripheral rods can be positioned individually or as an entire bank. Normally, only the central control rod would be activated by the automatic load-following control rod positioning system. Additional consideration should be given the relative merits of providing an automatic control rod sequencing program to provide improved power distribution in the core, minimize non-linearities between reactivity and rod position and simplify the operator's role.

8.1.3 Reactor Pressure Control

The automatic load-following rod-positioning control loop will consist basically of a sensor transmitting pressure signals to recorder and controller units. The controller will operate on deviations from the set point pressure to provide proportional control rod positioning over the load range. Integral error, or reset action, will be provided to eliminate any steady state error. Derivative error, or rate action, will also be considered to improve the system response to load changes. This could be accomplished by the controller operating directly on the pressure error signal. However, there will be a definite time constant associated with the system pressure response to a load change. Steam weight flow changes could be used, in a two-element control system, as an inherent anticipatory signal to initiate the desired corrective action.

An interlock with the nuclear instrumentation system is also suggested to limit rod-withdrawal on a high-flux signal, thereby permitting normal transient overshoot within the high-power scram level.

The steam bypass system would limit any unusual pressure transients and would also be useful during initial tests, startup and for decay heat removal. An automatic mode for this steam bypass control would be provided for proportional plus reset action in response to pressure excursions beyond a predetermined pressure setpoint.

After shutdown, pressure regulation by the rod control system is not available, and pressure would be maintained at design conditions, or reduced as desired, by use of the decay heat removal system.

8.1.4 Reactor Water Level Control System

A conventional three-element system will be provided to regulate the feedwater control valve position to hold the reactor water inventory within a narrow control band. Both proportional and reset action will be provided on level deviation from the set point. The difference between the steam and the feedwater flow measurements inherently provides anticipation of level changes with load.

One of three float-type mechanisms located in standpipes within the downcomer would provide level indication and the primary signal for level control. The standpipes should be designed to measure essentially the water inventory, independent of the swell height above the riser which varies with power level and pressure variations. If necessary, the steam weight flow signal can be used to provide any desired load-programming of the level detector output.

8.2 NUCLEAR INSTRUMENTATION

The nuclear instrumentation 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 Channels

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 watts 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 range channels furnish signals of 0.5 watts 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 receive 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 shall receive analog signals to:

1. Scram on reactor vessel high pressure.

2. Scram on reactor vessel low pressure.

3. Scram on reactor vessel high water level.

4. Scram on reactor vessel low water level.

5. Scram on low condenser vacuum.

6. Scram manually.

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 gamma 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, level, and temperature indications for the reactor vessel. Control and protective devices are installed for these parameters. Indications are given for the various 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, reactor vessel 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.

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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 specification was to integrate the fast neutron leakage from the primary shield and utilize bomb blast data to evaluate its transmission to one mile.[1] The analytical model is 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),

Σ = effective fast neutron removal cross section for water, 0.101 cm^{-1}

The total leakage is found from the following integral:

$$\phi_{total} = 2\pi R_o \phi_o \int_o^{2x_1} \left[1 + \cos \frac{\pi}{2} \left(\frac{x}{x_1} \right) \right] dx$$

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

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

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

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 70 percent of the limiting neutron flux from direct shield penetration. Neutron streaming through radial streaming paths and shield penetrations can then be allowed to contribute the remaining 30% 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 the one mile specification since they are captured long before they reach one mile. However, they do develop later to be 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.

1. Losses due to scattering 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 equaled water in the above model, a conservative assumption.

3. Snow scattering outside the reactor tunnel - The bomb blast radiation data utilized are for an infinite air medium. Introduction of the snow as a second scattering medium would probably account for about a factor of 2 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 which develop within the primary shield will not contribute to the effectiveness of the shielding. No account was made of these voids in the above model, a non-conservative assumption.

5. A better 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.

7. Neutron leakage through the top of the primary shield - Integrated fast neutron leakage through the top of the primary shield was arbitrarily set at the same value as that through the sides for preliminary design pruposes. An optimization between top and side leakage would be needed in final design.

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), and

3. No detectable radiation in camp living quarters (0.004 mrem/hr is taken as this limit).

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 feet (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)

In the previous section, D_{sn} and ϕ_{sn} were found to be 5.7×10^7 and 8.5×10^8 respectively. The integrated operating gamma leakage from the primary shield is about 6.2×10^7 mrem-cm^2/hr if it is designed to meet shut-down 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% of this will escape from the tunnel on one scattering. Using these figures in the above formulation gives:

$$D_{100} = 3.5\text{x}10^{-9} \text{x} 6.2\text{x}10^7 + 5.2\text{x}10^{-10} \text{x} 5.7\text{x}10^7 + 2.3\text{x}10^{-11} \text{x} 8.5\text{x}10^8 = 0.22 + 0.03 + 0.02 = 0.27 \text{ mrem/hr}$$

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. Attenuation of neutrons and gamma radiation in the reactor tunnel roof are omitted in this calculation as are neutron losses due to scattering within the tunnels.

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 added shielding on the reactor. The present calculations indicate a requirement for a controlled area on the snow surface for about 500 ft. surrounding the reactor.

Factors in this calculation which need additional study are:

1. Coefficients in the empirical air scatter formulation -The coefficients of ϕ_{sn} has some unusual characteristics which lead to questions 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. This assumption was conservative and could account for doses as great as one-half of the calculated doses.

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 ϕ_{sn}. They then contribute to the air capture gamma dose.

4. Neutron escape from the snow tunnels - The 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 designs of the primary shielding, reactor building, and tunnel configurations progress. 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 requirements. This requirement is interpreted to limit radiation leakage from the primary shield to 100 mr/hr at 2 hr after shutdown from an extended full power operation. Since ROC[4] Code library decks were available for 2.4 hr shutdown time, and since a dose reduction of about 20% occurs between 2.0 and 2.4 hr shutdown time, an equivalent preliminary design target was taken as 80 mr/hr at 2.4 hr after shutdown.

The primary shield optimization study indicates that about 4 in. of lead shielding must be added within the 7.5 ft radius primary shield to meet this criterion. Figure 9.3 gives the gamma dose rate at 2.4 hr 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 slection of lead for the PL-3 gamma shielding material.

The total operating gamma dose rate 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 RADIATION LEVELS IN EXTERNAL PLANT

Without shielding or added holdup, the dose rates from the steam line and the power plant equipment will be greater than the specified value of 1.2 mr/hr at points 3 ft from power plant components. In the absence of gross fuel cladding defects or failures the radiation will be essentially all from 7.35 sec N^{16}.

The N^{16} dose rates are predicted by scaling the data for EBWR on the basis of decay times, flow rates, power densities, relative virgin flux, and those assumptions found necessary in EBWR test No. 34 to match the calculated and observed dose rates in EBWR.[5] The results of calculations presented in a separate study[6] are summarized in Table 9.1.

Because of the uncertainties in geometry, the dose rates from components other than the steam line have been scaled from EBWR according to the calculated values of activity flow to the power plant equipment. This is valid, provided the equipment for each plant is sized for about equal steam velocity, as demonstrated by the calculation of dose rates 3 ft from the 6 in. EBWR steam line and 3 ft from the 4 in. PL-3 steam line (see Table 9.1). The steam velocities are similar at this point and the calculated dose rates are nearly proportional to the activity flow.

The steam line dose rate is based on a pipe extending infinitely in both directions, as is effectively the case in the pipe tunnel, but not near the turbine where only one end of the pipe is viewed. On this basis it may be predicted from Table 9.1 that the N^{16} dose rate from any single component at a 3 ft distance would be less than 6 mr/hr provided the air ejector condenser is shielded. If this unit were similar to the EBWR units, about 1-3/4 in. of lead would be required over a small volume to reduce the radiation level to that from other components.

The predicted dose rates are up to five times the levels specified for PL-3, indicating that accessibility might have to be restricted to 20 percent of the 84 hr week during continued full power operation. The dose rates will diminish appreciably at reduced power, varying nearly as the power level squared because of the higher recirculation ratio which occurs in the reactor at reduced power. Thus at 80 percent average load the radiation levels might be up to 3 times those specified; below 45% load continuous occupancy should be possible.

The N^{16} activity problem could be eliminated by adding a pressurized steam holdup volume before the turbine, but this appears unnecessary because the accessibility restrictions required without added holdup should not greatly inconvenience power plant operation. A volume of 78 cu ft, for example, would provide a reduction of four in activity, and this might consist of 100 ft of 12 in. pipe weighing about 3 tons.

The neutron production from N^{17} carried over in the steam was also considered from the standpoint of the neutron flux limitation at one mile from the plant, and found to be safe in this regard by at least a factor of 100.

TABLE 9.1
PREDICTED N^{16} DOSE RATES, MR/HR

Component	EBWR Contact	EBWR At 3 ft.	PL-3 Predicted at 3 ft.
6" Line at Steam Dryer	100	15	-
4" Steam Line at Turbine	-	-	9
Turbine Inlet	50	$\sim$ 5	$\sim$ 3
Condenser Hotwell	40	<10*	< 6
Air Ejector	400	<60*	<40
		EBWR	PL-3
Activity Flow in Steam Pipe, curies/sec		0.108	0.060
Specific Activity in Steam Pipe, dis/sec-gm		0.53×10^6	0.69×10^6

* Estimated reductions at 3 ft.

9.4 CONSEQUENCE OF FUEL FAILURE

The design criteria require safe routine operation even in the event of simultaneous release to the coolant of one percent of the activity of the end-of-life fuel elements. Subsequent interpretation requires instead that a safe orderly shutdown and transfer of load be possible following such release, and indicates that the one percent figure is somewhat high.

Calculations are presented in M-WRP-EED-62-1 for the time integrated dose received by an observer 3 ft from the steam line, which is assumed to be shielded only by insulation, when one percent of the Xe, Kr, I, Br, and Cs isotopes pass through the line. All other activities should be effectively removed in the steam-generating process and retained in the coolant. The dose is 43 mr from Xe and Kr and 216 mr from all five elements. The dose rate depends on the assumed duration of release; a 1000 second release would cause 160 mr/hr from Xe and Kr, and 776 mr/hr from all five elements.

The actual steam line dose rate to be expected following gross fuel failure is speculative; it depends on the leakage of freshly-created fission products during continued operation following the fuel failure as well as the manner of release of the previously accumulated inventory. Some indication of the dose rate to be expected is available from measurements made on BORAX IV during operation at up to 6 Mw thermal power subsequent to failure of about 1-1/4 percent of the fuel elements by collapse of the void volume region within the elements. The elements were thoria rods containing U-235 and the reactor had operated for sufficient time to essentially saturate the fission products.

The contact dose rate at the steam line was 400 mr/hr during 6 Mw operation subsequent to fuel failure.[7] With the N^{16} contribution subtracted and with an appropriate correction for distance, a dose rate of about 60 mr/hr is inferred at a location 3 ft from the steam line. Activities at the steam line, turbine, air ejector and other points had essentially vanished 17 hr after shutdown.

It is expected that the fission products dose rates near the power plant components could be several times greater than the normal N^{16} dose rates and that evacuation of the turbine buildings would be necessary in the event of extensive fuel failure. However, an orderly shutdown and transfer of load to other energy sources should be possible following the occurrence.

The fission products passing through the turbine will be collected at the air ejector and transferred to the off-gas handling system, the capability of which is discussed in Section 12.1.

The dose rates from normal fission product release, as may occur from small amounts of fuel smeared on the outside of the elements, are indicated to be negligible.

9.5 REFERENCES

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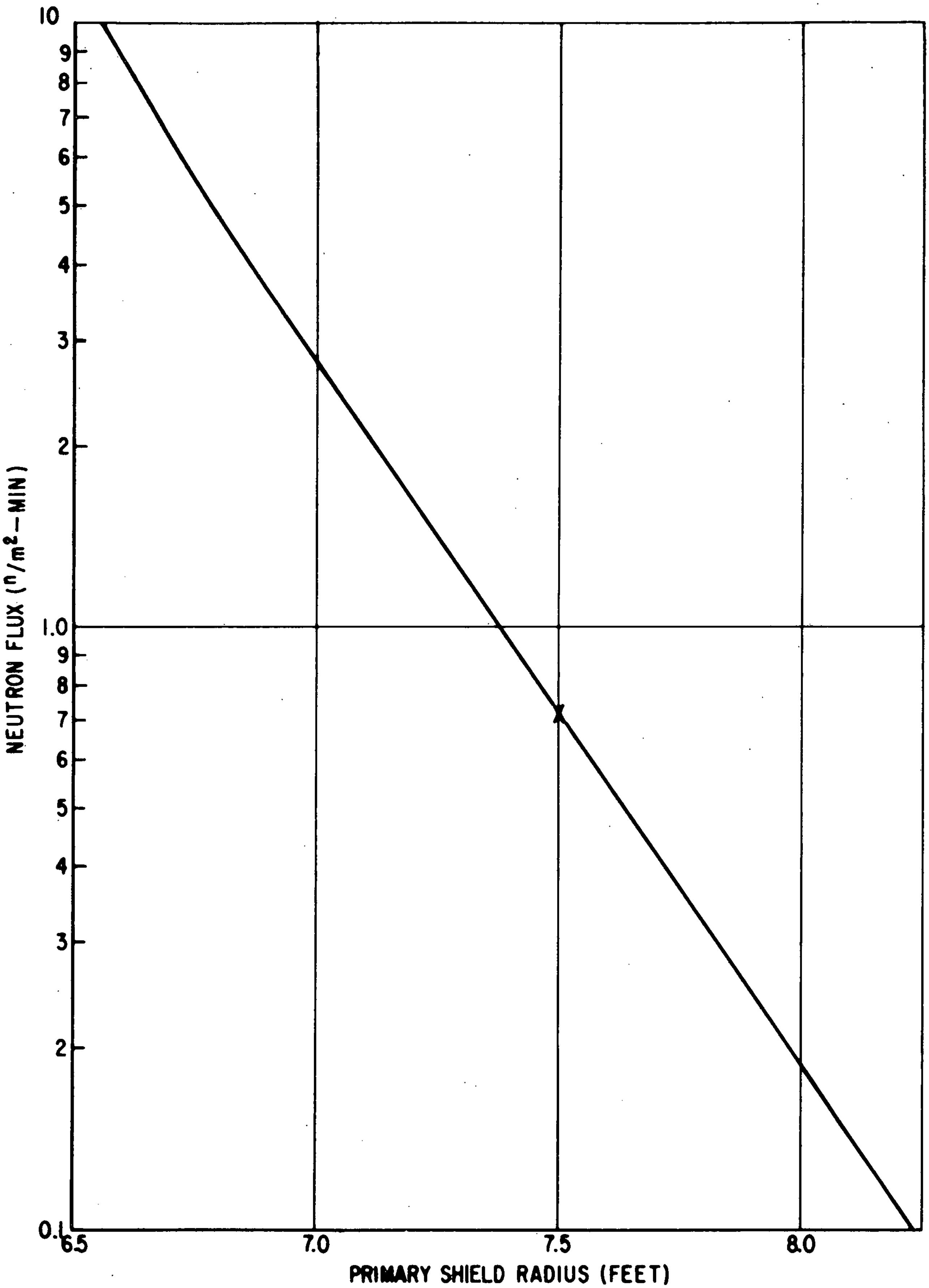

FIGURE 9.1 NEUTRON FLUX AT ONE MILE FROM DIRECT SHIELD PENETRATION NEUTRONS

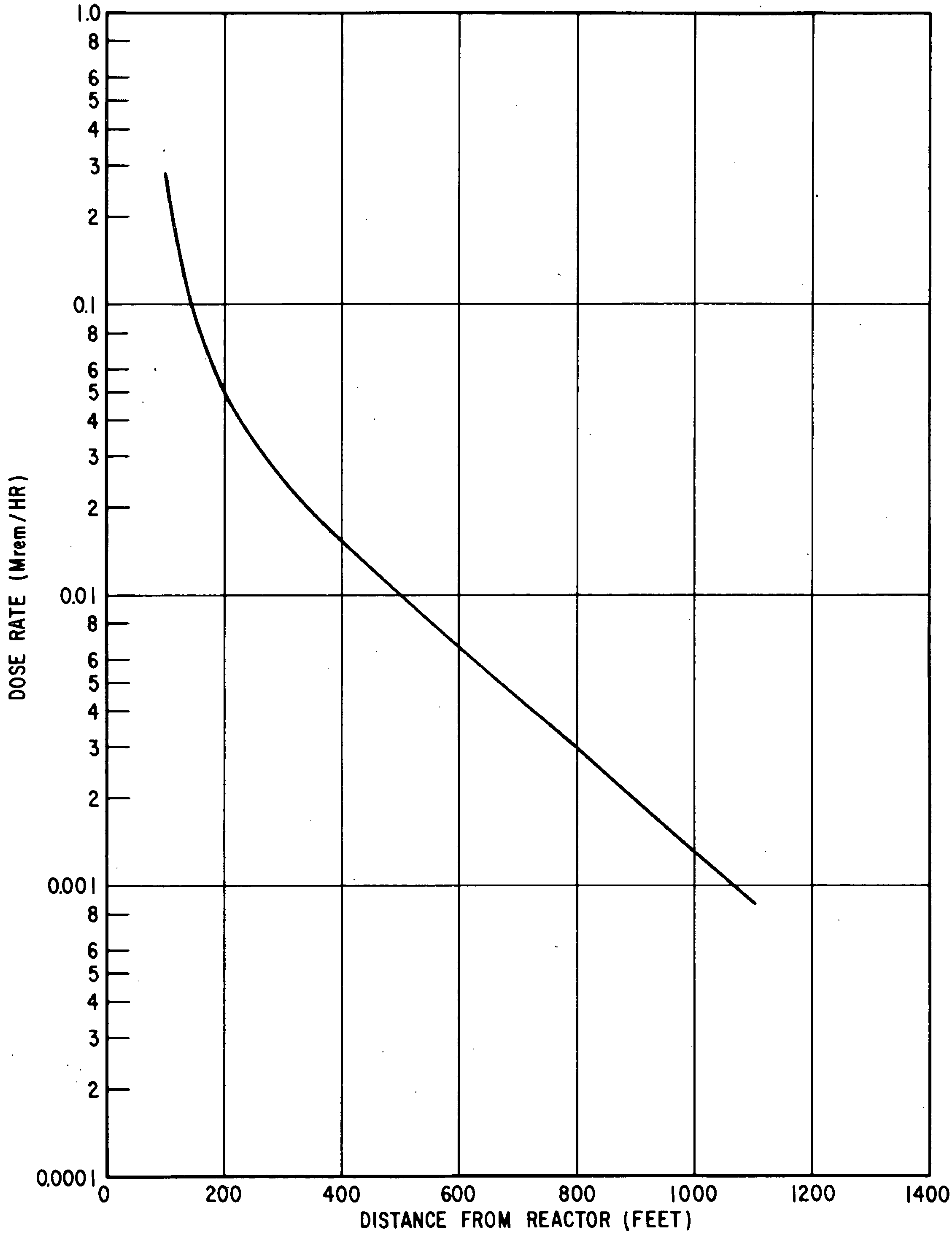

FIGURE 9.2 AIR SCATTER GAMMA DOSE ON SNOW SURFACE

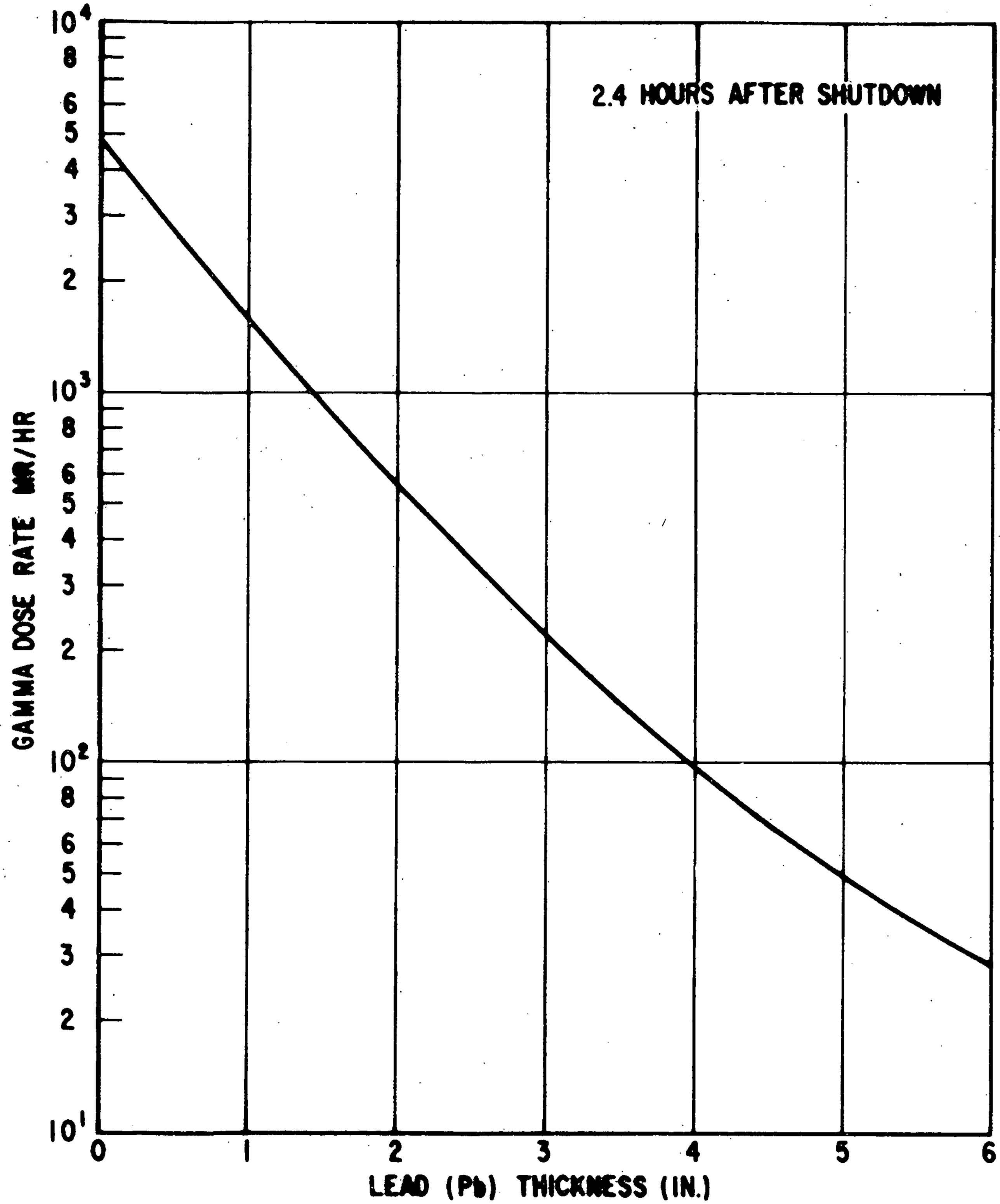

FIGURE 9.3 SHUTDOWN DOSE RATE VS. LEAD THICKNESS

10.0 REACTOR AUXILIARY SYSTEMS

10.1 COOLANT PURIFICATION SYSTEM

A schematic of the coolant purification system is indicated in Dwg. AEL-710. A low pressure system (25 psig) is preferred to a system operated at reactor pressure for the following reasons:

1. Most of the components are designed for low pressure (125 psig design).

2. Maintenance and replacement are less hazardous.

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

The only significant disadvantage of a low-pressure system is the requirement for continuous operation of a high pressure pump. However, if high pressure sealing water is required for primary system components (e.g., control rod drives), continuous operation of this pump is necessary.

Purification is accomplished by continuously diverting a small flow of water from the reactor, processing it and then returning it to the system. The flow or "blowdown" is approximately 1.5 gallons per minute of essentially high purity water contaminated with activity and corrosion products.

Operation of the purification system is completely automatic. Water is removed from the reactor, cooled to below 120°F and then reduced in pressure to 25 psig which is sufficient pressure to provide the required flow through the ion exchangers to the makeup tank.

A cation bed followed by a mixed bed arrangement is proposed for PL-3. This arrangement possesses several advantages:

1. Retention of the major portion of activity by the cation bed, thereby concentrating the activity and reducing shielding requirements.

2. Greater removal of activity than could be accomplished by a mixed bed alone.

Safety features of the system include:

1. A temperature controlled valve to shut down blowdown if temperature exceeds 130°F.

2. High and low pressure alarm.

3. High and low flow alarm.

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

5. A pressure relief valve to divert blowdown to the reactor shield tank at 100 psig (in the event of failure of other valves).

From the make-up tank, a pump raises the pressure to 50 psi above that of the primary system. This accomplishes two purposes: 1) returns flow to the primary system and 2) provides pressure for control rod drive seals (out leakage from the seals can be returned directly to the makeup tank or to an intermediate seal drain tank). Flow control is obtained through a flow control valve sensing makeup tank level.

Chemical treatment of the reactor coolant or intentional addition of chemicals would not be used.

10.2 DECAY HEAT REMOVAL SYSTEM

This system provides for the removal of decay heat generated in the reactor core after shutdown and facilitates depressurizing and controlled cool-down of the plant. During shutdown, when the pressure control available from the control rod system is unavailable, this system can be used to maintain design pressure. As shown in Dwg. AES-608, the decay heat removal system is integrated into the primary purification system.

Steam is removed from the main steam line through a pressure control valve, condensed, and discharged to the make-up tank. The temperature of the reactor will correspond to the saturation temperature at the pressure set at the pressure control valve, therefore the operator can control reactor temperature through a variable set point on the pressure control. The decay heat generated in the reactor will cause boiling, with the steam being vented through the pressure control valve. In this manner, the decay heat is continuously removed from the reactor. Continuous controlled lowering of the set point results in decreasing temperature as water is flashed to steam. In this manner the reactor operator controls the reactor temperature and cool down rate through pressure control, down to a reactor pressure of about 20 psi. The set point can also be increased to provide for an increase of reactor temperature.

The return of water to the reactor is through the use of positive displacement pumps. The pumping capacity must be sufficient for the maximum decay heat removal rate plus the cool down requirements. Since heat is removed by boiling, a return flow of 1000 lb/hr or 2 gpm will be sufficient, and this flow requirement diminishes very rapidly.

