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Proceedings of

**SYMPOSIUM
ON
SODIUM
REACTORS
TECHNOLOGY**

held in
Lincoln, Nebraska

May 24 - 25, 1961

UNITED STATES ATOMIC ENERGY COMMISSION
Division of Technical Information

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Sponsored by
U. S. Atomic Energy Commission
Division of Reactor Development
and
Consumers Public Power District
Lincoln, Nebraska

Preface

The Sodium Reactors Technology Symposium was held May 24 and 25, 1961, in Lincoln, Nebraska. The meeting was cosponsored by the U. S. Atomic Energy Commission and the Consumers Public Power District of Nebraska as part of the education and promotional function of the AEC Division of Reactor Development.

The symposium provided the occasion for various organizations and industry now working with or closely associated with reactor projects in which sodium is used as a heat-transfer medium to exchange current information on common problems. In addition to the status type of reports, the panel session in particular projected future thinking of the represented organizations. The differences of opinion expressed during the panel session in regard to the most promising fuel for sodium-cooled reactors is well worth reviewing and analyzing.

Full international participation was handicapped by the relatively short time allowed for announcements and invitations, as indicated by the list of attendees.

This document comprises the conference proceedings and includes the papers in the same order as listed on the agenda. The postmeeting answers to the questions submitted by the attendees and the list of attendees are located in the appendix.

In most cases the authors' prepared papers were used for this document. Those papers taken from the oral presentation by the court reporter, Albert Morwitz, were submitted to the authors for editing and approval. The program chairman has attempted to eliminate mistakes and misstatements in the process of editing, with particular attention to the transcribed discussion portion of the panel session.

Grateful acknowledgment is made of the services of the session chairmen, G. W. Wensch and C. A. Pursel; of the dinner chairman, T. A. Nemzek; and of the program committee, D. P. Rudolph and L. A. Redecke. Special recognition is given to C. D. Sayre for the arrangement of the conference and liaison with various organizations in Lincoln, Nebraska.

THOMAS P. HECKMAN*

Program Chairman

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*U. S. Atomic Energy Commission, Chicago Operations Office.

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Welcome

R. L. SCHACHT

Consumers Public Power District, Nebraska

I am Ray Schacht, General Manager of the Consumers Public Power District, Nebraska (CPPD). It is my pleasant duty to welcome all of you to Nebraska for this Symposium on Sodium Reactors Technology.

As you know CPPD is participating in the Hallam Nuclear Power Facility, which you will visit tomorrow; and we feel particularly pleased that this group has selected Lincoln as a place to come and study the problems of sodium technology.

This reactor development in which CPPD has an interest is something that is a bit different than the mere generation of power. To Nebraskans it is a symbol of doing something in the state in order to open up new areas of development. We, as are all other states, are looking for ways to expand our economy. We feel that this reactor project offers some possibilities for that purpose.

Because of the state-wide interest in this project, I am certain that the governor of the state, Governor Morrison, the mayor of this city; and other officials of the state will give you no less a hearty welcome than I do.

Perhaps some of you have wondered why CPPD has involved itself in this particular project. We are a relatively small organization as electric utilities go; and I think probably to understand why we became interested in it, you should know something about Nebraska. I know some of you have backgrounds in Nebraska.

I think probably one of the outstanding characteristics of Nebraska is that the conventional is unconventional. It is a state of paradoxes. It is a very conservative state. In many respects life was rugged for the pioneers in this state, and we are not too many generations away from those pioneers. Politically it has been a strong Republican state for many years. Its constitution prohibits any bonded indebtedness for the operation of the state itself. We have a one-house legislature. Also, Nebraska is an all-public-power state. Several famous men have come from Nebraska. Among them have been William Jennings Bryan, Charles Dawes, George W. Norris, and General Pershing.

That, I think, may give you some idea of the background of this state and perhaps may clarify my previous statement that we are interested in something more than just the mere generation of power; we are also interested in doing something for Nebraska.

I hope that your stay here for the next few days will be pleasant and productive. We hope that you will like the city of Lincoln, we hope that you will like the state of Nebraska and its climate—we shall try to control the weatherman.

Again I should like to say that it is a distinct pleasure to welcome each and every one of you to this meeting.

Introduction and Historical Summary

GLEN W. WENSCH
U. S. Atomic Energy Commission

Distinguished guests, gentlemen, and sodium lovers of America, it is an honor to share the platform today with a group of individuals who are becoming an increasingly scarce breed in our civilization today. I am, of course, referring to these pioneers who have shown a great deal of initiative in participating in a bold and advanced program of the civilian nuclear-power industry, the sodium-cooled-reactor program.

It is fitting to refer to the pioneering spirit in Nebraska which contributed so much to the growth of America following the Civil War through the Homesteading Act of 1862. Although Nebraska is noted primarily as a bountiful agricultural state, it has also been a fountainhead of literature, music, art, and drama. We all recall Old Jules in Mari Sandoz's novel about the early growth of this area. Nebraska has produced a number of outstanding leaders; among these are William Jennings Bryan, Roscoe Pound, General John Pershing, General Alfred Gruenther, and the Reverend Edward Flanagan of Boys Town. On the nearby campus of the University of Nebraska, you can see the statue of Abraham Lincoln by Daniel Chester French, the sculptor who produced the famous statue of Lincoln in the Lincoln Memorial at Washington, D. C.

From what I have said about Nebraska, it should be no surprise to find that one of the world's largest sodium-cooled power plants is located nearby. This plant, the Hallam Nuclear Power Facility, will produce the highest quality steam ever produced in a nuclear utility system and should maintain the traditions of boldness and vigor which are characteristics of this area. The Consumers Public Power District of Nebraska (CPPD), the operator of the plant for the AEC, is well known for its progressive outlook.

The conference we are attending today is one of a series held from time to time through which the AEC seeks to keep American industry informed on the state of development of civilian nuclear-power reactors. Because of the large number of research and development, reactor operations, and construction activities that are underway in the Sodium Reactors Program, the conference presentations will be based on a broad, rather than a detailed, basis.

We all know that power can be generated most efficiently by utilizing heat at high temperatures. We also know that some of the major design problems connected with the design of power reactors are centered around selection of the coolant that removes heat from the reactor core. I should like to review the basis for the AEC's continued belief that sodium is an excellent reactor coolant. Table 1 compares the important heat-removal properties of six coolants that the AEC has investigated.

Note that of the six coolants listed, three are capable of transferring high-temperature heat at atmospheric pressure. This points up one of the marked advantages of liquid-metal coolants. Low system pressures in the reactor and heat-removal systems greatly simplify design of vessels, piping, fuel-handling devices, procedures, and control-rod mechanisms.

At first glance lithium would appear to be a better coolant than sodium. However, costs, compatibility problems, and an unfavorable neutron-absorption cross section rule out lithium

as a coolant for use in civilian power reactors. The bismuth-lead eutectic is seriously handicapped by high pumping costs.

Figure 1 shows the liquid-phase temperature range at atmosphere pressure for a number of possible coolants. Note that all the liquid-metal coolants except mercury are solid at 100°F and therefore require system preheating. Note also that the boiling point of a sodium at atmospheric pressure is 1620°F. Because of its excellent heat-transfer characteristics, sodium at 1200°F is capable of generating superheated steam at 1050°F and 2400 psi.

Figure 2 compares attainable heat fluxes for the most common reactor coolants and shows the superb heat-transfer ability of sodium.

Table 1 — COMPARISON OF PROPERTIES OF REACTOR COOLANTS AT 500°F*

Material	Heat-transfer coefficient, Btu/(hr)(sq ft)(°F)	Heat capacity, Btu/(cu ft)(°F)	Pumping power for equivalent volume flow	Operating pressure, lb/in. ²
Water	6,030	56.5	1.0	1,500
Air	342	0.35		500
Helium	289	0.24		500
Sodium	15,200	17.4	1.3	14.7
Lithium	14,100	31.3	0.84	14.7
Bismuth-lead† eutectic	7,150	22.6	13.6	14.7

*From USAEC Report ORNL-2538, Aug. 29, 1958.

†55.5% bismuth-44.5% lead

Figure 3 shows the density and thermal-neutron-absorption cross section for a number of liquid-metal coolants. In each case these properties for sodium are quite satisfactory.

Table 2 compares costs of various liquid-metal coolants, based on information acquired to March 1961.

Sodium exhibits excellent compatibility with most metals and ceramics. The corrosion rates of steels, zirconium, uranium, and uranium alloys are extremely low in sodium below 1200°F. Corrosion rates of austenitic steels fall in the range 0.00 to 0.07 mg/cm²/month in sodium containing oxygen up to 150 ppm and at temperatures to 1300°F. This corrosion rate is equivalent to a maximum of 0.00004 in./year. From a corrosion standpoint the low-chrome ferritic steels are almost as good as the austenitic steels up to 1200°F. The corrosion rate depends primarily on the oxide concentration in the sodium, which can be easily controlled.

The use of sodium offers unique advantages from the safety standpoint. Because the reactor heat-removal system contains chemically compatible materials, no source of stored chemical energy is present, and it is possible to design the reactor without pressure-vessel containment. It has been demonstrated that a sodium coolant system retains most of the fission products released from a fuel-element rupture. This unique characteristic of sodium contributes further to the safety of sodium-cooled reactors. Disadvantages of sodium-coolant systems include the chemical reactivity of sodium with water, the need for system preheating, and inert-gas blankets.

The AEC interest in liquid-metal-cooled reactors goes back to 1945. During that year, both the Argonne National Laboratory (ANL), then called the Argonne Forest Laboratory, and the Los Alamos Scientific Laboratory (LASL) submitted proposals to the AEC's military predecessor, the Manhattan Engineer District, to design and construct such reactors. These proposals resulted in the construction and successful operation of the Clementine reactor by LASL and of the Experimental Breeder Reactor No. 1 by ANL, which I shall describe in a few moments.

In 1948 development of a slightly moderated liquid-metal-cooled reactor for submarine propulsion was undertaken at Schenectady by the Knolls Atomic Power Laboratory (KAPL). This work resulted in the eventual development and successful operation of the first power

Fig. 1—Temperature range for liquid phase of selected reactor coolants. ...., undesirable areas.

Fig. 2—Typical heat fluxes attainable for most common reactor coolants.

Fig. 3—Density and thermal cross section of selected reactor coolants. ...., undesirable areas. From USAEC Report ORNL-2538, Aug. 29, 1958.

Table 2—COSTS OF LIQUID-METAL COOLANTS*

Coolant	Cost, dollars per pound
Sodium	0.17
Lithium	11.00
Bismuth—lead eutectic (55.5%—44.5%)	1.30
NaK (56%—44%)	0.60
NaK (22%—78%)	0.80
Rubidium	390.00
Potassium	3.66
Mercury	2.86

*From Leon Green, Reactor Coolant Properties Table, Battelle Memorial Institute, Mar. 10, 1961.

plant of the Seawolf. The development of liquid-cooled-reactor systems and components under the KAPL project was closely paralleled by a program for development of a sodium-cooled graphite-moderated reactor at the Atomics International Division, North American Aviation, Inc. (AI). This program was originated in 1949; its objective was to develop a reactor that would produce plutonium at a minimum cost with electric power as a by-product. Later, as emphasis on reactor systems in the United States shifted from dual-purpose plutonium producers to power-only reactors, the AI program was realigned toward development of a high-efficiency power-production plant.

Thus development of liquid-cooled-reactor technology has been underway in this country for 16 years. Reactors operating in the thermal, intermediate, and fast regions of the neutron-energy spectrum have been built and successfully operated. I shall now briefly describe the reactors that have evolved as the program has progressed.

Clementine, the first cooled fast reactor was built at LASL. It was designed and built primarily to provide a high-intensity source of fast neutrons. In addition, it permitted study of the control of fast reactors, study of breeding possibilities, and limited study of power-production principles. The reactor was fueled with plutonium metal that was nickel coated and clad in mild steel. At a power level of 20 kw, the mercury coolant entered the core at 86°F and left at 212°F. Heat generated in the reactor was dumped by means of a mercury-to-water heat exchanger.

Rupture of a fuel element and subsequent contamination of the mercury with plutonium caused the reactor to be shut down and later dismantled in December of 1952 after approximately six years of operation. It accomplished the research and experimental objectives for which it had been built.

The second cooled fast reactor and the first nuclear reactor to produce electric power was the Experimental Breeder Reactor No. 1, which was designed and built by the ANL at the National Reactor Testing Station (NRTS). This reactor was based upon a design conceived by Enrico Fermi and Walter Zinn. The AEC approved EBR-1 as a construction project in November 1947. The reactor went critical in August 1951, and electricity was first generated in December 1951. The principal objectives sought through construction and operation of the reactor were testing of the possibility of breeding and confirmation of the hypothesis that a reactor could indeed operate in the high-energy-neutron range, using a high-temperature NaK heat-removal system.

Operation of the reactor experiment has been quite successful. Much knowledge about fast reactor kinetics has been obtained; and, during the entire period of operation, no significant component difficulties arose in the NaK system. The cause of the instability leading to the meltdown of the second core has been identified. Transfer-function measurements on the rigid third core indicate that the relatively nonrigid fuel elements of the second core bowed inward during positive temperature transients to provide a prompt temperature coefficient of reactivity. A fourth core of plutonium-1 $\frac{1}{4}$ % aluminum alloy is being fabricated at the present time.

The Experimental Breeder Reactor No. 2, designed and to be operated by ANL is being built at NRTS under the Civilian Power Reactor Development five-year program. This second-generation fast reactor rated at 20 Mw(e) is sodium cooled and will be fueled initially with U²³⁵. Power generated by EBR-2 will be fed into the NRTS grid.

The purpose of the experiments is to provide experience that can be extrapolated confidently to practical central-station reactors. The program objective is to develop and demonstrate principles of design and operation that can be used to achieve economic power production by full-scale fast breeder reactors through (1) establishing an economic fuel cycle, (2) ensuring safety of the reactor without sacrificing its economic potential, and (3) improving and simplifying liquid-sodium technology.

The Enrico Fermi Atomic Power Plant, near Monroe, Mich., is now almost completed. It is a privately financed fast breeder sodium-cooled-reactor plant which is capable of generating 100-Mw(net) electrical power and which will be the world's largest breeder reactor.

