APDA-102

Subject Category: REACTORS-RESEARCH AND POWER

UNITED STATES ATOMIC ENERGY COMMISSION

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INFORMATION REPORT BY ATOMIC POWER DEVELOPMENT ASSOCIATES COVERING WORK FOR THE PERIOD AUGUST 1, 1954 TO JANUARY 31, 1955

January 31, 1955

Atomic Power Development Associates
Detroit, Michigan

Technical Information Service, Oak Ridge, Tennessee

UNCLASSIFIED

APDA - 102

Price $ 0.55

Available from the
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Department of Commerce
Washington 25, D. C.

CLASSIFICATION CANCELLED
DATE 1-7-57
For The Atomic Energy Commission
H. F. Carroll
Chief, Declassification Branch

INFORMATION REPORT

BY

ATOMIC POWER DEVELOPMENT ASSOCIATES

January 31, 1955

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PREFACE

This report covers the work of the Atomic Power Development Associates for the period August 1, 1954 to January 31, 1955. Two previous research and development reports have been issued by the project; the first, Report DCDE-100, reviewed a prior report made to the Commission on December 1951 and described the objectives, the problems, and the progress made by the Project to December 1, 1953; the second, Report DCDE-101, covered the Project's activities for the period December 1, 1953, to July 31, 1954.

The present report is divided into 12 general sections, with the material herein being reported along APDA organization lines.

In addition to the work reported here, a considerable portion of the APDA Working Group's time has been spent in a consolidation of available information into a reactor design capable of immediate construction. The design that has evolved has been described in another report for consideration by the member companies of APDA, and others, as a basis for a reactor construction project. Neither the nature nor the details of that report are discussed here.

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SUMMARY

ADMINISTRATION

The name of the Project has been changed to "Atomic Power Development Associates," and the number of associates is being increased to a total of 33 companies.

A budget and work program for 1955 was prepared and approved. This budget, totaling $3,815,000, has been arranged to cover the cost of work on a long-range, high-performance reactor, as well as to maintain the position of being prepared to build the best possible experimental reactor in the near future.

PHYSICS

Day-to-day problems involving shielding design, criticality studies, activation calculations, radiation heating analysis, and reactor control were studied. Projects that are more long-range in nature, including safety studies, multigroup reactor calculations, and complete reactor shield design were started.

A fast reactor control and safety meeting was held in Detroit. Dr. H. A. Bethe presided at this meeting, and about 45 engineers and scientists interested in this phase of fast reactor technology attended. A report of the presentations and discussions will be issued.

MATERIALS

Preliminary irradiation tests have been run on certain of the fuel element alloys of interest to the project. The results indicated that a programmatic research investigation to determine optimum methods of melting, coating, and fabricating these alloys would be necessary and that other alloy systems should be investigated. Such a program has been established.

A significant amount of work has been completed in evaluating container and cladding materials in molten sodium and NaK. Fusion bonding of metals at point contact in sodium and NaK environments also has been studied.

FUEL ELEMENTS

Designs of composite, sodium-bonded elements have been produced and analyzed, and development work on the fabrication of fuel pieces has progressed. Encouraging results have been experienced on the integrally-cast type of fuel elements, circular pin-type elements, and in plate-type elements.

As part of the fuel fabrication development, effort has been aimed at obtaining techniques for producing fuel pieces with fine-grained, randomly-oriented, metallographic structure. Heat treatment has been studied as a means of obtaining these conditions for certain alloys of uranium; results to date are encouraging.

One reactor core arrangement under study utilizes a larger number of smaller elements. In general, the smaller size eases some of the difficult fabrication problems.

BLANKET DESIGN

A breeder blanket has evolved which appears to satisfy most of the basic design requirements. This design is acceptable as a neutron reflector and absorber; it can transfer all of the heat generated within it to flowing coolant without undue obstruction to this fluid flow and without producing unacceptably high-temperature areas; its structural strength appears great enough to allow it to remain in the reactor without failure of any of its parts for periods measured in years. Furthermore, its fertile material is in a form that appears best for aqueous reprocessing. This final consideration is important at present but may be less restricting as alternate processing techniques are developed.

MECHANICAL DESIGN OF REACTOR VESSEL

Reactor vessel outline drawings have been made for a design with coolant series-down-flow through the core section of the reactor and parallel-up-flow in the blanket. Preliminary work has started on the design of a reactor vessel for coolant parallel-up-flow through both core and blanket sections. Several methods of supporting, holding, positioning, and aligning fuel and blanket elements have been investigated.

MECHANICAL HANDLING EQUIPMENT

Two arrangements of fuel loading and unloading equipment have been designed and studied from the standpoint of cost reduction and over-all simplification. In evaluating the two designs referred to as the two-plug and single-plug concepts, such items as engineering complexity and dependence on or interference with other systems were considered. As a result of these comparisons, the single-plug concept has been selected for further study. The single-plug arrangement will be cheaper to build and offers more in the way of simplicity and reliability.

Basic engineering studies of the remainder of the fuel-element transfer system have reached the point where similar detailed conceptual studies are necessary. As these studies are made, detail designs of components for the balance of the system may be started.

INSTRUMENTATION AND CONTROLS

Studies have been carried out on both a hydraulic control actuator and an electro-mechanical control actuator. A model of the hydraulic actuator has been designed and is under construction. Tests, which will begin soon, have been designed to study the operation of such a control drive employing liquid sodium as an actuating fluid.

Control element designs have been made and are now being evaluated; in each design concept, an attempt is made to achieve the ultimate in simplicity and safe operation.

An analysis has been made of the transient coolant-flow distribution in a reactor during a scram. From this analysis, scram speeds have been determined.

LIQUID METAL AND STEAM-POWER SYSTEMS

Liquid metal and steam-power systems have been revised to reflect current design considerations. Preliminary drafts of intermediate heat exchanger specifications, valve specifications, and steam generator units have been completed. A full-length, once-through, liquid metal heated boiler test has been designed and is being constructed. This test will provide performance data under normal operating conditions and under transient conditions. Preliminary testing will be done to determine the severity of the NaK-water reaction in this type of a unit.

FUEL PROCESSING

Preliminary engineering design and evaluation studies were made for three types of reprocessing plants: aqueous reprocessing, fluoride volatility reprocessing, and pyrometallurgical reprocessing. The results show pyrometallurgical reprocessing to be most promising.

A more extensive theoretical study, based on data from Argonne National Laboratory, was made on the specific pyrometallurgical process that appears to be the most feasible. Plans are now being made to perform tests and studies to demonstrate process feasibility.

A conceptual design of a fuel disassembly and shipping facility has been made.

SITE AND SAFETY

A tentative site on which to place a reactor power plant was chosen and various layouts of the arrangement of structures on the site were made. Reactor safety, site safety, and site development studies are in progress.

In considering the health physics instrumentation problems, a survey and evaluation of present commercial instruments has been made. This is the first step in the health physics study program.

TEST FACILITY

A decision was made in July, 1954, to construct a test facility that will be a full-scale prototype of the final design and layout of the reactor and one sodium loop. This non-radioactive unit is considered necessary for testing final components, coordinated assemblies, and for training personnel for the reactor plant.

Space at the Delray Station of The Detroit Edison Company has been obtained as the site for this facility; clearing this space is in progress, and service facilities are being provided. Tentative layouts of the equipment have been made, and the design of the temperature control system is under way. It is expected that the facility will be in operation in the summer of 1956.

ADMINISTRATION

ORGANIZATION

The name of the Project was changed from "The Dow Chemical-Detroit Edison and Associates Atomic Power Development Project" to "Atomic Power Development Associates." This change was made in October, 1954 when the Dow Chemical Company requested Atomic Energy Commission approval to withdraw from the Project. Concurrently, the Project requested approval to add the following nine companies to the group:

<u>Electric Power Systems</u>

Boston Edison Company
The Connecticut Light and Power Company
New England Gas and Electric Corporation
Central Hudson Gas and Electric Corporation
Long Island Lighting Company
New York State Electric and Gas Corporation
Wisconsin Power and Light Company

<u>Engineering Organizations</u>

Commonwealth Associates, Inc.
Jackson and Moreland

The internal organization of the Project is still governed by the "Memorandum of Understanding" dated Feb. 1, 1954. That memorandum sets up the objectives and general organization of the Project and outlines the relationship between the Project Companies. Recognizing the magnitude of the effort confronting the Project and the need to proceed rapidly on as broad a front as possible, a special committee was set up to study the reorganization of the Project on a basis that would be commensurate with the work required. It has been concluded, that for the future, work can be carried on with greater effectiveness by a more formal organization; therefore, a decision has been made to form a non-profit membership corporation under New York law.

The operation of Atomic Power Development Associates is directed by a Management and an Executive Committee. Members of these committees are presidents and vice presidents of several of the associated companies. Reporting to them are the Legal, Economic, Finance, Technical, and Public Information Committees, all of which are similarly composed of top executives from the associates. The responsibility for day-to-day management of the Project is vested in the Project Manager. Figure 1 shows the relationship between the committees, the Project Manager, and the Working Group.

PERSONNEL

The total number of personnel in the APDA Working Group is 54. This figure is broken down as follows:

Administrative and Clerical		11
Technical		40
Physics	5	
Fuel & Blanket Elements	9	
Mechanical Handling & Design	5	
Materials	2	
Control & Instrumentation	4	
Liquid Metal & Steam-Power	5	
Fuel Separations	4	
Test Facility	2	
Site & Safety	1	
Economics	3	
Engineers in Training		3
TOTAL		54

This breakdown does not include consultants, subcontractor personnel, or the members of the management, technical, and advisory committees; however, it does include one person who is attending the Oak Ridge School of Reactor Technology, two people who are working at Argonne National Laboratory under an Industrial Training and Assistance Agreement, one person stationed at Brookhaven National Laboratory who is on loan to the Babcock & Wilcox Company to assist in the feasibility study of the Liquid Metal Fuel Reactor, and one person who is attending the University of Michigan.

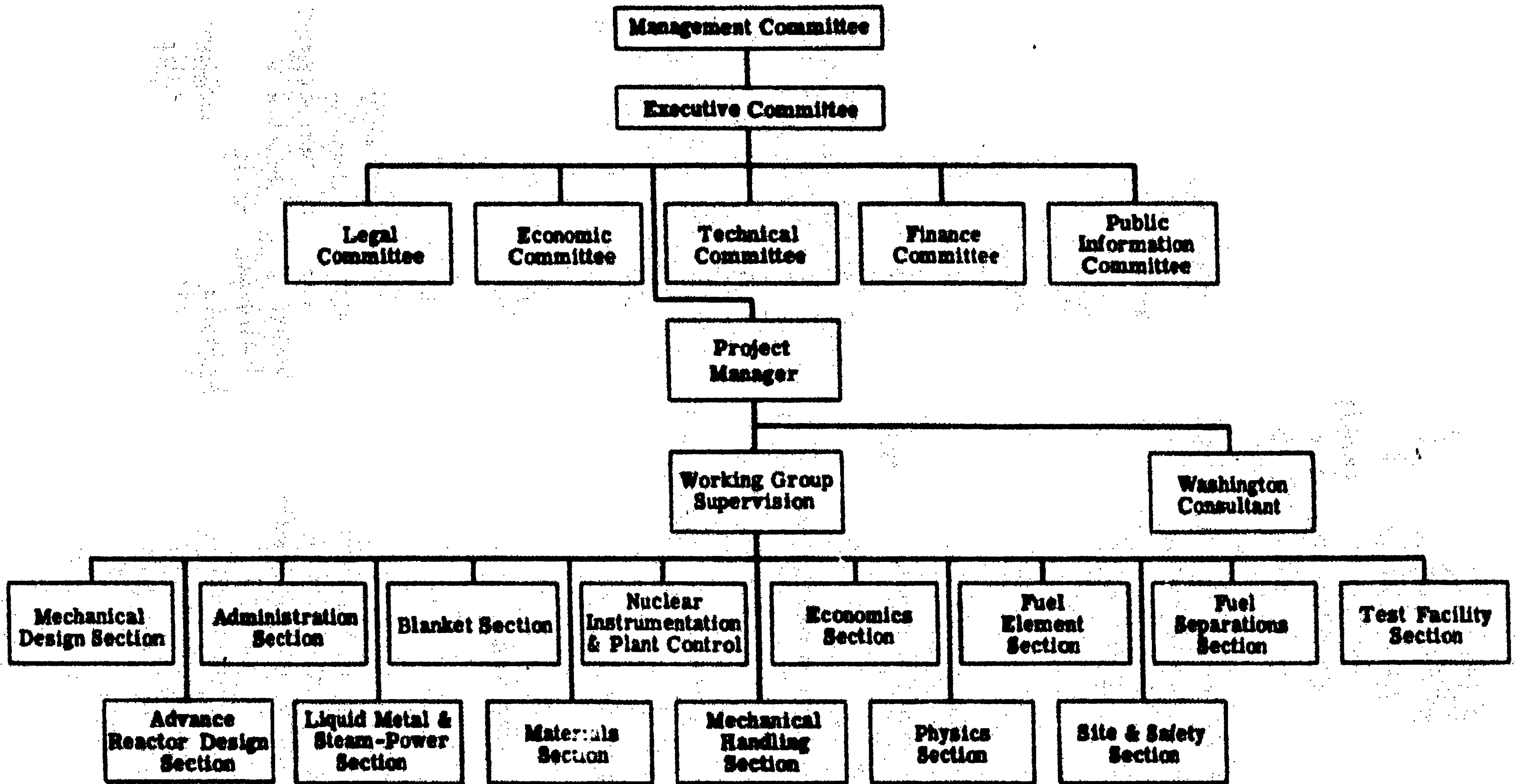


Fig. 1 - ATOMIC POWER DEVELOPMENT ASSOCIATES ORGANIZATION

PROJECT AGREEMENT WITH THE AEC

The agreement between the AEC and the Project, dated April 14, 1954, extends through January 31, 1955, and may be extended by mutual consent of both parties. A two-month extension of the agreement was arranged in January, 1955. Before the end of that relatively short extension, which terminates on March 31, 1955, the Associates plan to form the non-profit membership corporation mentioned above.

TECHNICAL PROGRAM

Except for the facility to permit testing of full-scale reactor components, there is no intention of starting a Project research and development facility. The research and development program for the fast breeder reactor design has been implemented by placing contracts with carefully selected companies, including some of the Project Company facilities. These APDA-supported programs are coordinated by the engineers in the Working Group who also make design studies and who follow pertinent research work being done under AEC sponsorship. Table I presents the relationships and activities of the Project associates and the Project contractors.

The main objective of APDA, as stated in the Memorandum of Understanding, is the development of atomic fuels to compete commercially with conventional fuels. To achieve this objective, research and development is necessary to complete a basic design for a high performance nuclear heat power reactor; analyses are required to determine whether such a reactor can be commercially competitive; components which might be required for such a reactor must be built and tested. The broad goal set early in 1954 was to be in a position to undertake a detailed design of a rather high performance fast reactor by early 1956. This first reactor might not be economic but would be an important step in that direction, with full expectation that later and improved versions would become competitive sources of power.

In order to provide some of the data required in several technical and engineering areas to advance the development of the liquid metal cooled, fast neutron, breeder reactor under study, a program estimated to cost $2,600,000 for 1954 was evolved. Of this amount, it was proposed that $2,300,000 would be borne by the associated companies, the remaining $300,000 representing work to be done for the Project by the Commission within the scope of the established Commission programs. Approximately $1,900,000 was spent by the Associates for the 1954 program, the immediate goal of which may be stated as the determination

	MAN POWER TO PROJECT WORKING GROUP	CASH TO PROJECT FUND	PAYMENT OF CONTRACT WORK	RESEARCH AND DEVELOPMENT	FUEL MATERIAL	FUEL FABRICATION	FUEL CORROSION TESTS	FUEL REPROCESSING	FUEL CONTAINER FABRICATION	TESTS OF LIQUID METAL COMPONENTS	FUEL CONTAINER CORROSION TESTS	REACTOR COMPARTMENT MECHANISMS	IRRADIATION TESTS	PLANT LAYOUT	REACTOR PHYSICS	OPERATION AND CONTROL	NUCLEAR PHYSICS	LEGAL	MANAGEMENT	REACTOR VESSEL DESIGN AND FABRICATION	PRIMARY SHIELD TANK DESIGN AND FABRICATION
PROJECT COMPANIES																					
ALLIS-CHALMERS MANUFACTURING COMPANY	X			X						X		X				X					
ATLANTIC CITY ELECTRIC COMPANY	X		X																		
THE BABCOCK & WILCOX COMPANY	X			X	X	X	X			X	X									X	X
BENDIX AVIATION CORPORATION	X			X											X	X	X				
*BOSTON EDISON COMPANY																					
*CENTRAL HUDSON GAS & ELECTRIC CORPORATION																					
THE CINCINNATI GAS & ELECTRIC COMPANY	X	X																			
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY	X	X	X																		
*COMMONWEALTH ASSOCIATES, INC.																					
*THE CONNECTICUT LIGHT & POWER COMPANY																					
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.	X		X																		
CONSOLIDATED GAS ELECTRIC LIGHT AND POWER COMPANY OF BALTIMORE	X	X																			
CONSUMERS POWER COMPANY	X	X	X																		
THE DETROIT EDISON COMPANY	X		X																X		
FORD MOTOR COMPANY				X		X															
GENERAL PUBLIC UTILITIES CORPORATION	X	X																			
THE HARTFORD ELECTRIC LIGHT COMPANY	X	X																			
*JACKSON & MORELAND																					
*LONG ISLAND LIGHTING COMPANY																					
NEW ENGLAND ELECTRIC SYSTEM	X	X	X																		
*NEW ENGLAND GAS AND ELECTRIC ASSOCIATION																					
*NEW YORK STATE ELECTRIC & GAS CORPORATION																					
NIAGARA MOHAWK POWER CORPORATION	X		X																		
PHILADELPHIA ELECTRIC COMPANY	X		X																		
POTOMAC ELECTRIC POWER COMPANY	X	X																			
PUBLIC SERVICE ELECTRIC AND GAS COMPANY	X	X																			
ROCHESTER GAS & ELECTRIC CORPORATION	X			X													X				
SOUTHERN SERVICES, INC.	X		X																		
THE TOLEDO EDISON COMPANY	X	X																			
UNITED ENGINEERS & CONSTRUCTORS, INC.	X																				
VITRO CORPORATION OF AMERICA	X																				
WISCONSIN ELECTRIC POWER COMPANY	X		X																		
*WISCONSIN POWER AND LIGHT COMPANY																					
*New Associates																					
PROJECT CONSULTANTS																					
BETHE, DR. HANS A.															X						
GOMBERG, DR. HENRY J.																X	X				
KING, PROFESSOR WM.																X					
MC QUILLEN, F. W.																		X			
PACKARD, J. E.																			X		
SMITH, PROFESSOR L. P.					X										X						
UPTON, A. E.																		X			
WHIPPLE, DR. G. HOYT																	X				
PROJECT CONTRACTORS																					
AMERICAN MACHINE & FOUNDRY CO.												X				X					
BATTELLE MEMORIAL INSTITUTE					X	X							X								
EFFICIENT ENGINEERING COMPANY, INC.												X									
FIRTH-STERLING, INC.					X	X															
GIFFELS & VALLET, INCORPORATED														X							
MINE SAFETY APPLIANCES CO.										X											
NUCLEAR DEVELOPMENT ASSOCIATES, INC.															X	X					
NUCLEAR METALS, INC.					X	X															
SMITH, HINCHMAN & GRYLLS, INC.												X									
SYLVANIA ELECTRIC PRODUCTS, INC.					X								X								
UNIVERSITY OF MICHIGAN								X													
AEC FACILITIES																					
ARGONNE NATIONAL LABORATORY					X			X					X								
IOWA STATE COLLEGE								X													
KNOLLS ATOMIC POWER LABORATORY													X								
LOS ALAMOS SCIENTIFIC LABORATORY					X																
MATERIALS TESTING REACTOR													X								
NORTH AMERICAN AVIATION, INC.								X													

TABLE 1
RELATION OF PROJECT ASSOCIATES AND CONTRACTORS

of the engineering feasibility and conceptual design of a reactor of the type under study by resolving certain major problems, such as the development of a satisfactory fuel element, the development of a rapid, economic system for processing irradiated fuel and blanket subassemblies, and the development of a design suitable for location in a populated area.