The primary purification system must be in operation to provide a return of water to the reactor. The steam condensed in the decay heat removal system is discharged to the makeup tank, tending to cause the tank level to rise. Since reactor blowdown rate is controlled by make-up tank level, the blowdown flow will automatically decrease to compensate for the additional flow into the tank, and the combination of vent flow and blowdown will therefore always be equal to the make-up rate. The reactor, the decay heat removal system, and the purification system comprise an essentially closed system, simplifying reactor water level control during shutdown. Indicators and alarms are provided on the reactor water level and make-up tank level to insure that adequate levels are maintained.

After the reactor pressure is reduced below approximately 20 psig, normal blowdown is no longer possible and it is necessary to use a booster pump. At this time decay heat removal is transferred to the primary purification system. When a reactor temperature of 200°F or less is obtained, the pressure control set point is run to zero to prevent any inadvertent pressure buildup.

10.3 EMERGENCY COOLING SYSTEM

The isolation valve on the main steam line is an integral part of the plant containment which must close when fault conditions warrant isolation of the reactor. Closure of this valve also isolates the reactor from the main heat sink and requires that the decay heat be removed from the reactor internally. Complete power loss could attend this condition, therefore it is desirable that operation of the emergency cooling system not be dependent on pumps.

Drawing AES-608 includes a schematic of the emergency cooling system. This system would be activated by the opening of a single valve which allows steam from the main steam line to flow to a coil in a tank where it is condensed. The condensate flows by gravity to the reactor through the feedwater line. The tank in which the coil is located would contain stagnant water sufficient for several hours cooling, and would have provisions for makeup.

Consideration would be given to automatic closure of the containment and simultaneous activation of the emergency cooling system. Selected scram signals such as low reactor pressure and low water level would close isolation valves on the main steam line and the purification outlet line and open the emergency cooling valve. The check valve on the feedwater line between the reactor and the purification return line would complete the isolation.

10.4 SHIELD COOLING AND CLEANUP SYSTEM

Heat generated in the shield water will be removed by use of cooling coils in the shield tanks. Water circulated through these coils transfers heat to the glycol system through an intermediate heat exchanger. Water from the tanks may be circulated through a demineralizer to provide for purification.

10.5 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 shutdown.

10.6 REFUELING AND FUEL TRANSFER SYSTEM

10.6.1 Refueling Objectives

The reactor must be refueled at regular scheduled intervals. This change of fuel 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 subject to adverse climatic conditions.

3. Minimum number of units to adequately do job.

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.

10.6.2 Removal of Fuel from Reactor

Spent fuel elements would be transferred by flask to spent fuel shipping casks located in the peripheral shield tanks. It is not desirable to utilize transfer chute or tube methods with this concept since it would require penetration of the reactor vessel. Such a penetration into a high pressure vessel is not practical and increases both the containment problem and hazard potential.

A typical transfer schematic, incorporating an indexing mechanism in the shield cover, is shown in Fig. 10.1. This indexing mechanism enables the refueling crew to directly engage the desired fuel element without use of tools other than those adapted to the transfer flask.

When transferring spent fuel elements by flask on the boiling water reactor concept, the procedure is as follows, referring to Dwg. AEL-740.

1. Pump water into reactor vessel until vessel is full.
2. Drain and remove upper shield from around vapor container cover.
3. Remove shielding around the control rod drive units and store.
4. Disconnect and store control rod drive units.
5. Unbolt and remove vapor container cover.
6. Remove control rod drive thimbles and delatch control rods.
7. Remove control rod extensions and racks.
8. Crack open nuts on reactor vessel.
9. Drain reactor vessel to flange level, remove and store cover nuts.
10. Remove and store vessel cover with housings attached.
11. Place fuel transfer cover shield on reactor flange.
12. Index shield cover to element selected for removal.
13. Place transfer flask on shield cover.
14. Open the viewing port, lower the grappling tool and engage the fuel element.
15. Close the viewing port and raise element into transfer flask.
16. Close the shield gate at bottom of flask.
17. Raise flask and move over shipping cask in shield tank.
18. Check locating adapter position and lower flask onto adapter.
19. Using extension tool, open shield gate in base of flask.
20. Using same extension tool, lower element into basket of cask.
21. Disengage element and retract tool into transfer flask.
22. Raise flask to operating floor level.

23. Move locating adapter on cask to next loading position.

24. Repeat steps 12 through 23 until all elements are transferred.

25. Remove shield cover and load new elements into reactor by hand tool.

26. Reassemble reactor by reversing steps 1 through 10.

A transfer flask conceptual design is shown in Dwg. AEL-731. This flask is designed to accommodate the modified SM-2 type 28 in. plate element for the reactor described in the appendix to this report, but the description is applicable to a comparable flask for the rod-type fuel assembly. The individual control rod and absorber grappling tools will penetrate the top center of the flask. Vertical movement of the grappling tools will be controlled by a handwheel on the end of the horizontal shaft. A spring loaded pawl engaging a ratchet wheel will prevent accidental dropping of the fuel element or absorber when it is being raised from the reactor or lowered into the storage container. This pawl is disengaged by hand force on the lifting handle button and is spring loaded to engage when the hand is removed. The manually operated shield gate at the bottom of the flask will be closed at all times when a fuel element is inside the flask. A sliding shield gate is used in this version rather than a rotary type, to save weight. If spent fuel elements are to be loaded in a shipping cask, the transfer flask will be lowered into the tank for transfer. If the spent fuel elements are to be discharged into a storage tank, a shielded cover will be placed on the top of the tank and the elements lowered into a receptacle and transferred by hand tool from there into the rack. The use of a cover eliminates potential hazard in the event of equipment failure. This flask loaded will weigh approximately 16,350 pounds and will be transported by crane. The estimated weight of a comparable transfer flask for the PL-2, rod-type element would be approximately 22,200 pounds. The heavier unit could be shipped disassembled at less than 20,000 pounds and assembled at the site.

10.6.3 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. The shielding thicknesses were tentatively determined by ROC code. According to present information, little advantage is gained in shielding thickness by allowing cooldown for one year rather than ninety days. The decrease in shielding for one year storage will be less than 0.4 in. under that for 90 day storage. The concepts are therefore based on radiation levels ninety days after shutdown of the reactor.

The PL-2 core consists of 24 rod-type fuel element assemblies and 9 cruciform control rods. Calculations based on presently available information indicate that a shipping cask for these elements will exceed the desired 20,000 pound maximum, unless no more than four elements are sent at a time. The four element cask would be in a 2 x 2 element arrangement, as shown at the bottom of Fig. 10.2, and six casks would be required. On those casks housing two cruciforms, causing a projection into the cask wall, depleted uranium would be used around the slot to bring the radiation to the same level as the balance of the cask. Since expected lifetime of the cruciform rods will be greater than the fuel elements, it may be possible to design casks on the basis of one cruciform rod per cask. A comparison of element arrangement for 4, 5, and 6 casks and their respective weights is shown in Fig. 10. 2. Assuming a shipping weight of 27, 500 pounds, the six element casks may be used and only four will be required per core.

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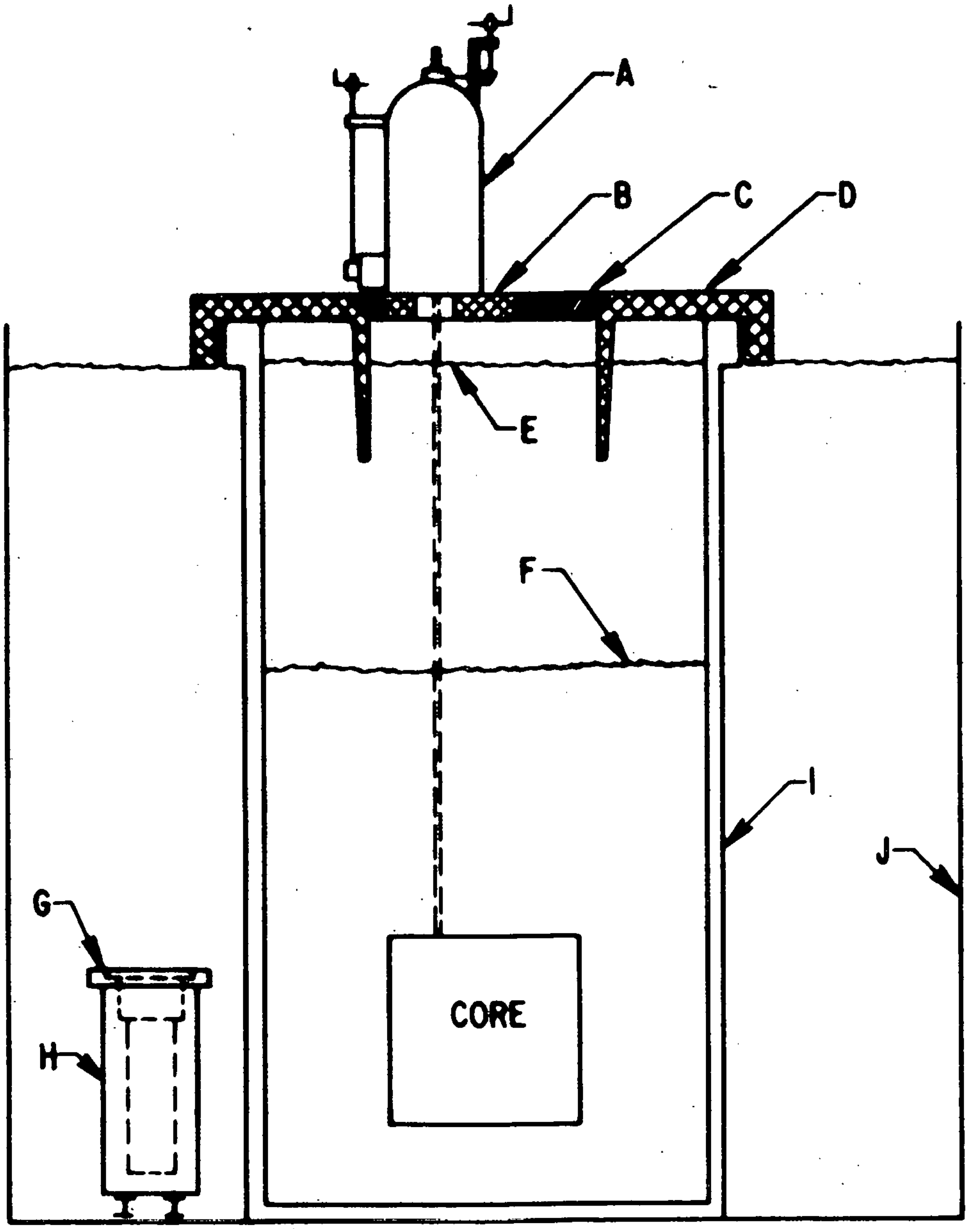
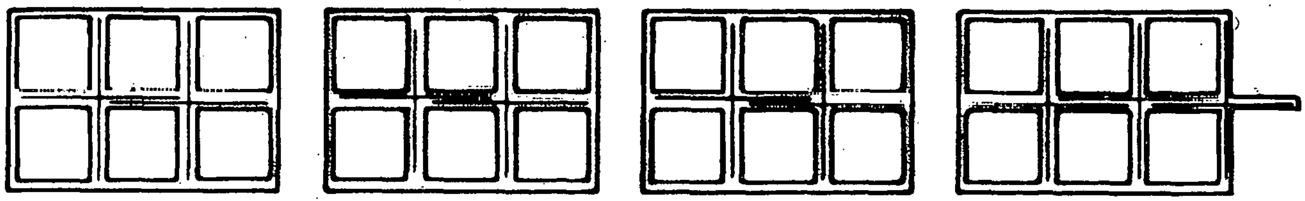


A- FLASK
B- ELEMENT GUIDE
C- SECTION GUIDE
D- SHIELD COVER
E- WATER LEVEL, TRANSFER
F- WATER LEVEL, OPERATING
G- LOCATING ADAPTER
H- SHIPPING CASK
I - REACTOR VESSEL
J- SHIELD TANK

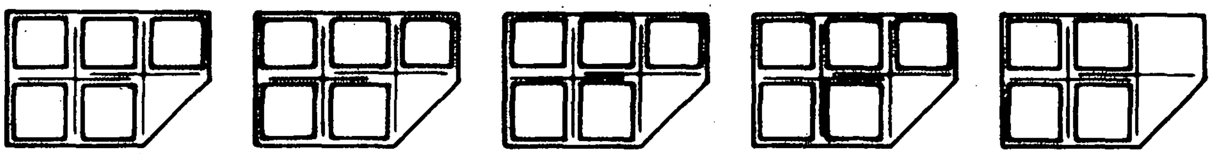
FIGURE 10.1 REFUELING CONCEPT USING TRANSFER FLASK

6 ELEMENT CASK (4 CASKS REQUIRED)



WEIGHT OF BARE CASK = 20,760#
WEIGHT OF LOADED CASK ON SKID = 23,160#

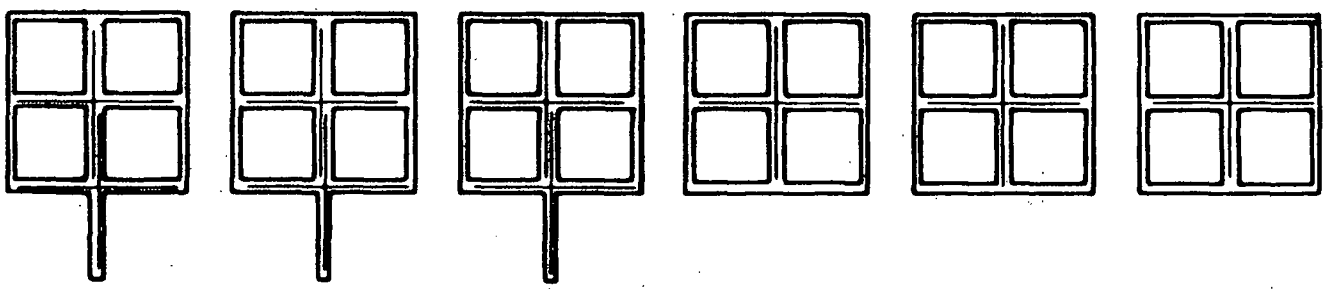
5 ELEMENT CASK (5 CASKS REQUIRED)



WEIGHT OF BARE CASK = 19,700#
WEIGHT OF LOADED CASK ON SKID = 21,900#

OPTIONS: 3 – SIX ELEMENT AND 2 – FOUR ELEMENT CASKS
2 – SIX ELEMENT AND 3 – FOUR ELEMENT CASKS

4 ELEMENT CASK (6 CASKS REQUIRED)



WEIGHT OF BARE CASK = 17,950# AND 18,400#
WEIGHT OF LOADED CASK ON SKID = 19,950# AND 20,460#

FIGURE 10.2 CASK SIZING FOR PL-2 CORE

11.0 POWER PLANT EQUIPMENT

11.1 STEAM SYSTEM AND HEAT BALANCE

The steam system consists of the main steam piping, a line separator, the steam turbine, the condenser and a turbine bypass line to the condenser. Steam generated in the reactor is utilized directly in the turbine. The steam leaving the dryer within the reactor vessel passes through the main steam piping to the turbine inlet throttle valve. A steam separator is located in the steam line just before this valve to remove moisture resulting from the line drop between the dryer and the turbine. The steam is then expanded through the turbine and 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 maintenance of 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 reactor feed pump.

A heat balance, showing the flow and state of the steam and condensate at the various points in the cycle, is presented in Fig. 11.1.

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 kw electrical at 6 in. Hg absolute using 585 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 eletric 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 reactor. For an ambient air temperature of $/34^{O}$F and a 1500 kw gross output of the turbine generator, the condenser will have a duty of approximately 18.7×10^{6} Btu/hr.

11.4 HEAT REJECTION SYSTEM

Waste heat from the main condenser and plant auziliaries will be transferred to a 60 weight percent solution of glycol in water. This solution will be circulated through the main condensers, turbine lube oil coolers and the condensate subcooler. The waste heat in the glycol water solution is rejected to the atmosphere in forced air cooled heat exchangers (air blast coolers). The heat rejection duty at full load and at $/34^{O}$F ambient air temperature will be approximately 21×10^{6} Btu/hr and the glycol flow through the air blast coolers will be approximately 1200 gpm at a mean temperature of approximately 105^{O}F. The air blast coolers are continuous plate fin cores with 5/8 in. tubing arranged to achieve optimum exchanger effectiviness. 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. Two air blast coolers, each one mounted on its own skid, will be used. Each cooler will be equipped with four exhaust fans which require a total of 7.5 kw per cooler. The glycol system will function with practically no operator attention during all plant conditions and offers inherent reliability from freezing.

11.5 CONDENSATE FEED SYSTEM

The condensate feed system utilizes two stages of pumping from the condenser hot well to the reactor. 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 reactor through the feedwater control valve. Thus, the two stages of pumping are accomplished by two pumps in series. This sytem of pumping has operated successfully on the PM-2A and has 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 blowdown cooler, 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 ft. The reactor feedwater pump will also have a capacity of 60 gpm but will have a TDH of about 1,500 ft. These pumps are centrifugal units with orificed recirculation provided to protect against high temperatures during shutoff conditions.

11.6 EXPORT STEAM 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 by others with 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 a pressure of 300 psia 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 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 the 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-744.

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 bank is composed of ninety-five nickel cadmium cells and supplies 115 volts DC for DC power requirements and to two rotary 2 kw inverters for AC voltage 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.

11.8 STARTUP AND SHUTDOWN POWER REQUIREMENTS

Auxiliary power requirements during periods when the plant is shut down 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 11.1.

TABLE 11.1
AUXILIARY POWER REQUIREMENTS

Item	Power Required (Kw) During Shutdown	Power Required (Kw) During Startup
Glycol Pump	-	36
Auxiliary Glycol Pump	2	-
Heat Rejection Fans	15	15
Feedwater Pumps	-	20
Auxiliary Oil Pump	-	3
Centrifuge Oil Pump	1	1
Condensate Pump	5	5
Waste Process System	4	4
Exciter	-	8
General Ventilation	14	14
Special Ventilation	3	3
General Heating	84	84
Special Heating	50	50
Tunnel Ventilation	30	30
Lighting	30	30
Evaporator Pump	-	-
Miscellaneous Power	15	15
Total Power Required	253	318

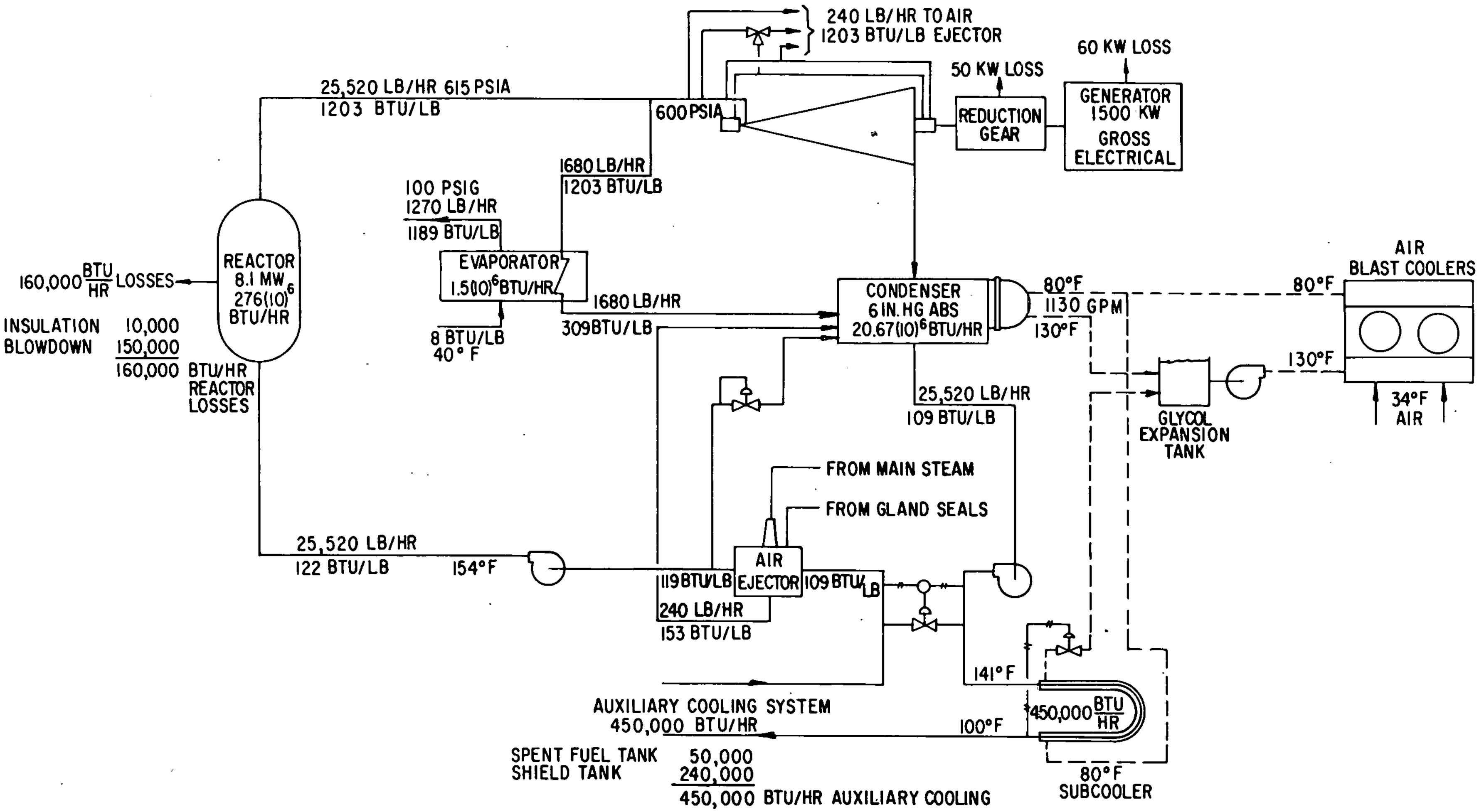


FIGURE 11.1 HEAT BALANCE

12.0 RADIOACTIVE WASTE DISPOSAL

12.1 GASEOUS WASTE COLLECTION AND DISPOSAL

The PL-3 specifications require that any gas discharge from the plant not exceed 4×10^{-14} $\mu c/cc$. Also, the plant should be designed to operate after the simultaneous release to the reactor coolant of 1% of the fission product inventory at the end of core life.

These requirements determine the need for an extremely large activity reduction factor in processing. Several processing systems that appear suitable and adequate are described.

For purposes of design of the gaseous waste system, it was assumed that:

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

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

A detailed study of applicable waste gas disposal systems will be presented in a forthcoming report.

12.1.1 Sources of Gaseous Activity

The sources of potential release of gaseous activity to the environment include:

1. Active gas present in the reactor coolant which is removed with the steam and which appears at the air ejector.

2. Any gaseous activity present in the pools.

3. Gases in liquid leakage from the primary system.

4. Sampling and automatic analysis equipment.

5. Tank vents.

The first item represents the major source of gaseous activity in a boiling water plant. In order for a BWR to be suitable for PL-3 application, it is necessary to provide a gaseous waste disposal system which will process activity released in the reactor, reducing its level to the required discharge limit of 4×10^{-14} $\mu c/cc$.

Gaseous activity present in the BWR coolant is stripped from the coolant by boiling. This activity is carried with the steam to the turbine and condenser. Non-condensable gaseous activity is removed from the condenser by a steam jet ejector. Also present at the ejector are other non-condensables such as hydrogen and oxygen from radiolytic decomposition of water, and any air in-leakage to the condenser. For a plant of PL-3 size, the total volume of hydrogen and oxygen would be approximately 0.5 standard cubic feet per minute (scfm). Air in-leakage would approximate another 0.5 scfm. This assumes that a turbine shaft seal arrangement similar to that shown in Dwg. AES-594 is used to prevent shaft air in-leakage from reaching the main condenser.

The non-condensables or "off-gas" present at the ejector are:

1. Fission gas activity from fuel cladding failure or tramp U-235 on the cladding exterior.

2. Induced activity from activation of oxygen and hydrogen in the core.

3. Hydrogen and oxygen from radiolytic decomposition of reactor water.

4. Air in-leakage to portions of the system below atmospheric pressure.

The 1.0 scfm resulting from items 3 and 4 above is by far the major volume of non-condensables present. Therefore, the BWR gaseous waste system must process a continuous flow of non-active gas containing a trace of active gas. However, this trace represents much greater activity than can be discharged directly from the plant.

12.1.2 Processing Methods

All methods of processing radioactive waste depend on one or more of the following:

1. Dilution
2. Holdup for decay
3. Containment

Dilution is probably the easiest method for disposing of radioactive gas. A gas with an activity above limits is mixed with a sufficient quantity of non-active gas so that the activity of the resulting mixture is then within limits. In the PL-3 plant, the air blast coolers provide a source of dilution air, varying between 200,000 and 400,000 cfm depending on air temperature and plant load.

The active gas or active components can be held-up before discharge to allow the activity to decay. Where volume is small or required decay time is short, the entire volume can be heldup. However, where volume is large or decay time long, it is more desirable to concentrate and hold up only the active components, and to allow the non-active portion to be discharged. This method is normally not applicable to isotopes with extremely long half lives, such as Kr^{85}.

Containment involves the storage of all gas containing activity. High pressure storage and liquefaction are two storage methods. For large volumes, liquefaction is preferred. A variation of containment would be to separate and contain only the active components.

Processing methods which apply one or more of these three principles are discussed in the following subsections.

12.1.2.1 Chemical Combination

Noble gases represent the major portion of the gaseous activity released from a reactor during normal operation and are essentially inert. However, traces of active elements such as iodine and cesium are also present, and these can be removed through chemical action. For example, chemically treated adsorption media have proven to be quite effective for the collection of iodine.

12.1.2.2 Recombination

Recombination of hydrogen and oxygen in the off-gases forms water vapor which is condensed, thereby reducing the volume of gas requiring processing. This can be accomplished by passing the gas through a catalytic bed.

12.1.2.3 Adsorption

The processing of active gas by adsorption is a procedure whereby the active components become attracted to and attached to the surface of an adsorber and held up. There are several possible mechanisms for the operation of an adsorption bed.

1. As a gas flow enters the bed, the active components are adsorbed, using up capacity. This process continues until the bed is saturated. At this point, the bed is no longer effective and must be regenerated or replaced.