Three separate organizations and the AEC are participating in the project. Atomic Power Development Associates is responsible for research, development, and design of the nuclear portion of the atomic plant. The Power Reactor Development Company is financing and building

the reactor facility, which it will own and operate. The third organization, the Detroit Edison Company, is building the conventional turbogenerator section of the plant and will operate it with steam purchased from the Power Reactor Development Company. The electrical power generated will be fed into the Detroit Edison system, which serves southeastern Michigan.

The Fermi reactor was initiated under the first-round invitation of the AEC Power Demonstration Reactor Program. The AEC assisted the project through research and development effort in the national laboratories.

Although sodium-cooled fast breeder reactors of the heterogeneous type, represented by EBR-1, EBR-2, and the Enrico Fermi Atomic Power Plant, are among the most promising for eventual commercially feasible production of electric power, the fact that they use solid fuels is an economic drawback.

In 1949 LASL discovered several low-melting eutectics between plutonium and other metals. Subsequently, a fast reactor concept was developed, based on the use of a molten plutonium-iron eutectic fuel, contained in tantalum capsules and cooled by sodium. The metallurgical incompatibility of molten plutonium and its alloys presented a major developmental problem. Extensive investigations were undertaken, which lead to the eventual use of tantalum for containment of the molten fuel.

LAMPRE is an experimental reactor rated at 1 Mw(t) and is located at LASL. Heat is transferred from the core to a primary sodium-to-air heat exchanger. This reactor is being used to investigate the feasibility of molten-plutonium-alloy fuels, the satisfactory containment of such fuels, fission-gas disengagement from the molten fuel, and the suitability of the type of reactor for future electrical-power generation.

Although it is not a part of the AEC civilian power-reactor program, an additional sodium-cooled concept is presently being investigated by AI on behalf of the Southwest Atomic Power Associates, which was formed by a group of southern and southwestern utility companies in 1957.

The feasibility of a 360 Mw(e) sodium-cooled-reactor system operating on a U^{233} -thorium fuel cycle is being studied. The system holds promise of breeding more U^{233} from thorium than is consumed. Although they are not directly involved in this project the AEC is observing this private development with keen interest.

The Sodium Reactor Experiment (SRE) was built for the AEC by AI under an early program for the development of civilian nuclear power. It is a sodium-cooled graphite-moderated thermal reactor, designed to produce 20 Mw(t). An adjacent steam generator and 6 Mw turbogenerator owned and operated by Southern California Edison Company utilizes the heat from SRE to produce electricity, which is fed to the Southern California Edison grid. The SRE generated its first electrical power in the summer of 1957. At full design power the primary coolant leaves the reactor core at 950°F. Superheated steam is produced at 825°F and 600 psig.

The SRE has furnished the technology on which the Hallam Reactor is based. In addition, it is contributing significant information on sodium technology that is generally applicable to sodium-cooled-reactor systems, including heat transfer, piping, valves, pumps, and sodium-service systems.

The Hallam Nuclear Power Facility is also being built under the first-round invitation of the AEC Power Demonstration Reactor Program; in this instance, as a joint project of the AEC and CPPD.

In this case the AEC has provided all necessary research and development and is constructing and will own the nuclear facilities. Consumers Public Power District contributed an amount approximately equal to the capital cost of equivalent-thermal-capacity steam-raising equipment to the cost of the nuclear facilities, provided a site and site improvements, designed and constructed the conventional power station adjacent to the nuclear facilities, and will operate the nuclear facilities for the AEC after the nuclear designer, AI, has completed test operations. This sodium-cooled graphite-moderated reactor plant is rated at 75 Mw(e) and 254 Mw(t).

The primary purpose of the Hallam Nuclear Reactor is to provide central-station operating experience with a sodium-graphite reactor.

We who are attending this symposium are interested in the present status and the future prospects of liquid-metal-cooled nuclear reactors. Those persons participating in the program

are dedicated to developing sodium-cooled power reactors to the point that they will produce power economically. You are about to hear descriptions of the existing technology and the problems still to be solved in order to attain this goal.

As I mentioned earlier, at the present state of the art, sodium has many advantages over other coolants for high-temperature heat removal. The limits of such high temperatures, as you will observe, are related to fuel burn-up, reactor-system design, and turbogenerator design.

Experimental Breeder Reactor No. 2 and Transient Test Reactor Facility

L. J. KOCH

Argonne National Laboratory

There are four subjects to be covered in the first part of this session. One hour has been allocated, of which I have half an hour. I shall devote approximately 20 min to the status report on the EBR-2 and 10 min to the status report on TREAT.

Dr. Wensch has already given a general description of the EBR-2 and has mentioned that it is under construction in Idaho. In just a moment I shall show you the status of the plant in several pictures. I think the significant point that Dr. Wensch made is that some of the features of the plant represent a bold or radical departure in design that hopefully advances technology, particularly for the fast sodium-cooled-reactor systems.

I should also like to show that we have exploited at least a few of sodium's favorable characteristics as a coolant, and I think the most significant exploitation has been with respect to the cost of sodium.

As you know, the price of sodium is only about \$0.17 per pound, and we are using about 80,000 gal of sodium in the EBR-2 primary system for reasons that I shall describe later.

One of the rather unique characteristics of fast reactors is that there is very little heat capacity in the reactor system itself. As a result of this characteristic, it is profitable to utilize a large mass of sodium to provide the heat capacity that the reactor itself does not possess. Thus the problem is one of ensuring the flow of this sodium through the reactor with the utmost reliability.

I shall review the EBR-2 very briefly for those of you who are either not familiar with it or who have not given it any attention for some time in order to provide a brief orientation. I should also like to discuss the status and some of the salient features of the plant.

Figure 1 is an artist's conception that shows the reactor, the primary system, and the containment vessel. The reactor is within the large tank, which we call the primary tank. Most of my discussion will relate to this part of the plant.

The heat is exchanged from the primary, or radioactive sodium system, to the secondary, or nonradioactive sodium system, and the sodium from the secondary system flows to the sodium boiler plant in which the heat in the sodium is transferred to water to form steam. The primary average mean mixed temperature of the sodium leaving the reactor is 900°F. The maximum temperature of the sodium leaving the secondary is 880°F. Steam is generated at 850°F and 1250 psig.

The steam is then transmitted to the power plant, which is essentially comparable to a conventional power plant of this size. It provides all the normal services for the entire plant,

and the centralized control room for the operation the reactor and power system is located in it.

Adjacent to the plant is one of the more unique features of the EBR-2, the fuel-cycle facility in which the fuel from the reactor will be processed, refabricated, and returned to the reactor.

I shall not be able to discuss any of the features of the fuel-cycle facility in the short time I have. However, Dr. Lawroski, at a future session, will describe much of the development work associated with this plant.

This fuel-cycle facility is a cooperative venture in that the fuel cycle selected and the reactor design are very intimately interrelated. As a result the fuel alloy is established by the pyrometallurgical process selected. The EBR-2 is fueled with a uranium "fissium" alloy that consists of the fission products which are not removed by the process and which remain in the base alloy.

The other buildings are essentially miscellaneous supporting facilities, the most important of which are the laboratories which support the fuel-cycle operations, including chemical analysis, isotopic analysis, etc., and which also provide facilities for water and sodium analyses. The remainder of the service facilities include the power substation, etc.

Figure 2 is an aerial photograph of the site taken last fall, and it shows that the site resembles the artist's conception shown in Fig. 1. At that time the fuel-cycle facility was not entirely enclosed. It is now fully enclosed. The status of EBR-2 as of this moment is as follows. Argonne National Laboratory accepted the reactor plant from the construction contractor May 15, 1961. We are now in the process of making final adjustments, calibrations, check-outs, etc., prior to loading the reactor for a series of dry critical experiments. We shall operate the reactor in two phases, the first being a critical experiment before the introduction of sodium, primarily to measure the sodium coefficient of reactivity, or the contribution of sodium to the reactivity of the reactor. These measurements will be made under isothermal conditions at room temperature and also at elevated temperatures. Later, by subtraction or difference of results with sodium present, we can accurately determine the sodium effect.

Because of the rather unique design of the primary system with all the critical components located inside the tank of sodium, it is also quite important for us to make final measurements on the neutron shielding, instrumentation, sensitivity, etc., so that we can rectify any deficiencies, if they occur, prior to filling the system with sodium, which then would make modification much more difficult.

There is still minor construction work to be done in the power plant, but it is essentially clean-up type work and minor modifications. Construction work of the sodium boiler plant and the fuel-cycle facility is still in progress and should be completed in a few months. We hope to make each part of the plant operational as they are accepted.

Incidentally the TREAT reactor and the TREAT control room also can be seen in the background of Fig. 2. The control building is approximately 0.5 mile away from the EBR-2.

Figure 3 shows the primary system. The primary tank is 26 ft in diameter and about 28 ft deep. This tank contains approximately 80,000 gal of sodium and is hung from the top structure to permit thermal expansion. All services are brought into the reactor through the top structure with no penetrations in the vessel below the top cover. It is suspended in the biological shield cavity.

The primary system consists of the reactor; the primary sodium pumps, of which there are two, with each supplying one-half the coolant flow of 8400 gal/min at full power; and a single intermediate heat exchanger where the heat is transferred to the secondary sodium, which then leaves in pipes through the building.

A second group of components are associated with fuel handling and removal and reactor control. Fuel removal is accomplished by mechanisms located in a pair of rotating plugs. The fuel assemblies are removed from the reactor, transferred while submerged in sodium to provide cooling and stored in a storage rack, which is also submerged in sodium to provide cooling.

Later they are removed from the system to the fuel-cycle facility for processing. The process schedule is established on the basis of a 15-day cooling period in sodium and then direct processing in the Fuel Cycle Facility, with no external storage for additional fission-product decay.

These various features dictated the basic design of the system. No radioactive sodium leaves the primary tank except for a 2-in. sodium line that runs to the cold trap in the sodium purification cell. The only shielding required in the plant is the basic biological shield around the primary tank and the shielding of the sodium purification cell

The reactor assembly shown in Fig. 4 consists of a lower grid assembly, the reactor vessel, and a top cover. The reactor cover must be raised and lowered to permit access to the internals for removal of the subassemblies, and, when it is lowered and locked in place by the three locking devices that clamp the cover to the vessel, it forms the upper coolant plenum for the outlet sodium.

The geometry of the reactor consists of a close-packed assembly of hexagonal fuel and blanket subassemblies, the enriched section or core, and the blanket, which is divided into two zones as shown in Fig. 4. The two zones in the blanket (inner and outer) provide the difference in coolant flow required because of the very large change in power generation or power density in the blanket radially from the center of the core.

The various types of subassemblies are supported in the lower grid and plenum assembly

Figure 5 shows more detail of the lower plenum arrangement into which the subassemblies are inserted. The core and the inner blanket are in the central region, and the outer blanket is in the outer region. The flow is split from the pumps into two systems: the high-pressure system, which flows through the outer blanket region and supplies coolant to the central region, and a low-pressure coolant by-pass system from the pumps, which, in turn, provides coolant to the annular plenum to cool the outer blanket.

There are a number of flow baffles to provide uniform flow distribution, particularly for the coolant in the outer blanket where the flow velocities are low and the pressures are low. A series of baffles are located in this area.

Figure 6 is a photograph of the reactor vessel and lower grid structure during construction. The reactor vessel was fabricated in two parts, the lower grid assembly and then the reactor vessel, which is bolted to it. Also shown is a part of the neutron shielding, which is about 30 in. thick and which consists of both borated and unborated graphite canned in stainless steel. It surrounds the reactor vessel to prevent neutron activation of the components in the primary tank and, in particular, the secondary sodium as it passes through the heat exchanger.

Figure 6 also shows some of the hardware, scaffolding, and structures that were erected inside the primary tank.

Figure 7 is an aerial shot taken from the building crane looking down on the reactor floor. The outer circle of floor plates essentially defines the diameter of the primary tank, which is hung immediately below this structure. The reactor can be seen at the bottom of the primary tank about 35 ft below the floor. The two pump drive motors for the primary pumps and a variety of nozzles through which the various components are brought into the reactor can also be seen, including the central opening in which the two rotating plugs are located. All the major components were installed in the system before these two plugs were installed because the opening for these plugs was the primary equipment access to the tank. This photograph was taken early this year.

Figure 8 is a photograph of the reactor cover, which is the third component of the reactor assembly. This is a view of the underside of it and shows the extensive thermal baffling required. During operation the underside of this cover (in the plenum that is formed when it is closed) is in contact with 900°F sodium while the entire unit is submerged in 700°F sodium.

Figure 9 is a top view of the cover. The two points at which the columns are attached for raising and lowering the cover are shown. The bearing for the large rotating plug can also be seen.

(Text continues on page 20)

Fig. 1—Artist's conception of EBR-2 facility.

Fig. 2—Aerial photograph of the EBR-2 site taken from the northwest quadrant.

Fig. 3—Primary system of EBR-2 reactor.

Fig. 4—EBR-2 reactor assembly.

Fig. 5—Pictorial of reactor-vessel grid assembly.

Fig. 6 — Primary tank of the reactor building, showing the inner shell of the reactor vessel in place.

Fig. 7—Primary tank on the operating floor of the reactor building, showing the reactor vessel and grid in place.

Fig. 8—Reactor cover.

Fig. 9—Top view of cover.

Figure 10 is another photograph of the rotating-plug area with the 12 control-rod drives that operate through the small rotating plug in place. The 12 drives are located on approximately an 18-in. diameter and are very tightly clustered. The cover lifting shafts can also be seen.

Figure 11 is a photograph taken somewhat later. Again the control-rod drives located on the plug and, in addition, the support bracket for the drives for the cover lifting mechanism just being positioned on the plug and surrounding the control-rod drive can be seen.

Figure 12 is a simple schematic of the flow system showing the coolant entering the primary pump from the bulk sodium in the tank, the split flow occurring after the coolant passes the pump through an orifice and an adjustable regulating valve for varying the flow between the high-pressure system and the outer blanket.

Figure 13 shows a drawing of the heat exchanger. The flow is through the shell side of the heat exchanger and out. One of the unique features of this design is that the bundle is completely removable, having its own shield plug and secondary sodium lines arranged without any piping connections in the tank.

