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CIVILIAN POWER REACTOR PROGRAM

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Book 6

Status Report on Sodium Graphite Reactors As of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

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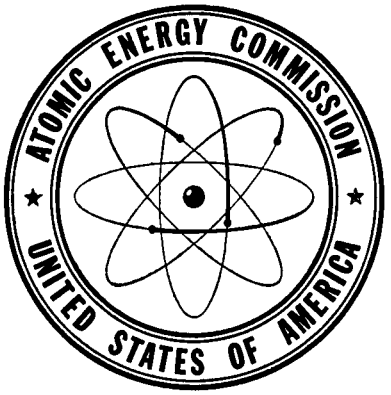

CIVILIAN POWER REACTOR PROGRAM

PART III

Status Report on Sodium Graphite Reactors as of 1959

Published: 1960

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SUMMARY

This report describes the current development status of the Sodium Graphite Reactor (SGR) concept. The development history is summarized and all important areas of development are discussed.

Development effort in the SGR dates from 1949 when analytical evaluation studies were started. Effort on sodium and NaK components had started previously, under the programs for development of the Submarine Intermediate Reactor (SIR) and the Experimental Breeder Reactor (EBR–I), and data available from these programs were used in the SGR evaluation work.

The discussion of the SGR development program is broken into three categories, (1) general research and development, dealing with reactor physics, fuels and materials, components, etc., (2) experimental reactors, operating or in some phase of design or construction, and (3) power demonstration reactors, operating or in a design or construction phase. These categories are not independent; e.g., an important part of the physics development has been accomplished through use of the Sodium Reactor Experiment (SRE), the only operating SGR.

Research and Development.—In the area of reactor physics, the important out-of-pile measurements have been made in exponential assemblies. Data obtained are thermal flux distributions, resonance escape probability, and buckling. The measurements were made with a range of fuel-to-moderator ratios, fuel enrichments, and rod diameters. Some measurements on single fuel elements in a block of graphite have also been made in an attempt to reduce the cost of exponential experiments. An SGR critical experiment facility has been built and the first experiments will be conducted in fiscal year 1960. One of the important measurements planned for this facility is the Doppler coefficient, as a function of temperature, in U–238 and thorium.

The physics experiments on SRE have been particularly fruitful in establishing data on reactor safety. Measurements have been completed, using reactor oscillator techniques, which firmly establish that the SRE is unusually stable, and which demonstrate the advantage of small overall temperature and power coefficients combined with an effective prompt fuel coefficient which is negative. Long term reactivity changes have also been measured in SRE, to about 1,000 MWD/T and the effects on critical mass and conversion ratio of modifications to the basic 7-rod fuel cluster have been studied.

The major effort in materials development has been concerned with fuels. The first core loading in SRE was fabricated from unalloyed uranium. A variety of test elements made of uranium alloys, both wrought and cast, have also been tested in SRE. The first core is reaching the end of its useful life, after exposure to about 1,000 MWD/T. Swelling of the uranium metal will soon force a core change, and a Th-U alloy loading has been fabricated.

The fuel materials development at SRE has been augmented by capsule tests in the Materials Testing Reactor (MTR). The general failure of metallic alloys to meet the requirements of long life at the desired operating temperature in an SGR, demonstrated in the SRE, is also concluded from the MTR experiments. Exceptions to this general failure are Th-U alloy and U–1.5 w/o Mo, but it is expected that the temperature limits on these materials, which are not yet firmly established, may be too low to give satisfactory performance in an advanced SGR.

The high coolant temperature and excellent heat transfer capabilities of sodium, major items in the appeal of the SGR concept, puts rather stringent requirements on fuel materials. Oxide is satisfactory in its ability to operate for extended periods at high temperature; it also

has satisfactory corrosion resistance in hot sodium. However, its thermal conductivity is undesirably low. The most promising fuel material for an SGR is the uranium monocarbide. Its density is somewhat higher than that of the oxide and its thermal conductivity nearly equal to that of the metal. Experience with this material is very limited. A few specimens irradiated in the MTR at maximum temperatures at or above 1,600° F. have demonstrated good resistance to radiation damage to the maximum exposure to date, about 6,000 MWD/T. Fission gas retention was excellent, and development of the monocarbide is being vigorously pursued. Currently, the most important development problem of the SGR program is a satisfactory fuel element. On the basis of the data available, the emergence of carbide appears to constitute a major advance in SGR development.

For construction materials, stainless steel is used almost exclusively in most of the current SGR designs. There is no important corrosion problem. Vessels are thin-walled and schedule 10 piping is commonly specified for piping systems such as those in the Hallam Nuclear Power Facility (HNPF) reactor. The major unknown in this regard is the problem of mass transfer at temperatures above 1,100° F. This is a problem from the point of view of transfer of radioactivity and is not a potential plugging mechanism.

Zirconium is performing satisfactorily in the SRE as a cladding for the graphite. The unalloyed material used in SRE is believed to have a temperature limit, fixed by creep strength, which would restrict sodium temperatures to about 1,050° F. for extended operation. Zirconium alloys are being developed which are comparable to 304 stainless in tensile strength at 1,050° F., however, and should soon be available for use in core structure. Generally speaking, zirconium cannot compete with stainless in terms of overall costs and has therefore been omitted from the Hallam reactor and tentatively omitted from the design of the advanced SGR.

Coolant chemistry has introduced no special problems in the reactors operated to date. Removal of oxide is necessary to prevent corrosion. A concentration of less than 30 ppm appears satisfactory in this regard. It has been easy to maintain a concentration of 10 ppm or less in the SRE with inexpensive, easily operated equipment.

The nuclear stability of the SGR has been intensively investigated in the SRE with very encouraging results. Much of the information developed has been used directly in the hazards analyses of the Hallam plant. One very useful result of the SRE experiments is the demonstration that the reactor transfer function can be accurately predicted by analytical methods, and the techniques and SRE results are applicable to some degree to all SGR's.

Satisfactory components for the sodium cooling system are currently available with the single exception of the steam generator. The cooling system for the NaK cooled Experimental Breeder Reactor (EBR–I) was operated without incident for a number of years.

The steam generator on the SRE has operated without incident for about 2 years. It uses double-wall tube construction with an intermediate fluid, mercury, in the annulus. An advanced type of double-wall steam generator, tested and proved under the SGR program, will be used in the HNPF. Currently the most advanced current steam generator design appears to be that used in the fast breeder reactor under construction by the Atomics Power Development Associates (APDA). This design, using single wall tubes, is based upon experiments conducted by APDA which indicate that failure of a tube will not be catastrophic. A rupture disk is incorporated into the design to permit the reactants to be exhausted from the steam generator without harmful effects on the rest of the system.

The development of satisfactory steam generators is well underway. The remaining problems are in the development of designs which will readily accommodate the tendency of sodium to introduce unusual stress problems, and which are moderately priced. The major effort required for other components such as control elements, valves, pumps, flow meters,

level indicators, etc., is that of increased reliability, simplified design, and cost reduction.

Experimental Reactors.—While EBR–1 and SIR have provided much information on the operability of liquid metal systems, the only operating sodium graphite reactor is the SRE. The reactor is of the "pool-type, canned moderator" design. It was designed to operate at 20 MWt and has been limited to a maximum of about 22 MWt by the heat sink provided. It was designed to dump heat in a sodium-to-air heat exchanger, but during the course of construction, a steam generator and turbine system were added by the Southern California Edison Co.

The SRE was built to test the SGR concept. It has been used principally as a fuel element and sodium component test facility. Immediately after startup, power was limited to 8 MWt by a thermal stress condition at the outlet nozzles of the reactor vessel. This condition was produced by excessive convective sodium flow following a scram. This was corrected by adding eddy-current brakes to the sodium piping. Since this modification, operation has been interrupted occasionally by failure of the organic cooled freeze seal on the primary sodium pumps. This condition appears now to have been solved by substitution of a NaK-cooled freeze seal.

The reactor is operated primarily as an experimental facility with power generation considered incidental. A number of physics tests have been conducted including pile oscillator measurements at full power. The fuel element program has been pursued intensively. Approximately 600 nondestructive fuel element examinations have been made in the adjacent hot cell. During the course of cleaning fuel elements for examination, a procedure involving a water wash, stainless steel cans on the fuel slugs have occasionally been collapsed, rendering them unsatisfactory for further exposure in the reactor. More satisfactory cleaning procedures are currently being sought. Important modifications to the SRE cooling system have been made in system simplification including elimination of most of the valves. Tests on the SRE have indicated many areas in which manufacturing tolerances can be relaxed. Valuable data on permissible thermal gradients have been developed, since the SRE has operated with a Δt of up to 480° F. across its 6-foot core. Tests have demonstrated that the power level can be varied safely at a rate of 20 percent per minute and that the reactor can readily be operated as a load-following heat source. Steam at 900° F. has been generated for an extended period at full power and 1,000° F. steam was produced for a short time at reduced power.

In spite of the emphasis on experimental activities, the SRE has logged about 2,200 MWD of operation and generated over 15,000,000 kw-hr of electrical energy. It has been and continues to be a valuable source of information for SGR development.

A second experimental SGR, the Sodium Cooled Reactor Experiment (SCRE) has been proposed which incorporates the lessons from the SRE and from the SGR development program into an advanced design. This reactor is of the "pool type, calandria-core" design. An important feature is the elimination of piping and piping galleries from the primary sodium system. The entire loop is contained in the reactor vessel. This reactor is in the preliminary design stage.

Power Demonstration Reactors.—The only power demonstration reactor in the SGR program is the Hallam Nuclear Power Facility for which ground was broken in April 1959. This reactor is identical to the SRE in concept. The zirconium cladding on the graphite has been eliminated in favor of stainless steel in this reactor. The first fuel loading will be fabricated of U–10 w/o Mo alloy with a proposed second core of uranium carbide.

HISTORY OF CONCEPT

In 1949, North American Aviation, Inc. (NAA), initiated the concept of the sodium graphite reactor with a study of various fuel, moderator, and coolant combinations for a reactor which would produce plutonium at minimum cost, and would utilize materials and technology available, at least to some degree, in United States industry.

Natural uranium systems were first investigated, but in 1950 it was demonstrated that slightly enriched fuel significantly improved performance. The study indicated that both plutonium and useful electric power could be produced by a thermal, heterogeneous reactor with low enrichment uranium as fuel, graphite as a moderator, and liquid sodium as coolant. As this study was completed, emphasis on reactor systems in the United States shifted from dual-purpose plutonium producers to power-only reactors. The previous work on SGR systems was reviewed by North American Aviation, and the SGR appeared attractive for this single objective also.

Simultaneously, development work on a sodium-cooled epithermal reactor was progressing under naval auspices, as an outgrowth of an epithermal breeder concept originated by Knolls Atomic Power Laboratory (KAPL) in 1946. Epithermal reactor physics, sodium technology and sodium component development were undertaken on a very substantial scale in what has been known as the Submarine Intermediate Reactor Program.

Additional sodium system and component work was undertaken by Argonne National Laboratory (ANL) in the period 1948 through 1952 in support of the EBR–I program. Although this reactor is NaK cooled, the problems are very similar, and a considerable technology of benefit to the SGR concept was developed. These development efforts were synthesized by NAA as much as security regulations at that time would permit. Certain development areas applicable to SGR's, which were not being covered by these programs, were then decided upon. These areas lay chiefly in problems associated with the higher top temperatures and higher temperature rises proposed for SGR's than were contemplated for either intermediate or fast reactors, the utilization of graphite in connection with a sodium-cooled reactor, and utilization of system components which more closely resembled American industrial practice than the highly specialized equipment developed for Navy use.

The reactor configuration decided upon by NAA, during 1952, as best suited for the initial SGR effort was a single-pass, sodium-cooled reactor with hexagonal graphite logs canned in zirconium. Fuel, selected on the basis of the best metallurgical information available from ANL and other sources in 1953, was unalloyed alpha-rolled, beta-heat-treated metallic uranium slugs enriched to 2.778 atom percent U–235. Coolant channels containing fuel elements were placed in the center of the graphite logs for convenient fabrication. This design was accepted by the AEC as a part of the original 5-year reactor development program, and in June 1954 a joint effort between NAA and the AEC was initiated. Intensive design of the sodium reactor experiment was started at that time and in April 1955 actual construction of the plant began.

Design of the SRE was influenced by two main considerations; provision of as flexible a reactor experiment as possible to provide the maximum amount of experimental information, and the use of standard equipment (or adaptations) wherever possible.

Major technology development programs in support of the SRE included investigations of graphite, cladding materials, and fuel elements, and the performance of these components in a

high temperature sodium system with the additional complication of a high temperature differential. Component development was limited chiefly to modifying commercial equipment for use in sodium systems, and proof under proposed SRE conditions of components developed under other sodium reactor programs. The technology of sodium and of materials in contact with sodium developed by SIR and EBR programs was followed closely during design and construction of the SRE, and developments from these programs were incorporated wherever possible. No extensive work was undertaken on the chemistry or thermodynamics of sodium, since efforts along these lines were in progress at both KAPL and ANL. Materials technology was taken directly from the SIR and EBR programs, as time would not permit extensive investigation of corrosion, erosion, or compatibility of construction materials prior to their selection and fabrication for the SRE.

Initially the SRE was planned solely as an experimental installation and no effort was made to utilize the power it would produce. An air blast heat exchanger was installed to dissipate the 20 MW of heat generated at design full power. In April 1956, the Southern California Edison Co. offered to install a steam generator and turbine system; this installation was made and at present either means of absorbing reactor heat can be used.

EBR—I commenced operation in 1951, and valuable information on the performance of liquid metal systems was obtained from this source. Generally, however, the heat transfer system components of this reactor were designed for the absolute maximum of reliability, with cost and size as secondary considerations. This was, of course, absolutely proper as uninterrupted performance of the reactor itself was of prime importance, and the heat rejection system more of a necessary nuisance than a significant part of the reactor experiment. The SIR Mark A prototype commenced operation in 1955, and almost immediately ran into difficulties in the steam generating system which limited the reactor complex to about 40 percent of full power. This reactor was operated for approximately 2 years.

The SRE was completed in March 1957, with first electric power generation in July 1957. Certain design deficiencies, associated with post-scram thermal stresses, were evident and full power operation was achieved in May 1958, after installation of additional equipment (eddy current brakes) to counteract the conditions causing excessive stress. The reactor has been operated for experimental purposes with no restrictions on power level up to 20 MW since that time.

In 1958, design was started on the Hallam Nuclear Power Facility, a 75 MWe sodium graphite powerplant for the Consumers Public Power District of Nebraska. Construction commenced in the spring of 1959, with completion scheduled for 1961. This reactor plant is essentially a scaled-up version of the SRE, incorporating the lessons learned from SRE operating experience.

The Enrico Fermi Power Plant fast breeder reactor and EBR–II programs were initiated in 1955. Development work associated with these programs is continuing today, and close association between NAA, APDA, and ANL assures an exchange of information on all facets of reactor technology, sodium technology, and component development.

A further concept, initiated by Atomics International, a Division of North American Aviation, Inc. (AI) in 1958 and supported by a group of private utilities, is an extension of the SGR concept. This is the Advanced Epithermal Thorium Breeder, a 360 MWe, sodium cooled reactor system operating on the U–233 thorium cycle. This reactor system, now in the active study stage, holds promise of breeding more U–233 from thorium than is consumed in producing power. Thus, current and past experience in all sodium cooled reactor programs are being factored into an SGR plant under construction today, and advanced concepts promise further cost reductions and improved neutron economy in the future.

A. CONCEPT DESCRIPTION

The SGR is a sodium cooled, graphite moderated, thermal neutron spectrum reactor. It is characterized by a low pressure heat removal system, with a high temperature rise in the coolant in passing through the reactor. A temperature rise of 300° to 600° F. is practical, thus reducing the pumping power necessary to remove reactor heat. A heterogeneous solid-fueled reactor in its present form, this system does not offer significant possibilities of breeding. Conversion ratios between 0.50 and 0.65 can be expected with reasonable mechanical designs, using fuel of low U–235 enrichment. High temperature fuels are required; in order to match the excellent heat removal characteristics of sodium, heat fluxes of the order of 1 to 3 million Btu/hr/sq-ft should be employed. Fuel material of high thermal conductivity is required to take advantage of the properties of sodium. Currently, the uranium monocarbide appears to offer the greatest possibility of long term radiation stability and high temperature performance. A vigorous uranium carbide development program is in progress.

Forced convection of the sodium coolant is required. Natural convection, of course, assists in circulating the sodium and is ample to remove decay heat from the reactor after a scram.

A two-loop sodium coolant system is employed to insure against a sodium water reaction with radioactive sodium, to exclude hydrogen from the reactor core and to avoid radiolytic decomposition of the boiler water in the steam generators by gamma radiation produced by Na–24 in the primary sodium system. Heat is transferred from radioactive sodium to nonradioactive sodium in an intermediate heat exchanger, and this nonradioactive sodium is in turn used to generate superheated steam. This serves the additional purpose of reducing shielded volume, as the radioactive sodium circuit can be compressed into as small a space as is feasible from engineering considerations. The nonradioactive secondary system may be arranged in any convenient configuration with no particular attention to radioactivity considerations. These benefits are not without cost; the additional coolant loop increases capital costs and results in a drop in steam temperature.

Several core configurations have been proposed, based on methods of separating sodium from graphite. The SRE and Hallam are "canned moderator" types, in which graphite logs are individually packaged and cooled by sodium. The "calandria" concept uses one large container pierced by fuel tubes, thus eliminating sodium from moderator regions. A third variation, the "re-entrant tube" type, with hot and cold sodium plenum chambers adjacent to each other, is being studied by General Electric.

GENERAL RESEARCH AND DEVELOPMENT

1. Experimental Physics

a. Physics Experiments Completed

Physics experiments pertinent to the SGR program include static measurements on subcritical lattices and static and dynamic experiments on the SRE. The fuel geometry of the SGR, consisting of multirod clusters, makes theoretical predictions of the flux distribution difficult. A considerable amount of information on intracell flux distributions has been obtained from a series of exponential experiments on lattices containing both Th-U alloy and slightly enriched uranium fuel (Tables I, II, and III and NAA–SR–3096). The results of these measurements are useful in predicting critical masses, conversion ratios, disadvantage factors, etc.

A related effort in subcritical experiments consists of measurements on single fuel elements in a graphite matrix, in the thermal column of a research reactor. These experiments yield essentially the same data as can be obtained from a full exponential experiment but with somewhat reduced accuracy. They are expected to be useful as an inexpensive means of screening possible fuel geometries; the technique is still being perfected using equipment now available at Atomics International.

The important SGR experimental physics information obtained to date has been acquired at the SRE. Experimental measurements of the isothermal temperature coefficient, control rod worths, and low power flux patterns have been made. The results of these measurements are described in NAA–SR–3341. A plot of the SRE isothermal temperature coefficient of reactivity versus temperature is shown in Figure 1. The reactor oscillator technique has been utilized to obtain the reactor transfer functions at power levels ranging from zero to full power. From these data certain parameters pertaining to the dynamic stability of the SRE have been established.

TABLE I.—MATERIAL BUCKLING OF GRAPHITE LATTICES

Fuel element			Assembly dimensions			Buckling, m^{-2}
Number of fuel rods	Rod diameter, inches	Fuel enrichment, w/o	Number of cells	Width, inches	Lattice spacing, inches	
7	0. 75	2. 78	36	42	7	7. 81 ±0. 21
7	0. 75	2. 78	25	47. 5	9. 5	9. 67 ±0. 17
7	0. 75	2. 78	16	48	12	8. 65 ±0. 09
6*	0. 75	2. 78	36	42	7	7. 69 ±0. 25
6*	0. 75	2. 78	25	47. 5	9. 5	9. 76 ±0. 16
6*	0. 75	2. 78	16	48	12	8. 37 ±0. 11
7	0. 75	Natural	36	42	7	−1. 51 ±0. 15
7	0. 75	Natural	36	57	9. 5	−0. 35 ±0. 15
7	0. 75	Natural	16	48	12	0. 526±0. 15
4	1. 00	0. 91	36	42	7	−0. 77 ±0. 64
4	1. 00	0. 91	36	57	9. 5	1. 67 ±0. 28
4	1. 00	0. 91	16	48	12	2. 24 ±0. 14

*¾-inch graphite rod filling central channel of element.

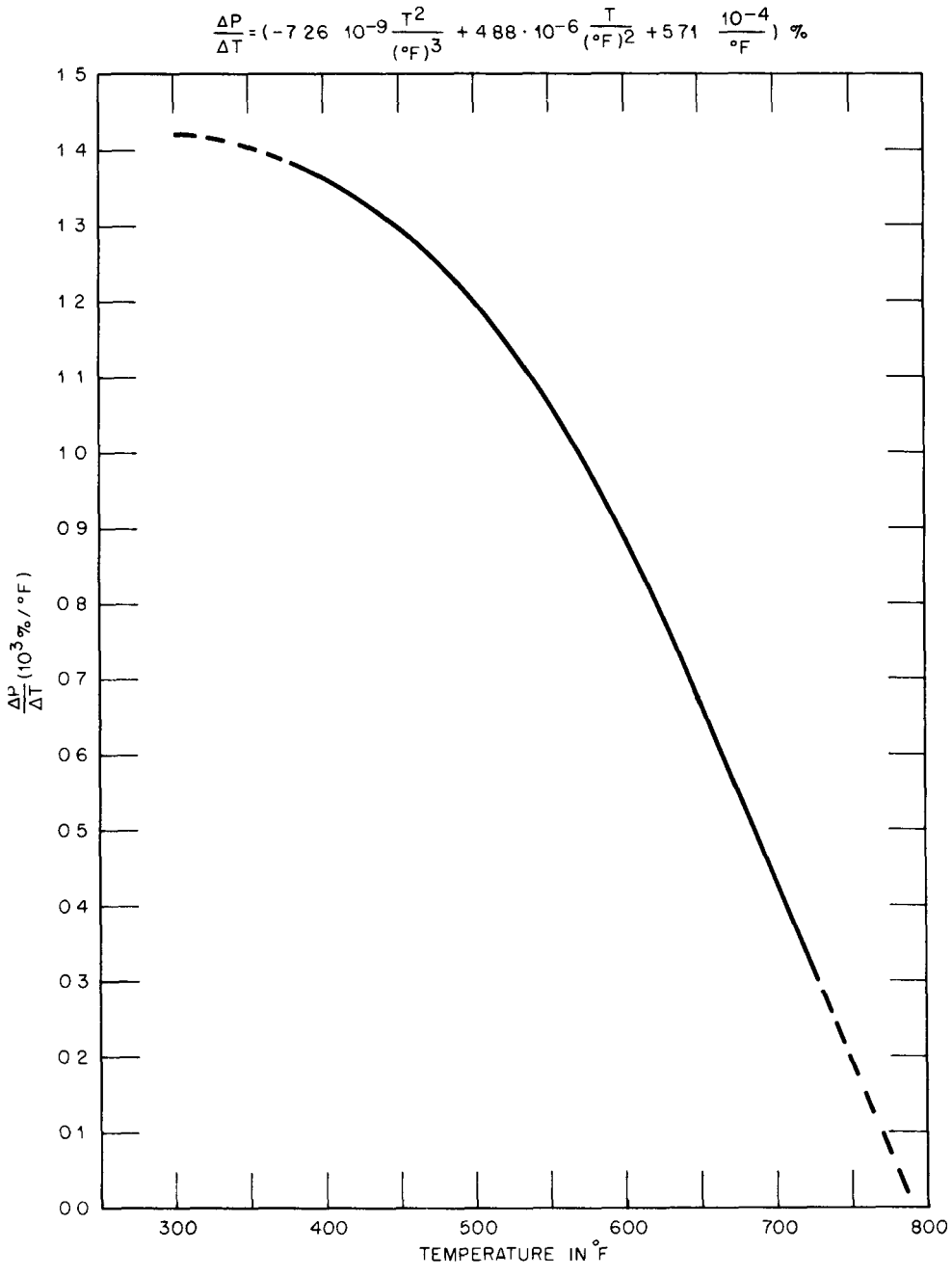

FIGURE 1.—SRE isothermal temperature coefficient of reactivity.

