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WKNL-49

REACTORS—POWER

UNITED STATES ATOMIC ENERGY COMMISSION

NUCLEAR POWER FOR PITTSBURGH; FINAL REPORT ON DUQUESNE–KIDDE NUCLEAR POWER STUDY

By
James J. Barker

November 10, 1955

Walter Kidde Nuclear Laboratories, Inc.
Garden City, New York

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Garden City, New York

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NUCLEAR POWER FOR PITTSBURGH

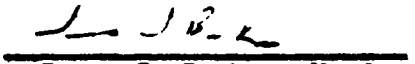
Final Report of

WALTER KIDDE NUCLEAR LABORATORIES, INC.

On

DUQUESNE - KIDDE NUCLEAR POWER STUDY

Prepared By:



James J. Barker, Head
Process Department

Study Contract
Duquesne - Kidde - AEC

November 10, 1955

Table of Contents

LIST OF TABLES

LIST OF FIGURES

FOREWORD

On December 9, 1953 the Duquesne Light Company and Walter Kidde Nuclear Laboratories, Inc. embarked on a joint study of nuclear power. Responsibility for direction of the study was vested in the Management Committee, whose membership was as follows:

Duquesne Light Company		Walter Kidde Nuclear Laboratories, Inc.
Chairman	R. B. Donworth	K. Cohen
Secretary	R. B. Oskin	W. I. Thompson
	E. M. Gue	
	E. M. Parrish	

Responsibility for accomplishing the study was delegated to the Joint Technical Staff, whose membership was as follows:

Walter Kidde Nuclear Laboratories, Inc.		Duquesne Light Company
Chairman	W. I. Thompson	R. B. Donworth
	K. Cohen	R. B. Oskin
	J. J. Barker (part time)	W. A. Conwell
	J. H. Frankfort "	E. M. Gue
	K. H. Puechl "	J. A. Tash
	G. B. Webb "	G. E. Moyer

The objective of this study was " . . . A study of the practical and economic use of nuclear energy for the production of electric power in the Pittsburgh district."

The study was divided into three phases: information survey of existing and possible future technology, preliminary economic study and choice of reactor for comparison, and detailed cost study of a specific cycle.

The Duquesne Light Company subsequently became engrossed by the Shippingport power project and the present study was terminated before it had been completed.

Visits were made by members of both companies to various AEC installations and power companies for the information survey. Progress was made in the first and second phases, but the third phase was essentially untouched.

This report covers the technical contribution of the Walter Kidde Nuclear Laboratories to this study, and is a summary of the work accomplished up to termination. The steam cycle information given in the appendix to Chapter III was provided by Duquesne Light Company.

Chapter I

INTRODUCTION

The Pittsburgh area, toward which this study is directed, is characterized by low fuel cost (presently about 20¢/MM btu) and relatively limited cooling water capacity. The demand for power is large, both by industrial and private users, and new power plants will probably be constructed in units of 200,000 KW or larger. Consideration of the cooling water problem has tended to push plant designs toward high temperatures and consequent high thermodynamic efficiencies, in spite of the relatively low cost of coal.

A preliminary survey of estimated nuclear power costs showed that nuclear plants of the types presently available or with any appreciable technological background are unlikely to compete in the Pittsburgh area with ordinary coal-fired plants now or in the next decade. As a result, the study was oriented toward more advanced concepts which might be developed in a later period. The investigations were therefore broad and speculative, with interest in the period from 10 to 20 years in the future when new processing and metallurgical technology might be available.

However, to provide a basis for comparison, a high-temperature thermal reactor, which it was thought might represent a near-future possibility, was investigated as a first step. The selection for this investigation was a graphite-moderated and sodium-cooled reactor, a type which has received considerable attention from other groups. This reactor is capable of delivering heat at temperatures high enough for good thermodynamic efficiency. Certain other possibilities were rejected for various reasons as follows. The Liquid Metal Fuel Reactor was considered, but data were not at the time available to permit an intelligent economic comparison. The various water-cooled reactors (with the possible exception of the supercritical water reactor) all produce saturated steam at fairly low temperatures and have maximum thermodynamic efficiencies not exceeding about 25%.

Chapter II, which is divided into two sections, presents the technical information that was developed for the study of the sodium-graphite reactor. Section 1 summarizes process, theoretical and mechanical studies and presents some preliminary information on fuel economics. Section 2 presents design and layout information for a complete power plant to produce 250,000 KW of electric power.

Chapter III presents a brief survey and analysis of the field of fast breeder reactors.

Chapter II

SODIUM-GRAPHITE REACTOR

Section 1
Exploratory Studies

A sodium-cooled graphite-moderated reactor was chosen for initial investigation because of its high potential thermal efficiency and because its technology is sufficiently advanced for reasonable projections of costs. Although the design study presented here is new, we have taken full advantage of the work of other organizations, notably North American Aviation and Monsanto Chemical Company, for technical information and design ideas on the sodium-graphite reactor concept. A partial listing of the references consulted is given at the end of this chapter.

The reactor fuel is nearly pure metallic uranium jacketed in zirconium. The moderator is in the form of graphite blocks, also encased in zirconium for protection from the coolant. The coolant is liquid sodium. Within this framework the studies covered various types of fuel elements (rods, plates, annuli), arrangements of fuel (clumping vs. even distribution), core sizes and shapes, and graphite-to-uranium ratios. There seems to be no economic reason to consider NaK as an alternate to sodium.

Mechanical Design

All slightly enriched, sodium-cooled, graphite-moderated power reactors have the following features:

1. large size.
2. high temperature.
3. low pressure.
4. large graphite block construction.
5. fine fuel element structure.

The large size is dictated by the criticality requirements of a graphite-moderated reactor. High temperature and low pressure are desirable for efficiency and economy and are possible because of the high boiling point of sodium. Large graphite block construction (fuel lumping) is necessary in order to reduce neutron absorption in U-238. Finally, the high coolant temperature and correspondingly high wall temperature of the fuel elements require that the fuel elements be thin in order to keep the peak temperature at the center of the fuel element at a reasonable level.

These features are exhibited by the sodium-graphite reactor design studies that have been made by Massachusetts Institute of Technology, North American Aviation, and Monsanto Chemical Company. As a starting point in our

own conceptual design, their reports were critically examined and the desirable features of each, where compatible, were incorporated in our design.

The major differences between the design presented here and the previous designs are the location of the thermal shields and the manner of support of the reactor vessel. The thermal shields have been placed inside the vessel to protect it and so that the heat generated in the thermal shields is readily available to the steam plant. The large thin-walled reactor vessel at high temperature presents a difficult problem in regard to supports. The support structure must be designed to accommodate the thermal expansion. The novel way of accomplishing this suggested here is to float the reactor vessel in a pool of dense liquid metal.

General Description

The core consists of vertical fuel elements arranged on a square lattice among blocks of graphite in zirconium cans, with sodium coolant flowing upward around the elements and between the graphite blocks. Figure 2.1 is a sketch of a vertical section through the reactor, reactor shell and concrete biological shield. Figure 2.2 is a sketch of a horizontal section through the reactor core, showing alternate arrangements for rod and for annular fuel elements. Pressures inside the vessel are small, since the coolant is sodium.

Coolant enters the vessel through pipes around its periphery near the top and flows downward through the thermal shield protecting the sides of the vessel. At the bottom of the vessel, the coolant flows radially inward through the bottom thermal shield to distributors which regulate the flow to the fuel elements in proportion to their heat generation. The coolant flows upward through the core and exits from the vessel through pipes around the periphery just above the inlet pipes. The sodium is prevented from contacting the top of the vessel by a blanket of inert gas which maintains a free surface of sodium inside the vessel above the level of the outlet pipes. The fuel elements are suspended by rods attached to plugs in the top of the vessel. Control rods are located inside thimbles which project downward through the core from holes through the top cover. The vessel is protected and heat losses are reduced by insulation between the thermal shield and the vessel wall.

The vessel floats in a bath of mercury, and provision is made for vertical and horizontal thermal expansion of the vessel by joints near its top and on the coolant pipes. A pin at the center of the bottom of the vessel which fits into a hole in the concrete shield keeps the vessel from shifting radially in the bath of liquid metal. The mercury is contained in a stainless steel liner mounted flush against the concrete biological shield.

Core

Graphite Blocks: The graphite moderator is canned in blocks so that a leak of sodium coolant will not contaminate all the graphite in the reactor. Should an individual block become contaminated, it may be removed