The major technical programs that were budgeted for 1954 are:

1. Experimental and analytical research and development of fuel elements, including fuel materials.

2. Development of a low cost fuel and blanket reprocessing system.

3. Development of reliable controls for a fast reactor.

4. Design and development of fuel and blanket handling equipment.

5. Design and development of the reactor proper.

The results of these studies are reported herein and in Report DCDE-101.

During the year, the Commission raised the possibility of building a reactor at an early date. In addition to the work reported here, a considerable portion of the Working Group's time and, to a lesser extent, of the Project contractors' time has been spent in a consolidation of information into a fast breeder reactor-electric plant design that is capable of immediate construction. This caused an acceleration of the technical program and has brought the Project to the point where it is in the position to effect a detailed design of a rather high-performance reactor at any time. The design that has evolved has been described in another report for consideration by the Associates, and others, as a basis for a reactor construction project. Neither the nature nor details of that report are discussed here.

1955 BUDGET AND OBJECTIVES

A budget and work program for 1955 was prepared and approved. This budget, totaling $3,815,000, has been prepared to cover the cost of work on a long-range, high-performance reactor, as well as to maintain the position of being prepared to build the best possible experimental reactor in the near future. The contemplated program was developed with the realization that to meet both these objectives, the tempo of the work would have to be stepped up considerably over that maintained during 1954. The activities planned include the continuation of the Working Group, the continuation of research and development contracts now in effect, the initiation of a number of new research and development contracts, and the beginning of a test facility to permit full-scale mock-up and the testing of reactor components.

A summary of the 1955 budget follows:

Controls	$ 350,000
Mechanical Handling	350,000
Site and Health Physics	95,000
Liquid Metal and Steam-Power System	195,000
Economics	45,000
Reactor - Structural	115,000
Fuel Element	520,000
Materials	330,000
Physics	280,000
Advance Design	130,000
Fuel Processing	400,000
Administration	305,000
	$3,115,000
Test Facility	700,000
	$3,815,000

The objectives of the 1955 program may be summarized as follows:

1. Establish designs for the principal items of mechanical equipment.
2. Make commitments to purchase prototypes of some of the principal components.
3. Have a component test facility well under way for testing the prototypes.
4. Establish one or more fuel elements capable of at least intermediate performance and adaptable to remote fabrication.
5. Establish a promising, if tentative, solution of the processing problem.
6. Arrange the activities of the Working Group so that the necessary emphasis can be given to an actual experimental breeder reactor project at any time.

MULTIGROUP CALCULATIONS

A series of criticality calculations have been performed to assist in the selection of an engineering design of a fast breeder reactor. Five-group diffusion theory equations for spherical geometry were used in the calculations. Results are given in Table II.

The general features of the design are:

a. Fissionable and fertile material are in the form of an alloy, either U-Cr or U-Zr.

b. The coolant is sodium.

c. The structural material is stainless steel.

d. The core is approximately cylindrical in shape and is surrounded by a breeding blanket.

e. Control may be accomplished by varying the nuclear properties of a region located between the core and the side blanket.

In calculations 1 through 4 and 12 through 19, the effect on critical mass and breeding gain of varying the core size and the volume percentages of uranium alloy and coolant has been investigated. The alloy in both core and blanket is uranium-chromium; the fuel is U-235.

A plutonium-fueled reactor is considered in calculation 5.

In calculation 6, the plutonium concentration in the blanket has been allowed to build up to 2% of the alloy atoms. In computing the conversion ratio, the burn-out of plutonium atoms has been included.

Control aspects have been considered in calculations 7, 8, and 9.

One scheme that has been considered for controlling the reactor is the motion of 3-region control rods that are located around the core periphery. The three regions are a Li-6 poison region, a void region, and an iron reflector region. The poison region is inserted for shutdown and scram, while the reflector region is positioned completely or partially adjacent to the core during normal operation. The primary function of the void region is to isolate the poison from the high flux region during normal operation so that efficient use of excess neutrons may be accomplished. The three control positions considered are the poison region full-in, the reflector region full-in, and an intermediate

Case Number	Spherical Dimensions (cm)			Composition (Volume Percent)										Atomic Ratio of U-238 to U-235 in Core	Conversion Ratio			Calculated Critical Mass (kg)	
				Core			Reflector				Blanket								
	Core Radius	Reflector Thickness	Blanket Thickness	Uranium Alloy	[illegible]	Coolant	[illegible]	Coolant	[illegible]	[illegible]	Uranium Alloy	[illegible]	Coolant		Core	Blanket	Total	U-235	Pu-239
1	30.5	0	35	33.4	16.4	50.2	-	-	-	-	59.8	9.9	30.3	2.3	0.218	1.077	1.294	734	-
2	40.0	"	"	"	"	"	-	-	-	-	"	"	"	1.4	0.338	0.986	1.319	330	-
3	50.0	"	"	"	"	"	-	-	-	-	"	"	"	1.65	0.162	[illegible]	[illegible]	[illegible]	-
4	21.5	"	"	"	"	"	-	-	-	-	"	"	"	0.8	0.0675	1.072	[illegible]	114	-
5	40.0	"	"	33.9	16.2	49.9	-	-	-	-	"	"	"	5.95	0.198	1.307	1.500	-	[illegible]
6	"	"	10	33.5	16.3	50.2	-	-	-	-	"	"	"	1.65	0.734	1.338	1.372	[illegible]	[illegible]
7	"	15	35	33.4	16.4	50.2	30	60	10	0	"	"	"	2.4	0.252	0.509	0.761	403	-
8	"	"	"	"	"	"	50	50	0	"	"	"	"	1.4	[illegible]	[illegible]	[illegible]	315	-
9	"	"	"	"	"	"	75	"	"	25	"	"	"	1.15	0.307	[illegible]	1.335	350	-
10	"	[illegible] 7.5 [illegible] 7.5	"	"	"	"	75	"	[illegible] 75% [illegible]	[illegible] 75% U-238	"	"	"	1.0	0.275	0.724	1.252	325	-
11	"	15	"	"	"	"	75	"	75% [illegible]	0	"	"	"	1.6	0.397	0.771	1.173	302	-
12	31.5	"	"	24.75	14.5	60.75	-	-	-	-	"	"	"	1.5	0.138	1.115	1.255	132	-
13	40.0	"	"	18.5	"	65.0	-	-	-	-	"	"	"	"	0.143	1.108	1.251	321	-
14	30.5	"	"	13.5	"	80.0	-	-	-	-	"	"	"	"	0.133	1.108	1.241	[illegible]	-
15	62.5	"	"	18.5	"	65.0	-	-	-	-	"	"	"	3.6	0.380	0.913	1.268	[illegible]	-
16	65.0	"	"	23.5	"	60.0	-	-	-	-	"	"	"	4.65	[illegible]	0.917	[illegible]	775	-
17	100.0	"	"	13.5	"	70.0	-	-	-	-	"	"	"	4.65	0.589	[illegible]	1.193	1610	-
18	35	"	"	33.4	16.4	50.2	-	-	-	-	"	"	"	0	0	[illegible]	[illegible]	74.5	-
19	85	"	"	43.5	16.5	40.0	-	-	-	-	"	"	"	9	0.764	0.414	1.178	1812	-
20	41.3	15	60	25.0	25.0	50.0	35	50	0	15	19.6	12.5	37.9	2.66	[illegible]	0.997	1.273	[illegible]	-
21	"	"	"	"	"	"	30	50	"	"	"	"	"	3.40	0.285	[illegible]	[illegible]	[illegible]	-

*No geometric corrections are included.

TABLE II - SPHERICAL MULTIGROUP CALCULATIONS reflector position with the void region also adjacent to the core.

The use of beryllium in the reflector region has been considered in calculations 10 and 11. A two-region reflector was used in calculation 10. The inner reflector contained 25% beryllium and the outer reflector 25% uranium. A one-region reflector containing 25% beryllium was used in calculation 11. Comparing beryllium and iron (calculations 8 and 11), it is seen that the use of beryllium results in a smaller critical mass but also a substantially smaller breeding gain. Relative to iron, beryllium lets fewer neutrons escape to the blanket and also degrades the neutron spectrum. The latter effect tends to increase α and decrease the relative number of U-238 fissions.

In calculations 1 - 19, uranium-chromium alloy has been used in both reflector and blanket.

In calculations 20 and 21, a preliminary engineering design has been considered in which a uranium-zirconium alloy was used. A comparison of the two calculations illustrates the disadvantage of having large amounts of iron in the reflector region: increasing the amount of iron tends to decrease the breeding ratio by decreasing neutron leakage to the blanket and by decreasing the number of U-238 fissions in the blanket through the degradation of the neutron spectrum.

REACTOR KINETIC BEHAVIOR FOR SAFETY STUDIES

An evaluation of the safety of a given reactor requires consideration of the possible behavior of the reactor and associated equipment under all conceivable circumstances. Of particular importance among these considerations is the stability of the reactor as provided by reactivity decreases caused by thermal expansion of the active lattice. This expansion operates independently of other control and safety devices to prevent runaway in the event that an accidental increase in reactivity occurs. If the reactivity is increased sufficiently to make the reactor prompt critical, the neutron density, and hence the power level, will increase exponentially and very rapidly unless or until the reactivity is reduced by thermal expansion or some other means.

Analyses were conducted to determine whether the rapid expansion of solid fuel elements during a prompt critical accident can stabilize a fast reactor of the type under study by the Project and to determine certain characteristics of the reactor during such an incident.

The reactor model analyzed was a right circular cylinder having a solid fuel alloy in the form of rods continuous over the height of the core.

The rods were separated by coolant channels; hence, the assumption was made that during an accident no accumulative expansion displacements were propagated radially through the core. Further, the heat generated during the short times under consideration was assumed to remain in the fuel rods. The stabilizing mechanism was, therefore, entirely the longitudinal expansion of the fuel rods. Inertial forces and elastic deformations of the rods were taken into account. Analyses were made for various linear rates of introduction of excess reactivity.

The results show that the stabilizing mechanism of fuel rod expansion will take the reactor off a fast period in the event of a prompt critical accident if the reactivity insertion rate is less than \$50 per second. If the total excess reactivity inserted is less than \$1.50, the accident will be controlled by thermal expansion alone. For higher rates of insertion, the inertial forces are large, and it appears that the ability of the mechanism to stop an accident is limited by inelastic deformations occurring in the fuel rods.

For amounts of reactivity larger than \$1.50 but added at a rate of less than \$50 per second, the power rise is sufficiently damped for a long enough time to conceivably allow a suitable fuse or control device to operate before appreciable damage to the reactor has occurred.

MEETING ON FAST-REACTOR CONTROL AND SAFETY

On November 10 and 11, 1954, a meeting was held at Detroit for the mutual interchange of information and ideas on fast-reactor control and safety. This meeting was attended by approximately 45 engineers and scientists who are working on fast reactor technology, with Dr. Hans A. Bethe of Cornell University presiding. A transcript of the meeting has been prepared; following final editing, it will be issued as an APDA report.

The general feeling of those attending the meeting seemed to be that fast reactors can be designed to be safe, predictable, and readily controlled. The principal fears of accident lie in areas that are common to all reactors, such as (1) loss of coolant or coolant flow causing meltdown and (2) start-up accidents resulting from prompt criticality at power levels so low that period measurement is in large error.

MATERIALS

With the natural growth of the Project, problems associated with behavior of materials have become more numerous and complex. It became apparent that a staff having specialized experience in a wide range of reactor materials would be necessary in order to provide consultation and assistance and to establish programmatic research investigations of materials problems as necessary. Accordingly, a Materials Section was established in the Working Group.

CORE MATERIALS

Programmatic research to develop core material having acceptable mechanical, physical, and nuclear properties, including minimum dimensional growth resulting from neutron exposure, has been underway for a considerable period of time. Programs have been established at Battelle Memorial Institute and at Sylvania Electric Products Company. Other programs of interest to APDA have been conducted at the various AEC National Laboratories; the results of those programs have been made available to APDA and the work has been followed closely.

Core alloys of promise are binary systems of chromium, molybdenum, and zirconium with uranium. Battelle Memorial Institute has investigated chromium-uranium alloys rather thoroughly, and pertinent data are given below.

Effect of Allotropic Transformations

Thermal cycling tests to determine the effects of allotropic transformations have been run on alloys of uranium containing 4-1/2, 5, and 5-1/2% wt chromium. The 5% chromium represents the eutectic composition, and the others are slight deviations from the eutectic. The eutectic alloy freezes at a constant temperature as a single phase of gamma uranium and chromium. Each of these compositions was cycled 30, 60 and 250 times within the following ranges: between 300F and 1022F within the alpha phase, between 300F and 1275F through the alpha-beta transformation region, and between 300F and 1450F from the alpha through the gamma region.

Results of the alpha phase tests indicated that dimensional and thermal stability were maintained as long as both the eutectic and non-eutectic materials were not subjected to allotropic transformations.

The 5% chromium eutectic was severely distorted when transformed repeatedly between the alpha and beta phases (300 - 1275 F). The surfaces of the samples wrinkled slightly during the first 30 cycles; additional wrinkling took place and the samples showed evidence of slight bumping after another 30 cycles; the material was severely distorted, bent and bumped after a total of 250 cycles. The 4-1/2 and 5-1/2% non-eutectic compositions apparently enhanced the dimensional and thermal stability of the alloy as the material was cycled across the alpha-beta transformation regions. The distortion in the non-eutectic alloys was retarded and did not begin until after a 60 thermal cycles, and the over-all distortion was not as severe.

On thermal cycling from the alpha region through the beta phase and into the gamma phase and back again (300 - 1450 - 300 F), the chromium alloys showed little distortion or length changes; however, the materials showed evidences of fine cracks after 60 cycles and were severely cracked after 250 cycles. Again, the eutectic 5% chromium material was more severely cracked than the 4-1/2 and 5-1/2% non-eutectic materials.

The results of these thermal cycling tests indicate that the eutectic chromium is more susceptible than non-eutectic compositions to distortion as a result of repeated allotropic transformations. As long as no transformations take place (materials are kept within the alpha region or below 1220 F for these compositions) the materials are inherently stable during thermal cycling.

Grain Refinement

Laboratory experience has suggested that small-grained microstructures are desirable in metallic nuclear fuels. In accordance with existing metallurgical observations on metals, an effort was made to minimize the grain size of cast chromium-uranium specimens by two methods: small ternary alloy additions and heat treatment.

Of the eight elements used as grain growth inhibitors, molybdenum and possibly niobium gave the finest grain size. Various heat treatments were explored; chill casting produced a fine grain size but also produced undesirable "hairline" cracking because of the severity of quenching. Less drastic heat treating methods showed greater promise. For example, cooling the eutectic alloy from the gamma phase to a temperature in the high alpha phase permits the solid state allotropic transformation to occur isothermally with a minimum of thermal stresses.

Alloys Produced by Powder Metallurgy

An alloy development program, utilizing powder metallurgy techniques and processes, has been established at Sylvania Electric Products Company. An inherent advantage of this process is that a fine-grained material is produced with a minimum of preferred orientation. Theory and limited experience has indicated that material prepared by this process should be stable when exposed to neutron radiation. Also, powder-metallurgy-produced material has shown some improved thermal cycling performance.

This investigation has been largely confined to a 1.2 to 1.4% wt Mo-U alloy and to a 2% wt Nb-U alloy. Because the niobium did not impart a sufficient improvement in its alloy characteristics, further study on this alloy system was curtailed. Experimental data on the molybdenum alloy investigated confirmed improved resistance to grain growth and thermal cycling distortion; therefore, it is planned to investigate additional alloys in which the molybdenum content is increased to 3-1/2 weight percent. Since the effect of beta transformations are minimized, this alloy will also be evaluated for its behavior under irradiation.

EFFECTS OF IRRADIATION

It is recognized that radiation-stable fuel, blanket, and control materials are required before a reactor core can be judged acceptable. The dimensional stability of reactor materials in nuclear radiation fields is of prime importance. The effects of excessive fission gas generation, effects of fission element build-up and the over-all changes in physical and mechanical properties also must be determined.

In view of the unsatisfactory results obtained on the first group of fuel materials irradiated, reported in DCDE-101, development effort has been directed toward producing small-sized, randomly-oriented grain structures. Heat treatments and alloying additions have been studied as a means of realizing these conditions before additional radiation damage tests are performed. Promising combinations of alloys and fabrication procedures have been evaluated and are awaiting tests under nuclear irradiation.

MECHANICAL PROPERTIES OF STRUCTURAL MATERIALS

Since no reliable data are available on the short-time mechanical properties of stainless steels in the temperature range 400 F to 700 F, it is felt that an investigation of the mechanical properties of types 304 and 347 stainless steels in this range would be of importance. Specimens have been prepared at the Babcock & Wilcox Research Center to determine values of proportional limit, yield strength, and Young's modulus at these temperatures. Fatigue properties, both mechanical and thermal, are also under investigation for these materials.

CORROSION TESTS

Mass Transport Tests

Tests have been completed on the counterflow, single-tube, tube-and-shell heat exchanger operated at the Mine Safety Appliances Company laboratories at Callery, Pennsylvania. The high heat flux test loop facility and its operation with high velocity sodium and NaK have been described in the previous progress report (DCDE-101, pages 25-27). A summary of the test results is given immediately below.

Thirty-Day Test on Zirconium Tubing - Weight and dimensional determinations showed no appreciable gain or loss of weight or change in wall thickness. Analysis of the sodium and NaK cold trap contents indicated that the amount of zirconium tube material lost to the liquid metals during the 30-day test was negligible. A metallic deposit, found on the outer or NaK surfaces, was analyzed as having a 75% iron content. The deposit is believed to have been caused by mass transport due to the coexistance of a temperature differential and a metal (type 316 stainless) having temperature-dependent solubilities in NaK, rather than by any characteristic peculiar to the zirconium tube.

Metallographic examination of the zirconium tube showed a hydride-like structure at the grain boundaries, penetrating to as much as half the thickness of the tube from the sodium contacting side. Examination under polarized light showed that recrystallization had taken place in all sections of the tube.

Thirty-Day Test on Modified Globeiron - The composition of the modified Globeiron was:

Element	Percent by Weight
Carbon	0.029
Chromium	0.90
Molybdenus	0.54
Maganese	0.26
Iron	98.27 (by difference)

Weight and dimensional determinations, made on the specimen tube before and after exposure, showed that, within experimental accuracy, there was no appreciable change in weight or wall thickness. Cold trap analysis showed that there was no appreciable increase in iron or chromium content.

A metallic deposit, similar in location, appearance, and composition to that found on the zirconium specimen tube, gave support to the theory that the deposit formation was caused by mass transport taking place in the loop piping. Metals that were soluble in NaK went into solution at high temperatures (~ 1200F found in the furnace coil and in the hot leg of the loop) and deposited where ever cooling took place.

The flowmeter section was particularly loaded with the mass transport products. This section was bridged across, in the direction of the magnetic field, with approximately one-third the cross-sectional area of the 1-inch pipe occupied by the metallic mass. It is believed that the suspended metallic particles were caught in the magnetic field of the flowmeter, and the build-up was gradual during the entire life of the loop.

Short-period Test on Titanium - Two separate tests on commercially pure titanium were completed. The first titanium specimen tube failed after only a few days exposure in the test loop. A visual examination of the damaged area, which was located at the hot end of the exposed section of the tube, suggested the possibility that the failure was caused by imperfections in the tube material. Another test on titanium was conducted to check the results. In this second test, a specimen tube, which was carefully checked for imperfections and found to have none, also failed after a few days exposure in the loop.

At the conclusion of each test, the sodium and NaK cold traps were found to contain large amounts of titanium. Metallographic examination of the first tube showed that the material in the damaged area of the tube had become embrittled as a result of oxide migration along the grain boundaries.

It was theorized from the observations, examinations, and analyses that a combination of corrosion and erosion were, in the most part, responsible for the tube failures. The corrosion took place when the titanium, which is a good oxygen "getter", absorbed the minute amounts of oxygen from the sodium in the NaK. The oxide on the surface of the tube afforded a protective coating against further corrosion for a larger portion of the tube; however, at the hot end of the tube, where the NaK turbulence was great, the titanium oxide was continuously eroded away. The process continued until the tube wall became so thin that it could no longer withstand the sodium pressure on the inner surface, and the tube ruptured. Material embrittlement, caused by oxide migration along the grain boundaries, contributed to the failure.