2. The second possibility starts in the same manner, with the bed adsorbing the active gas components. However, as additional active gas enters the bed, it displaces those active components already adsorbed. The displaced activity is readsorbed a short distance along the bed. In this manner, the entering active components act to push the previously adsorbed activity through the bed. When activity finally breaks through, it is the oldest activity in the bed. Based on this model, an adsorption bed could be used indefinitely to provide for holdup and decay.

In reality, the mechanism of an adsorption bed is somewhere between (1) and (2). The prediction of holdup time for continuous operation is a little uncertain; however, it is possible to predict the earliest time at which activity will appear in the effluent when a new bed is put into operation. This prediction of "breakthrough time" is used as the design basis for the adsorption systems proposed for PL-3.

The temperature of an adsorption bed greatly effects its operation. The lower the temperature, the smaller the bed for a given set of conditions. The two remaining bed variables are pressure and total gas flow rate. Where the active components represent only a small fraction of the total gas present, the bed design is controlled by the total gas flow. This nonactive gas, or "sweep gas", then determines the bed design flow rate. It is desirable to hold this sweep gas flow rate to a minimum, since the bed size is directly proportional to the total gas flow rate. The size of the bed is inversely proportional to the total pressure.

12.1.2.4 Distillation

Fractional distillation of liquified air can be used to remove krypton. The krypton content of the still effluent can be limited to 0.1 ppm.

12.1.2.5 Other Methods

Other processing or collection methods such as clathrate formation and liquid stripping are available for use with active gas. However, sufficient information is not available for immediate application to PL-3.

12.1.3 Conceptual Processing Systems

Two alternate systems are proposed for processing the BWR gaseous wastes, one a cycling adsorber system and the other a continuous adsorber system. A final choice of systems could be made only after a complete design analysis which is beyond the scope of this work.

12.1.3.1 Cycling Adsorber System

Because it is possible to predict minimum breakthrough time for an adsorption bed, two beds can be connected in parallel with one on stream and the other on standby. When activity breaks through, the standby bed can be placed in operation while the other is degassed and then placed on standby. In this manner active gas is separated from non-active gas, which can be discharged to the environment. The desorbed active gas is collected and shipped off-site for disposal.

A schematic of this system is shown in Dwg. AES-601. Operation of the system is as follows:

1. The off-gas is recombined to remove hydrogen and oxygen. Ejector steam serves to dilute the gas below the explosive limit and cool the recombiner.

2. A condenser removes the major portion of water vapor and steam. This condensate is drained back to the primary system.

3. The gas is passed through a small freezer operating at tunnel temperature. Two freezers are alternated in this service; one is removing water vapor while the other is being heated to melt the ice. This water is also drained back to the primary system.

4. The gas is then passed through two of the adsorber beds in series. The beds operate at tunnel temperature (-10°F). A third bed is available for standby.

5. A radiation monitor and a flow meter after the adsorber beds indicate the activity and amount of the gaseous waste being discharged to the air blast cooler discharge stack. This allows activity breakthrough to be detected and corrected.

6. At breakthrough, the first bed is removed from service, the second placed first in line and the standby bed placed in the secondary position. This changing of beds is accomplished through valves, and takes only a few seconds.

7. Beds removed from service are degassed as follows:

   a. The bed is heated and lowered to a pressure of 1 mm Hg. The low pressure and elevated temperature will desorb the active gas.

   b. A nitrogen purge through the bed displaces the desorbed gas and carries it to a cold trap.

   c. A small liquid nitrogen cooled adsorption bed (cold trap) removes the active gas from the nitrogen purge. This bed can easily be contained and shipped.

8. An alternate method for desorbing a bed might be to use superheated steam. In this case, the steam is removed from the desorbed gas by condensation. Then the active gas is compressed into bottles for off-site disposal. Steam regeneration eliminates the need for extra low temperatures. However, problems could be encountered with condensation and freezing in the bed being desorbed. While the cycling capability must be included to provide for the processing of high off-gas activity, it may be needed only infrequently during plant operation. Therefore, if the Kr-85 activity is low enough, and the adsorption beds have provided enough hold-up time to allow the shorter half-life activities to

decay, then the bed effluent might possibly have low enough total activity for direct discharge to the air blast cooler stack.

12.1.3.2 Continuous Adsorption System

A continuous adsorption system is based on the principle that an adsorption bed will provide a definite holdup time for active gas components, which can be calculated. Such a system is of little use for activity with an extremely long half life such as Kr-85. All other active gases decay to an acceptable level in a reasonable length of time. Therefore, a continuous adsorption system is dependent upon the Kr-85 activity of the off-gas. To make the system workable under all conditions, an additional Kr-85 collection system must be provided.

A schematic of the basic system is presented in Dwg. AES-600. The recombiner, condenser and freezer operate identically to that described for the cycling adsorber system.

After the freezer, a small replaceable bed removes remaining moisture and chemically active components such as the halogens and cesium. Following this small primary bed are the secondary beds which adsorb the radioactive noble gases and provide required holdup time for decay. After decay, the gases are discharged through a radioactivity monitor and a flow meter to the stack. In the event of high activity (representing Kr-85, since all other activity is presumed to have decayed), the discharge to the stack and the bed effluent are directed to a krypton-85 processing system.

12.1.4 Krypton-85 Processing Systems

1. One positive method of preventing the discharge of high activity to the stack is to store the entire off-gas flow. Because a rather large volume is involved, this can be accomplished by storing it as a liquid. When the bed effluent activity is once again at a low level it can be placed back on stream to the cooler stack. The accumulated liquid is shipped off-site for disposal.

2. Fractional distillation of liquid air, as described in Dwg. AES-599, can be used to limit the krypton content in air. However, because of its complexities, the system is not considered suitable for PL-3 application.

3. Through the use of very low temperatures, it is possible to have a small size adsorption bed with a breakthrough time of over 6 months (see Dwg. AES-597). In the event of high activity due to Kr-85 or some other source, the entire off-gas flow is diverted through such a small supercooled adsorption bed (cooled to between -260°F and -295°F with either liquid nitrogen or methane). A cold trap precede the bed to remove any water vapor or carbon dioxide. When exhausted, the bed is shipped offsite for disposal. This system alone could provide a complete gaseous waste disposal system with all fission gas activity being retained in the adsorption bed, however it would be less reliable than a combined system because of the total dependence on maintenance of low temperatures.

12.2 LIQUID WASTE DISPOSAL

PL-3 specifications require that any liquid discharged from the plant will not exceed 10^{-8} μc/cc. Evaporation is the only practical method of processing radioactive liquid wastes to this specification. The evaporation effluent or processed waste can be reclaimed and used as primary system or pool make-up.

The arrangement of the liquid waste disposal system is indicated in Dwg. AEL-710. The various wastes are collected in a storage tank where they can be mixed and sampled and pH adjustments made. After adjustment, the waste is processed by evaporation. The condensate is passed through a demineralizer (to guard against carryover) and stored for reuse or discharge. The activity and other solids in the evaporation feed concentrate in the evaporator. When a predetermined quantity of solids have been accumulated, the evaporation process is interrupted and the concentrated contents of the evaporator sent directly to a disposal drum which will be precharged with a dry concrete mixture. By going directly to the disposal drum, problems associated with storing and pumping a highly active sludge, which may solidify, are eliminated. After adding the evaporator bottoms to the drum, the contents are mixed by rolling and then allowed to stand while the concrete sets. The contents are self-shielded, however additional shielding will be provided if required.

This liquid waste disposal system is designed to process any waste which may reasonably be expected from a PL-3 type plant. All equipment is mounted on a single skid (see Dwg. AEL-747) in an arrangement which provides for easy access to components for operation and maintenance. Also the layout makes use of water storage capacity to provide additional shielding of the components.

12.3 SOLID WASTE DISPOSAL

All 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.

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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-738. 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 be similar to those now installed at the site and will consist of vertical ladders in 5 ft dia 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 and 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 to this 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 tape 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 pre-construction 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.

Drawings 7385-SK-S-8 illustrates a preliminary arrangement of foundations and superstructure of the primary building for a boiling water reactor with the reactor enclosed in a single vertical containment vessel.

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 system 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 about 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.g. 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°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. dia, inserted 20 to 25 ft down 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 permissible 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 a prolonged period 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 space setting of a thermostat (T_1) and with the air conditioner 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 shutdown 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 shutdown 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 affored 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 and at 0°F, and approximately 300 psig. Normally the electric heater will maintain the internal temperature of the storage unit at 0°F. Should the tunnel temperature rise above 0°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 pushbutton (7) located at each building exit, or automatically by heat-actuated devices (8) located in the vicinity of potential fire hazard. 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 wherever possible rather than outside in the below-freezing temperatures. Pitch of the system will be a minimum of 1/8 in. per ft, 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, floor, 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, an 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 $-80^{\circ}F$. 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 facilities will contain the equipment necessary to prepare low-activity samples for 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. A 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 FACILITIES

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.

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 a load-following control rod positioning system. 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.

These 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, who will also be required to direct the processing and disposal of radioactive waste. This will be an intermittent operation on a batch basis.

Refueling will be scheduled every four years, utilizing the entire crew for five 12 hr days.

The simplicity of the direct cycle BWR will result in minimum maintenance requirements for the reactor plant. Standard power plant equipment and auxiliaries are specified. Normally scheduled maintenance following the manufacturer's recommendations results in minimum downtime and manpower requirements. On-line maintenance of power plant equipment in the areas of the turbine inlet, the condenser hot well and the air ejectors will be complicated somewhat by the radiation fields in these locations. Maintenance will be routine when the plant is shutdown.

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

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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 74 shipping modules, each one a single aircraft load. The total consists of 28 modules of foundations, buildings and support facilities, 24 modules for the reactor system and 22 modules for the power conversion system 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 system and the power conversion system 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-eight plane loads will be required during the first (1962-1963) construction period and 46 will be required during the second (1963-1964) 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 some of the spare parts, spare equipment and operating supplies during 1962. The above totals do not include the shipping requirements for the emergency diesel fuel oil. Based on 10 percent 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 in order to expedite erection and to insure startup of the plant on schedule.

TABLE 16.1
SHIPPING MODULES

Contents	Number	Weight-lb	Cubage-ft^3
Foundations, Buildings and Support Facilities			
Foundations and Jacking	10	157,600	15,100
Buildings and Superstructure	11	206,200	19,000
Support Facilities	7	108,940	10,640
TOTAL	28	472,740 lb	44,740 ft^3
Reactor System			
Reactor Vessel	1	17,300	960
Vapor Container			
Lower	1	19,000	960
Upper	1	18,000	960
Lead Shield:			
Bottom	1	18,700	960
Bottom Cylinder	1	19,500	960
Top Cylinder	1	19,000	960
Shield Tanks and Misc. Equipment	3	57,000	2880
Reactor Vessel Head and Shield Tank	1	19,000	960
Fuel Transfer Cask	1	19,500	960
Fuel Shipping Casks (23,160 lb each)	8	185,280	4480
Spare Parts and Equipment	2	39,600	1920
Miscellaneous	3	59,400	2880
TOTAL	24	491,280 lb	19,840 ft^3

	Number	Weight-lb	Cubage-ft^3
Power Conversion System and Auxiliaries			
Turbine	1	15,450	686
Generator	1	15,000	210
Turbine Auxiliaries	1	16,700	728
Condenser	1	20,000	1792
Air Blast Coolers	2	39,000	3840
Secondary Auxiliaries	1	19,500	960
Control Console	1	13,600	960
Switchgear	1	13,100	960
Waste Processing	1	20,000	1792
Chemical Laboratory and Maintenance and Decontaimination Equipment	1	11,800	1760
Auxiliary Power (2-250 Kw units)	2	40,000	864
Piping	2	31,000	3840
Auxiliary Electrical	2	37,450	3810
System Supplies	1	20,000	400
Lead Shielding for Waste Disposal System	2	30,500	640
Spare Parts (tools and misc.)	1	20,000	1280
Spare Components	1	20,000	1280
TOTAL	22	383,100 lb	25,802 ft^3

16.2 CONSTRUCTION SCHEDULE, MANPOWER AND EQUIPMENT

The PL-3 plant will be constructed in two phases; (a) erection of the foundations, buildings and support facilities, and (b) 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 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 reactor and plant systems components 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.

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 16.2
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
TOTAL	68	
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
TOTAL	70	

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

	Number	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 Sup't.	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 arive for start up 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, timer 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
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 costs 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	$ 100,000
Core Structure	15,000
Vapor Container	99,000
Control Rod Drives	81,000
Shielding	120,000
Shipping Casks	65,000
Air Blast Coolers (2)	90,000
Turbine Generator	250,000
Condenser	55,000
Secondary Auxiliaries	43,000
Turbine Auxiliaries	35,000
Waste Processing Equipment	58,000
Chemical Laboratory	27,000
Maintenance Facility	55,000
Diesel Generators (2)	78,000
Switchgear	43,000
Control Console	280,000
Radiation Monitoring System	33,000
Reactor Core (First Core Costs)	270,000
*Miscellaneous Items	406,000
	$ 2,113,000
Support Facilities	665,000
	$ 2,778,000

Spare Components

Control Rod Drive	$ 10,000
Control Rod Basket & Rack	5,000
Nuclear Instrumentation	20,000
Second Reactor Core	180,000
	$ 215,000

Spare Parts

Fuel Oil - 10% Supply for 18 months	$ 10,000
Reactor Vessel Studs, Nuts and Bolts	6,000
Trip and Control Valves	10,000
Process Instrumentation	40,000
Nuclear Instrumentation	23,000
Control Console	15,000
Turbine and Generator	110,000
Diesel Generator	16,000
	$ 230,000

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

17.2 FUEL CYCLE COSTS

Fuel cycle costs were computed for the reference PL-2 BWR core based on the following assumptions:

1. Four year core life at 0.8 plant factor (25.5 MWYRS).

2. Fabrication costs are based on the average of the latest quotations for two complete cores and include 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. Plutonium credit at $9.50 per gram less $1.50 per gram for conversion to metal, and 1% losses.

5. Chemical reprocessing plant charges @$17,600 per day, $5.60 per KgU for conversion to UF_6 and 1.3% losses.

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

7. Four spent fuel shipping casks @ 23,160 lb each, $1,200/ton shipping charge.

8. A use charge of 4-3/4% is applied only to the two full cores plus 10% spare elements 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.

TABLE 17.2
FUEL CYCLE COSTS FOR REFERENCE BWR

Cost Item	4 yr. Core	$/per Core	$/per Year
1. Fuel Burnup Cost			
Kg U Initial	$ 1140		
Kg U Final	1125.28		
Initial Enrichment	4.8%		
Final Enrichment	3.9%		
$ per Kg U Initial	509.70		
$ per Kg U Final	396.71		
Initial U Value	581,058		
Final U Value	446,410		
Plutonium Credit	21,463		
Burnup Cost		$ 113,185	$ 28,296
2. Core Fabrication Cost		225,000	56,250
3. Reprocessing Costs			
Conversion to UF_6	6,302		
Plant Costs	55,264		
Reprocess Losses	5,803		
Total Reprocessing Costs		67,369	16,842
4. Shipping Costs			
Fresh Fuel	1,000		
Spent Fuel	55,584		
Total Shipping Costs		56,584	14,146
FUEL CYCLE COSTS WITHOUT USE CHARGE		$ 462,138	$ 115,534
5. Fuel Use Charges			
In-Reactor Core	97,609		
Spare Core	110,401		
10% Spares 1 Core	11,040		
Total Use Charges		$ 219,050	$ 54,762
TOTAL FUEL CYCLE COSTS		$ 681,188	$ 170,296

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; it is assumed that all the operator trainees will be trained together and that 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 will have as yet operated together as a crew. Emphasis must be given to qualifying the trainees as PL-3 operators and also to 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. There is no comparable training facility for BWR, and the possible alternates, BORAX, VBWR and EBWR, are basically research facilities. Extensive use will therefore be made of a simulator to familiarize the trainees with PL-3 performance and operation. This indoctrination will be supplemented by instruction and tours at the BWR facilities.

All trainees will then receive 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, will be approximately ten months.

TABLE 18.1
PL-3 OPERATOR'S TRANING PROGRAM

Session	Training Description	Location	Instruction Hours
1	Integrated Crew Training	SM-1	40
2	Plant Information and Orientation	SM-1	80
3	Simulator Operation	SM-1	120
4	BWR Facilities Tour	Sites	160
5	Test Site Orientation	Sch/dy	40
6	Plant Systems	Sch'dy	40
7	Plant Procedures	Sch'dy	40
8	Specialty Training	Sch'dy	
8A	Health Physics Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
8B	Chemistry Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
8C	Electrical Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
8D	Instrument Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
8E	Mechanical Specialty		
	(a) Classroom & Laboratory		40
	(b) Practical		160
	Review of all Specialties (10-15 hrs/each)		40
9	Assembly, Disassembly and On-Site Testing	Sch'dy	40
10	Plant Operation and Testing	Sch'dy	160
11	Plant Buildings	Sch'dy (J & M)	20
12	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, Plant Information and Orientation

The objective of Session 2 will be the familiarization of the trainees with the PL-3 plant by comparisons with the SM-1 plant. A classroom lecture series will acquaint the trainees with the basic differences in functional performance and plant systems. Conference discussions will also be held to compare the operational procedures of PL-3 with SM-1. Prints, photographs and cut-away drawings will be used as training aids. This session will involve 80 hrs of lectures and discussions.

18.1.3 Session 3, Simulator Operation

This session will be tailored to complement the simulator operation course work of the SM-1 Operator's Training Program by developing a general knowledge of the PL-3 plant through practical exercises and demonstrations on the PL-3 BWR simulator. This 160 hr session will familiarize the trainees with the PL-3 arrangement of instrumentation and controls and will provide simulation and practice of both normal and abnormal plant operational procedures. Conference discussions will treat the interpretation of results, diagnosis and corrective actions.

18.1.4 Session 4, BWR Facilities Tour

A four-week period is allocated for BWR facilities tour by the PL-3 operator trainees. The facilities which might be scheduled are BORAX IV, VBWR and EBWR. The main purpose of the tour would be to furnish the trainees with a general working knowledge of BWR systems and the opportunity to discuss operational techniques, problem areas, etc. with experienced personnel who are also familiar with the trainee's specialty categories.

18.1.5 Session 5, Test Site Orientation

Prior to Session 5, the trainees will be given approximately one week in which to re-locate to Alco's Schenectady Plant. Session 5 will resume the training program with 40 hours of indoctrination at the stateside test facility for the PL-3, consisting of 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.

18.1.6 Session 6, 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.

18.1.7 Session 7, Plant Procedures

This session will follow Session 6 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.

18.1.8 Session 8, 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 8A, Health Physics, and 8B, 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 hr 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 prove 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 participate except to give guidance where necessary.

18.1.8.1 Session 8A, 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.

18.1.8.2 Session 8B, 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. Radio-chemistry procedures will be taught in the Alco laboratory. Session 8B will be divided into 20 hrs of lectures and 20 hrs of laboratory testing.

18.1.8.3 Session 8C, 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.

18.1.8.4 Session 8D, Instrument Specialty

Instrument specialists will receive their specialty training in 40 hrs of lectures and on-the-job instrument training during plant testing. Session 8D lectures will cover the theory of operation of all plant instrumentation and control systems, with comparisons being made to the SM-1 systems. Manufacturers' literature will be the principal training aids. A minimum of 40 hrs 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.

18.1.8.5 Session 8E, Mechanical Specialty

Forty hours of lectures and demonstration will be given mechanical specialists 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.

18.1.9 Session 9, Assembly, Disassembly and On-Site Testing

All operator trainees will attend 40 hrs 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.

18.1.10 Session 10, 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 hr of this on-the job training will be received as an integrated military unit within Alco's operations group.

18.1.11 Session 11, Plant Buildings

Representative of Jackson and Moreland will give 20 hr of class-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.

18.1.12 Session 12, 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 and gaseous waste disposal systems at Alco's stateside facility. This training will be given at opportune times during intial checkout of the equipment by the vendor's 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 hrs 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 hrs of lectures and group discussions of the details of disassembly, packing, and assembly of the plant. The sequences 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 hr 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 conference 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 jobside 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.

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APPENDIX A

DESIGN INFORMATION FOR A HIGHLY ENRICHED BOILING WATER REACTOR

1.0 INTRODUCTION

A substantial design analysis effort was expended in the course of the evaluation study on development of a direct-cycle natural circulation boiling water reactor concept using fully enriched, plate-type fuel.

Design information developed for this reactor is summarized here to present fully the findings of the BWR studies and to facilitate direct comparison of the effects of full versus low enrichment fuel in a boiling water reactor for this application.

The reactor power plant would differ from the plant described in the previous text only in the design of the reactor complex. It would use the same power plant equipment, auxiliaries, and services, and it would operate at the same conditions of power level, steam flow, and feedwater return.

The differences in the two concepts are confined to the reactor core, the reactor vessel and its internals, and the dimensions of the vapor containment and shielding.

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2.0 REACTOR DESCRIPTION

The arrangement of the reactor complex is shown in Dwg. AEL-722*. Design information for the reactor is summarized in Table A.1.

The reactor vessel is similar to that for the rod-type reactor except for its smaller diameter. The vessel internal diameter is 53-1/2 in. for its entire length. The description of the vessel, provisions for refueling, and considerations of water level variation and steam separation are essentially as described for the rod-type reactor.

Dimensional changes in the vapor containment and shield radii are evident from comparison of Dwgs. AEL-722 and AEL-740.

The reactor core consists of 60 SM-2 type flat-plate elements, modified in length, fuel-bearing plate width, and number of plates (see Dwg. AEL-746). The elements are fastened together in groups of three and four, with each group shrouded by 1/16 in. stainless steel boxes which are permanently fastened to the lower grid plate. The shroud boxes are fastened to each other above the core to form slots for the control rod travel and all plates of the boxes except those at the core-reflector boundary are cutaway over the active fuel region to reduce the steel content of the core and to facilitate flow in the rod gaps.

The fuel elements rest on the lower grid plate and they are not held down. The lower grid plate is supported by the core support structure which is welded to the vessel.

The fuel subassemblies each contain 8 fuel plates with 30 mil meat and 5 mil clad of Type 347 stainless steel. The active length of the fuel plate is 28 in. The meat consists of highly enriched spherical UO_2 fuel particles and ZrB_2 burnable poison dispersed in a prealloyed Type 347 stainless steel matrix. After pressing, vacuum sintering, and coining, the core is placed in a Type 347 picture frame, and Type 347 cover plates are welded on the top and bottom. This assembled billet is metallurgically bonded by hot and cold rolling.

Eight fuel plates are assembled into grooved side plates and are welded by automatic TIG procedures. Three or four of these sub-assemblies are welded to end fixtures, making a completed fuel assembly.

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* Dwg. AEL-722 depicts an earlier design concept using cruciform rods with one rod in the center of the core. This required that the drive shafts for the rod drives be on two levels. This is unnecessary in the present design; otherwise the drawing represents accurately the reactor described.

Tee-shaped rods are incorporated to eliminate a center rod in this particular core geometry, thereby reducing the maximum worth of a single rod. Eight control rods are provided, which differ only in shape and length from the cruciform rods described for the low enrichment reactor. The rods would be gripped at the centroid of their cross-section. The rod drives would be identical to those described for the low enrichment reactor.

TABLE A.1
DATA SUMMARY - HIGH ENRICHMENT PLATE TYPE REACTOR

General Characteristics	
Power Level	8.1 Mw
Design Life	2.3 yrs at 6.36 Mw average power
Reactor Pressure	615 psia
Feedwater Temperature	154°F
Fuel Type	Flat plate UO_2-SS matrix
Vessel ID	53.5 in.
Vessel Material	Carbon steel, clad with stainless
Vessel Fast Neutron Exposure	
Power Density	27.9 Kw/liter reactor 31.5 Kw/liter coolant 41.3 Kw/liter active coolant
Core Description	
Active Fuel Length	28 in.
Equivalent Core Diameter	28.4 in.
Number of Elements	60
Fuel Element Array	8 x 8 with corners missing
Plates/element	8
Overall Plate Thickness	0.040 in.
Clad Thickness	0.005 in.
Gap Thickness	0.319 in.
Fuel Element Shroud Boxes	1/16 in. SS around four elements - 5.875 in. inside
Pitch of 4 Element Cell	6.5 in.
Rod Gaps	0.5 in.
Fuel	
Fuel Enrichment	0.9317
Fuel Loading	25 kg U^{235}
Meat Volume	16.716 liters
Meat Composition	22.69 w/o UO_2 in SS
Initial B^{10} Loading	31 gm
Control Rods	
Number	8
Shape	Tee-rods
Span	12 in. long side, 6 in. short side
Thickness	0.250 in.
Material	Ag-Cd-In, Ni-plated and partially canned in SS
Thermal and Hydraulic Data	
Riser Height	3 ft
Average Void Fraction	0.14
Maximum Exit Void Fraction	0.40
Downcomer Velocity	1.18 ft/sec
Core Flow Rate	645 lb/sec
Total Steam Generated	7.48 lb/sec
Average Heat Flux	58,500 Btu/hr ft^2
Maximum Heat Flux	316,000 Btu/hr ft^2
Maximum Centerline Fuel Temperature	545°F
Nuclear Data	
Metal/Water Ratio:	
Within Plate Lattice	0.1865
Average Core Without Rods	0.1298
Reactivity in Voids	2.6% Δ k/k
K_{eff} Initial, Hot, Voided, No Xe	1.112
K_{eff} Midlife, Hot, Voided, No Xe	1.136
Shutdown K_{eff}, End of Life 4°C	0.891
Shutdown K_{eff}, End of Life 4°C Maximum Worth Rod Full Out	0.935
Void Coefficient at End of Life, 4°C	$-6.0 \times 10^{-4} \Delta\rho/\%$ void
Temperature Coefficient at End of Life, 4°C	$-1.3 \times 10^{-5} \Delta\rho/°F$

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3.0 NUCLEAR ANALYSIS

3.1 SELECTION OF CORE SIZE

To obtain reasonable assurance of stable operation, it is advisable to design for no greater reactivity in voids that has been attained in reactors of similar fuel time constant which have deomonstrated stable operation at similar pressure. Because of the relatively small fuel time constant associated with the thin fuel plates of a highly-enriched reactor, it is necessary to design for less reactivity in voids than in the case of a low enrichment reactor using UO_2 rod-type fuel.