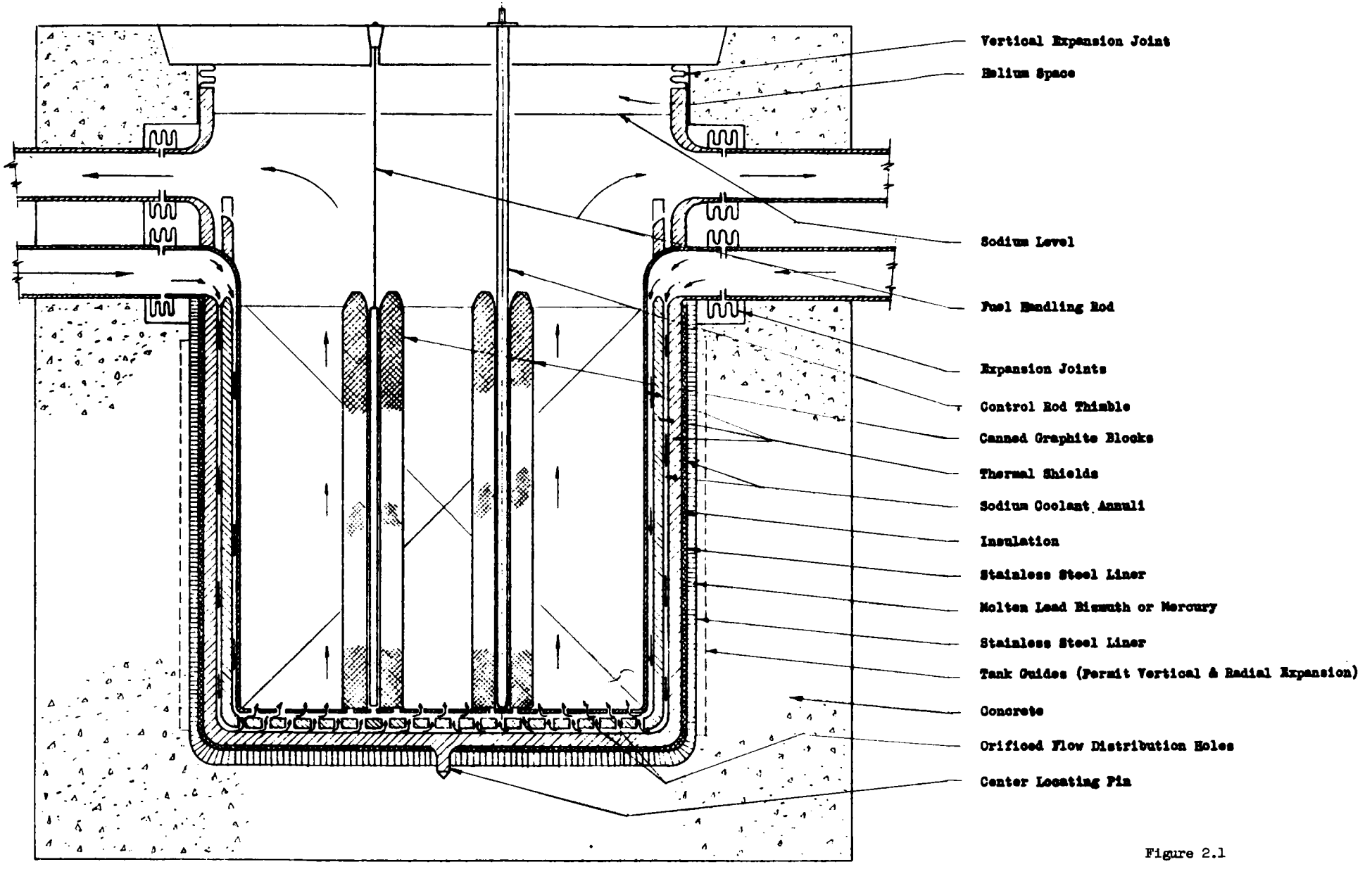


Figure 2.1

Sodium-Graphite Reactor Vertical Section

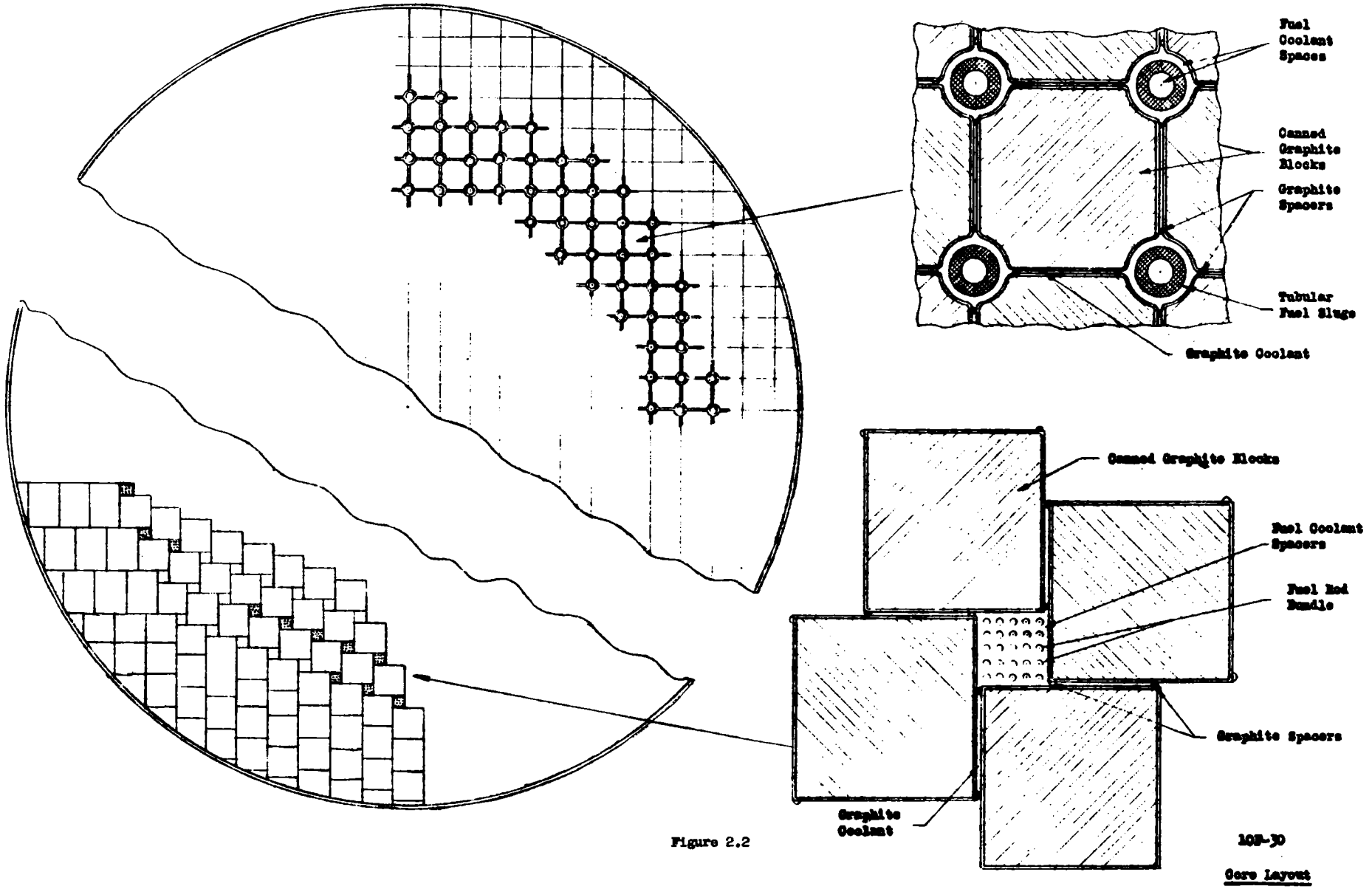


Figure 2.2

108-30

Core Layout

and replaced through the top of the vessel. There should be little effect of irradiation on graphite at the temperatures of interest.

The bottoms of the canned graphite blocks fit into a waffle-like structure for accurate location. The tops of the cans are heavy zirconium caps which fit together closely and align the upper portion of the core. The bodies of the canned blocks are somewhat smaller than the end fittings so that sodium may flow between the cans to cool them.

Control Rods: Metal thimbles suspended from the top of the reactor vessel house the control rods to isolate them from the reactor coolant and inert gas systems.

Fuel Elements: Figure 2.2 shows two arrangements of the core, one using rod elements and the other using tubular elements. Box-like elements, made of plates, could be adapted also, but the manufacturing advantages inherent in the circular geometry of the rod and the annular elements made these the first choice. The rod cluster design permits a good distribution of coolant and fuel, but introduces additional handling and spacing difficulties.

The elements are jacketed with zirconium to contain fission products and prevent the sodium from eroding or corroding the uranium fuel.

The fuel elements are suspended in the core by handling-rods which are permanently attached to the tops of the elements and extend upward to the plugs in the apertures in the top of the vessel. Loading of the fuel elements is expedited by the use of guide tubes which extend from the top of the core to the holes in the top of the vessel.

The fuel rods are relatively long and should be sufficiently flexible to avoid jamming due to warpage. Since the zirconium jackets will probably have no tendency to warp by themselves, the necessary flexibility can be provided by the use of short fuel slugs instead of continuous fuel rods.

Vessel

The reactor runs at high temperatures and is bulky and heavy. However, the internal pressure is low because of the low vapor pressure of the sodium coolant.

Vessel Support: Thin vessel walls are desirable because of cheapness, ease of fabrication and reduction of thermal stresses caused by the heat generated by radiation. By floating the vessel in a pool of mercury the structure is supported without heavy construction for support members. Although the vessel is free to expand in the pool, it is located laterally by a pin at the center of the bottom which fits into a hole in the concrete biological shield. Vertical movement upon a change in load in the vessel is minimized by keeping the clearance between the vessel wall and the mercury tank small.

Thermal Expansion: Thermal expansion is provided for by expansion joints on the coolant pipes and around the periphery of the vessel at the top.

Thermal Shield: A thermal shield surrounds the sides and bottom of the reflected reactor core and protects the vessel wall from most of the radiation from the reactor. The sodium pool above the reactor core serves as a thermal shield for the upper portions of the vessel. The thermal shield is cooled by the sodium entering the vessel. Insulation between the thermal shield and the vessel minimizes heat losses from the reactor and reduces the vessel wall temperature.

Biological Shield: The reactor vessel is surrounded by a concrete biological shield. The small amount of heat generated by the residual radiations reaching the biological shield is removed by cooling pipes imbedded in the concrete.

Reactor Physics

Reactivity calculations were made for a number of cases to cover the range of interest. The reactor was taken to be a right square cylinder with the core surrounded by a graphite reflector 2 ft. thick. The clumped fuel elements were calculated as a homogeneous mixture of uranium, sodium and zirconium. The sodium to uranium volume ratio was fixed at 1.0 and the zirconium to uranium volume ratio fixed at 0.4. Results are given for graphite to uranium volume ratios from 15 to 35, uranium inventories from 50 to 150 metric tons and fuel clump radius from 2 to 6 cm.

Multiplication Constants: Multiplication constants for infinite lattices in the hot clean condition are given in Table 3.1. The fast fission factor was calculated assuming no interaction between fuel clumps. The resonance escape probability was obtained from experimental data in the Reactor Handbook. Thermal utilization was calculated by diffusion theory, using a neutron temperature of 0.0928 e.v. The same neutron temperature was used to calculate the number of fast neutrons produced per thermal neutron absorbed in the homogeneous fuel clump. Table 2.1 shows that, for a given enrichment, the multiplication constant increases with graphite to uranium ratio and with fuel clump radius, within this particular domain.

Enrichment Required for Criticality: Table 2.2 gives the enrichments required for criticality in finite reactors. Poisons, expressed as per cent increase in k needed for criticality, were taken to be 2.7 for xenon, 0.7 for samarium and 0.6 for residual control. Leakage was computed for right square cylindrical reactors with a reflector savings of 50 cm., the values for τ and L^2 being as follows:

Constants for Leakage Calculation

Volume Ratio: Graphite to Uranium	Fermi Age, τ, cm.^{2}	Diffusion Area, L^2, cm.^{2} Volume Ratio: (Na + Zr)/U		
		1	2	3
15	382	95.9	116	136
20	371	123	148	172
25	360	150	180	208
35	352	202	242	282

Table 2.2 shows that the enrichment required for criticality decreases as fuel clump radius, uranium inventory and graphite to uranium ratio increase. For the largest reactors, the enrichment required (0.78 atom per cent 235) is close to natural abundance.

Initial Conversion Ratio: Table 2.3 gives values for the initial conversion ratio (atoms of plutonium produced per U_{235} atom destroyed). The conversion ratio decreases as the fuel clump radius and the graphite to uranium ratio increase, but it increases with uranium inventory. The conversion ratio ranges between 0.83 and 1.013 for the cases considered.

Reactivity Lifetime: Table 2.4 gives the reactivity lifetimes for the various cases. The lifetime as here defined is that point at which a reactor starting out just critical (after xenon and samarium equilibrium has been reached) again becomes just critical with the initial amount of shim control. The lifetime varies in the same manner as the conversion ratio: decreasing with increasing clump radius and graphite to uranium ratio, but increasing with increasing uranium inventory. The lifetimes range from 3000 to 12,000 megawatt-days per ton of uranium for the cases considered.

The lifetimes were predicted assuming a fission product cross section of 100 barns per fission (excluding xenon and samarium) at .025 e.v. with a 1/v variation, and an α (capture to fission ratio) for Pu^{239} of 0.60. Since estimates of the fission product cross section vary from 50 to 200 barns per fission, and estimates of α for Pu^{239} vary from 0.48 to 0.65, depending on the neutron temperature and the treatment of the Pu^{239} resonance at 0.3 e.v., any estimate of fuel lifetime is uncertain. However, if the fuel lifetime is overestimated by 50 per cent, the calculated lifetime can be obtained by over-enriching the fuel initially at an additional fuel cost of only about 20 per cent.

Another assumption is that the irradiation occurs in a uniform neutron flux; i.e., all portions of the reactor receive the same amount of irradiation per unit time. This uniformity can be approached only by redistributing the fuel within the reactor during the radiation cycle. Unless the fuel is redistributed, a cosine flux in a single direction reduces the lifetime about 25%.