Dynamic Corrosion Testing

Additional work was conducted to investigate dynamic corrosion and mass transport of various structural materials under operating conditions which simulate an actual reactor power plant.

The Babcock & Wilcox Research Center conducted tests in which [illegible] was subjected to the following conditions for 1000 hours:

Velocity	30 ft/sec
Heat flux (hot specimen)	665,000 Btu/(hr)(sq ft)
(cold specimen)	100,000 Btu/(hr)(sq ft)
Hot sodium temperature	1,067 F
Cold sodium temperature	450 F

Analysis of the test results indicated that the dynamic corrosion and mass transfer of the low carbon iron was very minor and almost undetectable except for the minor effect of decarburization.

This program is being expanded toward the evaluation of various low carbon, low alloy steels and a stainless steel. These materials include composition such as a 1/2% chromium - 0.15% maximum carbon; 2-1/4% chromium - 0.01% maximum carbon; 2-1/4% chromium - 0.15% maximum carbon; and sensitized type 304 stainless steel. The tests would be run for 1,000-hour periods for at least a year.

Static Corrosion Testing

Three separate tests on the following materials were conducted at The Babcock & Wilcox Company Research Center to determine the corrosion resistance of various structural materials to liquid sodium at a temperature of 1100 F and to determine whether the materials are satisfactory for use in a reactor power plant:

1. Stainless steel type 440 C.
2. 18-8 (types 304, 310, 316, 318, 347) stainless steels.
3. Nickel alloys, such as A-Nickel, L-Nickel and Inconel.

4. Zircaloy - 2.
5. Carbon steel (SAE 1010).
6. Globeiron. (extra low carbon iron)
7. Modified Globeiron (extra low carbon plus small additions of Cr and Mo).

The basic apparatus used for these tests was a modified enclosed rotor, liquid-metal pump. Provisions were made for both rotating and stationary specimens, the rotating specimens being attached to the pump shaft and driven with a velocity of 55 fps at the tips. The stationary specimens were located on the pump casing.

Results of these tests are as follows:

1. The 300 series stainless steels had very low weight gains as a result of the exposure to sodium at 1100 F. It is felt that this weight gain is a function of the amount of oxygen present in the sodium and could be reduced by further purification of the sodium used.

2. There is no detrimental effect on corrosion resistance of type 304 stainless steel due to welding and/or sensitizing, and the corrosion rates obtained were even less than those experienced with unwelded type 304 stainless steel.

3. All of the carbon steels showed a loss in weight during the test, and this loss in weight was primarily caused by decarburization of the material. Since the Globeiron specimens contained far less carbon than the SAE 1010 steel, their weight loss was smaller.

4. The chromium-containing materials showed relatively low corrosion rates. It appears that these materials may be used for sodium systems (in place of the 300 series stainless steels) in so far as their corrosion resistance is concerned. The reaction of these materials to sodium was found to be similar to that of both the carbon steels and the 300 series stainless steels: they lost weight through decarburization and had a tendency to gain weight by the formation of chromium oxide.

5. The nickel-base materials do not appear to be as suitable for use in sodium systems as the other materials tested because they are susceptible to preferential depletion of minor constituents.

6. The limited data with Zircaloy-2 is not sufficient to draw any final conclusions, but it appears that this material was far more seriously attacked than other generally acceptable structural materials.

SELF-WELDING IN SODIUM

The effect of sodium, at temperatures between 500 and 1,000 F, on the self-welding or fusion characteristics of two metals forced together is under investigation at the Nuclear Power Test Laboratory of the Allis-Chalmers Manufacturing Company. To date, a pilot unit has been used to test type 347 stainless steel against itself.

Test equipment has been constructed to include ten test units. The pressure loading is accomplished by a double lever arm arrangement which has an over-all mechanical advantage of 78.5 to 1. The test program that is now underway has been limited to cover the following materials, since they are expected to be used in the reactor:

1. Type 347 stainless steel vs type 347 stainless steel.
2. Stellite-6 vs Stellite-1.
3. Type 347 stainless steel vs Inconel-X.
4. Colmonoy-6 or Colmonoy-7 vs Stellite-6.
5. Inconel-X (fully hardened) vs Inconel-X.
6. Low carbon iron (Armco) vs Stellite-6.
7. Chromium plated type 347 stainless steel vs low carbon iron (Armco).
8. Kennametal-150A carbide vs Carboloy-44A carbide or Carboloy-907 carbide.
9. V2B (precipitation hardening) stainless steel vs SAE52100 bearing steel.
10. Ampco-18 aluminum bronze alloy vs Stellite-3 or Stellite-6.

VALVE MATERIAL TEST

A test to determine the effect of prolonged operation in sodium of a piston-type Johnson valve was performed at the Nuclear Power Test Laboratory of the Allis-Chalmers Manufacturing Company. The operation of the valve involved sliding surfaces of a 4-3/16-inch piston, hard-faced with Stellite #6, against the base of a type 347 stainless steel valve body.

Operation was effected in a closed circuit by pressurizing the underside of the valve piston and venting the chamber above the piston to atmosphere. Reversing the procedure caused the piston to move in the opposite direction. The piston stroke was approximately 1-13/16 inches. Valve construction was such that the piston traveled through a 1-inch axial length of a completely surrounded cylinder with a 0.005-inch clearance, and the remainder traveled through a vaned area of six equally spaced vanes 5/16 inch wide. The piston was free to rotate while moving axially.

A summary of valve operations is given in Table III. The valve was immersed for 696-1/2 hours in hot sodium during which time it was operated smoothly for 219-1/3 hours through 7,263 cycles. The working surfaces showed no scratches -- only light scrub marks around the bore; the metal was merely discolored by a combination of heat and sodium. A measurement of the valve piston with a micrometer showed that its dimensions were the same as before the test started.

TABLE III

Summary of Operations of Johnson Valve in Hot Sodium

Temperature - F	Operating Time Hours:Minutes	Cycles at Temperature
500	23:57	1034
550	23:31	931
600	19:02	1209
650	14:44	774
700	52:17	2088
750	23:37	406
800	21:18	205
900	8:43	131
1000	32:12	485
-	Test Discontinued	-

FUEL ELEMENTS

For the past six months, the fuel element design and development program has proceeded along the lines indicated in DCDE-101. Designs of composite, sodium-bonded elements have been produced and analyzed, and development work on the fabrication of fuel pieces has been continued. Progress has been made in the development of integrally cast type of elements.

A circular pin-type fuel element design, patterned after the ANL design for EBR-II, has been added to the plate and radiator types. Although the pin-type element does not allow a thermal performance as good as the other designs, it is felt that work at Commission facilities has indicated solutions to the problems of fabricating satisfactory elements of this type. Therefore, this design has been added to the others as a back-up effort.

The fuel fabrication development has been aimed at obtaining techniques for producing fuel pieces with fine grained, randomly oriented metallographic structures. Heat treatment, as a means of obtaining these conditions, has been studied with some success for cast U-Cr eutectic alloy, wrought 2% wt Zr-U, and alloys of U-Nb and U-Mo, prepared by powder metallurgy methods.

A new reactor core arrangement utilizing 121 elements instead of 19 has led to the use of smaller cross sections in each element. In general, the smaller size eases some of the fabrication problems.

The fuel element designs have been modified slightly since the last report as follows:

1. The core and axial blanket sections are mounted together in one continuous outer structure in a manner similar to the pin-type element shown in Figure 2.
2. The elements are much smaller in cross section.
3. A square cross section is now being considered for the flat-plate design.

REACTOR HEAT TRANSFER

Analyses of the thermal performance of the composite elements have been made. A re-evaluation of hot-channel factors was completed

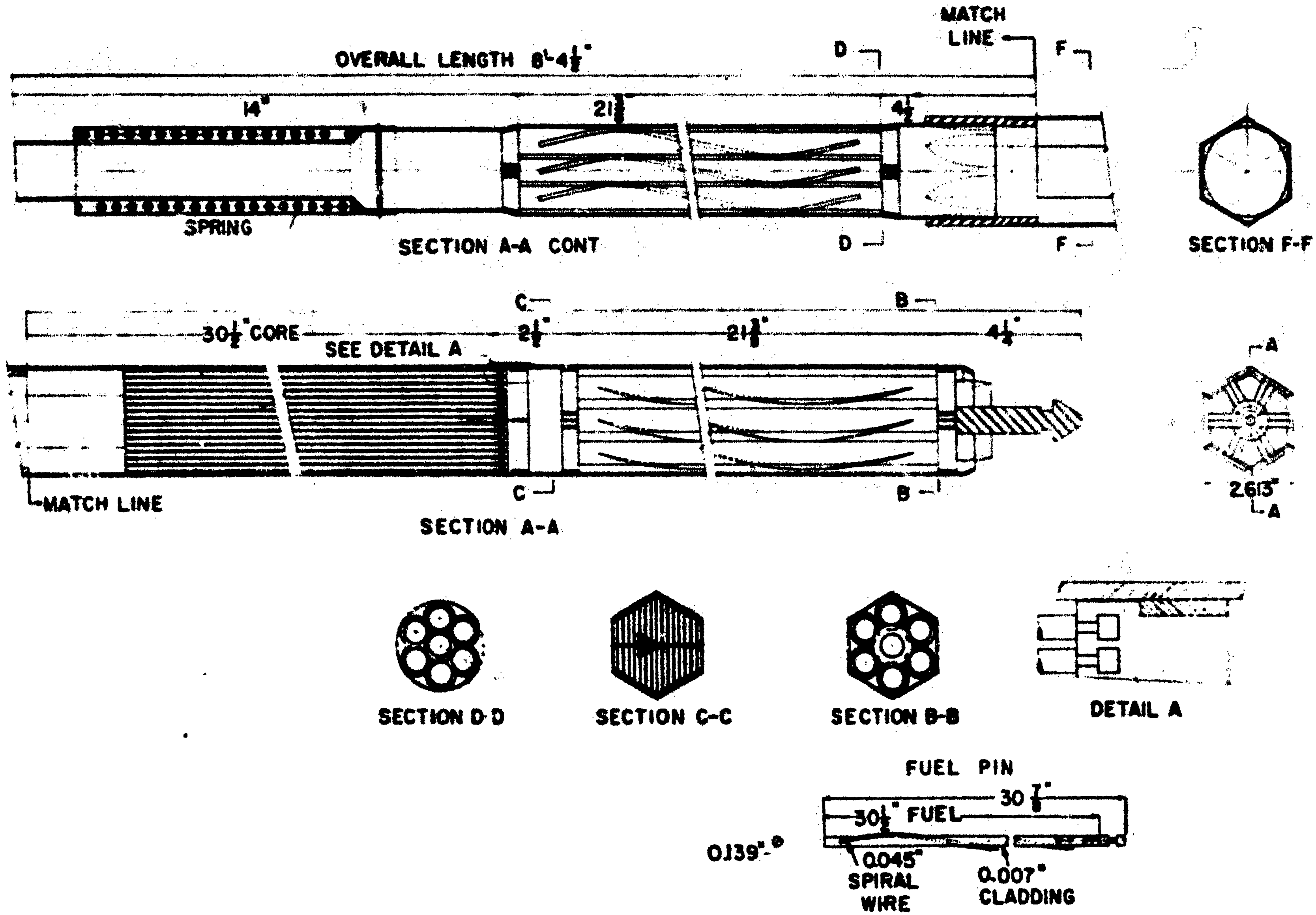


Fig. 2 — PIN-TYPE FUEL ELEMENT SUB-ASSEMBLY

to determine what performance could be assured if a reactor were to be build immediately. Larger individual uncertainty factors were used and applied simultaneously so that their effect was the worst possible. Somewhat lower power output and coolant exit temperatures resulted.

The design of the various element geometries are being optimized for good heat transfer. It is anticipated that a more detailed analysis of the application of hot-channel factors and more nearly optimum designs will permit the production of an element capable of excellent thermal performance.

FUEL ELEMENT FABRICATION

Integral Casting with Sodium Bond

Tubular radiator-type elements have been cast at Nuclear Metals, Inc. of uranium-chromium eutectic alloy around oxide coated Globeiron tubes. These elements were 1.8 inches across the hexagonal flats and 24 inches long. They were cast using an annular end-gate and with the six corner tubes left out of the mold. In the casting recently poured with the corner tubes included, 18 inches of the 24-inch-long mold filled. There is considerable optimism that complete mold filling can be obtained in the near future by varying casting conditions. A concentrated study is in progress to determine the effectiveness of tube positioning devices. At present, about 85% of the ligaments are within the ± 0.003-inch tolerance. About 6% are oversize.

Integral Casting with Metallurigical Bond

Small sections of radiator-type elements have been cast at Nuclear Metals, Inc. using 2% wt Zr-U alloy and zirconium tubes. Sufficient work has been done to indicate the feasibility of this method of fabrication. It appears that mold filling can be obtained with a wider range of casting conditions than is possible with U-Cr eutectic alloy casting. These materials are being studied primarily in an effort to obtain a metallurgical bond between tube and fuel material; a study of this bond integrity is being made.

Plate-Type Fuel Elements

Development work on the fabrication of fuel plates for the plate-type elements has continued at five laboratories.

The Ford Motor Company Research Laboratory has been casting uranium-chromium eutectic alloy into graphite coated steel dies with a cavity of 0.040" x 3.5" x 26". With gravity casting, full-length penetration has been achieved, but the full width has not filled. It is believed that the application of vibration to the steel die will promote better filling by reducing the heat transfer rate between melt and die; therefore, equipment for vibrating the die is being assembled. Other die coatings will also be tested.

The Babcock & Wilcox Research Laboratory is developing a method of randomizing the grain orientation and refining the grain size of mechanically-worked 2% wt Zr-U plates. The device will consist of a quartz mold open at both ends, surrounded by an induction heating coil and blanketed by inert gas. The worked plates are passed through the mold from top to bottom, being remelted by the induction coil and rapidly cooled in the lower part of the mold. Because of randomizing the grain orientation by remelting and the formation of fine grain structure by rapid cooling, the emergent plate is expected to have properties equal to those of the "chill-cast" pins developed by Argonne National Laboratory.

Battelle Memorial Institute has cast grooved plates from U-Cr eutectic using hot graphite molds, molds prepared by the lost wax technique of investment casting, and metallic molds. Good gravity-poured castings have been made up to about 6 inches long in the investment-type molds. Pieces 24 inches long have been cast in hot graphite but show a small amount of porosity at the center. Acceptable detail has been obtained in 12-inch pieces that have been made by using centrifugal casting techniques and metal molds; however, sudden cooling by the mold produced cracking in the U-Cr eutectic alloy.

Nuclear Metals, Inc. has cast 0.40" x 3.5" x 27" uranium-chromium eutectic flat plates in hot graphite molds. The grain size was excessively large, but heat treatment techniques have been found that achieve a reduction in the size.

Flat plates of 2% wt Zr-U alloy clad with 0.002" of zirconium have also been fabricated. The fuel alloy and its cladding were first co-extruded in cylindrical form and then rolled to 0.037" x 1.625" x 20" plate. Thermal cycling tests have not damaged the bond between the fuel and the cladding, and preliminary heat treatment studies give promise of yielding fine grains, randomly oriented.

Grooved plates, 3-1/2" square, have been pressed by Nuclear Metals from uranium-chromium eutectic flat plates; the range of temperatures and pressures required to result in acceptable groove detail has been determined.

Sylvania Electric Products Company has rolled flat plates of 1-1/2% wt Mo-U alloy using powder metallurgy techniques. This alloy was chosen after considerable investigation of many promising alloys. The molybdenum alloy was chosen because the molybdenum inhibits grain growth at elevated temperatures; the 1-1/2% wt Mo-U composition was chosen because there was indication that this composition eliminates the beta phase in which some grain growth occurs.

Perforated Wafer Composite Element

The development of techniques for casting perforated wafers has been concluded at Battelle Memorial Institute; the method is thought to be feasible for fabricating this type of element. A number of successful wafers have been cast using a graphite mold assembly and electrophoretically coated steel pins as the hole cores. Pictures of a typical wafer and an assembly are shown on Fig. 3 and 4.

The method of casting wafers that is being pursued at The Babcock & Wilcox Company, using tubular graphite cores and graphite molds, has produced encouraging results. One of the castings was clad with stainless steel, filled with bonding sodium, and closed by welding. This method of casting is being extended to make full-length elements.

Safety considerations have ruled out the use of wafers that are not strongly bonded together. Methods of joining have been briefly investigated and may prove to be reasonable; however, further effort on wafer elements has been temporarily suspended since other elements appear equally attractive from other standpoints and do not require that the fuel pieces be bonded together.

Pin Fabrication

The Babcock & Wilcox Company Research Laboratory is developing methods of fabricating bare fuel pins by the working and remelting process mentioned under fuel plate development. It is expected that the work will progress on pins until such time as the technology can be fully applied to fabrication of fuel plates. At present, wrought fuel pins are being remelted and chilled in evacuated quartz tubes.

Nuclear Metals, Inc. has extruded and swaged 0.125" diameter 2% wt Zr-U alloy pins with 0.007" zirconium cladding. Heat treatment studies made by that group have shown that these pins can be heat treated to obtain a small grain size with little orientation. Preliminary studies also have been made in remelting the alloy within the cladding by zone melting. This method of fabrication appears quite promising.