Scoping studies established that the reactivity associated with voids in a fully-enriched reactor of given power level is very sensitive to the size of the core. The amount of reactivity in voids may be regarded as the product of the average void coefficient of reactivity and the average void fraction in the core, each of which tends to vary inversely with core size. The void coefficient of reactivity for a given fuel concept, and thus for a given solid-to-water ratio, depends on the fast neutron leakage characteristics, and hence on core size. This is particularly true in the case of fully-enriched cores, because they do not have the additional degree of freedom afforded low-enrichment cores through the U^{238} resonance absorption effects, which depend on cell configuration rather than core size. The average void content for a given power level also varies with core size, thus the reactivity associated with voids is doubly dependent on the size of the core.

The full enrichment core must be large enough to minimize the reactivity in voids, yet small enough to meet the specific PL-3 requirement that the void and temperature coefficients be negative under all expected shutdown conditions.

The preliminary scoping studies[1][2] led to selection of a core 28 in. high by 28 in. in diam which satisfies these criteria, consistent with the requirements of hydraulics and heat removal, core lifetime, fuel-bearing capacity, burnup limitations, and control.

3.2 SUMMARY OF NUCLEAR CHARACTERISTICS AND METHODS OF CALCULATION

The nuclear characteristics of the core were analyzed on the basis of uniform voids corresponding to the average void content predicted by STREAC calculations. The axial power distribution assumed in the STREAC calculations corresponded to that in PL-2 at initial operation. The void content was distributed in the active coolant in the fuel elements. At 8.1 Mw the void content is 18.5% in the active coolant and 14.1% averaged over the total coolant including the water in the rod gaps.

Reactor analysis was performed using two-dimensional two-group diffusion calculations with the IBM 7090 version of the PDQ code(3). Epithermal cross sections were calculated using the MUFT III code(4) which utilizes a Fourier Transform method to calculate the epithermal neutron flux spectrum based on the slowing down distribution from a point source, then averages the cross sections over this spectrum. Thermal cross sections were obtained using the SOFOCATE code(5). The SOFOCATE code incorporates a Wigner-Wilkins flux spectrum calculation and averages the cross sections over this calculated spectrum. Thermal disadvantage factors for thermal cross section homogenization were calculated using the two-dimensional P-3 Code(6), which is basically a superposition of several P-3 approximation solutions to the one-dimensional, single velocity transport equation.

The critical mass of the hot, voided, clean core was determined to be 12.5 kg of U^{235}. Additional fuel for burnup and other lifetime considerations increases the fuel requirement to an initial loading of 25 kg. In order to meet the stuck-rod requirement, supplementary control is provided by 31 gm of B^{10} homogeneously distributed in the fuel meat in the form of ZrB_2. The reactivity associated with voids at beginning of life with all rods out was calculated to be 2.6% $\Delta\rho$ for the design value of 14.1% average core void content. The temperature defect to 4°C at beginning of life, with all rods out, is 1.2% $\Delta\rho$. Uniform burnup studies show that maximum core reactivity will occur at about 2500 MWD, or roughly at the middle of core life. At this time the shutdown K_{eff} at 4°C with the maximum worth rod stuck full out is 0.935, giving a shutdown margin of 6.5% ΔK. The control rod worth calculations involve some uncertainty associated with the epithermal rod worth for the Ag-In-Cd rod material, therefore a conservative shutdown margin is desirable.

The uniform burnup analysis shows a maximum lifetime of 3.17 yrs, corresponding to 7400 MWD. The effect of non-uniform burnup is expected to reduce lifetime by a factor of 1.3 to 1.5, indicating a core life of 2.44 to 2.11 years.

3.3 REACTIVITY AND LIFETIME

Burnup calculations were performed on the basis of uniform burnup of uranium and boron and a constant uniform void content, with Xe, Sm and accumulation of other fission products considered. As shown in Fig. A.1, reactivity maximizes at about 2500 megawatt-days due to the more rapid rate at which the boron is depleted initially, relative to the fuel.

The calculation indicates a lifetime of 7400 MWD, equivalent to 3.17 yr at an average thermal power of 6.36 Mw. Prediction of lifetime is speculative in the absence of detailed studies which include the spatial variation of burnup and the effects of rod movement, however it is expected that these effects will reduce the lifetime by a factor of 1.3 to 1.5, indicating a core lifetime of 2.44 to 2.11 years.

Fuel loading, lifetime, and reactivity data are presented in Table A.2 where the end-of-life data are based on a "nominal" lifetime estimate of 2.3 years.

It may be noted that the void fraction and the reactivity in voids should diminish with life as rods are withdrawn and power shifts upward in the core. However, the reactivity recovery should be less than in PL-2 because the reactivity in voids is less. Also, the calculation neglected the reduction in flux depression within the fuel plate and the softening of the thermal spectrum which will accompany burnup, and which should tend to lengthen lifetime.

These considerations suggest that the estimate is conservative and that the core will easily meet the minimum lifetime of 2 years.

3.4 ROD WORTH AND STUCK ROD CRITERION

The worst case from the standpoint of ability to meet the stuck rod criterion will be at midlife of $4^{\circ}C$ where the reactivity of the core will be at its maximum and the worth of the control rods at minimum. The effective multiplication of the uncontrolled core for this worst case (mid-life, $4^{\circ}C$ without xenon) is 1.185.

Two-group neutron-diffusion theory calculations for control rod worth were made for this case, using the standard "black" logarithmic boundary conditions for the thermal group, and treating the rod as a diffusion region in the epithermal group. The control rod cross sections in the epithermal group will be dependent on leakage spectrum, and thus on core size and metal-to-water ratio. There was insufficient time to do any detailed analysis of the important geometric and energy self-shielding effects which occur in the rod. However, investigation of the available literature [7][8][9] which presents calculational methods and experimentally determined parameters for similar rod materials led to the choice of $D = 0.765$ and $\sum_a = 0.13$ for use as the epithermal parameters.

The results of these calculations give an effective multiplication constant at midlife, $4^{\circ}C$, without xenon, for all rods in, of 0.891. With the most reactive rod stuck out (a corner rod), the effective multiplication constant is 0.935, giving a 6.5% Δk shutdown margin for the worst stuck rod case to be encountered.

Despite the uncertainties associated with the epithermal rod worth calculations, there is sufficient shutdown margin to allow for considerable error. Also, the values presented are for the case of uniform burnup, which overestimates the reactivity to be controlled at midlife by an amount estimated to be about 1% Δk.

3.5 REACTIVITY COEFFICIENTS

The total void and temperature defects have been calculated for the beginning of life condition with all rods out. The effective multiplication constant is plotted as a function of average core void content in Fig. A.2. The reactivity in voids between 0 to 14.1% average core void content is indicated to be 2.6% $\Delta\rho$, corresponding to an average void coefficient of -0.184% $\Delta\rho$ per percent void. The total temperature defect down to $4^{\circ}C$, with all rods out, is 1.2% $\Delta\rho$, giving an average temperature coefficient of $-2.7 \times 10^{-3} \Delta\rho$ per $^{\circ}F$.

TABLE A. 2
REACTIVITY AND LIFETIME DATA

U-235 Critical Mass, Clean core, Hot, voided	12.5 kg
Initial Core Loading of U-235	25 kg
Corresponding w/o of UO_2 in fuel meat	22.69%
Fuel Enrichment	0.9317
Estimated Lifetime	2.3 yr *
End of Life Loading of U-235	18 kg
End of Life Loading of B_{10}	5 gms
Average Fuel Burnup	28%
Average B_{10} Burnup	84%
Average Atom Burnup in Meat	1.30%
K_{eff}, Initial, No Xe, Hot with 14.1% Avg. Voids	1.112
K_{eff}, Midlife, No Xe, Hot with 14.1% Avg. Voids	1.136
K_{eff}, Midlife, No Xe, 4°C	1.185
Equilibrium Xe, Sm, End of Life	3%
Maximum Xe, Sm, End of Life**	3.6%

* At average thermal power of 6.36 Mw.

** Maximum occurs 10.4 hr after shutdown.

The more nearly positive reactivity coefficients will occur in the cold condition at end of life when the rods will be farthest out of the core and the absorption by fuel and boron is minimum. In this condition the effective core size will be at its maximum for 4°C and hence, the negative leakage effects will be a minimum.

Calculations of the temperature and void coefficients at 4°C have been performed with end-of-life average core compositions corresponding to 5500 MWD, with all rods out. The resulting void coefficient is $-6.0 \times 10^{-4} \Delta\rho$ per percent void. This void coefficient is more nearly positive than will actually be encountered since rods must be partially inserted to the just critical position to overcome the vapor and temperature defects, thus enhancing the leakage characteristics and making the void coefficient more negative.

The temperature coefficient calculated for the all rods out condition is very slightly positive. With the rods inserted to the critical position the temperature coefficient becomes negative due to the increased leakage and is estimated to be $-1.3 \times 10^{-5} \Delta\rho$ per °F.

3.6 POWER DISTRIBUTIONS

Power distributions and peaking factors have been obtained from PDQ (r, z) and (x, y) calculations at beginning of life conditions. The smeared core (r, z) calculation indicates a radial maximum-to-average power ratio of 1.45. The (x, y) calculation and the assumption that the axial maximum-to-average ratio is the same as that for the PL-2 give a maximum local peak-to-average ratio of 5.4.

The peaking factors at beginning-of-life conditions are presented in Table A.3. As fuel is depleted with burnup and control rods are withdrawn, the power peaking factors will be reduced considerably.

TABLE A.3

POWER PEAKING FACTORS AT BEGINNING OF LIFE

Radial maximum-to-average (smeared core)	1.45
Axial maximum-to-average (with rods, from PL-2)	2.3
Maximum local peak-to-average	5.4
Hottest channel to average channel	1.8
Hottest channel to average channel in hottest 4 element assembly	1.38
Hottest 4 element assembly to average 4 element assembly	1.3

3.7 FAST NEUTRON EXPOSURE OF REACTOR VESSEL

Calculations were performed, as described in Section 7.6, for the full enrichment reactor. The results of the calculations indicate the maximum vessel exposure, over a 20 yr life at an average power level of 6.4 Mw, to be 9.0×10^{18} nvt, approximately the value selected to conform to the design criterion applied to carbon steel.

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4.0 THERMAL AND HYDRAULIC ANALYSIS

4.1 DESIGN CONDITIONS AND ASSUMPTIONS

The thermal and hydraulic analyses of the full enrichment plate-type core were performed using the STREAC code in a manner similar to that discussed for the low enrichment rod-type core. The design conditions and assumptions were the same as presented in Section 6.3 except for the power distributions. The PL-2 beginning-of-life axial distribution was used, but the radial distribution was taken from a uniformly voided PDQ (r, z) calculation. Four regions were used to represent the radial distribution. The number of boxes per radial region were selected so as to best approximate the radial power distribution, and they do not represent actual fuel assembly groupings in the core.

4.2 RESULTS

The results of the thermal and hydraulic analysis of the high enrichment core are presented in Tables A.4 and A.5.

The average steam void fraction in the active coolant is 0.185 and it includes subcooled voids approximated by a straight line drawn tangent to the void distribution curve from the point where heating starts. Subcooled voids by this method account for 0.015 of the total average void fraction.

The downcomer velocity was determined to be 1.18 ft/sec based on the 53-1/2 in. inside diameter of the vessel. Although this is below the carryunder threshold velocity of 1.2 - 1.3 ft/sec,some experimental work would be required to reduce the uncertainty in this calculated value.

The exit steam void fraction from the hottest region was calculated to be 0.320. This is an average value based on uniform heating in the fuel assembly. Because of local power peaking near the control rod water gaps, there will be hot channels having greater exit void fractions than the average value. The maximum exit void fraction from the hottest channel was calculated to be approximately 0.40, based on a 1.4 ratio of the maximum channel power to the box average power.

The pressure losses around the convective loop are given as a percentage of the total loss in Table A.6. The major portion of the pressure gain occurs in the riser, this being 61.4% of the total gain.

The core inlet loss is the major loss term and is subject to considerable uncertainty. An experimental measurement of this loss on a flow model would be required to verify the design.

TABLE A. 4
THERMAL AND HYDRAULIC DATA

Reactor Power	8.1 Mw
Power Density in Active Coolant	41.3 Kw/liter
Feedwater Temperature	154°F
Effective Feedwater Temperature with Carryunder	208°F
Assumed Downcomer Carryunder	5%
Core Flow Area	2.97 ft^2
Core Hydraulic Equivalent Diameter	0.573 in.
Heat Transfer Area	472 ft^2
Riser Height	36 in.
Downcomer Flow Area	10.9 ft^2
Total Coolant Flow Rate	645 lb/sec
Total Steam Flow Rate	7.48 lb/sec
Average Exit Quality	1.16%
Core Inlet Subcooling	3.0°F
Downcomer Velocity	1.18 ft/sec
Average Steam Void Fraction in Active Coolant	0.185
Exit Steam Void Fraction in Hottest Region	0.320
Stability Multiplier for Hottest Box	2.10
Average Heat Flux	58,540 Btu/hr ft^2
Maximum Heat Flux (5.4 Max/Avg)	316,000 Btu/hr ft^2
Maximum Fuel Centerline Temperature	545°F

TABLE A.5
THERMAL AND HYDRAULIC DATA FOR RADIAL REGIONS

Radial Region	1	2	3	4	Core Average
Number of Boxes	1	4	4	6	15
Radial Power Multiplier	1.43	1.21	0.927	0.837	1.00
Inlet Velocity ft/sec	4.78	4.63	4.32	4.19	4.38
Fractional Bulk Boiling Length	0.856	0.841	0.815	0.804	0.820
Exit Quality, percent	1.67	1.40	1.06	0.96	1.16
Exit Steam Void Fraction	0.320	0.287	0.243	0.227	0.253
Average Steam Void Fraction	0.238	0.212	0.176	0.163	0.185

TABLE A.6
COMPONENTS OF TOTAL PRESSURE LOSS

Core Inlet Loss	43.1%
Subcooled Boiling Fraction Loss	5.7
Bulk Boiling Friction Loss	34.8
Boiling Acceleration Loss	18.3
Core Exit Loss*	-14.3
Riser Friction Loss	2.5
Downcomer Loss	9.9
Total Loss (53.9 psf)	100.0%

* The negative core exit loss indicates a net static pressure gain across the expansion.

4.3 EFFECTS OF FEEDWATER TEMPERATURE

The thermal efficiency of the power plant cycle may be improved by the incorporation of regenerative feedwater heating. Despite the reduction in reactor power and the possible improvement in core lifetime, no feedwater heaters were used in the PL-3 design, partly because of the adverse effect high feedwater temperatures has on the thermal-hydraulic performance of the reactor. The results of a study made to determine the effects of reactor power and feedwater temperature on average void fraction, exit void fraction and downcomer velocity are shown in Fig. A.3.

Heat balances were performed for two secondary systems:

a. A non-extraction system with no feedwater heating and

b. An extraction system with two stages of feedwater heating.

The non-extraction system requires a power of 8.1 Mw at a feedwater temperature of 154°F. 7.54 Mw is required for the extraction system at 310°F feedwater temperature. These operating points are shown on Fig. A.3 and indicate the penalties in achieving lower power by this means. Since the fully-enriched core has been calculated to have a lifetime somewhat greater than two years using the non-extraction cycle, this cycle was selected.

5.0 STABILITY CONSIDERATIONS

As discussed in Sections 6.5 and 7.3, stability considerations for a natural circulation boiling water reactor make it advisable to limit the reactivity in voids and the hydraulic performance characteristics to those conditions for which stability has been demonstrated under similar reactor conditions or in applicable convective loop experiments.

BORAX II was a fully-enriched natural circulation reactor with thin Al-U fuel plates having a time constant less than that for the PL-3 plate-type fuel. It operated stably with 3.2% Δk in voids at a pressure of 300 psig.[10][11] At this pressure, greater reactivity in voids could have been accommodated satisfactorily, had it been available. The fully enriched core for the PL-3 application is predicted to have 2.6% $\Delta k/k$ in voids with rods withdrawn. The reactivity in voids at the initial operating rod position is probably nearly as great as in BORAX II, but because of its higher design pressure, this design should be adequate from the standpoint of coupled nuclear-hydraulic stability.

The calculated maximum channel exit void fraction of 0.40 is considerably less than that known to cause hydraulic instability in electrically heated loop tests and is approximately equal to that measured in natural circulation reactors which have operated stably. The factor by which the power of the hottest box must be increased to cause the flow rate to decrease with increasing power is calculated to be approximately 2.1. From these considerations the plate-type core appears adequate from the standpoint of hydraulic stability.

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6.0 FUEL CYCLE COSTS

Fuel cycle costs were computed for the full enrichment BWR core based on the following assumptions:

1. 2.3 year core life at 0.8 plant factor (14.6 MWYR).
2. Fabrication costs are based on the average of the latest quotations for two complete cores and include 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 kgU for conversion to UF_6 and 1.3% losses.
5. Shipping charges for fresh fuel Schenectady to P.O.E. only.
6. Five spent fuel shipping casks @20,560 lb each, $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 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 a 0.8 plant factor.

TABLE A. 7
FUEL CYCLE COSTS FOR FULL ENRICHMENT BWR

Cost Item	2.3 yr Core	$/per Core	$/per year
1. Fuel Burnup Cost			
Kg U Initial	26.88		
Kg U Final	20.88		
Initial Enrichment	93%		
Final Enrichment	88%		
$/per KgU Initial	12,720		
$/per Kg U Final	12,000		
Initial U Value	$341,914		
Final U Value	$250,560		
Plutonium Credit	Nil		
Burnup Cost		$ 91,354	$ 39,719
2. Core Fabrications Cost		$310,000	$134,782
3. Reprocessing Costs			
Conversion to UF_6	$ 864		
Plant Costs	$ 52,800		
Reprocess Losses	$ 2,691		
Total Reprocessing Costs		$ 56,355	$ 24,502
4. Shipping Costs			
Fresh Fuel	$ 1,000		
Spent Fuel	$ 61,680		
Total Shipping Costs		$ 62,680	$ 27,252
Fuel Cycle Costs Without Use Charge		$520,389	$226,255
5. Fuel Use Charges			
In-Reactor Core	$ 32,246		
Spare Core	$ 37,295		
10% Spares of 1 Core	$ 3,729		
Total Use Charges		$ 73,270	$ 31,857
TOTAL FUEL CYCLE COSTS		$593,659	$258,112

7.0 REFERENCES

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11. "Civilian Power Reactor Program: Status Report on Boiling Water Reactor Technology as of 1959," Part III, U. S. Atomic Energy Commission, TID-8518(5), 1959.