Coolant and Cladding Volume: For the range of interest, the effect of varying coolant to uranium and zirconium to uranium volume ratios on the enrichment required for criticality may be approximated as follows:

$$\frac{\Delta(\text{Enrichment})}{\Delta(\text{Volume Na/Volume U})} = 0.09 \text{ atom per cent } 235$$

$$\frac{\Delta(\text{Enrichment})}{\Delta(\text{Volume Zr/Volume U})} = 0.08 \text{ atom per cent } 235$$

The reactivity lifetime is not strongly affected by variations in the sodium or zirconium volume ratios because the loss in conversion ratio is compensated by the higher enrichment.

Table 2.1

Multiplication Constants

R \ r_o	k_∞, hot, clean					
	2	4	6	2	4	6
	E = 0.72			E = 0.85		
15	0.725	0.830	0.899	0.778	0.890	0.965
20	0.823	0.917	0.972	0.885	0.985	1.043
25	0.885	0.961	1.005	0.950	1.034	1.081
35	0.960	1.011	1.030	1.033	1.087	1.107
	E = 1.00			E = 1.25		
15	0.827	0.946	1.027	0.889	1.018	1.103
20	0.941	1.047	1.108	1.012	1.127	1.193
25	1.013	1.099	1.149	1.089	1.183	1.236
35	1.100	1.157	1.179	1.186	1.246	1.271
	E = 1.50			E = 2.00		
15	0.938	1.075	1.166	1.009	1.156	1.250
20	1.068	1.190	1.260	1.147	1.277	1.353

Notes:

E = uranium enrichment, atom per cent 235 in 235 + 238.
r_o = fuel clump radius, cm.
R = graphite volume/uranium volume

Volume Na/Volume U = 1.0
Volume Zr/Volume U = 0.4

Fuel clump homogeneous mixture of U, Na and Zr.

No structure aside from fuel-coolant assemblies.

Neutron temperature = 0.0928 e.v.

Table 2.2

Enrichments Required for Criticality

T \ r_o	Enrichment Atom Per Cent U-235 in Uranium					
	2	4	6	2	4	6
		R = 15			R = 20	
50	---	1.74	1.29	1.74	1.20	1.00
100	---	1.58	1.20	1.60	1.12	0.95
150	---	1.53	1.16	1.54	1.09	0.92
		R = 25			R = 35	
50	1.31	1.01	0.90	0.99	0.87	0.83
100	1.23	0.96	0.86	0.94	0.84	0.80
150	1.19	0.93	0.84	0.92	0.82	0.78

Notes:

r_o = fuel clump radius, cm.
R = graphite volume/uranium volume.
T = metric tons of uranium (2205 lbs./etric ton)

Volume Na/Volume U = 1.0
Volume Zr/Volume U = 0.4
No structure aside from fuel-coolant assemblies
Fuel Clump homogeneous mixture of U, Na and Zr
Right square cylinder; reflector savings 50 cm.

Poison	% k
Xe	2.7
Sm	0.7
Control	0.6

Table 2.3

Initial Conversion Ratio

T \ r_o	2	4	6	2	4	6
		R = 15			R = 20	
50	—	0.953	0.941	0.930	0.930	0.913
100	—	0.997	0.981	0.968	0.968	0.947
150	—	1.013	1.001	0.986	0.986	0.969
		R = 25			R = 35	
50	0.913	0.906	0.882	0.889	0.864	0.829
100	0.948	0.940	0.913	0.921	0.889	0.856
150	0.966	0.959	0.929	0.936	0.906	0.873

Notes:

See notes in Table 2.2.

Conversion Ratio = atoms Pu produced per U-235 atom destroyed.

Table 2.4

Reactivity Lifetime

T \ r_o	Reactivity Lifetime, Thousands of Megawatt-Days per Ton of Uranium					
	2	4	6	2	4	6
		R = 15			R = 20	
50	—	11.0	7.80	10.1	6.96	5.40
100	—	12.0	8.46	10.7	7.50	5.84
150	—	12.4	8.93	11.1	7.85	6.21
		R = 25			R = 35	
50	7.07	5.25	4.28	4.85	3.78	3.07
100	7.63	5.76	4.64	5.22	4.12	3.36
150	7.91	6.04	4.83	5.43	4.27	3.55

Notes:

See notes in Table 2.2.

Fission product cross section (excluding Xe and Sm) taken as 100 barns at 0.025 e.v. with 1/v variation.

Capture to Fission ratio for Pu^{239} taken as 0.6.

Uniform neutron flux.

Fuel Economics

The information on criticality and lifetime was used to obtain estimates of fuel costs for a simple, once-through operation in which the fuel is discarded after use. These studies were reported in detail in the interim report*. If the used fuel had no salvage value, fuel costs ranged between 2.5 and 4 mills/kw hr, which is clearly not competitive with coal at 20¢ per million Btu in modern plants in this size range. Even with a net salvage value of 50 per cent of the initial charge, fuel costs ranged between 1.5 and 2.9 mills/kw hr.

It appeared clear that feedback of reprocessed spent fuel would be necessary to a position competitive with coal, but whether such a scheme would be sufficient was not established.

Reactor Heat Removal

For the reactor heat removal studies, it is assumed that the core contains no structural materials other than the 40-mil-wall zirconium coolant tubes through the graphite moderator and the fuel jackets. The maximum-to-average heat flux ratios are 2.2 overall and 1.5 radially, with the axial flux distribution being a cosine function. The uranium fuel elements are clad with 20 mils of zirconium and the uranium-zirconium bond thermal resistance is assumed negligible. The maximum uranium temperature is 1100°F, which is safely below the transformation range. The coolant flow is distributed in proportion to the heat generation in each element.

The effects of fuel element shape, uranium inventory, core shape, graphite to uranium ratio, coolant temperature level and flow rate, and type of coolant on fuel element size and spacing, volumes of coolant and cladding material, overall core dimensions and coolant pressure loss and pumping power requirements were investigated by means of a case study.

Table 2.5 gives results of case studies for a reactor producing 750 MW of heat. Columns 1, 2 and 3 show the effects of fuel element shape: clusters of rods compared with annular elements and with flat-plate elements. Columns 4 through 7 show the effects of uranium inventory and graphite to uranium ratio, and columns 8 through 11 give the effects of coolant temperatures. Column 12 indicates the effects of reactor core shape. Column 13 compared with column 1 shows the effects of substituting eutectic NaK for Na as the coolant. Fuel clump radius was not specifically studied, since it affects only the lattice spacing and therefore graphite temperatures, which do not appear critical. Other process parameters are unaffected.

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* WKNL-36 "Interim Report on Duquesne-Kidde Study of Nuclear Power in Pittsburgh".

Effect of Fuel Element Shape: Three types of fuel elements were investigated: (1) solid rods; (2) annular cylinders (hollow rods); (3) flat plates. With the uranium inventory, power level, flux distribution, coolant conditions, etc. fixed, there is only one thickness, for each type of element, that satisfies the heat removal conditions. The dimensions of the elements are given in Figure 2.3 and the results of the comparison are given in columns 1, 2 and 3 of Table 2.5.

At the conditions of this comparison, the differences between elements are small, with flat plates showing a slight saving in power for circulation of coolant. At other conditions the picture will be different, so it is well to consider general effects. Flat-plate elements are most versatile, process-wise, since they can be made to fit any conditions of coolant to fuel and moderator to fuel ratio. Solid rods are less versatile, since they are limited by a minimum coolant to fuel ratio which is greater than zero. The annular cylinder is the least versatile of the three, since the thickness of the annulus is fixed by the inventory and power level of the reactor while the coolant to fuel ratio depends on the radius of the element. At larger fuel clump radii the coolant to fuel ratio can be reduced only by inserting another element inside the annular cylinder. Cooling the annular element on only its inner surface leads to impracticable designs.

Effect of Uranium Inventory and Graphite to Uranium Ratio: For a particular coolant velocity and temperature range, the effects of varying inventory are shown in columns 1, 4 and 7 of Table 2.5, and the effects of varying graphite to uranium ratio are shown by comparing columns 1,5 and 6.

Increasing the inventory increases the size of the core, the element diameter, the pressure loss and pumping power, and the absolute amounts of coolant and cladding material in the core, but decreases the number of elements and the clearance between fuel rods, the heat flux, and the coolant to fuel and cladding to fuel volume ratio.

Increasing the graphite to uranium ratio increases the size of the core, the clearance between elements and the coolant volume, but decreases the number of elements and the pumping power.

Effect of Coolant Temperature: Comparison of column 1 with columns 8, 9 and 10 shows the effects of coolant temperatures with the fuel element diameter fixed, and comparison of columns 1 and 11 shows the effects when the core size is fixed.

Decreasing the exit temperature results in a smaller reactor, but increases the pumping power. Increasing the inlet temperature results in a larger reactor as well as greater pumping requirements. Raising the temperature level of the coolant by 100°F increases the uranium inventory by 70 per cent and increases the core diameter to about 20 ft., but the pumping power is reduced.

Table 2.5

Comparison of Various Liquid-Metal Cooled, Graphite Moderated Reactors

Bases for Calculations

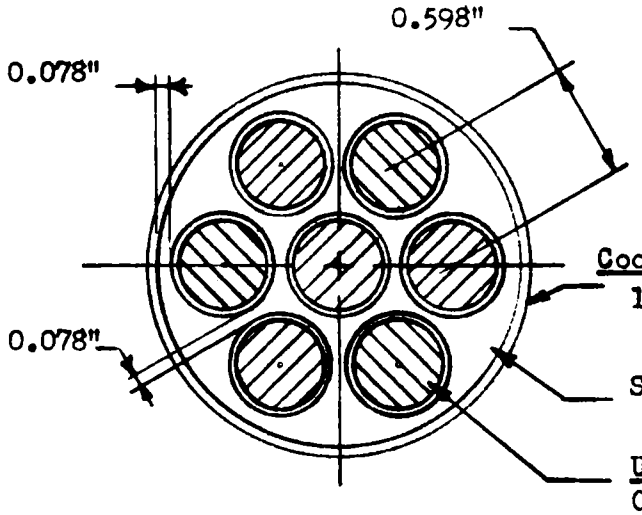
Heat Level, MW	750	Thermal Conductivity, Btu/hr.ft.^{2}($^\circ$F/ft.)	
Maximum Uranium Temperature, $^\circ$F	1100	Uranium	15.0
Core Shape	Cylinder	Zirconium	11.0
Max. to Avg. Heat Flux Ratio, Overall	2.2	Uranium-Zirconium Bond Resistance Assumed zero	
Radial	1.5	Core consists of coolant tubes through graphite; i.e., no structure other than coolant tubes	
Axial	Cosine Function		
Zr Cladding Thickness, mils	20		
Zr Coolant Tube Wall Thickness, mils	40		