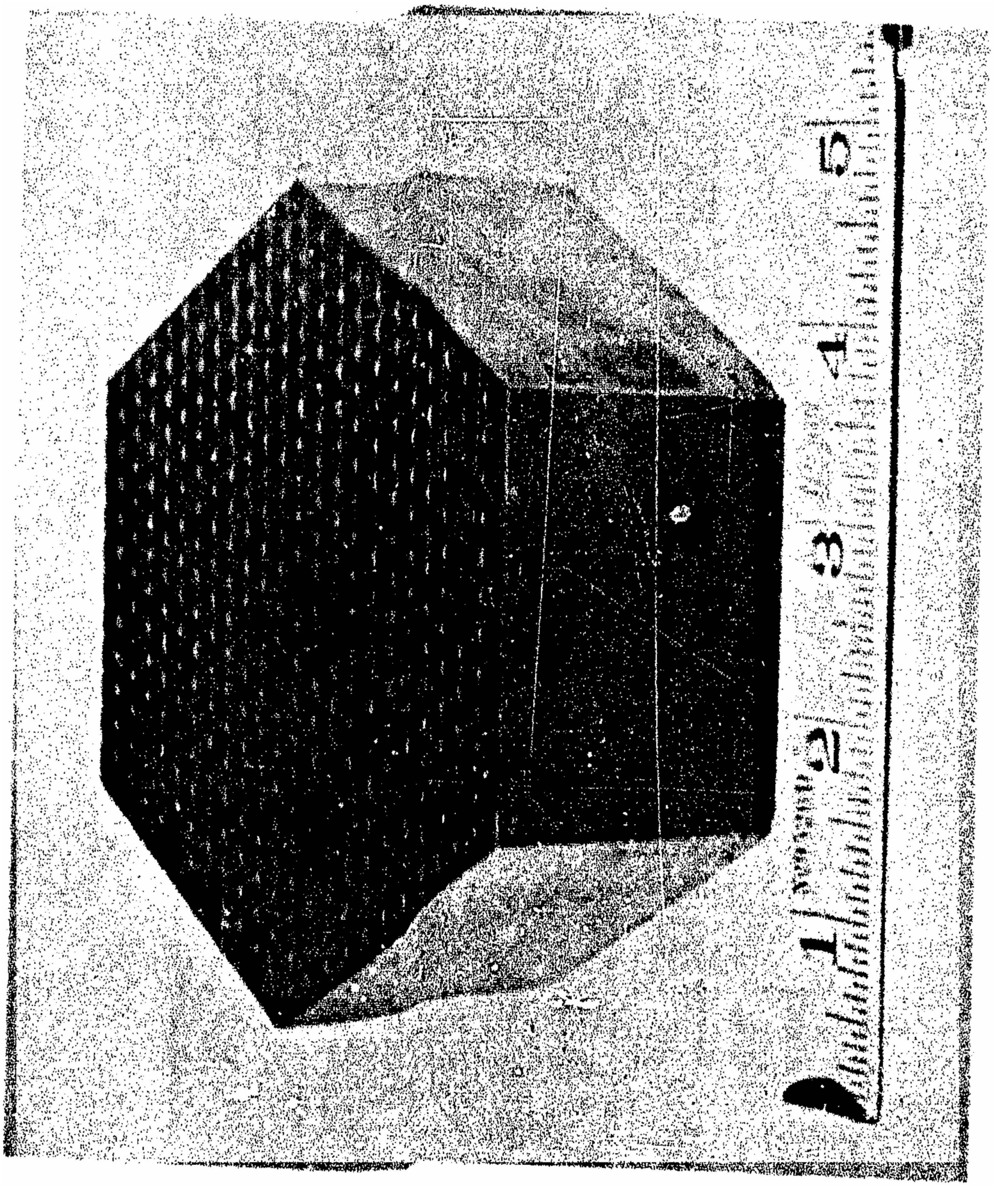


Figure 3 - Cast Fuel Wafer

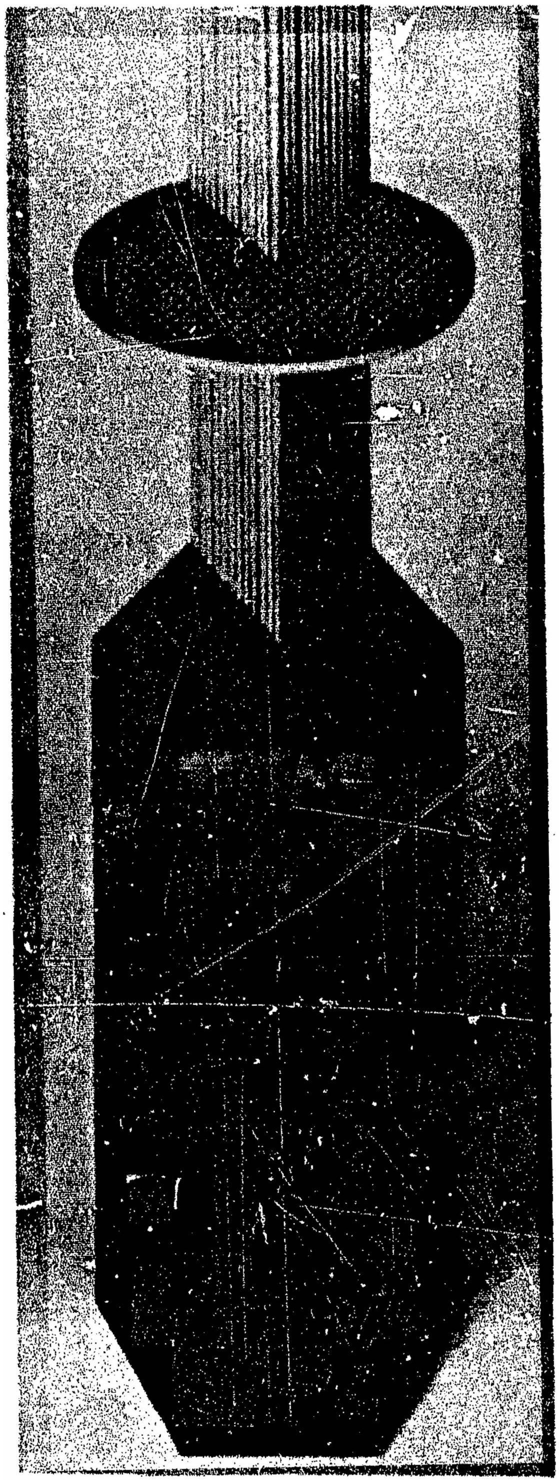


Figure 4 - Assembly of Cast Wafers

RADIATOR FABRICATION

Modine Manufacturing Company has made preliminary fabrication studies to determine some of the problems involved in brazing flat-plate type of radiators. The results have been encouraging. Sample radiator assemblies are shown in Figure 5. It is planned to fabricate at least one of the designs to indicate the magnitude of the problem to be overcome and to obtain a figure for a realistic cost of the structural part of the element.

REMOTE FUEL ELEMENT ASSEMBLY

A fuel element assembly-machine feasibility study has been completed, and the final report is being prepared by the American Machine & Foundry Company. Three types of composite elements were studied: the flat-plate, the grooved-plate, and the wafer. The study shows that all three types can be remotely assembled with reasonably close clearances between the fuel and its containing structure.

TESTING PROGRAM

Thermal Conductance Test of Sodium Bond

The Babcock & Wilcox Research Laboratory has designed a test apparatus for measuring the thermal conductance of thin sodium bond layers between cladding and fuel materials. The apparatus is nearly assembled, and results are anticipated early in March. This apparatus is unique in that it appears possible to measure bond conductances of 100,000 Btu/(hr) (sq ft) (F) with an accuracy of better than $\pm 25\%$.

Continuity of Sodium Bond

Another test being conducted at The Babcock & Wilcox Company Research Laboratory is one to determine whether sodium will completely fill small gaps when imprenated at high temperature; this method is being considered as a means for filling sodium-bonded fuel elements. The initial tests will be made with stainless steel rods in stainless steel tubes, and later tests will use natural uranium rods in stainless steel tubes. The steel tubes are 32 inches long and 0.165 inches OD with a 0.010 inch wall thickness. Radial clearances of 0.001, 0.002, 0.003, 0.004 inch, and a force fit will be tested. The presence or absence of sodium in the gap will be determined by peeling the stainless clad away from the rod. Preliminary results are expected in February 1955.

Hydraulic Tests

In order to determine pressure drops, flow distribution, hydrodynamic forces, etc., in the reactor a series of hydraulic tests is being planned. Since the friction factors are

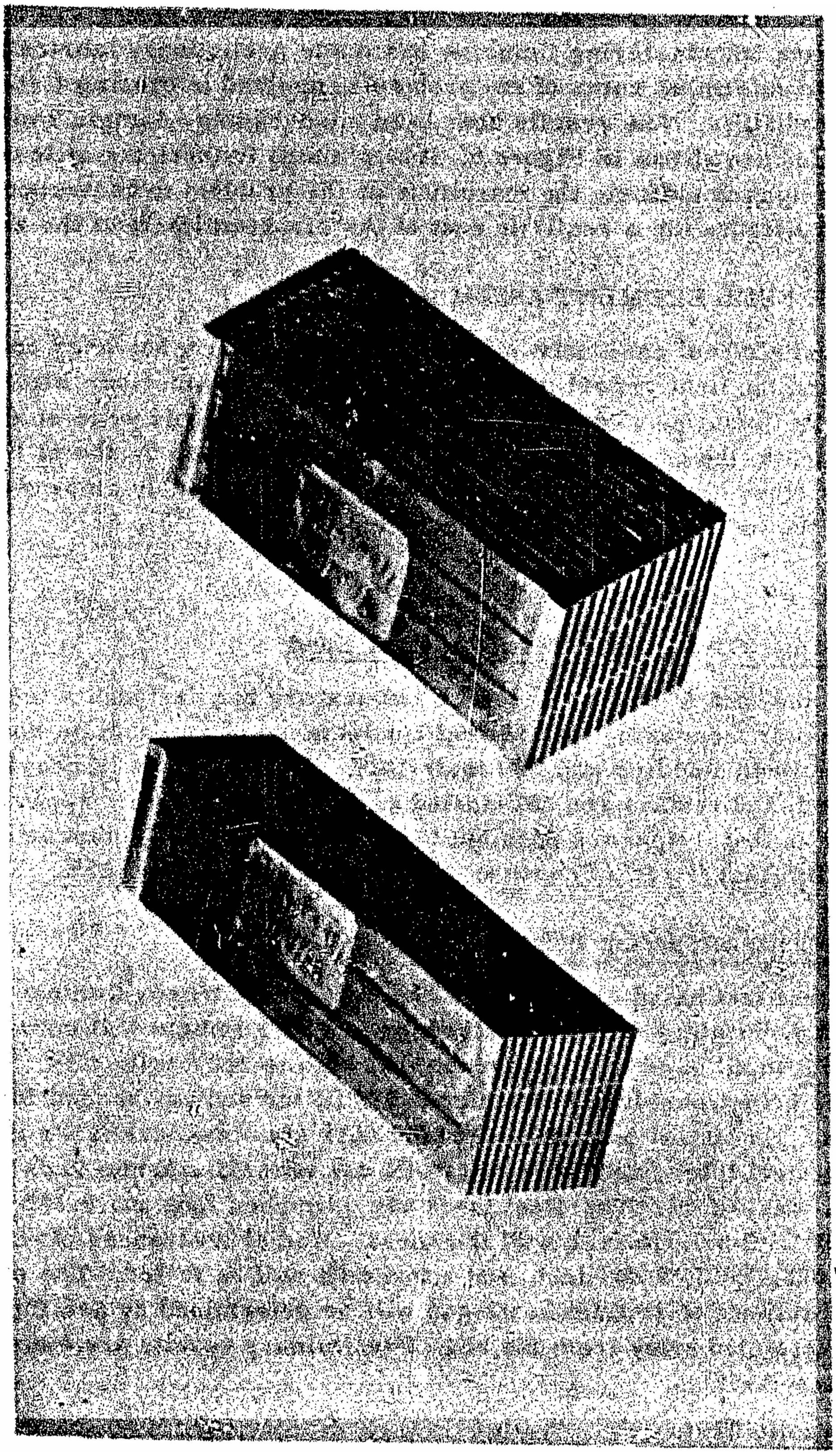


Figure 5 - Flat-Plate Element Radiator Structure

practically constant over the range of Reynolds numbers considered for both warm water and 700 F sodium, it appears that water tests will produce realistic results. The test equipment will be sufficiently flexible to allow separate tests to be run on various components, such as core elements, blanket elements, hold-down and positioning devices, etc. Personnel at several hydraulic laboratories will be consulted in order to determine the best approach for the tests.

FUEL ELEMENT ECONOMICS STUDIES

An economic evaluation of various processes for fuel-element fabrication has been started. The information being developed for this study of each fabrication process includes:

1. Process flow chart.

2. Capital, maintenance, operating and fixed charge costs on:

    a. Fabrication equipment.

    b. Element assembly equipment.

    c. Inspection and test equipment.

    d. Remote handling equipment.

    e. Auxiliary services equipment.

3. Fuel element structure cost, delivered to site.

4. Product storage, equipment layout, and building volume.

5. Inventory tie-up and losses of fuel and other materials in process.

It is expected that the results of this evaluation will indicate which fabrication processes have more eventual promise of low cost fuel-element manufacture.

BLANKET DESIGN

A breeder blanket has been designed that, it is believed, will satisfy many of the basic requirements of any reactor blanket. The design is acceptable as a neutron reflector and absorber; it can transfer all of the heat generated within it to flowing coolant without undue obstruction to this fluid flow or without producing unacceptably high temperature areas; its structural strength appears great enough to allow it to remain in the reactor without failure of any of its parts for periods measured in years. Furthermore, its fertile material is in a form which appears best for aqueous processing. The final consideration is important at present but may be less restricting as alternate processing techniques are developed. The significant features of this blanket are tabulated in Table IV.

The flow of coolant in the radial blanket (See Figure 6, page 45) is upward while that in the axial blankets is downward. No hold-down device is necessary to prevent floating of the radial blanket sub-assemblies since the small pressure drop across this part of the reactor is sufficient to overcome the force of gravity. Positioning the blanket sub-assemblies is accomplished by using either long dowel-like pegs at the bottom of each sub-assembly or by using shorter pegs at the bottom and an "egg crate" structure at the top.

A uniform velocity is obtained throughout the radial blanket since no attempt is made to distribute coolant. This procedure is acceptable in the present case because heat generation rates are not large and the waste of coolant in the radially-outermost regions of the blanket can be tolerated.

Heat transfer calculations were made assuming that the blanket programming was the simplest one possible; that is, an element was placed in one spot in the blanket and allowed to stay there undisturbed until it had attained its maximum permissable burn-up. Furthermore, it was assumed in the heat transfer and coolant flow calculations that the worst possible condition occurred when all sub-assemblies contained 0.2 percent burn-up.

TABLE IV

BLANKET SPECIFICATIONS

Item	Radial	Axial
Type of construction	Pins	Pins
Diameter of pins, inches	0.645	0.645
Length of pins, inches	85.5	21.5
Pin material	Natural uranium	
Method of fabricating pins	Alpha rolling	
Heat treatment	Beta quench	
Cladding thickness, inches	0.010	0.010
Cladding material	Stainless steel	
General shape of pin groups	Hexagonal	
Number of pins per sub-assembly	19	7
Across flats dimension of sub-assembly, inches	3.99	2.50
Number of sub-assemblies	468	121
Sub-assembly container thickness, inches	0.0625	0.0625
Pitch of pins, inches	0.735	0.915
Composition		
a) Uranium, % by volume	56.8	35.1
b) Sodium, % by volume	31.5	50.8
c) Stainless steel, % by volume	11.7	14.1
Weight of sub-assembly, lb	400	110
Maximum uranium temperature, F	1020	- - -
Maximum cladding temperature, F	620	- - -
Maximum temperature rise of coolant, F	70	3
Pressure drop of coolant, psi	9.5	7.5
Velocity of coolant, ft/sec	13	30
Burn-up, % of total atoms	0.2	0.2
Plutonium build-up, % of total atoms	0.4	0.4

REACTOR VESSEL

The Babcock & Wilcox Company and Allis-Chalmers Manufacturing Company made reactor vessel designs in which the coolant flows in series through the radial blanket and then down through the fuel section. An evaluation of these designs indicated that several features required modifications; they have been incorporated into a revised Babcock & Wilcox drawing and are shown in Figure 6.

Preliminary work has started on the design of a reactor vessel for coolant parallel-up-flow through both fuel and radial blanket sections.

FUEL AND BLANKET ELEMENT SUPPORT AND HOLDING STRUCTURE

Several methods of supporting, holding, positioning, and aligning fuel and blanket sub-assemblies that have been or are being investigated are:

1. All sub-assemblies clamped; bottom pin supports used.
2. Fuel sub-assemblies clamped; blanket sub-assemblies aligned and held at the top; bottom pin supports used for both types of sub-assemblies.
3. All sub-assemblies supported by a dowel-like structure with top alignment and holding structure.
4. All sub-assemblies supported by a dowel-like structure with top alignment and holding structure. In addition, "egg crate" matrix tubes are positioned between the sub-assemblies.

Figure 7 is a photograph of the model of a cluster of seven sub-assemblies supported and held as stated in method No. 3. The model, as shown, is to be reconstructed to utilize the latest concepts and recommended design changes, such as utilization of the lower axial blanket section of the sub-assembly as the supporting member (dowel-like structure).

DESIGN WORK

The Babcock & Wilcox Company is progressing on the design work

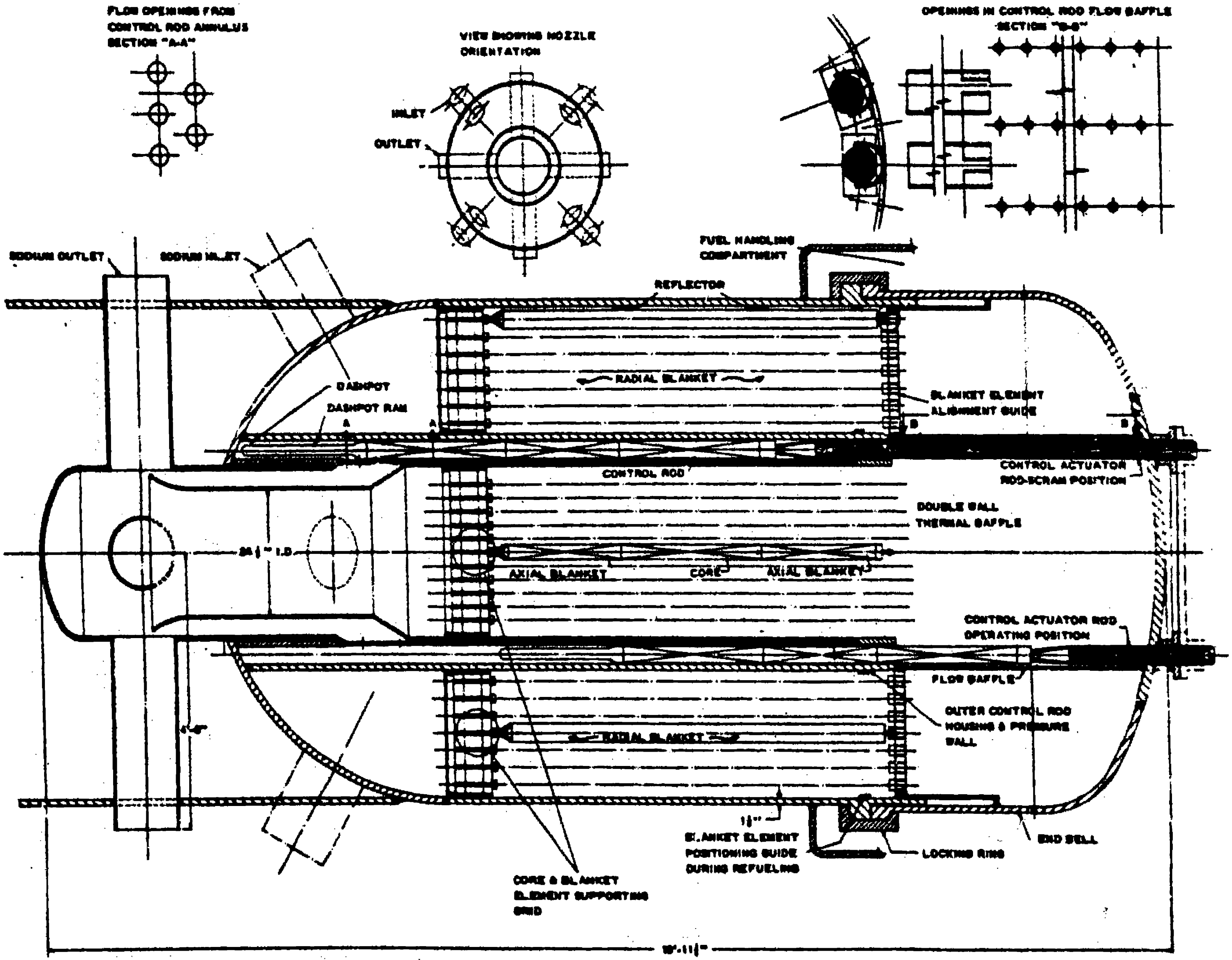


Fig 6– REACTOR VESSEL
(SECTIONAL ELEVATION)

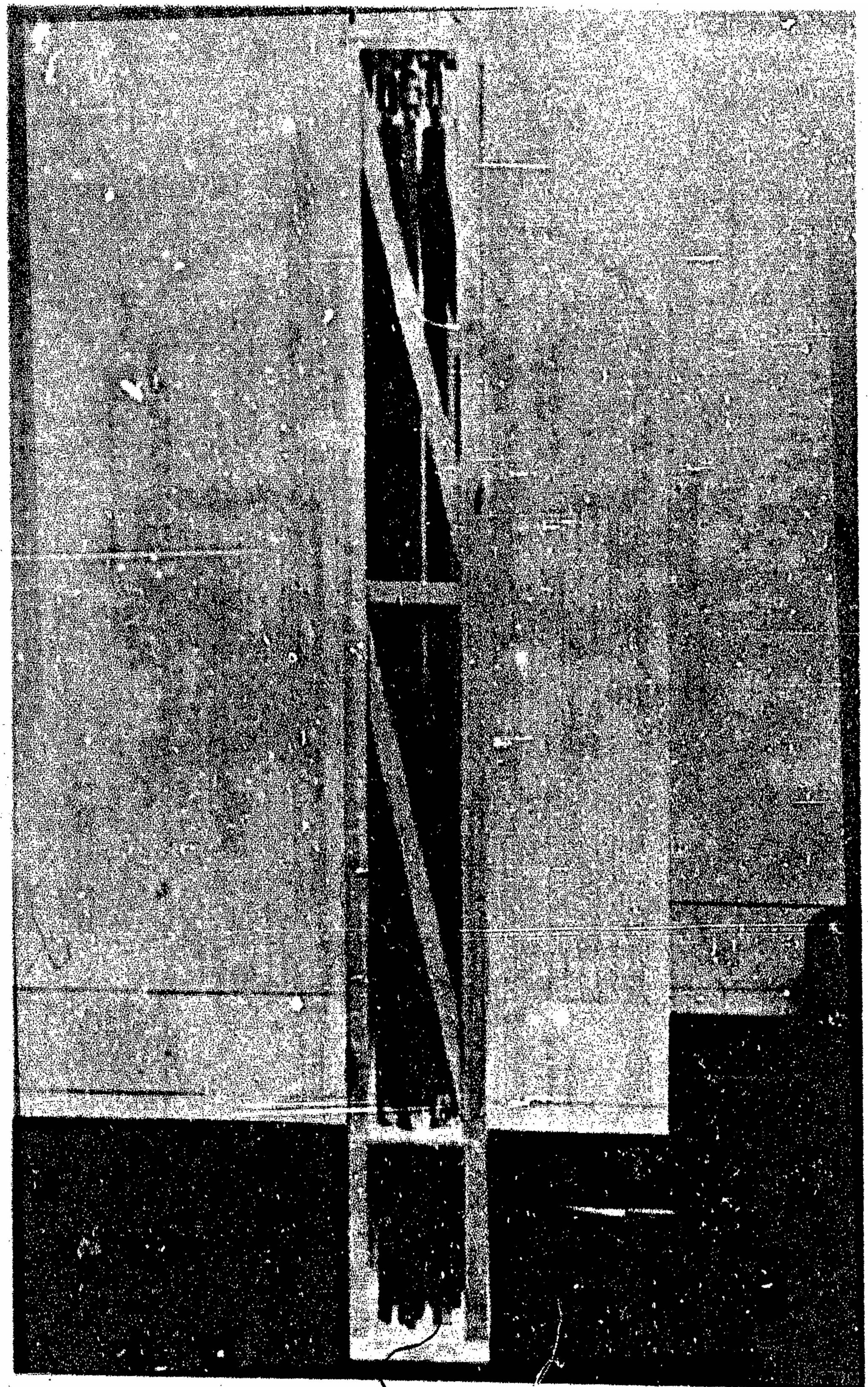


Figure 7 - Fuel and Blanket Sub-Assembly Support and Holding Structure

required before detailed application specifications and drawings for the proposed reactor vessel, reactor structure, and the primary shield tank can be issued. The scope of this work includes the following items:

1. Design of the top support for the vessel, thermal shielding, support structure for fuel and radial blanket sub-assemblies and thermal shielding, and thermal shielding around reactor vessel and primary-shield tank.