Figure 14 is a photograph of the operating floor taken May 15, following acceptance of the plant. Subsequent to this the floor plates were removed to provide access for final checking, calibration, installation of instrumentation, etc. This photograph is representative of the appearance of the plant for operation and it presents a reasonably neat and clean arrangement with all the services carried in the subfloor below the floor plates. The one nozzle shown in this figure which is still open is the heat-exchanger shell. Also shown is the ladder for access into the tank. In addition, the control rods, the cover lifting mechanism, the gripper and hold-down unit for fuel removal, and the drive for the storage rack in which the fuel is stored for 15 days prior to processing can be seen.

Figure 15 is a photograph taken from the other side of the building. The pump drive motor, the storage rack drive, and the manual drive for the transfer arm, which transfers the subassemblies between the reactor and the storage basket, can be seen.

I shall conclude my discussion of the EBR-2 in order to maintain the schedule, and I shall say just a few words about TREAT.

TREAT is not a sodium-cooled reactor; therefore, in a sense, it does not belong in this program on sodium-cooled reactors, except for a very important feature: that its primary objective is to test fast reactor fuel elements under transient conditions and to determine their characteristics. In this respect it is very definitely a part of the sodium-cooled-reactor program. It has developed that the TREAT is very useful for experiments other than those concerned with fast reactor safety. However, I shall confine my remarks primarily to the fast reactor safety aspect of the program.

The reactor has been in operation for some time and has been thoroughly described so that I shall only attempt to review the status of some of the experimental work being performed using TREAT.

As shown in Fig. 16, this reactor is essentially a cube of uranium-graphite fuel, homogeneously mixed and contained in 4-in. square fuel cans arrayed in a close-packed lattice. This arrangement provides considerable versatility and access into the reactor. The individual 4-in. square subassemblies can be removed from the reactor individually or in groups, permitting rather flexible insertion of vertical experiments. Access is provided horizontally into the reactor by a series of ports. With proper location of subassemblies, viewing ports into the central region of the reactor can be provided. Also there is a considerable experimental area below the top cover in which loops, piping, and instrumentation can be brought out of the reactor. It is a versatile and accessible experimental facility.

The control rods are located below the reactor in a subpile room and these are high-performance and high-velocity type units to permit the wide variety of power excursions to be run. The main attribute of the reactor is in its ability to withstand very fast excursions, and its self-limiting power characteristics effected by temperature. The very rapid rate of rise of temperature in the entire assembly results from uniform dispersion of uranium in the graphite and that these fuel components are in intimate contact with each other.

(Text continues on page 27.)

Fig. 10—Photograph of the rotating-plug area.

Fig. 11—Photograph of the rotating-plug area taken later than the one shown as Fig. 10.

Fig. 12—Reactor inlet and outlet piping system.

Fig. 13—Sodium-to-sodium heat exchanger.

Fig. 14—Operating floor.

Fig. 15—Photograph of operating floor taken from the side opposite to the one shown in Fig. 14.

Fig. 16—Reactor perspective (TREAT).

Fig. 17—Dry opaque single-element capsule.

The most common type of experiment run in the reactor to date has been static capsule tests performed inside a dummy subassembly of the normal TREAT configuration (4 in. square). A central region is provided in which a static capsule can be located with an experimental fuel element located in the capsule. Figure 17 is a drawing of an EBR-2 type fuel element located in a graphite capsule that, in turn, is contained in a zirconium sleeve, a stainless sleeve, and then in the can of the dummy subassembly itself.

These are called "dry capsules" since only an inert-gas atmosphere surrounds the fuel element.

Figure 18 shows the results of a series of very early EBR-2 type fuel-element experimental transients. There was no failure of the clad fuel element. The EBR-2 fuel elements are approximately $\frac{3}{16}$ in. in diameter. The fuel pin is contained in a stainless-steel tube with an 8-mil bond of sodium.

The fuel temperatures are measured by thermocouples attached to the outside surface of the clad. These samples were 530, 945, 965, and 1000°C. It is interesting to note that although there was no external failure to the fuel element, there was an extensive amount of eutectic alloy formed between the uranium metal and the stainless-steel clad.

Figure 19 shows two more EBR-2 type fuel elements that were tested to destruction. The significant factor here is that the temperatures were very little higher than for the previous tests. These temperatures were at 1000 and 1030°C (as measured with some inaccuracy). The fuel element has failed destructively, and there has been violent expulsion of fuel and sodium. This phenomenon has later been demonstrated to be the result of sodium vaporization with attendant internal pressure causing ejection of the molten fuel. The uranium and cladding were expelled and physically ejected against the graphite capsules in which they were contained.

Figure 20 is a similar test run on a Fermi type fuel element. This element also is a small-diameter pin, but it differs in that it is zirconium clad and metallurgically bonded directly to the uranium metal. There is no sodium bond and no fluid to vaporize and form a pressure. Zirconium does not interact with molten uranium; therefore, no eutectic formation occurs. The two samples shown in Fig. 20 reached a temperature of 1100°C, which was about 100°C higher than the temperatures reached in the previous failures. In this figure one can see that uranium has left one sample; the other fuel element is entirely intact. The molten uranium has seeped out through long cracks. Such an occurrence is found quite consistently on this type of fuel element. This element has only a 4-mil zirconium jacket; and, under very fast excursions, the jacket will develop small cracks due to the thermal stresses that are generated. It can be seen that a small amount of uranium in its molten stage has left the cladding through the small cracks. Actually a further examination revealed that about 1 in. of the fuel at the top had run down, leaving about a 1-in. void at the top. This is a good indication, of course, that the uranium was actually molten inside the fuel element and penetrated only at the cracks that existed.

In Fig. 21 Fermi type elements taken to destructive excursions are shown. The temperatures are not known because the thermocouples broke off the cladding during the excursion. However, the integrated reactor power during the experiments was more than 50% greater than was used in the previous experiment. In this figure one can see failures of gross destruction by molten dissolution of the cladding and also puddles of uranium formed inside these catchers inside the capsule. The results indicate more of a static type failure than the ejection type failures observed in the previous figures.

I should like to say a few words about some of the work either just beginning or in progress or planned for the near future. Figure 22 is a drawing of a transparent capsule for viewing the types of events that we have just shown. The previous experiments described are "before" and "after" types. This type of capsule is inserted in the reactor and oriented with a horizontal slot through the reactor. High-speed motion pictures can then be taken through a series of prisms and mirrors.

Two experiments have been conducted to date with high-speed motion-picture photography. These were both on EBR-2 type fuel elements and the initiation of failure can be seen

Fig. 18—Results of early EBR-2 type fuel-element experimental transients.

Fig. 19—Two EBR-2 type fuel elements tested to destruction.

Fig. 20—Test run on a Fermi type fuel element.

Fig. 21—Fermi type elements taken to destructive excursions.

Fig. 22 — Transparent single-element meltdown assembly.

and timed. Improvement of the experimental technique, lighting, etc., are still required to improve resolution. At present only the ejection of sodium vapor as the fuel element fails can be seen in these early experiments. This vapor masks any fuel movement. It is believed that with improved lighting, reflection, absorption, etc., the resolution of the photographic technique can be improved.

Figure 23 shows a capsule for an in-pile stagnant sodium test; it is very similar to the one previously discussed, except that the fuel element is contained in the lower section and is immersed in a static sodium bath. The heat-removal characteristics during the excursion are different.

Only a few of these tests have been conducted to date, but the generalization can be made that if the sodium remains, the failure characteristics are quite different, whereas, if the sodium is vaporized or ejected, then failure characteristics are very similar to the early dry-capsule experiments.

One interesting observation was made in one of these experiments. The fuel was molten inside the stainless-steel capsule, but the stainless-steel capsule was not penetrated because the heat was being removed from it fast enough by the sodium to prevent eutectic formation, and the capsule did not fail.

Figure 24 represents the next planned experimental facility, a single-pin flowing-sodium capsule again contained in a square TREAT type dummy subassembly. The subassembly contains the entire loop. A small E.M. pump will circulate the sodium around a single fuel element in a closed recirculation system. Four such loops are now under construction. One is about ready for out-of-pile testing before insertion in the TREAT reactor.

Figure 25 describes the most ambitious experimental facility presently planned for TREAT. This is an in-pile sodium loop capable of accepting a full EBR-2 subassembly. It has a pumping capability of 200 gal/min of sodium through the system. It includes pumps, tanks, heat-exchange equipment, etc. This loop is presently being designed. Some of the major components, including the pumps, rectifiers, some of the quick-opening valves, and instrumentation have been acquired.

I shall conclude with the observation that TREAT is a far more versatile facility than indicated by the very few examples of experiments that I have described. It has found considerable usefulness in other programs, including work being performed on water-metal reactions as a part of the reactor-safety program.

Fig. 23—Single-element stagnant-sodium capsule.

Fig. 24—TREAT single-element-package sodium loop.

Fig. 25—TREAT in-pile sodium loop capable of accepting a full EBR-2 subassembly.

Experimental Breeder Reactor No. 1

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The EBR-1 was designed in the period 1948 to 1950, roughly—I am talking about the specific reactor, EBR-1, not the developmental work that went before it. It was designed (1) to prove the concept of breeding by actual measurement (by making measurements after radiation of fuel by chemically reprocessing it and then arriving at values) and (2) to prove out the concept of cooling a reactor with liquid metal and utilizing the heat in the production of steam.

The reactor was built in 1951, went critical that fall, and produced the first useful power in December of 1951. The first use of the electricity produced provided power for its own facility. We cut off the outside power and continued to operate with electricity supplied by this facility. This is the first time, as far as we know, that this had been done with nuclear energy.

The reactor, after some time of operation, proved that the system could operate stably. It had very few faults. In fact, we are operating today with a history of literally no failures, except for one which occurred shortly after start-up. That failure was a leak in a heat exchanger between the primary and secondary liquid-metal system. The leak was traced to a faulty weld, which was subsequently repaired.

The result of measurements of the breeding gain indicated that the reactor just about held its own. In fact, the numbers were 1.00 plus or minus 0.01. In making these measurements, we found out how to improve this reactor and increase the breeding gain to numbers like 1.2 to 1.3. In fact, by making measurements with other materials such as plutonium samples, we found that we could breed with ratios as high as 1.5 to 1.6.

The first core was operated for some three million kilowatt hours. It behaved very well except for a gradual loss of reactivity, which was attributed to damage of the fuel elements. This damage was the result of radiation that caused the slugs to lengthen and decrease in diameter. It resulted in spite of the fact that the fuel elements had been proof-tested ahead of time by thermocycling. We knew that normally uranium is a very poor metal for this purpose, and it was expected to grow and deform. However, the EBR-I fuel was heat-treated to minimize this effect under thermal cycling. Irradiation in the reactor caused it to deform nevertheless. Therefore a second core was designed which had 2 per cent zirconium-alloy fuel. This composition proved to be far more stable. In the design of the second core, we eliminated ribs, which were a part of the original fuel-element design to simplify removal from the reactor. We then found we had a reactor that was very stable except when we tried to operate it above design power, which was 1 Mw.

We ran at 1200 kw very nicely, but when we tried to run a little above that or to throttle the coolant flow slightly, the reactor would become unstable and tend to oscillate. It exhibited what we call a prompt positive power coefficient. If we did anything such as suddenly cutting the flow by a small amount, we would find an increase of power that was in a direction opposite

to what would be expected because the average temperature went up, and it had an isothermal temperature coefficient that was negative.

In attempting to learn more about this, we performed a number of experiments; and in the final one accidentally melted the core. At the time it did not seem much of an incident. No evidence of any damage was immediately noticeable, but subsequent inspection raised the question as to the safety of fast reactors. We were apparently only a very small margin away from a nuclear incident of possibly major proportions. I suppose it is fate, but we were scheduled to shut the reactor down permanently 30 days after the incident; but, because of the incident, it became obvious that an additional program was necessary to prove that fast reactors are not inherently unstable and unsafe.

It was decided that the reactor would be rebuilt and be called Mark III and that we would try to make it as stable as possible to indicate and prove that fast reactors could be built safely. Such a reactor was designed and built, and it operated very stably. In one manner of investigating reactors, a process of oscillating them by introducing a small amount of reactivity and then measuring the effect on the reactor is used. I am sure many of you are familiar with this. We found by extrapolation that the reactor was stable at many times its designed power.

We then proceeded to modify the reactor to make it simulate more nearly the Mark II loading, which suffered from a flexibility of fuel rods. We found that although the reactor was not unstable even in this case, stability was lessened. I shall briefly go on and tell you the status of the reactor today and then go back and describe the reactor.

This reactor has been operated for some three million kilowatt hours, even though the fuel elements were designed for a short running time with maximum rigidity and stability. There was no clearance for fuel-element growth; we did not expect high burn-up, but the burn-up on this core has been as much as that we had on the first core. We presently operate it at high power only for continuous short runs to irradiate various materials, which have included uranium-plutonium carbide samples.

We have irradiated prototype elements for the Mark IV core, which is next on the reactor program and have used the reactor as a training device for our EBR-2 operators.

Currently it is in semistandby status. We use it for special experiments and special irradiations; and its crew and staff are assisting in EBR-2 training and will assist in EBR-2 start-up.

The reactor, shown in Fig. 1, is very small and has been called a football, and it is about the cross-sectional size of one. It is surrounded by a natural-uranium reflector blanket and a concrete shield. It is cooled by a primary system of sodium-potassium. We use the eutectic alloy, which of course, becomes highly radioactive, and therefore is enclosed in concrete cells.

The secondary system, or we might call it an intermediate system, also contains eutectic sodium-potassium.

The third system is water and, of course, steam. The primary system has two pumps: an electromagnetic one and a mechanical one. Only the electromagnetic one is shown in this figure. Either one can carry the full load, so that we get operating experience with both kinds. The secondary has only a mechanical pump.

We run the mechanical pump in the secondary system and the electromagnetic pump in the primary system for operating experience. We have had no difficulty with either. I might add that 12 years ago we could not buy such pumps; in fact, we had to build these pumps ourselves. Today, however, we could buy them.

The third system consists of the water heater, steam generator, and superheater. We generate 400 lb of steam at 550°F. Obviously this is a very low performance system, but we were not interested in efficiency or inviting trouble. The steam generator was built very conservatively. Its heat-exchanger tube consists of a triple wall of nickel, copper, and nickel bonded together and has given us absolutely no trouble.

Fig. 1—EBR-1.