	Effect of Fuel Element Shape			Effect of Uranium Inventory and of Graphite to Uranium Ratio				Effect of Inlet and Exit Temperatures of Reactor Coolant				Effect of Core Shape	Effect of Reactor Coolant
Column	1	2	3	4	5	6	7	8	9	10	11	12	13
Uranium Inventory, Metric Tons	75(1)	75	75	50	75	75	150	62	84.5	128	75	75	75
Graphite to Uranium Volume Ratio	25	25	25	25	15	35	25	25	25	25	25	25	25
Coolant to Uranium Volume Ratio	1	1	1	1.32	0.86	1.11	0.63	1	1	1	1	1	1.58
Zirconium to Uranium Volume Ratio	0.37	0.35	0.34	0.48	0.37	0.37	0.24	0.37	0.37	0.37	0.486	0.37	0.45
Core Diameter, ft.	17	17	17	14.9	14.5	18.9	21.2	15.9	17.6	19.8	17.0	18.7	17
Core Height, ft.	17	17	17	14.9	14.5	18.9	21.2	15.9	17.6	19.8	17.0	14.0	17
Fuel Element Type	7-Rod Clusters	Annular Cylinder	Flat Plates	7-Rod Clusters	→								7-Rod Clusters
Number of Fuel Elements	940	1,180	690	1,129	1,100	847	727	830	1,019	1,280	1,574	1,135	1,240
Number of Fuel Element Sub-Units	6,580	1,180	2,070	7,903	7,700	5,929	5,089	5,810	7,133	8,960	11,018	7,945	8,680
Element Dimensions:		(See Figure 1)											
Fuel Rod Diameter, without Cladding, ins.	0.480	----	----	0.382	0.480	0.480	0.690	0.480	0.480	0.480	0.370	0.480	0.416
Fuel Rod Pitch (Triangular), ins.	0.598	----	----	0.509	0.580	0.610	0.786	0.598	0.598	0.598	0.469	0.598	0.573
Clearance between Fuel Rods, mils	78	----	----	87	60	90	56	78	78	78	59	78	117
Inside Diameter of Coolant Tube, ins.	1.87	----	----	1.61	1.81	1.92	2.41	1.87	1.87	1.87	1.46	1.87	1.837
Lattice Spacing, ins.	5.88	5.25	6.85	4.71	4.65	6.90	8.46	5.88	5.88	5.88	4.53	5.88	5.17
Reactor Coolant	Sodium	→										Sodium	NaK(22-78)
Coolant Temperatures, $^\circ$F: In	500	500	500	500	500	500	500	500	600	600	600	500	500
Out	900	900	900	900	900	900	900	800	900	1,000	1,000	900	900
Total Coolant Flow, lbs./hr.	21.0×10^6	21.0×10^6	21.0×10^6	21.0×10^6	21.0×10^6	21.0×10^6	21.0×10^6	27.8×10^6	28.1×10^6	21.2×10^6	21.2×10^6	21.0×10^6	30.6×10^6
Maximum Coolant Velocity, ft./sec.	19.6	----	----	19.6	19.6	19.6	19.6	29.6	24.5	14.8	20	16.35	19.6
Inner Passage(s)	----	25.0	20.9	----	----	----	----	----	----	----	----	----	----
Outer Passage(s)	----	17.1	16.9	----	----	----	----	----	----	----	----	----	----
Frictional Pressure Drop in Tubes, psi	26.6	24.2	23.1	23.0	29.1	25.8	34.8	52.6	39.3	18.1	36.4	15.5	16.85
Ideal HP for Frictional Pressure Drop	759	690	657	655	830	735	992	1,970	1,505	526	1,060	440	770
Zirconium Inventory, lbs.: Cladding	9,800	7,000	8,800	8,220	9,800	9,800	13,400	8,050	11,200	15,900	12,600	9,800	11,400
Coolant Tubes	10,700	11,900	10,300	9,730	10,300	11,000	13,500	8,350	12,100	17,100	14,000	10,700	13,800
Total	20,500	18,900	19,100	17,950	20,100	20,800	26,900	16,400	23,300	33,000	26,600	20,500	25,200
Maximum Heat Flux, Btu/hr./sq.ft.*	369,000	536,000	502,000	435,000	369,000	369,000	273,000	450,000	329,000	234,000	280,000	369,000	318,000

(1) Initial Design Case

* Based on Outside Surface of Cladding

Figure 2.3
Fuel Elements *

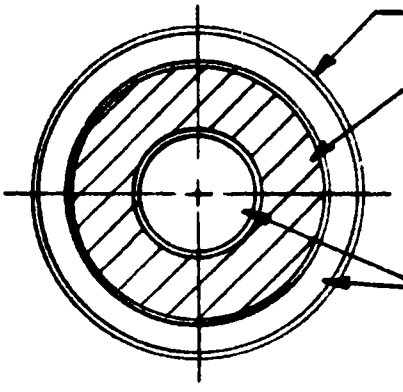


ROD CLUSTER

Coolant Tube
1.87" I.D. x 0.040" Zr Wall

Sodium Coolant

Uranium Rod
0.48" I.D. with 0.020" Zr Cladding

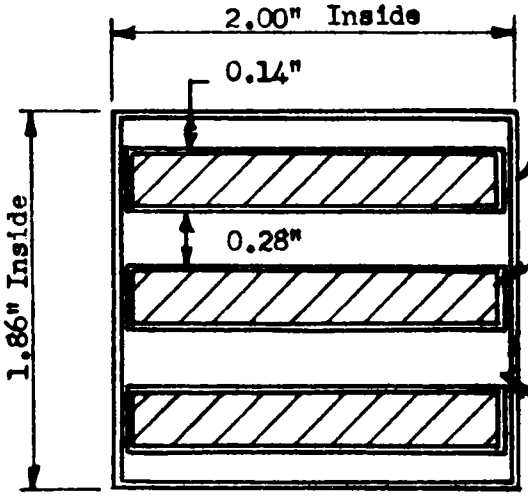


ANNULAR CYLINDER

Coolant Tube
1.66" I.D. x 0.040" Zr Wall

Uranium Annular Cylinder
0.68" I.D. x 1.32 ins. O.D.
with 0.020" Zr inside and outside

Sodium Coolant



FLAT PLATE

Coolant Tube
0.040 Zr wall

Uranium Plate
0.30" ins. x 1.92" ins.
with 0.020" Zr cladding

Sodium Coolant

* See columns 1, 2 and 3 of Table 2.5 for design bases.

For the same inventory, raising the coolant temperature decreases the element diameter and the clearance between elements, but increases the pumping power, the number of elements and the cladding volume.

Effect of Core Shape: The effect of changing the core shape from a square cylinder to that of a cylinder with a height equal to three-quarters of its diameter is shown by columns 1 and 12. The sodium velocity is decreased from 19.6 to 16.4 ft./sec. and the pumping power decreases by 42 per cent. The number of fuel elements increases by 21 per cent, but they are 14 ft. long instead of 17 ft.

Comparison of Sodium and Eutectic NaK as Coolants: Column 13 of Table 2.5 shows the effects of using eutectic NaK (22 w/o Na) as the coolant instead of sodium (column 1). Eutectic NaK melts at 12°F, and is therefore more adaptable than sodium, which melts at 208°F. However, its poorer heat transfer properties result in a greater number of smaller fuel elements and greater volumes of coolant and cladding material.

Potassium absorbs more neutrons than sodium, its price is 15 to 20 times as much per pound, and it is more reactive chemically, but it requires somewhat less shielding. The high melting point of sodium may be used to advantage, since by freezing a small section of a system local maintenance may be performed without draining the whole system.

Section 2
Plant Design Studies

To provide a basis for cost estimates, the sodium temperatures were set at 925°F leaving and 625°F entering the reactor, and the reactor heat was estimated at 615 megawatts. The design studies that were made on the primary circuit, the fuel handling system and the waste disposal system are described below. The studies performed by Duquesne Light Company on the steam plant are given in the appendix at the end of this section.

Primary Circuit and Auxiliaries

Flowsheets

Primary Circuit and Auxiliaries: Figure 2.4 is a process flowsheet for the primary circuit. Sodium coolant flowing at a rate of 54,400 gpm leaves the top of the reactor at 925°F and enters four 18" OD schedule 80 pipe lines connected to four sodium circulating pumps. Each sodium pump is rated at 13,800 gpm at a head of 150 ft. and will require 435 theoretical horsepower. A spare pump is provided in the event of mechanical failure of one of the operating pumps. From the pumps, the sodium flows through the intermediate exchangers and is returned to the bottom of the reactor at 625°F. The pumps are placed at the reactor exit to minimize the height of the sodium pool in the reactor. If the pumps were placed at the reactor entrance, a head of 26 ft. of sodium would be required above the reactor core, with a pressure of 1 atm above the pool. This pool height could be diminished by increasing the helium pressure above the pool, but this is undesirable since high helium pressure would mean increased leakage at the reactor shield plugs. However, by placing the pumps at the reactor exit, the height of the sodium pool need only be sufficient to prevent loss of prime to the pump. Cavitation is not encountered by installing the pumps in this manner, since the vapor pressure of sodium at 925°F is only 3.5 mm of mercury. In addition to removing heat from the fuel rods, the sodium flowing through the reactor core will be used to cool the control rods.

A cold trap is provided in the sodium circuit for the continuous removal of excess sodium monoxide. In the event complete drainage of the sodium system is required, a 20,000 gallon dump tank, connected to the sodium piping, is located below the lowest level of the reactor circuit. A 5000 gallon gravity feed tank is provided for charging sodium into the system and to supply emergency coolant to the reactor core should a loss of circulating sodium occur. Accidental loss of sodium from the reactor core would not only cause the fuel elements to melt, but would create the possibility of a reactivity accident, since poisoning of the core would be reduced without a compensating reduction in moderation. Safety considerations, therefore, must not only provide emergency cooling should power to the pumps fail, but must insure against loss of sodium from the reactor core.

Electrical resistance heaters are installed on the sodium pipe lines to provide heating during startup and to prevent sodium freezing during shut downs. Similarly, the gravity feed tank and sodium dump tank are pro-