2. Alternate design of head-closure locking device and of seal between vessel and head closure.

3. Evaluation of alternate methods of supporting, holding, positioning, and aligning fuel sub-assemblies.

4. Redesign of reactor vessel in such a manner as to allow for visual or x-ray inspection of all parts of the vessel prior to radioactive operation.

5. Design of a helium cooled, primary-shield tank, including supports, that will contain the sodium in the event of a reactor vessel leak, serve as a primary-shield tank containing the so-called "thermalizing shielding," and act as a safety zone in the event of a nuclear incident.

6. Design of the control element housing to adequately serve as a control element guide, as a pressure wall, and if necessary, as a dashpot.

7. Determination of the design stress limits; perform the stress analysis of critical parts of the vessel. This analysis includes the effects of superimposed forces such as pipe reactions and control element mechanical shock.

8. Design of fuel sub-assembly flow distribution plates.

9. Design of thermocouple installations, electrical heating of the vessel, and, if necessary, an emergency poison section and melt-down container.

10. Preparation of plans for invitation to bid and help in preparation of specifications.

11. A special study for a coolant parallel-up-flow design to include the following: the design and evaluation of a hold-down plate for core and other elements, the use of separate inlets for core and blanket coolant, and the design of pressure walls between the axial and radial sections.

SPECIFICATIONS

Preliminary drafts of three sections of the reactor vessel specification have been written: the scope, design data, and design stress limits.

The two arrangements of fuel loading and unloading equipment, described in report DCDE-101, have been further studied from the standpoint of cost and over-all simplification. These concepts, referred to as the two-plug concept and the single-plug concept, also were examined to determine the feasibility and problems associated with each concept as well as their compatibility with other related components of the fuel element transfer system.

The more detailed studies conducted during the past six months have terminated in the selection of the single-plug design; cost studies prepared for both concepts indicate that the single-plug arrangement will be considerably cheaper to build and offers more in the way of simplicity and reliability.

Basic engineering studies of the remainder of the fuel element transfer system have reached the point where similar detailed conceptual studies are necessary in order to fix the concept so that detail designs of components for the rest of the system may be started.

REACTOR COMPARTMENT ARRANGEMENT

Design studies of the components associated with alternate reactor-compartment arrangements indicated that the single-plug concept was much simpler for all components except the offset handling crane. Since the position of each fuel element in the reactor necessitates a different angular position of the plug and crane, it was thought necessary to provide mechanical orientation of each element over its position so that the hexagonal-shaped element would be properly oriented with respect to the hexagonal void into which it would be placed. This mechanical orientation was to be accomplished by providing a mechanism which would pivot the gripper head through an included angle of 60°. However, analysis of the design of this mechanism revealed that a rotary face-seal and bearing surfaces, under sodium and in a zone of extremely high radiation flux, would be necessary. It is not possible to predict adequately the effects of this environment. This factor, coupled with the belief that undue complexity without easy access for inspection and maintenance constitutes inherent unreliability, led to a search for some other solution to the problem.

Effort was directed toward devising a means for causing the elements to orient themselves as they enter the hexagonal voids in the reactor. An element-self-orientation test rig was designed and fabricated to demonstrate the feasibility of such a procedure. The results of the first design are sufficiently encouraging to permit elimination of those mechanical complexities in the offset crane that made the single-plug concept questionable.

The single-plug arrangement then became the much better choice from all aspects and has been selected as the design concept for the reactor plant.

Plan and elevation drawings of the element handling mechanism for a reactor having the series-down-flow coolant pattern are shown in Figures 8 and 9, respectively. The handling equipment consists of a large rotating plug and an offset crane mounted in the plug. The end-bell closure for sealing the reactor during operation is suspended beneath the movable shield plug and is actuated by drive mechanisms mounted above the plug. The plug is, in turn, supported by a top flange that rests on ball bearings. Motor-driven actuators are provided for rotating and locating the plug and offset crane.

Adjacent to the reactor handling compartment is a tank filled with sodium in which fuel and blanket elements will be allowed to decay. The decay time chosen is such that elements will not melt due to residual heat generation when they are removed from the sodium. Elements will be placed in the decay pit in receptacles after removal from the transfer rotor. The transfer rotor has spaces for fresh elements so that the offset crane can insert an irradiated element and immediately grasp a fresh element for insertion into the reactor.

A liquid-metal loop-type seal and an expandable mechanical seal are located under the flange of the shield plug to prevent escape of radioactive vapor and gas from the reactor compartment into the space above the plug where control actuating mechanisms are located. The entire space above the plug is covered by a thin shell containing an inert atmosphere in order to prevent leakage of air into the sodium filled compartment below the plug.

ALTERNATE REACTOR-COMPARTMENT ARRANGEMENT

An alternate coolant flow configuration, parallel-up-flow, was studied intensively and has been found to possess certain very definite advantages. The pressure-containing end-bell closure is replaced by a simpler fuel element hold-down plate. This arrangement allows a greatly simplified set of drives and actuators, located above the shield

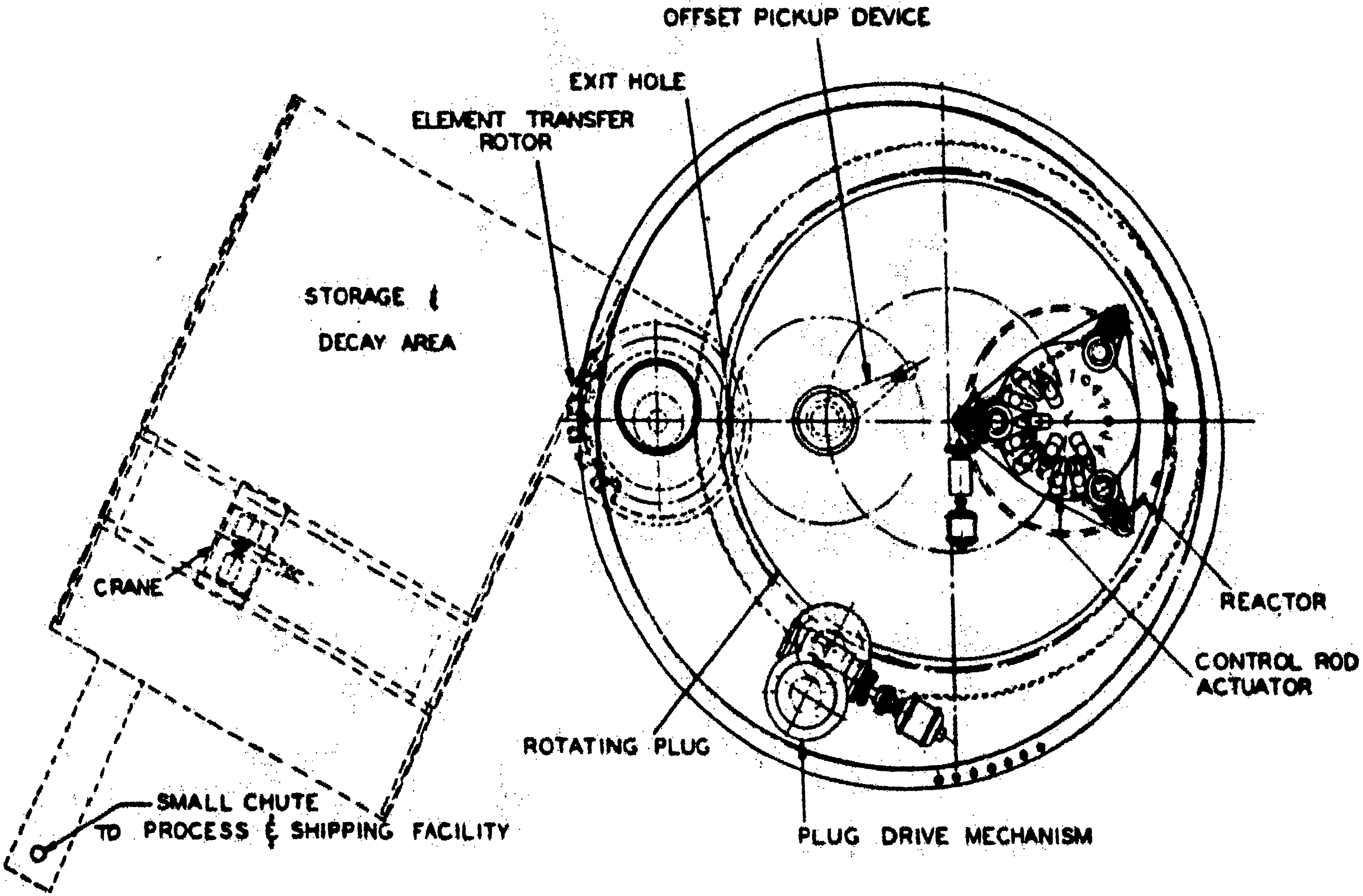


Fig. 8 — ELEMENT HANDLING MECHANISM (PLAN)

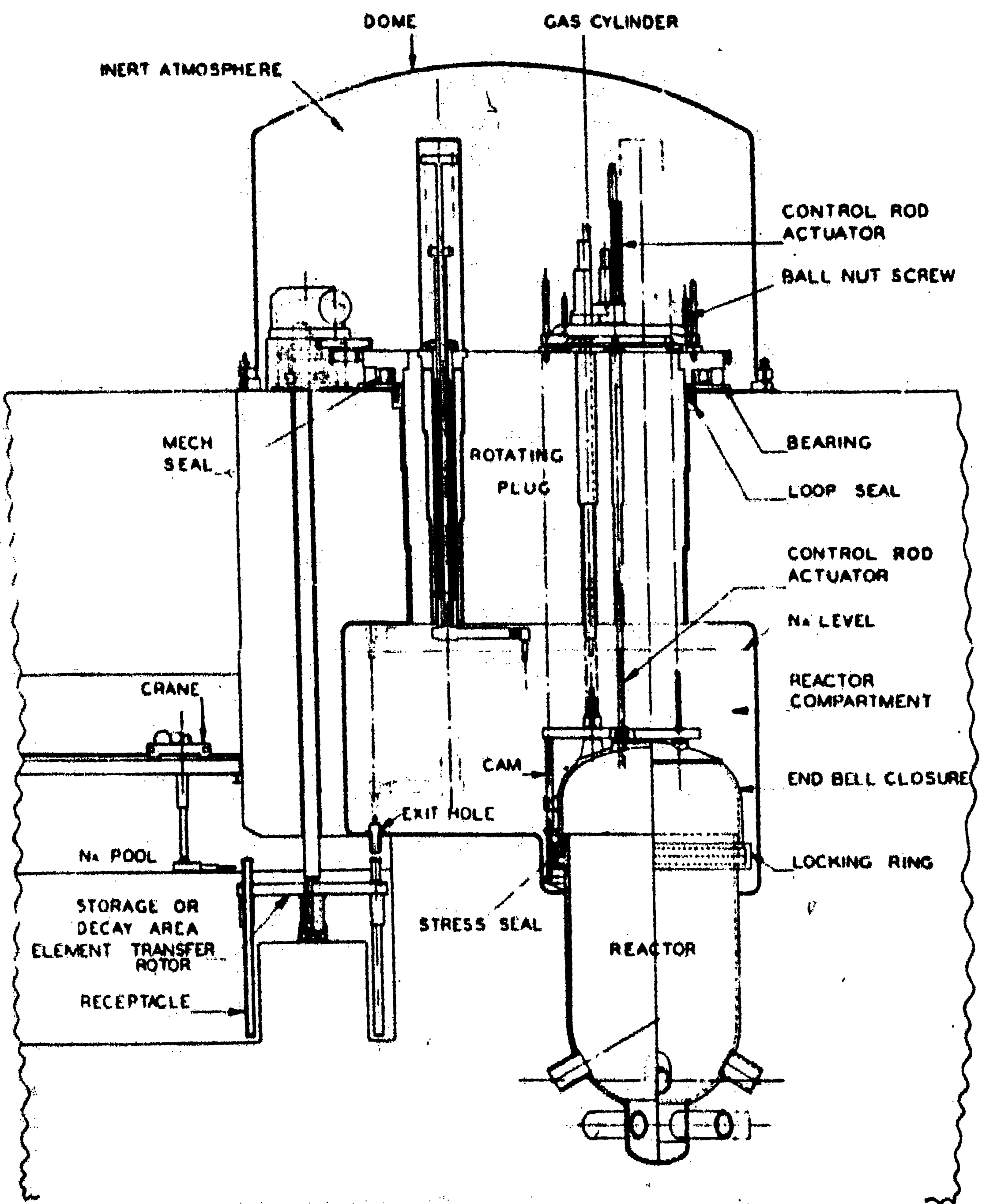


Fig. 9 – ELEMENT HANDLING MECHANISM (ELEVATION)

plug, to actuate the vessel closure and simplifies the problems arising from sealing against high pressure, high temperature sodium.

The mechanical arrangements for series-down-flow or parallel-up-flow through the reactor require the same general procedure during reactor shut-down. The various steps required during one complete cycle of fuel and blanket loading and unloading are:

1. Lower all control rods to scram position.
2. Unlatch all control rods from actuator rods.
3. Raise all actuator rods to clear top of reactor vessel.
4. Reduce sodium flow to balance pressure across reactor-vessel closure.*
5. Release pressure in stress seal band.*
6. Unlock vessel closure locking-ring.*
7. Raise vessel closure clear of reactor.
8. Position rotating shield plug and offset crane to place gripper head over proper fuel or blanket element.
9. Lower pick-up arm, grasp element, and withdraw.
10. Rotate plug and crane to position element for exit from reactor compartment.
11. Place element in empty receptacle in transfer rotor.
12. Revolve transfer rotor to position fresh element under gripper head.
13. Grasp fresh element and lift into reactor compartment.
14. Rotate shield plug and offset crane to position over void in reactor.
15. Lower crane to place element in reactor.

These steps are repeated until the desired reactor unloading or loading is accomplished. The shield plug and crane are then rotated into the reactor-operating position, the vessel closure is lowered into place, the control rods are grasped by the control rod gripper mechanisms, and the reactor is ready for start-up.

* Series-down-flow arrangement only.

The transfer rotor contains a one-week normal supply of fresh fuel and blanket elements. After the reactor is again in operation, the work of placing the elements in racks in the decay pit can proceed.

After the decay period has elapsed, the decay-pit crane deposits receptacles at the reactor building exit, from which they are transported to the shipping area outside the reactor building. Fresh elements are brought into the reactor building through this same port, so that there is one system for both charging and discharging fuel.

The preliminary work of the past few months has brought out the particular sets of problems associated with various transfer system arrangements; more detailed studies are proceeding to obtain solutions to these problems. Evaluations will then be possible so that the final over-all transfer system concept can be fixed.

CONTROL PHYSICS

The reactivity available in 18 three-region control rods has been calculated; the results are as follows:

Regional Change	Reactivity Up-Flow Reactor	 Down-Flow Reactor
Iron to Vacuum	- 3.6%	- 3.3%
Vacuum to Poison	- 13.5%	- 10.8%
Sodium to Poison	- 14.5%	- 11.5%
Central Fuel Element	0.5%	

Rotating control elements that are mechanically similar to those studied for SIR but containing materials compatable to fast reactor control have been investigated to a point where it appears that they are inadequate for reasonable-size elements. The main problem to solve would be that of moving sufficient material over a distance great enough to produce adequate reactivity changes.

MECHANICAL DESIGN OF CONTROL ROD

The control rod, which is shown in Figure 10 is rectangular in shape and consists of six major components: they are the latch, fast-neutron poison section, void section, reflector section, dashpot ram, and fluid bearings.

Latch

Two types of latches have been designed; one design consists of a pair of splines, one spline of which is straight and the other helical. The splines are inserted into the adapter attached to the rod by allowing relative angular movement between the two splines. They are latched by locking the two spline shafts together.

The other design is the finger-gripping-type. On driving down to latch, the fingers are in open position until the bearing shoulder is reached. Then, additional down motion snaps the fingers outward to the load carrying position, where they are locked by a solenoid operated toggle device.