The Mark III core is shown in Fig. 2 to give you an idea of the reactor's appearance. The core of the reactor is inside a tank. It is actually a double-wall tank, about 20 in. across, and it is surrounded by an outer blanket we call the cup containing natural uranium canned in stainless steel.

Fig. 2 — Mark III core and reactor.

Fuel elements are contained in hexagonal boxes in Mark III. The core has a center box with six surrounding it, and the inner blanket consists of twelve around the core. This is where Mark III differs from Mark I and Mark II.

Each box has room for 37 elements. We use 36 elements and a center special rod we call the tightening rod. This rod forces the fuel elements outward against the box, tightens them up,

and gets rid of the dimensional instability effect (flexibility effect) that was observed in Mark II.

The containers themselves, which make up the core, are clamped in an array by outer clamps.

Figure 3 shows the cross section of the core and inner blanket. One must remember that the outer air-cooled blanket is outside. In this figure a typical cross section showing 36 elements and a tightening rod can be seen.

The Mark III loading was designed so that the coolant could flow in series, as it did in the Mark I and II, down through the outer blanket into a lower plenum up through the core and out or parallel down through downcomers and up and in parallel through the core and blanket. There was a feeling that the coolant flowing in series led to instability because of temperature feedbacks.

We had a throttle arrangement so that we could vary the flow between the two and arrive at a desirable condition where the outlet temperature was the same in each region.

Figure 3 also shows other things, such as the source and the oscillating rod, which were made part of the experiment.

Figure 4 shows a smaller view of the subassembly. In it we can see the fuel elements. These fuel elements are approximately 0.4 in. in diameter and are made of highly enriched uranium with 2 per cent zirconium. The zirconium jacketing was clad by a coextrusion process that gave absolute contact between fuel and jacket and permitted no room for expansion within.

The Mark IV core will contain plutonium fuel. It is currently under construction. The reactor will look exactly like the Mark III, except that the seven inner elements will contain 60 fuel elements instead of 36. This arrangement is necessary because of the poor thermal conductivity characteristics of the plutonium alloy we propose to use. This alloy consists of plutonium with 1.25 per cent aluminum. This is a delta stabilized alloy with reasonably good characteristics. The small size of the core rules out the use of fuels having a high percentage of alloying metals as are proposed for larger reactors.

The technology developed at present would not allow us to coextrude—or, I should say, no one has facilities for coextruding plutonium—so that we have to be satisfied with an earlier type of design, something similar to Mark I. This design consists of a zirconium-alloy outer jacket containing plutonium pins. The jackets will have ribs on them to provide the stability and rigidity that we had in the Mark III core.

The plutonium core is not a prototype. We do not expect anyone to build future large reactors with this alloy, but it will enable us to make reading gain measurements comparable to the first ones. We shall be able to make measurements of a kinetic nature and determine whether the stability of this reactor with different material is different from the Mark III.

We intend to put the reactor in standby status following approximately a 1-year operation, and unless there is a need for the reactor, we shall not operate it again. We expect to concentrate on EBR-2 experiments. However, should it be desirable, we may possibly operate the reactor with ceramic elements such as carbide oxides, but, again, in this use the disadvantage of this reactor is its small size and its low power. We feel that EBR-2 will be a much more useful tool for radiations of that sort.

In regard to the oscillating experiments, they occupied some two million kilowatt hours. After that length of time, the fuel had undergone considerable change dimensionally. We saw different effects than we did initially. After that time we have operated for perhaps another million kilowatt hours, but we have not performed any major oscillating experiments during that time.

A question was asked about the result of the irradiation on the oscillating experiment. The irradiation caused the fuel elements to swell. In this case they did not grow in length; they shortened and increased in diameter and gave us a considerably more rigid arrangement. Measurements indicated the reactor was less susceptible to motions. The reactor had a very prompt negative coefficient, and this was even more evident as the reactor tightened.

Fig. 3—Cross section of Mark III core and inner blanket.

Fig. 4—View of subassembly.

I might put it another way. We found that at low powers the power coefficient was not linear. It was lower at low powers than it was at higher powers. To us this fact indicated that there was an inherent looseness that obviously is due to mechanical or engineering factor. One cannot have an absolutely tight core, but it tightened up as the power and temperature of the core increased. With continued operations the tightening effect also took place because of fuel swelling.

There has been a number of reports published, which can be used as references, on the results of the Mark III kinetics experiments. They have been published very recently.

I might add that the Mark III core was used to irradiate plutonium and uranium carbides and prototype elements. The postmortem examination looks pretty good. This irradiation resulted in no obvious damage to the sample containers or jackets. No difficulty was experienced in this way.

Fast Power Reactor Experiments in ZPR-3 with Metallic and Ceramic Fuels

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The Argonne National Laboratory fast power-reactor critical facility, the Zero Power Reactor No. 3 (ZPR-3) is a flexible facility for the study of the important reactor-physics characteristics of fast reactor assemblies. It is possible to load into the machine any desired configuration of materials and essentially any desired composition of the reactor core, even permitting the detailed shapes of engineering designs to be reproduced in the assembly. Its flexibility is limited only by the size of the machine, which is approximately a 6-ft cube; the load bearing capacity of the tables, which is about 25 tons each; and the inventory of the materials on hand. We have about 575 kg of enriched uranium available for use in the ZPR-3.

The over-all picture of the machine is seen in Fig. 1. This picture was taken during an experiment to measure the distribution of fissions through a core postulated as a result of a partial meltdown of the Enrico Fermi core. This was one of the experiments performed to investigate the safety aspects of the Fermi reactor. You will note a tube running through the center line of the assembly. Fission counters were remotely traversed through this tube after the halves of the machine had been brought together and the reactor was critical at a low power. Note that the materials themselves are loaded into the assembly in drawers that are placed in the square matrix tubes. Control and safety of the reactor are assured by five control safety rods in each half, four acting as safety rods and one as a control rod. The four safety rods must be fully inserted before the machine can be brought together, and the one control rod must be fully withdrawn. Each half is fitted with a neutron source that must be inserted into the machine before the halves can be brought together, thus ensuring proper monitoring of the approach to critical. A typical core composition is illustrated in Fig. 2. This particular drawer is loaded to mock up the Enrico Fermi Reactor. The left-hand drawer, which is loaded into the front of the matrix half, is filled with $\frac{1}{8}$ -in. plates of U^{235} , U^{238} , zirconium, molybdenum, and aluminum to a core half-height of 15 in., simulating the core half-height of the Fermi reactor. The back part of the drawer simulates the transition region between core and blanket and the blanket region. The right-hand drawer, which is loaded into the back of the matrix half, simulates the remainder of the blanket region. One can see the flexibility with which this critical facility can be loaded to simulate many reactor types.

A better idea of the construction of a loaded reactor is shown in Fig. 3. This is a view of one-half of the machine, loaded with a cylindrical core with the axis running vertically. One row of drawers is pulled out of the machine to reveal a highly enriched central cylinder surrounded by a more lightly enriched annular region.

Recently the ZPR-3 has been used to study the characteristics of large dilute fast power reactors fueled with uranium oxide or uranium carbide. The core compositions of these recent assemblies are listed in Table 1, together with the composition of assembly 20, which

was a study of the Fermi core, and assembly 25, an all-uranium assembly consisting of 1 part U^{235} to 10 parts of U^{238} . Assembly 29 is a mockup of a typical uranium oxide fueled fast reactor, where the UO_2 composition was mocked up by loading aluminum oxide plates together with the uranium plates. Stainless steel was used as the structural material in the core and aluminum was used as a mockup of sodium coolant. In assembly 30 the oxide content has been reduced to mock up UO . This assembly and the next assembly were used in a study of the effect

Table 1—CORE COMPOSITIONS OF RECENT ASSEMBLIES

Assembly	Core descriptive title	Critical core dimensions, length $\times$ di- ameter, cm	Core composition, vol.%, density						
			U ²³⁵	U ²³⁸	Aluminum	Stainless steel-304	Oxygen	Sodium	Carbon
			18.75	19.0	2.7	7.85	2.55	0.97	1.43
20*	Fermi	76.4 $\times$ 79.3	6.1	19.0	25.1	14.3†			
25	U ²³⁸ diluent	76.3 $\times$ 85.2	7.1	74.0		9.3			
29	Oxide, UO ₂	71.3 $\times$ 90.0	4.97	9.97	24.4	24.7	14.5		
30	Oxide, UO	62.2 $\times$ 82.8	5.92	9.06	23.4	24.6	7.2		
31	Metal	62.2 $\times$ 90.4	5.81	9.14	23.5	24.5			
32	Stainless steel diluent	61.1 $\times$ 52.2	9.3	0.7		81.0			
33	Stainless steel + Na diluent	61.1 $\times$ 53.4	9.3	0.7		63.6		15.8	
34	Carbide	86.5 $\times$ 91.9	4.67	10.3	25.5	24.6			10.6

* Reduced density blanket. All other assemblies had 83.3 per cent U^{238} and 7.3 per cent stainless-steel blanket. Blanket at least 12 in. thick.

† Also contains 5 per cent molybdenum and 4.3 per cent zirconium.

of oxygen on the system. As will be seen later, calculations do not predict the correct critical mass for these assemblies. Assembly 32 was built in an attempt to study the effect of stainless steel on the critical mass of these systems. In an assembly loaded with enriched uranium and stainless steel only, it was hoped that any error in the stainless-steel cross section could be emphasized. Assembly 33 was a modification of this assembly to include as much sodium as was available. We would have liked to have built an assembly loaded only with enriched uranium and sodium to study the effects of possible errors in sodium cross section on the critical mass of the machine. Assembly 34, the most recent assembly built in the ZPR-3, is a typical uranium carbide fueled power reactor with stainless steel as a structural material and aluminum used to mock up sodium coolant. Table 2 gives the critical masses obtained for these assemblies experimentally. For assembly 29, the uranium oxide fueled assembly, the initial loading was calculated to give a critical mass of 332 kg. With the composition originally envisaged, the machine did not go critical. The loading was limited by the available supply of aluminum oxide to provide the oxygen for the system. The estimated critical mass obtained from extrapolating the reciprocal count rate as the machine was built up was 460 kg. The composition of the machine was finally changed by adding more fuel uniformly throughout the core and a critical mass of 421 kg was obtained for this more highly loaded machine. In assembly 30 a mockup of a UO loading gave an experimental critical mass of 395 kg, whereas the calculation gave 288 kg. Assembly 31, an all-metal core, gave a much higher critical mass than calculated. I believe that this illustrates that the trouble is not with the oxygen content of the machine, but with the diluteness of the loading. In assemblies 32 and 33, in which the effects of stainless steel and of sodium were studied, the critical masses were higher than calculated. This was also true for the carbide loading, assembly 23, where an experimental critical mass of 503 kg was obtained compared to the calculated mass of 376 kg.

The calculations mentioned above were made with cross-section set No. 58, an 11-group set. The state of affairs is not quite as bad as it seems, because the people at Atomic Power

Development Associates, using a 16-group set of cross sections patterned after the set of Roach at Los Alamos, have obtained much better, almost phenomenal, results. We are now undertaking similar calculations at Argonne.

However, neither set gives consistently good fission ratios when compared with experiment.

There is still much work to be done in the reactor-physics study of large dilute fast power reactors.

Table 2—CRITICAL MASSES OBTAINED FOR ASSEMBLIES

Assembly	Core descriptive title	Critical mass*	
		Experiments	Calculations
20	Fermi	431	
25	U ²³⁸ diluent	582	779
29 (initial)	Oxide, UO ₂	460†	332
29 (final)	Oxide, UO ₂	421	
30	Oxide, UO	395	288
31	Metal	463	344
32	Stainless steel diluent	227	176
33	Stainless steel + Na diluent	238	187
34	Carbide	503	376

* Calculated critical mass corrected for shape factor and heterogeneity.

† Extrapolated.

Fig. 1—ZPR-3.

Fig. 2—Typical core composition.

Fig. 3—Loaded reactor.

Enrico Fermi Atomic Power Plant

R. W. HARTWELL

Power Reactor Development Company

I wish to promptly acknowledge the fact, as was explained in the introduction, that the Enrico Fermi Atomic Power Plant represents a combined effort, including our Power Reactor Development Company (PRDC); the fine services of Atomic Power Development Associates (APDA), Nuclear Design Group; Commonwealth Associates for the more conventional design work; United Engineers, our constructors; and the very important role that Detroit Edison has played in the design and construction of the turbine portion of the plant.

Construction

Construction of all buildings has been completed. The aerial view shown above shows the progress that has been made on cleanup of the site. Roads and sidewalks have been laid and grading is complete. Landscaping of the grounds is about finished.

In the reactor building, the primary sodium loops are completed and filled. The sodium filling operation was successful with all equipment, valves, liquid-level indicators, and temperature instruments working with complete satisfaction. The sodium pumps have been installed and checked. The overflow pumps have been placed in service and cold-trapping of the

primary sodium has been completed. The ventilation system is in use. Modification work on the graphite shielding continues.

In the fuel and repair building, work is continuing on the steam cleaning system, the access tube to the transfer rotor tank, and the cut-up pool vacuum-cleaning system. The fuel-transfer cask car has been received and is now in place in the fuel and repair building where it is undergoing tests and minor adjustments.

Fig. 1—Plant control room.

Installation of control-room instrumentation (see Fig. 1) is in the final stages. It will be necessary to transfer some of the permanent plant instrumentation to the control room after the single sodium-loop test program has been completed.

In the steam-generator building, all the equipment has been installed and the piping has been completed. About 20,000 gal (approximately two-thirds of the requirements) of the sodium has been placed in the secondary-system storage tanks. Figure 2 shows the steam generators and the secondary sodium pumps as they appear above the operating floor.

The turbine-generator in the Detroit Edison Company section of the plant is completely erected as is shown in Fig. 3. It has been started up and synchronized numerous times to check out all equipment and instrumentation. To conduct the turbine tests, two 250-psi package boilers, each with a capacity of 50,000 lb/hr, have been installed by the Detroit Edison Company. These boilers will be removed after the reactor is in operation.

Fuel

D. E. Makepeace Division, Engelhard Industries, Inc., is the supplier of two complete core loadings and one-half of the blanket requirements. Sylvania-Corning Nuclear Corp. is the supplier of the other half of the blanket requirements. Essentially all the blanket material is in storage at the plant, and the first core loading of 100 subassemblies is about ready for shipment from D. E. Makepeace. Thirty-five subassemblies are completely assembled, and D. E. Makepeace is now producing two per day. All the uranium-molybdenum fuel pins and stainless-steel parts are available in stock.