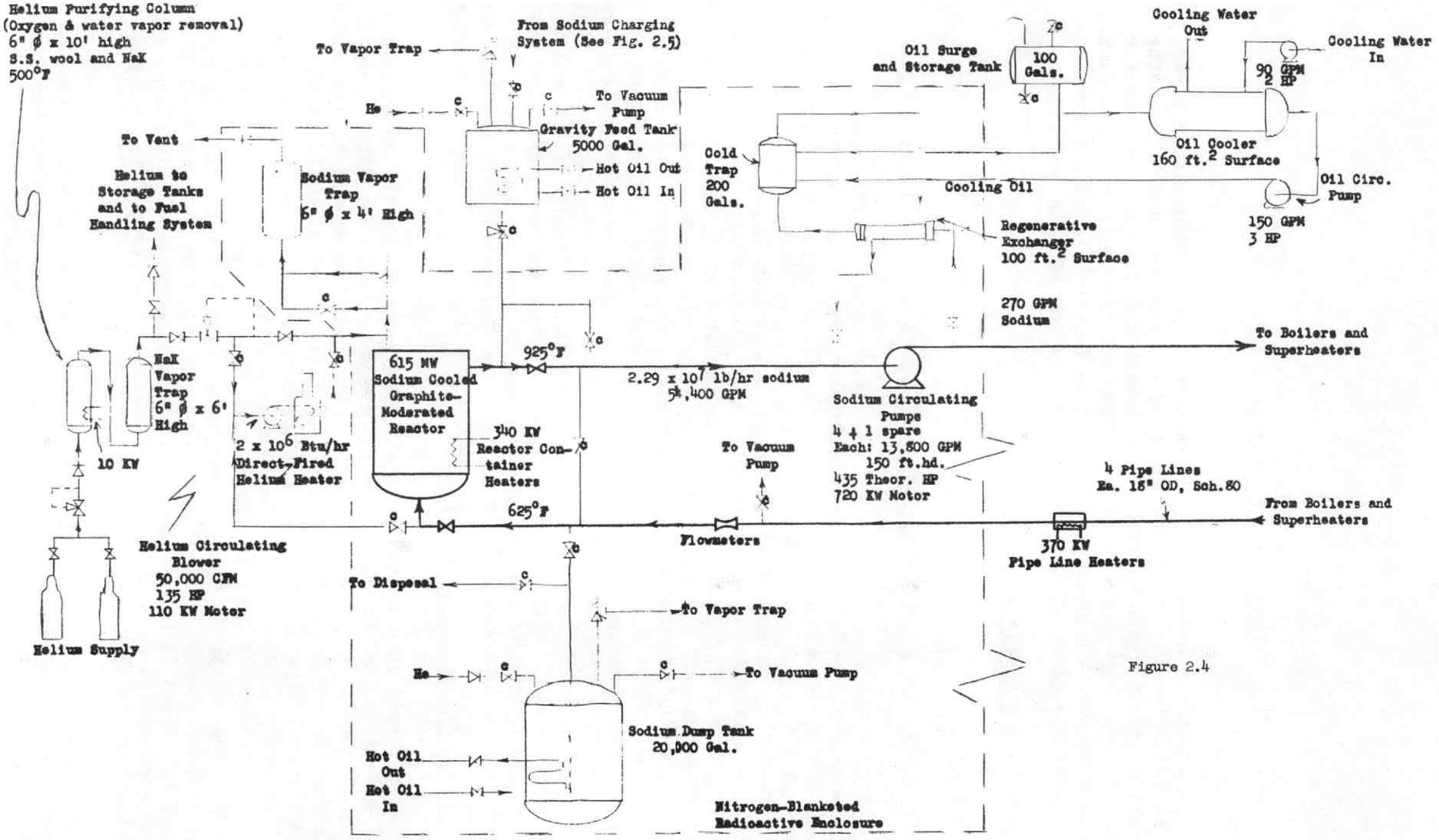


Figure 2.4

Notes: 1. All vessels and piping 304 S.S., insulated, traced and instrumented.
2. ⋈ denotes normally closed valve.
3. Total power plant sodium charge 110,000 lbs.

vided with hot oil coils. All vessels and piping are constructed of 304 S.S. and are insulated to minimize heat losses during operation.

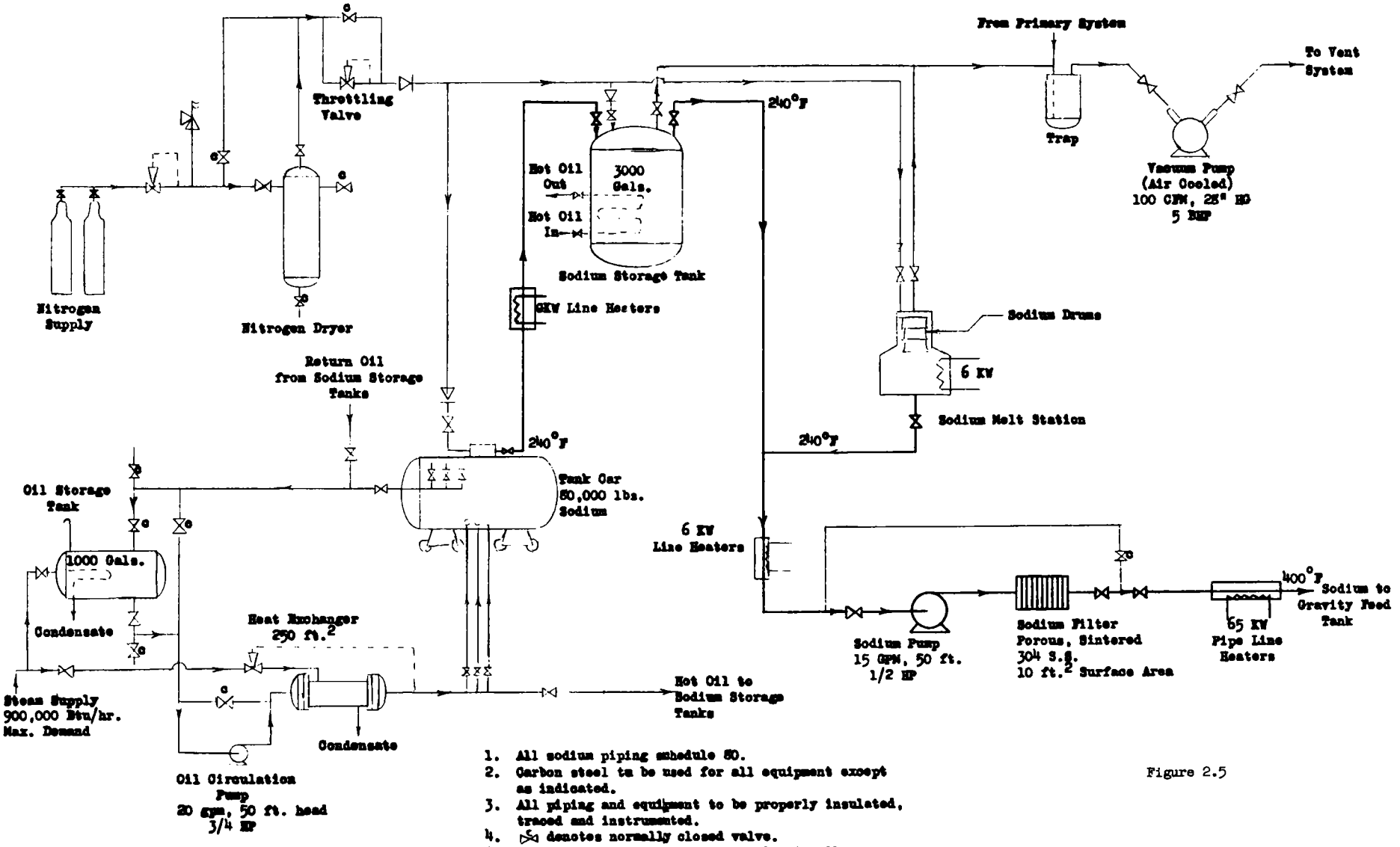
All free sodium surfaces in the system will be blanketed with oxygen-free helium. A helium purifying column, 6 in. ID and 10 ft. high, packed with stainless steel wool and filled with eutectic NaK is used for removal of oxygen. Incoming helium is bubbled up through the NaK and the column is electrically heated to 500°F to promote the reaction between NaK and any oxygen and water vapor in the helium. Supplementing the purifying column is a tower, 6 in. ID and 6 ft. high, designed to trap out any entrained NaK from the helium. This tower is packed with stainless steel wool, is partially filled with NaK, and operates at room temperature. In operation, the trap reduces the NaK content of the helium to that corresponding to the equilibrium vapor pressure at room temperature. A similar trap is required for the effluent gases from the reactor and other vessels which may contain sodium.

In addition to providing an inert atmosphere, helium will be used during startup as a means of preheating the reactor core. A direct fired heater, rated at 2×10^6 Btu/hr., is used to heat the helium and a blower is used to circulate the helium through the core. During startup, sodium is by-passed around the reactor and is preheated with the pipe line heaters. The reactor shell is also preheated at this time. When all items are brought to the desired temperature, the sodium by-pass valves are closed and sodium valving to and from the reactor is opened permitting sodium to enter the reactor core. The sodium displaces the helium used to preheat the core and the helium is vented to the stack until the sodium reaches the level of the exit piping. The reactor is then ready for operation.

All equipment in the reactor complex containing radioactive sodium will be within a nitrogen-blanketed enclosure, indicated by the dashed line on Figure 2.4.

Sodium Charging System: Figure 2.5 is a process flowsheet for the sodium charging system. The initial sodium charge is supplied from tank cars. A hot oil circulating system is used to melt the sodium in the tank cars (and to provide heat to the sodium dump tank and gravity transfer tank — Figure 2.4. Steam is used to heat the oil by means of coils in the oil storage tank and a 250 $ft.^2$ heat exchanger. A 100 CFM vacuum pump is used to evacuate equipment before adding sodium, and dry nitrogen is used for blanketing the system and to provide motive force to molten sodium. A sodium melt station designed to melt sodium directly from 55 gallon drums is installed to supply makeup sodium. A 6 KW heater is used at the melt station.

During the charging operations, sodium from the tank car is displaced with nitrogen to a 3000 gallon holdup tank where it is held for four hours to allow settling out of oxides. The sodium is then pumped through a sintered, 304 stainless steel filter and a 64 KW pipe line heater and into the reactor system. Carbon steel may be used as the material of construction for a major part of the sodium charging system equipment since the temperatures are low.



1. All sodium piping schedule 80.
2. Carbon steel to be used for all equipment except as indicated.
3. All piping and equipment to be properly insulated, traced and instrumented.
4. ⋈ denotes normally closed valve.
5. Total power plant sodium charge 110,000 lbs.

Figure 2.5

Fuel Handling and Waste Disposal System: Operations illustrated on the flowsheet shown in Figure 2.6 are described in detail below.

Fuel Handling System

An important factor in the cost of power will be fuel reprocessing. For greater economy, a continuous fuel cycle will probably be more desirable than a once-through operation. If we assume an average burnup of 7000 MWD/ton of fuel, the elements will require replacement at an average rate of one every 22 hours. Obviously, the ability to refuel the reactor during operation will be desirable.

Fuel Handling Operations: The ability to refuel while the reactor is running requires remotely-operated equipment. Provision must be made for the following operations:

1. Irradiated fuel elements must be removed from the reactor under an inert atmosphere by means of a fuel handling mechanism.

2. The removed fuel elements must be cooled until the heat generation within the elements has fallen off appreciably. This will take about 30 minutes from the time the element is withdrawn from the neutron flux. During this period, an external cooling system must be attached to the fuel handling mechanism.

3. The irradiated fuel elements then must be transferred to a cleaning station where any adhering sodium can be flushed off.

4. The flushing agents (steam and water) plus removed sodium must be collected and stored until the Na^{24} activity has decayed to safe levels.

5. Cleaned, irradiated fuel elements must be stored until fission products activity and heat generation has been reduced to a safe level.

6. After storage, the spent fuel elements must be transferred to a loading station for shipment to the fuel reprocessing plant.