TABLE II.—RESULTS OF INTRACELL FLUX DISTRIBUTION MEASUREMENTS

Fuel elements			Lattice spacing, inches	Average relative thermal flux				
Number of fuel rods	Rod diameter, inches	Fuel enriched, w/o		Central rod	Outside rod	Center rod / Outer rod	Alum.	Graph.**
7	0. 75	2. 78	7	1. 20	1. 98	0. 61	3. 20	7. 06
			9. 5	1. 22	2. 15	0. 57	3. 57	9. 63
			12	1. 24	2. 27	0. 55	3. 93	11. 96
6*	0. 75	2. 78	7	2. 23	2. 13		3. 27	7. 02
			9. 5	2. 35	2. 24		3. 69	9. 41
			12	2. 56	2. 41		4. 04	12. 02
7	0. 75	Natural	7	1. 07	1. 36		1. 66	2. 84
			9. 5	1. 06	1. 36		1. 72	3. 35
			12	1. 09	1. 39		1. 77	3. 80
4	1. 00	0. 91	7		1. 26		1. 68	3. 01
			9. 5		1. 20		1. 67	3. 38
			12		1. 29		1. 81	3. 99

*¾-inch graphite rod filling central channel of element.
**Includes flux in central rod if of graphite.

The SRE steady-state power coefficient has been determined to be:

$$\frac{\delta\rho}{\delta N_0}=\left(\frac{3}{N_0}-0.37\right)\times10^{-4}\pm50\%\ \frac{\Delta K}{K}/\text{MWT}$$ at a constant 1,400 GPM flow rate.

$$\frac{\delta\rho}{\delta N_0}=-\frac{3.5\times10^{-4}}{N_0}\frac{\Delta K}{K}/\text{MWT}\pm50\%$$ at a constant temperature difference of 335° F

where $\frac{\delta\rho}{\delta N_0}$ is the power coefficient and N_0 is the initial power level in megawatts.

The contribution of the fuel and fuel coolant to the total power coefficient was found to be:

$$\left(\frac{\delta\rho}{\delta N_0}\right)_{FC}=-(1.2\pm.1)\times10^{-4}\frac{\Delta K}{K}/\text{MWT}$$ at a constant 1,400 GPM flow rate

$$\left(\frac{\delta\rho}{\delta N_0}\right)_{FC}=-\left(\frac{7.3}{N_0}+0.8\right)\times10^{-4}\frac{\Delta K}{K}/\text{MWT}\pm20\%$$ at a constant temperature difference of 335° F

where $\left(\frac{\delta\rho}{\delta N_0}\right)_{FC}$ is the contribution of the fuel and fuel coolant to the total power coefficient and N_0 is the initial power level in megawatts.

The time constants associated with the moderator and fuel are tabulated in table IV.

The time constants associated with the fuel channel exit thermocouples and the fuel thermocouples are 1.8 ± 0.2 seconds.

The isothermal temperature coefficient of the moderator at an average coolant temperature of 680° F. is:

$$(1.2\pm0.6)\times10^{-5}\frac{\Delta K}{K}/F°$$

And the isothermal temperature coefficient of the fuel and coolant combined is:

$$-(0.65\pm.13)\times10^{-5}\frac{\Delta K}{K}/F°$$

The apparent overall heat transfer coefficient between the graphite moderator and sodium coolant was deduced to be:

$$180\pm35\ \text{BTU/Hr-Ft}^2\text{-}F°$$

In addition the reactor oscillator and noise analysis techniques have been utilized to obtain a value of 0.525 milliseconds for the prompt neutron lifetime in the SRE.

The reactor oscillator has shown the SRE to be an inherently stable reactor at all power levels. This stability is due to the prompt negative contribution of the fuel to the total power coefficient. It should be noted that the power coefficient is not the sum of the separate temperature coefficients but rather the sum of the separate coefficients weighted by the change in temperature for that component per megawatt change in power. The positive contribution of the graphite moderator to the total power

TABLE III.—TWO-GROUP CORE AND REFLECTOR DATA

	Dry	Wet	Hot-poisoned
SRE Lattice Constants			
η	1.827	1.823	1.753
ϵ	1.043	1.043	1.043
p	0.855	0.854	0.848
f	0.8523	0.8144	0.8403
k	1.389	1.321	1.302
L^2	173 cm^2	167	195
D_2	1.021 cm	0.960	0.973
r	425 cm^2	371	370
D_1	1.14 cm	1.07	1.08
Axial Reflector			
L^2	606 cm^2	553	670
D_2	1.08 cm	0.979	0.991
r	479 cm^2	384	378
D_1	1.20 cm	1.11	1.12
Radial Reflector			
L^2	2160 cm^2	1470	1840
D_2	0.985 cm	0.930	0.940
r	367 cm^2	338	336
D_1	1.06 cm	1.01	1.02
Critical Loading			
Theoretical	21.1 fuel clusters	28.4	33.2
Experimental	22.2	32.7	

TABLE IV.—SRE TIME CONSTANTS

	Time constant in seconds	
Component	Constant flow of 1,400 GPM.	Constant T of 335° F.
Moderator	600±50	800±80 at 5 MWt and 600±60 at 20 MWT.
Fuel	9±2	20±3 at 5 MWt and 9±2 at 20 MWt.

coefficient in no way affects the safety or stability aspects of the reactor since the graphite time constant is so large (600 seconds). Determination of the power coefficient provides experimental information which will be of great use in predicting the stability of future SGR's. The general agreement between the measure parameters, while oscillating at power, and those determined by theoretical method give evidence of the ability to create a mathematical model upon which a theoretical analysis of reactor stability may be effected.

Measurements to determine the long-term reactivity changes have been performed, and an empirical relation has been developed for the prediction of these changes. Measurements have been made using fuel elements whose exposures range from 200 to 850 MWD/Tonne, and the results are shown in Figure 2.

Experiments at the SRE to demonstrate the effect of utilizing sodium to replace the central rod in a seven-rod Th-U cluster, containing 7.6 w/o uranium (93 percent enriched), have shown that this replacement results in small loss in reactivity. The significance of this small reactivity loss can be demonstrated by considering that the calculated wet critical loading of the SRE with the Th-U elements is 23.5 seven-rod clusters. By replacing each central rod with sodium, the critical loading will be increased only by 0.8 clusters; the total number of fuel rods required would be reduced by 11 percent. As a result of these measurements, experiments have been designed to utilize various materials in the central rod for the purpose of investigating the better utilization of the central low flux space in the fuel cluster. The materials chosen for the experiments are helium, graphite, stainless steel, sodium, beryllium, and pure thorium.

Documents Issued

NAA–SR–3341—"Low Power Physics Experiments on the Sodium Reactor Experiment," R. W. Campbell.

NAA–SR–3765—"Measurement of the SRE and KEWB Prompt Neutron Life-Time Using Random Noise and Reactor Oscillation Tech-

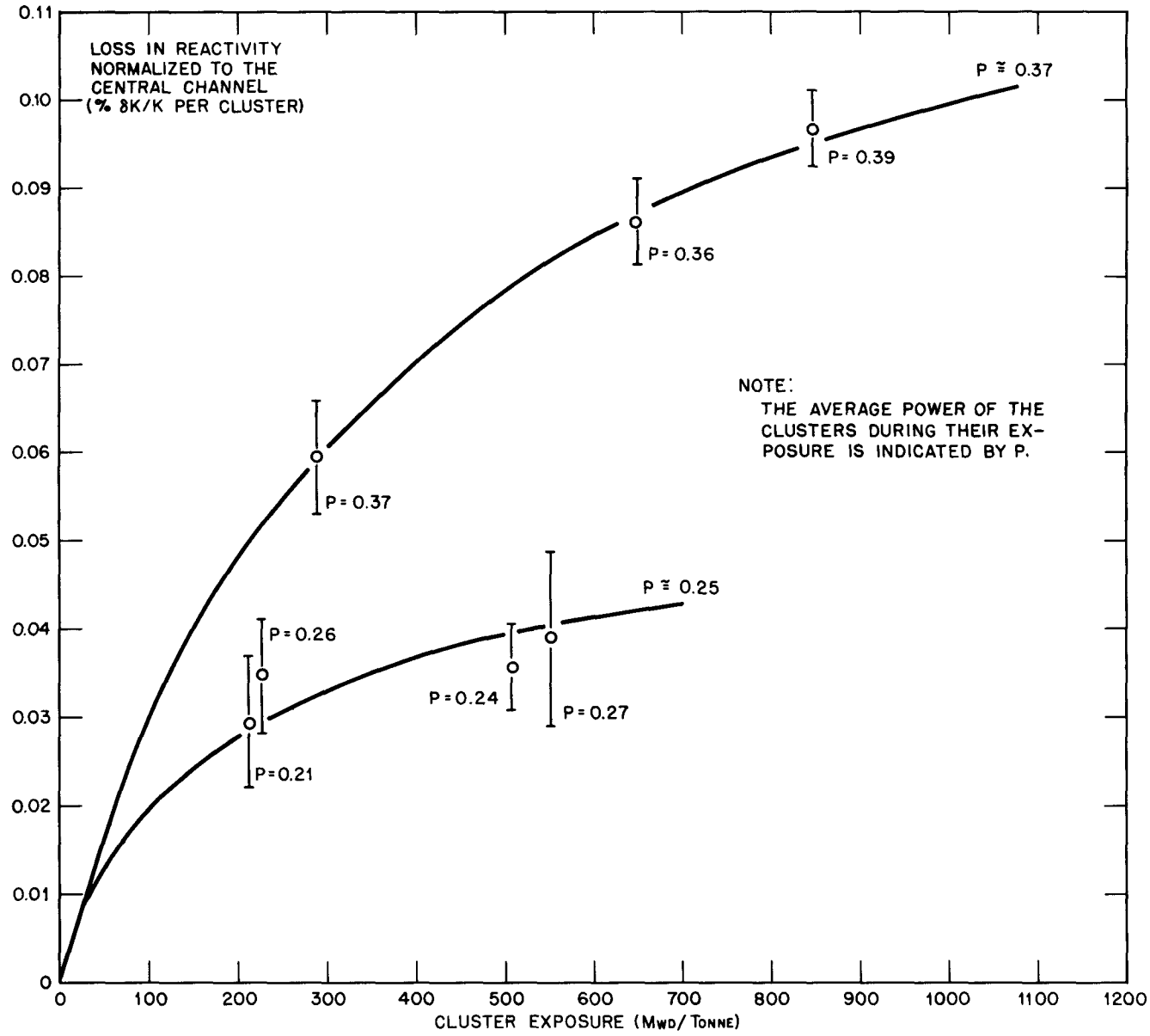

FIGURE 2.—Loss in reactivity for a cluster as a function of power and exposure.

$$\frac{\Delta P}{\Delta T} = \left(-7.26 \cdot 10^{-9} \frac{T^2}{(°F)^3} + 4.88 \cdot 10^{-6} \frac{T}{(°F)^2} + 5.71 \cdot \frac{10^{-4}}{°F} \right) \%$$

niques," C. W. Griffin and J. G. Lundholm, Jr.

b. Physics Experiments Underway

The single rod measurements and the exponential work is continuing. The critical assembly facility for SGR development has been completed, fuel ordered, and a series of general critical experiments are planned for the near future. Physics experiments on the SRE are continuing. Measurements of the long term reactivity effects will continue to the end of the life of the current metallic uranium fuel loading. Plans have been made to insert a new fuel loading of Th-U alloy early in fiscal year 1960. All of the experiments which have been conducted with uranium fuel loading will be repeated with the Th-U alloy core loading. In addition, experiments have been planned to determine the axial and radial flux distributions with the new SRE core loading. The experiments will be conducted and their results may effect fuel savings and an increased fuel lifetime. Studies of neutron economy will be made in order to realize an optimum configuration which is technically and economically compatible with available fuel materials.

Documents In Preparation

NAA–SR–3762—"Measurement of the SRE Zero-Power Frequency Response," C. W. Griffin, J. G. Lundholm, Jr., E. R. Meise.

NAA–SR–3763—"Determination of Nuclear Parameters of the SRE by Frequency Response Measurements Utilizing Pile Oscillator Techniques," J. G. Lundholm, Jr., C. W. Griffin.

NAA–SR–3764—"Control Rod Calibration by Frequency Response Techniques as Used on the SRE," J. G. Lundholm, Jr., C. W. Griffin.

2. Fuel and Materials Properties

a. Work Completed

Sodium exhibits excellent compatability with most metals and ceramics, including both cladding and fuel materials. The corrosion rates of steels, zirconium, uranium and uranium alloys, thorium and thorium alloys, uranium carbide, etc., are extremely low in sodium below 1,200° F. Corrosion rates of austenitic steels fall in the range 0.00 to 0.07 mg/cm²-mo in sodium containing oxygen up to 150 ppm and at temperatures up to 1,300° F. This is equivalent to a maximum of 0.00004 inch per year. Corrosion rates of ferritic steels containing at least 2¼ percent Cr fall in the range 0.02 to 0.11 mg/cm²-mo in sodium containing oxygen up to 100 ppm and at temperatures up to 1,200° F. These corrosion rates are based on 2,000-hour tests. The corrosion rates continuously decreased during the 2,000-hour exposure. Consequently the rate for very long time operation will be less than that indicated above. Uranium, thorium, and zircaloy corrosion rates are correspondingly low. The corrosion rates depend primarily on the concentration of oxide in the sodium. Experience with SRE has demonstrated that oxide concentration below 10 ppm can readily be maintained in an operation reactor. The factors affecting the choice of fuel for sodium cooled thermal reactors are stability under irradiation at high temperature, thermal conductivity, neutron economy, uranium density, availability, and fuel cycle costs. In order to take advantage of the heat transfer characteristics of sodium, a fuel element should be capable of providing heat fluxes of 1 to 3 million Btu/hr-ft² to high temperature sodium. The surface of the element should operate at temperatures well in excess of 1,200° F.

A number of low alloy metallic uranium elements have been irradiated in the SRE. Even at the relatively low center temperature reached in the SRE (1,000° F.) these materials have generally been demonstrated to be unsatisfactory for extended exposure. The only exception to date is the U–1½ Mo cast alloy, which has shown surprisingly good dimensional stability under SRE conditions at low burnup. A résumé of irradiation data on metallic and carbide fuel materials is presented in Table V.

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TABLE V—SWELLING DATA FOR URANIUM AND URANIUM ALLOYS

Material	Specimen size	Metallurgical state	Burnup (percent of all atoms)	Surface temperature		Central temperature		Percent volume increase	Percent volume change per a/o burnup
				°C	°F	°C	°F		
NAA-SR-3411									
Th 10 w/o U	375" x 1 5" pins	Cast	0 3	[2] 535	995	[3] 659	1,219	[1] 1 9	6 3
Do	do	do	0 3	[2] 463	866	[3] 570	1,059	[1] 0 47	1 6
Th 11 w/o U	do	do	0 3	[2] 514	957	[3] 633	1,172	[1] 0 78	2 6
Do	do	do	0 3	[2] 533	991	[3] 657	1,214	[1] 0 86	2 9
Do	do	Cast and swaged	0 55	[2] 469	876	[3] 577	1,071	[1] 2 9	5 27
Do	do	do	0 60	[2] 526	979	[3] 648	1,198	[1] 4 7	7 8
Do	do	do	0 63	[2] 521	970	[3] 642	1,187	[1] 4 1	6 5
Do	do	do	0 66	[2] 550	1,022	[3] 678	1,252	[1] 5 1	7 7
Do	do	do	0 68	[2] 495	924	[3] 610	1,130	[1] 7 7	11 3
Do	do	do	0 94	[4] 427	800	[4] 521	970	[1] 3 2	3 4
Do	do	do	1 00	[4] 454	850	[4] 554	1,030	[1] 2 7	2 7
Do	do	do	1 10	[4] 482	900	[4] 593	1,100	[1] 4 4	4 0
Do	do	do	1 11	[4] 482	900	[4] 593	1,100	[1] 4 8	4 3
Do	do	do	1 14	[4] 482	900	[4] 593	1,100	[1] 4 1	3 59
Do	do	do	1 13	[4] 482	900	[4] 593	1,100	[1] 4.9	4 3
ANL-5664									
Unalloyed uranium	25 x 75 pins	Powder metallurgy plus thermal cycling before irradiation	0 7	[2] 320	608	[2] 540	1,004	[6] 31 5	45 0
Do	do	do	0 43	[2] 200	392	[2] 340	644	[6] 1 29	3 0
Do	do	do	0 44	[2] 170	338	[2] 280	536	[6] 0 95	2 16
Do	do	do	0 53	[2] 220	428	[2] 370	698	[6] 0 93	1 75
Do	do	do	0 58	[2] 250	482	[2] 420	788	[6] 1 19	2 05
Do	do	do	0 39	[2] 180	356	[2] 290	554	[6] 0 48	1 23
Do	do	do	0 64	[2] 270	518	[2] 450	842	[6] 8 96	14 0
Do	do	do	0 64	[2] 270	518	[2] 460	860	[6] 5 76	9 0
Do	do	do	0 64	[2] 280	536	[2] 470	878	[6] 15 36	24 0
Do	do	do	0 66	[2] 300	572	[2] 510	950	[6] 23 76	36 0
Do	do	do	0 27	[2] 120	248	[2] 180	356	[6] 0 25	0 93
ANL-5736									
U 2 5 w/o Mo	25" x 1" pins	Cast, solid cylinders	0 48	[5] 578	1,072	[2] 668	1,234	[6] 2. 10	4 4
Do	do	do	0 42	[5] 509	948	[2] 589	1,092	[6] 2 38	5 7
Do	do	Cast, with 030" axial hole	0 40	[5] 496	925	[2] 556	1,032	[6] 0 94	2 3
Do	do	do	0 49	[5] 583	1,081	[2] 683	1,261	[6] 1 38	2 8
Do	do	Wrought, and furnace cooled from 850° C	0 42	[5] 496	925	[2] 556	1,032	[6] 2 19	5 2
Do	do	do	0 51	[5] 578	1,072	[2] 668	1,234	[6] 2 14	4 2
Do	do	do	0 52	[5] 583	1,081	[2] 683	1,261	[6] 1 16	2 2

See footnotes at end of table.

TABLE V.—SWELLING DATA FOR URANIUM AND URANIUM ALLOYS—Continued

Material	Specimen size	Metallurgical state	Burnup (percent of all atoms)	Surface temperature		Central temperature		Percent volume increase	Percent volume change per a/o burnup
				°C	°F	°C	°F		
Geneva Paper 81									
Unalloyed uranium	0 1" x 0 1" buttons	Arc melted	13 [11] 34	[5] 500	932	[2] 500	932	[1] 4 0 [1] 10 9	31 0 32 0
Do	do	do	10 [11] 37	[5] 800	1,472	[2] 800	1,472	[1] 2 9 [1] 14 1	29 0 39 0
Do	do	do	10 [11] 39	[5] 700	1,292	[2] 700	1,292	[1] 2 9 [1] 20 1	29 0 42 0
Do	do	do	27	[5] 700	1 292	[2] 700	1,292	[1] 11 5	43 0
Do	do	do	26	[5] 500	932	[2] 500	932	[1] 8 1	31 0
Do	do	do	26	[5] 500	932	[2] 500	932	[1] 11 8	45 0
Do	do	do	34	[5] 800	1,472	[2] 800	1,472	[1] 84 5	248 0
Do	do	do	35	[5] 600	1,112	[2] 600	1,112	[1] 146 0	418 0
Do	do	do	41	[5] 780	1,436	[2] 780	1,436	[1] 65 4	160 0
Do	do	do	41	[5] 600	1,112	[2] 600	1,112	[1] 4 4	10 7
Geneva Paper 622									
U-10 w/o Mo	1" x 75" pins	Extruded and gamma heat treated	0 8	[7] 538	1,000	[3] 575	1 068	[1] 4 1	5 1
Do	do	do	0 8	[7] 538	1,000	[3] 575	1,068	[1] 4 4	5 5
Do	do	Hot rolled, swaged, and gamma heat treated	1 7	[7] 704	1,300	[3] 745	1,373	[1] 28 0	16 5
Do	do	do	1 7	[7] 704	1,300	[3] 745	1,373	[1] 32 0	18 8
Do	do	Hot rolled, swaged, and transformed	0 6	[7] 663	1,225	[3] 700	1,292	[1] 2 8	4 6
Do	do	do	0 6	[7] 663	1,225	[3] 700	1,292	[1] 4 1	6 8
Do	do	do	0 6	[7] 663	1,225	[3] 700	1,292	[1] 4 5	7 5
Do	do	do	0 6	[7] 663	1,225	[3] 700	1,292	[1] 5 1	8 5
Do	do	Extruded and gamma heat treated	1 3	[7] 588	1,090	[3] 630	1,166	[1] 30 0	23 1
Do	do	do	1 3	[7] 588	1,090	[3] 630	1,166	[1] 24 7	19 0
Do	do	Extruded and transformed	1 3	[7] 588	1,090	[3] 630	1,166	[1] 87 9	67 6
Do	do	As extruded	1 3	[7] 388	730	[2] 470	878	[1] 2 3	1 8
Do	do	do	2 0	[7] 505	942	[2] 600	1,112	[1] 3 6	1 8
Do	do	Extruded and gamma heat treated	1 2	[7] 388	730	[2] 470	878	[1] 2 7	2 2
Do	do	do	2 2	[7] 505	942	[2] 600	1,112	[1] 4 0	1 8
Do	do	do	1 4	[7] 388	730	[2] 470	878	[1] 2 9	2 0
Do	do	do	2 0	[7] 505	942	[2] 600	1,112	[1] 4 0	2 0
Do	do	do	1 2	[7] 316	600	[2] 400	752	[1] 2 7	2 2
Do	do	do	1 2	[7] 316	600	[2] 400	752	[1] 2 5	2 1
Do	do	do	1 2	[7] 316	600	[2] 400	752	[1] 2 3	1 9
Do	do	do	1 2	[7] 316	600	[2] 400	752	[1] 2 1	1 7
Letter BMI to AI dated April 29, 1959									
Uranium carbide	375" x 2"	As cast	[12] 0 20	[3] 410	770	[2] 721	1,330	0 7—[10] [1] 0 9	3 5 —[12] 4 5
Do	do	do	[12] 0 21	[3] 282	540	[4] 427	800	[1] 2 5	[12] 11 9
Do	do	do	[12] 0 63	[3] 491	916	[2] 746	1,376	0 6—[10] [1] 2 0	95—[12] 3 17
Do	do	do	[12] 0 77	[3] 371	701	[5] 538	1,000	[1] 2 5	[12] 3 24

WASH-741								
U–10 w/o Nb–4 w/o Zr	.04" x 1." x 6."	Plates double arc melted, hot rolled, and transformed.	0.61	[4] 363	635	[3] 379	715	[9] 2.9	4.8
Do	do	do	0.66	[3] 366	691	[3] 383	721	[9] 1.7	2.6
Do	do	do	0.72	[3] 370	698	[3] 387	728	[9] 2.9	4.0
Do	do	do	0.70	[3] 419	785	[3] 435	815	[9] 1.5	2.1
Wash 741									
U 10 w/o Nb	.05" x .65" x 5" plates clad with .030" Zr[2].	Double arc melted and hot rolled	0.69	[3] 327	619	[3] 343	649	[9] 2.2	3.2
U 10.4 w/o Nb	do	do	0.33	[3] 369	696	[3] 386	726	[9] 1.0	3.0
U 10 w/o Nb	do	Double arc melted, hot rolled, and transformed.	0.69	[3] 410	770	[3] 427	800	[9] 3.9	5.7
Do	do	do	0.72	[3] 410	770	[3] 427	800	[9] 18.9	26.3
Do	do	Double arc melted and hot rolled	0.80	[3] 413	775	[3] 429	805	[9] 22.4	28.0
Do	do.[5]	do.[5]	0.77	[3] 413	775	[3] 429	805	[9] 14.6	19.0
U 12 w/o Nb	do.[7]	do.[7]	0.72	[3] 417	783	[3] 434	813	[9] 19.8	27.5
U 10.4 w/o Nb	do.[7]	do.[7]	0.56	[3] 421	791	[3] 438	821	[9] 11.0	19.5
U 10 w/o Nb	do.[7]	do.[7]	0.85	[3] 435	814	[3] 451	844	[9] 9.4	11.0
Do	do.[7]	do.[7]	0.95	[3] 461	862	[3] 478	892	[9] 32.4	34.1
U 12 w/o Nb	do.[7]	do.[7]	0.95	[3] 467	872	[3] 483	902	[9] 35.5	37.3
U 10 w/o Nb	do.[7]	do.[7]	1.03	[3] 477	890	[3] 493	920	[9] 37.1	35.9
Do	do.[7]	do.[7]	1.03	[3] 480	896	[3] 496	926	[9] 45.6	44.4
Do	do.[7]	do.[7]	1.06	[3] 488	910	[3] 504	940	[9] 46.5	43.9
U 12 w/o Nb	do.[7]	do.[7]	1.06	[3] 489	912	[3] 505	942	[9] 44.1	39.8
SRE data									
U–2 wt % Zr	0.75" x 6"	As cast	0.047	[2] 301	575	[2] 368	695	[1] 0.5	10.64
Do	do	do	0.060	[2] 321	610	[2] 407	765	[1] 0.4	6.60
Do	do	do	0.072	[2] 340	645	[2] 443	830	[1] 0.4	5.50
Do	do	do	0.084	[2] 382	720	[2] 501	935	[1] 3.7	44.05
Do	do	do	0.085	[2] 401	755	[2] 521	970	[1] 4.6	54.12
Do	do	do	0.083	[2] 416	780	[2] 538	1,000	[1] 5.6	67.47
Do	do	do	0.076	[2] 432	810	[2] 540	1,005	[1] 4.8	63.16
Do	do	do	0.055	[2] 449	840	[2] 529	985	[1] 2.7	49.09
Do	do	do	0.042	[2] 457	855	[2] 518	965	[1] 1.4	33.33
Do	do	do	0.032	[2] 462	865	[2] 507	945	[1] 1.0	31.25
U 1.5 wt % Mo	do	do	0.047	[2] 301	575	[2] 368	695	[1] 0.1	2.13
Do	do	do	0.060	[2] 321	610	[2] 407	765	[1] 0.2	3.33
Do	do	do	0.072	[2] 340	645	[2] 443	830	[1] 0.2	2.77
Do	do	do	0.084	[2] 382	720	[2] 501	935	[1] 0.6	7.14
Do	do	do	0.085	[2] 401	755	[2] 521	970	[1] 0.9	10.59
Do	do	do	0.083	[2] 416	780	[2] 538	1,000	[1] 0.9	10.84
Do	do	do	0.076	[2] 432	810	[2] 540	1,005	[1] 1.0	13.16
Do	do	do	0.055	[2] 449	840	[2] 529	985	[1] 0.9	16.36
Do	do	do	0.042	[2] 457	855	[2] 518	965	[1] 0.7	16.66
Do	do	do	0.032	[2] 463	865	[2] 507	945	[1] 0.5	15.62

See footnotes at end of table.