The problem of the fuel-pin support was resolved through the combined efforts of Westinghouse Electric Corporation and APDA. Seventeen grid structures, which are spaced along the 30-in. pin length, replace the wire spacers that were used in the original design. This assem-

Fig. 2—Steam generator building.

Fig. 3—Turbine-generator.

bly provides ample pin support, is rigid, and produces favorable hydraulic conditions. These pin-support assemblies (birdcages) are being fabricated and delivered to D. E. Makepeace for the assembly of core subassemblies.

Component Test Facility

Atomic Power Development Associates, Inc., has been conducting the nonnuclear test of major reactor components and associated facilities. This test program will be complete on June 1, 1961, and the facilities will then be turned over to PRDC. The components and equipment consist principally of the reactor vessel, rotating plug, surrounding primary shield tank, fuel-handling and -reloading equipment and mechanisms, reactor control devices, and the primary sodium-coolant loop No. 2, which includes the pump, check valve, by-pass throttle valve, and piping. Important performance information has been provided by this test work. The primary sodium pump has been operated in excess of 1500 hr up to operating conditions as high as 130 per cent of rated loop flow. The fuel-handling system has been completely calibrated at 500°F, which is the normal operating temperature. Two dummy fuel elements have been removed from the system, cleaned, and inspected. The filters in these elements had collected very small amounts of material, attesting to the cleanliness of the system.

A number of dummy subassemblies have been moved around within the reactor during the installation of control and safety rods. One of the dummy subassemblies stuck in the process of being removed. By remote maintenance the dummy was removed in a matter of a few days. Inspection indicates that the dummy was not built in accordance with specifications.

During this test program some design modifications were shown to be necessary. These modifications have been made and the equipment now performs satisfactorily.

Preoperational Testing

Preoperational testing in all other sections of the plant is being conducted by the plant operators. This testing of a number of components and systems has been completed. Tests on other components are now under way and test procedures for the remaining components are being prepared. The purpose of these tests is to demonstrate that the equipment as installed meets the design specifications and operates satisfactorily. In addition, the blanket fuel subassemblies are being unpacked and properly stored as soon as they arrive from the fabricator's plant. The fuel gauging fixture has been installed and the gauging of the blanket subassemblies is in progress.

Examples of components and systems that have been tested are as follows: the essential bus; the 50-kva 3-unit motor-generator set; the 250-kw emergency diesel-generator set; the safety system 3-unit motor-generator set; the heating system on the secondary sodium fill lines, service system, and storage tanks; the secondary sodium system for leak tightness; and the humidity and temperature control equipment for the blanket subassembly storage vault. Limited tests have been conducted on the primary and secondary sodium pumps and their pony motors.

Tests are being conducted on the below-floor ventilation system in the containment building, the ventilation system for the cold-trap room and sodium tunnel, and the ventilation system in the fuel and repair building. In addition, tests are in progress on the inert-gas system and various components in the fuel building pertaining to handling, gauging, and cleaning of fuel.

Preoperational test procedures have been prepared for the waste-gas system, high-pressure-air supply system, emergency-control air compressor, secondary sodium pumps, and secondary sodium system. These tests will be conducted as soon as practical.

Construction Permit Proceedings

The U. S. Court of Appeals for the District of Columbia issued an order on June 10, 1960. In their decision they laid down two rules: (1) in issuing a construction permit, the AEC must make the same safety findings as it would in issuing an operating license and (2) Congress intended a finding of compelling reasons for the location of a power reactor near a large population center. A petition for a rehearing *en banc* was filed on June 29, 1960. The court denied a rehearing on July 25, 1960. On July 26 PRDC filed a motion for a stay of 20 days from the date of the court's order and until final disposition of the case by the U. S. Supreme Court. A stay was granted by the court panel on Aug. 8, 1960.

A petition for certiorari was filed with the U. S. Supreme Court on August 12. On November 14, the Supreme Court granted a writ of certiorari in our construction-permit proceedings. They limited argument to two issues: (1) the adequacy of the construction-permit findings and (2) the site-location question. The Supreme Court eliminated the argument as to the standing of the unions as intervenors. A time of 3 hr, 1½ hr for each side, was allotted for oral arguments. Briefs were filed by the Solicitor General for the government and by PRDC on Feb. 13, 1961, and by the unions in March 1961. Oral arguments in the case were heard on Apr. 26 and 27, 1961, and it is expected a decision will be handed down sometime before adjournment of the court in June 1961.*

Fig. 4— Training simulator in relay room.

Plant Organization

All operators and maintenance personnel are now on the job at the site. Many of these men have been in a training status for more than two years. Those to be licensed are completing their final courses, and it would be expected that they will take their license examination within the next few months. The training simulator (see Fig. 4) has been very helpful in the training program. This device tests the operator during simulated failures in the plant.

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*The U. S. Supreme Court decision of June 12, 1961, upheld the Fermi construction permit and the AEC authority to have issued the permit.

Schedule

The Hazards Summary Report will be submitted to the AEC next month; and, assuming a favorable Supreme Court decision regarding the construction permit, an application for an operating license will be filed in July. The plant will be ready to go critical this fall.

Cost

The utility companies' members of this audience will be interested in costs. I therefore tell you that this project, exclusive of the AEC monies that have been spent in our behalf, total \$84 million. These monies have been provided by the APDA companies for research and development, by PRDC for the nuclear portion of the plant, and by Detroit Edison for the turbine-generator portion of the plant.

Status

What are we doing today? I think you can see that the construction is essentially complete. We are in what we refer to as our preoperational test period. Sodium was introduced into the system after the system had been heated last December, and sodium has been circulated by APDA in a special test loop that was installed, wherein we may induce heat through a separately fired boiler. We have just completed a 1000°F isothermal test. This means that the reactor vessel with all three loops, the intermediate heat exchangers, and the three primary pumps have all been subjected to temperatures up to a 1000°F for an extended period of time, 10 days or so, in the process of going up to that temperature and coming down.

The No. 1 primary pump has run approximately 1600 hr. We have had some difficulties during the preoperational tests that I shall identify. Time does not permit my going into these difficulties in detail.

There is a simulated core in the reactor. We refer to it as dummy elements with filters in them. These were installed primarily to test the cleanliness of the system and also to mock up the pressure drop through the core for purposes of the operation of the No. 1 pump.

Two of these dummy subassemblies were removed and inspected and found to be very clean. This was gratifying, and those of you that have followed the sodium-system experience at Dounreay would, I think, be rightfully concerned. We certainly can tell you that our system was clean.

On the other hand, when we got around to installing the second of two safety rods that we wanted to test at 1000°F, we had difficulty removing one of the subassemblies. I think eight in all were moved around by the handling system, and then one stuck. We had to use mechanical equipment to remove it. I think this trouble might be measured in terms of the length of time it took to remove the subassembly. It was removed in about three or four days with the use of some special equipment that was made.

I should also identify a problem that we had in the operation of the offset handling mechanism. Its shielding and primarily the clearances for the shielding, as a function of our evaluation of the Dounreay experience, was found to be improper; therefore, we rebuilt the shielding for the offset handling mechanism.

I think the other important problem that we have encountered in these preoperational tests—and one that is causing us some discomfort right now—is the rebuilding of the graphite shield that surrounds the reactor vessel and is internal to the primary shield tank. The reactor vessel and the piping did not grow precisely in accordance with the calculations, and we therefore found that the clearances we had allowed were insufficient. This made it necessary for us to go in to remove some 68 per cent of the graphite block, which had been meticulously shaped and placed. Now with sodium in the system and the system at temperature, we are in the process of replacing the graphite; and, if you were to visit the site, you would see

discouraging piles of graphite around the reactor building. My resident engineer tells me that this can be cleaned up in two to three months.

During the testing process by the APDA people in the reactor portion of the plant —and I should explain to you that that particular portion of the plant will be turned over to PRDC on June 1, 1961 — the PRDC operating people, now all assembled at the plant, have conducted tests on all the other systems. These include the various gas systems that I have identified, the ventilation system, the electrical system, the feed and steam systems, etc.

Los Alamos Molten Plutonium Reactor Experiment No. 1

R. E. PETERSON

Los Alamos Scientific Laboratory

The Los Alamos Molten Plutonium Reactor Experiment No. 1, LAMPRE-1, is an experiment to investigate some of the problems associated with the use of a molten plutonium alloy as a fuel for a high-specific-power high-temperature sodium-cooled fast breeder reactor. LAMPRE-1 has now been in operation for about two months at powers up to 100 kw, or about 10 per cent of the nominal design rating.

LAMPRE-1 is in itself not a high-specific-power device, it is not a breeder, and it is not a prototype of a commercially useful reactor. Its purposes are given in the following list:

1. To demonstrate, if possible, means of reliably containing a molten-metal fuel alloy in a reactor environment.
2. To provide a fast-neutron-flux high-temperature sodium-cooled test facility for examining the characteristics of a variety of possible molten-metal fuel alloys.
3. To determine the behavior of fission products in such fuels.

It should be clearly recognized that the design, construction, and successful operation of LAMPRE-1 will constitute only a modest first step toward the goal of a practical, molten-plutonium-alloy-fueled fast power reactor and breeder.

In this status report two main aspects of the LAMPRE program will be covered. First, the device will be described very briefly from the "hardware" point of view. Second, the reactor start-up will be reviewed, and some of the associated data will be presented.

Hardware and Design

Figure 1 is a cross-sectional view of the LAMPRE installation. The reactor and the sodium-coolant loop are contained in two adjacent cells. The top of the reactor is accessible from the working area above the reactor cell. Control-rod drives are located in the basement under this cell. The heat exchanger, which is a simple air radiator, removes heat directly from the primary sodium, which cools the reactor core. An air-cooled borated-graphite shield surrounds the reactor vessel.

Figure 2 is a schematic drawing of the sodium-coolant loop; the customary sodium dump tank and cold-trap facilities are provided. Final purification of the coolant to the required standards is effected by zirconium-filled hot traps operating at about 700°C. There are two electromagnetic pumps in parallel. System preheating is provided by electrical resistance heaters. Additional power may be put in by means of a heating transformer whose secondary

Fig. 1—Cross-sectional view of the LAMPRE-1 system.

Fig. 2—LAMPRE-1 coolant loop. Schematic.

winding is part of the sodium loop. Total heat leak at 500°C is about 35 to 50 kw. There is only one valve in the system that is continuously exposed to sodium. The loop has operated successfully for a total time of about eight months, much of this period above 450°C.

Fig. 3—LAMPRE-1 reactor vessel.

Fig. 4—Fuel-capsule charger for LAMPRE-1.

Figure 3 is a photograph of the reactor vessel. It is constructed of 316 stainless steel and is about 12 ft long and $10\frac{1}{2}$ in. in diameter. Inlet and outlet coolant pipes are attached above the level of the core, and the bottom portion of the vessel is of double-wall construction. Coolant flow is downward along the inner surface of the vessel, into a plenum, and then upward through the core. The top end of the vessel is fitted with a 10-in.-diameter sliding vane valve, which permits access to the core components.

Although fuel replacement to accommodate burn-up is not required in the low-power LAMPRE system, it is essential, if the reactor is to fulfill its fuel-testing goal, that the fuel-containing portion of the core components be readily replaceable. A mechanism for extracting or inserting core components is attached to the top of the reactor vessel. This mechanism, or capsule charger, lifts core components out of the reactor vessel and transfers the fuel section out of the charger housing through a gas lock. Core components are completely contained during all phases of transfer operations so that, even if a core component is defective, neither

Fig. 5 — LAMPRE-1 control-element assembly.

Fig. 6 — LAMPRE-1 fuel capsule (tantalum-0.1 wt. % tungsten).

Fig. 7 — LAMPRE-1 fuel-capsule and -handling assembly.

plutonium nor fission-product contamination will be released. A photograph of the capsule charger without its lead shielding is shown in Fig. 4. Motions of the internal parts of the charger mechanism are monitored visually by means of periscopes.

Reactivity control of LAMPRE-1 is effected by means of vertically movable neutron reflectors external to the reactor vessel. The shim, which is the largest of the five control elements, is an annular stainless-steel cylinder (16 in. long and $4\frac{5}{8}$ in. thick) that surrounds the reactor vessel; it may be raised and lowered relative to the reactor core. Four nickel control rods (3.7-in.-diameter cylinders) are used for fine control; these move in tubular guides lo-

cated in notches on the shim. The assembled shim and control-rod guide-tube system is shown in Fig. 5. The reactor vessel was subsequently installed in the center of this assembly. Hydraulic cylinders located in the basement beneath the cell are used to move the control elements. Only the shim can be "scrammed." At the present time one control rod has been removed and an oscillator rod inserted in its place.

The plutonium-iron alloy in the first LAMPRE core is contained in tantalum capsules, each of which holds about 170 g of the fuel alloy. Figure 6 is a photograph of one of these fuel capsules. It is about $\frac{7}{16}$ in. in diameter, has a wall thickness of 0.026 in., and the over-all length, including the tantalum plug at the top and the threaded stainless-steel adapter, is about 9 in. The fuel occupies the lower 6 in. of the capsule; the void space above the fuel is provided to accommodate the gaseous fission products that, it is hoped, will escape from the fuel.

A complete core component consists of a capsule and its handle. Such an assembly is shown in Fig. 7. Its over-all length is about $10\frac{1}{2}$ ft. The upper part of the stainless-steel handle is hexagonal, whereas the section adjacent to the capsule is cylindrical.

Fig. 8 — Bottom locator plate of LAMPRE-1 core.

Each of the core components rests on and is positioned by a locator plate in the reactor vessel (see Fig. 8). The conical tips of the capsules engage the depressions in the locator plate, and the coolant flows through the holes drilled in the locator plate.

Radial restraint at the upper ends of the capsules is provided by the enlarged "spline" sections on the capsule handles. The upper, hexagonal portions of the handles form a close-packed array in the reactor vessel and thus serve as part of the shielding for the core.

The reactor vessel and locator plate will accommodate 199 of these assemblies. Only about 140 capsules are required to make a critical mass, and the remaining 60 handles are fitted with solid stainless-steel dummy capsules that form part of the core reflector.