7. New fuel elements must be stored prior to insertion in the reactor.

Fuel Element Design: As presently conceived, the fuel elements will have hanger rods which are permanently attached to the top of the elements. These rods will extend upward to a stepped plug which is locked in

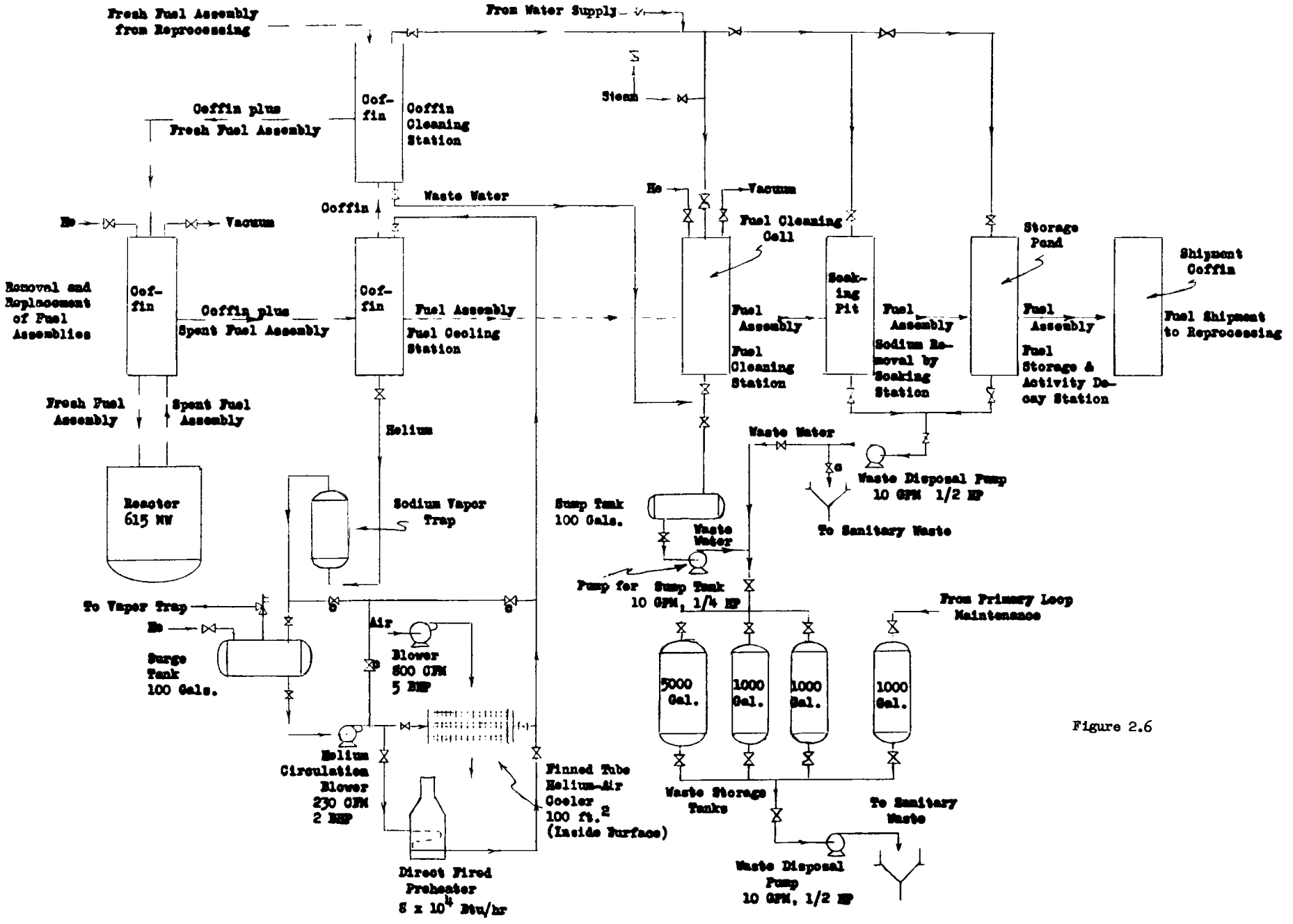

Fresh Fuel Assembly from Reprocessing
From Water Supply
Steam
Coffin plus Fresh Fuel Assembly
Coffin
Coffin Cleaning Station
Waste Water
He
Vacuum
Removal and Replacement of Fuel Assemblies
Coffin plus Spent Fuel Assembly
Fuel Assembly
Fuel Cooling Station
Fuel Cleaning Cell
Fuel Cleaning Station
Soaking Pit
Sodium Removal by Soaking Station
Storage Pond
Fuel Storage & Activity Decay Station
Shipment Coffin
Fuel Shipment to Reprocessing
Fresh Fuel Assembly
Spent Fuel Assembly
Helium
Reactor 615 MW
Sodium Vapor Trap
Sump Tank 100 Gals.
Waste Disposal Pump 10 GPM 1/2 HP
To Sanitary Waste
Pump for Sump Tank 10 GPM, 1/4 HP
To Vapor Trap
Surge Tank 100 Gals.
Air
Blower 800 CFM 5 HP
From Primary Loop Maintenance
5000 Gal.
1000 Gal.
1000 Gal.
1000 Gal.
Helium Circulation Blower 230 CFM 2 HP
Finned Tube Helium-Air Cooler 100 ft.2 (Inside Surface)
Waste Storage Tanks
To Sanitary Waste
Direct Fired Preheater 8 x 10^4 Btu/hr
Waste Disposal Pump 10 GPM, 1/2 HP

Figure 2.6

place in the top section of the shield. This design is similar to designs proposed by North American Aviation and Monsanto and will permit ready removal of the fuel elements. Guide tubes which extend from the top of the core to the bottom of the cover plate can be used to simplify the loading and unloading operations.

Fuel Handling Procedure: A study of North American Aviation progress reports and the reports of the Monsanto group suggests a fuel handling procedure as follows:

1. The fuel handling coffin is positioned over a fuel element to be removed by means of alignment devices in the reactor shield. An overhead bridge crane is used to move the fuel handling coffin.

2. A seal is established between the lower section of the fuel handling coffin and the reactor shield. (The fuel handling coffin will be made in two sections: a long upper section, which will contain the fuel elements and which is under a helium atmosphere, and a short lower section which is used for making and breaking seals with the reactor shield and other apparatus.)

3. The seal chamber is evacuated and a helium atmosphere is established in the lower section of the fuel handling coffin. The helium pressure established in the seal chamber will be slightly higher than that in the reactor or in the upper section of the fuel handling coffin. This is to prevent leaks out of the reactor.

4. A seal door connecting the two sections of the fuel handling coffin is opened and the fuel handling coffin is now ready to receive the fuel element.

5. Through the use of remote controls, a cable is lowered and attached to the shield plug connected to the fuel element. The attachment may be by means of a mechanical grapple or with a magnetic clutch.

6. A winch is started and the fuel element is lifted from the reactor core into the coffin by means of the shield plug and hanger rod.

7. A fresh fuel element, with a hanger rod and shield plug attached, which was transferred to a separate compartment of the fuel handling coffin prior to Steps 1 to 6, above, is now lowered into the reactor core and the shield plug is locked in place.

8. The seal door between the two sections of the fuel handling coffin is closed and the seal at the reactor shield is broken.

9. The fuel handling coffin is transferred to the fuel element cooling station, where the heat generated in the fuel element is removed by circulating cooled helium through the coffin for at least 30 minutes. (Further details on the cooling operation are given in the section below on heat removal from an irradiated fuel element.)

10. When the irradiated fuel element has cooled, the fuel handling coffin is moved to the next station and the spent fuel element is transferred from the coffin to a portable fuel cleaning cell.

11. After removal of the fuel element, the fuel handling coffin is cleaned and prepared to receive a fresh fuel element for the next cycle. Although a replacement rate of one element every 22 hours has been assumed above, it is understood that this is an average rate and that some elements may have to be removed as frequently as once every 11 hours. However, sufficient leeway is still present, as it is estimated that Steps 1 to 11, above, can be completed within five hours.

Fuel Cleaning Cell and Fuel Handling Coffin: The sodium adhering to the fuel element is flushed off with steam and water from spray nozzles in the portable cleaning cell. For the Sodium Reactor Experiment, North American Aviation proposes a cleaning cell 24 ft.-9 in. high consisting of a 5 ft.-10 in. section of 5 in. pipe, a 16 ft.-10 in. section of 3 in. pipe and a 2 ft.-1 in. section of 1-1/2 in. pipe. The cell is fitted with spray nozzles and is valved for steam,water, helium,vacuum, and drainage. A small carriage mounted on wheels supports the cell in a water-filled canal and is used to transfer the cell to wall-mounted brackets that position the cell over a drain connection in a storage pond. Snap rings are used to attach the service lines to the cell and the sodium is washed off the element by water-spraying. Following the sodium removal, the cleaning cell is moved to a fuel removal station where the fuel element is supported by an overhead crane and the cleaning cell lowered into a deep well. The fuel element is then placed in a water soaking pit, held there for about 12 hours, and finally transferred to a water-filled storage basin where it is held for loading in a shipping coffin. Water used to clean the fuel element of sodium is collected and held for eventual disposal when radioactivity has sufficiently decayed.

A fuel cleaning cell similar to the North American Aviation design described above may be used. However, although the diameters of the fuel element bundles for the two designs are of the same order, there is an appreciable difference in the element lengths. The fuel elements of the Sodium

Reactor Experiment are 7 feet-8 inches long and the total length of the slugs plus guide fins, hanger rods, and upper shield plug is 22 feet-3 inches. For the proposed design, the active length of uranium slugs will be 17 feet. Therefore, the total length of elements plus hanger rod plus upper shield plug will be about 35 to 40 feet, and the length of the cleaning cell will be of the same order.

Similarly, the dimensions of the fuel handling coffin for the Sodium Reactor Experiment are 4-3/4 feet in diameter at the base, 2 feet in diameter at the top and 29-1/2 feet in height. The diameters of the fuel handling coffin for the proposed design can be of the same order, but the height may be 50 feet or more. Corresponding to the rather large heights of the cleaning cell and handling coffin will be the sizes of the overhead cranes, the reactor enclosure, the fuel storage pond and other pieces of fuel handling equipment.

Careful study to minimize costs and to reduce complexities of operation will be required on all these pieces of equipment.

Alternate Designs: Because of the inconvenience involved and large equipment required to handle a fuel element-hanger rod combination, it may be desirable to use a coupling device between the element and the rod so that they may be separated when the fuel element is transferred to the fuel handling coffin. While a fuel element is being removed from the reactor, the hanger rod is disconnected and then may either be connected to the fresh element prior to charging, or may be discarded. In either case, the activity of the hanger rod must be considered in the development of equipment and handling techniques.

The necessity of removing and replacing fuel elements while the reactor circuit is on-stream creates many problems in addition to those attendant to the complex, remotely-operated fuel handling operations. Each of the 900-odd fuel assemblies has connected to it a plug in the reactor shield which must be removed and replaced during refueling. These plugs, and the recesses they fit into, must be designed so that there is no leakage of helium or sodium vapor from the reactor, and so that removal and replacement of the plugs by remotely-operated mechanisms from the fuel handling coffin may be accomplished in a short time.

The requirement of rapid and easily manipulated plug-securing gadgetry probably obviates the possibility of obtaining a seal that is impervious to leakage. It may be possible, however, to construct a compartment that is attached and sealed to the top of the reactor and which contains helium at a pressure higher than that of the vapor in the reactor. Helium, therefore, could only leak into the reactor, and the seals on the plugs would be required only to minimize gas seepage. A port in this compartment, which would allow the fuel handling coffin to enter and leave, would likewise be required only to minimize leakage, since helium leaking through it would not have originated in the reactor. This compartment, of course, must be high enough to accommodate the 50 ft. high coffin, in addition to the cranes and positioning devices.

Heat Removal from an Irradiated Fuel Element: An irradiated fuel element will continue to generate heat after being discharged from the reactor. This heat may be removed by circulating cooled helium through the fuel handling coffin. Data given in the North American Aviation report NAA-SR-260 show that the power level of a fuel element one minute after removal from the core is 3.7 per cent of operating power and 1.1 per cent of operating power 30 minutes after removal. Over a 30 minute period, the average power level of a fuel element is about 2.5 per cent of operating power. For the proposed design, the power of the central element is 800 KW and the average heat generated by a fuel element in the fuel handling coffin will be 20 KW (68,300 Btu/hr.).

The helium used for removing this heat from the fuel handling coffin must have an outlet temperature less than 621°F (the melting point of the lead shield on the coffin). In addition, the inside wall temperature at any point in the helium loop must be greater than 208°F (the freezing point of sodium) to avoid plugging. A flow rate of 270 lbs./hr. of helium at an inlet temperature of 300°F and an exit temperature of 500°F will be adequate for this cooling requirement.

Radioactive Sodium Waste Disposal System

A system for disposing of waste radioactive sodium is indicated in Figure 2.6. When fuel element assemblies are removed from the reactor, sodium coolant clinging to the fuel rods and their appurtenances is removed by spraying with steam and water. Small amounts of sodium remaining after the spraying operation are removed in the soaking pit. The coffin used in removing and transporting the fuel rod assembly is also cleansed of any sodium that may have contaminated it.