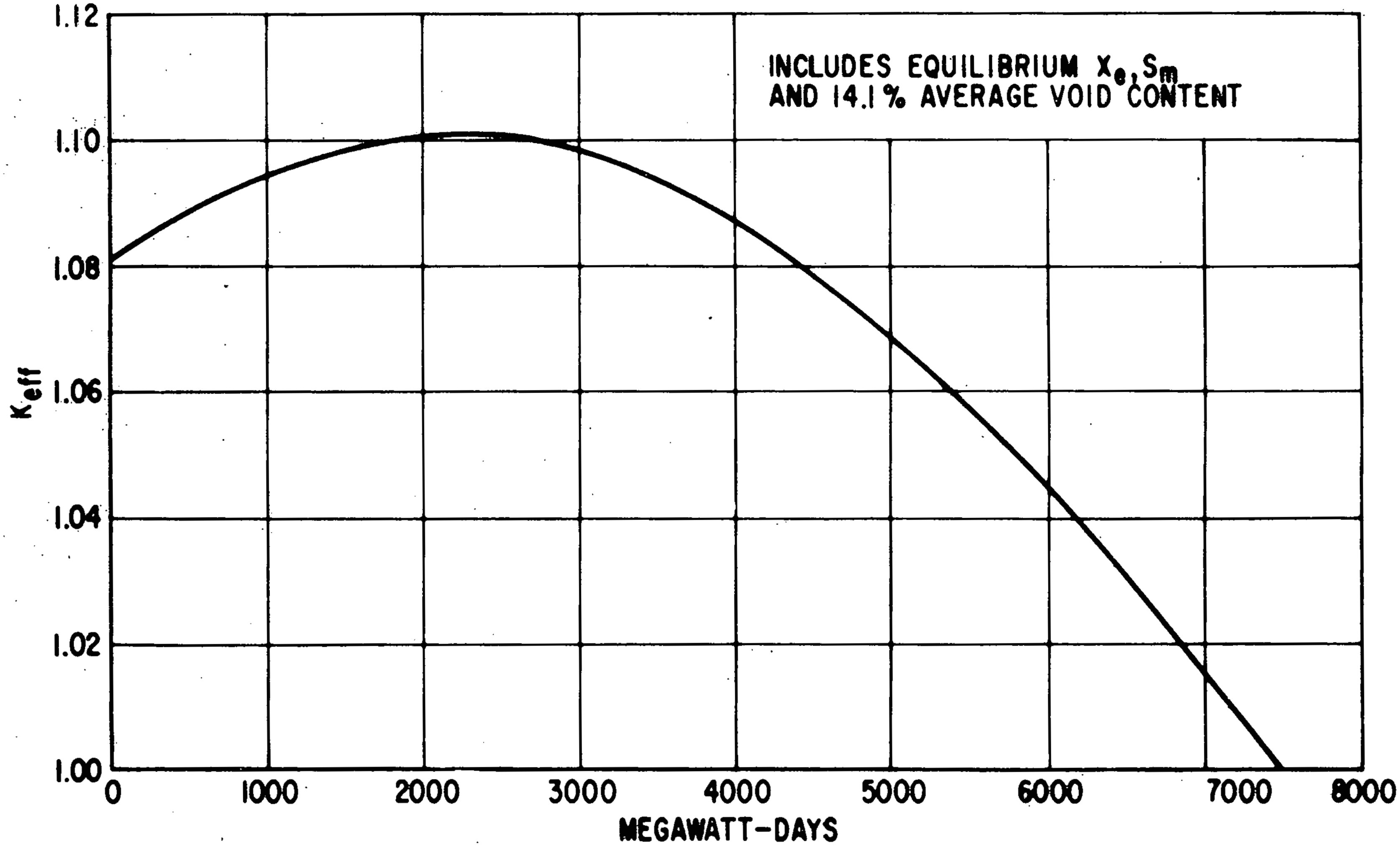


FIGURE A.1 VARIATION OF Keff WITH UNIFORM BURNUP

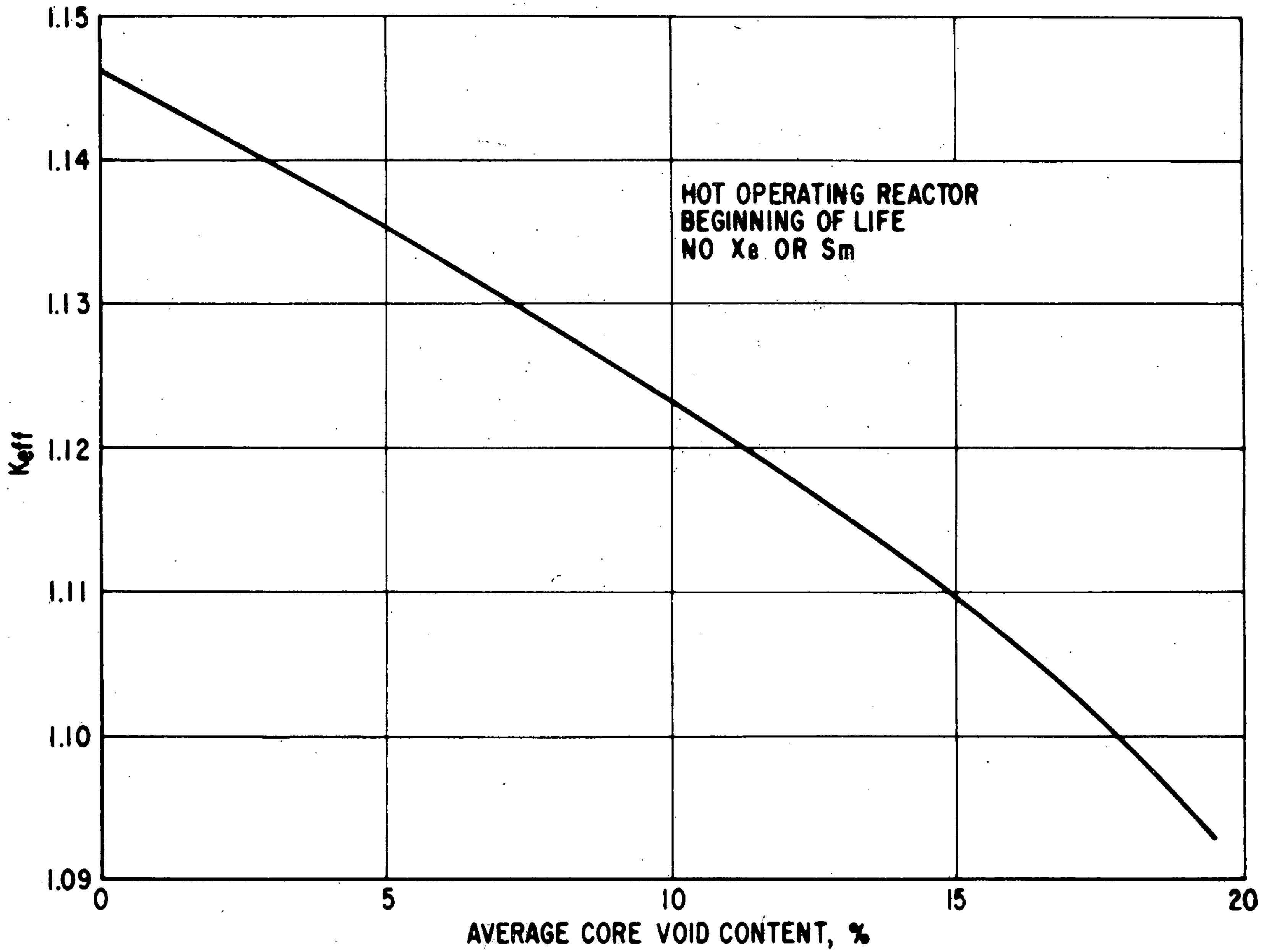


FIGURE A.2 EFFECT OF AVERAGE CORE VOID CONTENT ON K_{eff}

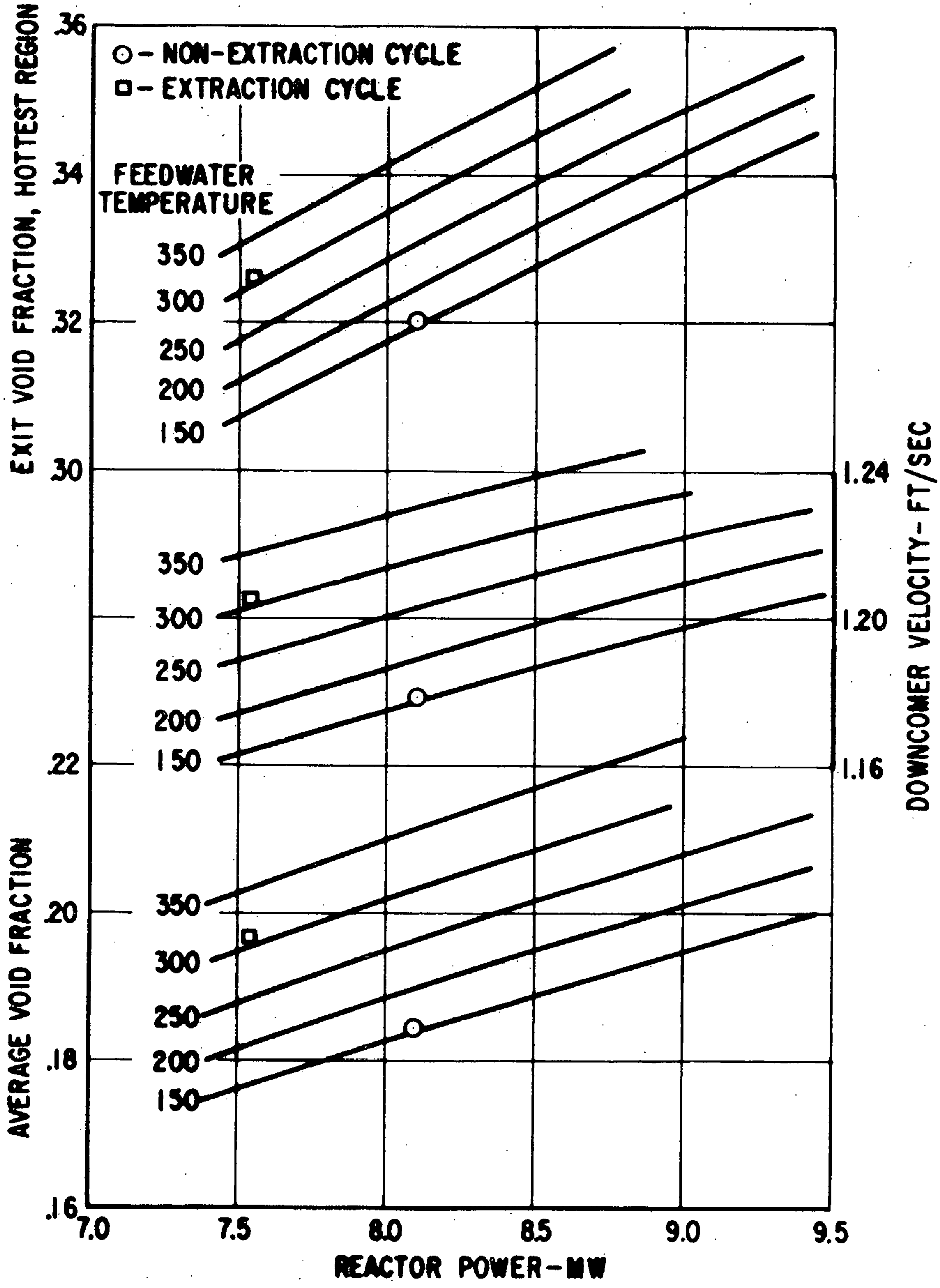
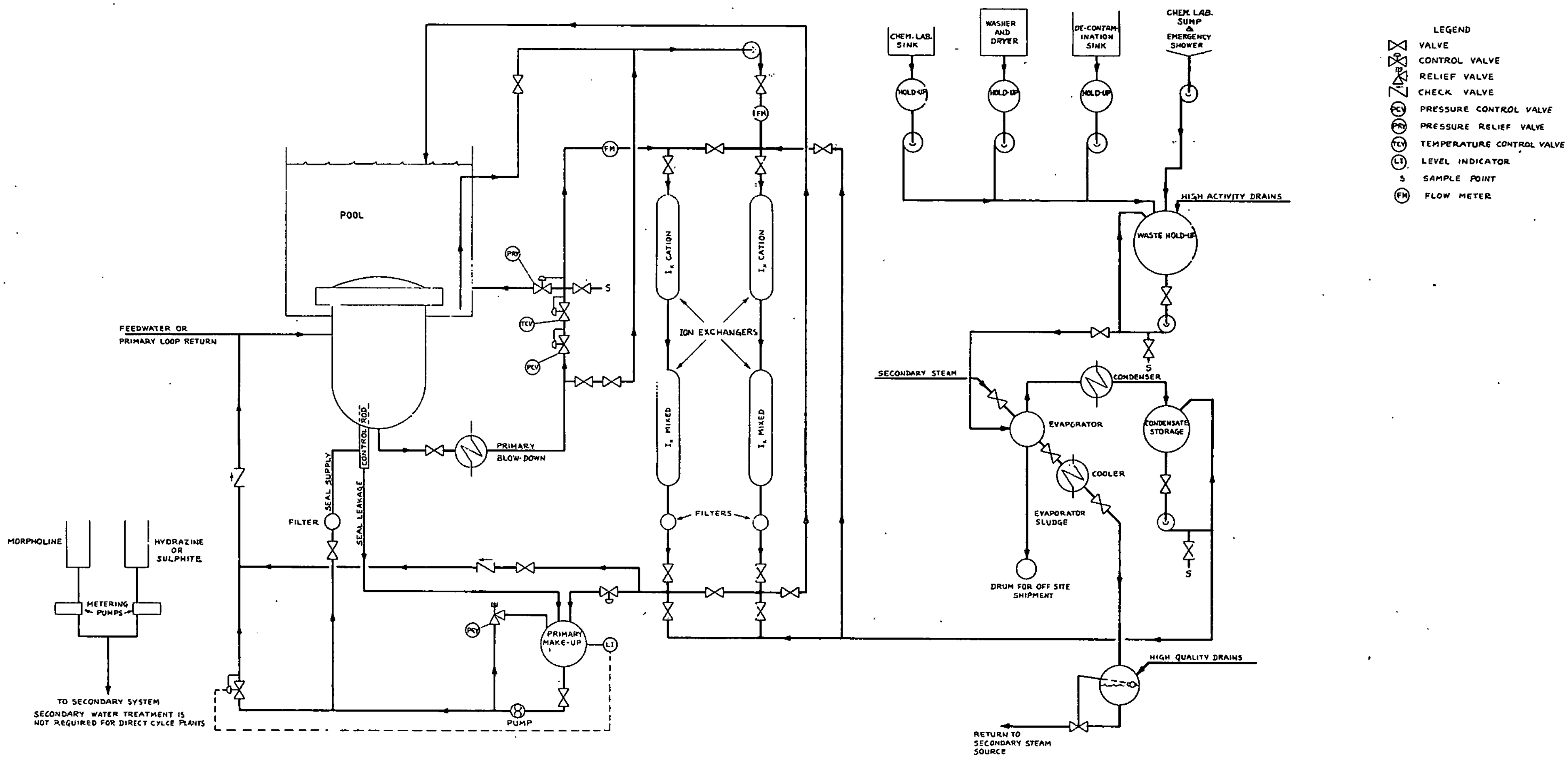


FIGURE A.3 EFFECTS OF REACTOR POWER AND FEEDWATER TEMPERATURE

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		H.F.	



PRELIMINARY

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPT.
SCHENECTADY, N. Y., U. S. A.

SCALE	—
REF.	
MATERIAL SPEC.	
DR.	B. Parker 12-1-61
TR.	
CHK.	
APPR.	Feb 23 '62
APPR.	Feb 22 '62
MET.	

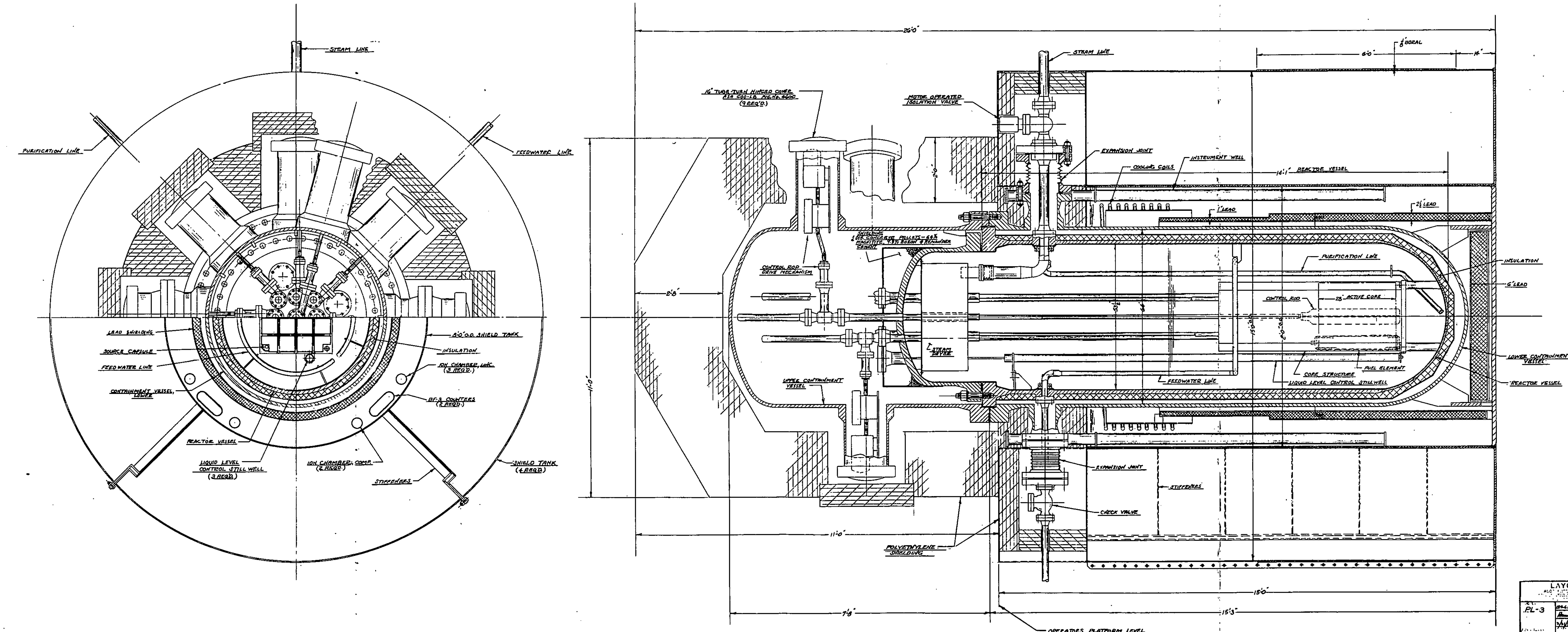
UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES. TOLERANCES ON FINISHED FRACTIONAL DIMENSIONS TO BE ± ________

√ FINISH AS INDICATED IN MICROINCHES.
∕1 MACHINE FINISH - ROUGH
∕10 FLAME CUT OR SAW

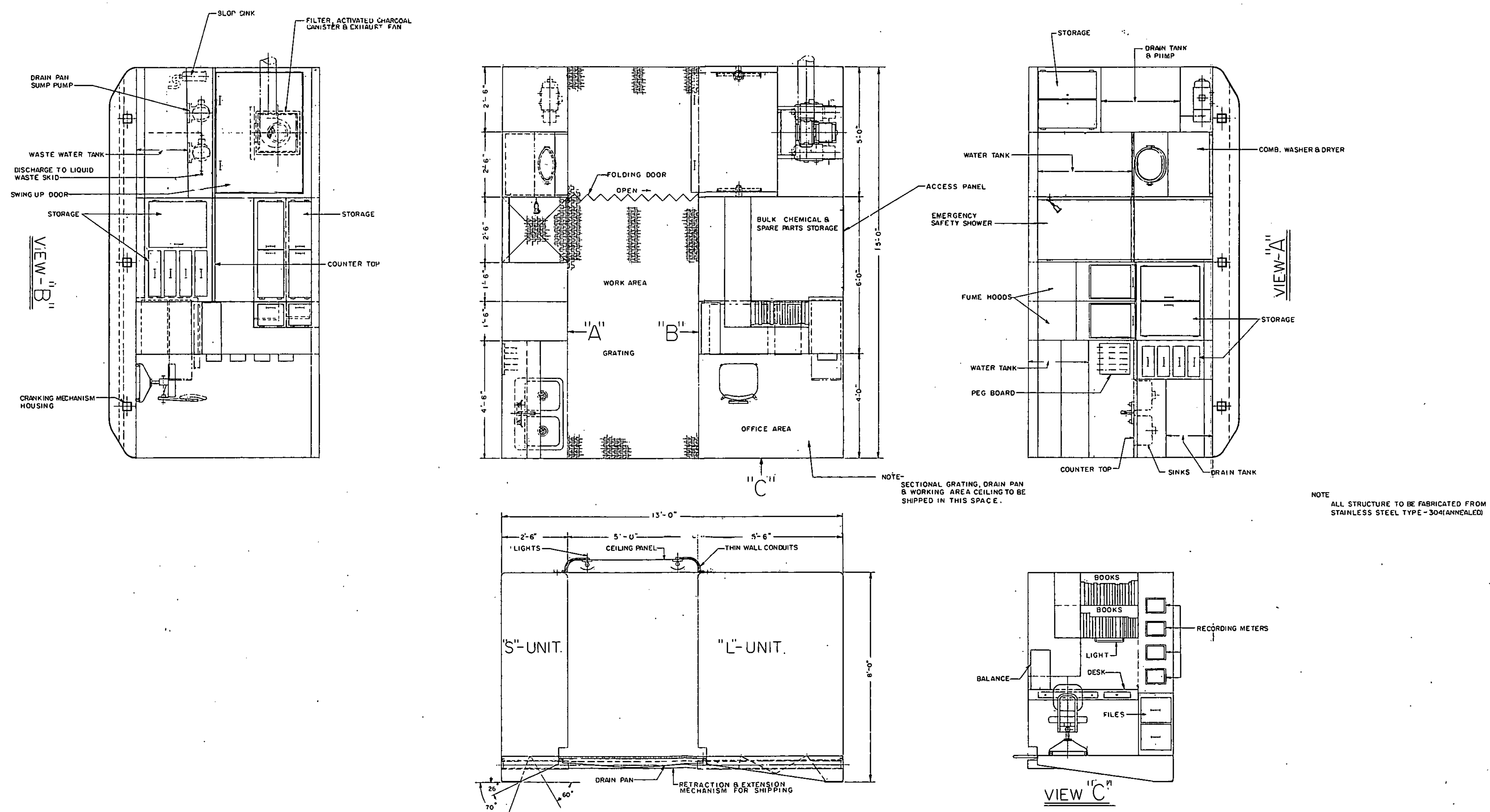
NAME WATER TREATMENT AND PROCESSING SCHEMATIC INTEGRATED CONCEPT FOR BWR OR PWR SYSTEMS

PART NO. AEL-710

PART NO. AEL-710



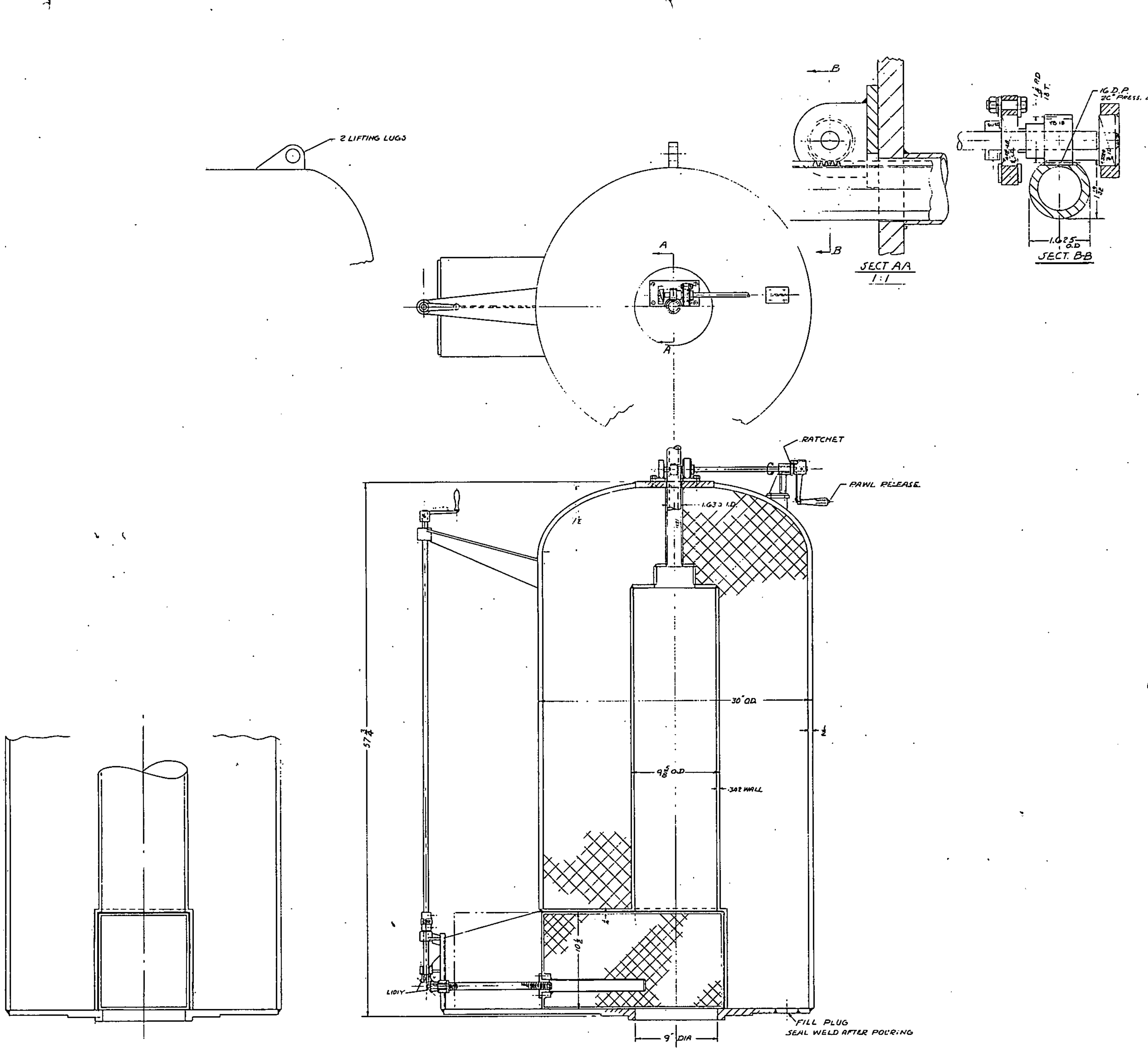
LAYOUT
PL-3
TITLE: BWR COMPLEX (DIRECT CYCLE) CONCEPT-[illegible]
LAYOUT NO.: AEL-722



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. SCAUTERBURY	
AUTHORITY-	DATE- 2-27-62	APPROVED- G.E.H.	
	DATE-	APPROVED-	
TITLE- CHEMICAL LABORATORY SKID			
LAYOUT NO.- AEL-729			



WEIGHT OF STEEL PARTS	1844 LBS
WEIGHT OF LEAD SHIELDING	14470 "
TOTAL WEIGHT	16314 LBS

PRELIMINARY

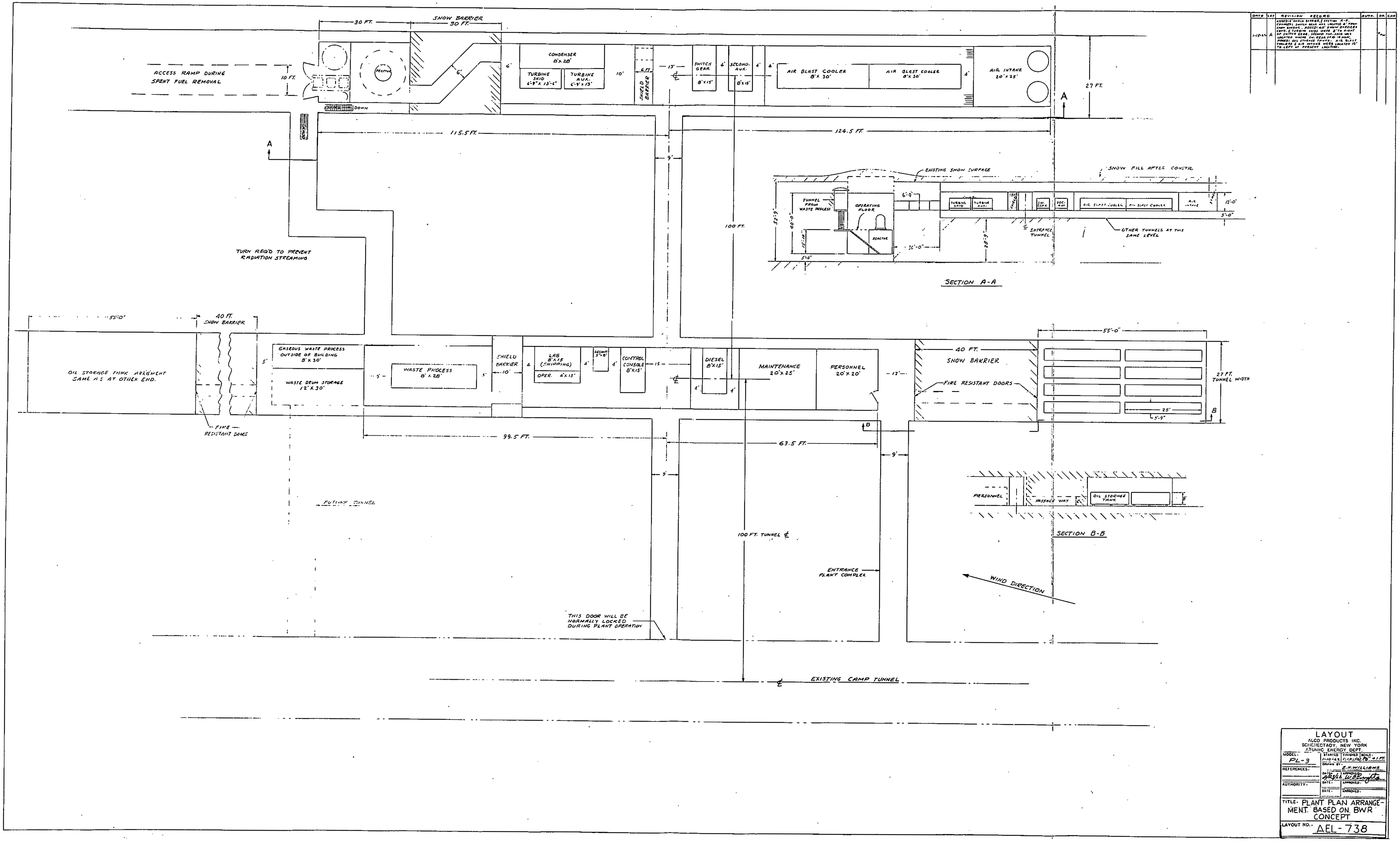
LAYOUT

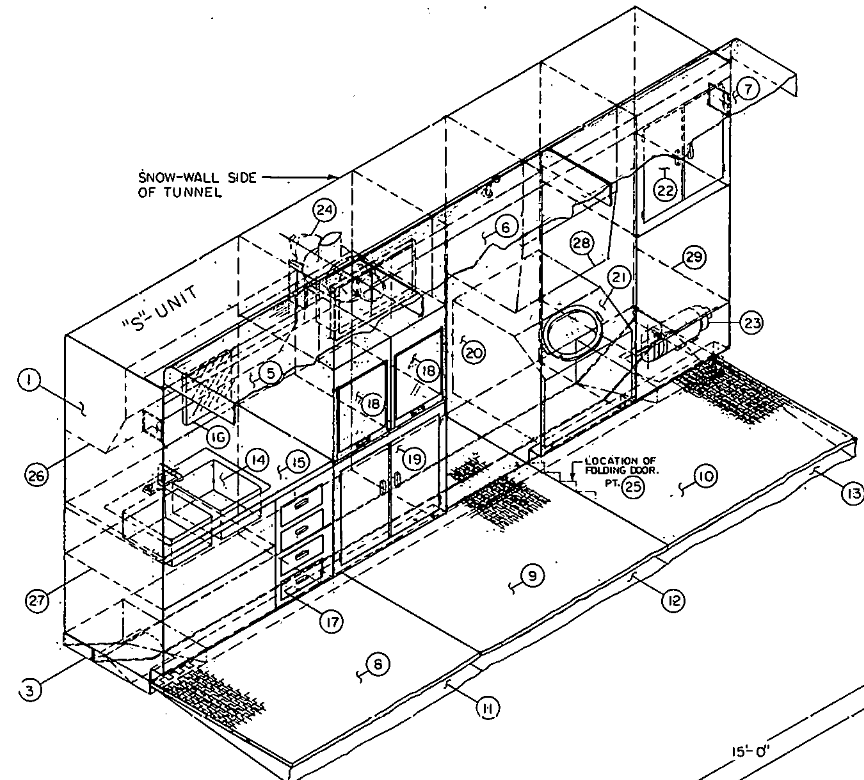
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