TABLE V —SWELLING DATA FOR URANIUM AND URANIUM ALLOYS—Continued

Material	Specimen size	Metallurgical state	Burnup (percent of all atoms)	Surface temperature		Central temperature		Percent volume increase	Percent volume change per a/o burnup
				°C	°F	°C	°F		
SRF data—Continued									
Uranium	0 75″ x 6″	As cast	0 047	[2] 301	575	[2] 368	695	[1] 0 2	4 26
Do	do	do	0 060	[2] 321	610	[2] 407	765	[1] 0 4	6 66
Do	do	do	0 072	[2] 340	645	[2] 443	830	[1] 0 3	4 17
Do	do	do	0 084	[2] 382	720	[2] 501	935	[1] 0 7	8 33
Do	do	do	0 085	[2] 401	755	[2] 521	970	[1] 0 8	9 41
Do	do	do	0 083	[2] 416	780	[2] 538	1,000	[1] 1 2	14 46
Do	do	do	0 076	[2] 432	810	[2] 540	1,005	[1] 1 1	14 47
Do	do	do	0 055	[2] 449	840	[2] 529	985	[1] 1 0	18 18
Do	do	do	0 042	[2] 457	855	[2] 518	965	[1] 0 7	16 66
Do	do	do	0 032	[2] 463	865	[2] 507	945	[1] 0 7	21 88
Do	do	Alpha rolled and beta heat treated	0 047	[2] 301	575	[2] 368	695	[1] 0 1	2 13
Do	do	do	0 060	[2] 321	610	[2] 407	765	[1] 0 2	3 33
Do	do	do	0 072	[2] 340	645	[2] 443	830	[1] 0 5	6 94
Do	do	do	0 084	[2] 382	720	[2] 501	935	[1] 1 8	21 43
Do	do	do	0 085	[2] 401	755	[2] 521	970	[1] 2 2	25 88
Do	do	do	0 083	[2] 416	780	[2] 538	1,000	[1] 2 6	31 32
Do	do	do	0 076	[2] 432	810	[2] 540	1,005	[1] 2 5	32 90
Do	do	do	0 055	[2] 449	840	[2] 529	985	[1] 2 1	38 18
Do	do	do	0 042	[2] 457	855	[2] 518	965	[1] 1 3	30 95
Do	do	do	0 032	[2] 463	865	[2] 507	945	[1] 1 4	43 75
Th 5 4 wt % U	do	Extruded and swaged ~30% R A	0 072	[2] 301	575	[2] 360	680	[1] 0 1	1 39
Do	do	do	0 093	[2] 321	610	[2] 396	745	[1] 0 3	3 23
Do	do	do	0 111	[2] 340	645	[2] 429	805	[1] 0 4	3 60
Do	do	do	0 129	[2] 382	720	[2] 488	910	[1] 0	0
Do	do	do	0 130	[2] 401	755	[2] 507	945	[1] 0 8	6 15
Do	do	do	0 127	[2] 416	780	[2] 521	970	[1] 0 2	1 57
Do	do	do	0 117	[2] 432	810	[2] 527	980	[1] 0 3	2 56
Do	do	do	0 084	[2] 449	840	[2] 518	965	[1] 0 2	2 38
Do	do	do	0 064	[2] 457	855	[2] 507	945	[1] 0 4	6 25
Do	do	do	0 050	[2] 463	865	[2] 501	935	[1] 0	0
U 1 2 wt % Mo	75″ x 1″	Powder compacted	0 047	[2] 301	575	[2] 368	695	[1] 0	0
Do	do	do	0 060	[2] 321	610	[2] 407	765	[1] 0	0
Do	do	do	0 072	[2] 340	645	[2] 443	830	[1] 0 3	31 94
Do	do	do	0 084	[2] 382	720	[2] 501	935	[1] 1 8	21 43
Do	do	do	0 085	[2] 401	755	[2] 521	970	[1] 1 9	22 35
Do	do	do	0 083	[2] 416	780	[2] 538	1 000	[1] 3 2	38 55
Do	do	do	0 076	[2] 432	810	[2] 540	1 005	[1] 2 1	27 63
Do	do	do	0 055	[2] 449	840	[2] 529	985	[1] 2 3	41 82
Do	do	do	0 042	[2] 457	855	[2] 518	965	[1] 1 5	35 71
Do	do	do	0 032	[2] 463	865	[2] 507	945	[1] 0 8	25 0

[1] Based on density measurements
[2] Average calculated temperature
[3] Average measured temperature
[4] Brief thermocouple life required that these temperatures be calculated
[5] Authors estimate
[6] Based on length and diameter changes
[7] Based on estimates from Don Calkins
[8] Based on estimates from R Fillnow W A P D
[9] Based on increase in plate thickness
[10] On two different pieces of same sample
[11] Reirradiation of same sample at same temperature
[12] Percent of uranium atoms only

Additional tests of metallic uranium and thorium-uranium alloys have been made in the MTR. In the MTR tests, the temperature range of the uranium specimens was 900 to 1,300° F. at low burnup. The thorium-uranium alloy specimens (Th-10 w/o U) irradiated at 1,200° F. have shown good dimensional stability up to 11,000 Mwd/t. Table V also indicates good stability under SRE conditions at low burnup. With the exception of thorium-uranium alloy, these tests have generally proven metal fuels to be unsatisfactory for the advanced SGR in which center temperatures above 1,400° F. are required. A full core loading of Th-U alloy has been fabricated and is ready for insertion in the SRE.

UO_2 has been examined for application to sodium cooled reactors, and an experimental 19-rod element has been inserted into the SRE. A tubular element has been fabricated and will be inserted shortly. Extensive effort on UO_2 at other sites has indicated good radiation stability at high burnup at low surface temperatures. Because of the very low thermal conductivity (~1 Btu/hr-ft-° F.) of UO_2, it is of limited interest to high-performance sodium systems, and the fuel element irradiation constitutes a completion phase of previous work. It should be noted that oxide can be used in a sodium cooled reactor with a performance equal to or better than its performance in other reactors, and high sodium temperatures can be obtained from this fuel material. However, heat fluxes available from an oxide element would be no greater than those which can be obtained with a more conventional coolant, and full advantage cannot be taken of the superior heat transfer properties of sodium.

Uranium monocarbide, with a melting point of 4,300° F. and a conductivity of about 14 Btu/hr-ft-° F., is being intensively investigated for use in sodium cooled reactors. The conductivity of UC is about 14 times greater than that for UO_2 at fuel temperatures of interest to the SGR. Only limited information is available at the present time on the stability of the carbide under high temperature irradiation. Four specimens have been irradiated in the MTR, two of these to about 2,000 MWD/T and two to about 6,000 MWD/T.

Fission product retention was excellent in all specimens. The uranium carbide development program has compiled preliminary data on compatibility with Na and NaK, thermal shock, some physical properties, and considerable arc-melting fabrication experience. These data are now being analyzed; no unfavorable characteristics have been observed qualitatively.

b. Fuel and Materials Work Underway

Although considerable effort on other fuels at lower temperatures conditions will be of some interest to the sodium reactor fuel program, the most applicable work with high surface temperature fuel materials is being done in the Enrico Fermi, EBR–I, EBR–II, SRE, and ASGR programs. Recent awards of the Fuel Cycle Development Program contain additional development work on uranium carbide, UC+PuC, and uranium compound investigations. Irradiation of the metallic uranium elements in the SRE will be continued until replacement of the core with a complete Th-U alloy loading in the summer of 1959. Upon discharge from the reactor, these elements will be examined for swelling and fission product release. Irradiation of metallic specimens in MTR is also continuing with emphasis on the more promising Th-U and U-Mo alloys. Increased basic investigations are underway to determine the cause of swelling in metallic elements.

A series of static capsule exposures of UC in the MTR is scheduled for fiscal year 1960, as indicated in Table VI.

TABLE VI.—UC IRRADIATIONS IN MTR FOR FISCAL YEAR 1960

Carbon (wt %)	Burnup (Mwd/t)	Insertion date	Discharge date
5.0	10, 000	1–59	9–59
5.0	15, 000	12–58	1–60
4.6	3, 000	6–59	9–59
4.8	5, 000	9–59	1–60

Only one of the above specimens is intended to be of exact stoichiometric composition (4.8 w/o C). It is designed to have a central temperature of about 3,000° F. Additional nonstoichiometric specimens covering the range from 3 to 6 percent carbon will be used in the determination of physical properties, fabricability, etc. In addition to the MTR tests, three 4-rod clusters will be made of arc-cast slugs 0.8 inch in diameter and 4 inches long. Under the ASGR program a single rod of UC will be irradiated in SRE as part of a 7-rod cluster. The progress of other ceramic and cermet development is summarized in Table VII.

TABLE VII.—CERAMIC AND CERMET FUELS DEVELOPMENT

Compound of cermet	Development of fabrication technique in progress	Irradiation planning in progress	Irradiation tests in progress
UO_2 (19-rod)	(*)	X	X
UO_2 (tubular)	(*)	X	
UC (BMI)	X	X	X
UC (AI)	X	X	
UAl_2	X	X	
U_3Si_2	X	X	
UN	X		
UP	X		
US	X		
UB_2	X		
UO_2 or UC, U–60 wt % Nb.	X		
UO_2 or UC + unalloyed U	X		

*Completed

Only limited effort is in progress on UO_2 for sodium cooled reactors because of the superior properties of UC for this system. Current work on UO_2 is expected to establish the limits of mechanical designs using clad pelleted UO_2 fuel in long continuous lengths.

In the area of fuel-cladding materials, work is underway on the development of high temperature zirconium alloys for use in fuel cans at temperatures up to 1,050° F. Alloys have been developed with a creep rate at 1,050° F. of 1 percent in 1,000 hours at a stress of 10,000 psi. The yield strength at 1,050° F. is 46,000 psi.

These figures may be compared with those for 304 stainless steel, for which the 1 percent creep in 1,000 hours at 1,050° F. is obtained with a stress of 20,000 psi, and the yield strength at 1,050° F. is 25,000 psi. Nonstructural claddings in the form of films produced by such techniques as vapor deposition, flame-spraying or plating, are also under development for application to fuel materials of high creep strength.

Structural materials for the reactor and coolant system are also under investigation. The conditions under which excessive nitriding will take place in sodium systems with a nitrogen cover gas (rather than the more expensive helium) are being studied in loops operating at 1,200° F. In these experiments careful attention is being paid to the aspects of impurities in the sodium. Ferritic steels with adequate strength and corrosion resistance for use in sodium systems at 1,100°–1,150° F. are under development. Low-cost welding procedures and simple cleaning procedures utilizing sodium as the cleaning and descaling agent are under development.

3. Heat Transfer Experiments

a. Work Completed

Heat transfer experiments with liquid metals have been actively prosecuted since 1948, with the inception of the Navy-sponsored GENIE program at KAPL. Steam generator models were tested for 5 years under various auspices during this program. In 1952, a series of experiments designed to demonstrate the capabilities of large, superheated, steam generators by model tests was undertaken; 10 percent capacity model steam generators were tested under Navy auspices.

Simultaneously, a general liquid metal heat transfer research program was initiated at both KAPL and Mine Safety Appliances (MSA). This information was developed to provide analytical information for core heat transfer

purposes and liquid-metal-to-liquid-metal heat exchanger designs. These efforts, covering 1951 to 1956, are widely reported, and are summarized in the Liquid Metals Handbook, Volumes 1 and 2.

Experimental data on the overall heat transfer coefficient, for the heat exchangers which have been tested in SRE operations, are presented in Table VIII. In each case the data apply to full power (20 MW) operation. A curve showing the variation in heat transfer coefficient with sodium flow rate for the intermediate heat exchanger appears in Figure 3. Comparable data on the two steam generators are not available.

The poor experimental performance of the main intermediate heat exchanger is indicative of the inefficient surface utilization and bypass flow condition which exists in the present equipment. The quoted value of the coefficient U is not representative of that obtainable in a sodium-to-sodium heat exchanger of proper design.

Additional heat transfer work applying to specific engineering applications, such as heat removal from frozen sodium seals, both static and rotating has been completed at Atomics International. This work was undertaken as a part of component development.

TABLE VIII.—OVERALL HEAT TRANSFER COEFFICIENT FOR SRE HEAT EXCHANGERS

Heat exchanger	Overall heat transfer coefficient (U)* (BTU/HR-FT²-°F)	
	Experimental	*Predicted*
Main Intermediate	730	870 (unbaffled shell side). 1,140 (full baffled shell side).
B & W Steam Generator.	286	
C.E. Steam Generator:		
Boilers	476	470.
Superheater	181	125.

* Based on outside surface area of heat exchanger tubes.

References

R. W. Lockhart et al., "Experiments with Sodium Heated Steam Generators," KAPL–1450 (Conf.).

"Experimental Breeder Project Report," Feb. 1, 1950 through March 31, 1951, ANL–4554, Et seq.

Reports in Preparation

NAA–SR–3775, "Thermal Performance and Structural Evaluation of the SRE Main Intermediate Heat Exchanger," K. W. Foster.

NAA–SR–3969, "Preliminary Performance Test of a Natural-Circulation Double-Tube Steam Generator," R. D. Welsh.

b. Heat Transfer Experiments Underway

The most significant heat transfer experiment presently in progress is, of course, the Sodium Reactor Experiment. Both core and component heat transfer experience is being generated daily. This is the most valuable type of experience as it represents actual operating conditions. Numerous improvements on intermediate heat exchanger design have already been derived from operating the SRE, as well as a more thorough understanding of the heat transfer associated with cold traps, hot traps, and other purification devices.

Concurrently, an apparatus designed to investigate the boiling of sodium is being constructed. Operation of this device is scheduled for the fall of 1959, at which time it is expected that additional information, both on limiting (burnout) heat fluxes in the reactor core and other experimental information essential for proper backup of hazards analyses, will be produced.

A figure-8 loop to test a "helicoil" type heat exchanger is in existence, and a 100-kw unit is being installed for test under thermal shock conditions and to test mechanical integrity under violent thermal transients. Incidentally, of course, the heat transfer rate of this apparatus in liquid metal will also be tested.

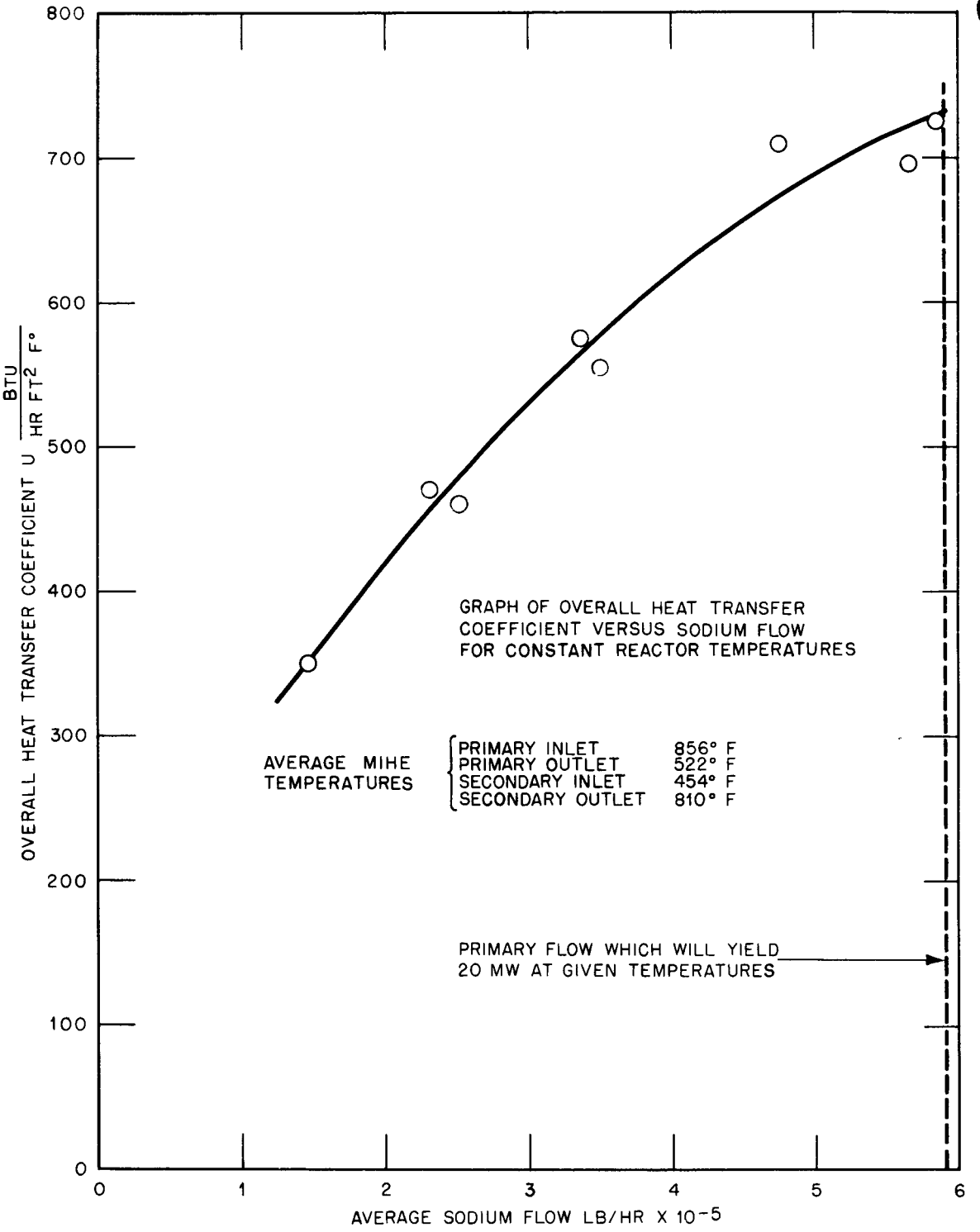

FIGURE 3.—SRE intermediate heat exchanger performance.

Small scale heat transfer experiments are in progress to indicate whether the sealing of liquid metal valve seats by freezing is practical. Initial experiments have been completed, which led to the satisfactory performance of seals in the sodium filter device now in use at AI. These experiments also contribute to the knowledge of heat transfer to liquid metal in small annuli, and to the knowledge of convective cooling by liquid metals.

The largest liquid metal thermal shock machine known to be in existence has been completed and is now in operation. This is the Large Components Test Loop at Atomics International's field laboratory facilities. Transient heat transfer data, derived from instrumentation of components placed in the core tank (35-foot height by 8-foot diameter), which can be flushed with sodium 500° F. above ambient temperature at a rate of several thousand gallons per minute, will contribute to the experimental verification of theoretical heat transfer analyses. The first experiments on full size core components for the Hallam reactor are in progress. Nozzle configurations and other critical components are fully instrumented and are being tested at the same time as the core components.

Test of a liquid-metal to water heat exchanger, a three megawatt model of an experimental steam generator-superheater combination, is in progress at MSA under contract to AI. Heat transfer coefficients are being obtained from this experiment, which is primarily a test of reliability and transient performance of a proposed large-scale steam generator for the Hallem Nuclear Power Facility.

4. Fluid Flow Experiments

a. Work Completed

Liquid metal system hydraulic experiments were undertaken by KAPL and MSA as a part of the SIR development program, which extended from 1948 to 1956. These data are widely reported in classified literature, and summarized to a large extent in the Liquid Metals Handbook. Dimensionless parameter correlations of liquid metal behavior up to temperatures of the order of 850° F. have been derived at MSA, KAPL, and Oak Ridge National Laboratory (ORNL) and are now reported in the literature. These are summarized in the Liquid Metals Handbook. It is considered that sufficient hydraulic and fluid flow information has been developed to support routine design of liquid metal heat transfer systems and equipment utilizing forced convection. Since liquid sodium is quite similar to water in its flow characteristics in forced convection new component designs are usually checked experimentally using water. It is then possible to make reasonably accurate prediction of the flow characteristics for sodium.

A large area of uncertainty still exists, however, in the field of natural convection under low flow conditions. Natural convection circulation loops are set up even in small pools of sodium because of the high thermal conductivity, high film coefficients, and moderate density change of liquid with temperature. Because the behavior of these fluids is not as yet completely understood, there is a continuing need to perform both scale model and full scale experiments to insure, as far as possible, that heat transfer equipment will perform as predicted.

b. Fluid Flow Experiments Underway

Fluid flow experiments are now being undertaken on a mockup of the calandria core, which will be operated first with water, and later with sodium. The objectives are to determine flow and temperature distribution patterns around the core and to determine its response to temperature transients. The electrically heated mockup will be about ¼-scale of a proposed experimental reactor (SCRE); data will be correlated by dimensionless parameters to the reactor experiment.