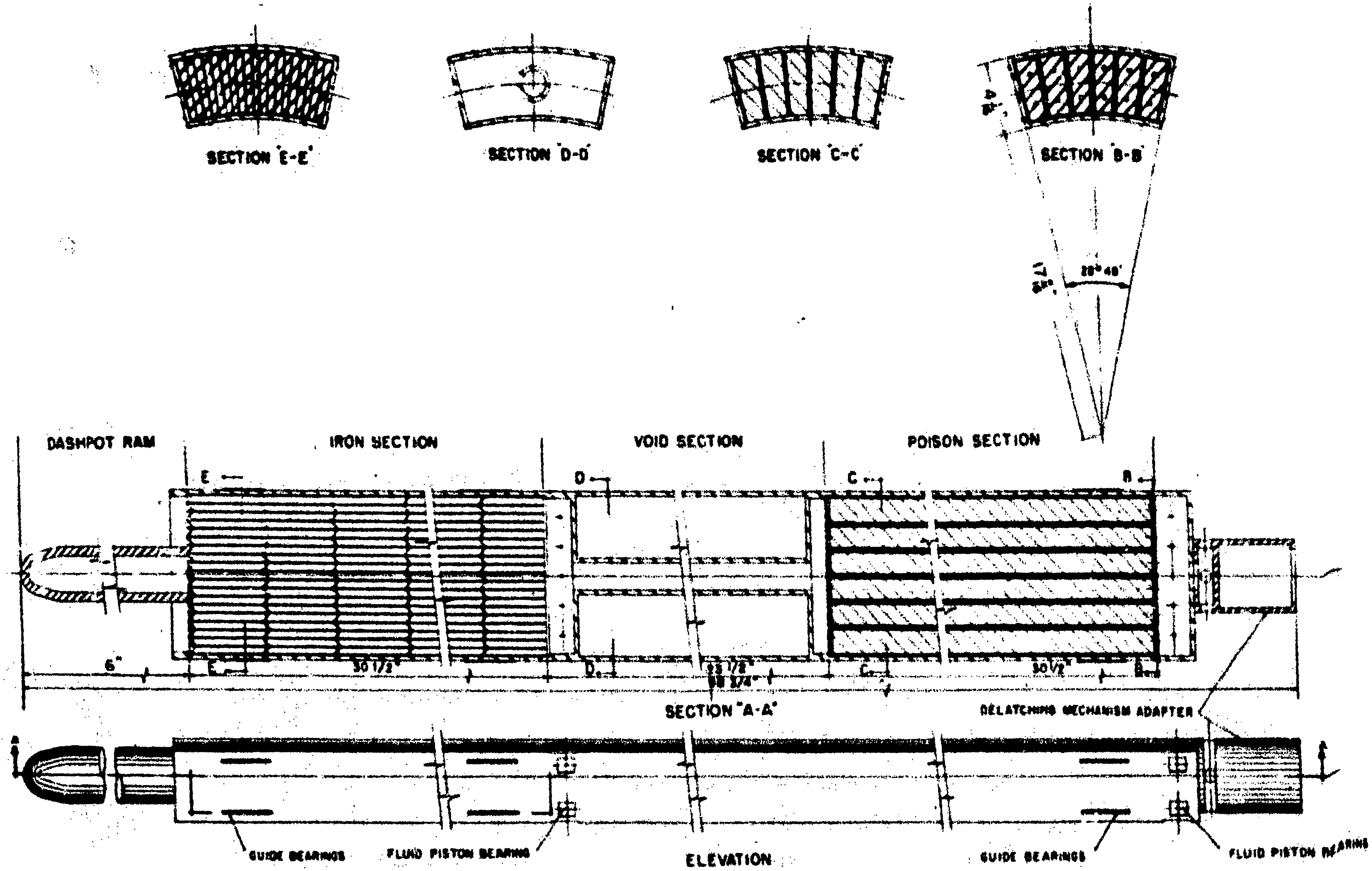


Fig. 10 CONTROL ROD

Fast-Neutron Poison Section

The poison section consists of 150 "calrod"-type pins. These pins would be made by packing Li_2^6O powder into stainless steel tubes and then drawing through a die to obtain 90% of theoretical Li_2O density. A pin diameter of 1/4 inch is contemplated to give a heat transfer rate high enough to keep Li_2O in the solid state. Each pin will have a gas space at its top to contain tritium gas formed and allow for Li_2O expansion. An alternate poison section design, using boron-10, is in progress as a back-up effort.

Void Section

The void section, containing as little structural material as possible, acts as a nuclear void. A pipe running through its center is used to transport coolant from the poison section to the reflector section.

Reflector Section

The reflector section consists of stainless steel blocks with the coolant flow holes distributed across the blocks proportional to the gamma-ray induced heat, as shown in Section EE of Figure 10. This is done to obtain uniform temperature at any given section and thus prevent bowing.

Dashpot Ram

The dashpot ram is a bullet-shaped extension to the control element designed to act as a plunger in the dashpot. It has a 6-inch effective length.

Fluid Bearings

The control rod is equipped with fluid piston type of bearings at two axial locations. These bearings will center the control rod in its channel during normal operation and during its scram fall.

HYDRAULIC CONTROL-ROD ACTUATOR TEST

A model of the sodium operated actuator for control rod drives is being assembled at Allis-Chalmers Manufacturing Company. The first tests will use water as the operating medium to check over-all operation and to determine leakage rates around the piston. It is then planned to operate the actuator with sodium using actual positioning devices and equipment as designed for the final system.

BELLOWS AND SEALS

A test rig has been built by American Machine and Foundry Company for the purpose of testing, in sodium vapor, the operation of bellows and seals intended for use on control rods and on the handling crane. It also includes an arrangement for testing piston ring type of seals to determine the leakage rate and operation in sodium. Combinations of several different piston ring materials will be used to determine effectiveness and compatibility.

SCRAM DYNAMICS SIMULATION

In the latter part of 1954, it became necessary to determine the coolant flow characteristics through the reactor core during scram conditions. Due to the non-linear nature of the mathematical equations describing the scram phenomena, the problem was set up and solved on the Bendix Aviation Corporation Research Laboratories' Reeves electronic analog computer (REAC).

The prime purpose of this simulation was to obtain reactor core coolant velocities as a function of time and scram time for the control rods. From these data, the time interval involved in dropping from full flow to some minimum safe velocity was determined. This time interval must be known in order to properly design the auxiliary power supply controls that must operate within this time interval.

Design information from which heat transfer during reactor scram or shutdown may be computed was also obtained. Further insight was obtained on the reversal of flow in the core due to the ram effect of the control rods during a scram.

The transient response during 80 different reactor scram conditions was obtained on the REAC. The variables plotted against time during scram were coolant velocity through core, pressure drop across the core, coolant velocity in piping to the intermediate heat exchanger, position of control rods, and pump pressure. Runs were made varying the pressure decay time of the pump, time after pump failure at which rods were scrammed, and the position from which the rods were dropped.

Figure 11 illustrates the configuration of the reactor system simulated. Figure 12 shows two typical scram histories; the solid lines indicate the transient responses obtained when the rods were dropped 0.25 seconds after the pumps had failed, and the dashed lines indicate the response obtained from a scram without pump failure. In both cases, the core velocity slows down as the piston effect of the rods is felt; and in the case where the pumps fail, the flow direction reverses.

TRANSIENT CONDITIONS ON ONCE-THROUGH-TYPE STEAM GENERATOR

Tests will be made on the once-through, counterflow, shell-and-tube

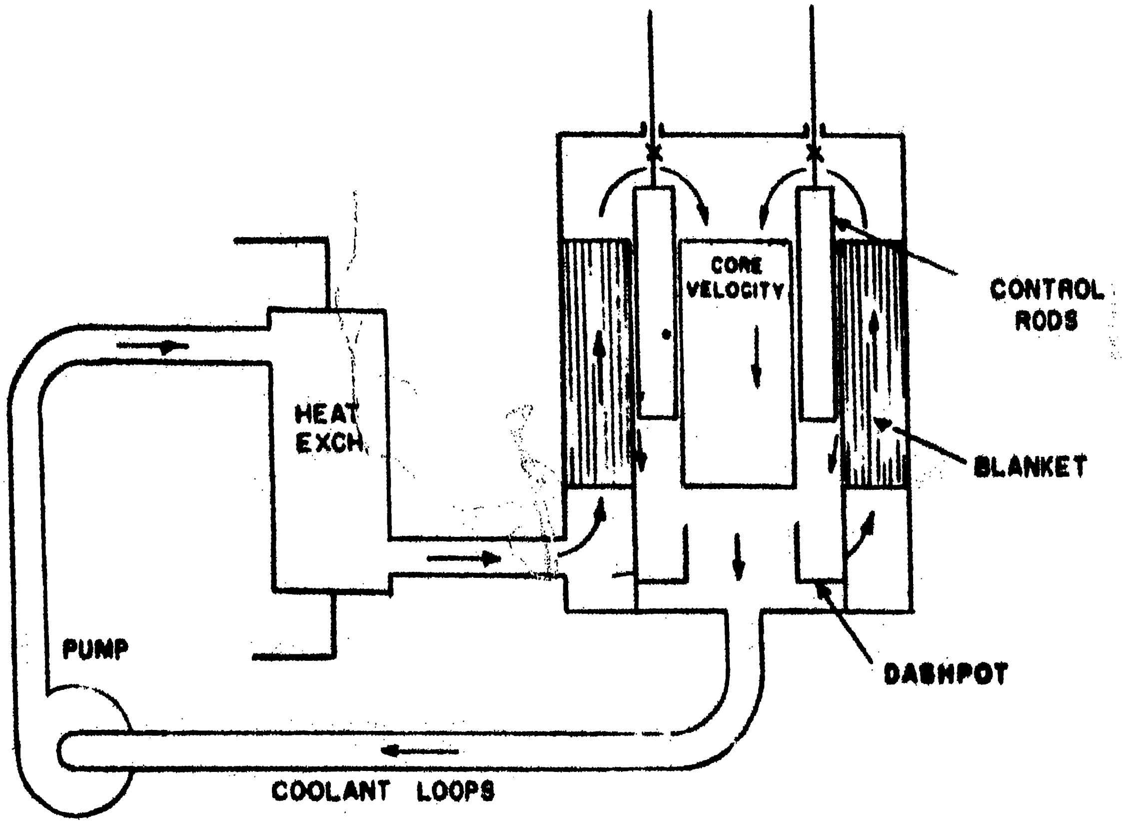


Fig. 11— SCRAM DYNAMICS SYSTEM

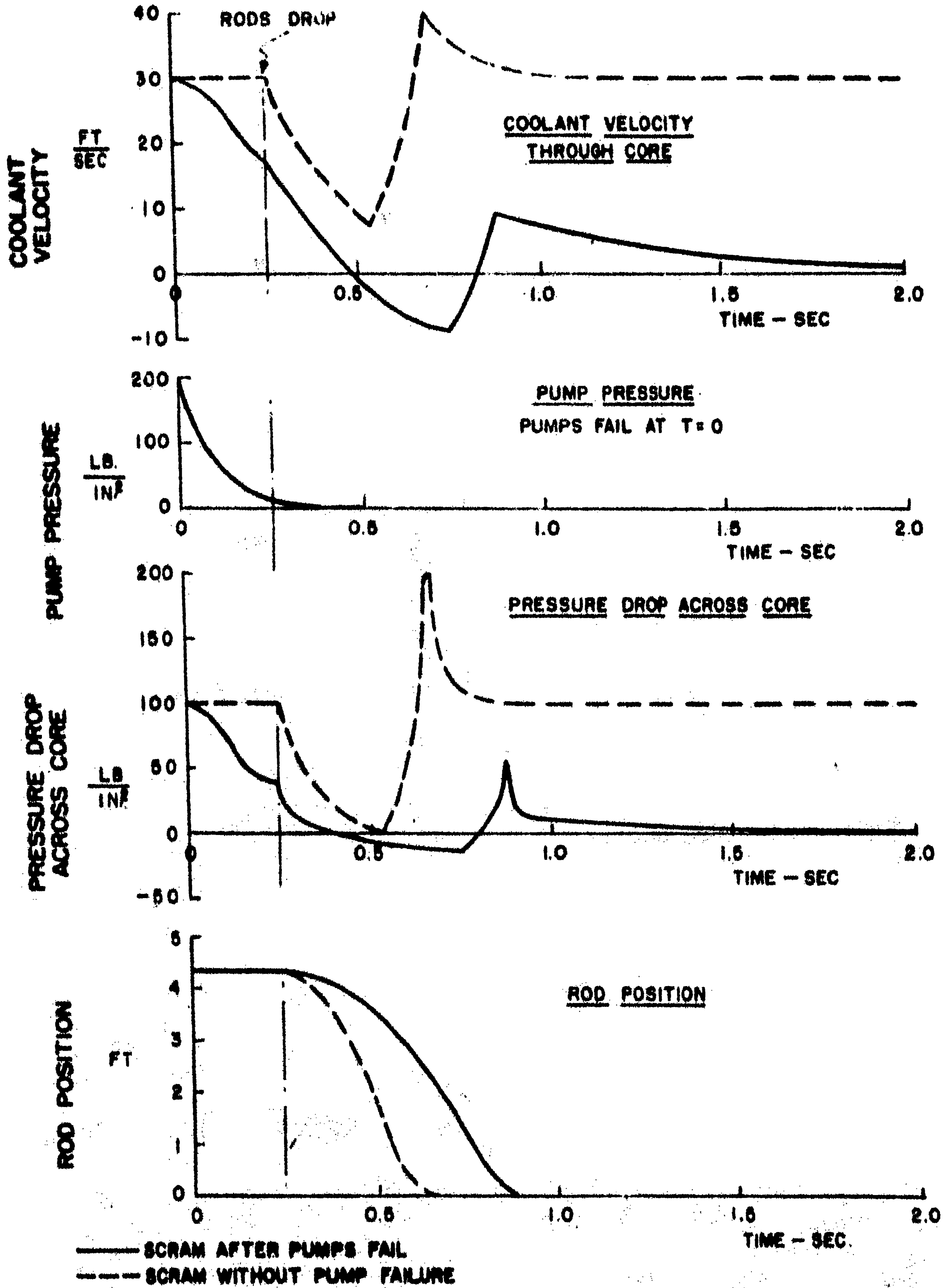


Fig. 12 — TYPICAL SCRAM HISTORIES

boiler shown in Figure 18, page 70 to determine transient behavior of all inlet and outlet parameters of the boiler. These tests will be conducted during the over-all test of this steam generator, described in the section entitled "Steam Generator Test" on page 68.

The transient data obtained from this test will allow equations to be written that will best describe the boiler operation. These equations will then be duplicated in a simulator and will permit the establishment of suitable control methods for the boiler and steam end of the system as well as contribute to over-all stability studies.

ALTERNATE REACTOR-CONTROL STUDIES

Alternate conceptual designs are being investigated in an effort to obtain a simpler control system and to obtain the ultimate in safe operation. Included in these studies is an effort to reduce the size and weight of the control elements.

LIQUID METAL AND STEAM-POWER SYSTEMS

A schematic flow diagram of the liquid metal and steam-power systems, revised to reflect current design considerations, is shown in Figure 13. This design is based on the following conditions:

Reactor Power, mw(heat)	306
Sodium Temperature	
Entering Reactor, F	560
Leaving Reactor, F	810
Sodium Flow, lb per hr	13.55×10^6
NaK Temperature	
Entering Steam Generator, F	760
Leaving Steam Generator, F	510
NaK Flow, lb per hr	16.7×10^6

LIQUID METAL SYSTEM

The intermediate heat exchangers are "U" tube, counterflow, shell-and-tube units with sodium inside the tubes and NaK on the shell side. A plan view of the heat exchanger, together with current data, is shown in Figure 14. Pump requirements have been revised to reflect current design considerations, and preliminary drafts of pump and valve specifications have been completed. Schematic line diagrams of the auxiliary systems for both the sodium and NaK systems have been completed.

STEAM-POWER SYSTEM

The design of the steam-power system is basically the same as that outlined in report DCDE-100. Current design data are as follow:

Steam Conditions	
Pressure, psia	600
Temperature, F	740
Flow, lb per hr	1,082,000
Feedwater Temperature, F	430

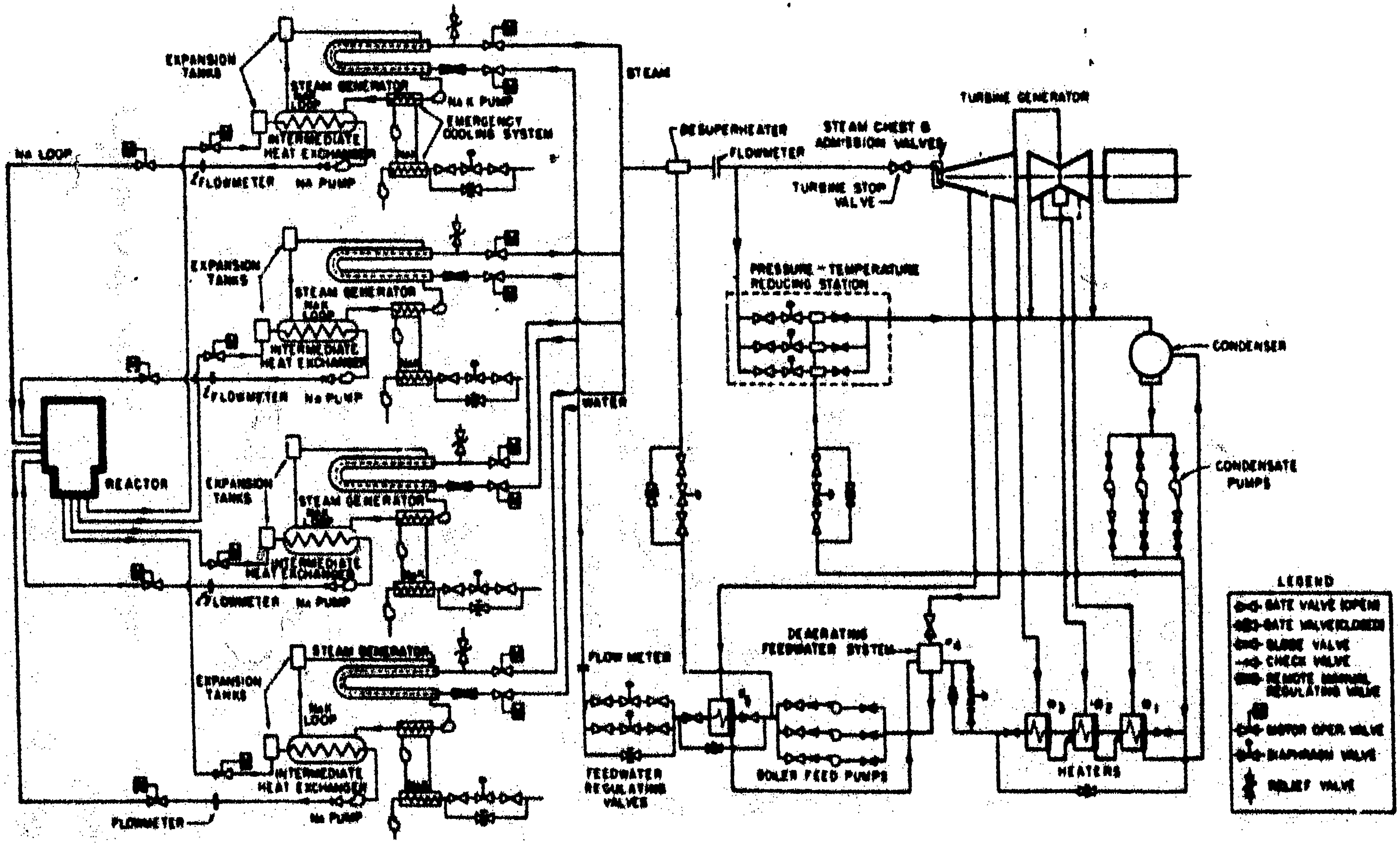
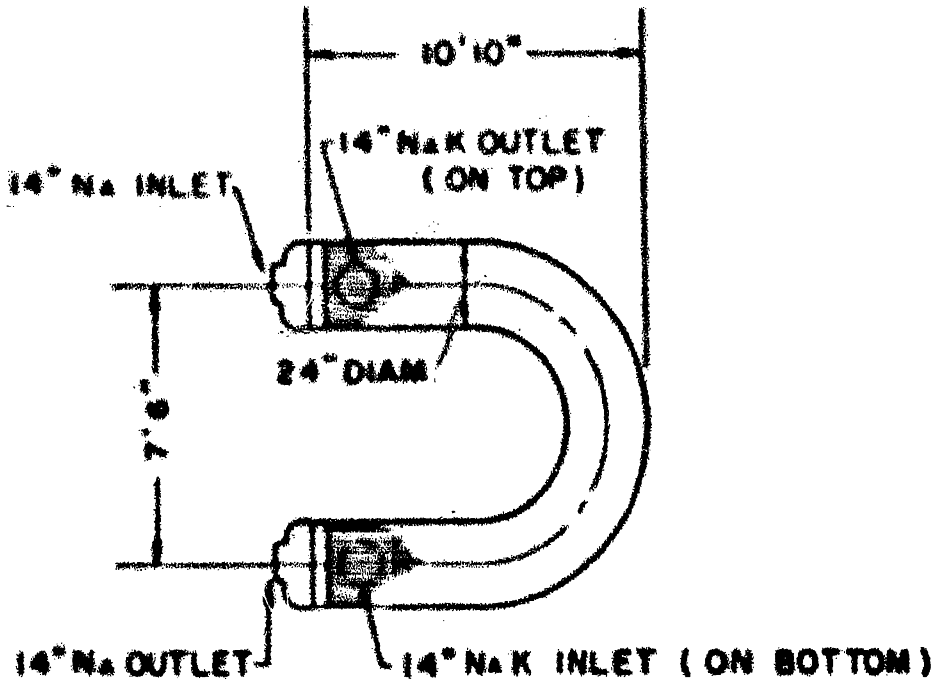


Fig. 13 — LIQUID METAL and STEAM — POWER SYSTEMS

NUMBER OF UNITS	4
TUBE DATA	
NUMBER PER UNIT	665
OUTSIDE DIAMETER, IN	0.625
WALL THICKNESS, IN	0.0625
PITCH (TRIANGULAR), IN	0.875
EFFECTIVE LENGTH, FT	20
MATERIAL	"A" NICKEL
HEAT TRANSFER AREA, SQ FT	2150



PLAN VIEW

Fig. 14 — INTERMEDIATE HEAT EXCHANGER

Regenerative Feedwater Heating, stages	5
Condenser Pressure, in. Hg abs	1.5
Turbine - Generator Unit	
Rating, Kw	100,000
Type	Tandem-compound, double exhaust flow
Speed, rpm	3,600
Condenser pressure, in. Hg abs	1.5
Net Station Output, kw	88,000

<u>Steam Generator Units</u>

The steam generators are once-through, counter-flow, shell-and-tube units having a single wall tube to separate the water and NaK. Water and steam are inside the tubes; NaK is on the shell side. An elevation drawing of this unit, together with pertinent design data, is shown in Figure 15.