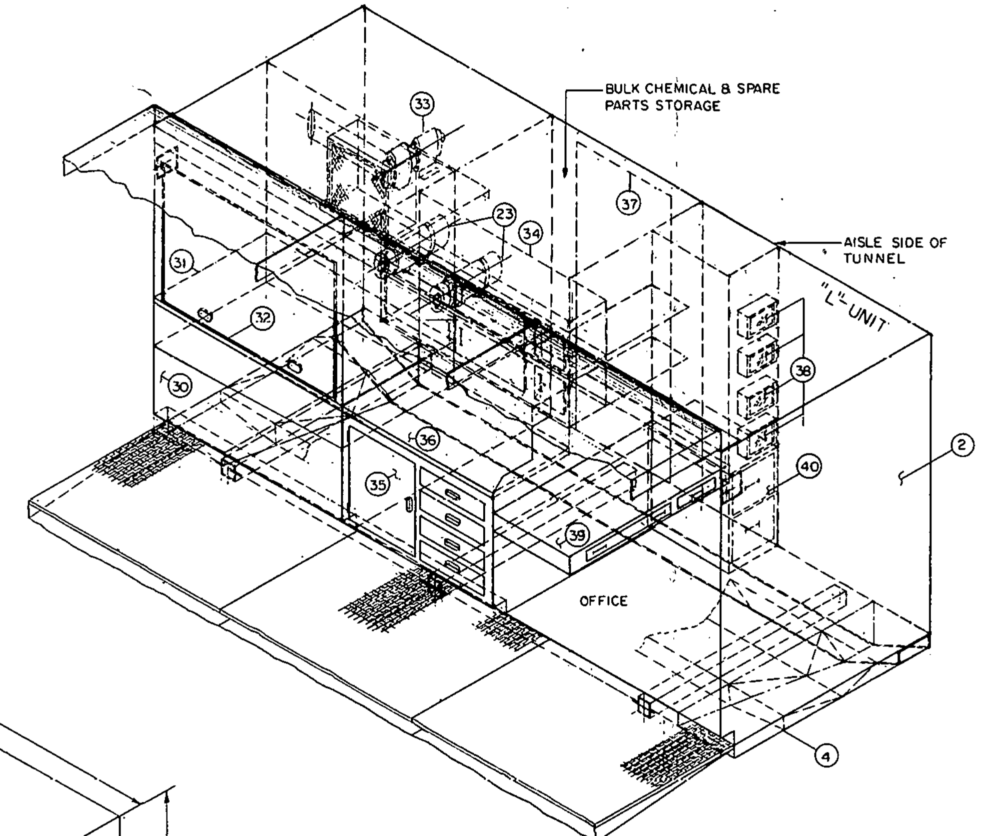
MODEL: PL-3

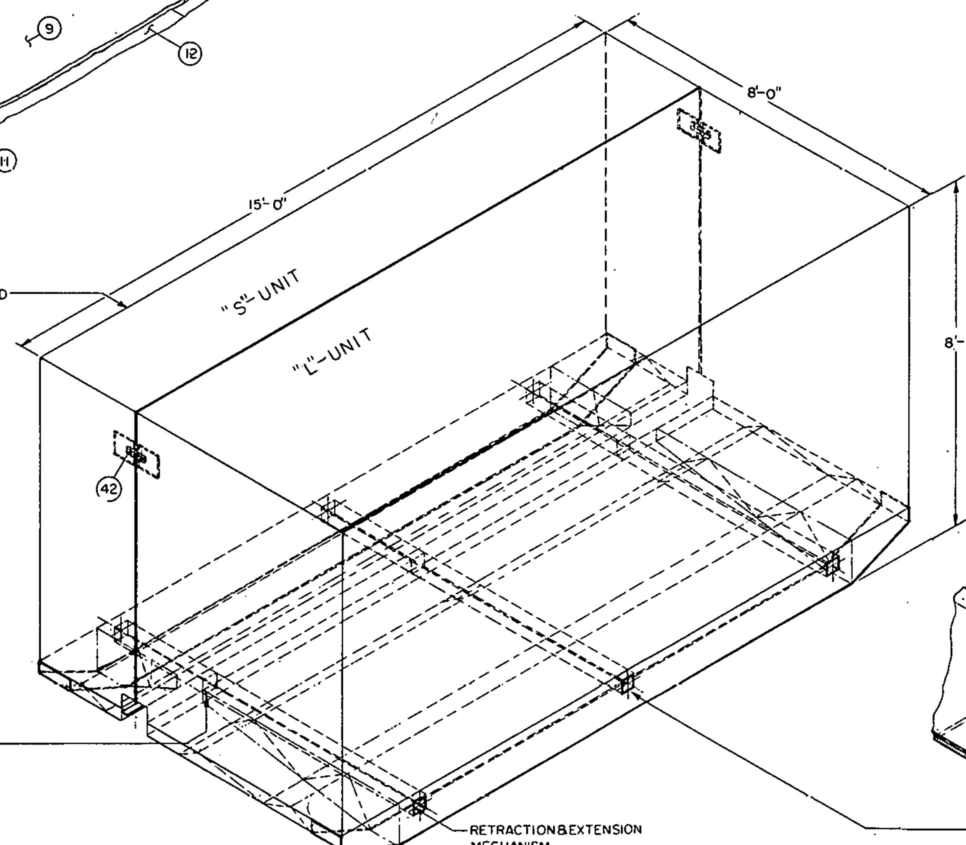
TITLE: FUEL TRANSFER FLASK
PWR CONCEPT
MODIFIED SM-2 PLANT

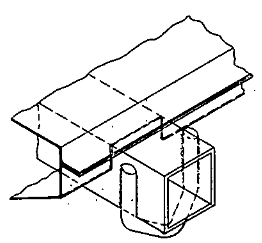
LAYOUT NO.: AEL 731


SNOW BARRIER
30 FT.
ACCESS RAMP DURING
SPENT FUEL REMOVAL
10 FT.
REACTOR
DOWN
CONDENSER
8' x 28'
TURBINE SKID
6'-9" x 13'-6"
TURBINE AUX.
6'-9" x 13'
SHIELD BARRIER
SWITCH GEAR
8' x 15'
SECOND. AUX.
8' x 15'
AIR BLAST COOLER
8' x 30'
AIR INTAKE
20' x 25'
27 FT.
115.5 FT.
124.5 FT.
9'
100 FT.
TURN REQ'D TO PREVENT
RADIATION STREAMING
EXISTING SNOW SURFACE
SNOW FILL AFTER CONSTR.
TUNNEL FROM WASTE PROCESS
OPERATING FLOOR
REACTOR
ENTRANCE TUNNEL
OTHER TUNNELS AT THIS
SAME LEVEL
SECTION A-A
55'-0"
40 FT.
SNOW BARRIER
OIL STORAGE TANK ARRGMENT
SAME AS AT OTHER END.
GASEOUS WASTE PROCESS
OUTSIDE OF BUILDING
8' x 30'
WASTE DRUM STORAGE
12' x 30'
WASTE PROCESS
8' x 28'
SHIELD BARRIER
LAB
8' x 15'
(SHIPPING)
OPER. 8' x 15'
DECONT.
5' x 8'
CONTROL CONSOLE
8' x 15'
DIESEL
8' x 15'
MAINTENANCE
20' x 25'
PERSONNEL
20' x 20'
SNOW BARRIER
FIRE RESISTANT DOORS
27 FT.
TUNNEL WIDTH
FIRE
RESISTANT DOORS
99.5 FT.
63.5 FT.
FUTURE TUNNEL
PERSONNEL
PASSAGE WAY
OIL STORAGE TANK
SECTION B-B
100 FT. TUNNEL ℄
ENTRANCE
PLANT COMPLEX
WIND DIRECTION
THIS DOOR WILL BE
NORMALLY LOCKED
DURING PLANT OPERATION
EXISTING CAMP TUNNEL
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.
MODEL- PL-3
STARTED 1-10-62
FINISHED 1-10-62
SCALE- 1/8" = 1 FT.
DRAWN BY- E.H. WILLIAMS
REFERENCES-
AUTHORITY-
DATE-
APPROVED-
TITLE- PLANT PLAN ARRANGEMENT BASED ON BWR CONCEPT
LAYOUT NO.- AEL-738



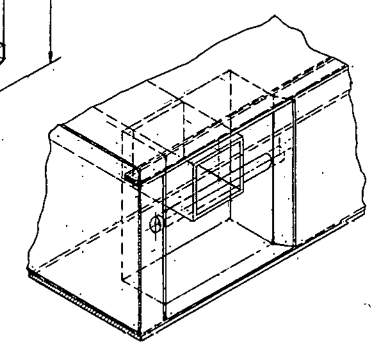






TIE DOWN FITTING (2-PLACES) "S"-UNIT
SCALE- 1/4 SIZE

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-758	LOWER STRUCTURE "L"-UNIT
3	1	AEL-758	LOWER STRUCTURE "S"-UNIT
2	1		UPPER STRUCTURE "L"-UNIT
1	1		UPPER STRUCTURE "S"-UNIT

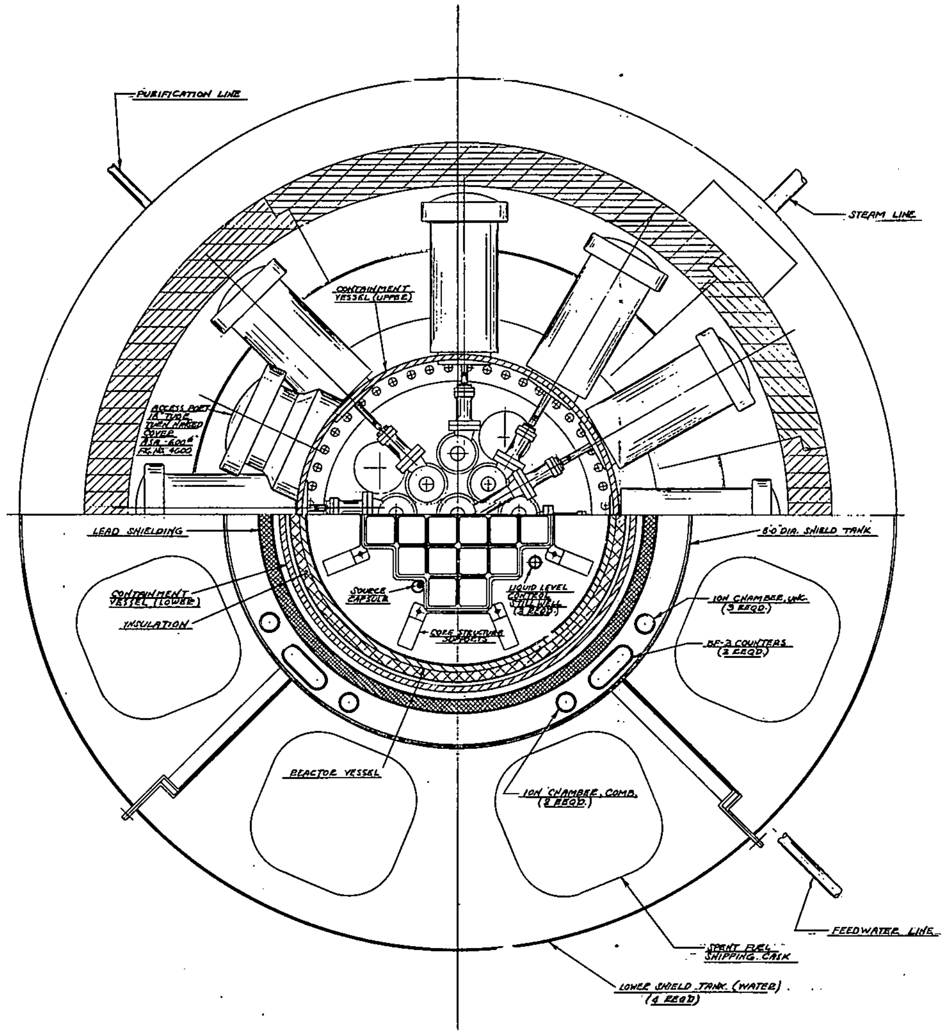
PRELIMINARY

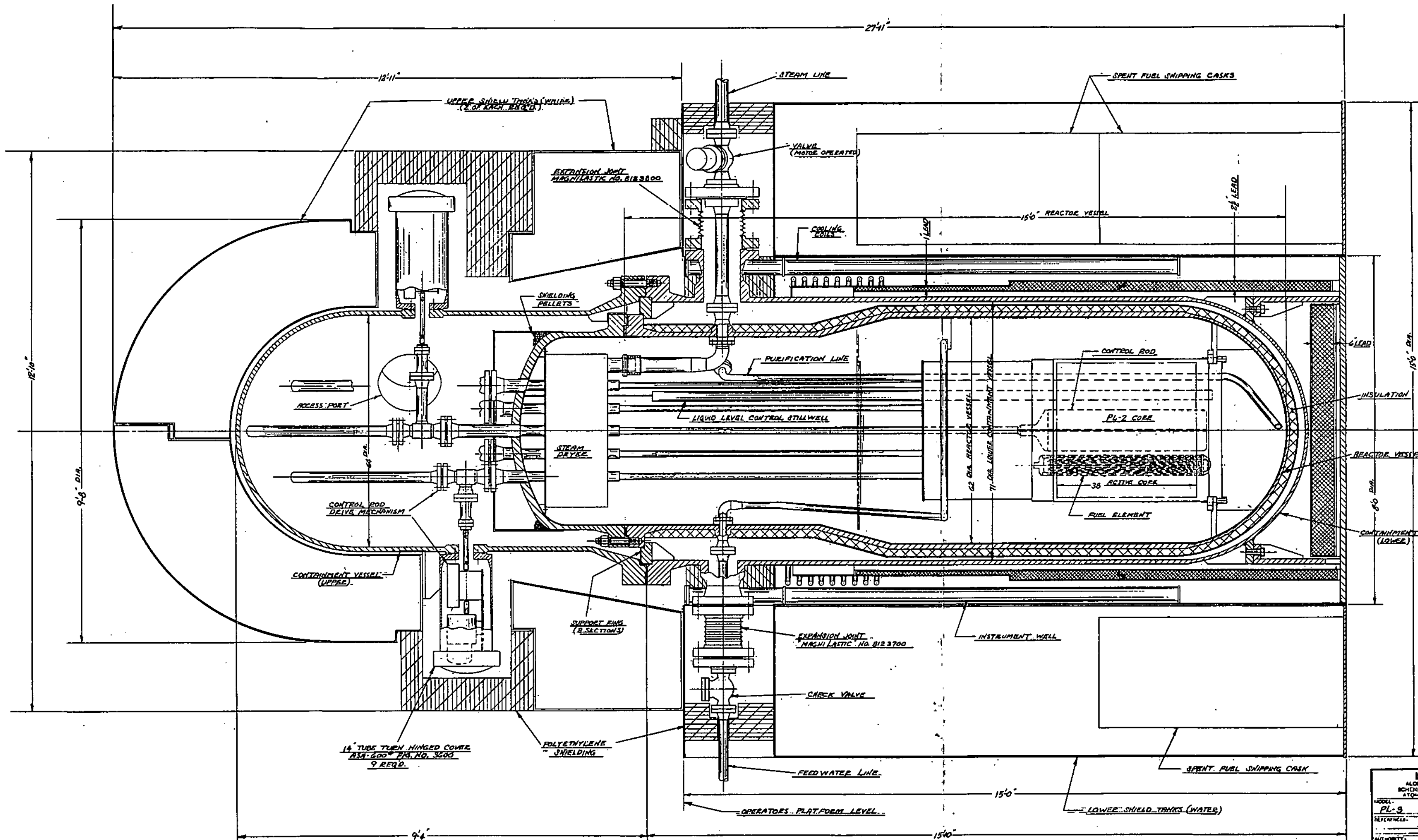
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL- PL-3	STARTED 1-3-62	FINISHED 1-18 62	SCALE- 1/4" = 1-FT
	DRAWN BY G. L. Boyd.		
REFERENCES- AEL-729	DATE- 1-19-62	APPROVED- W. Scantlebury	
AUTHORITY-	DATE- 2-27-62	APPROVED- G.E.B.	
	DATE-	APPROVED-	

TITLE-
CHEMICAL LABORATORY SKID. (ISOMETRIC)

LAYOUT NO.- AEL-739



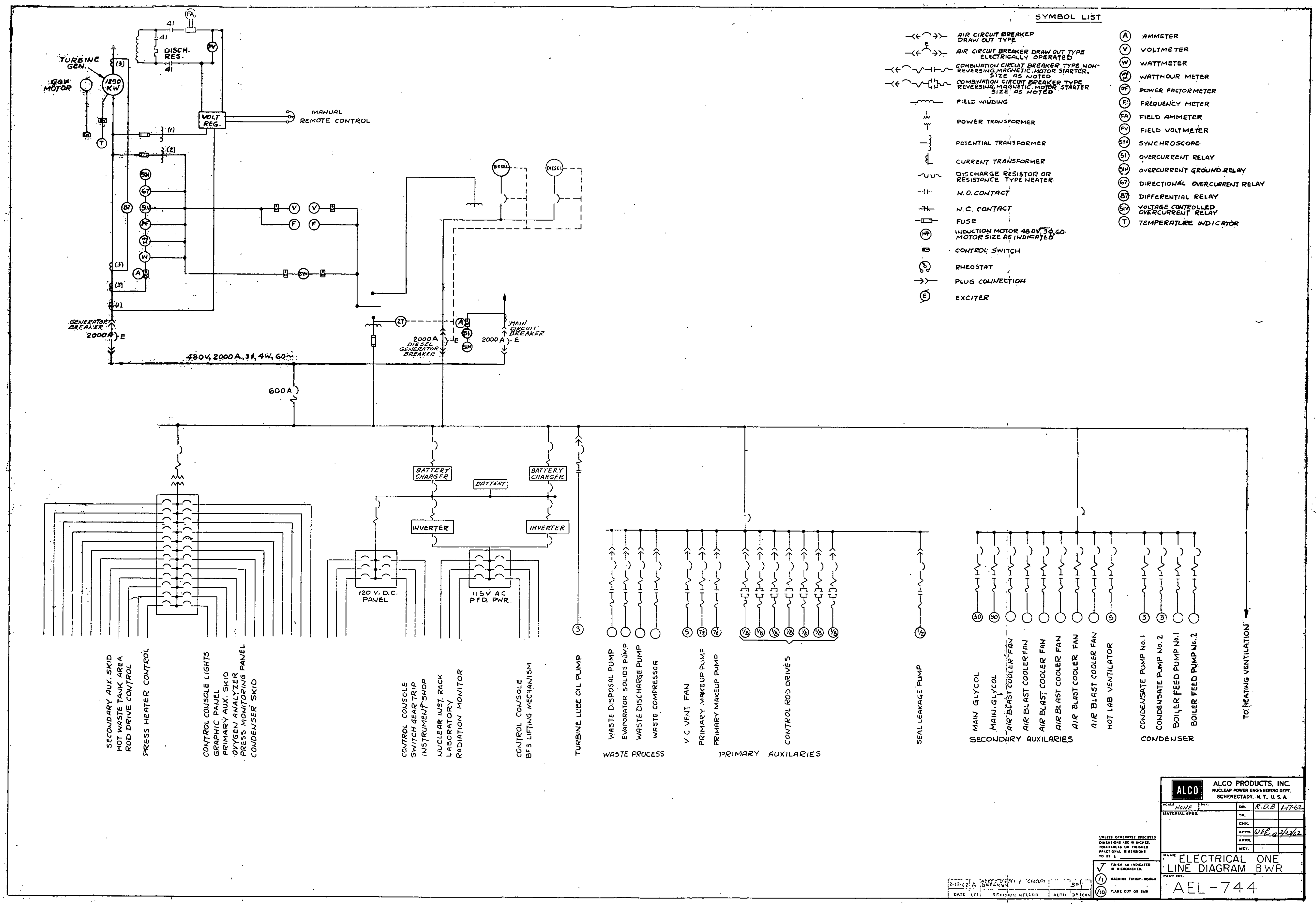


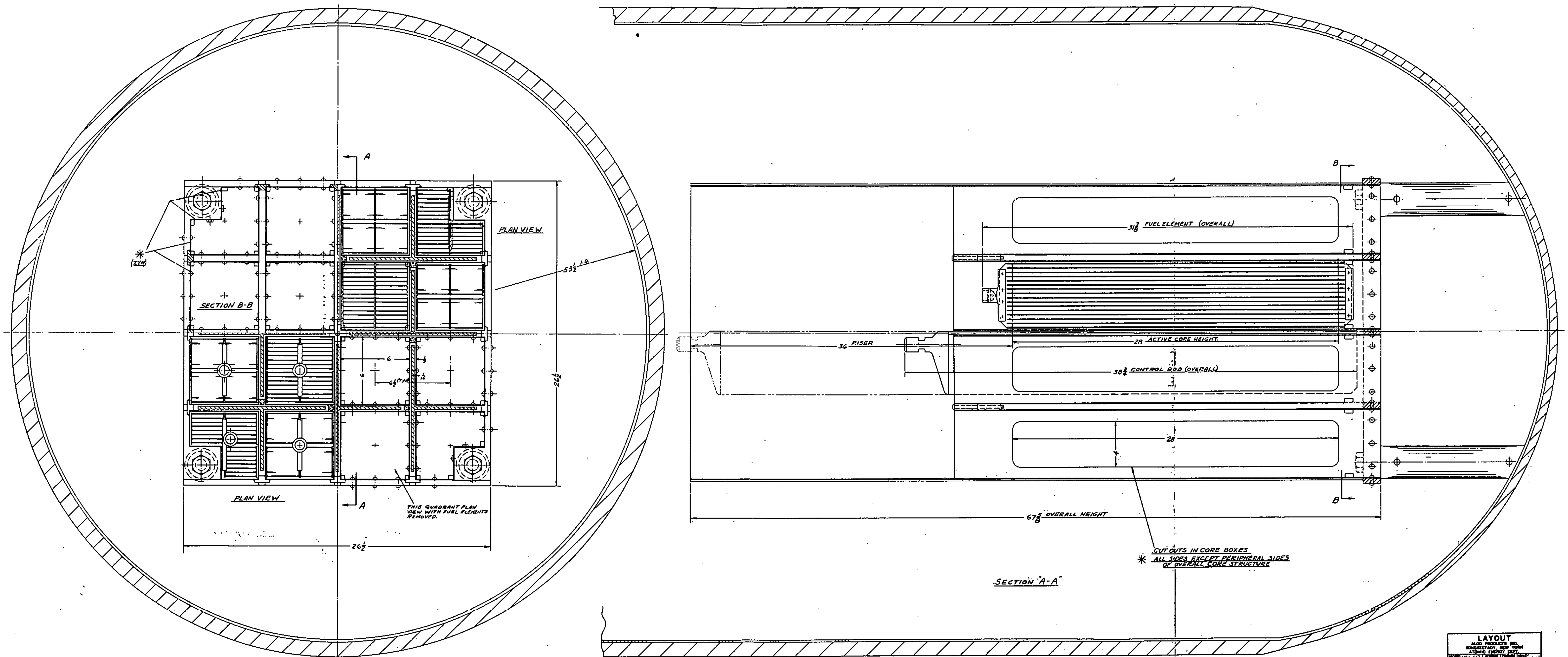
LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL - PL-3
REFERENCES -
AUTHORITY -

TITLE - BWR COMPLEX

LAYOUT NO. - AEL-280


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
GEN MOTOR
DISCH. RES.
VOLT REG.
MANUAL
REMOTE CONTROL
DIESEL
DIESEL
GENERATOR BREAKER 2000 A
480V, 2000 A, 3Φ, 4W, 60~
2000 A DIESEL GENERATOR BREAKER
2000 A
MAIN CIRCUIT BREAKER
600 A
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
CONTROL ROD DRIVES
PRIMARY AUXILARIES
SEAL LEAKAGE PUMP
MAIN GLYCOL
MAIN GLYCOL
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.
SCALE NONE
DR. R.D.B 1-17-62
ELECTRICAL ONE LINE DIAGRAM BWR
AEL-744