Minor studies involve determination of flow patterns in narrow annuli, and internal circulation patterns within heat exchangers. These circulations are caused by temperature gradients within the liquid metals. These efforts are being undertaken at Atomics International;

parallel efforts are being carried out at Argonne National Laboratory and by Atomic Power Development Associates in support of their respective fast breeder programs. By exchange of information and close cooperation it will be possible to derive fundamental information from all these experiments as they approach completion.

5. Coolant Chemistry

a. Work Completed

Liquid metal coolant chemistry is relatively simple. No decomposition products need be considered, as sodium is an elemental substance and transmutation to magnesium-24 by neutron absorption is extremely slow. Difficulties with sodium chemistry stem from impurities rather than the metal itself. Typical impurities are listed in Table IX.

TABLE IX.—IMPURITIES IN A TYPICAL SPECIMEN OF COMMERCIAL SODIUM

Impurity	*Approximate concentration (p p m)*
Ca	10
Li	5
K	50
C	10
Fe	50
Si	100
Cu	5
Mo	10
Mg	(1)
Mn	(1)
Nb	(1)

[1] Below detection limits (5–10 p p m)

Development of analytical techniques to determine quantitatively the amount of impurities present is in its infancy. The difficulties stem from the chemically active nature of liquid metals, which require that they be handled under an inert atmosphere which may itself not be pure.

Use of liquid metal as a heat transfer fluid in process industries, such as the production of tetra-ethyl lead, has been practiced for many years by the Ethyl Corporation and by duPont. Temperatures of interest to these processes, however, are about 450° F., and have involved no special techniques other than use of inert gas atmospheres. Investigation of high temperature liquid metal chemistry and technology began in 1948 at the Knolls Atomic Power Laboratory, Oak Ridge National Laboratory, and Argonne National Laboratory. Handling of liquid metals at temperatures up to 1,200° was investigated, with most of the work done at about 850° F. Means of removing oxygen from sodium and sodium-potassium alloys were developed. These early developments involved "cold traps" to precipitate the oxide at lower temperatures than existed in the process system. "Hot traps" for final system clean-up were developed by AI in 1956. Cold traps reduce oxygen concentration to about 10 ppm; hot traps reduce oxygen concentration to an undetectable level.

Analysis for oxygen in sodium has always been difficult due to lack of confidence that the sodium sample was not contaminated during removal. Laborious techniques, chiefly mercury amalgamation schemes and high temperature distillation methods, have been developed for quantitative analysis of oxygen as sodium oxide. Accuracy of ± 10 ppm, down to concentrations of 20 ppm oxygen, can be achieved by these methods. These techniques have been perfected over the years. No significant improvement in sampling methods have been made, however, and therefore confidence in oxygen analysis is only as good as the confidence in the sample. Other impurities in sodium have not proved troublesome at temperatures up to 1,000° F. with helium and argon cover gases. The noble gases do not, of course, react with liquid metals and apparently do not dissolve to any appreciable degree in the metal. Because of the inconvenience associated with purging and maintaining noble gas atmospheres, it is desirable to use more common gases such as nitrogen. Nitrogen apparently does not react with sodium, nor does it dissolve in any significant amount in pure sodium. Evidence of nitriding in nitrogen cover systems has, however, been occasionally reported. Tentative information indicates that, quite possibly, transport of nitrogen

in liquid metal systems is due to impurities such as carbon, lithium, and calcium. The exact role of these impurities is not yet understood.

Atomics International, Knolls Atomic Power Laboratory, and Mine Safety Appliances have been concerned with the phenomena of mass transfer in liquid metal systems since 1952. Experiments in all of these laboratories indicate that, up to about 1,000° F., mass transfer is not a problem. Above this temperature mass transfer can occur. The effects are "nuisance effects" rather than real system hazards up to at least 1,200° F. Transport of radioactive atoms may eventually prove a nuisance in accessibility to liquid metal reactor primary systems after shutdown. However, radiation levels in the SRE, after 2 years of operation, are 2–10 Mr/hr 10 days after shutdown, and this activity is apparently due to fission products as determined by gamma-ray spectrometer. No effects such as plugging of small pipes and orifices by deposition of mass transfer products is expected on the basis of the evidence now at hand. An excellent summary of mass transfer in sodium is presented in "Chemical Engineering Progress," No. 20, Volume 53, 1957.

Sodium appears to be a very noncorrosive substance. The product of a sodium-water reaction, sodium hydroxide, is, of course, extremely corrosive when in a water solution. All the evidence accumulated at KAPL, ANL, ORNL, and AI since 1952 indicates that, if sodium is reasonably free of oxide, it is essentially inert to the usual materials of construction in high temperature systems. Decarburization of austenitic steels has not been observed. However, the ferritic steels containing less than 5 percent chromium decarburize to low levels (about 0.02% C) in sodium systems which contain the austenitic stainless steels. When carbon transfer does occur the rate-controlling step may be either the diffusion of carbon out of the ferritic steel or into the austenitic steel, depending on the relative areas involved. The diffusion coefficient for decarburization of the ferritic steel is about 1×10^{-10} cm^2/sec at 1,050° F. while the diffusion coefficient for volume diffusion into the 304 SS is about 1×10^{-12} cm^2/sec. Modification of the low chromium steels with strong carbide forming elements such as Nb, Ti, or V will prevent the carbon transfer from the ferritic to austenitic steels.

In summary, liquid metal coolant chemistry appears to be no significant problem if proper attention is paid to the system design, recognizing the chemical reactivity of alkali liquid metals with oxygen.

b. Coolant Chemistry Work Underway

Experiments on coolant chemistry are now being performed at AI, in an effort to determine the role of sodium impurities as carriers of nitrogen. These experiments are required to confirm the suitability of nitrogen as a cover gas, in place of more expensive helium now in use. Mass transfer experiments are also underway in the SRE, in a facility designed for this purpose. Analytical techniques for the determination of carbon in the sodium have been developed and are being perfected in support of the SRE, and techniques for the analytical determination of oxygen are being undertaken at other laboratories.

General Electric Co. is engaged in a study of sodium impurities, and the technology of sodium in general. These studies are primarily aimed at determining the role of oxide in plugging small sodium lines. The growth of sodium oxide crystals is of some importance, and a technique and apparatus have been developed for crystal growth. The aim of this study is to be able to control the growth of sodium oxide crystals so that they can be precipitated from the system in a predictable fashion. Samples of oxide plugs formed in small lines in the SRE have been provided to GE, and are being analyzed both chemically and crystallographically.

The General Electric Co. is also studying the phenomenon of mass transfer of metals in sodium at elevated temperatures. This work has just commenced, and no results are yet available. The technology of sodium at temperatures up to 850° F. is rather well understood; future sodium systems, however, will

operate at temperatures above 1,000° F., and this temperature range is under intensive investigation by Argonne National Laboratory, Atomics International, and to some extent, the General Electric Co. It is expected that the current work will bring some general information within a year, and demonstrate the direction which must be followed to provide simple and reliable systems at temperatures up to 1,200° F., within the next 2 to 3 years. It should be noted that these problems are common to all sodium cooled reactors.

6. Reactor Safety

a. Work Completed

The only operating sodium graphite reactor' the SRE, has been demonstrated to be extremely stable. Operating experience has shown that very little control rod movement is required when operating at full power under automatic control. An example is the recent observance of only 2½ minutes of regulating rod motion in the SRE during 24 hours of full power operation at steady state. The small, overall temperature coefficient of reactivity renders unnecessary large amounts of control to compensate for the temperature rise to operating conditions. In the SRE the change in reactivity between startup (~350° F.) and operating at full power is about 0.2 percent, and this change is positive. This reduces the excess reactivity and the number of control rods required for an SGR. The negative, prompt, fuel temperature coefficient produces the stability of the reactor. The results of the pile oscillator experiments on SRE have shown the contribution of the fuel and fuel coolant to the power coefficient to be always negative and to have a time constant of 9 seconds. The total steady-state power coefficient is negative at all power levels for the case of a constant temperature difference across the core, and is negative at power levels above 8 Mw for the case of constant flow of 1,400 gpm. Below 8 Mw, the steady-state power coefficient is positive because of the moderator. However, the long time constant of the moderator (600 seconds) implies that any instability would be of a very low frequency and presents no control problem of any consequence. The analyses of the reactor transfer function obtained with the pile oscillator have firmly established that the SRE is completely stable in the entire range of reactor periods investigated (50 Ms to 2,000 sec) at all power levels.

Hazards analyses of SGR systems have revealed that this reactor type is particularly insensitive to the "runaway" excursion. In this situation the worst result is partial melting of the fuel elements and rapid shutdown of the reactor. While extensive cleanup would be required, there would be no release of activity outside of the primary cooling system, and no hazard to the reactor environs. There are no materials in the reactor which provide a source of chemical energy for explosive accidents and no large storage of energy as in a high pressure coolant.

A more hazardous accident is the improbable one in which a small amount of excess reactivity is inserted over an extended period of time with no corrective action taken, an excursion properly termed "walkaway" incident. The hazard in this type of accident arises from the possibility of boiling the sodium contained in the upper coolant plenum to the extent of reaching sodium vapor pressures which could rupture the primary cooling system. This incident is discussed in NAA–SR–3175 "Hazards Summary for Thorium-Uranium Fuel in the SRE". A considerable amount of information is required on heat transfer to sodium vapor, Doppler coefficient as a function of temperature and other areas of uncertainty before a completely realistic anaylsis of the "walkaway" accident can be made. A development program which is currently being formulated to obtain this information is discussed below.

On the basis of information obtained to date from operation of and experiments on SRE, the sodium graphite reactor is unusually stable and easy to operate in a safe manner.

b. Reactor Safety Work Underway

At present the major source of information on the safety of SGR is the SRE, and physics

experiments on this reactor are continuing. Experiments in noise analysis for prompt neutron lifetime and experiments with the pile oscillator for power coefficient and stability, which have been conducted on the metallic uranium core, will be repeated on the Th-U alloy loading. In addition, a comprehensive program in reactor safety for the SGR is currently being detailed. Experiments are planned to measure the Doppler coefficient at high temperatures in metallic uranium, thorium and compounds of these materials. Laboratory experiments in sodium boiling are planned and an apparatus to determine burnout heat fluxes in pool boiling is in fabrication. The safety program calls for an extension of the sodium work to include the determination of burnout heat fluxes in a dynamic system, heat transfer in two-phase flow, properties of sodium vapor and heat transfer in condensation upon horizontal surfaces such as the underside of the top shield plug of the SGR.

Fuel meltdown in an excursion, while effective in stopping a reactor transient without hazard to the reactor environs, would result in an extended shutdown to permit clean up of and perhaps substantial repair to the reactor core and primary system. Melting of a "fusible link" in the fuel element or its support as a means of inserting a poison into the reactor is preferred. A program has been planned for the development of a built-in safety device for use in the SGR.

It is expected that this safety program will be carried out in fiscal years 1960 and 1961 and that the results will complete the essential data required for the design of safe, reliable plants

7. Components and Auxiliary Systems

a. Work Completed

The development of liquid metal components had its start in the SIR program. The work which has been performed at Argonne National Laboratory, in support of the experimental breeder reactor program, and at Atomics International, in support of the sodium graphite reactor program, has provided a backlog of experience in the design and manufacture of liquid metal heat transfer components. This work is discussed below:

(1) *Electromagnetic Pumps*—Liquid metals lend themselves to the use of electromagnetic pumps, and these have been developed to a rather high degree. An inherent limitation on this type of pump is its pumping efficiency. It would appear that a theoretical efficiency of approximately 45 percent can be achieved if the design is optimized, and an actual efficiency of 44 percent was achieved on the SIR main coolant pumps. EM pumps of this type are available from vendors, as a result of SIR and EBR program developments. The design parameters and technology of linear EM pumps are well known and need not be further discussed.

(2) *Mechanical Pumps*—EM pumps are generally expensive and, as their efficiencies leave something to be desired, a parallel program on mechanical pumps has been underway at both Argonne and Atomics International. Atomics International has specialized in the freeze-seal pump, in which the conventional stuffing gland, which seals the rotating shaft and prevents leakage of the fluid being pumped, is replaced by a continuously shearing film of frozen sodium. This type of pump is installed in the SRE. Its characteristics are rather well known, and possibly its major difficulty is that it relies upon a freeze-seal cooling system which is not related to the pump or reactor. Consequently, failure of this cooling system could result in reactor shutdown. Argonne National Laboratory and Atomic Power Development Associates, in cooperation with pump vendors, have developed and tested so-called liquid-metal-bearing pumps, in which the liquid metal being pumped provides a "pressure pad" type of bearing lubrication at the impeller. This type of pump has performed well on test, and is specified for the Fermi reactor installation and part of the EBR–II plant.

These pumps have a hydraulic efficiency of 70–80 percent and an overall efficiency of perhaps 60 percent as a maximum from bus-bar to fluid. In a reactor system in which low temperature rises are specified, and liquid metal flow is large, a mechanical pump may well be more desirable than the electromagnetic type from economic considerations. In any event, both types are available, and current developments promise to provide refinements which will make maintenance both direct and remote much simpler.

(3) *Valves*—Liquid metal valves are unique in their requirements for extremely high temperature operation. Valves developed on military programs have had very rigid requirements as to across-the-seat leakage. It is not at all clear that such valves, which are extremely expensive, are required in conventional powerplants. Some form of stem sealing is necessary. Two current methods have been developed to a point of feasibility. One is the bellows seal which takes the place of the stuffing box at the operating shaft. This seal must sustain considerable flexure, being expanded or compressed each time the valve is operated. A second type stem sealing is the freeze seal. This is similar to the pump seal noted above, but is somewhat simpler in that the shaft is rotated only occasionally. Each of these types is in use, in the sodium reactor experiment and in EBR–I. No difficulties have been experienced with the freeze stem valves, and bellows development has advanced to the point, through tests in the SIR program and at AI, at which considerable confidence is placed in certain vendors designs of these bellows. Twelve-inch valves procured for the HNPF, with freeze sealed stems, have shown zero leakage after 300 cycles from 500 to 1,000° F. at a differential pressure of 35 psi.

(4) *Heat Exchange Equipment*—As noted above, the heat exchange information is meager and sometimes inappropriate for the design of liquid metal heat exchangers, both liquid metal to liquid metal and liquid metal to water. Continuous development of liquid metal heat exchangers on widely fluctuating scales of effort has been underway since 1948, but only now is the realization finally appearing that liquid metal must be treated as a substance unique, and not as a common hydrogeneous fluid with slightly different film transfer properties. It is in heat exchange, of course, that liquid metals show their best capabilities; as film coefficients of 2,000 Btu/hr-ft^2 °F. are obtainable with low fluid velocities. At the same time, this brings problems, as sudden transients are rapidily reflected in the materials which make up the heat transfer apparatus. Thermal stresses result and conventional equipment, with less attention paid to thermal transients, often suffers when subjected to liquid metal service. A family of heat exchanger concepts has been developed over the past few years, but no heat-exchanger equipment has yet been built which is truly oriented to liquid metals, although some has been conceived and designed. The Fermi plant embodies possibly the most advanced liquid metal to liquid metal heat exchangers yet devised, and these appear to reflect most of the criteria believed necessary for successful fabrication of this type of equipment. It is true that some liquid metal heat exchange equipment such as embodied in the EBR–I, has performed faithfully for a period of about 8 years. This equipment, however, was designed for maximum reliability with very little concern paid to cost, size, and complexity considerations. Though not a model for future heat exchange equipment, it does demonstrate that equipment can be made to perform reliably for many years under conditions of high temperature and high temperature gradients.

(5) *Piping and Component Heating*—Because of the elevated melting point of most

liquid metals (208° F. in the case of sodium) it is necessary to preheat the system and components prior to initially filling, or refilling after draining. The SIR systems used resistance heating, and this practice was repeated in the SRE. Several other schemes are available. ANL has developed an induction heating scheme whereby austenitic piping is surrounded by ferritic piping, and the ferritic piping is heated by induction. Transformer heating, in which a loop of nonmagnetic piping material is used as a one-turn secondary of a current transformer, has been developed by AI. Several variations of this scheme are available, depending on the application. The schemes are simple and require no further development, only design for a particular application.

(6) *Controls and Control Mechanisms*—Use of "thimbles" is possible in low pressure reactors. This permits the use of control mechanisms which are not exposed directly to flowing sodium. With graphite moderation, undesirable flux peaking effects are minimal because of the long diffusion length. A problem associated with thimble type control rods is dissipation of the heat generated in the poison by neutron capture. In the SRE, the heat is removed by radiation and conduction across a helium gas gap.

Control and safety rod drive mechanisms for SGR's evolved from designs previously applied to other nuclear reactor systems. For instance, the SRE uses drives based on the recirculating ball-nut principles common in water cooled reactors. Accelerated construction schedules rendered original designs, specifically adapted for SGR system use, impractical for the SRE. While the SRE type design (described in "Sodium Graphite Reactors" by Starr and Dickinson) has proven entirely satisfactory, the rod is bulky and requires removal from the reactor face whenever fuel changing operations are undertaken. It is considered highly desirable to be able to: (*a*) bring up the fuel handling cask without removal of the drive mechanisms and (*b*) to be able to load and unload fuel with safety rods withdrawn and cocked.

Difficulties in devising control and safety rod drive mechanisms using previously developed techniques, but contained entirely below the reactor face, have been engendered by the high temperature existing above the sodium pool, and by the relatively small diameter volume into which the mechanism could be placed. It is therefore necessary to go to different concepts of control and safety rod mechanisms, adapted to the specific needs of the low pressure, high temperature SGR systems. Two ingenious mechanisms have been developed to date, illustrated in Figures 4 and 5, which permit a clean reactor face and also will allow loading of the reactor with safety rods withdrawn and ready for insertion should accidental overloading be approached.

Low system pressure permits retention of the simple mechanical seals used in the SRE system, and elaborate means of containment of a high pressure radio active coolant are unnecessary.

(7) *Instrumentation*—Reliable sodium process information, such as level, flow, temperature, and pressure is available for systems at temperatures below about 900° F. Above this temperature difficulties associated with high temperature insulation begin to appear. It appears that there is no obstacle to the use of electromagnetic flow meters at 1,200° F. in the near future, but considerable development remains before level indicators and pressure gages can be expected to perform reliably in this temperature range. Possibly the most pressing immediate need in this regard is an accurate pressure gage, which could also be used as a flow indicator in connection with a differential pressure cell. No such instrument is currently available.

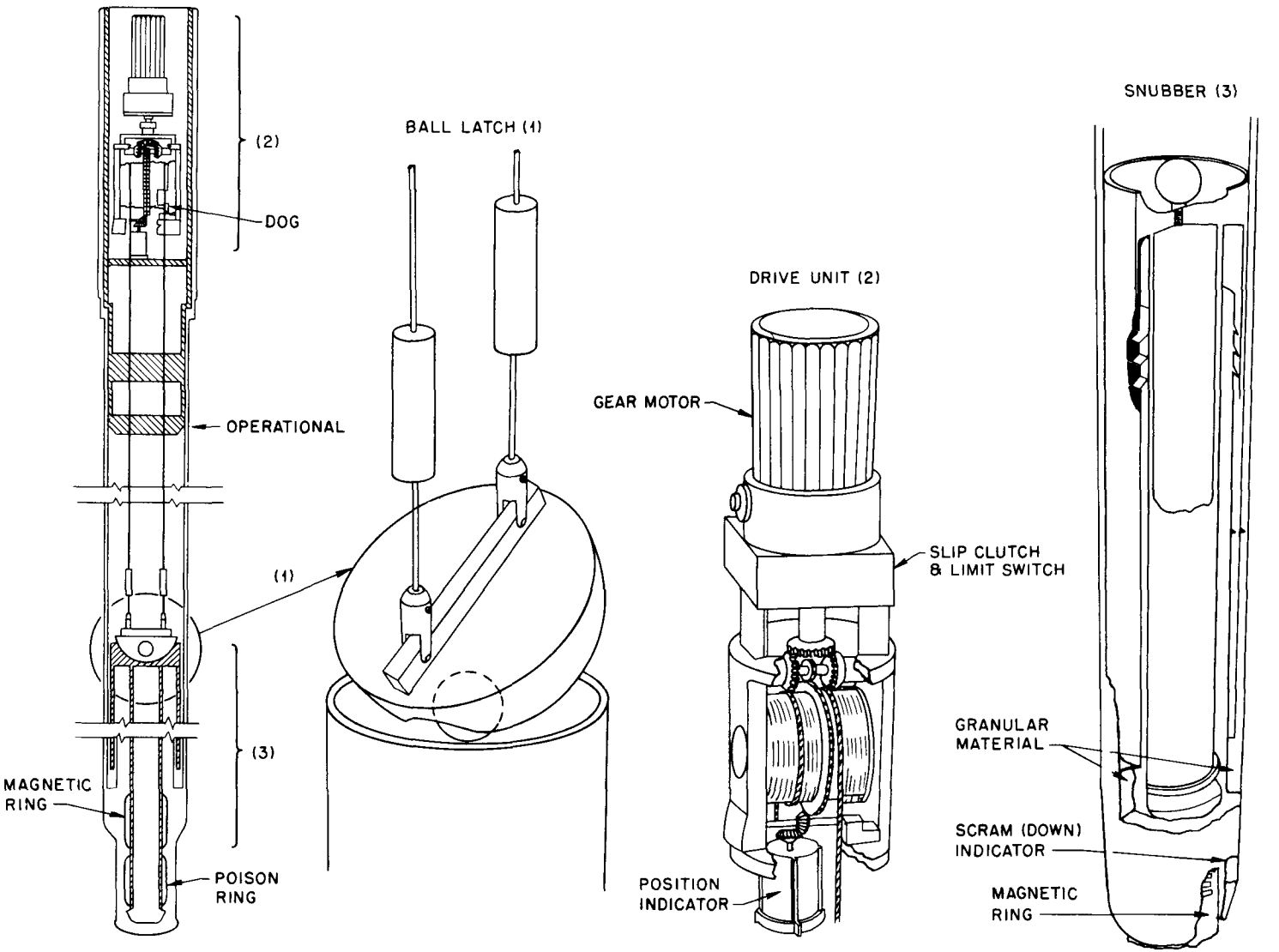

FIGURE 4.—Ball latch safety rod.