PLANT LAYOUT AND ARRANGEMENT STUDIES

The parameters for the plant layout studies remain the same as stated in report DCDE-101. An elevation drawing of the reactor plant and generating plant, revised to reflect current design considerations, is shown in Figure 16. Alternate plant layout studies have been made using different sets of design parameters.

NaK-WATER REACTION TESTS

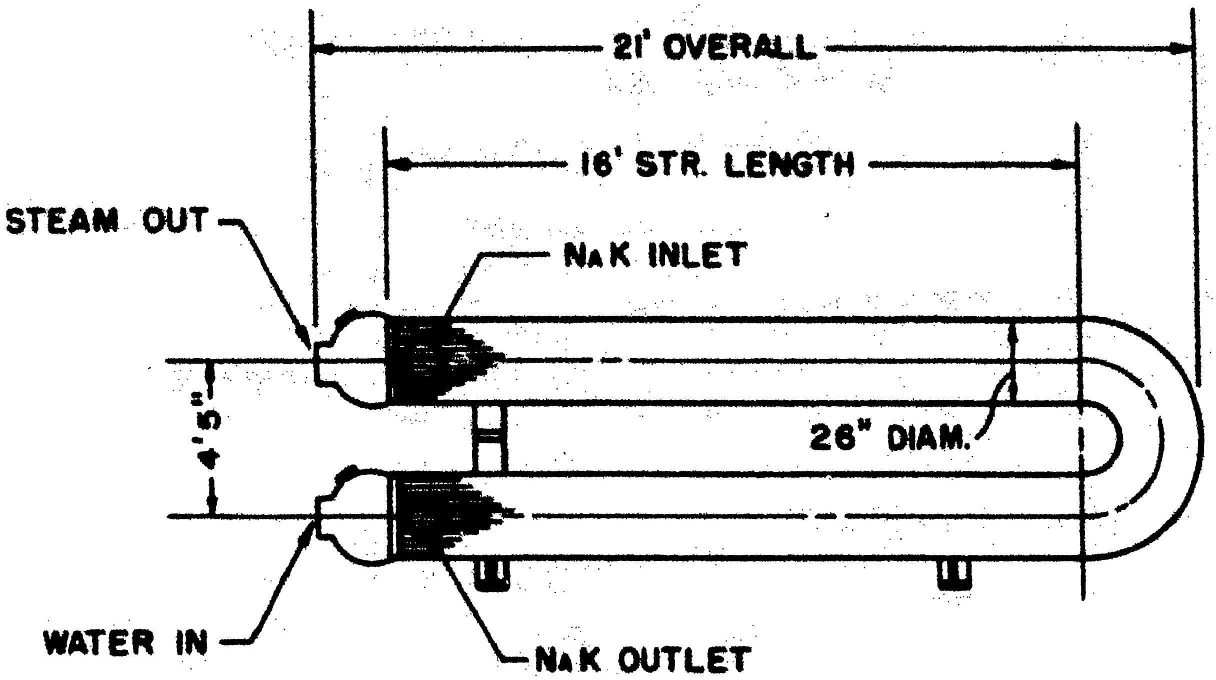
The design of a test facility to perform a series of NaK-water reaction tests is in progress. The primary functions of the tests will be to obtain pressure and temperature data for quantitive NaK-H_2O reactions caused by a tube failure, to test the emergency safety system, and to investigate effects of the NaK-water reaction on adjacent tubes when one tube in a shell-and-tube heat exchanger ruptures.

The basis for the design of this facility is as follows:

1. Small, inexpensive, shell-and-tube heat exchangers with water inside the tubes and NaK on the shell side will be used.

2. Instruments must have sufficiently fast response time to provide pressure and temperature-time characteristics data.

NUMBER OF UNITS	4
TUBE DATA	
NUMBER PER UNIT	871
OUTSIDE DIAMETER, IN.	0.5
MINIMUM WALL THICKNESS, MIL.	0.050
PITCH, IN.	0.75
EFFECTIVE LENGTH, FT.	38.2
MATERIAL: STAINLESS STEEL	TYPE 304
HEAT TRANSFER AREA, SQ. FT.	4350



ELEVATION

Fig. 15 — ONCE-THROUGH-TYPE STEAM GENERATOR

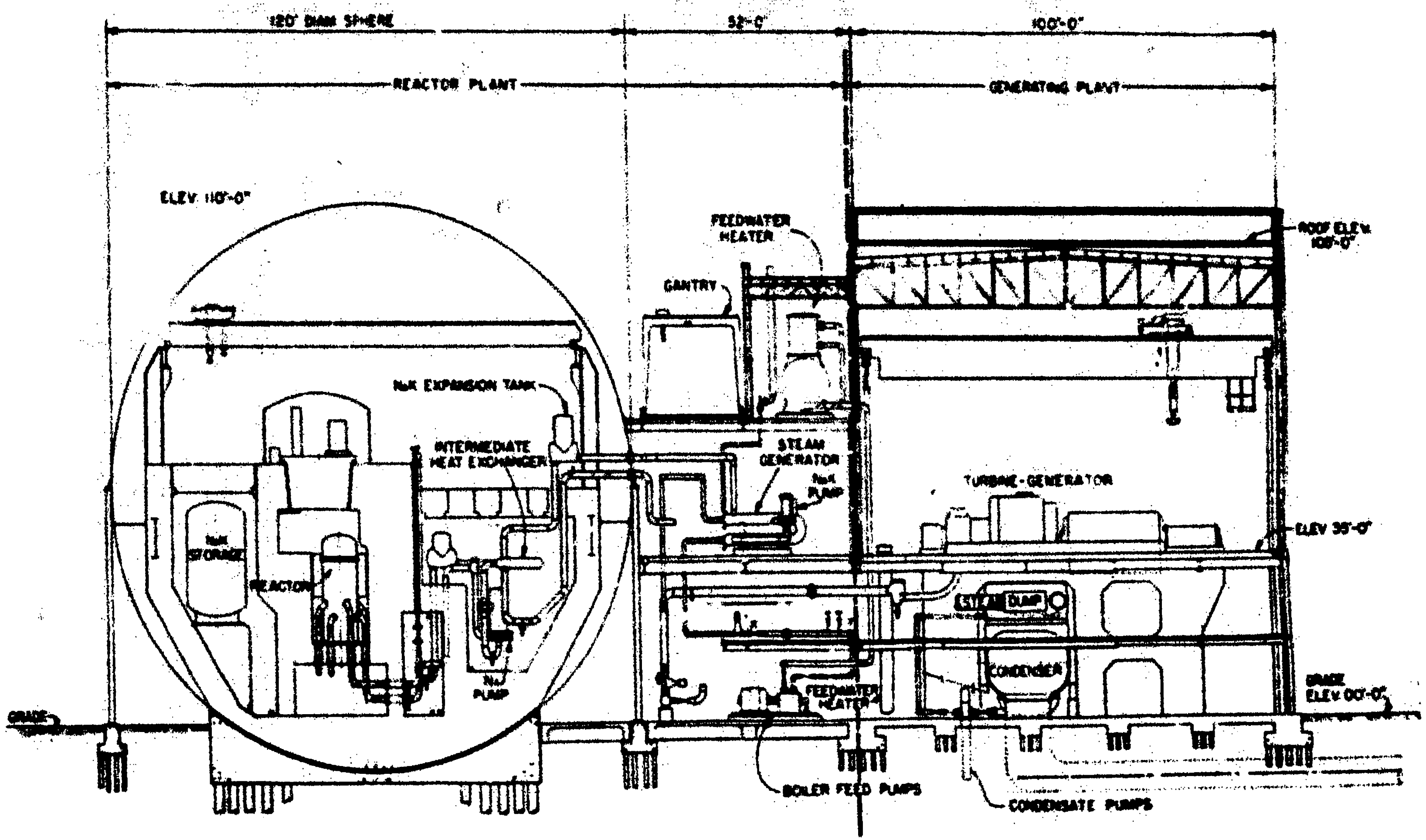


Fig. 16 — REACTOR AND STEAM PLANT (ELEVATION)

3. System equipment will provide water at a pressure of 1200 psig and at a temperature of 175 F to 400 F; NaK will be supplied over the pressure range of atmospheric to 50 psig and a temperature range of 450 F to 700 F.

4. A means of causing a tube failure in order to produce the NaK-water chemical reaction is required; for example, a tube could be notched to cause rupture at a predetermined pressure.

STEAM GENERATOR TEST

Since a once-through steam generator unit of the type being considered has never been built, a decision was made to conduct a performance test of a pilot unit of the same basic design. The objectives of the test were outlined in report DCDE-100.

The design of the boiler test facility is in progress. A schematic diagram of the boiler test facility is shown in Figure 17. The unit to be tested is the same length as that shown in Figure 15; it will, however, contain 7 tubes instead of 871. A schematic sketch of the test boiler, together with design data, is shown in Figure 18. Fixed thermocouples are built into the unit to measure NaK and tube-metal temperatures. A movable thermocouple is utilized to measure water and steam temperatures in the center tube as shown in Figure 19. The boiler will be tested at the following design conditions:

Steam Pressure, psi	1200
Steam Temperature, F	800
Feedwater Inlet Temperature, F	175
Steam Flow, lb per hr	2770
NaK Inlet Temperature, F	900
NaK Outlet Temperature, F	450
NaK Flow, lb per hr.	30,200

In addition to testing the unit at design conditions, additional tests will be run at the following conditions:

Steam Pressure, psi	600	400
Feedwater Inlet Temperature, F	420	360
NaK Inlet Temperature, F	760	666
NaK Outlet Temperature, F	510	446

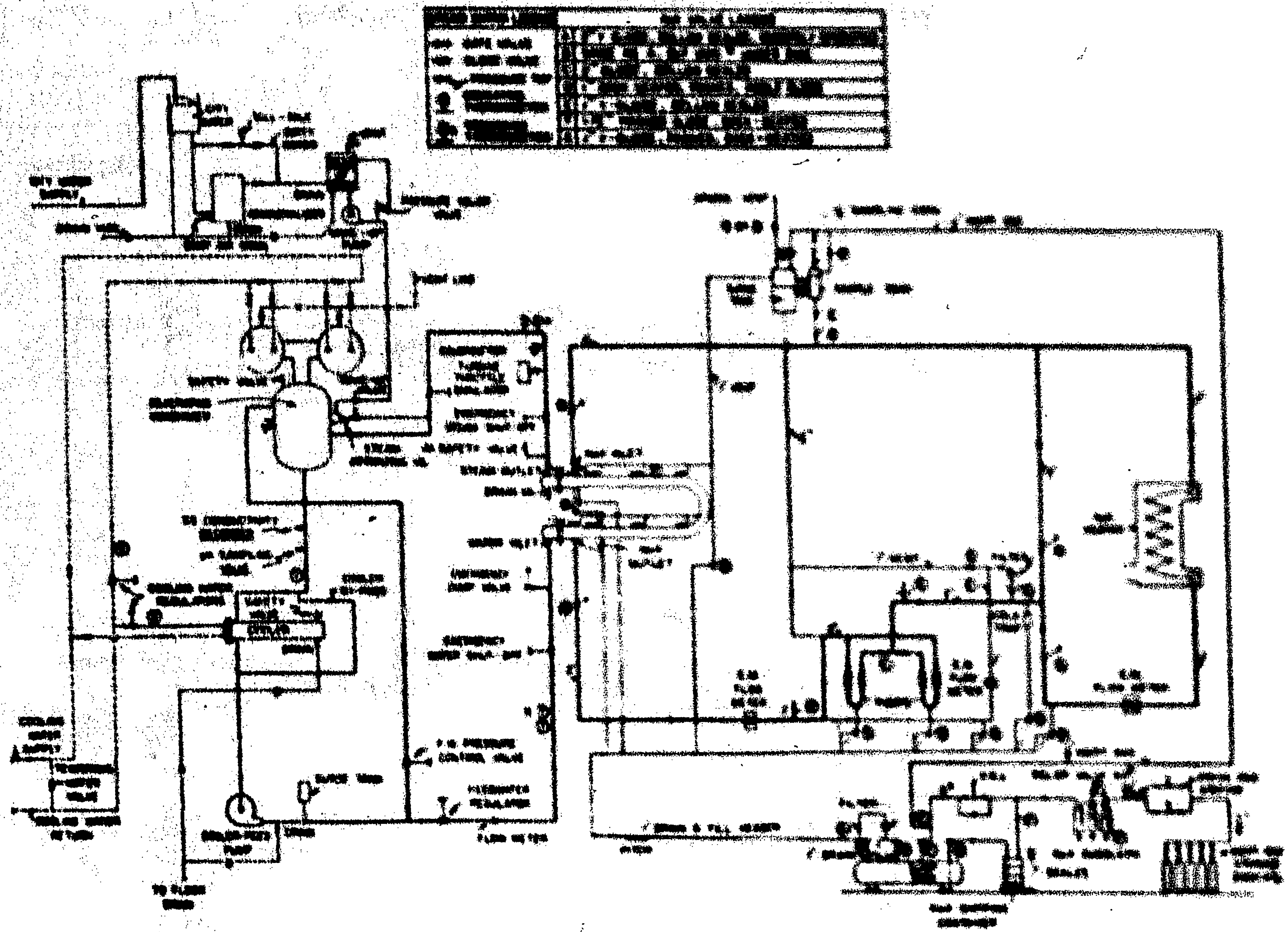


Fig. 17 – DIAGRAM OF BOILER TEST FACILITY

TUBE DATA	
OUTSIDE DIAMETER, IN.	0.5
MINIMUM WALL THICKNESS, IN.	0.050
NUMBER OF ACTIVE TUBES	7
PITCH (EQUILATERAL), IN.	0.75
EFFECTIVE LENGTH, FT.	48.1
DESIGN PRESSURE, PSIA	1300
DESIGN TEMPERATURE, F	1000

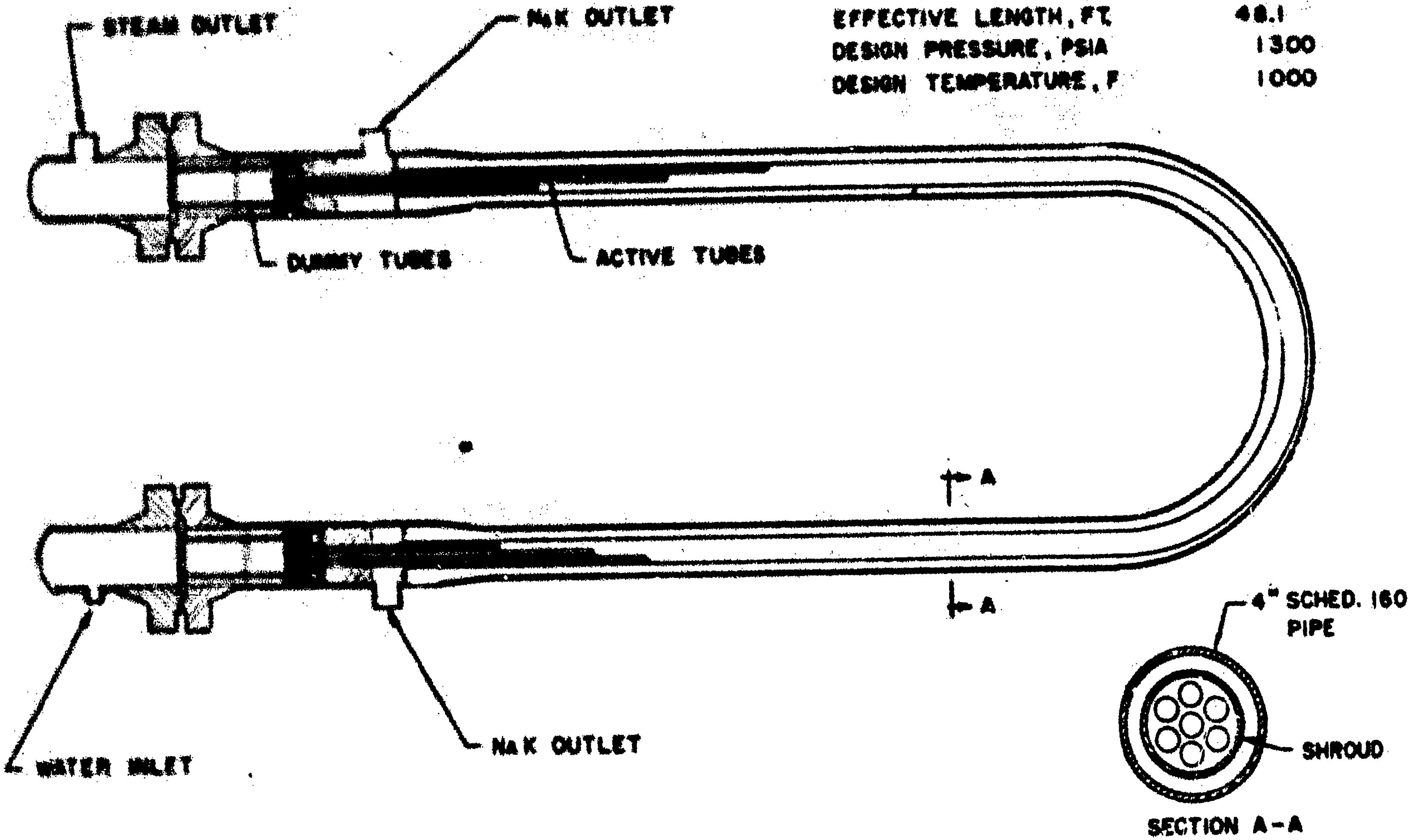


Fig. 18 — ONCE-THROUGH-TYPE TEST BOILER

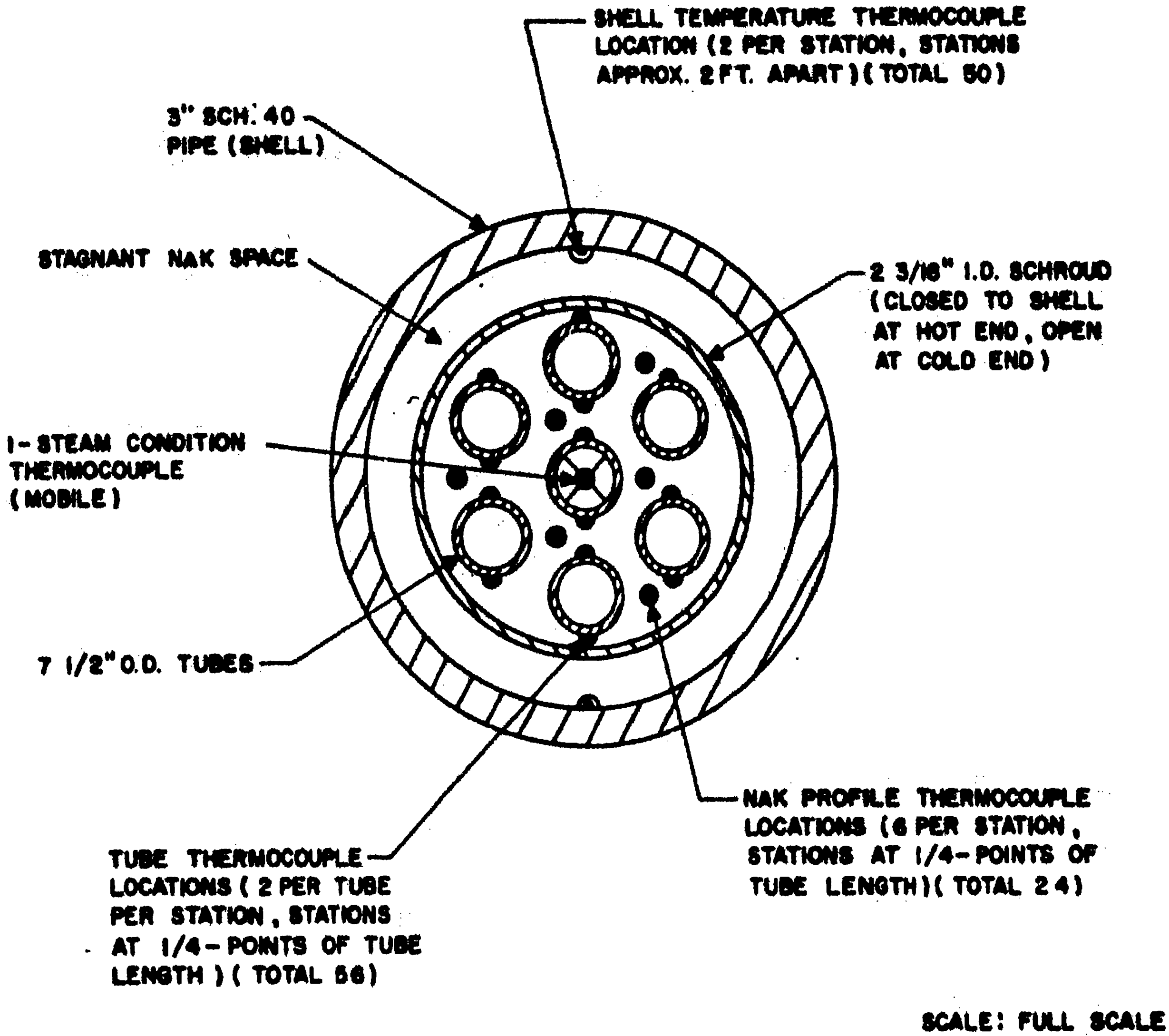


Fig. 19 — CROSS SECTION OF TEST BOILER SHOWING THERMOCOUPLE LOCATIONS

The major effort on fuel and blanket reprocessing has been an analysis of studies made to determine the relative merits of on-site and off-site reprocessing. For the on-site reprocessing studies, a longer range program, The University of Michigan Engineering Research Institute compared the cost of an aqueous reprocessing scheme (see Report DCDE-101, p. 53) with fluoride volatilization and pyrometallurgical reprocessing. Based on this comparison and on an extensive theoretical study in which ANL data was used, an evaluation of processing methods has been made that verifies earlier indications that a simple, pyrometallurgical (melting-slagging) operation offers the optimum condition for economic operation of a fast power-breeder reactor; such a plant must, of course, be backed up by an off-site, aqueous plant for "scrap" recovery.