Selected capsule handles are equipped with thermocouples for measuring the temperature of the coolant at the inlet and outlet to the core. The capsule-charger mechanism can connect and disconnect the thermocouple lead wires should it be desired to withdraw or replace one of these core components.

Table 1 — LAMPRE-1 SYSTEM DATA

Sodium flow rate, 133 gpm	Fuel Pu-Fe eutectic alloy (~90 at % Pu), melting point $\cong 410^{\circ}\text{C}$
Average coolant inlet temperature, 450°C	Fuel in core = 51.5 vol %
Average coolant outlet temperature, 563°C	Sodium in core = 33.5 vol %
Average ΔT in fuel, 200°C	Tantalum in core = 15.0 vol %
Average heat flux through fuel capsule wall, 150 watts/cm^2	Fuel density $\cong 16 \text{ g/cc}$ at 500°C
Average specific power, 40 watts/g	Fuel-expansion coefficient $\sim 110 \times 10^{-6}/^{\circ}\text{C}$
	Neutron flux $\cong 3 \times 10^{14} \text{ neutrons/sec/cm}^2$ fast
	Median energy of neutrons producing fission $\cong 1 \text{ Mev}$ (calculated)
	Critical mass at operating temperature $\cong 24 \text{ kg}$

Table 2— LAMPRE START-UP CHRONOLOGY

Cold, dry critical	January 1960	(No sodium, solid fuel, 80°C)
Warm, wet critical	February 1961	(Sodium solid fuel 160°C)
Hot, wet critical	March 1961	(Sodium, molten fuel 480°C)

A description of production and evaluation methods for the fuel capsules is beyond the scope of this report. Some phases of the work will be mentioned in a paper by W. R. Wykoff in a later session; however, the following comments may be appropriate here:

1. Hot-trapped sodium does not appear to attack the fuel capsules in laboratory tests.
2. Successful containment of the fuel by the capsule material depends on many variables. Some of these are
  - a. Temperature.
  - b. Impurities or additives in the fuel alloy.
  - c. Impurities in the capsule material.
  - d. Heat treatment of the capsule material.
  - e. Temperature distribution in the system.

It has been found, for instance, that alloying tungsten with the tantalum seemed to improve the capsule quality. Most of the capsules in the first LAMPRE-1 core are made of a tantalum-0.1% tungsten alloy, and not of pure tantalum. Unfortunately these capsules have not had the heat treatment that current laboratory experiments indicate is necessary for reliable containment at the temperatures characteristic of 1-Mw operation. It has therefore been decided that the first core should be run more conservatively, at 300 to 400 kw, so that the capsule temperatures do not exceed 525 to 550°C .

Some data that were not specifically noted in this description of LAMPRE-1 are summarized in Table 1. Those parameters which are power dependent are given for the 1-Mw case.

Reactor Start-up

The chronology of all the operations that may be called a part of the reactor start-up is indicated in Table 2.

The "cold, dry" critical experiments were performed before any sodium had been charged to the system, in fact, before the construction of the reactor system was complete. At that time no tantalum capsules suitable for operation with molten fuel were available. Solid fuel was loaded into tantalum capsules available from various test programs, and a core was built up using these components. Rod and shim worths were determined, a transfer-function analyzer was tested, etc., at a core temperature of about 80°C . The reactor was then unloaded.

By October 1960 the reactor system was complete. Sodium had been purified by cold-trapping and had been charged to the reactor loop to begin loop cleanup and sodium hot-trapping. The system operated at temperatures of from 350 to 500°C until early 1961. Capsule-charger operation was studied, and a few fuel capsules were inserted into the reactor to ensure that results of capsule tests performed in small laboratory loops were comparable to results obtainable in the reactor loop.

Fig. 9—Density vs temperature. LAMPRE-1 fuel alloy

The “warm, wet” critical experiments were carried out in February 1961. One of the objectives was to determine the reactivity worth of the core sodium. Ideally this would have been obtained from a measurement carried out as near to 80°C as possible (perhaps 120°C). Safety considerations however indicated that it would be better to do these critical experiments well above the 122°C alpha–beta phase transition in the fuel (see Fig. 9). Accordingly a temperature of 160°C was selected for this phase of the start-up experiments. A measurement was also made of the critical mass with all control elements in the least reactive position. This mass establishes the permissible upper limit on reactor loading.

Following the warm, wet critical experiments, preparations were made to raise the system temperature, melt the fuel, and make a final approach to a critical loading at 480°C. Calculations indicated that the critical mass at 480°C would be substantially the same as that observed at 160°C. The reason for this is that there is an increase in density of the fuel as it melts; and, furthermore, as the fuel melts, it should completely fill the capsules, eliminating some voids that existed in the core while the fuel was in the solid state. In view of this situation, the core was unloaded until the neutron multiplication was about 10 at 160°C with all control elements in their minimum-worth position.

Neutron multiplication was monitored as the system temperature was raised to 500°C. As was expected the multiplication decreased slowly until the melting point of the fuel was reached.

It then increased, as indicated in Fig. 10. When the melting was complete, a decreasing multiplication was again observed as the temperature rose.

Before making the final approach to a critical configuration at 480°C, it seemed desirable to verify that the core was in its most stable and compact configuration. It was known from laboratory experiments that in an isothermal system, the fuel does not immediately fill the container even though the temperature is well above the melting point. Rather, voids in the fuel may occur, perhaps because of gas entrapment. Such a condition is illustrated in Fig. 11, which is a radiograph of two capsules melted in the reactor, then withdrawn, frozen, and radiographed. One of these capsules apparently contains a void near the tip; the second has a void near the top. In the laboratory such voids had been eliminated, or at least reduced, by jolting or jarring the capsules while the fuel was molten. It appeared that this would be a desirable thing to do in the reactor as well.

Before beginning this operation, however, the core loading was increased until a shim-up rods-up multiplication of 100 was achieved. Rods and shim were then lowered; and the capsules were jolted against the locator plate, one at a time, by means of the capsule charger. The multiplication was remeasured. Since it was found to have increased, the procedure was repeated until no further increase in multiplication could be produced. The total gain in reactivity was ~ \$1.3. Fuel loading was then resumed, and a routine approach to critical was achieved at a temperature of 480°C on Mar. 27, 1961.

A brief summary of the three critical experiments, measurements of isothermal temperature coefficients, etc., is given in Table 3.

Table 3—LAMPRE-1 START-UP EXPERIMENTS

Case	M_c	Shim worth	Rod worth	Cents/g fuel	Temperature coefficient
Cold, dry	25 35	5 1 kg of fuel	\$2 3	0 62	-1.1 cents/°C
Warm, wet	22.98	4 0 kg of fuel	\$2 2	0 68	-1.32 cents/°C
Hot, wet	23 37	Not measured	\$2 3	0 65	-3.21 cents/°C
Sodium worth at 80°C (experimental) ~ 2.94 kg of fuel					
Maximum permissible loading ~ 27 7 kg					

Since completion of the zero-power measurements, the reactor has been operated at progressively higher powers. Transfer-function measurements have been made in the interval 1 to 50 kw and also at 100 kw at various coolant flow rates. No anomalies in the transfer function have been observed.

The response of the reactor to power demand has been explored by making successively larger power demands. (The power demand was initiated by simply switching off the desired amount of electrical heating power in the coolant loop.) Data from a 30-kw demand experiment are shown in Fig. 12. The power coefficient is found to be ~ 0.6 cent/kw. Similar data were obtained for the case of a 50-kw demand. However, the power overshoot (~ 20 per cent) did not damp out in one oscillation. Doubling time was ~ 45 sec.

Two power runs at 50 kw, one for a 32-hr period and the other for a 60-hr period, were performed. In both instances, a steady persistent decrease in reactivity was observed. The rate of reactivity loss was about 0.6 cent/hr. None of the reactivity lost during the run was regained after the reactor was shut down.

There are at least two possible explanations for this effect. The first, which is hardly reasonable at such low power, is that gaseous fission products are accumulating in the fuel and reducing its density. In the case of one fuel capsule that was removed from the reactor and examined with a collimated energy-sensitive gamma-ray detector ("gamma-scanner"), it appeared that xenon was observed in the region above the fuel. If the data are reliable, this would indicate that gaseous fission products are escaping from the fuel, at least to some extent.

Fig. 10 — Multiplication vs. time and temperature during core melting. LAMPRE-1.

Fig. 11 — LAMPRE-1 fuel capsules after melting in test loop.

Fig. 12 — A 30-kw power-demand experiment. LAMPRE-1.

Fig. 13 — LAMPRE-1 capsules melted in test loop. The capsule on the right was held at 625°C for ~500 hr. The capsule on the left was subjected to an axial temperature gradient of 475°C at the bottom and 625°C at the top for a similar time.

The second possible explanation for the observed loss of reactivity is that impurities in the fuel are segregated at the top of the fuel and perhaps displace some fuel upward. Such phenomena have been observed in isothermal laboratory tests. Figure 13 is a radiograph of two fuel capsules; the one on the right in the picture was held at 625°C isothermally for an extended period, and the second one was subjected to an axial ΔT of 150 to 475°C at the bottom and 625°C at the top for the same length of time. In the isothermal case, impurities were collected near the top of the fuel; whereas, in the second case, they were found by chemical analysis to be concentrated in a very small region at the bottom of the capsule. At the low power of 50 kw, fuel capsules in the reactor probably behave very much like the isothermal test capsule; impurity segregation at the top of the capsules may then form a froth of low-density fuel, which would lead to the observed decrease in reactivity.

The next steps in the start-up program will, according to current plans, consist of raising the power to 100, 200, and 400 kw. Oscillator experiments and power demand tests will be done at each power. The effect of coolant flow, and hence ΔT in the core, will be studied to see if reactivity losses are correlated with thermal conditions in the core.

Sodium Reactor Experiment

L. E. GLASGOW

Atomics International

The Sodium Reactor Experiment (SRE) was built and is being operated for the Atomic Energy Commission by Atomics International, a Division of North American Aviation, Inc. The design of the plant was begun in July 1954 and the construction was completed in March 1957.

This plant is a 20-Mw(t) sodium-cooled graphite-moderated low-enrichment thermal reactor. It is of the tank type with a graphite moderator contained in individual hexagonally shaped 0.035-in.-thick zirconium moderator cans. The cans are 11 in. across the flats and 10 ft long. The 0.17-in. space between the cans is filled with sodium. Forty-three of these assemblies contain 2.8-in.-diameter axial tubes to accommodate the fuel clusters. The active core is a 6- by 6-ft right-circular cylinder. The core tank is 19 ft high and 11 ft in diameter. The two primary (radioactive) and two secondary (nonradioactive) mechanically pumped cooling circuits are external to the core vessel. The primary and secondary loops are thermally coupled by means of sodium-to-sodium shell-and-tube heat exchangers. The design temperatures for the primary loops are 500°F for the reactor inlet and 950°F for the reactor outlet. The secondary-loops design temperatures are 460 and 900°F. Flow in the main loops is 1100 gal/min.

The principal advantages of sodium-cooled systems are (1) a safe, simple, easily contained reactor system operating at pressures less than 1 psig, (2) thermodynamic cycle efficiencies in excess of 42 per cent, (3) ease of operation and control, (4) simple, direct-contact maintenance, and (5) complete chemical compatibility of the reactor coolant and the materials of construction.

The main purpose of the SRE is to test the technical feasibility of a large sodium-cooled reactor system operating at high temperature. The technical evaluation is based on a broad range of operating experience, both scheduled and unscheduled. This experience is supple-

mented with specific engineering tests aimed at defining any weakness or uncertainty in the plant indicated by analysis or plant performance characteristics.

The over-all technical feasibility of the concept has been established by more than three years of operations and tests. During this time this relatively small [6.0 Mw(e)] experimental plant has, in conjunction with the Southern California Edison Experimental Station, produced 15,331,050 kw-hr of electrical energy for the Southern California grid. During fiscal year 1959, when the experimental program was primarily limited to biweekly fuel inspection, the plant achieved a 75 per cent availability factor.

The plant has operated for extended periods with an outlet sodium temperature of 1000°F; and, during one test run, it operated at 1065°F in order to test the characteristics of the steam generator at a steam temperature of 1000°F.

The experimental high-temperature-reactor fuel program requires that the reactor be frequently shut down for fuel-element removal and examination. This activity, combined with experimental and normal scrams, has resulted in the equivalent of 20 years of normal start-up experience. As a result of this experience, normal reactor start-up time following a scram has been reduced from 2 hr to 20 min. Stresses in the core and core tank due to thermal transients are avoided by controlling the flow rate and maintaining the critical system temperatures constant. The flow is controlled by means of eddy-current brakes in both the primary and secondary systems.

Tests to determine the ability of the reactor to follow load show that the reactor can easily follow load swings at the rate of 20 per cent per minute on manual control. The limit on the rate of power change is determined by the steam turbine rather than by the reactor.

The steady-state operating stability of the SRE was demonstrated by noting during a 144-hr period that integrating timers on the control rod showed only 3.5 min of control-rod movement. The dynamic stability of the reactor at all frequencies up to 20 cycles/sec has been shown by pile oscillator tests made at power. The power coefficient for Core I is

$$- \frac{3.5}{N_0} \times 10^{-4} \left(\frac{\Delta K}{K} \right) / \text{Mw}$$

at a constant temperature of 335°F across the reactor, where N_0 is the power level at which the coefficient is measured.

Measurements of the radial power distribution in the core indicate a ratio of outer channels to inner channels of 0.70 as compared to a predicted value of 0.50. This provides for better utilization of the fuel and will in future reactors, where the fuel temperatures are expected to be limiting, permit additional total power from a given core.

Our experience with specific components has been as follows.

Fuel Elements

The SRE first-core fuel elements have been structurally reliable. Irradiation of about 1000 Mwd/ton produced swelling that increased the diameter of the rods about 3 to 4 mils. This simple alpha-rolled beta-heat-treated uranium fuel would appear to have a life no better than about 2000 Mwd/ton. The second-core fuel, which is in the reactor now, is a thorium - 7.6 wt.% uranium alloy; and it is expected to last to approximately 5000 Mwd/ton. Advanced fuels of high thermal conductivity and high temperature stability such as uranium carbide are indicated for advanced sodium reactors. These fuels should be good to at least 25,000 Mwd/ton. Central temperature of the slugs has proved to be ~100°F lower than expected. At 1000°F reactor outlet temperature and 20 Mw(t), maximum slug temperature is about 1100°F.