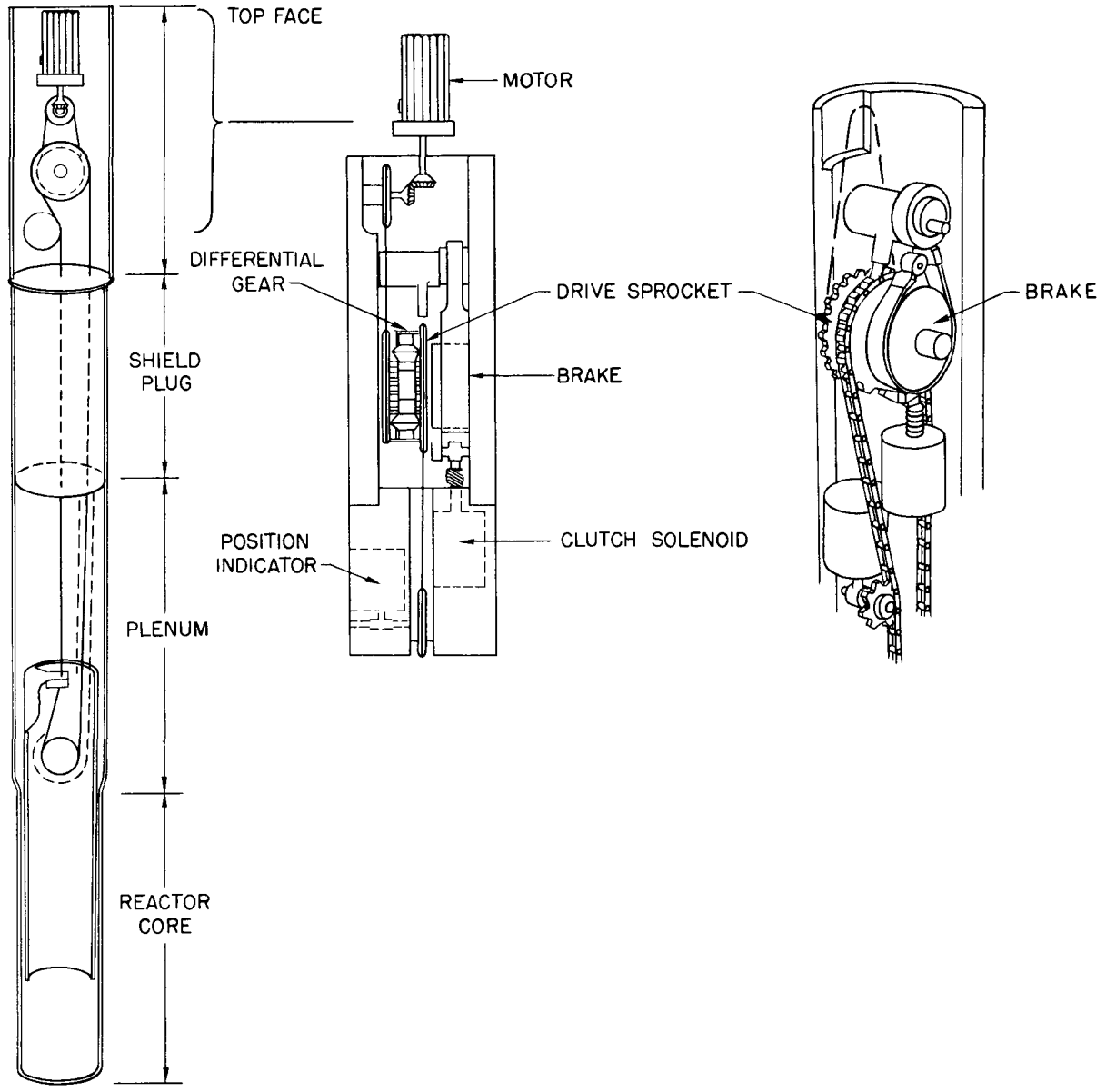

FIGURE 5.—SGR chain-drive shim-safety rod.

(8) *Inert Gas Handling*—The chemically reactive nature of most liquid metals in oxygen creates a demand for an inert gas atmosphere wherever a free surface exists in the reactor system. These inert gas atmospheres are low pressure systems, generally less than 10 psi gage, and present a problem only insofar as gas makeup is concerned. Noble gases, such as helium, have been used to date. Fragmentary evidence garnered from the SIR program indicates that nitrogen is a satisfactory substitute for the more expensive helium at temperatures up to 1,000° F. Experiments are currently in progress at Atomics International to verify the

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satisfactory performance of nitrogen at temperatures above 1,000° F. Other users of sodium systems are planning to use argon as a cover gas.

(9) *Maintenance of System Components*—The chemically reactive nature of liquid metals makes removal and maintenance of mechanical components which have been immersed in coolant somewhat more difficult than if the coolant were not a chemically reactive substance. AI has perfected the technique of removing sodium-coated components into plastic bags purged with an inert atmosphere. Primary and secondary mechanical pumps are handled this way in a routine fashion when radioactivity has decayed away. Development of components which are not wetted by sodium (primary pumps) is active at Atomics International; both Argonne National Laboratory and Atomic Power Development Associates are sponsoring work in this area. Improved system concepts and components which lend themselves to direct maintenance have been demonstrated in mockups. An example is the "overflow" sodium-cooled reactor system under study by Atomics International.

(10) *Fabrication Methods*—Alkali metals are excellent cleaning agents; notable in this regard is the use of sodium hydride for descaling steel. Consequently, it appears the "surgical" approach to reactor system construction might well be modified for these systems, providing, of course, that quality control of components is maintained. Large and small scale sodium experiments have recently been constructed at Atomics International using conventional methods of assembly. Techniques such as are practiced in oil refinery construction have been employed. It is apparent that the low pressure, nonhermetically sealed liquid metal coolant systems can be fabricated by conventional techniques with attention paid to degreasing (to reduce carbon content) but otherwise free of the necessity for elaborate rinsing, sealing, and maintenance of a "clean room" atmosphere during construction. System cleanup would, of course be accomplished prior to reaching criticality in the core to prevent unnecessary activation of impurities in the sodium, which would become a nuisance later from a maintenance standpoint. This is considered a major step, and contributes toward the eventual reduction of cost of liquid metal systems.

b. Components and Auxiliary Systems—Work Underway

In addition to the particular component development program undertaken by Atomics International (ASGR and Hallam), the Argonne National Laboratory (EBR-II), and APDA (Fermi reactor) there is in existence a sodium components development program which is designed to provide support for all sodium cooled reactors. This program is designed to produce components suitable for producing superheated steam at 1,050° F. and 2,400 psi. All types of heat exchange and system components are being developed in support of this objective encompassing steam generators, intermediate heat exchangers, sodium purification devices, and sodium instrumentation, as well as basic studies on sodium technology and mass transfer as noted above. These programs are generally carried out under contract to suppliers of allied equipment in other fields. For example, steam generator and intermediate heat exchanger contracts are in existence with Alco Products and Griscom Russell; valve contracts exist with Crane Co., Ohio Injector, and Westinghouse. A mass transfer conaract has been let to General Electric, one for materials studies with MSA Research, and hydrogen detection studies are proceeding at Atomics International. These programs are necessarily of a general nature; however, individual programs in support of sodium graphite reactors are in progress at Atomics International. An example is the program in effect at the SRE, designed to demonstrate the ultimate power

limit of this experimental reactor. Preliminary studies indicate that this reactor, which has a nominal power rating of 20 Mw, can be operated at 50 Mw thermal power with appropriate modifications to main and auxiliary heat transfer systems. Simultaneously, a system simplification program is underway at the SRE, which has already resulted in the removal of more than two-thirds of the primary system valves. Circulating system cold traps have been either removed or simplified.

The SGR advanced development program is attempting to demonstrate that certain valves are in fact unnecessary, and can be replaced with "syphon breaks" on the suction side of the lines, thus further simplifying all sodium cooled reactor systems, including the SGR. Development of system strainers is currently underway, which will permit the use of less elaborate cleaning procedures with resultant diminution of expense of fabricating these systems. Materials development is also in progress, designed to permit the use of low alloy steels at the high temperatures desired from modern sodium-cooled reactor systems, and sodium graphite reactor systems in particular. Substantial systems fabrication savings are expected from this work, which is being demonstrated on a large scale in the large components test loop. As a matter of interest, low alloy steels cost approximately 30 cents a pound, as against the $1.10–$1.25 pound for the austenitic steels currently in vogue for high temperature sodium systems. Needless to say, all development programs are being undertaken with a view to reducing the cost of fabrication and operation and to increasing the reliability of sodium graphite reactor systems. These systems are currently reasonably reliable; the above effort is being devoted primarily toward their simplification from "laboratory" into process systems.

8. Fuel Charging and Discharging

a. Work Completed

The SGR type of reactor, in its present configuration, lends itself to rather simple fuel handling procedures. A low pressure (0.5 to 2 psi) inert gas cover over the top sodium pool permits use of simple rubber gas seals. The vapor pressure of sodium at normal operating temperatures is so low (a few millimeters of mercury) that no special pressure containment features are required. In the SRE, HNPF, and their proposed successors, fuel is suspended from plugs in the top shield which are sealed with neoprene O-rings. The top of the plugs is at the reactor floor. The complete assembly of shield plug, hanger rod, and fuel cluster, is drawn up into a shielded fuel handling coffin. This fuel handling coffin contains an inert atmosphere, and during the transfer process the atmosphere over the top sodium pool communicates freely with the cask atmosphere. The transfer cask is sealed to the reactor face to prevent contaimination of the reactor atmosphere during fuel transfer.

An average of 2 hours is required to effect a complete exchange of one fuel element in the SRE; improvements in this time are expected in Hallam as more than one fuel element can be handled at a time in their fuel handling cask. No special precautions or preparations are required for fuel transfers in this type reactor, other than clearing the reactor face of any projections (such as control rod drive mechanism), which would interfere with sealing the cask to the reactor face. Irradiated fuel elements which are to be examined in a hot cell are normally cleaned of sodium to minimize the probability of sodium-air reaction in the hot cell atmosphere. Currently this cleaning operation is performed by steam and water; recent SRE experience indicates that this method is not completely satisfactory and has resulted in damage to four (out of 583) fuel elements so cleaned. In a power reactor, such as Hallam, cleaning would not normally be required until final disposition of the fuel for reprocessing.

It should be noted that SGR's have so far not been designed for changing fuel while operating at power. Neutron streaming would necessitate more elaborate shielding than is presently employed, as only gamma radiation is now contained by the cask shield. With

relatively long burnup fuels, such as carbides, and with the relative ease and rapidity of SGR refueling, it is not now considered worthwhile to devise elaborate schemes to refuel under power. Rearrangement and/or discharge of irradiated fuel can, it is anticipated, be accomplished during semiannual or annual reactor shutdown periods for maintenance of other reactor components and power generation equipment.

In summary, the low pressure systems made possible by use of sodium as a coolant, combined with the rather large (10–16 inches) lattice spacing of SGR fuel elements permits rapid and relatively simple loading redistribution, and discharge of SGR fuel.

b. Work Underway

As previously noted, the washing procedure for sodium-coated components, primarily fuel elements, has proved somewhat unsatisfactory. The bulk of sodium may be removed by agitation or emulsification in a variety of fluids; butyl alcohol is quite successful in cleaning off this wetted film, forming sodium butylate and hydrogen. In order to remove the film of sodium from a fuel element, a chemical reaction appears to be necessary. Water, of course, will react forming sodium hydroxide and hydrogen. Moist carbon dioxide will react rather slowly with sodium to form sodium carbonate and hydrogen. No cleaning agent has been discovered to date which does not release hydrogen, and therefore the possibility of a hydrogen-oxygen explosion is present. Refined cleaning methods, using moist CO_2 to form a noncorrosive sodium carbonate reaction product (rather than caustic NaOH) are currently in progress in a mockup of the SRE wash cell. Concurrently, APDA is experimenting with ultrasonic cleaning of fuel elements in a hydrocarbon bath, which appears partially satisfactory in removing the wetted sodium film. Neither technique has been perfected to date. It is apparent, however, that a substitute for the present steam and water wash of fuel elements, ancillary to the fuel handling process, must be developed. This problem will become even more urgent with the advent of uranium carbide fuel, which itself reacts more vigorously with water.

Fuel handling techniques themselves have been adequately demonstrated for SGR's of the configurations now under consideration. Design changes in fuel handling machinery will, of course, be required for different reactors, as their dimensions will be different. It appears, however, that future fuel handling techniques will be design variations on the well-grounded principles established to date.

9. SGR Systems

a. Work Completed

The SGR system for producing electric power consists, as in every reactor system, of a heat source, a heat transfer system, a steam generating system, a power conversion system, and auxiliary system to provide such services as are necessary to all the foregoing.

The reactor heat source has been adequately described; the power conversion system more nearly approaches conventional fossil fired steam plants than any other current reactor system and therefore requires no additional explanation. The heat transfer system in current and advanced SGR designs, both those built by Atomics International and studied by GE–APED, is a two-fluid arrangement. Radioactive primary sodium transfers its heat to a nonradioactive secondary system through an intermediate heat exchanger, resulting in a temperature drop of 30–100° F. depending on the efficiencies and characteristics of the heat exchanger. This, of course, results in a doubing-up of components, as pumps, piping, and accessory valves are necessary in both primary and secondary systems. The general requirement for a two-fluid system is set up by the chemical incompatibility of sodium and boiler water, which requires caution in system design in order to prevent reaction products from entering the reactor core and possibly plugging small orifices and fuel element passages. Plugging of fuel passages will result in rapid overheating of affected fuel elements, with deleterious effects

in the reactor itself. Elaborate precautions in steam generator may be taken to prevent such sodium-water reactions, in which case it might be thought, superficially, that the secondary loop would be unnecessary, and that steam could be generated directly from radioactive sodium.

SGR heat transfer systems have generally been designed as three loop systems in large central station power plant applications such as Hallam. This is done to provide maximum assurance that at least one system will be available for removal of decay or "afterglow" heat from the reactor. The SRE and SIR systems were two-loop arrangements; in case of the SIR this was done to conserve space, and in the SRE simply because it is a relatively low power experimental unit in which continued electric power generation is not paramount.

Development of the heat transfer system has been aimed at simplification; elimination of primary piping and valving is proposed in future systems, however, all current and past systems are simple piped and valved arrangements similar to those used with fluids of lesser heat transfer properties.

Auxiliary systems perform the following functions: sodium purification, provision of inert gas atmosphere over sodium, cooling of critical areas and components, preheating of piping to above the melting point of sodium, and liquid and gaseous waste disposal. Sodium purification is concerned primarily with removal of oxide. The technique of cold trapping was developed early in the SIR and EBR programs; it depends on cooling a side stream of sodium to the point where sodium oxide (Na_2O) will precipitate from molten sodium, will be strained out of the stream and retained for future disposal. This scheme is quite satisfactory for reducing oxygen constants to the neighborhood of 10 ppm. Purification beyond this limit of 10 ppm by cold trapping alone is difficult because of the mechanical difficulties of reducing and maintaining side stream sodium temperatures to a few degrees above the freezing point of sodium. With an all stainless system, this purity level of sodium is sufficient. However, when zirconium is present in the system, as it is in SRE (as moderator cladding) it is considered desirable to remove the last vestiges of oxygen from the system in order to prevent oxidation of the zirconium. Therefore, hot traps have been developed at AI, and installed in the SRE; these hot traps reduce the oxide content of sodium to a point where it is undetectable by any known chemical analysis method. The hot traps operate by absorption of oxide from sodium on large surfaces of zirconium foil which operate at a temperature well above that existing on the zirconium surfaces to be protected from oxidation. In the SRE the hot traps operate at 1,200° F., and the maximum zirconium temperature in the core is approximately 1,000° F. Pressure to operate these side streams (hot and cold trap) is normally provided by the pressure drop across the main primary pump.

Because of the chemical reactivity of sodium with oxygen, it is necessary to provide a chemically inert gas atmosphere over any free surface of sodium which exists in the system. The noble gases are satisfactory from a chemical standpoint, but only helium has been used over the core tank pool, as other noble gases are subject to neutron activation and present handling hazards of varying degrees. Therefore, helium has been extensively used in the SIR and SRE systems. Nitrogen is also inert to sodium, but because of the high temperatures at which sodium systems operate, the spectre of nitriding is present. To date, it appears that this should not be a problem at temperatures of 1,000° F or less when only steel is present in the system, but the role of impurities in sodium as transporters of nitrogen is so little known that caution is still exercised in the use of nitrogen as a low cost cover gas.

Cooling must be provided for areas within the reactor system where excessive heat might be generated, and for locations where it is desirable to freeze sodium (e.g., frozen sodium seals at pump and valve operating shafts). Cooling is normally provided to biological shields, to remove heat generated by absorption of radiation. In central station type

plants, the biological shields normally contain a high proportion of concrete, and therefore should be kept below 150–200° F. to avoid dehydration. Water is, of course, an ideal coolant, but its chemical reactivity with sodium presents a problem. Organic fluids, such as toluene and tetralin have been used, as they do not react violently with sodium. They do, however, decompose into free carbon and hydrogen, and organic substances such as naphthalene when in accidental contact with sodium. The presence of free carbon in sodium is undesirable because of its potential tendency to carburize steels, hydrogen is undesirable when zirconium is present as the zirconium may hydride and become brittle. Other organic decomposition products may result in minor hazards and nuisances. NaK appears to be a coolant which is generally compatible with sodium, performs excellently in heat transfer, and requires no special heating systems due to its low melting point. It does, of course, require an inert atmosphere due to chemical activity. NaK cooling systems are gaining favor in both SRE and Hallam plants because of the difficulties noted above with use of organic coolants.

Preheating is necessary to raise the temperature of all systems containing sodium to above the melting point of sodium. Original development work carried out for the SIR programs used resistance heaters in close proximity to coolant pipe and vessels; this scheme was followed in the SRE primarily because no additional development work was considered necessary. Experience with both SIR and SRE systems has indicated that this heating scheme is satisfactory once it is "shaken down." It is however, rather expensive to install (for instance, an estimated $600,000 was expended on SRE heating systems) and many adjustments are required before uniform heating can be achieved. Other preheating systems have been subsequently developed; ANL has originated a scheme whereby nonmagnetic sodium piping is surrounded by a shield of magnetic material, which is in turn heated by induction. This scheme is being used extensively on both EBR–II and the Fermi plant. AI has pioneered a scheme called "transformer heating" in which piping loops are used as a short-circuited single turn secondary of a current transformer. This scheme is ideally suited to simple loops, but loses some of its simplicity when applied to complicated multiloop systems. Combinations of these schemes can be used, combining the best features of each and minimizing the difficulties in each, in current sodium systems such as Hallam.

Radioactive waste in the SRE is generated primarily by the washing of fuel elements. Significant quantities of radioactive sodium hydroxide are created by this process, which must then be stored to await the decay of the 15-hour sodium-24 activity. These requirements are, of course, lessened in a central station power plant, where fuel elements are not being continually washed for later observation in hot cells, etc., and then placed in the reactor. Stainless steel fittings and tanks have proved adequate for waste material service in the SRE; waste is discharged for normal low level disposal when activity is sufficiently decayed. The potential presence of fission products in sodium, resulting from possible rupture of a fuel element, must be considered. Therefore, shielded tanks have been provided in the SRE, Fermi, and Hallam plants to accommodate sodium in the primary system which may have picked up fission product activity. To date, this has not proved to be a serious problem in the SRE due to the chemical compatibility of uranium and sodium. Occasional traces of fission gases have been observed; these have been stored in stainless steel gas storage tanks until the activity has decayed to an undetectable level. Ten thousand eight hundred standard cubic feet of radioactive gas storage is provided at the SRE; this capacity has never been fully utilized in SRE operation to date. Provision must be made, however, in any such system for handling at least two changes of gas throughout the system, which might be con-

taminated and require purging. There are no waste requirements known to exist in sodium systems other than those noted above.

b. Work Underway

SGR system work is constantly being reviewed with the advent of improved designs. Specific areas which are now actively being prosecuted are those which will lead to simplification of the rather complicated SRE and Hallam systems. The SRE itself is under continual study with this objective in mind; a notable achievement in this regard has been the removal of 14 out of 21 service system valves from radioactive or otherwise inaccessible areas.

Substitution of nitrogen for helium would result in considerable saving in operating cost, as helium now costs 8 cents a standard cubic foot commercially and 2 cents a standard cubic foot from the Government. As previously noted, concern exists over nitriding of sodium systems, because of the high temperature at which they are operated. Thin sections, such as bellows (in valves) and fuel element cladding might be susceptible to damage by nitriding. The role of impurities in sodium is under experimental study at Atomics International, as previous work by the duPont Co. at Niagara Falls (Nucleonics, June 1958) has shown that there is less than 1 ppm solubility of nitrogen in sodium. It appears that calcium and carbon act as agents in the transport of nitrogen through the system; quantitative information is not yet available to support these qualitative results. A thorough understanding of the nitrogen transport phenomenon is expected to lead to use of nitrogen as a cover gas throughout the sodium systems, in addition to its present use in inert atmosphere in piping galleries.

The use of tetralin and other organics as coolants has proved quite undesirable; experiments are now in progress to determine the engineering parameters for design of NaK cooling systems, preferably free convection systems with heat rejection to ambient atmospheres. These are short term projects; engineering results are expected this year.

C. EXPERIMENTAL REACTORS

1. Reactors Completed—Submarine Intermediate Reactors

No SGR power demonstration reactors have yet been completed although construction of the 75 MWe Hallam plant has commenced. The Submarine Intermediate Reactors, both the land based prototype and the seagoing version, have been completed. Although these are not thermal reactors, they are sodium cooled and have a resemblance to the SGR system. Some of the operating experience is therefore applicable.

The SIR Mark A, a land based prototype, developed by the Navy, for the U.S.S. *Sea Wolf*, was a sodium cooled, beryllium moderated, uranium-oxide fueled reactor. Operation commenced in early 1955 and was concluded in early 1957. This reactor was designed specifically for naval use, and employed highly enriched fuel. This was necessary to reduce powerplant size to a minimum for shipboard propulsion. Totally enclosed electromagnetic pumps, were employed to circulate the sodium. A 100-hour full power run was accomplished on this reactor after initial testing was completed. Very shortly after completion of this 100-hour run, indication of leaks in superheaters and steam generators was observed. Presence of leaks was confirmed, and reactor power was limited thereafter to the generating of saturated steam. Superheaters were bypassed. The reactor and system components, such as pumps, valves, cold traps, etc., performed quite satisfactorily throughout the operating period of this plant; the steam generators were, of course, partially inactivated. Steam generator leaks in the evaporating section were contained by carrying a "leak sealing" compound commonly employed in steam locomotives. This temporary expedient permitted operation at moderate power levels; sodium temperature and temperature gradients were reduced to minimize thermal stresses in the steam generator.

The seagoing model of the SIR, the USS *Sea Wolf* (SSN 575) underwent a very similar history. Shortly after full power operation there were indications of leaks in the superheater section of the steam generators.

The superheater sections of the steam generators were isolated, some repairs were made to the evaporator sections, and the *Sea Wolf* operated for 2 years on saturated steam. Performance of all components in the reactor system (except for the above noted deficiencies in steam generators) have been reported to be completely satisfactory. This is evidenced by the report that, during the operating period of the *Sea Wolf* following steam generation modification and repairs, at no time was it necessary to enter the sealed section of the reactor compartment. (KAPL–2008 Conf.).

Significant differences between the SIR and the SGR systems are as follows:

a. The SIR reactor operated at a mean fission neutron energy higher than that of the SGR.

b. SIR utilizes highly enriched fuel; SGR's utilize slightly enriched fuel.

c. The SIR was beryllium moderated; SGR's are, of course, graphite moderated, leading to larger core sizes.

d. SIR utilized a different method of control.

e. The SIR utilized a different primary loop arrangement than SGR.

f. SIR operated with a lower temperature rise through the core than the SGR's.

g. The SIR was necessarily designed for extreme compactness for shipboard use; SGR's as land based central type stations, are not as severely restricted in space and shielded volume requirements.