A preliminary study and cost estimate was made for a complete off-site, aqueous reprocessing system since this mode of reprocessing may be used during initial operation of the APDA fast breeder reactor.

ON SITE REPROCESSING

Fluoride Volatility Process

The University of Michigan completed a technical and economic evaluation of the Fluoride Volatility Process as applied to the treatment of APDA fuel. The basic process essentially is the same as that which is in the pilot plant stage of development at Argonne National Laboratory.

Briefly, the process consists of dissolving declad elements in BrF_3 to form volatile UF_6, TeF_6, and IF_5 and non-volatile PuF_3 and fluorides of the remaining fission products; the non-volatile materials stay in the dissolver. The UF_6 product is separated from the other volatile components by fractional distillation and is reduced to metal by conventional methods. This basic process does not provide for the separation of Pu from fission products since it was assumed that this separation could be performed, as follows, in an aqueous solvent extraction plant: remove by volatilization most of the residual BrF_3 from the Pu and fission products in the dissolver; add aluminum nitrate as a complexing agent; send the resulting mixture to a Purex-type extraction plant where product $Pu(NO_3)_3$, free of fission products, is obtained.

The University of Michigan estimated that a capital investment of $8,000,000 would be required for a fluoride volatility plant capable of processing 75,000 pounds per year of enriched fuel and blanket materials in separate processing lines. The estimated annual operating cost of such a plant, including amortization at 10 per cent a year, is $2,650,000.

Pyrometallurgical Processes

The University of Michigan completed a technical and economic evaluation of two pyrometallurgical (PM) schemes for the processing of APDA fuels. The first process, called "Pyro-Extraction Process," assumed a continuous extraction of uranium fuel with a molten flux composed of magnesium and magnesium chloride. After extraction, the molten uranium would be cast into ingots and sent to a fuel-element fabrication plant. The flux would be continuously purified by distillation and the slag, which contains the Pu and fission products, cast into crucibles for shipment for off-site, aqueous reprocessing. The blanket elements also would be processed in off-site, aqueous facilities.

The second process, called "Pyrometallurgical Process with Aqueous Leg," assumed the same PM plant described above for processing fuel elements to which a conventional aqueous solvent extraction plant is added for processing blanket elements and PM slags.

The University of Michigan estimated the capital cost of a PM plant to be about $2,000,000; the estimated cost of the PM plant with aqueous leg is approximately $5,000,000. The predicted annual operating costs, including amortization, are about $420,000 for the PM plant and $1,340,000 for the combined PM-aqueous plant. The amortization period assumed was 9 years for both plants.

It is believed that the University's estimates are somewhat optimistic, however, these estimates confirm the APDA Working Group's conclusion that an on-site, pyrometallurgical process is the most economical approach to the reprocessing required in connection with the operation of a fast breeder reactor.

APDA PROCESS-STUDIES

Coupled with the University of Michigan results, the process studies made by APDA have confirmed the fact that the most suitable reprocessing scheme for the contemplated reactor is a simple, on-site melting and slagging operation that is backed up by an off-site, aqueous, "scrap" recovery system even though this method requires "hot" fuel element refabrication.

In evaluating the process, a study was made of the build-up of contaminants in recycled APDA fuel; the calculated values, which are based on preliminary experimental data that indicate 5% of the uranium metal would be removed each cycle as slag, are shown in Figure 20. In order to limit the contaminant build-up, particularly Mo, Ru, and Pu, to about 6%, an additional 5% bleed-off of bulk metal was taken after the twentieth cycle. For this chart, the fuel is assumed to be only uranium; the addition of alloy constituents would, of course, modify the curves to some extent.

OFF-SITE PROCESSING

Depending on the reactor construction schedule and the rate of development of a usuable PM process, the possibility exists that initial fuel reprocessing will have to be done in an off-site, aqueous plant. For this reason, the feasibility of reprocessing the core of an APDA-designed reactor in the Chemical Processing Plant at The National Reactor Testing Station has been investigated with the Idaho Operations Office and the Phillips Petroleum Company; it was ascertained that some minor changes and two major additions to the plant would be required in order to utilize the CPP facility for this purpose. A new dissolver section would be needed due to fuel composition, capacity, and criticality considerations. The other major addition would be a complete plutonium extraction, purification, and shipping section. The addition of these facilities to the existing plant appears technically feasible; and their cost, which is approximately 5% of the total plant cost, is not excessive. No insurmountable problems are expected with the chemistry of the process, but certain research will be required to pinpoint process conditions.

The total cost for this mode of reprocessing, which includes shipping, amortization, and inventory charges in addition to the direct charge for reprocessing, appears prohibitive for anything but an interim arrangement. On an interim basis, that is, during initial plant operation, the utilization of these facilities may be necessary.

Conceptual designs have been prepared for the transfer and shipping facility, to be located adjacent to the reactor building, that will handle partial dissassembly, shipping, and receiving of core and blanket elements. The design of this facility includes provisions for:

Inspection - Checking elements for dimensional stability after irradiation.

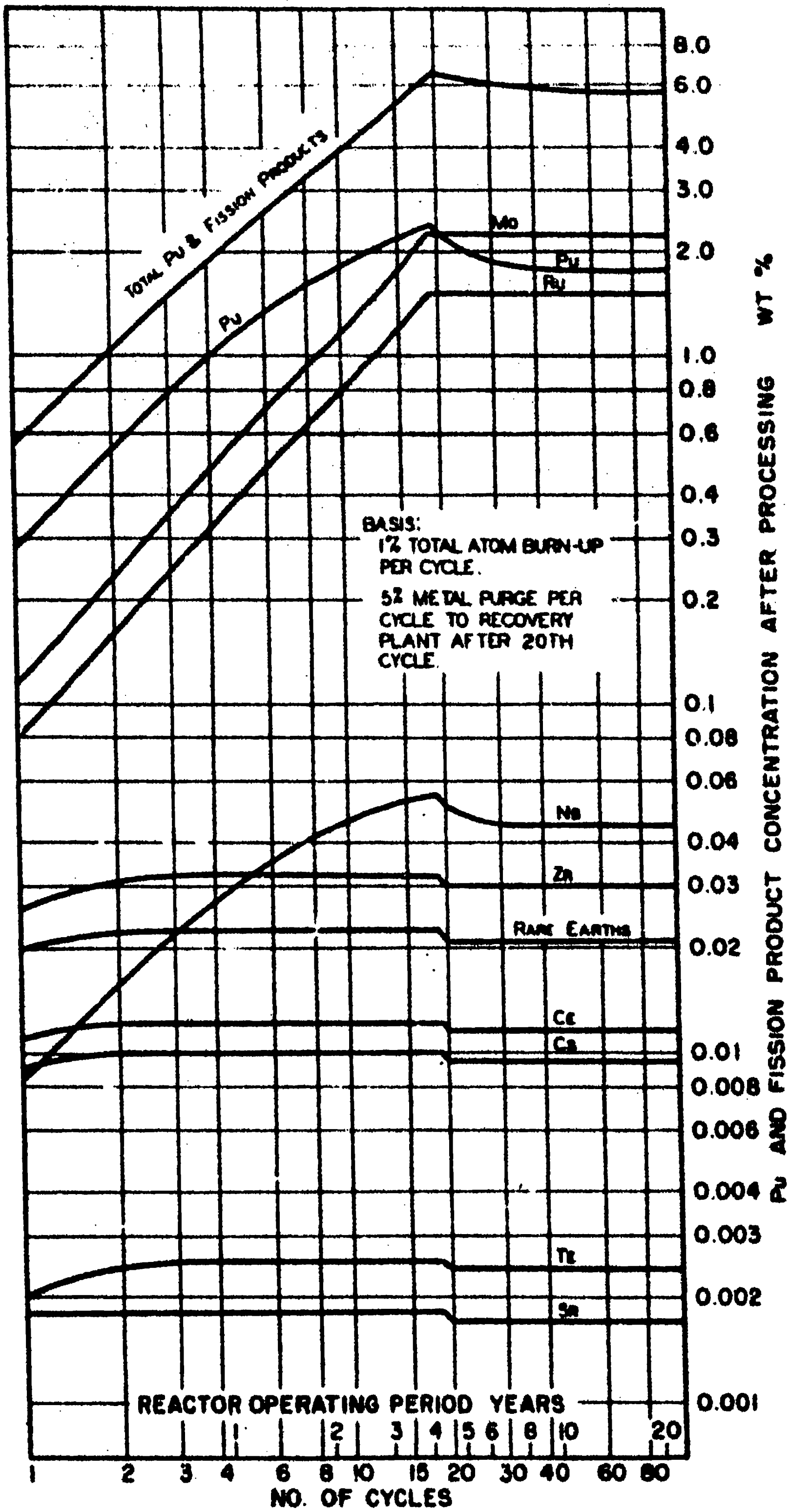


Fig. 20- Pu AND FISSION PRODUCT BUILD-UP FOR OXIDE SLAGGING PROCESS

Disassembly	-	Separating core and blanket elements prior to shipment to their their respective processing sites.
Pot Loading	-	Loading and capping the transfer pot that would be placed in the shipping cask.
Shipping	-	A shielded railroad area with shipping-cask handling equipment.

The facility was designed to be operated and maintained completely by remote means. Since this work area must have an inert gas atmosphere, all transfers to and from the area are made through air locks. The design is such that this facility could be expanded to include a decanning operation and/or a complete reprocessing plant if deemed desirable.

FULL-SCALE TEST FACILITY

A decision was made in July 1954 to construct a test facility that will be a full-scale prototype of the final design and layout of the reactor and one sodium loop. This non-radioactive unit is considered necessary for testing final components and coordinated assemblies as well as for training personnel for the reactor plant.

Space at the Delray Station of The Detroit Edison Company has been approved as the site for this facility. The equipment will be installed in a portion of Warehouse No. 6 that was formerly occupied by No. 10 Turbine-Generator Unit; clearing of this space is in progress.

Tentative layouts of the equipment have been made, and design of the NaK control system is under way. Service facilities are being provided. Operation of this test facility is scheduled to begin approximately one year after the design of the reactor plant is frozen. Based on the present schedule, the facility should be in operation in the summer of 1956.

DESCRIPTION

The full-scale test facility will consist of the reactor pressure vessel, one 14-inch diameter sodium loop, and auxiliary equipment as shown in Figure 21. Since only one of the four sodium loops of the reactor plant will be used in this facility, the reactor vessel will have three-fourths of its internal volume and flow area blanked-off with dummy fillers; the active quarter will contain simulated fuel elements, control rods, and blanket elements. The mechanical handling plug, installed over the reactor vessel, will be equipped with several control rod actuators and the remote operated fuel element handling device. The single sodium loop will be equipped with an 8600 gpm prototype pump as well as appropriate valves, surge tank, sodium storage and gas sealing systems. The intermediate heat exchanger will function only as a temperature controller and will, therefore, be only a small coil transferring heat between the sodium and the control NaK loop.

Essentially, the facility will be operated as an isothermal test loop, deriving the major part of its heat input from pump energy. The NaK control loop will be capable of removing excess heat to maintain temperature levels of 450 F to 1000 F; it will add heat to the sodium loop during start-up and low flow rate operation since, at those times, the pumping energy would be insufficient to heat sodium to the desired

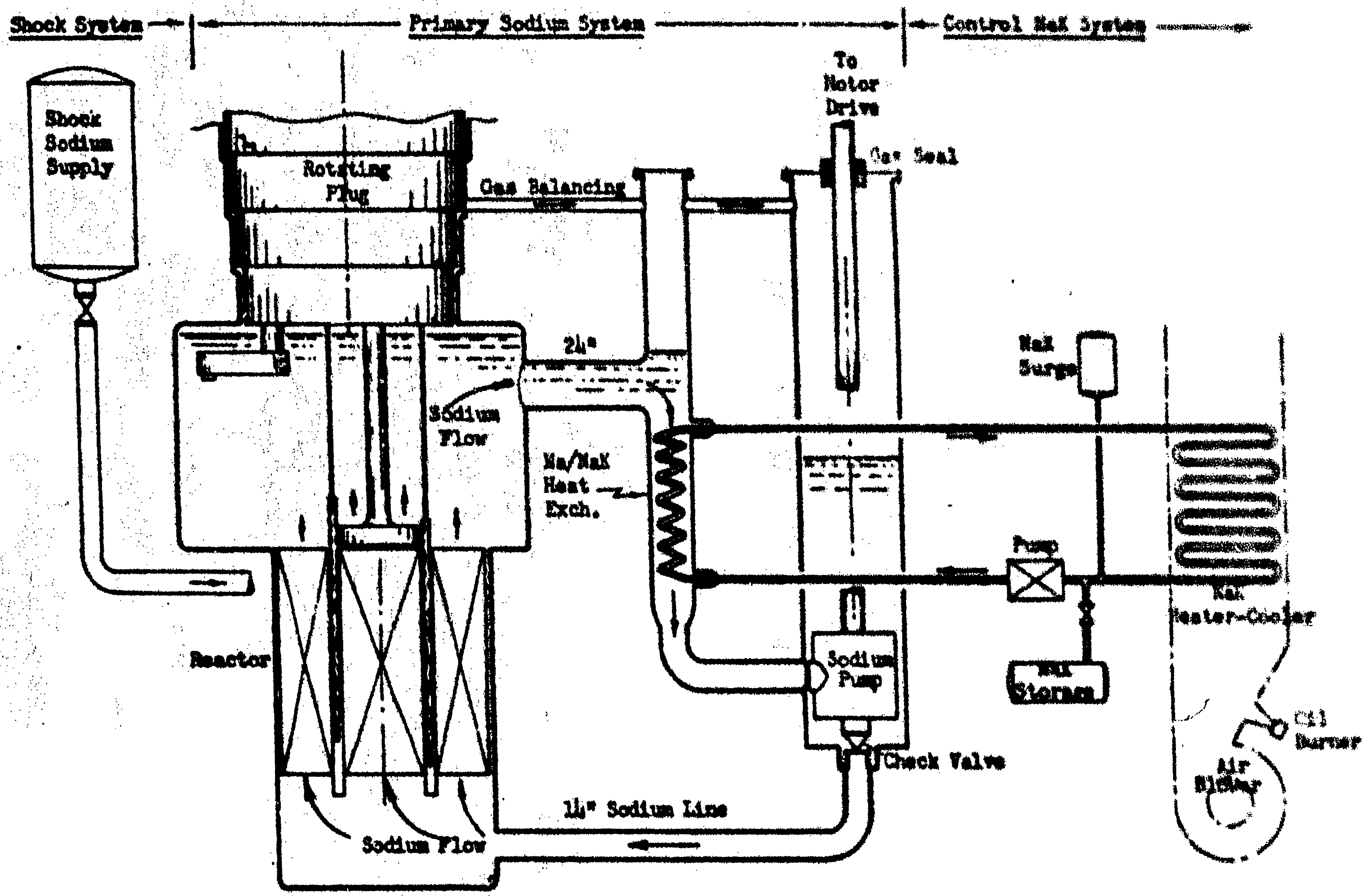


Fig. 21 — FLOW DIAGRAM OF FULL SCALE TEST FACILITY

temperature. It also will be desirable to test under conditions of transient temperature changes anticipated for the reactor plant. Either the sodium surge tank or an auxiliary shock supply tank will be used to introduce the desired shock temperatures into the sodium system for about a 30-second duration.

The present plan for fire protection for this relatively large sodium system is to put a containment tank and piping outside the insulation of the main reactor vessel and the 14-inch sodium piping. Under normal operation, air will be circulated through the resulting annulus; in the event of a sodium leak and fire, the ventilating air would be shut off and an inert gas introduced to snuff out the fire.

FUNCTIONS

In addition to training personnel in a "cold" facility on equipment that duplicates major components of the reactor plant, it is expected that the mock-up will be used to test and evaluate the following items:

1. Sodium system performance including system pressure drops, temperature distribution, flexibility, filling and draining procedures, sodium inertia, and sodium impurity control.
2. Reactor control system performance including scram time and resulting vessel stresses.
3. Mechanical handling system performance including the effect of distortion on indexing.
4. Shock-temperature effects on all components.
5. Diffusion-bonding of fuel elements.
6. Over-all coordinated functioning of the mechanical handling system, control system, and other reactor components.
7. Miscellaneous tests that appear desirable from time to time.

SITE AND SAFETY STUDIES

SITE AND PLANT LAYOUT

A tentative site on which to place a reactor power plant was chosen and various layouts of the arrangement of structures on the site were made.

The plant site, a filled plot of approximately 20 acres, is located in a swampy area on the northwest shore of Lake Erie. The swamp or marsh is not land-locked but is protected from the lake by sand bars and small islands. It drains into the lake via large shallow bodies of water. A causeway would be built to provide access by road and rail and to serve as a right-of-way for the transmission line. Condenser cooling water would come from a dug canal extending into deep water in the lake and would be discharged into the swamp.

The nearest center of population is three and one-half miles away. In order to have adequate exclusion area for protection and public safety, about 1,200 acres of swamp should be controlled. This would prohibit anyone from living within three-fourths of a mile of the reactor.

HEALTH PHYSICS

The first phase of the program to design and develop satisfactory health-physics instrumentation, a survey and evaluation of present commercial instruments, is near completion. Until the results are known, it is impossible to determine future work required, although it appears that certain weaknesses occur in existing designs, particularly for fast neutron and air monitoring instruments.