Fuel-handling Experience

Our fuel-handling experience is equivalent to approximately 20 years of normal operation and has led to the design and construction of an improved fuel-handling cask. The original fuel-

handling cask, although adequate for the purpose, required excessive maintenance and was a single-purpose device. A new cask which eliminates these difficulties and provides greater flexibility for a variety of shielding tasks has been built and operated. The principal design changes include complete streamlining of the cask interior to prevent fuel hang-up, the elimination of sliding O-ring seals where the surfaces can become contaminated with sodium, and provisions for easily disassembling the cask in sections for maintenance and decontamination.

The original fuel-washing system, which utilized a direct water wash, worked perfectly the first 1100 times, but during the 1101 wash a pressure excursion blew the shielding plug from the wash cell. Subsequent investigations disclosed that the 2.5-in.-diameter 6-ft-long cylindrical hold-down tube, a part of the fuel assembly, failed to drain clear of sodium. This failure introduced about ten times more sodium than the cell was designed to handle. The wash cells have been redesigned to utilize steam as the washing medium.

Main Intermediate Heat Exchanger

An unexpected result from the first power run was the 50 per cent excess log mean temperature difference observed on the main intermediate heat exchanger. Between power runs 60 thermocouples were distributed over the length and around the girth of the heat-exchanger shell. The temperature curve rose uniformly at the straight portions of the exchanger but was horizontal around the bend, indicating that no heat transfer is occurring in that section. Examination of the construction photographs disclosed a wide gap between the tube bundle and the shell; and, since this section is not baffled, the fluid simply by-passed the tubes. The loss of this amount of heat-transfer area accounts for the high log mean temperature difference. Transient data taken on this equipment disclosed that rather complete stratification occurs on the shell side of the exchanger. A replacement for the present heat exchanger, embodying improvements based on these studies, is on hand. In the interim the present exchanger will be operated without modification.

Steam Generator

During the initial operation of the once-through sodium steam generator, it was noted that the upper tubes were running approximately 200°F hotter than the lower tubes. This temperature difference is initiated by a small amount of temperature stratification in the shell-side sodium, which causes the upper tubes to run slightly hotter than the lower tubes. The effect is amplified because the effective superheater length of the upper tubes is increased, which in turn increases the tube pressure drop and decreases the water flow. The process continues until the flow to the upper tubes is virtually cut off. The opposite sequence of events occurs in the lower tubes. This difficulty was resolved by the boiler manufacturer and the Southern California Edison Company with the addition of orifices at the water-tube sheet. The orifices provide enough self-regulation to limit the top-to-bottom temperature difference to about 50°F. The steam generator has performed perfectly since the modification.

Control and Safety Rods

The control- and safety-rod performance has been excellent. The maximum dimensional change observed on the control rods was 0.006-in. diametrical change after 2140 Mwd (7.5 per cent B^{10} burn-up). Maintenance on the rod drives has been extremely low, requiring a brake replacement on one of the four rod drives. There have been no difficulties with the four safety rods.

No difficulty has been experienced in maintaining the O_2 concentration below 10 ppm by means of the cold traps. As the cold traps become filled they are simply replaced. This has been done about eight times and without difficulty.

Sodium Valves

The SRE is equipped with two kinds of sodium valves. One type uses a frozen-sodium shaft seal that is similar to the pump shaft seal. This type has operated with 100 per cent reliability. The other type is a bellows-sealed valve. Eight of these in the service lines have had to be replaced owing to damage to the bellows resulting from solid-sodium extrusion through the pipe during preheating operations. We have never had a failure of a bellows-seal valve on lines that remain heated. All the bellows-seal valves that were in the main primary loops have been eliminated.

Sodium Pumps

The SRE sodium pumps are conventional centrifugal pumps with modified shaft seals. The shafts are sealed against sodium leakage by freezing off a 20-mil annulus of sodium around the shaft; as the shaft turns it continuously shears the solid sodium in the annulus. The total integrated time logged on all four pumps amounts to 75,000 hr with an over-all reliability of 95 per cent. Although all four pumps are identical, our experience with them has been quite varied. Any given pump will run completely trouble-free for long periods (~10,000 hr) while one or more of the remaining pumps is experiencing shaft-seal difficulties.

The several trouble areas with the shaft seal have been shaft binding, solid-sodium extrusion, and gas in-leakage into the sodium system. The fact that the pumps can perform perfectly on occasion suggests that there are some not yet understood details that need to be remedied. The 95 per cent pump reliability is not adequate for power-plant operation, especially in the primary system where there is an approximately 4- to 10-day delay to gain access to the pumps for maintenance. Recently designed shielded handling equipment can reduce this waiting period to approximately one day, which is still excessive. For future reactors the reliability of the freeze-seal type pump must be greatly improved or a different pump type, such as a free-surface pump or an electromagnetic pump, is strongly indicated.

It was the failure of the shaft seal on the main primary pump which introduced tetralin into the main primary sodium.

Core Maintenance Experience

The tetralin leak that developed in the main primary pump added approximately 4 gal of tetralin to the main primary sodium. The tetralin decomposed into hydrogen and carbon. The carbonaceous material lodged in the fuel channels, partially obstructed the flow, and provided local coolant by-passing of the fuel. This resulted in overheating and damaging the fuel elements to the extent that they swelled and stuck in the channels. Of the 43 fuel elements in the reactor, 13 were damaged. The upper halves of 10 of the broken elements were removed by a modified standard procedure utilizing the new fuel-handling machine. A special cylindrical probing tool was devised to explore the possibility of grappling and removing the remaining fuel from the 10 channels involved. These probings revealed that the fuel had distorted sufficiently to completely take up the channel clearance (nominally 29 mils), and they could not be removed without damaging the moderator can. Therefore it was decided to remove the lower part of the fuel element and moderator assembly as a single unit. The hanger rods and shield plugs of the two stuck elements (R-24, R-76) were separated from their respective fuel clusters by rotating the shield plugs. This action sheared the hanger rod at the shear pin that is located near the top of the moderator cans. Following the removal of the 30 sound fuel elements, one complete but defective element, and the upper halves of the 10 parted elements, the sodium was drained from the core, and electrical heaters were added to maintain the core at 350°F. The helium cover gas was replaced with argon.

At this time inspection of the core was begun. The radiation level at the bottom of the loading face shield was measured and found to be 43 r/hr. One of the 3-in. plugs in the top shield was removed by using the fuel-handling cask, and a glass window $\frac{1}{2}$ in thick was in-

stalled. The radiation level above the glass was about 50 mr/hr. The contamination in the cover gas was determined to be about $4 \times 10^{-3} \mu\text{C}/\text{cm}^3$ of Kr^{85} in helium; replacement of argon reduced activity to about $7 \times 10^{-4} \mu\text{C}/\text{cm}^3$. The condition of the top of the core was examined with the aid of special viewing devices constructed for this purpose. This examination revealed a layer of black, flocculent, carbonaceous-appearing material on the tops of the cans. On top of this layer were about 82 separate fuel slugs scattered at random, some bits of fuel cladding, and several pieces of wire wrap. This material had fallen on top of the core during the removal of damaged fuel clusters. Location of this debris was mapped, and preparations were made for its removal.

The fuel slugs and cladding fragments were picked up by articulated grapples fitted through the 3-in. fuel-plug openings in the top shield and placed in a 2.5-in.-diameter 3-in.-long bucket suspended from a shield plug in an adjacent corner channel. The buckets were removed by means of the fuel-handling machine, canned, and stored with the other damaged fuel. Viewing was accomplished by means of a "corescope," which was simply a 3-in.-diameter 10-ft-long tube with a lens sealed in the bottom end and a telescope mounted in the upper end. Lighting was provided by means of mercury-vapor lamps suspended from shield plugs. Removal of most of the flocculent carbonaceous material was accomplished by vacuuming the tops of the cans. The vacuum cleaner consisted of a commercially available blower, a nozzle, glass-fiber flexible hose, filter and settling tank, and shielding. The filtered gas was discharged back to the reactor. One filter change was needed to keep the radiation level below 90 r/hr at the surface of the filter. Roughly 3 lb of carbon, estimated from the associated radioactivity, were removed from the system by this means.

Preparations for the moderator can removal through the 40-in. plugs in the loading face shield went on concurrently with the core and system cleanup. Although the moderator replacement equipment had been largely designed and most of the parts manufactured, it was necessary to complete the assemblies and thoroughly check them out. In addition, means were developed to detect sodium in-leakage to individual moderator cans to determine which cans had failed. The two most useful techniques were (1) use of an induction probe lowered through a thimble placed in the moderator-can fuel channel to detect the electrical resistance change expected in sodium-saturated graphite and (2) a probe of the can heights to detect the 1 to 1.5 per cent dilation of graphite accompanying sodium absorption. Both methods indicated leaks in cans R-10, which had lengthened $\frac{7}{8}$ in., and R-42, which had elongated $1\frac{1}{2}$ in.

The moderator-can encapsulating station was equipped for viewing the cans and obtaining zirconium samples. After the cans were inspected, they were sealed in argon-filled metal storage capsules. The capsules were then transported to the solid-waste storage building.

All the equipment used was gas tight; seals between the various assemblies were checked for leak tightness at each step. All seals were double; spaces between the seals were purged to the reactor radioactive vent system. These precautions prevented escape of radioactivity from the core into the work area.

During the moderator-can removal operation, all the cans were inspected for carbon deposit. None was observed on the sides of the removed cans or on the six adjacent can panels. A light carbonaceous deposit was noted on the bottom of some cans. Observations of the top of the gird plate through the void created by the removed can disclosed a very thin and somewhat mottled dark coating. The amount observed did not warrant the additional manipulative and exposure hazard that would have been necessary for its removal.

By means of a special draining device, the sodium level in the lower plenum was reduced to approximately $\frac{1}{4}$ in. in the deepest area. This left several mirror-surfaced sodium puddles and exposed about one-half of the core tank bottom which was perfectly clean and bright. Two fuel slugs were observed on the floor of the tank; these were removed.

After it was established that the system was reasonably clean, it was refilled with sodium, heated to 600°F, and sodium circulated for seven days to check for any additional leaky cans. Three were found: R-32, R-44, and R-45. These were replaced. It was subsequently established that two of the three cans (R-32 and R-44) failed because of mechanical damage to the short unprotected zirconium pump-out tubes during fuel-slug grappling operations. The third can (R-45) was not examined in sufficient detail to establish the point of sodium entry. This made a total

of 16 cans replaced. A second sodium circulation test and height probe showed that all the cans were sound. The critical loading was conducted, the cans were again tested and found to be intact, thus validating critical-loading measurements. Any future can failures can be detected by changes in reactivity.

It is interesting to note that it took approximately one week to replace the first moderator can whereas the last can was replaced in a little less than 2 hr.

It was observed that carbonaceous material taken from the core had a specific activity about 10^5 times as great as that of the sodium. This phenomenon provides a ready means of locating carbon deposits in the piping and equipment. Measurements made in a 10-in.-diameter thimble that penetrates the main gallery showed that the gamma field at the main intermediate heat exchanger decreased from 2.5 to 1.8 r/hr while the cold trap increased from 68 to 80 r/hr over the same time interval (August 18 to 25, 1959), indicating that the activity was being transferred from the piping into the cold trap. After the gallery shield blocks were removed, a detailed survey was made of the piping. Over most of the pipe runs, the activity level was in the range of 0.1 to 0.2 r/hr. At several specific locations the radiation levels were somewhat higher. A sodium circulation test was performed to reduce the radiation levels. The resulting reduction in radiation levels at the control points were as follows: hot-trap economizer, 7.0 to 0.95 r/hr; main primary pump discharge, 18 to 7.0 r/hr; and bottom of the pump case, 0.4 to 0.16 r/hr. At the same time the radiation level of the replaced cold trap increased from 2 to 11 r/hr, again indicating the transport of carbon from the piping to the cold trap.

Piping System

Maintenance or modification of the sodium piping has proved to be remarkably straightforward, simple, and safe. On the primary loop the sodium activity is allowed to decay for about 10 days. The system is then cooled permitting the sodium to solidify in the pipe. A conventional pipe cutter is used to cut through the stainless-steel piping. The sodium is then removed from the piping, and the sodium in the pipe is dug back about 6 in. with the use of a spatula. The new part is then welded in the line, and the weld is checked for leaks with helium and x-rayed. There has been no piping maintenance on the primary sodium system, only modifications. On the secondary system four valves have been replaced and the system has been modified from time to time for experimental purposes.

Current Activities

We are at present studying the dynamic characteristics of the second SRE core at low power. At the completion of the studies, the reactor temperature and power will be raised, and hot trapping of the sodium to remove dissolved carbon will begin. Full power will not be achieved until after the carbon content is reduced below 18 ppm.

Conclusions

The following conclusions have been reached:

1. The technical feasibility of the sodium-graphite reactor concept has been amply demonstrated up to 1000°F.
2. Maintenance work, whether it involves replacing a primary system valve or a part of the core, can be accomplished in a simple, direct, and safe manner.
3. The containment features of the SRE primary system were proved by retaining all the fission fragments released from the twelve severely damaged fuel elements.
4. With the exception of the inert gases, all the fission fragments remained with the sodium. Ninety-five per cent of these have been removed from the sodium by cold trapping.
5. The components that were developed for the SRE have proved to be functionally satisfactory. Improvements in the reliability aspects of the SRE pumps and valves should provide a plant availability factor in excess of 90 per cent.

Figure 1

Figure 2

Fig. 3—MIHE shell temperatures 18 min after scram.

Fig. 4—Frozen seal. Liquid sodium at 1200°F. 30 hp, 1200 rpm.

Fig. 5—Mark II corescope.

Fig. 6—Bottom of R-55 fuel element.

Figure 7

Fig. 8—Moderator removal equipment.

Hallam Nuclear Power Facility

R. C. GERBER
Atomics International

The Hallam Nuclear Power Facility (HNPF) project was authorized late in 1957 under the first round of power demonstration reactors. The nuclear plant is owned by the U. S. Atomic Energy Commission and will be operated by the Consumers Public Power District of Nebraska (CPPD) at their Sheldon Station site 20 miles south of Lincoln. The nominal rating of the nuclear plant is 75,000 kw(e), produced from 710,000 lb/hr superheated steam at 800 psig and 825°F. Plant equipment is designed 6 per cent above this rating to ensure meeting the initial objective, hence a rating of about 80,000 kw(e) is expected to be achieved.