2. Operating Reactors—SRE

a. Description

The Sodium Reactor Experiment is an SGR of the "canned moderator" type in which the graphite logs are individually canned in zirconium to exclude the sodium coolant. The core coolant flows upward through fuel channels which pierce 43 of the graphite logs. Additional canned graphite logs form the reflector. The fuel elements are suspended from the top shield plug and are removed by means of a fuel handling cask. The primary sodium is collected in a pool above the core from which it is pumped through a heat exchanger to heat nonradioactive sodium in the secondary loop. This sodium is used in turn to generate steam in a steam generator-turbine system supplied by the Southern California Edison Co.

b. Purpose

The SRE, as its name implies, is an experimental installation. The steam generating equipment was installed by the Southern California Edison Co. after construction of the reactor had started. The control systems are not as yet integrated, and the Edison plant is currently operated as a "reactor-following" station. Integration of control and operation of the SRE-Edison complex as a load following system is scheduled for fiscal year 1960. A great variety of instrumentation has been provided throughout the plant to provide as complete a record of operating characteristics as possible. Maximum flexibility for performance of both reactor and system experiments has been provided. The SRE is in no sense a prototype of a central station powerplant; the generation of electricity is strictly incidental. The physics and engineering data have, of course, been extensively used by designers of the Hallam Nuclear Power Facility plant and in the preparation of more advanced designs. Power and flux distribution, fuel rod worths, temperature and power coefficients, and similar experimental information, without which accurate design of other sodium graphite reactors would be impossible, have been derived from this facility. Irradiation of large amounts of fuel at high temperature provides statistical information on fuel material behavior which is far superior to that which can be obtained from capsule type experiments in test reactors. Experimental fuels of many types have been and are now installed in the SRE. Facilities for examining these fuels and evaluating their performance are available in the SRE building. The experimental facilities permit rapid removal and inspection of fuel. The SRE has fulfilled its basic mission in providing:

Experimental information on the physics of SGR systems,

A high temperature, high temperature gradient source of sodium for liquid metal system and component experiments, and

A high temperature fuel irradiation facility.

c. SRE Chronology

April 1955—Construction started.

February 1957—Construction completed.

March 1957—First dry critical experiment.

April 1957—First wet critical experiment (sodium in core).

July 12, 1957—Initial power generation (8 MWt, 2 MWe).

August through November—Critical and power experiments.

November 1957—SRE dedicated.

January 1958 through April 1958—System modifications completed to permit operation at full power; construction and installation of eddy current brakes to reduce post scram convection flow of coolant.

May 1958—Full power operation (20 MWt, 6 MWe).

May 1958 through May 1959—Continued power operation and experimental work. Over 15,000,000 KWH or electricity generated, total reactor energy output 2,200 MWD. Examination and evaluation of standard (unalloyed) uranium fuel and experimental fuels continued during this time. Numerous engineering experiments in support of HNPF development work were performed, primarily in the areas of control rod heat generation, oxide fuel evaluation, and power stability tests. Pile oscillator tests by two separate methods indicated reactor stability.

May 1959—Reactor operated with 1,010° outlet temperature and 480° temperature rise across core at full power.

May 1959—1,000° F. steam generated with reactor outlet temperature at 1,060° F., maximum sodium temperature from hottest channel 1,080° F.

A detailed chronology of the reactor operating history from inception through initial full power operation is contained in Chapter 7 of "Sodium Graphite Reactors," by R. W. Dickinson and C. Starr, one of the presentation volumes for the Geneva Conference of 1958.

d. SRE Significant Design Parameters

(1) *System (quantity)*	*Value*	*Limitation reason*
Kwt	20,000	Operating restriction from ACRS—not permanent; 50,000 kw capability apparent.
Kwe	6,000	7,500 KW capacity of turbine.
Thermal efficiency	29%	825° temp. limit on installed turbine.
Specific power, kw/Kg U–235	240 (avg)	Reactor power level.
Power density, kw/ft³	176.5	Reactor power level (parameter not germane to this type of reactor).
Peak to avg. power ratio	1.625	Core geometry.
Operation during fuel loading	No	Removal of control drives to operate cask, neutron streaming through cask.
Core size	6′ x 6′	Sufficient for power required—no inherent limitation.
(2) *Physics*		
Reactivity available	$3\%\frac{\Delta k}{k}$	Enrichment of first fuel loading.
Control rods/MWe	.667	None known.
Core tolerances	Close	Assembly of canned moderator.
Temp. coefficient of reactivity	1×10^{-5} Δk/° F @400° F., $.2\times10^{-5}$ Δk/° F @750° F., (see fig. 1).	Graphite temperature.
(3) *Fuel and Material*		
Central fuel temperature	1,100° F. (U fuel)	Transition temp. of uranium.
Fuel exposure MWD/MT	1,100 avg	U fuel limited by radiation damage to ~1,500 MWD/T.
Fuel element tolerance	±.001 diameter and length of slug. ±.001 diameter of cladding.	
Refuel cycle	2 hrs per element	Handling cask designed for single element transfer.
Core fuel inventory	300 kg U	43 fuel positions designed into core.
System fuel inventory	Not applicable.	
Corrosion and erosion of clad and structural materials.	Negligible	Sodium, steel and zirconium compatible chemically.
Conversion ratio	0.51 (initial)	Small reactor and decreased eta due to high temperature.
(4) *Coolant*		
Temperature	1,010° F. reactor outlet	Structural strength of Zr moderator cladding.
Pressure	40 psi max	None known.
Δt across core	480° F	Intermediate heat exchanger capacity.
Boiling temperature in sodium	1,620° F	Saturation temperature of sodium.
Radioactivity	0.28 curie/cc primary coolant	Dependent on flux level.
Corrosion and erosion	Negligible so far	Long term mass transfer effects may appear.
Chemical reactions	Sodium reacts with air and water. Neutral gas atmosphere over sodium required. No reaction with fuel or clad.	Not applicable.

(4) *Coolant*—Continued	*Value*	*Limitation reason*
Dissociation rate	Negligible	1% of sodium will transform to Mg—24 in 100 years of full power operation.
Purification	~30 ppm oxygen should be maintained.	Cold traps can reduce 0_2 to 10 ppm. Hot traps can reduce 0_2 to essentially zero.
Leakage	Negligible	Significant only in case of major break; not expected.
Maintenance	Radioactive system maintenance must await N^{24} decay; inert atmosphere should be maintained.	Half life of Na—24, chemical activity of Na with oxygen.
(5) *Moderator*		
Temperature	1,000°–1,200° F	Nuclear considerations only (energy of thermal neutrons).
Pressure	Solid moderator (graphite)	Not applicable.
Boiling temperature	Not applicable to this type reactor.	
Radioactivity	Negligible	Activity contributed by minor impurities.
Corrosion and erosion	Negligible	Graphite clad in zirconium. Sodium penetration of graphite introduces neutron poison if cladding fails. No exothermic reaction.
Reaction with coolant	None if cladding remains sound.	1% volume increase if impregnated with sodium. No reaction.
Dissociation or decomposition	None known	See above for effects of cladding failure.
Leakage	Solid moderator	Not applicable.
Maintenance	None so far	Moderator can-handling cask available for removal in case of clad failure.
(6) *Heat Transfer*		
Clad surface temp	525°–1,010° F	Set by coolant temp.
Hot channel factors (max.)	1.07	See Chap. 3–2.2 in "Sodium Graphite Reactors," by R.W. Dickinson and C. Starr, presentation volume for the Geneva Conference of 1958.
Variation of coolant tube diameter	1.07	
Variation of orifice diameter	1.01	
Eccentricity of fuel element	1.05	
Radial asymmetry of core	1.02	
Deviation from predicted radial power.	1.05	
Total hot channel factor	1.12	
Average heat flux, (Btu/ft² hr)	209.000	Reactor power level (10^6 possible).
Maximum heat flux, (Btu/ft² hr)	340,000	Reactor power level (10^6 possible).
(7) *Components and Auxiliary Systems*		
Refueling	Flexible; can be done in 2 hours per element after reactor shutdown.	Crane speed and cask operating time.
Heat exchangers	20 megawatt capacity for IXH; 30 megawatt capacity for steam generator.	Heat transfer surface limited in IHX; feedwater stability problems in steam generator.
Thermal stresses	Core tank nozzles and IHX overstressed during uncontrolled scram transient controlled by eddy current brake.	Overcooling of system due to excessive thermal convective sodium flow.
Chloride corrosion	None to date	Once-through steam generator design demands ~0.5 ppm impurities.

(7) *Components and Auxiliary Systems*—Continued

	Value	*Limitation reason*
Fission product circulation	Detectable but too low to measure accurately.	Post-shutdown accessibility to system. Not currently a problem.
Activity levels	5 mr/hr after Na–24 decay	Mass transfer effects over long time (5 years) may increase activity.
Safety	Excellent	Low pressure, compatible materials, prompt negative temperature coefficient.
Maintenance	Good	Na–24 activity decay, possible mass transfer effects (long term); requires attention to prevent possible sodium-moisture reaction.

e. Reactor Section and P & I

A vertical reactor section of the SRE appears in Figure 6. A schematic P&I diagram is presented in Figure 7.

f. Operating Experience with SRE

(1) Thermal energy generated—2,200 MWD.
(2) Electrical energy generated—15,000,000 kilowatt-hours.
(3) Coolant activity levels.
  (*a*) At full power; 0.28 curies/cc Na–24 gammas.
  (*b*) At shutdown; 0.28 curies/cc Na–24.
  (*c*) 24 hours after shutdown; ~0.16 curies/cc.
  (*d*) During maintenance; negligible Na–24 activity, 1–2 mr/hr mass transfer activity.
(4) Equipment Problem—The intermediate heat exchanger has not performed according to specifications. An LMTD of 90° F. is required rather than the design value of 60° F. Thermal circulation of sodium within the shell has been observed, leading to after-scram stresses. A replacement heat exchanger of improved design has been ordered. Difficulties have been experienced with the freeze seal pumps. Tetralin, which is used to cool the seal, has been introduced into the primary circuit; which does not create any hazard but decomposes into amorphous carbon which may permit carburization at elevated operating temperatures. Some bellows-sealed valves have failed at the bellows; these difficulties were traced to improper operation and further occurrences are not expected. Initial cold trap design was of insufficient capacity; redesigned equipment is now installed and is capable of removing any amount of oxide so far introduced into the system. The most significant equipment problem was a high stress condition which existed around core tank nozzles after scram. This condition has been corrected by installation of eddy-current brakes which limit post-scram convective sodium flow and reduce thermal transients in the nozzle vicinity to an acceptable value. No other equipment problems of a nonroutine nature have been experienced.
(5) *System Problems:*
  (*a*) Load following—The SRE has been operated as a load forcing, rather than a load-following machine. However, test transients of up to 20 percent power increase per minute have been induced and successfully followed manually with the reactor. The success of these tests has created confidence in the load following characteristics of the reactor, and a load-following control system has been designed. This system is being installed and will be completed early in fiscal 1960. After its installation, power operation of the SRE will be computer-programed from the Southern California Edison load dispatching station.
  (*b*) Transient problems—The transient convective sodium flow following scram has been the principal problem encountered to date. Thermal convec-

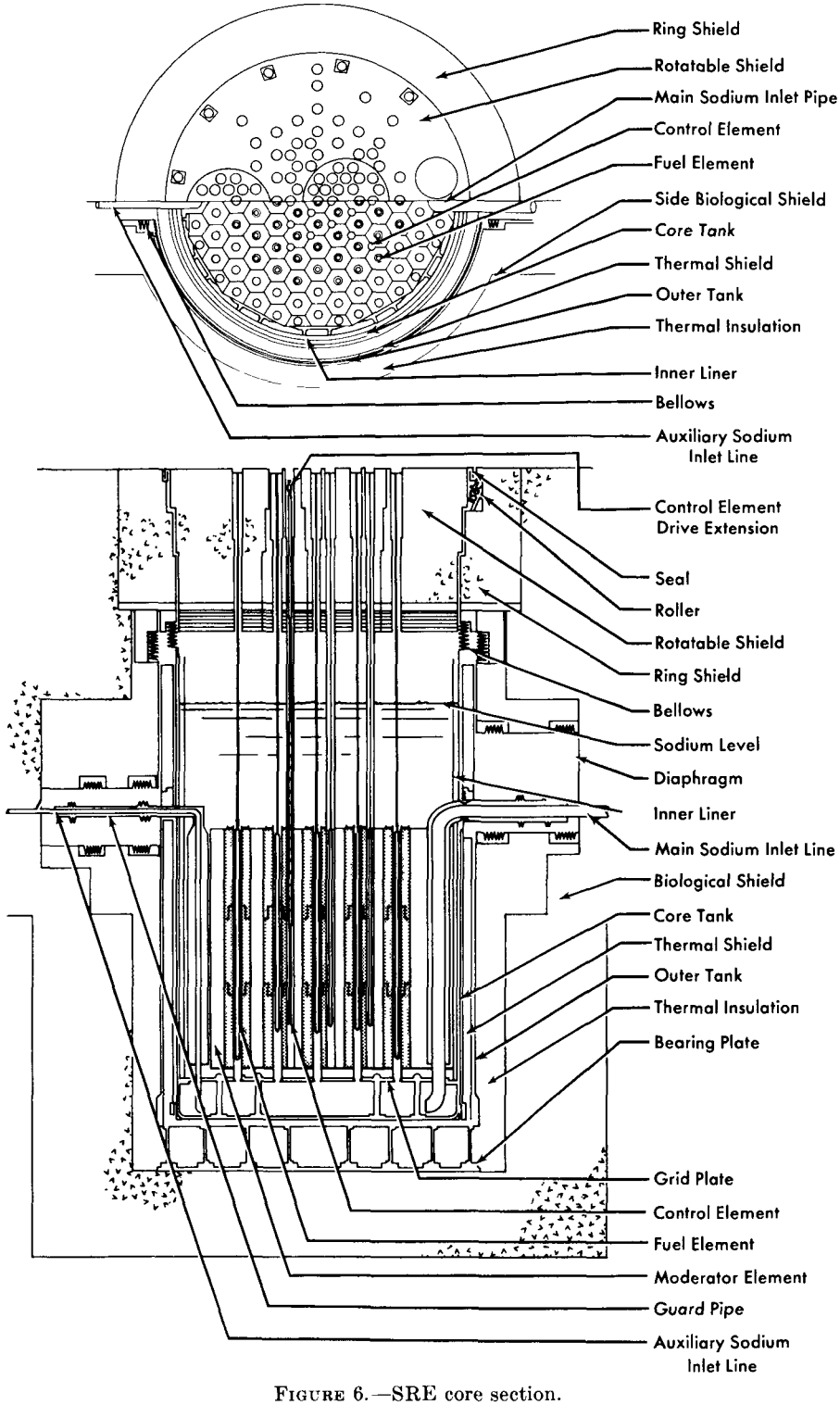

FIGURE 6.—SRE core section.

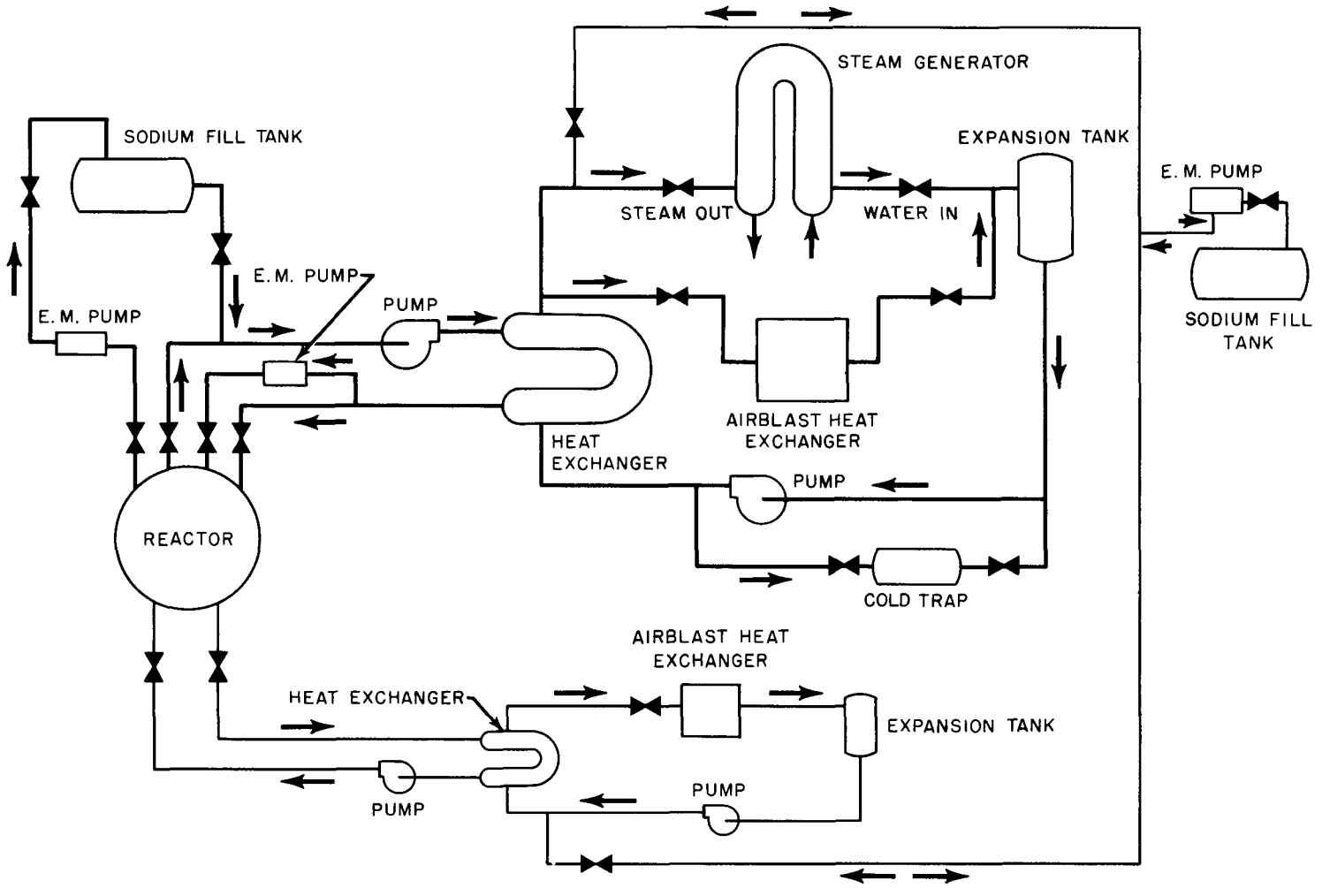

FIGURE 7.—SRE P and I diagram.

tive flow is most difficult to compute accurately during design of a liquid metal system, and a means for controlling the post-scram flow was not initially incorporated into the SRE system. Later installation of this equipment has eliminated this major transient problem. The reactor itself has proven exceptionally stable and easy to control under transient conditions.

(6) Fuel Failures—Fission product gases have been detected in the inert gas atmosphere above the top pool of the SRE. Quantitative measurements of this activity have so far proved impossible due to the low level of activity; detection has been by gamma ray spectrometer only. It is not known whether this activity is due to minute imperfections in fuel cladding, or is a result of "tramp uranium" left on the outside of the fuel elements during fabrication operations.

A procedure has been developed for cleaning sodium from fuel elements removed from the core. This operation is performed after fuel element removal, and has no connection with reactor operation. It is required only when fuel elements are to be examined or disassembled in the SRE hot cell. Four fuel elements have been damaged in this washing operation (out of 583 washing operations performed). Damage has consisted of crushing the cladding above the fuel-bearing region of the fuel rods and is due to some form of overpressure. No fuel has been lost by this damage, but once it has occurred, the fuel elements are unsuitable for restoration to the core. Intensive efforts to improve the fuel element cleaning procedure are now in progress to avoid future incidents of this type.

Expansion of the unalloyed uranium fuel has apparently occurred in the most highly irradiated (center) sections of the central fuel elements. The 10-mil clearance between fuel slug and cladding has been closed by fuel expansion. A maximum of 4 mils distortion of the cladding has been observed after 1,100 MWD/T irradiation. No rupture of the fuel cladding has occurred; observation of reference fuel elements is continuing in order that irradiation may be stopped before reaching cladding failure.

(7) General Maintenance—Maintenance of the SRE system has proved surprisingly easy. Techniques for removing components which have been immersed in sodium, such as coolant pumps, have been developed to the point of great confidence in their use. Once the sodium-24 activity has decayed, maintenance operations on piping systems, including both cutting and welding operations, are performed on a routine basis. The sodium is simply frozen in the pipe concerned, and the cut area is purged with an external source of nitrogen gas when the sodium is exposed. The sodium is cut back from the work area and the opening taped until rewelding operations are performed. Many such operations have been performed to date. When large pipes, such as the 6-inch primary coolant piping are to be cut, the sodium is drained from the system prior to cutting, but no special measures are taken to remove residual sodium. Oxide formed in the pipe by these operations dissolves in the sodium when the system is heated and filled. The oxide has proved to be easy to remove in cold traps, which keep oxide content below 10 ppm in the primary system. The main obstacle to quick repair of the sodium system is awaiting decay of the sodium-24 activity. By advance scheduling of down-periods, it has been possible to minimize the outages necessary to permit decay. Other work requiring the reactor to be subcritical or just critical is scheduled for the decay period. Access to the primary system may be expedited by draining the radioactive sodium back to the shielded fill-and-drain tanks. If urgently required, access can be had to the piping galleries about 5 days after shutdown. Normally,

a decay period of about 10 days is taken.

Maintaining of the sodium system in the SRE has been quite simple in general. System and component designs based on improving those areas which have been somewhat troublesome can be expected further to simplify maintenance procedures in future sodium cooled reactor systems.

3. Experimental Reactors Underway (SCRE)

a. Description

The Sodium Cooled Reactor Experiment (SCRE) has been proposed as a test reactor which incorporates the essential features of the most promising advanced SGR. The principal design advances to be investigated in this experimental reactor will be the following:

(1) "Calandria" core vessel—the graphite moderator will be contained within a stainless steel vessel. The tubes which form the fuel element channels will pass through close-packed hexagonal graphite logs and be welded to the upper and lower heads of the calandria. Differential thermal expansion of the tubes will be provided for by means of a bellows at the top of each tube. This design will avoid the use of individually canned moderator elements, resulting in improved neutron economy and lower core fabricating costs.

(2) "Overflow" primary sodium system—the primary sodium will be pumped upward through the core into the top sodium plenum. The sodium from the top plenum will overflow a baffle wall downward through the intermediate heat exchanger, which will be located in an annular region around the calandria vessel. Electromagnetic pumps will return the sodium to the lower plenum under the reactor core. In this design all of the primary sodium system will be confined to the reactor vessel, eliminating piping and galleries for the primary system. The result will be an increase in the safety of the reactor and a substantial saving in construction cost.

(3) Titanium hydride neutron shield—A neutron shield, composed of iron, titanium hydride, and boron steel will prevent activation of the secondary sodium in the intermediate heat exchangers. The shield will form the baffle wall over which the primary sodium flows to the intermediate heat exchangers.

(4) Intermediate heat exchanger—A heat exchanger of unusual design will be tested It will consist of a bundle of tubes bent into a series of U-bends with a baffle between bends. A header at each end of the tubes will support the tubes which are otherwise free to expand. The design will permit easy replacement of the exchanger and permits considerable economy in manufacturing and material costs.

(5) Electromagnetic sodium pump—A novel electromagnetic pump will be incorporated into the design. The part of the pump which bears the windings will be located against the outside of the reactor vessel below the intermediate heat exchangers. The return path for the magnetic field will be a block of stainless-clad iron laminations (without windings) located inside the reactor vessel, so that the magnetic field will travel through the stainless steel wall of the reactor vessel. The throat of the pump will consist of an annulus formed by thin sections of stainless sheet, which will be supported on one side by the vessel wall and on the other side by the iron core. The section of the pump which carries the windings will not be in contact with sodium and can be replaced with minimum difficulty.

The postulated advantages of this system are:

(*a*) Advantage is taken of all thermal stratification and natural convection in the primary sodium.

(*b*) All major nozzles have been removed from the reactor thus minimizing stress problems and promising greater reliability.

(*c*) All moving parts such as pump impellers and valves have been removed from the primary circuit thereby increasing reliability.