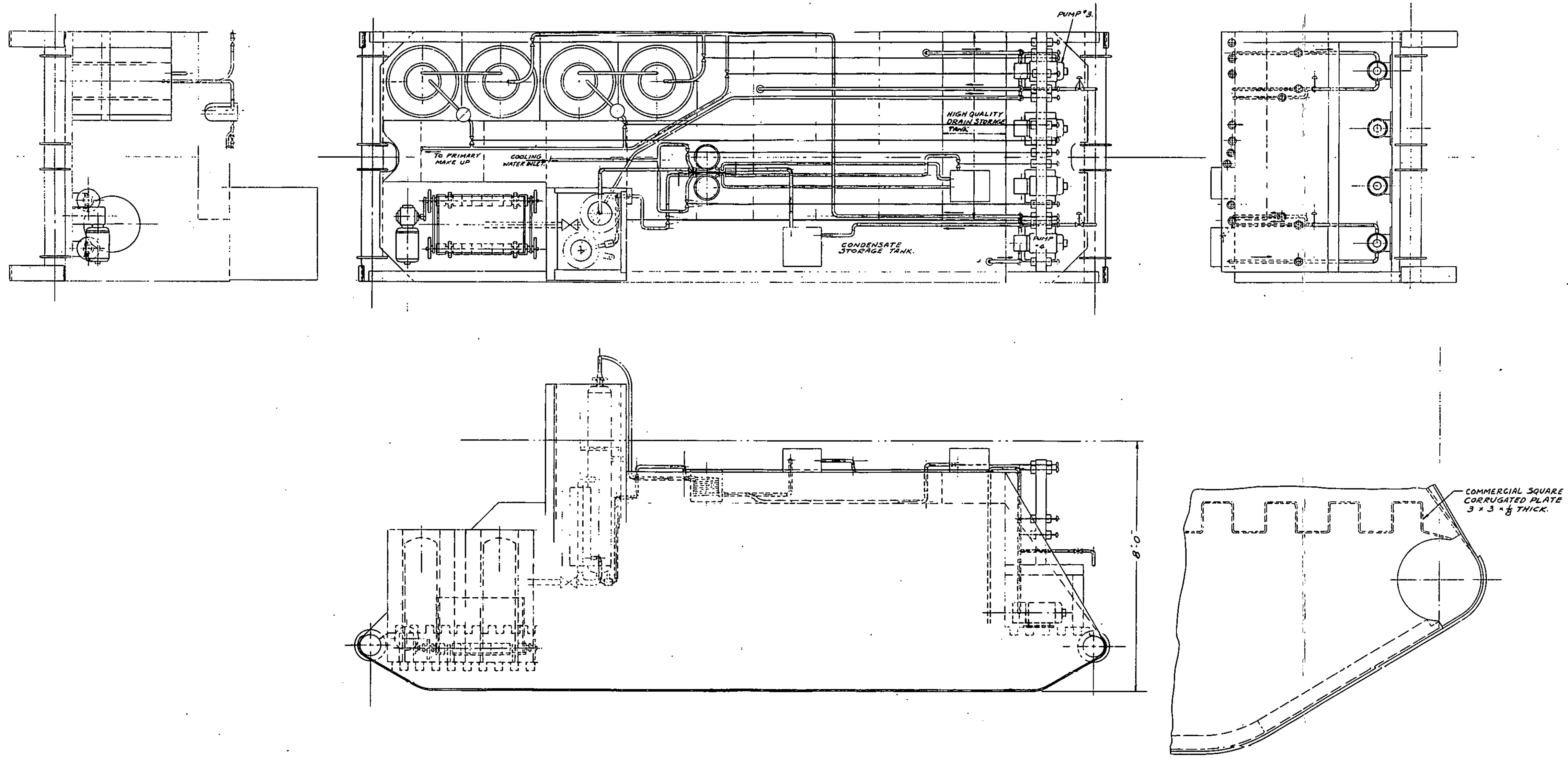


LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL: BWR-PL-3

TITLE: CORE STRUCTURE CONCEPT B-9

LAYOUT NO.: AEL-746

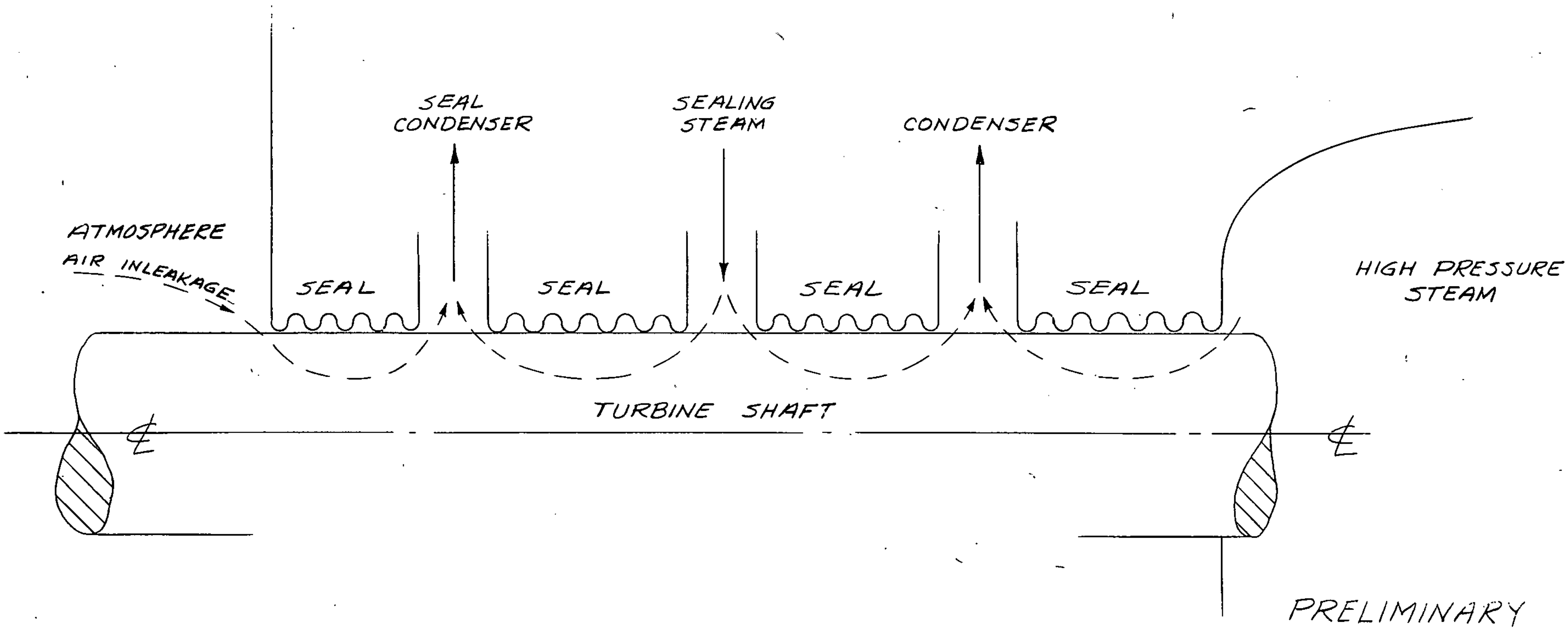


LAYOUT
ALCO PRODUCTS INC.
SCHENECTADY, NEW YORK
ATOMIC ENERGY DEPT.

MODEL- PL-3	STARTED 1-18-62	FINISHED	SCALE- 3/16"-1'-0"
REFERENCES-	DRAWN BY- MUNAFO		
	DATE- 3/23/62	APPROVED- W.B. Enright	
AUTHORITY-	DATE-	APPROVED-	
	DATE-	APPROVED-	

TITLE- WASTE PROCESSING SKID

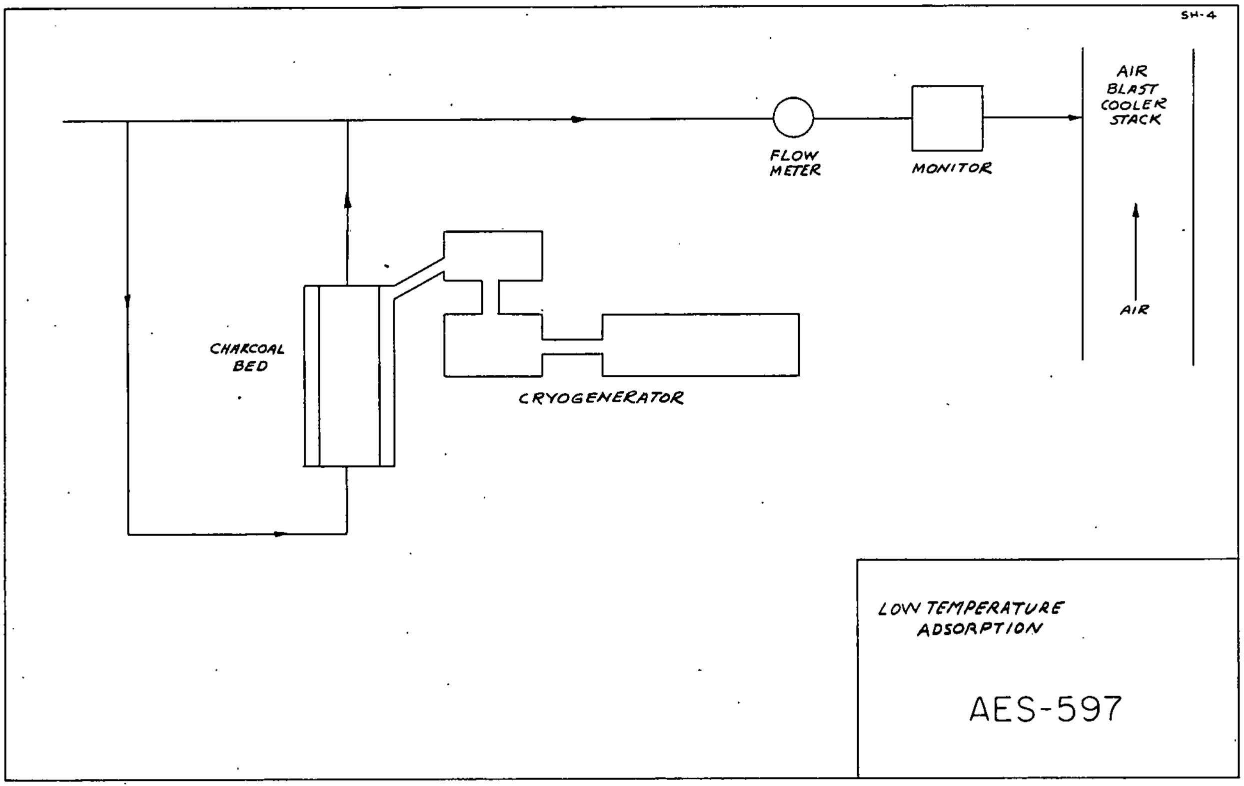
LAYOUT NO.- AEL-747

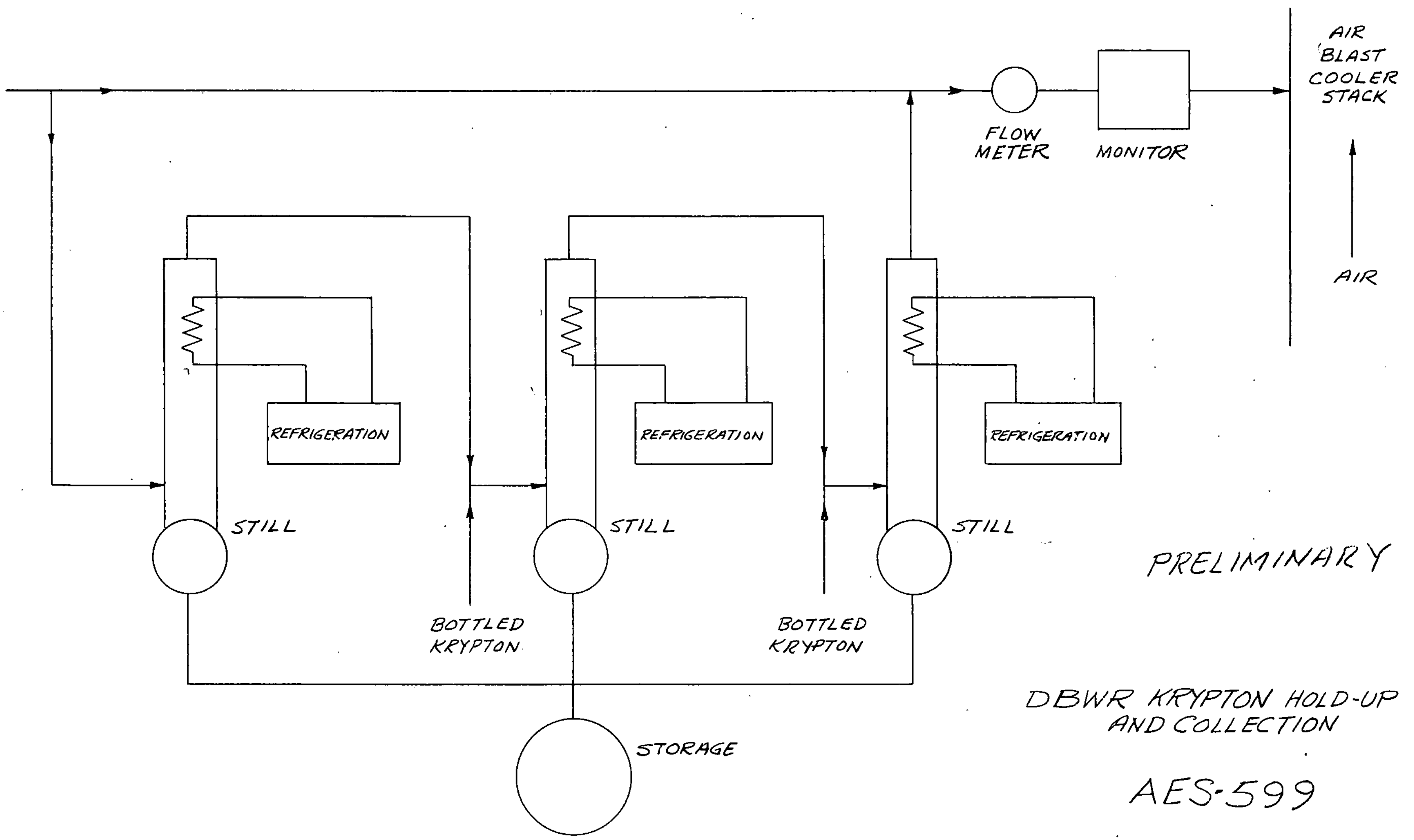


PRELIMINARY

DBWR TURBINE SHAFT
SEAL ARRANGEMENT

AES-594

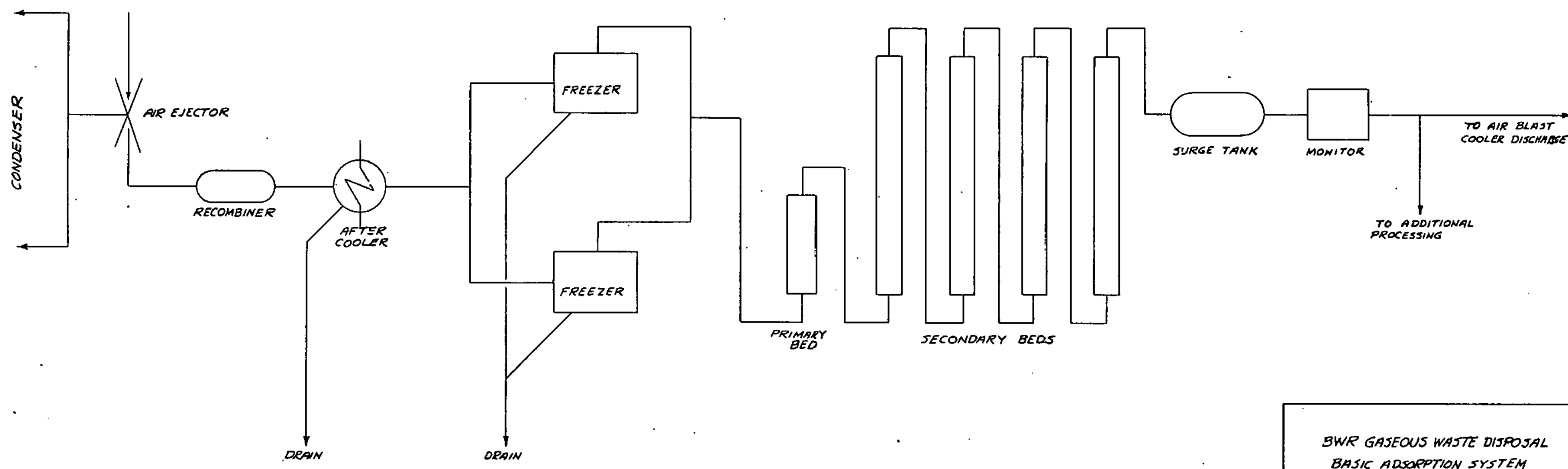

SH-4
AIR
BLAST
COOLER
STACK
FLOW
METER
MONITOR
AIR
CHARCOAL
BED
CRYOGENERATOR
LOW TEMPERATURE
ADSORPTION
AES-597



PRELIMINARY

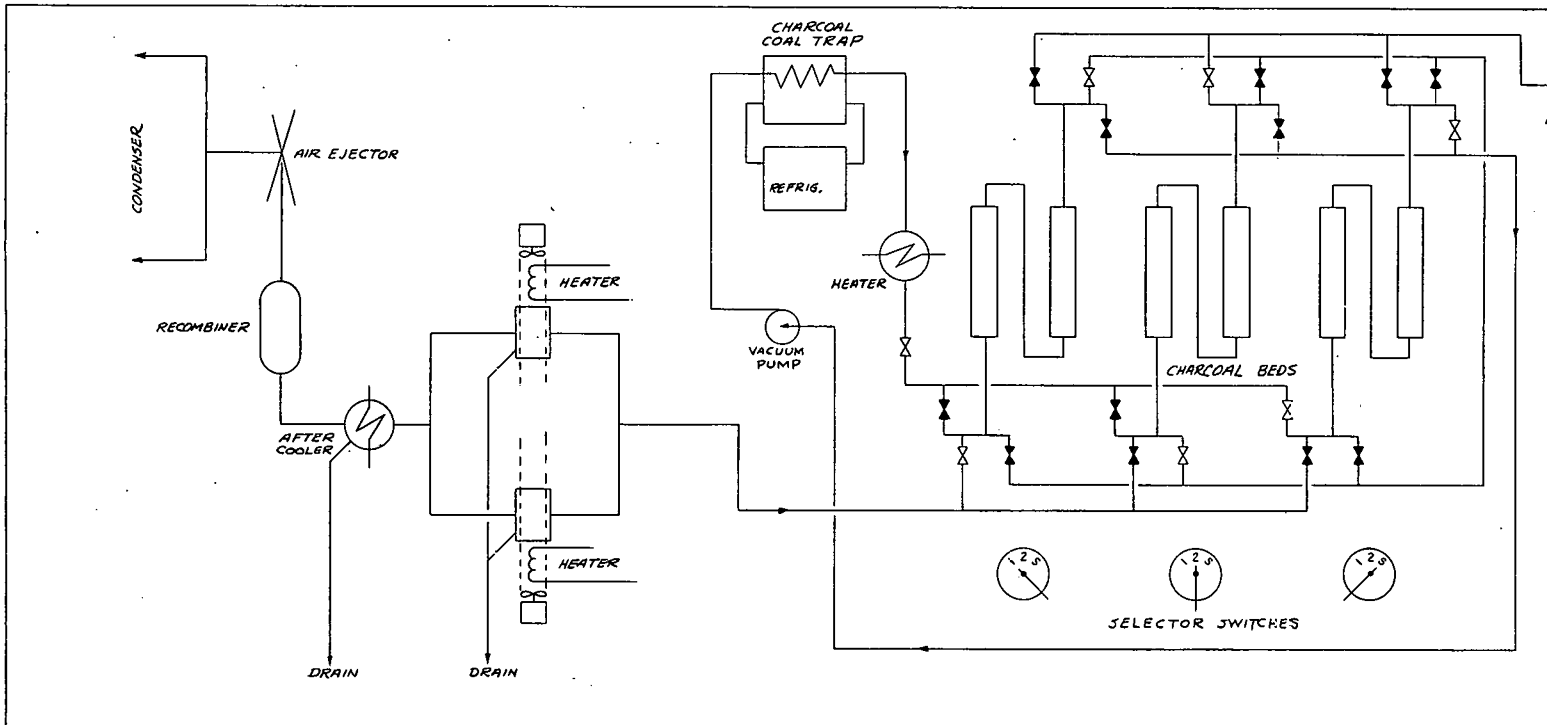
DBWR KRYPTON HOLD-UP AND COLLECTION

AES-599



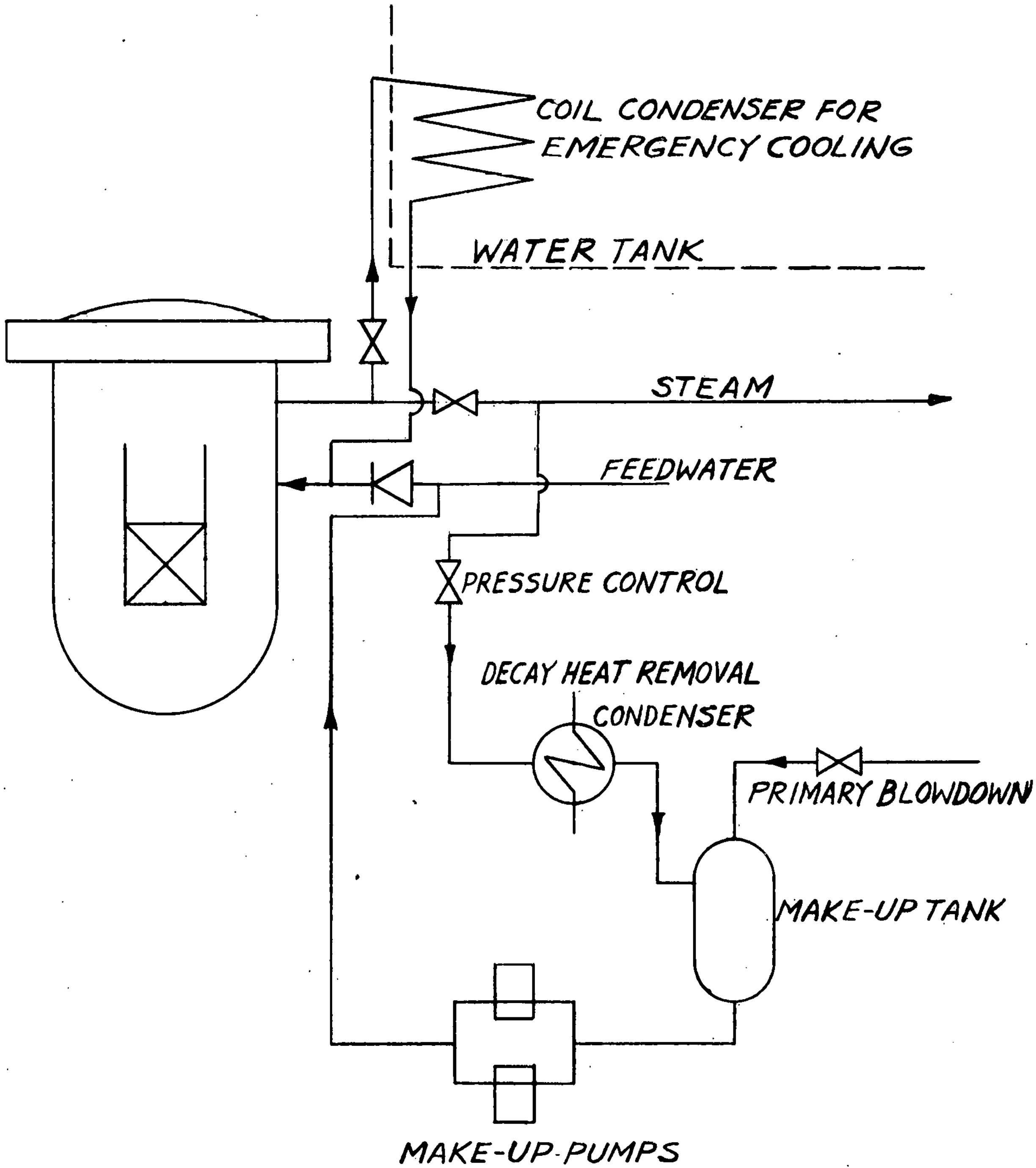
BWR GASEOUS WASTE DISPOSAL
BASIC ADSORPTION SYSTEM

AES-600



BWR CYCLING ADSORPTION BED GASEOUS WASTE DISPOSAL

AES-601

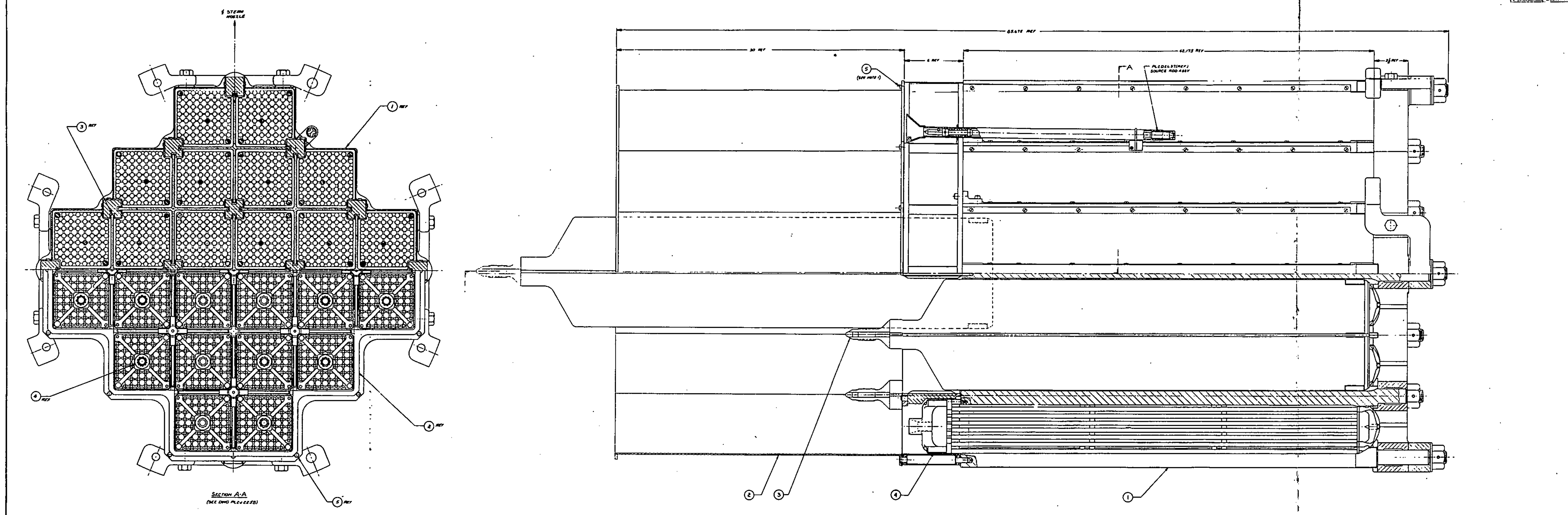


AES-608

DECAY HEAT REMOVAL & EMERGENCY
COOLING SYSTEMS

BILL OF MATERIALS

ITEM	PART NUMBER	NAME	QTY	MATERIAL	REMARKS
	PL2J2258	CORE ASSEMBLY (PL-2 CORE)			
1	PL2J2257	CORE SUPPORT ASSEMBLY	1		
2	PL2J2255	SHROUD, UPPER	1		
3	PL2J225G	CONTROL ROD BLADE	3		
4	PL2J2250	FUEL ASSEMBLY	14		
5	[illegible]	BOLT, HEX, HD.	12	[illegible]	[illegible]