It is proposed to pipe these wash waters containing sodium (as NaOH) to storage tanks and hold them there until the radiation has decayed to allowable limits, at which time the water will be dumped into the sanitary waste water system. Neutralization is unnecessary, since the solution is quite dilute.

The amount of sodium clinging to the fuel element was estimated by ratioing the amount experienced at KAPL for the SIR fuel rod assembly on a fuel assembly surface area basis. The 7-rod fuel cluster would, therefore, retain 216 gms. of sodium. It was further assumed that 50 gallons of steam plus water are used to wash and rinse the fuel assembly. This waste water, therefore, has a concentration of 1.15×10^{-3} gm. Na/ml soln. The radiation intensity of the solution t hours after removal from the reactor is approximately

$$0.00615\, e^{-.0468\, t}\ \frac{\text{curies}}{\text{ml soln.}}$$

According to the Radiological Health Handbook of the U. S. Public Health Service, the maximum permissible contamination in water for continuous exposure for Na^{24} is 8×10^{-3} microcuries/ml. This tolerance was used as a basis for the effluent of the waste disposal system. To meet this tolerance, the waste solution must be retained in storage tanks for 12 days before being dumped.

Since a fuel bundle will be removed about once per day, two-1000 gallon tanks suffice as the storage volume. Thirteen days' wash water would be piped to a tank and held there for 12 days after the last day's batch was introduced, and then would be dumped into the sanitary sewage system. The two tanks would be filled and discharged in succession. One additional 1000 gallon tank is to be provided to handle sodium-contaminated water resulting from the periodic maintenance and repair of the primary loop. Contingencies for storing particularly active solutions are provided for by a 5000 gallon storage tank. Periodic effluent from the soaking pit and storage pond may be sent either to holdup tanks or directly to sanitary waste, depending on its activity. The above equipment for the radioactive waste disposal is shown in Figure 2.6.

The waste disposal system is predicated on Na^{24} contamination only. If there is activity in the wash water due to the corrosion products formed by the action of sodium on the zirconium and stainless steel, or due to the impurities in the sodium itself, then the manner of waste handling and disposal would be different. It has been fairly well established that the corrosion of both zirconium and stainless steel by sodium consists of the reaction of the base metal with the oxide dissolved in the sodium, with the formation of an oxide film on the metal. The tenacity of this film, and, therefore, the amount of corrosion products in the sodium, cannot be predicted without additional experimental data.

Miscellaneous Process Equipment

By-Pass Cold Trap: A cold trap is installed to remove sodium oxide from sodium to reduce corrosion. The inlet sodium to the cold trap is initially cooled in a regenerative heat exchanger by the exit sodium from the trap. Further cooling of the inlet sodium is accomplished in the crystallizing tank by circulating a heat transfer oil through cooling coils. In addition to the crystallizing tank, which should have a volume of about 200 gallons, the following equipment will be required for operation of the cold trap.

1. A 100 gallon oil storage and surge tank.

2. An oil to water heat exchanger of about 160 $ft.^2$ heat transfer surface area.

3. A 90 gpm water circulating pump.

4. A 150 gpm oil circulating pump.

5. Necessary piping and fittings for the water and oil circuits, including cooling coils in the crystallising tank.

6. A sodium to sodium regenerative heat exchanger of about 100 ft.^{2} heat transfer surface area.

Cleaning and Leak Testing Equipment: During the initial startup of the reactor, equipment will be required for cleaning and leak testing the auxiliary circuits. Hot water will be used for an initial flushing of equipment to remove foreign matter. Following the hot water flushing, a detergent solution will be used to remove grease. The equipment will then be dried by circulating hot nitrogen. Finally, the equipment is leak tested with a helium mass spectrometer. Much of the equipment to be used during the normal operation of the reactor complex can also be used for cleaning and leak testing purposes.

Table 2.6

Heating Requirements for Sodium-Graphite Reactor Installation

	Electrical KW Installed	Burners MM Btu/Hr.
Reactor Cooling System	380	3.16
Sodium Charging System	83	**
Fuel Handling System	---	0.08
Cleaning and Leak Testing System	---	3
	463	6.24

** 0.9 MM Btu/hr. maximum steam heating demand for hot oil system.

Table 2.7

Pumping Power Requirements for Sodium-Graphite Reactor Installation

	Brake Horsepower	
	Item	Totals
Reactor Coolant System		
Main Sodium Pumps (at 50% eff.)	3480	
Helium Blower	135	
Cold Trap Pumps	5	3620
Sodium Charging System		
Sodium Pump	0.5	
Oil Circulating Pump	0.75	
Vacuum Pump	5.0	7
Fuel Handling System		
Helium Blower	2	
Air Blower	5	
Water Pump	0.5	
Sump Pump	0.5	8
Cleaning and Leak Testing		
Vacuum Pump	5	
Water Circulation Pumps	10	15
Total		3650

Appendix to Chapter II*

Steam Cycle

The steam plant contains a boiler and superheater, a reheater between a high-pressure turbine and an intermediate-pressure turbine coupled together and turning a generator, a separate generator run by a low-pressure turbine exhausting to the condenser, and five feedwater heaters using steam extracted from various points in the turbines. The steam pressure at the inlet to the high-pressure turbine was set at 1250 psig, the condenser pressure was taken to be 1.5 inches of mercury absolute, and the heat rate for the plant was calculated for six cases with various combinations of superheated steam temperature and reheat steam temperature. Size and cost data for the reheater, the boiler and the superheater for typical conditions were supplied by the Griscom-Russel Company.

Table 2.8 summarizes the steam cycle data for the six cases. Figure 2.7 is a heat balance flow sheet for a typical case.

Table 2.9 gives size and cost data for the main heat exchangers.

* From information supplied by Duquesne Light Company.