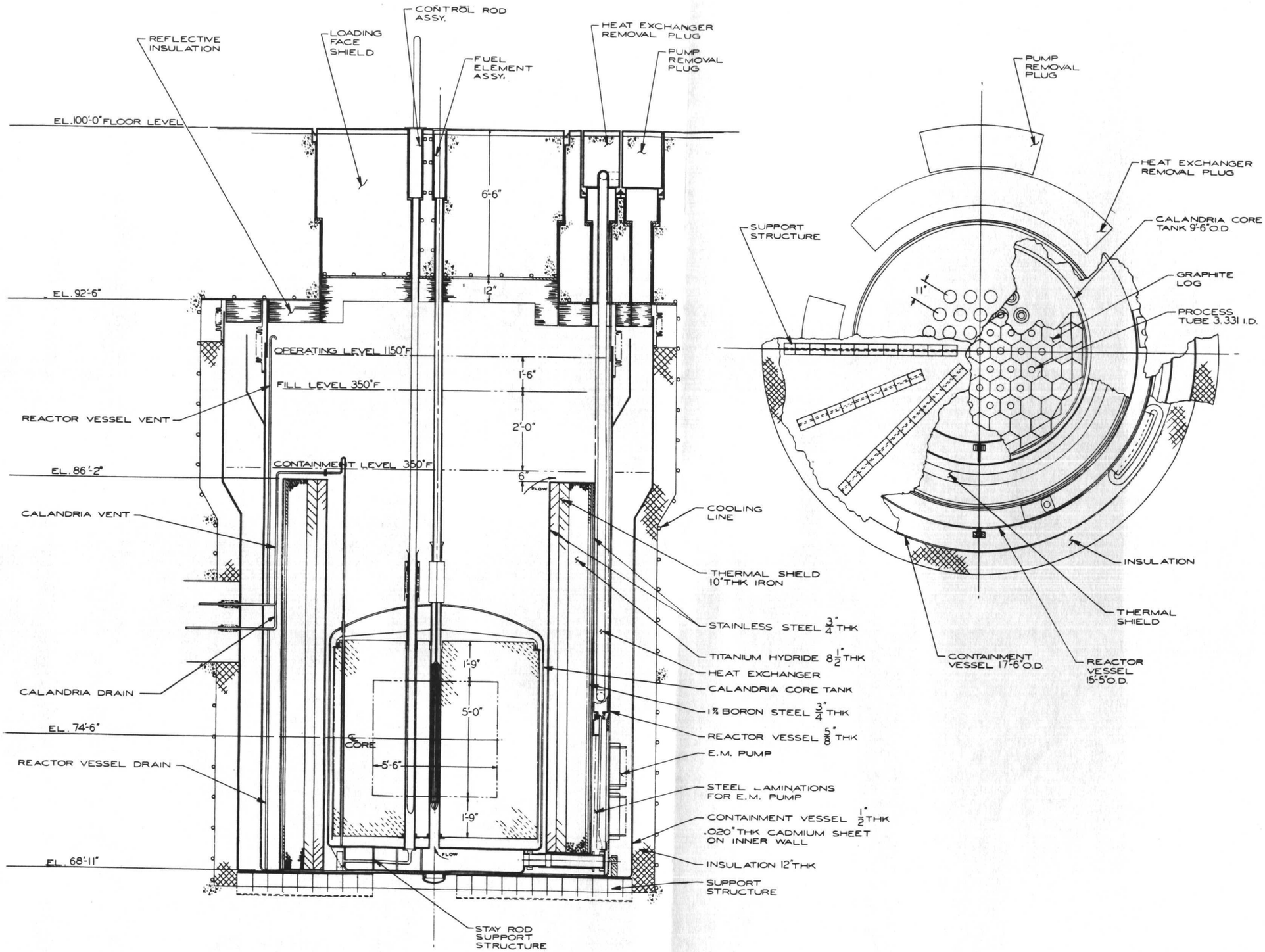

FIGURE 8.—SCRE vertical section.

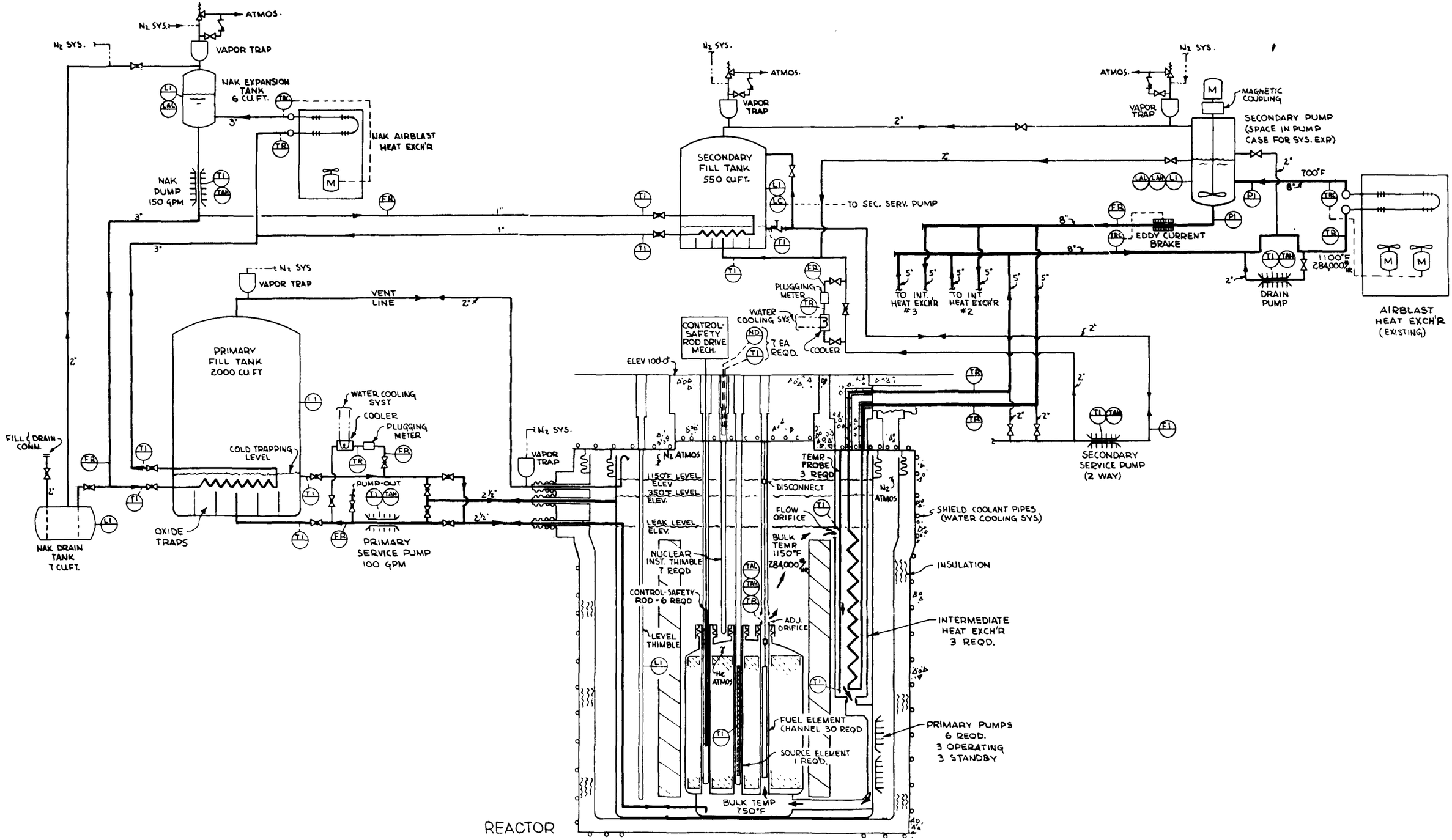

FIGURE 9.—SCRE P and I diagram (part I).

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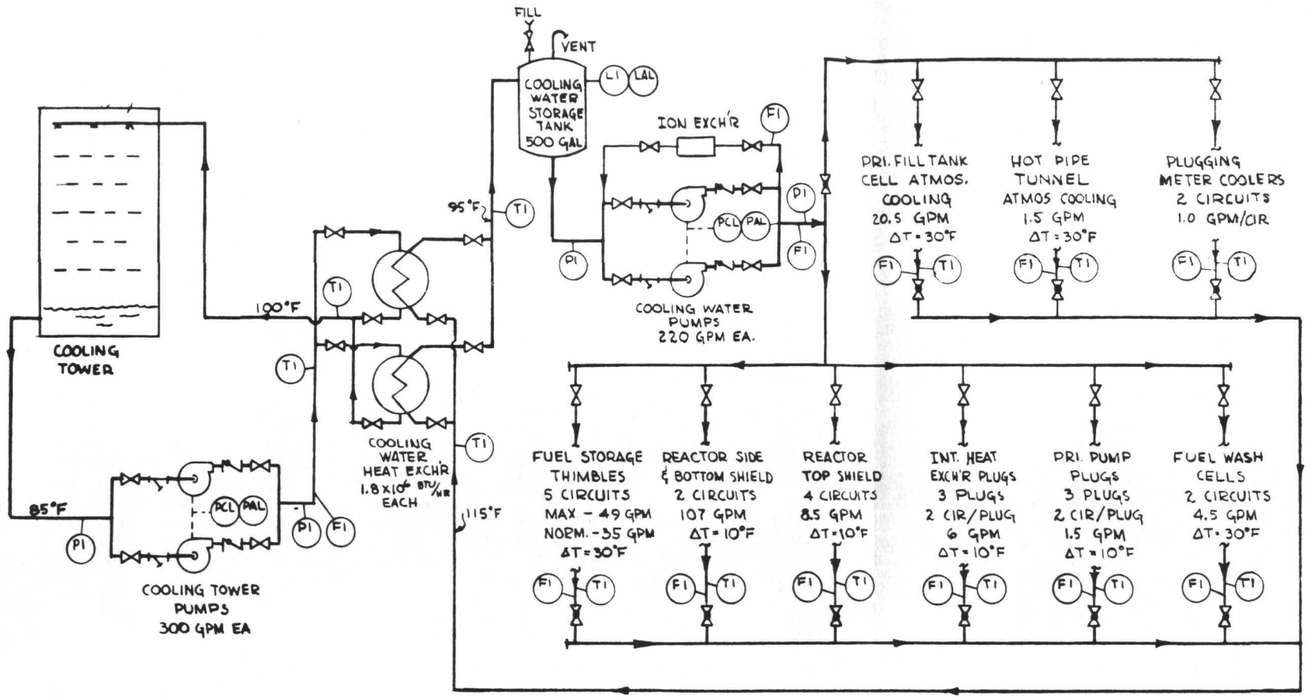
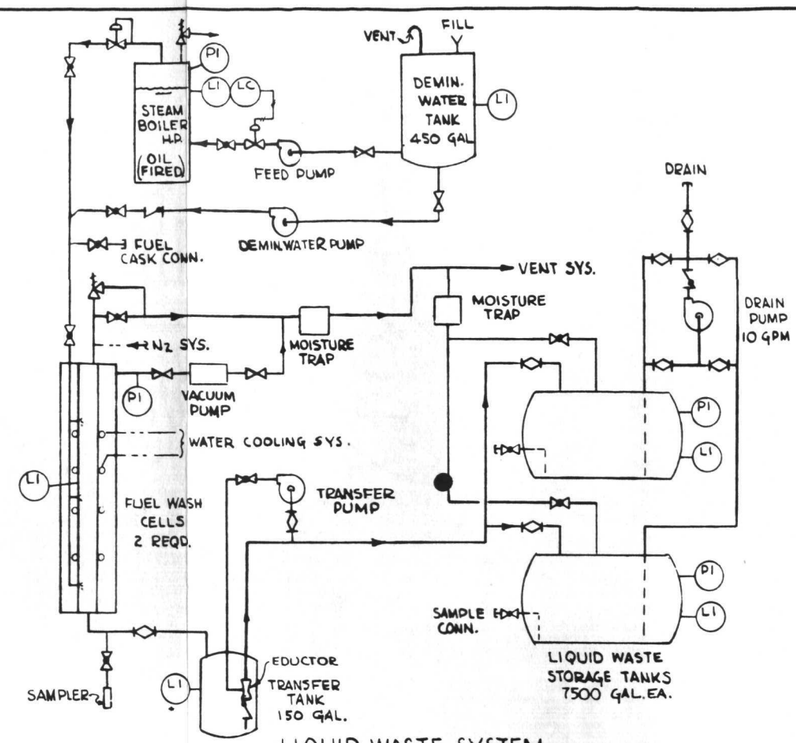
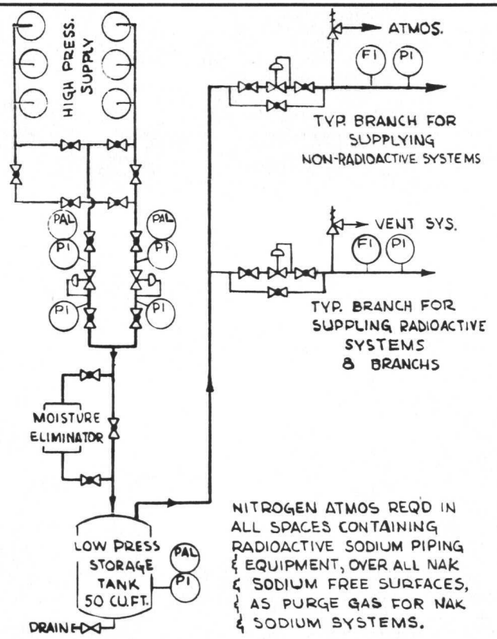


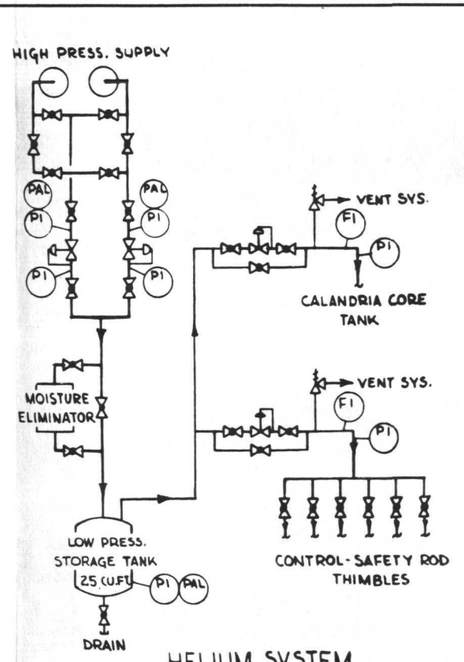
WATER COOLING SYSTEM



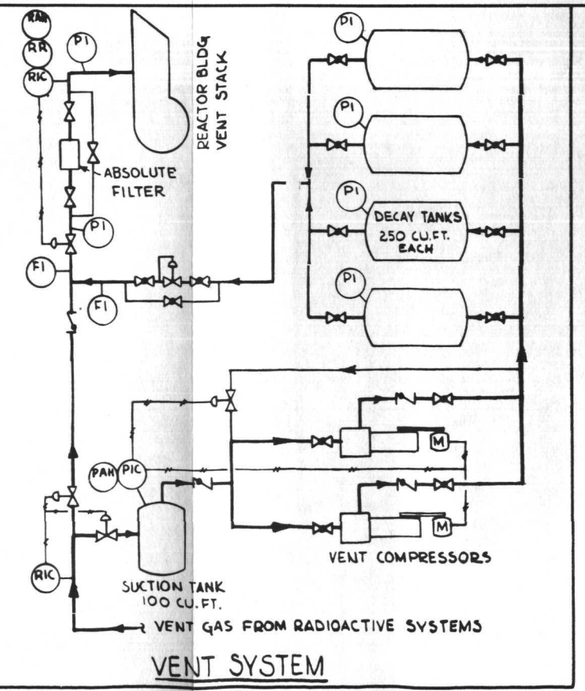
LIQUID WASTE SYSTEM



NITROGEN SYSTEM



HELIUM SYSTEM



VENT SYSTEM

FIGURE 9.—SCRE P and I diagram (part II).

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(*d*) This general arrangement should result in important reductions in cost without sacrificing quality by permitting the use of less expensive construction techniques than have been employed in the past.

b. Purpose

The purpose of this reactor would be to provide engineering information on which to base the design of a full scale power plant. More specifically the reactor will provide a heat source and radiation field to demonstrate the following:

(1) The reliability of the calandria concept as modified to fit the requirements of an SGR.
(2) The temperature of the graphite at specific points in the core and the reflector and the effect of these temperatures on reactor physics and materials in a radiation field.
(3) The merits of the intermediate heat exchanger as conceived for this reactor design.
(4) The effectiveness of the neutron shield in preventing activation of the sodium in the secondary system, and the long term compatibility of the shield materials at high temperature in a radiation field.
(5) The fabrication feasibility and the problems connected with this reactor design including the reactor vessel, calandria, neutron shield. EM pumps, and heat exchangers.
(6) The convective circulation paths and flow, and thermal gradients in the wall of the reactor vessel and other components.
(7) The potential of a plant of this design as a source of competitive nuclear power.

c. Chronology

The SCRE is in the preliminary design phase. Some development work has been completed on the major components. A prototype pump was built and tested in a geometry which simulated that of the reactor vessel wall. The pump produced a flow of 280 gpm against a head of 8 psi. When operated from a 60-cycle power supply, the pump performed with an overall efficiency of 5.4 percent as predicted. Because of the geometry of the pump throat, the optimum frequency of the power supply is about 15 cps, which would yield a pump efficiency of about 20 percent. Frequency converters would be used in the full scale plant.

Development work on the titanium hydride shield is also in progress. Short term compatibility of TiH pellets in 1,200° F. sodium has been established. Neutron attenuation measurements were made on a representative section of the proposed shield, and the data are currently being analyzed.

A small (66-inch diameter by 70-inch high) calandria tank complete with process tubes and bellows has been fabricated. This work was done under the program for the SRE second core, but the experience applies directly to the SCRE.

d. Significant Design Parameters

A study now underway at AI is concerned with establishment of the appropriate power level for SCRE. The design parameters in the following table are based on use of carbide fuel and a thermal rating of 40 MW.

A vertical section and a P&I diagram for SGRE are presented in figures 8 and 9.

TABLE X.—DESIGN PARAMETERS FOR SCRE

(1)	Coolant	Sodium.
(2)	Moderator	Graphite.
(3)	Thermal power rating (kwt)	40,000.
(4)	Electrical power rating (kwe)	13,600.
(5)	Thermal efficiency	34%.
(6)	Coolant temperature and pressure.	625°–1,200° F. 40 psi max.
(7)	Core size	4.88′ dia., 4.75′.
(8)	Power density (kwt/ft³ of core)	343.
(9)	Peak-to-average power ratio	1 2 axial. 1.4 radial.
(10)	Fuel enrichment	7%.
(11)	Conversion ratio	0.3.
(12)	Specific power (kwt/kg of U-235).	490.
(13)	Central fuel temperature	1,950° F.
(14)	Core inventory (MT)	1.2.
(15)	Δt (° F.) across core	575° F.
(16)	Average heat flux (Btu/ft²-hr)	4.8×10^5.
(17)	Maximum heat flux (Btu/ft²-hr)	8.2×10^5.

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D. POWER DEMONSTRATION REACTORS

1. Power Demonstration Reactors Underway—Hallam

a. Description

The Hallam Nuclear Power Facility (HNPF) consists of a 254 MWt Sodium Graphite Reactor and the associated equipment for generating and delivering 800 psig, 825° F. steam to a turbogenerator. The site, in southeastern Nebraska 19 miles from Lincoln, was selected with special consideration to requirements for a nuclear powerplant.

Atomics International Division of North American Aviation, Inc., has been assigned the technical responsibility for design of the nuclear facility. Consumers Public Power District (CPPD) will provide the turbogenerator and will operate the reactor under contract with the AEC.

The reactor is sodium cooled, graphite moderated, and will use slightly enriched uranium-10 w/o molybdenum alloy as a fuel for the first core. Structurally, the arrangement of the reactor, similar to the SRE, is known as a "tank-type, canned-moderator" to distinguish it from other proposed SGR configurations. The reactor core is located below the level of the main operating floor of the reactor building. The graphite moderator is separated from the sodium by encasing graphite logs in stainless steel. Rod-type fuel elements are suspended from plugs in the shielding above the reactor core. Control rods operate in thimbles that are similarly supported. Drive mechanisms for these control units are supported on a carriage over the shield. Refueling is accomplished by a mechanized, shielded cask which is supported from a gantry, and which can be indexed to engage the shield plugs after removal of the control actuator carriage. A radioactive primary sodium system transfers heat from the reactor to a nonradioactive sodium system which generates steam.

The HNPF is a load-following, automatically controlled system between 20 percent and 100 percent of design power. Provision is made for semiautomatic operation at all power levels. In addition, a plant protective system provides for rapid detection of any off-normal condition in the plant and automatically takes corrective action. A more complete description of HNPF can be obtained from NAA–SR–2960 "Preliminary Proposal for the Hallam Nuclear Power Facility, Hallam, Nebraska" and NAA–SR–3379, "Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility."

Information obtained from construction and operation of the SRE as well as other advancements in SGR technology have dictated design features in the HNPF which differ from the SRE design. A major departure from the SRE design is the use of stainless steel for the moderator cladding. This change permits higher coolant temperatures and takes advantage of the cost differential between steel and zirconium. The fuel channels have been relocated to the corners of the hexagonal moderator element to improve neutron economy. Other significant design changes from the SRE are the use of single speed shim-rod drives and the removal of essentially all sodium service equipment and valves from the primary and secondary system areas. Some of these changes have also been incorporated in the SRE since construction was completed.

The initial fuel loading at HNPF will be an alloy of uranium-10 w/o molybdenum. It is tentatively planned to refuel with uranium monocarbide fuel elements.

b. Purposes

The following statement of the purpose of the Hallam Nuclear Power Facility is abstracted from the CPPD–AEC contract: "The parties

MASTER SCHEDULING
ISSUED 6-23-59

NORTH AMERICAN AVIATION, INC.,
ATOMICS INTERNATIONAL DIVISION

HALLAM NUCLEAR POWER FACILITY
CONDENSED SCHEDULE STATUS CHART

PROJECT A-81
G.O. 7508
7518
7533

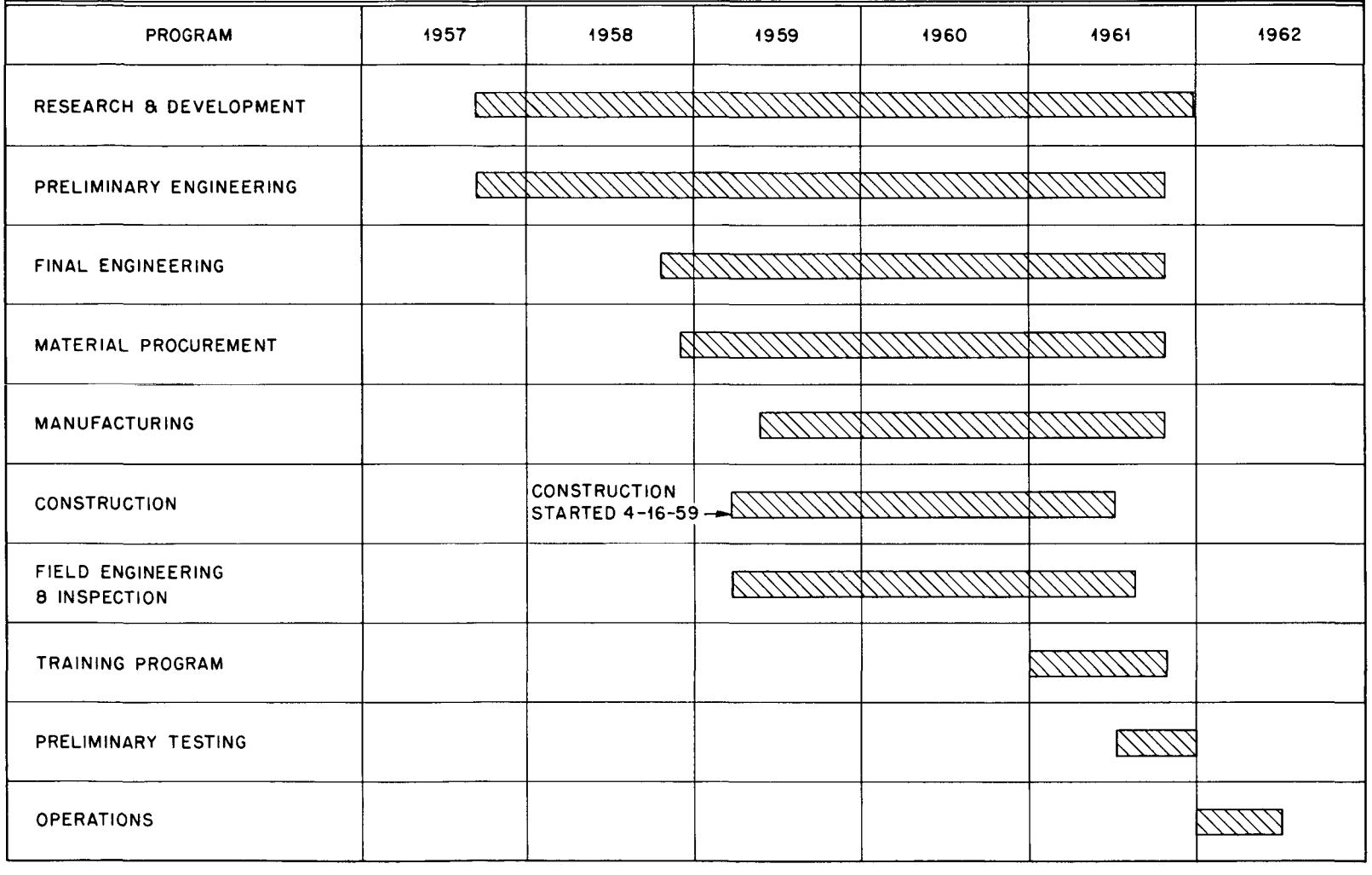


FIGURE 10.—HNPF schedule.

intend to demonstrate the economic and technical practicability of central station power utilizing a sodium graphite nuclear reactor by operating the plant for the primary purpose of producing the maximum amount of electric energy therefrom, thereby obtaining technical knowledge and practical experience in the use of nuclear power and possible byproducts for civilian purposes. In so doing, it is intended to utilize and extend the knowledge obtained through the Sodium Reactor Experiment (called the SRE), which North American Aviation, Inc., is presently placing into operation for the commission. It is anticipated by the parties that this demonstration will permit a major advance toward realization of civilian nuclear power and that such information will lead to further advances in subsequent power reactors."

c. Chronology

In 1955, having reviewed various types of power reactors, CPPD submitted a proposal to the AEC based on the sodium-cooled graphite-moderated reactor (SGR). The outcome of the proposal and the objectives of the project are embodied in the contract, AI(11-1)-513 entered into by CPPD and the AEC, September 20, 1957.