1 TORQUE ITEM 5 TO 30 INCH LBS AT ASSEMBLY AND LOCK WITH 1/4 LONG TACK WELD.

PL2J2258

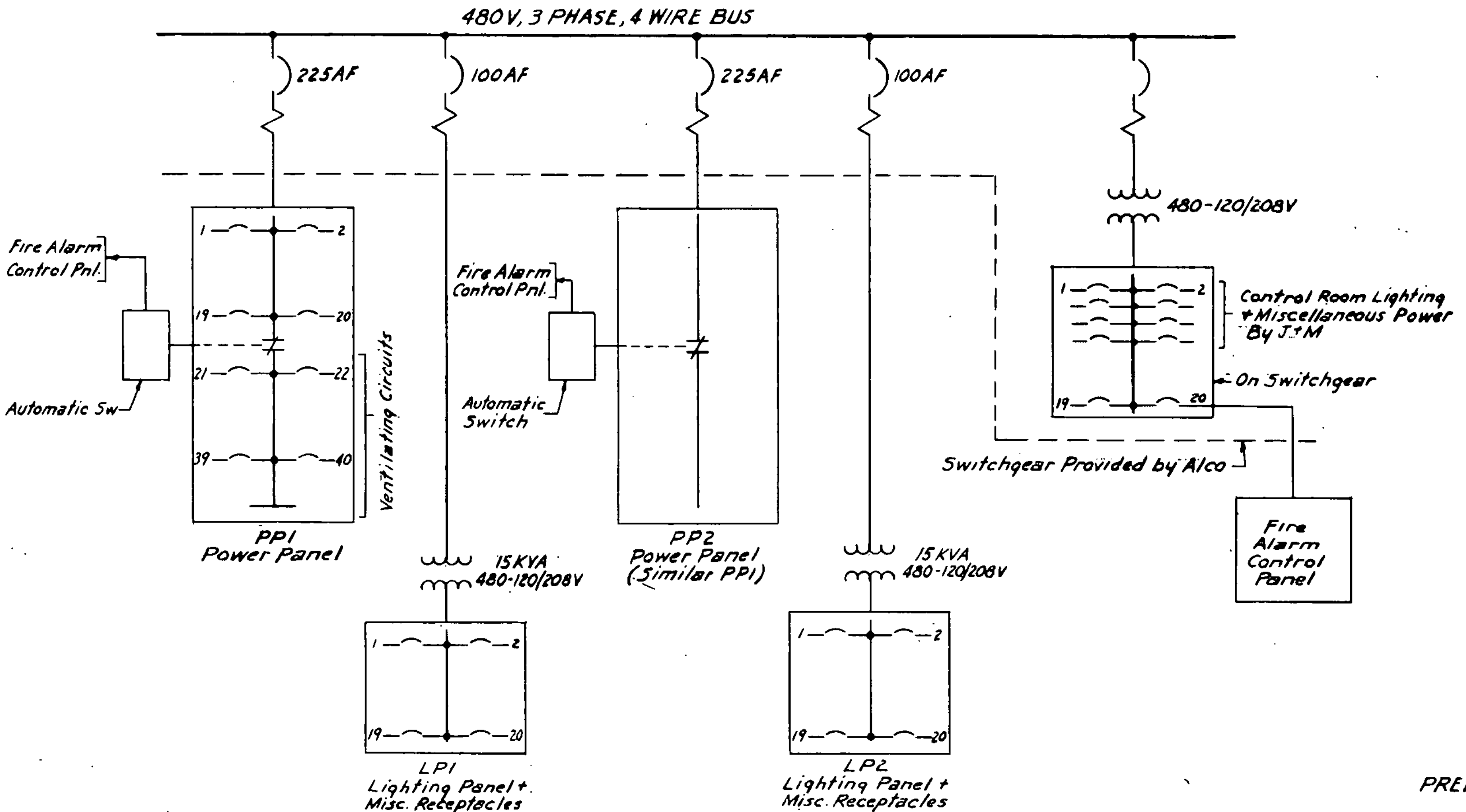
APPROVED BY

PREPARED UNDER AEC CONTRACT AT(30-3)-327

COMBUSTION ENGINEERING, INC.
NUCLEAR DIVISION

CORE ASSEMBLY
PL-2 CORE

PL2J2258



PRELIMINARY

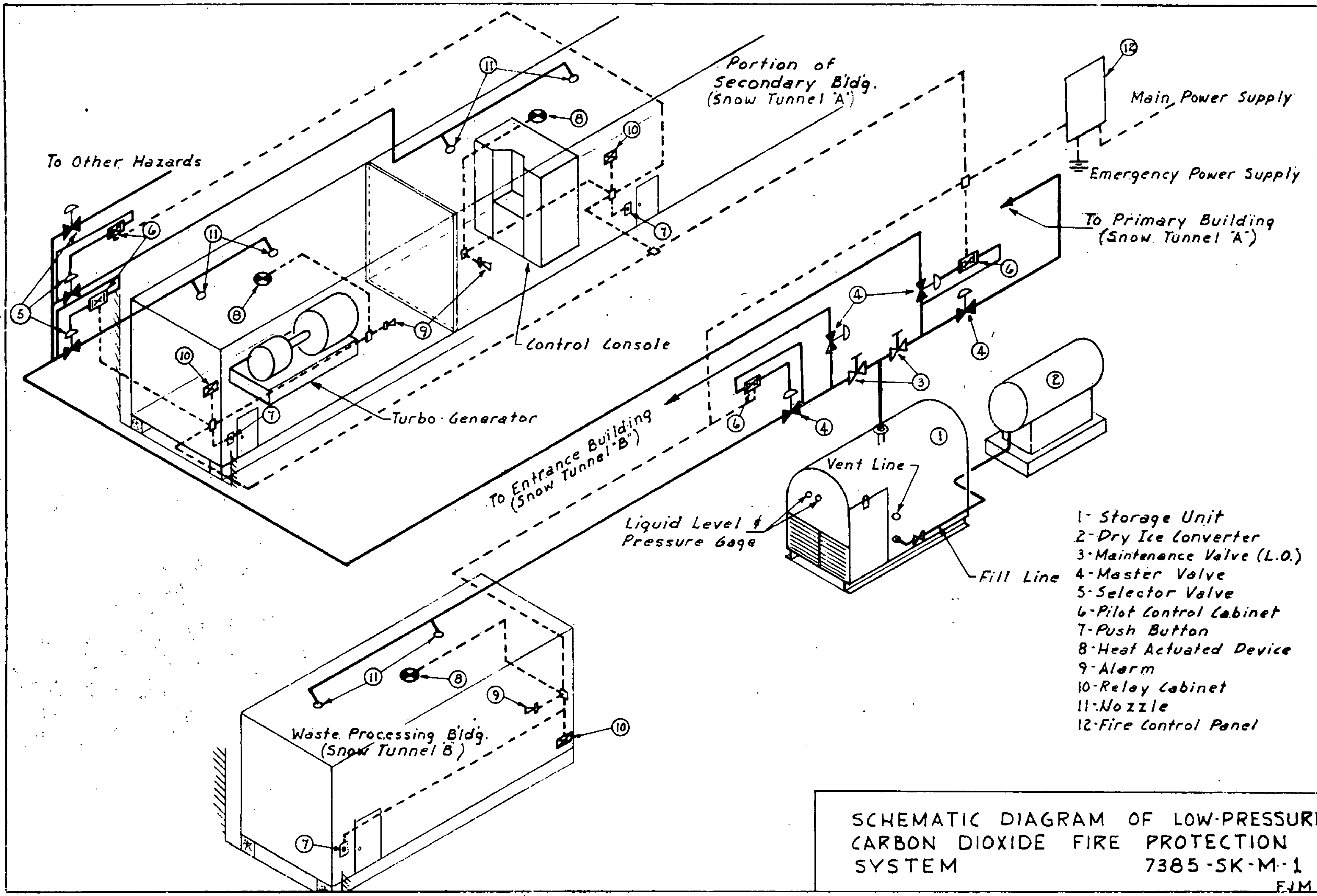
NO	DESCRIPTION	CHKD.	APPD.	DATE

REVISIONS

SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	NO.
	DS				1-10-62	7385-2	7385-SK-E-1

JACKSON & MORELAND INC., ENGINEERS

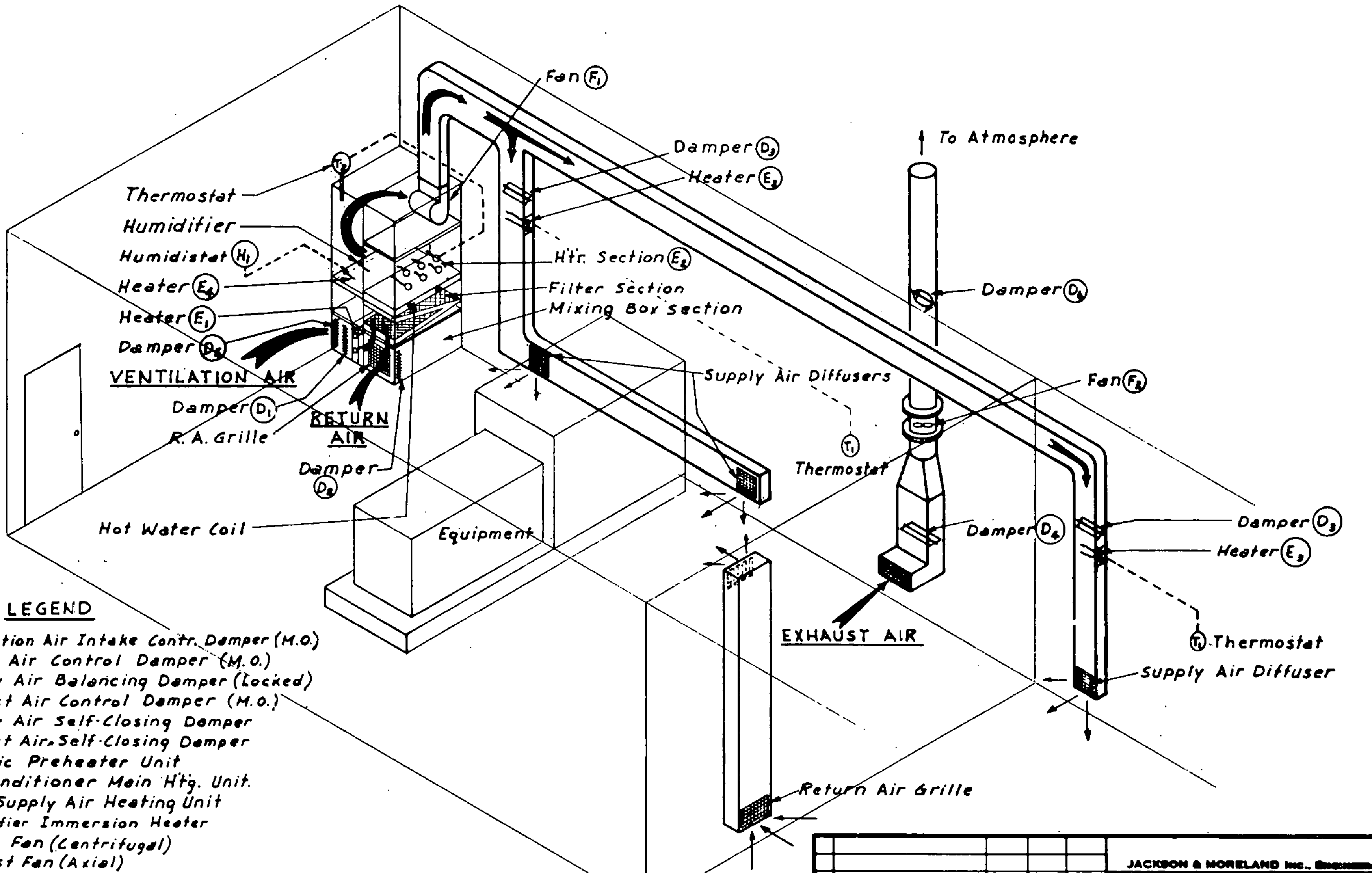
ONE LINE DIAGRAM
LIGHTING & MISC. POWER



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

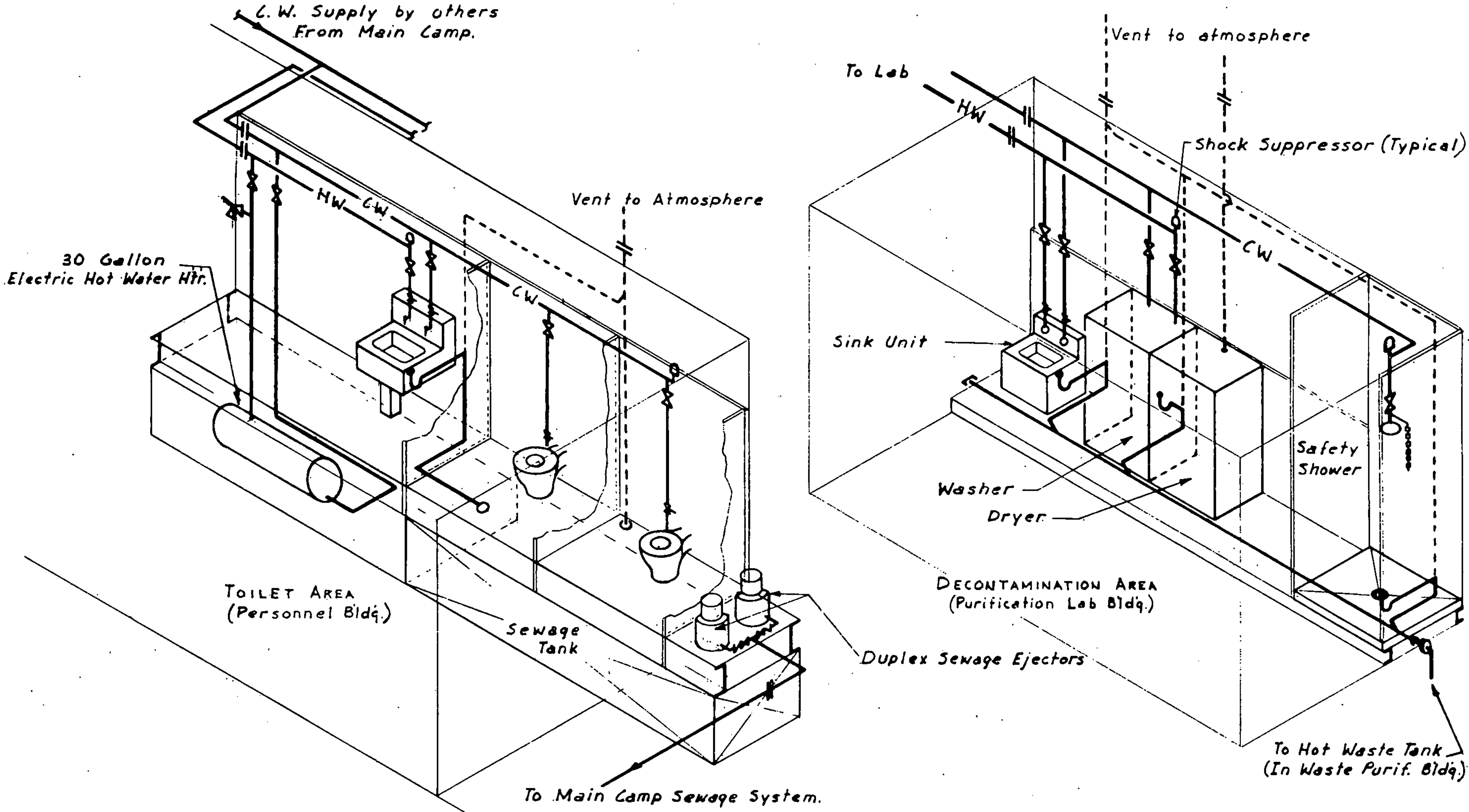
NO	DESCRIPTION	CHKD.	APPD.	DATE

REVISIONS

JACKSON & MORELAND Inc., Engineers

TYPICAL ARRGT OF HEATING & VENTILATG SYS

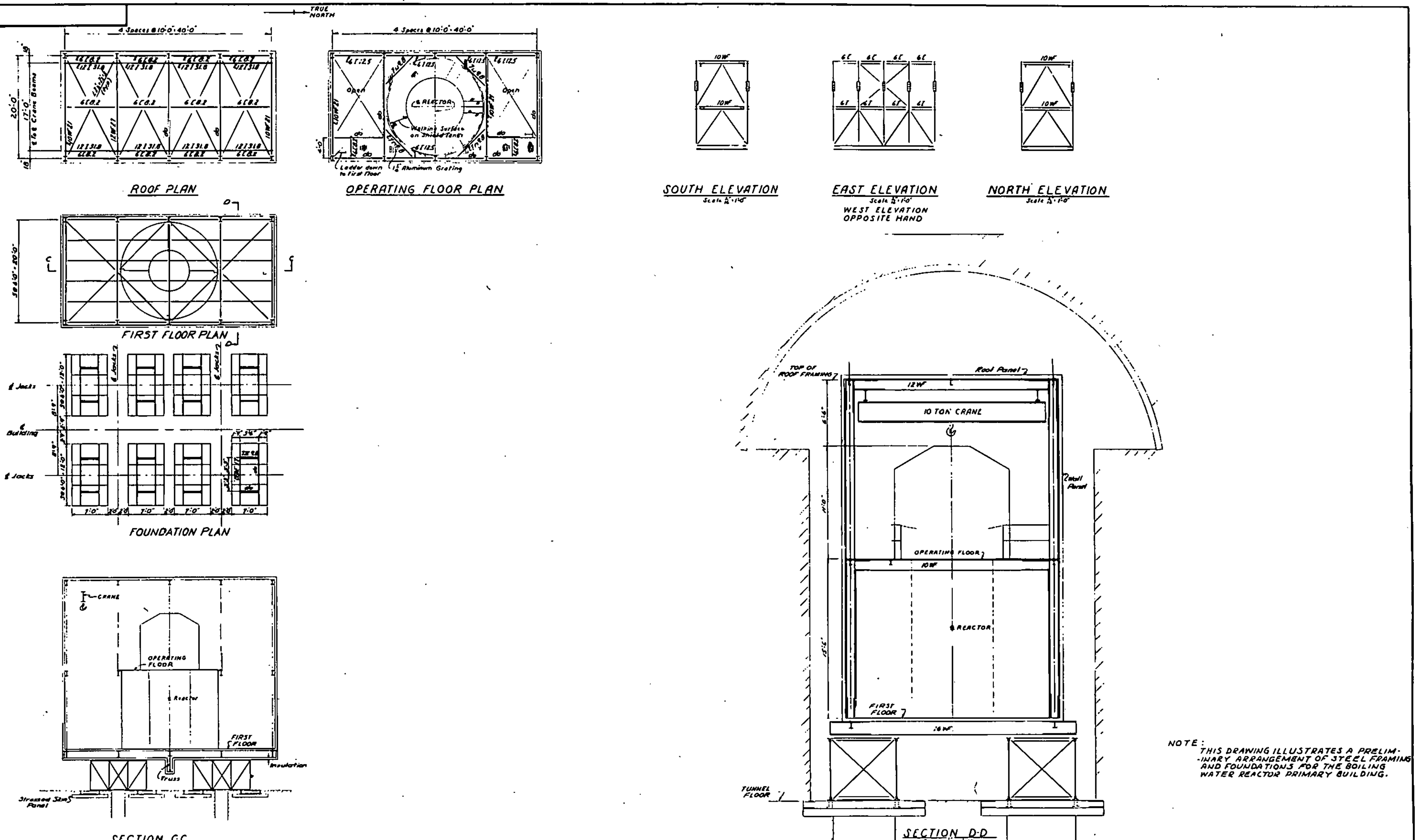
SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	NO.
NONE	FJM					7385	7385-SK-M-2



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



NOTE: THIS DRAWING ILLUSTRATES A PRELIMINARY ARRANGEMENT OF STEEL FRAMING AND FOUNDATIONS FOR THE BOILING WATER REACTOR PRIMARY BUILDING.

Title of Reference Drawings	Number
Plan Plan Arrangement Based on B.W.R. Concept	AEL-738
B.W.R. Complex Direct Cycle Concept B-3	AEL-696

APPROVED .... 19...

PROJECT MANAGER

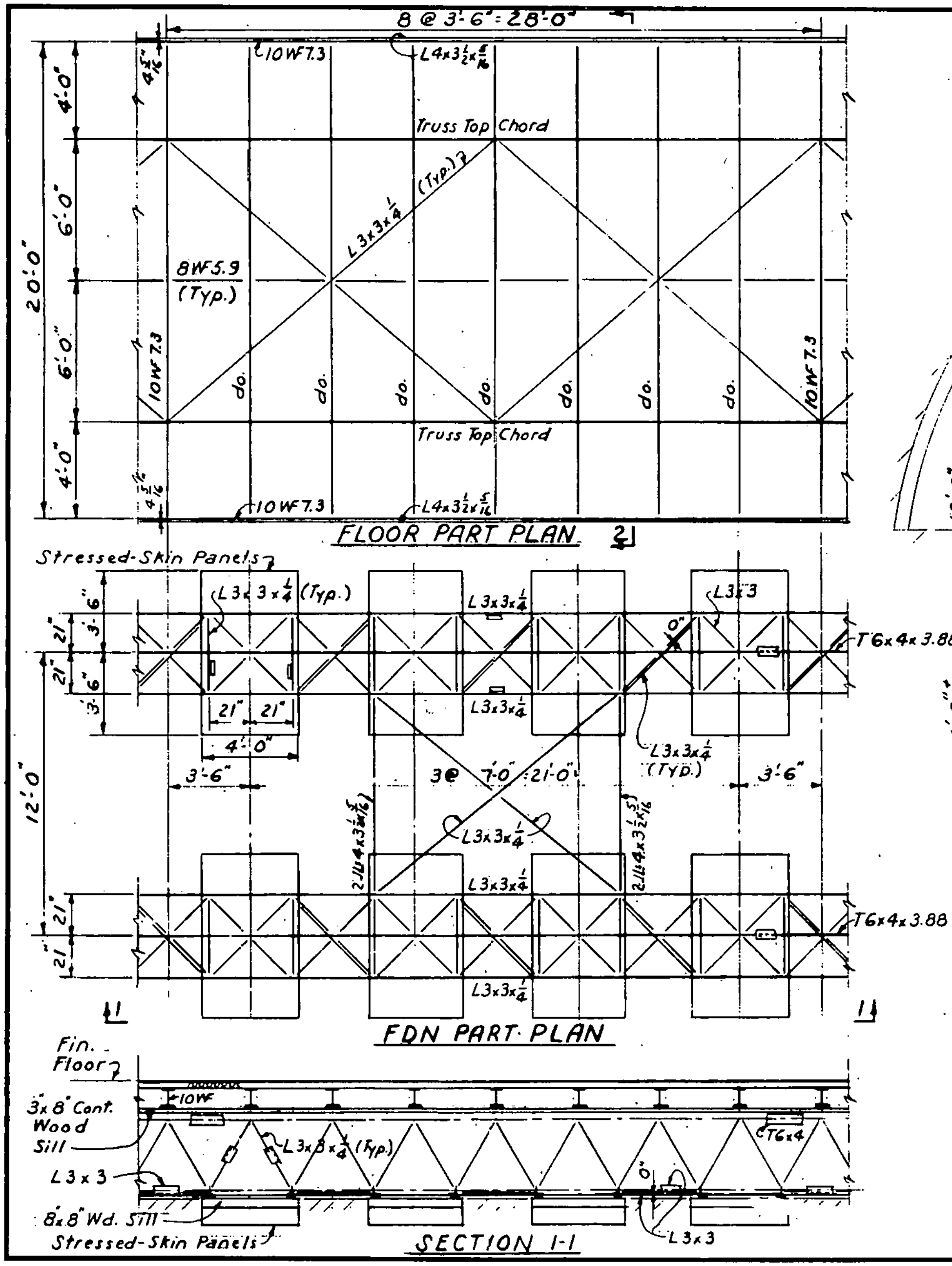
ALCO PRODUCTS INC.

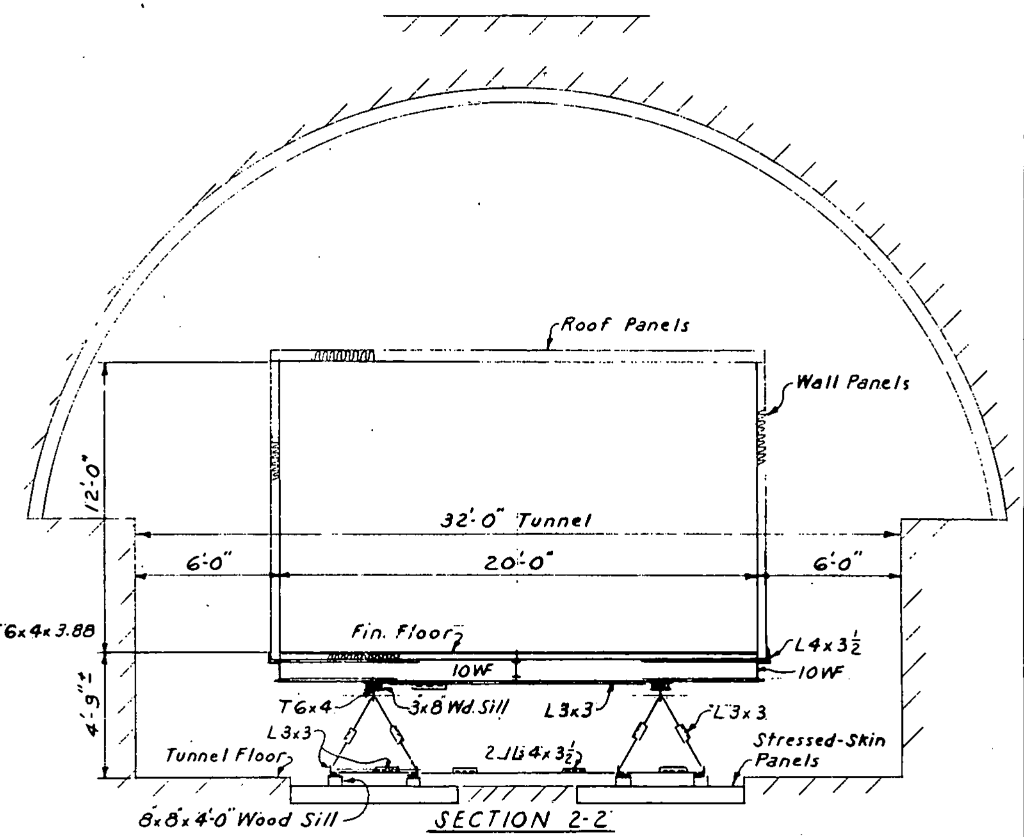
JACKSON & MORELAND, INC., ENGINEERS BOSTON — NEW YORK

PL3

BOILING WATER REACTOR PRIMARY BUILDING

Record Number	Scale	
7385	IN. = 1 FOOT	7385-SK-S-8





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
PL3
ENTRANCE BUILDING

SCALE	DRAWN	CHKD.	DES. SUP.	APPROVED	DATE	RECORD NUMBER	
3/16" = 1'-0"					2-13-62	7385	NO. 7385-SK-S-11