DUQUESNE LIGHT COMPANY
Atomic Power Development Department

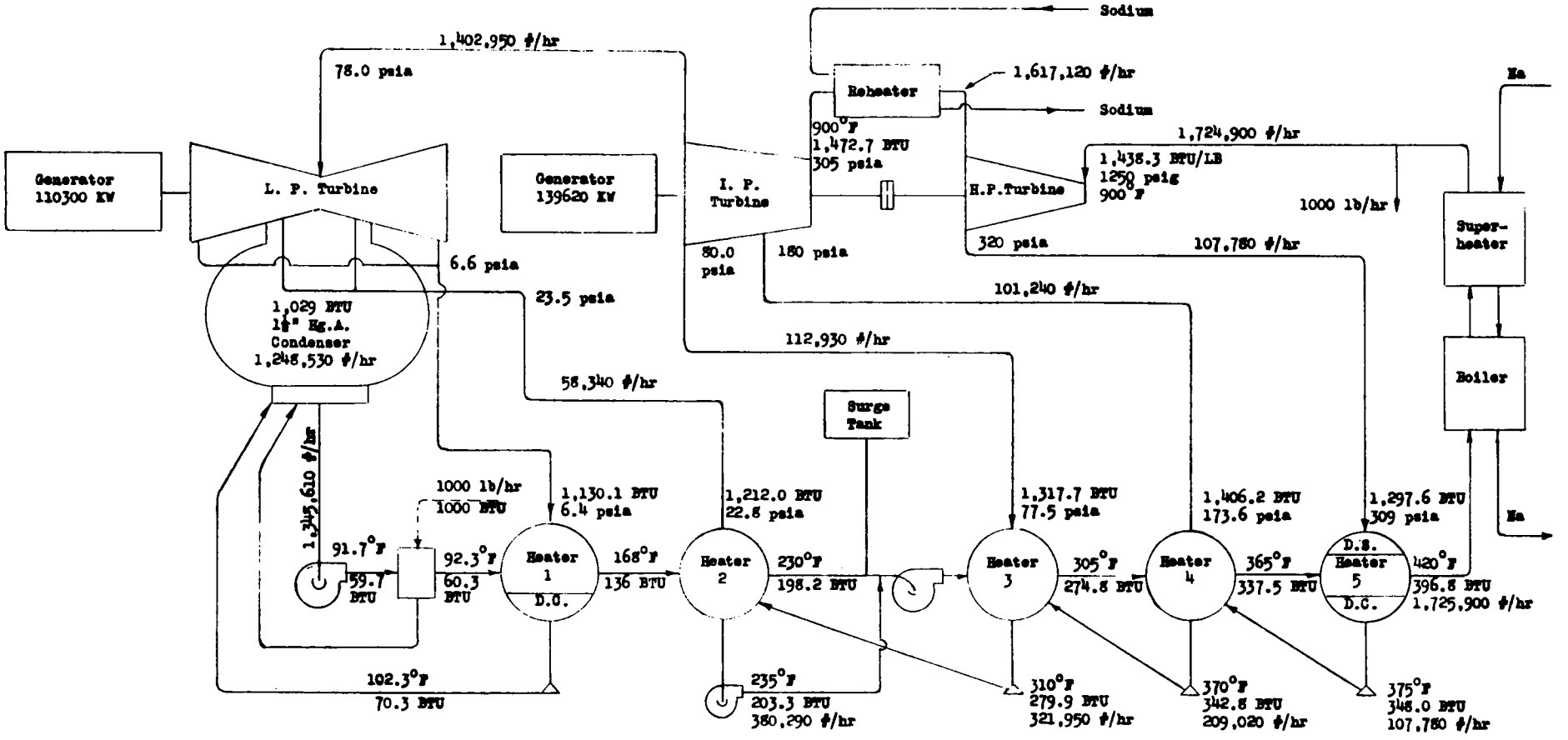
Table 2.8

Duquesne-Kidde Study
Steam Cycle Data

Case		I	II	III	IV	V	VI
Turbine Capacity	KW	250,000	250,000	250,000	250,000	250,000	250,000
Turbine Heat Rate	BTU/KWHR	8,346	8,386	8,406	8,431	8,480	8,555
Turbine Conditions							
Pressure	PSIG	1,250	1,250	1,250	1,250	1,250	1,250
Initial Temperature	°F	900	900	900	850	850	800
Reheat Temperature	°F	900	850	800	850	800	800
Back Pressure	Hg	1½"	1½"	1½"	1½"	1½"	1½"
Flow Data	1000 LB/HR						
Steam Flow to Turbine	"	1,724.9	1,769.1	1,816.3	1,797.7	1,846.7	1,878.1
Steam Flow to Reheater	"	1,617.1	1,658.6	1,702.8	1,681.6	1,728.4	1,754.6
Steam Flow to Condenser	"	1,248.5	1,274.3	1,303.6	1,292.4	1,322.8	1,343.0
* Hotwell Flow	"	1,345.6	1,374.7	1,407.4	1,394.3	1,428.4	1,450.3
Extracted Steam Flow (1) (L.P.)	"	96.1	99.4	102.8	100.8	104.6	106.3
(2)	"	58.3	58.1	62.2	58.4	62.9	62.2
(3)	"	112.9	120.6	123.1	122.1	125.1	128.1
(4)	"	101.2	106.2	111.1	107.9	113.0	114.9
(5) (H.P.)	"	107.8	110.5	113.4	115.2	118.3	123.6
Air Ejector & Gland Steam (Est.)	"	1.0	1.0	1.0	1.0	1.0	1.0
Boiler Feedwater Flow	"	1,725.9	1,770.1	1,817.3	1,798.7	1,847.7	1,878.9
Enthalpy	BTU/LB						
Steam to Turbine	"	1,438.3	1,438.3	1,438.3	1,407.7	1,407.7	1,376.0
Steam to Reheater	"	1,297.6	1,297.6	1,297.6	1,274.3	1,274.3	1,249.8
Steam from Reheater	"	1,472.7	1,446.5	1,420.4	1,446.5	1,420.4	1,420.4
Steam at No. 1 Extraction Point (L.P.)	"	1,130.1	1,117.5	1,104.0	1,117.5	1,104.0	1,104.0
Steam at No. 2 Extraction Point	"	1,212.0	1,196.2	1,181.4	1,196.2	1,181.4	1,181.4
Steam at No. 3 Extraction Point	"	1,317.7	1,297.2	1,278.4	1,297.2	1,278.4	1,278.4
Steam at No. 4 Extraction Point	"	1,406.2	1,383.2	1,362.4	1,383.2	1,362.4	1,362.4
Steam at No. 5 Extraction Point (H.P.)	"	1,297.6	1,297.6	1,297.6	1,274.3	1,274.3	1,249.8
Condensate Leaving Hotwell	"	59.7	59.7	59.7	59.7	59.7	59.7
Feedwater Leaving No. 5 Heater	"	397	397	397	397	397	397
Moisture Content of Exhaust Steam	%	6.9	8.2	9.5	8.2	9.5	9.5
Extraction Stage Pressures	PSIA						
Heater No. 1	"	6.4	6.4	6.4	6.4	6.4	6.4
Heater No. 2	"	22.8	22.8	22.8	22.8	22.8	22.8
Heater No. 3	"	77.5	77.5	77.5	77.5	77.5	77.5
Heater No. 4	"	173.6	173.6	173.6	173.6	173.6	173.6
Heater No. 5	"	309	309	309	309	309	309
Heat Flow	10^6 BTU/HR						
Boiler Heat to Turbine	"	1,802.7	1,848.9	1,891.6	1,817.3	1,866.8	1,839.9
Reheat to Turbine	"	283.1	246.9	209.1	289.7	252.5	299.3
Total Heat	"	2,085.8	2,095.8	2,100.7	2,107.0	2,119.3	2,139.2
Turbine Efficiency	%	40.8	40.7	40.5	40.4	40.2	39.9
Reactor Data							
Na Temperature Leaving Reactor	°F	925	925	925	925	925	925
Na Temperature Entering Reactor	°F	625	625	625	625	625	625
**Na Flow to Boiler	1000 #/HR	20,080	20,600	21,090	20,290	20,800	20,480
**Na Flow to Reheater	1000 #/HR	3,160	2,860	2,245	3,230	2,820	3,340
**Total Reactor Heat	10^6 BTU/HR	2,148	2,158.8	2,164	2,170	2,182.8	2,202

* Drains from No. 1 Heater Flow to Condenser.

** These values are 3% higher than Turbine Heat to allow for losses.



Gross Generation - 249,920 KW Heat Rate - 8,346 BTU/KWHR

Turbine Cycle Steam Conditions
Pressure - 1250 psig
Initial Temperature 900°F
Reheat Temperature 900°F

Figure 2.7

GEM & RBO
Steam Plant Conditions
Case I

10R-32

Table 2.9

Heat Exchanger Data[1]

	Reheater		Steam Generator		Superheater	
Steam Temperatures:						
Initial	850°F	800°F	850°F	800°F	850°F	800°F
Reheat	800°F	800°F	800°F	800°F	800°F	800°F
Cost	$400,000	$350,000	$575,000	$575,000	$200,000	$150,000
Surface Sq. Ft.	71,600	61,200	19,200	19,148	7,900	4,500
Shell Diameter, I.D.	68"	68"	60"	60"	33"	33"
Effective Tube Length	Finned 388"	Finned 331"	268"	270"	110"	64"
No. of Shells	1	1	4	4	4	4
No. Tubes per Shell	1,230	1,250	1,100	1,085	1,100	1,085
Passes	Single	Single	Single	Single	Single	Single
Tube Material	Bimetallic Carbon Outer 304 SS Inner		304 SS	304 SS	304 SS	304 SS
Shell Material	Carbon Moly.	Carbon Moly.	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel

(1) Calculated and cost estimated by the Griscom-Russel Company.

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Chapter III

FAST BREEDERS

Fast breeders (and fast converters, which are closely related to them) are a very broad subject, and it is not easy to give a consistent story, particularly since those most active in the fast reactor field are themselves pursuing different routes for different reasons. However, it is possible to discern some considerable areas of agreement.

The initial impetus behind breeder reactors was a consideration of the relation between total fissionable material supply and the world's total power needs. A study of the latter shows that only by using the bulk of the fertile material, notably U-238, can the long-term power demands of the world be met after the year 2000. While this motivation might not be very important to the present generation, the possibility of extending and expanding our fissionable material supply is advantageous to breeder reactors because of government interest in weapons and in long-range planning. Thus, breeder reactors have received, and will receive, very considerable government support.

The idea of a breeder reactor leads to consideration of fast reactors for the following well-known reason: plutonium 239, in addition to its thermal fission cross-section,has a large thermal capture resonance. When plutonium fissions, it produces about 3 neutrons. However, in the thermal region the competing process of neutron capture reduces the average number of neutrons produced per absorption in plutonium below 2. This makes it impossible to breed plutonium 239 in a thermal reactor. In reacting plutonium 239 with fast neutrons, the probability of capture is much less, and the average number of neutrons per absorption in plutonium 239 is considerably above 2 neutrons. Breeding with plutonium 239 is feasible in fast reactors.

Thus, consideration of the total requirements of fissionable material, plus the peculiar properties of plutonium 239, which is the fissionable daughter of the bulk of the world's fertile material, U-238, has stimulated interest in fast breeder reactors.

In Table 3.1 are assembled some data on fast converters and breeders which have been recently discussed. In this table, a plant investment of $225/KW has been taken for the Dow-Detroit reactor. This is not an estimate; it is a stand-in for the correct number and may ultimately be lower or higher.

The table does not show remarkable price advantages for breeder reactors at the present stage of technology (that is, with the reactor costs $225/KW and up). It is hard to escape the conclusion that the enthusiasm

for breeder reactors in the near future is based on the expectation of operating them as converters and obtaining a bonus by charging more for plutonium than the price of the U-235 which they burn. Nevertheless, the long-range potential of the breeding system should not be overlooked.

It will be observed from this table, as indeed would be evident a priori, that power costs for fast reactors are composed of the same items as power costs for thermal reactors. These are, respectively, capital charges on investment, labor, maintenance and supplies, fuel rental charges, fuel make-up charges, chemical processing and fabrication, and a credit for the value of the spent fuel. The principal advantage which the designers of fast breeder reactors claim is that, unlike thermal reactors, where the value of the spent fuel is always less than the value of the original fuel, in breeder reactors the value of the product is greater than the initial value.

Looking at this more closely, the annual cost of burning nuclear fuel, aside from chemical processing, is a rental charge plus (or minus) the net loss (or gain) in value of the fuel over a year's cycle. Fast reactors tend to have a high fuel inventory, and thus, high annual rental charges, since these are proportional to the value of the inventory. If a reasonable rental rate is 4%, then the fuel must increase in value at least 4% per year to counterbalance this charge. If the rental rate were higher, a proportionately higher turnover rate would offset the increased charges, assuming the breeding gain was constant. These considerations are independent of the assumed price of plutonium. Thus, fast reactor designers are pushed to very high neutron and thermal fluxes. This in turn requires unusually resistant fuel elements, and is the present main technical obstacle to the production of power from fast breeders. However, should this difficulty be overcome (and one has every reason to expect that with the gradual progress of technology it will be overcome) breeder reactors, only assuming that there is a market for plutonium which expands at the government-established rental rate, will show a cost advantage over thermal converters with identical $/KW investment because of the cancellation of all fuel charges except chemical processing.

The question of the $/KW cost of the reactor and power system will now be considered. Fast reactors may have solid fuel or liquid fuel. The fluid fuel fast reactors are difficult to design because of the required high concentration of fuel in the solution. There are few solvents (iron, nickel, chromium or fused chlorides are some of these) which can contain appreciable amounts of uranium or plutonium in solution. The melting points of the resulting solutions are quite high, and the fluids will probably be difficult to handle and very corrosive. More thought has been given to solid fuel reactors cooled with liquid metal. At high energies, the cross-sections for neutron absorption become much more regular than in the thermal region. The choice of structural materials for fast reactors is therefore less limited. Stainless steel appears to be entirely suitable as a material for construction. Likewise, there is no need for materials of low atomic weight to moderate the neutrons. A fast reactor is usually very dense, and this results in a small reactor vessel.

It is only fair to set beside these possible advantages some possible disadvantages. The principal one is that solid fuel fast reactors must contain exceedingly fine-grained fuel elements, such as a large number of pins of extremely small diameter. The control of fast reactors may be more difficult than thermal reactors, because of the lack of selective absorbers of extremely high cross-section. There are also serious control questions brought about by the short neutron lifetime, the temperature coefficient, etc.

All of these considerations show that the design problem for fast reactors is quite different from that of thermal reactors, and one may hope — and the designers of fast reactors do hope — to build reactors at least as inexpensively as thermal reactors of the same power output. A similar consideration applies to the potential advantage which breeder reactors have of reducing fuel costs. It might turn out to be completely chimerical to hope to operate a reactor at sufficiently high flux and sufficiently high breeding gains to offset the high inventory charges in fast reactors.

Table 3.1

Power Costs for Fast Converters and Breeders

80% Service Factor

Designer	Bechtel-Pacific	California Research & Development	Dow-Detroit	Dow-Detroit
Type Fuel	Converter U in Pin Type Elements	Breeder Molten Pu-U-Ni	Converter U-Cr Solid	Breeder Pu-U Solid
Reactor Heat, MW	500	500	600	600
Electrical Output, KW	145,300	173,000	190,000	190,000
Conversion Ratio, Gross	1.38	1.39	1.25	1.25
Conversion Ratio, Net	.95	1.25	1.16	1.16
Investment, $/KW				
Fuel(1)	81.8	120	48	74
Plant	351	269	225(4)	225(4)
Charges, mills/KWH				
Capital Charges on Plant at 15%/yr.	7.52	5.76	4.8	4.8
Labor, Maintenance and Supplies	0.80	0.77	0.71	0.71
Sub-Total	8.32	6.53	5.51	5.51
Fuel Costs:				
Rental at 4%/yr.	0.47	0.68	0.27	0.42
(at 10%/yr.)	(1.17)	(1.71)	(0.68)	(1.06)
Fabrication	1.55	0.07	0.53	0.53
Chemical Processing	0.47	0.56	0.53	0.51
Makeup	3.68	0.03(3)	3.19	——
Sub-Total	6.17	1.34	4.52	1.46
	(6.87)	(2.37)	(4.93)	(2.10)
Total	14.49	7.87	10.03	6.97
	(15.19)	(8.90)	(10.44)	(7.61)
Credit from Sale of Plutonium(2), mills/KWH	3.06	0.50	3.43	0.48
Cost of Power at Busbar, mills/KWH	11.43	7.37	6.60	6.49
	(12.13)	(8.40)	(7.01)	(7.13)

(1) Uranium fuel costs based on U_3O_8 at 15 $/lb. and 50 $/kg separation work ($22.50/gm). Plutonium feed costs taken to be the same.

(2) Selling price — $22.50/gm. (3) Blanket losses in processing. (4) Not an estimate.