The Atomics International Division of North American Aviation, Inc., was directed to proceed with engineering and development on October 28, 1957, under the terms of Contract AT(11-1)Gen-8. The Bechtel Corp. was designated by the AEC as the architect-engineer for the nuclear facilities in April 1958. CPPD engaged Stearns-Roger Manufacturing Co. as architect-engineer for the conventional power plant facilities.

The HNPF schedule provides for 50 months from the start of title I design to initial power operation. Construction of the facility started in April 1959. System and area acceptance and preoperational testing by operating personnel will begin during the latter part of 1961. Electric power generation is scheduled for June 1962.

Future plans for Hallam include an irradiation facility which would employ radioactive sodium from the reactor as a source of gamma rays, for sterilization of agricultural products raised in the vicinity.

The Hallam scheduling is presented in Figure 10.

d. Hallam Significant Design Parameters

System	*HNPF*	*Reason*
Electrical power:		
Net	76 MW	Contract agreement.
Gross	80 MW	
Thermal power:		
Nominal	240 MW	Net thermal efficiency of steam cycle.
Design	254 MW	
Net thermal efficiency	31.6%	Steam temperature and pressure.
Specific power (avg)	260kw/gm U-235	*Economic optimization*—Economic advantage to fuel subdivision and smaller core offset by increased fuel cycle cost. Optimization based on U-10 w/o Mo alloy.
Power density:		
Core	132 kw/ft^3	(This parameter not germane to SGR's).
Fuel	1.95 kw/in.^{3}	Economic optimization.
Peak-to-average power ratio	2.5	Determined by fuel disadvantage factor (enrichment and rod size) and reflector savings.
Reactor operating during refueling.	No	Access to reactor face prevented by control rod drives—decay heat removal—reactivity pertubations by fuel removal.

System—Continued	*HNPF*	*Reason*
Core:		
Diameter	13.5 ft	No apparent limit, size set by core design of HNPF.
Height	13.5 ft	
Volume	1930 ft^3	
Physics		
Reactivity available excess reactivity (hot, clean) ($\Delta k/k$).	5.5%	Reactivity requirements—2.5% equilibrium Xe, Sm poisoning 2.5% fuel depletion, 0.5% control.
Control rods/MWe	.408	
Shim-regulating rods/MWe Safety rods/MWe	.25	Control rod worth and excess reactivity.
	.158	Separate shim-safety rods.
Core tolerances	Close (±.010″) across flat of hex. moderator logs.	Canned moderator design requires close tolerances because of assembly difficulties.
Temp. coefficient of reactivity:		
Fuel temperature	-3.05×10^{-5}/° F	Doppler coefficient.
Sodium temperature	$+0.5\times10^{-5}$/° F	Sodium density.
Graphite temperature	$+2.2\times10^{-5}$/° F	Neutron temp and change in microscopic cross-section.
Fuel and materials		
Central fuel temp. (max):		
U–10 w/o Moly	1,260° F	Decrease in strength and increase in swelling of fuel if operated above this temperature.
UC	1,600° F	Radiation data currently limited to 1,600° F. max. fuel temperature.
Fuel exposure (avg):		
U–10 w/o Mo	3,000 MWD/MT	Radiation damage to fuel.
UC	10,000 MWD/MT	Loss of reactivity.
Enrichment (a/o U–235):		
U–10 w/o Mo	2.85%	Sufficient reactivity to achieve expected burnup.
UC	3%	Sufficient reactivity to achieve expected burnup.
Fuel element tolerances	Nominal	Liquid metal bond (0.020″ on radius) permits use of as-cast fuel slugs.
Refuel cycle:		
Operating time between refueling:		
U-Mo	6 mos	Fuel element lifetime.
UC	1 yr	Convenience of Electric Utility Co.
Shutdown time for refueling	24-hr+1 hr for each element changed.	Time required to reduce sodium temperature, operate cask and then return to operating conditions.
Core fuel inventory	33 MT-U	Economic optimization of nuclear and heat transfer engineering studies.
System fuel inventory	68 MT-U	Average exposure of fuel removed—300 MWD/MTU—80% plant factor.
Corrosion of clad and structural material.	Negligible	Negligible corrosion at these temperatures if sodium impurities, especially oxide are kept at 60 ppm or less. Experience with SRE indicates less than 10 ppm of oxide can readily be maintained.

Fuel and materials—Continued	*HNPF*	*Reason*
Erosion of clad and structural material.	Negligible.	
Initial conversion ratio	0.51	Relatively large portion of epithermal fission in low eta region, high neutron leakage (~ 6%), use of Mo in fuel and stainless steel moderator cans.
Coolant		
Exit temperature	945° F	Hottest channel limited to 1,050° F.
Pressure	¼ psig	No pressurization required.
Δt across core	338° F	Saturation temperature of steam cycle.
Boiling temperature in system	1,620° F	Boiling temperature of sodium at atmospheric pressure.
Radioactivity	.16 curies/gm—Na	Decay of Na-24.
Corrosion	Negligible	Effect of mass transfer undetermined.
Erosion	Negligible.	
Reaction with:		
Oxygen	$4Na+O_2=2Na_2O_2$	Inert atmosphere and exclusion of moisture required.
Water	$2Na+2H_2O=2NaO_2+H_2$	
Fuel	No reaction.	
Clad	No reaction.	
Moderator	Absorption of sodium by graphite.	Minor swelling and distortion of graphite.
Dissociation or decomposition	None.	
Purification	Removal of oxide by cold trapping.	Prevention of corrosion.
Leakage	Negligible.	
Maintenance	Cold trap replacement. Occasional replacement of small valves. Routine pump inspection and maintenance.	Decay of Na-24.
Moderator		
Temperature (avg)	450° C	Neutron energy—no structural limitation.
Pressure	Not applicable.	
Boiling temperature	Not applicable.	
Radioactivity	Negligible	Activation of impurities.
Corrosion	Clad with SS	Absorption of sodium in graphite would cause swelling and distortion of graphite.
Erosion	Clad with SS.	
Reaction with:		
Air	Inert dry atmosphere	Graphite oxides at elevated temperatures in air.
Water	Maintained in moderator elements.	
Fuel	No reaction with fuel.	
Clad	Carburization of SS.	
Dissociation or decomposition rate	Not applicable	Most radiation damage annealed during operation.
Purification	Not applicable.	
Leakage	Not applicable.	
Maintenance	Moderator element replacement possible.	

Heat Transfer	*HNPF*	*Reason*
Clad surface temperature (max)	1,000° F	Outlet coolant temperature.
Hot channel factor	1.17	Flux contingency 10%. Thermal conductivity cont. 7% (Other contingency factors distributed in heat transfer calculations).
Average heat flux	121,000 Btu/hr ft.^{2}	
Maximum heat flux	302,000 Btu/hr ft ^{2}	Fuel temperature.
Components and Auxiliary System		
Refueling—U–10 w/o Mo	Semi-annual	Expected life of central fuel element.
—UC	Annually	Convenience of utility.
Heat exchangers	IHX—single wall U-tube Steam. Generators—double wall tubes.	Need for separation of sodium and H_2O.
Thermal stresses	50° F per hour—maximum temperature change for sodium.	Stresses in moderator elements and other components.
Chloride corrosion	No problem	Steam generation fabricated of 2½ Cr—1 Mo steel. Superheaters fabricated of 5 Cr—½ Mo steel.
Fission product circulation	No problem	Primary loop shielded—fuel compatible with coolant—only small amount of fission product release expected with fuel cladding failure.
Safety	No problem	No pressurization—compatible materials.
Maintenance	Direct maintenance of pumps and heat exchangers. Radioactive core components replaced when necessary.	Decay of Na-24.

A cross section view and P & I diagram for the Hallam reactor are presented in Figs. 11 and 12.

2. The 300 MWe Canned Moderator Reactor

The 300 MWe Canned Moderator Reactor is essentially the 75 MWe reactor (based on Hallam) scaled to four times the net electrical power. Steam conditions and sodium temperatures are identical to those of the 75 MWe plant. Although similar in all respects, the two reactors have certain differences worth noting. To obtain four times the power from a core only slightly larger, a 61-rod fuel element was chosen. The rod diameter is smaller, which improves heat transfer from fuel to coolant. Coolant flow is carried in 4 loops versus 3 for the 75 MWe size. A slightly higher fuel enrichment is required, due primarily to the additional stainless steel of the 61-rod element and the larger volume of sodium in the core.

Auxiliary equipment consists of 4 primary and 4 secondary pumps, 4 intermediate heat exchangers, 8 steam generators, one turbine-generator, 2 operating feedwater pumps, 1 stand-by feedwater pump and 5 feedwater heaters.

Net plant efficiency increases from 31.6 to 34.0 percent for the 300 MWe size. This gain is due to the higher efficiency inherent in larger turbogenerators and to an improved feedwater heating cycle.

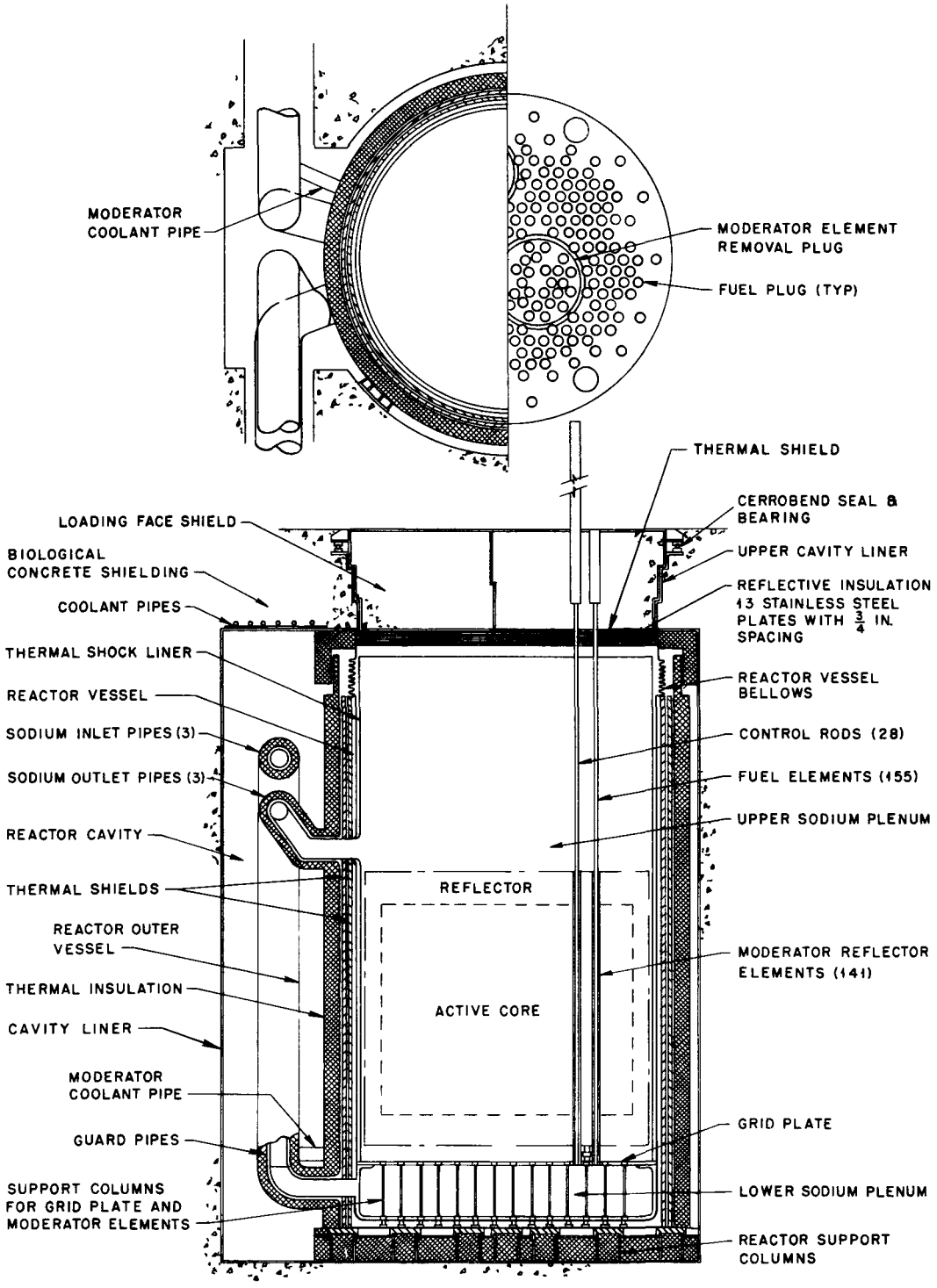
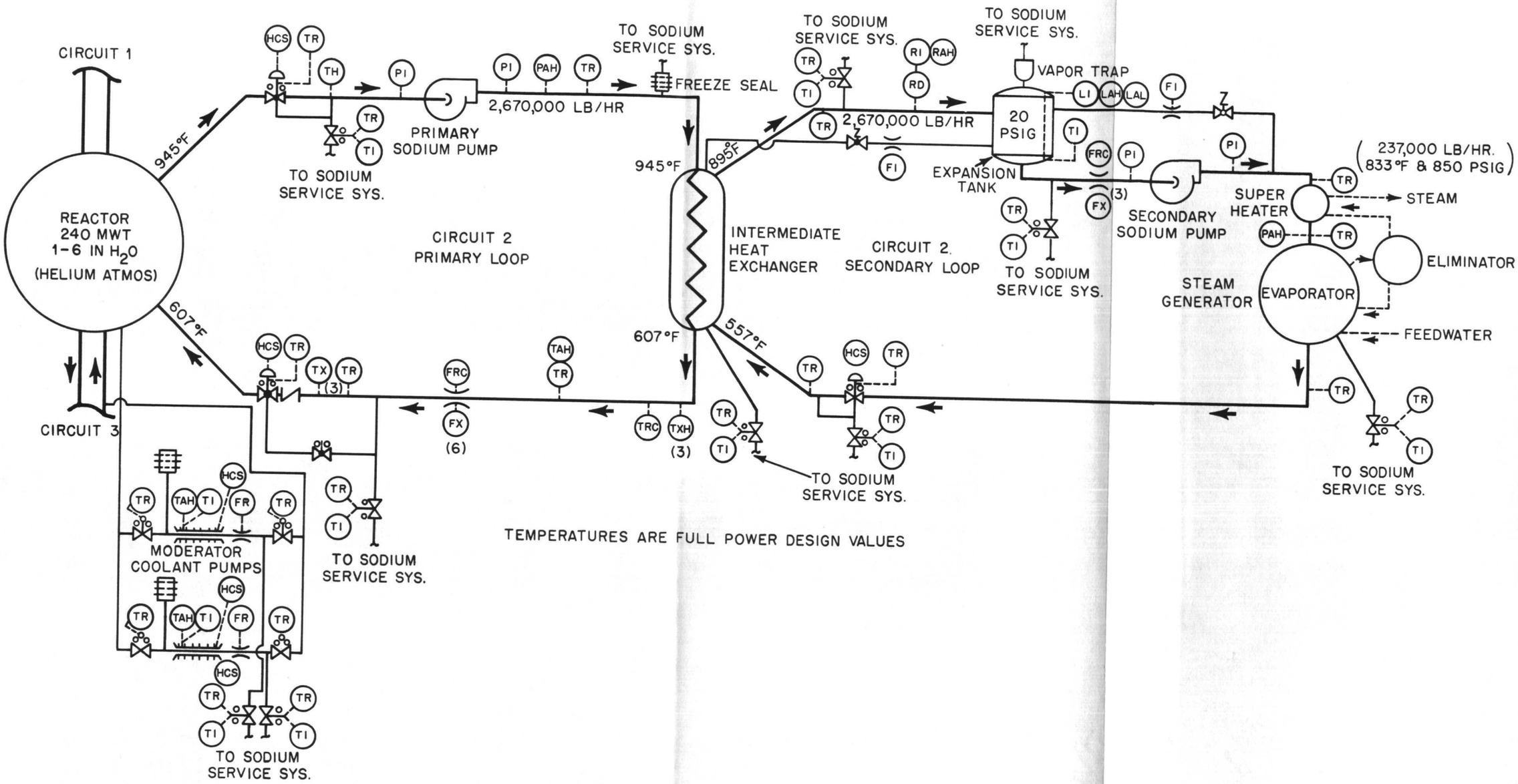


FIGURE 11—HNPF vertical section



TEMPERATURES ARE FULL POWER DESIGN VALUES

FIGURE 12.—HNPF P and I diagram.

TABLE XI —300 MWe SIGNIFICANT DESIGN PARAMETERS

System

Electrical power	
Nominal net	300 MW
Design net	315 MW
Thermal power	
Nominal	884 MW
Design	928 MW
Net thermal efficiency	34 0%
Average specific power U-235	0 57 kwt/gm U-235
Uranium	18 kwt/kg-U
Power density	
Core	350 kw/ft ^{3}
Fuel	4 12 kw/in ^{3}
Peak-to-average power ratio	2 5
Reactor operation during refueling	No
Core.	
Diameter	15 0 ft
Height	15 0 ft
Volume	2,650 ft ^{3}

Physics

Reactivity available Excess reactivity (hot, clean) ($\Delta k/k$)	8 7%
Control rods/MWe	111
Shim-regulating rods/MWe	068
Safety rods/MWe	043
Core tolerances	Close

Fuel and Materials

Central fuel temperature (maximum)—U-10 w/o Mo	1,260° F
Fuel exposure (average)—U-10 w/o Mo	3,000 MWD/MT
Enrichment (a/o U-235)—U-10 w/o Mo	3 15%
Fuel element tolerances	Nominal
Refuel cycle	
Operating time between refueling—U-Mo	3 mos
Shutdown time for refueling	24 hr+1 hr for each element changed
Core fuel inventory	31 6 MT-U
U-Mo alloy	57 3 MT U-10 w/o Moly
System fuel inventory	82 5 MT-U
Corrosion of clad and structural material	Negligible
Erosion of clad and structural material	Negligible
Initial conversion ratio	0 478
Rods per fuel element	61
Fuel slug diameter	0 33 in
Sodium bond gap	0 10 in
Stainless steel cladding	0 010 in
Number of fuel elements	218

TABLE XI.—300 MWe SIGNIFICANT DESIGN PARAMETERS—Continued

Coolant

Exit temperature	945° F.
Pressure in upper plenum	¼ psig.
Δ t across core	338° F.
Boiling temperature in system	1,620° F.
Corrosion	Negligible.
Erosion	Negligible.
Reaction with:	
Oxygen	$4Na+O_2=2Na\ O_2$.
Water	$2Na+2H_2O=2Na\ O_2+H_2$.
Fuel	No reaction.
Clad	No reaction.
Moderator	Absorption of sodium by graphite.
Dissociation or decomposition	None.
Purification	Removal of oxide by cold trapping.
Leakage	Negligible.
Maintenance	Cold trap replacement—occasional replacement of small valves. Routine pump inspection and maintenance.
Sodium loops	4.
Flow per loop	7.3×10^6 lb/hr.
Nominal pipe size	24″ ϕ Schedule 10.

Moderator

Temperature (average)	450° C.
Pressure	Not applicable.
Boiling temperature	Not applicable.
Radioactivity	Negligible.
Corrosion	Clad with SS.
Erosion	Clad with SS.
Reaction with:	
Air	Inert dry atmosphere.
Water	Maintained in moderate elements.
Fuel	No reaction with fuel.
Clad	Carburization of SS.
Dissociation or decomposition rate	Not applicable.
Purification	Not applicable.
Leakage	Not applicable.
Maintenance	Moderator element replacement possible.

Heat Transfer

(1) Clad surface temperature (max)	1,000° F.
(2) Hot channel factor	1.17.
(3) Average heat flux	164,000 Btu/hr ft².
(4) Maximum heat flux	410,000 Btu/hr ft².

TABLE XI.--300 MWe SIGNIFICANT DESIGN PARAMETERS--Continued

Steam Cycle Equipment

(1) Steam generator	2 per loop.
Duty per loop	7.35 x 10^8 Btu/hr.
Steam produced per loop	725,000 lb/hr.
Steam pressure	850 psig.
Steam temperature	833° F.
Feedwater temperature	410° F.
(2) Turbine generator system	One required.
Heat rate	9,500 Btu/kw-hr. gross.
Gross generator capacity	315 MW.
Net electrical to lines	300 MW.
Steam enthalpy	1415.4 Btu/lb.
Feedwater enthalpy	385.8 Btu/lb.
Total steam flow	2.9 x 10^6 lb/hr.
Specific volume steam	0.849 ft^3/lb.
Main steam piping	20-inch diameter schedule 80–2 loops.

Components and Auxiliary System

(1) Heat exchangers	IHX –single wall U-tube Steam generators — double wall tubes.
(2) Thermal stresses	50° F. per hour—max. temp. change for sodium.
(3) Chloride corrosion	No problem.
(4) Fission product circulation	No problem.
(5) Safety	No problem.
(6) Maintenance	Direct maintenance of pumps and heat exchangers. Radioactive core components replaced when necessary.
(7) Primary and secondary pumps:	
Capacity/pump	18,000 gpm.
Pumping power each	725 hp.
Head	150 ft.
Type	Freeze seal.
Shaft HP	650 hp.

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