The final design of any power reactor is the result of a series of compromises to incorporate previous experience, to minimize power cost at operating conditions, and to serve as a prototype for expected future reactors. Therefore the HNPF design reflects the design experience of the Sodium Reactor Experiment (SRE), the requirement for approximately 75,000 kw(e) at minimum total cost, and the predicted trends in future sodium graphite reactors. The design has also been influenced by an important objective of demonstrating reliable operation of this type plant. This design feature was expressed in the contract between the AEC and CPPD as follows: "to demonstrate the economic and technical practicability of control station power utilizing a sodium-graphite nuclear reactor, by operating the plant for the primary purpose of producing a maximum amount of electrical energy therefrom. . ."

You have just heard Mr. Glasgow's discussion of the status of the SRE. I plan to cover the status of the HNPF by discussing the major differences between the SRE and HNPF reactors, giving a summary of development specifically related to HNPF, a brief description of reactor facility, and a schedule of remaining effort.

Figure 1 shows the over-all Hallam program from its beginning in late 1957 to the expected plant turnover to CPPD in late 1962. Component development, fuel development, and preliminary design were initiated simultaneously. At that time the basic fuel was uranium oxide, but the core was sized to uranium-molybdenum, uranium carbide, or thorium-uranium fuels. Following a review of the Title I design, during September and October 1958, the initial fuel loading was changed to uranium-molybdenum, and development was started on uranium carbide as a possible replacement fuel. Construction was started in April 1959 and is scheduled for July 1, 1961, completion.

Design Based on the Sodium Reactor Experiment

The design concept is based on the SRE, scaled up by approximately a factor of 10 in rating. The basic sodium-cooled graphite-moderated thermal core concept utilizing removable graphite elements in a zero-pressure tank type vessel is retained. Major differences are the result of following the objectives previously mentioned. Some of these differences are:

1. The moderator cladding material was changed from zirconium to stainless steel. This change was made primarily to gain experience in the fabrication of the stainless-steel sheaths since it is expected that SGR's operating at higher temperatures will require stainless steel. Zirconium and Zr-II are marginal at the 1000°F outlet sodium expected for future operation of the HNPF core. A design change from the SRE centrally located fuel elements to the HNPF fuel location in the moderator periphery reduced relative neutron capture in the sheath material and made the material change feasible.

Fig. 1—Over-all Hallam Nuclear Power Facility program.

2. The initial fuel loading for HNPF will be uranium-10 wt.% molybdenum instead of the initial unalloyed uranium or the present thorium-uranium in the SRE. Uranium-10 wt.% molybdenum was selected to provide long burnup (3 to 6 years per element) using a fuel with extensive developmental testing and well-known reprocessing and refabrication methods. Uranium carbide is being considered as a replacement fuel.

3. Variable orifice flow adjustment for each fuel element has been added to HNPF fuel elements. Fixed orifice with adjustment in a maintenance cell has proved to be time consuming in the SRE. The addition was made to increase total operating time and should result in more uniform fuel-element outlet temperatures.

4. Shim and safety rods have been combined into single rods. This combination was originally considered desirable for the SRE, but was not utilized owing to heat-removal questions. Tests in the SRE have shown that the combination is feasible for HNPF. This area is one in which HNPF is not expected to be a prototype for a higher specific-power reactor.

5. Free-surface sodium pumps have been selected for HNPF. The SRE uses freeze-seal pumps. A tetralin leak in the main SRE primary pump caused the fuel damage mentioned by Mr. Glasgow. Although the SRE solution to the problem, which calls for a revision of the freeze seal to sodium-potassium, is satisfactory for safety purposes, the net result is a more expensive pump. Therefore the lower cost free-surface pump was selected for HNPF.

6. Kettle type steam generators with double-walled bayonet tubes have been selected for HNPF. The SRE can reject heat through air-blast heat exchangers, a once-through steam

generator, or a pancake-coil steam generator originally made for SIR. The HNPF selection was made at a time when water-side leaks were causing difficulties in the sodium steam-generator field. The design selected is based on the Knolls Atomic Power Laboratory and Navy choice for replacement SIR units. The choice was made to maximize operating time. Current AEC programs sponsor development to prove reliability of less expensive units so that I do not expect HNPF will serve as a prototype in this area.

Development

A development program for direct support to HNPF design and operation was established as a part of the original program. Development models of all major reactor equipment were fabricated before final design of the production units. Operational tests were conducted to obtain operating characteristics and an estimate of component life. Development is essentially complete on control rods, moderator elements, sodium pumps, intermediate heat exchangers, small sodium components, steam generators, sodium instruments, protective system, uranium-molybdenum fuel-element hydraulics, and fuel handling. Results from the component development increased our confidence that the plant will run with a high degree of reliability.

Development is continuing in the fuel field. Both uranium-molybdenum and uranium carbide are being irradiated to assist in determination of allowable burnup of the operating elements. Seven uranium-molybdenum capsules are in the MTR. Four of these are running below and three above the expected maximum operating temperature of 1250°F. The maximum burnup is now about 8000 Mwd/metric ton of uranium. Four uranium carbide capsules remain in the MTR and WTR. One uranium carbide capsule with burnup of about 8000 and another with about 20,000 Mwd/metric ton of uranium have been removed from the MTR and are now ready for hot-cell examinations. Fabrication development is just starting on uranium carbide fuel elements so that replacement element design can be released in February or March of 1962. Uranium carbide slugs have been fabricated for exponential experiments and these tests are now starting on a core of mixed uranium-molybdenum and uranium carbide elements. Ten full-sized uranium carbide fuel elements are being fabricated to gain fabrication experience and for possible irradiation at HNPF.

Reactor Description

Construction is nearly complete on the nuclear facility and has been completed on the conventional boiler-turbine side. This meeting has been timed to give you an excellent opportunity to tour the site when nearly all the components are in place but when the reactor core and process cells are still accessible.

The reactor building (see Fig. 2) houses the reactor core and structure, heat-transfer systems, fuel storage, fuel-handling equipment, and auxiliary service systems. This building is of steel-frame construction with metal siding. Roof and walls are designed to form as tight an enclosure as is possible with conventional materials. The building is 275 ft long with an 80-ft-wide 70-ft-high bay over the reactor. Beneath the operating floor are the reactor, primary heat-transfer system, and fuel-storage cells. Under-floor compartments are steel lined to limit helium and nitrogen leakage. Steam generators and associated components of the secondary heat-transfer system are located in three separate enclosures at the north end of the reactor building. Adjoining the reactor plant is a control room serving both the conventional plant and the nuclear facility. Fuel is handled above floor by a lead-shielded fuel-handling cask mounted on a gantry that operates between the reactor and the fuel-storage areas.

Figure 3 shows the reactor in more detail. The core is contained in a stainless-steel vessel 33 ft deep by 19 ft in diameter. Building blocks for the core are hexagonal prisms of stainless-steel-sheathed graphite. These moderator elements are 16 in. across the flat of the hexagonal by 18 ft high, have been fabricated at the site, and are now being installed in the reactor vessel. The last elements should be installed on May 26, 1961. A loading-face shield

Fig. 2—Reactor building.

Fig. 3—Detailed view of reactor.

is supported by rollers near the upper surface to close the top of the reactor vessel. This loading-face shield is now located just north of the dust-proof cover over the reactor vessel. Iron-limonite concrete was poured into the loading-face steel shell last week, and the shield will be moved into place in about three weeks. Fuel elements, control rods, and core instruments are suspended from shield plugs in the loading-face shield and located in scallops built into the corners of the hexagonal moderator elements. Heat is removed from core elements by an upward flow of sodium from the plenum shown under the grid plate and then through the elements into a free-surface pool over the core. Three primary sodium pipes enter the reactor cavity through diaphragm seals that separate the helium atmosphere of the cavity from a nitrogen atmosphere of the process cells. Inlet pipes discharge into the lower plenum and exit just above the moderator elements. An outer vessel is built around the reactor vessel to contain sodium leakage and maintain a safe coolant level in the event of a leak in the main vessel. Guard pipes cover primary pipes to channel sodium into the outer vessel. The cavity area is blanketed with helium and held at the same pressure as inside the reactor vessel.

Figure 4 shows a cutaway view of a moderator element and, by section, the relative locations of fuel, graphite, and control rods.

Figure 5 is the uranium-molybdenum fuel element. An 18-rod cluster is supported in a Zircaloy-2 process tube. The process tubes seat in the grid plate to channel sodium through fuel rods. Fuel rods are about 0.6 in. in diameter and filled with slugs that stack 13.5 ft high.

Fig. 4—Cutaway view of a moderator element. Relative locations of fuel, graphite, and control rods.

Elements are suspended from shield plugs and hanger tubes. A variable orifice plug is located at the element outlet to provide for adjustment to maintain constant sodium outlet temperature.

Figure 6 is a section of the control rod. Nineteen of these elements will be used in the core. The poison rod operates in a Zircaloy-2 thimble that maintains a helium atmosphere over the poison column. Poison elements are gadolinium oxide and samarium oxide canned in Hastelloy. Elements are operated from a ball-nut-and-screw mechanism mounted in a housing above the loading-face shield. The housing is supported on a carriage that is currently located in its maintenance area east of the reactor. The control rods are now being proof-tested in our development tower at Santa Susana. They will be shipped to HNPF during the next two months.

Figure 7 shows a primary pump. The primary pumps are rated at 7200 gal/min at about 160 ft head. Pumps are of free-surface type with the surface set by an overflow pipe. Reactor atmosphere is connected to each primary pump to maintain operating pressure. The motor and magnetic coupling are located above the reactor floor. A steel shot-filled column fills the case to provide shielding over radioactive sodium. Secondary pumps are similar except for the shield section.

Figure 8 shows an intermediate heat exchanger. Primary sodium enters the top and flows downward through straight tubes. Secondary sodium is in the shell side. A bellows expansion joint at the center of the shell compensates for differential expansion.

Figure 9 is a steam generator. Units shown from the top down are an eliminator, a superheater, and an evaporator. The main feature of the evaporator and superheater is the double-walled bayonet type tube with a helium monitor space between to detect either sodium or water leakage. Process equipment is installed.

Figure 10 includes a photograph of the fuel-handling machine as it was tested at Santa Susana and cutaway sketches of the internal mechanisms. One fuel element is installed on the fuel-handling machine from a storage cell. Then the machine is located over the reactor and a gas lock made to the loading-face shield. A used fuel element is grappled and pulled up into the machine. The used element is rotated 180° and centered over a helium blower for cooling if needed. In this position the replacement fuel element can be lowered into the reactor. The grapple seals the bottom of the machine for moving across the reactor floor.

Figure 11 summarizes the heat-removal conditions in the primary and secondary loops. A reactor rating of 240 Mw(t) to produce 75,000 Mw(e) is expected. Three heat-removal circuits are used with about 2,700,000 lb/hr of sodium for each. Sodium enters the reactor at 610°F and exits at 945°F. This heat is transferred in the intermediate heat exchanger, with sodium leaving at 895°F and entering again at 557°F. The steam produced is at 850 psig and 833°F.

Table 1 summarizes reactor data, and Fig. 12 summarizes process data. The design case, including the 6 per cent safety factor, is given.

Table 1 — DESIGN DATA

Reactor		Sodium Systems 256, Mw(t)	
Nominal full power	240 Mw(t)	Number of circuits	3
Design base rating	256 Mw(t)	Flow rate, per circuit	2.8×10^6 lb/hr
Moderator	Graphite	Temperatures	
Active core size	13 ft diameter $\times$ 13.5 ft high	Primary hot leg	945°F
Reactor vessel size	19 ft diameter $\times$ 33 ft high	Primary cold leg	610
Fuel Material	U-10 Mo	Secondary hot leg	895
Enrichment	3.6% U ²³⁵	Secondary cold leg	557
Number of elements		Steam systems, 256 Mw(t)	
Initial	137	Steam flow, total	752,000 lb/hr
Maximum	182	Throttle pressure	800 psig
Peak exposure, objective	15,000 Mwd/metric ton of uranium	Throttle temperature	843°F
Control rods	19	Estimated net output	80 Mw(e)
Poison material	Gd ₂ O ₃ - Sm ₂ O ₃		

Remaining Effort

The remaining effort on HNPf includes safeguards, preoperational testing, postcritical testing, power operation, and design of replacement fuel elements. The final summary Safeguards Report was issued in mid-April. We expect to review it soon with the AEC Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards. Figure 13 shows the effort here at the site. Preoperational testing started in April on subsystems. Plant construction will be complete about the first of July. Atomics International operator examinations should be held during August. A public hearing should be held in September, and a dry critical should take place in September or early October. At that time the heat-transfer systems will be filled with sodium and the related preoperational tests completed. The wet critical will take place about the end of the year; low-power tests will be performed through the first part of 1962; and CPPD will assume operation of the plant late in 1962.

Speaking for Atomics International and Bechtel and Peter Kiewit Sons', Inc., we are proud of the record of building the Hallam Reactor as scheduled four years ago and confidently expect it to operate safely with a high operating factor.

Fig. 5—Uranium—molybdenum fuel element.

Fig. 6—Sections of control rod.

Fig. 7—Primary pump.

Fig. 8—Intermediate heat exchanger.

Fig. 9—Steam generator

Fig. 10—Photograph of the fuel-handling machine and cutaway sketches of the internal mechanisms.

Fig. 11—Summary of heat-removal conditions in the primary and secondary loops.

Fig. 12—Summary of process data.

Fig. 13—Hallam Nuclear Power Facility testing and start-up program schedule.

Integration of Hallam Nuclear Power Facility into Consumers Public Power District of Nebraska System

W. P. VENABLE

Consumers Public Power District of Nebraska

The integration of the Hallam Nuclear Power Facility (HNPF) into Consumers Public Power District of Nebraska (CPPD) Eastern System is broader than the subject because the Eastern System transmission lines are integrated into the transmission grid covering the eastern two-thirds of Nebraska.

Nebraska is an all-public-power state. The electric services are supplied to the customers by public power and electric membership associations or by individually owned and operated municipal plants.

With the exception of Omaha and the immediate area, the Eastern System of CPPD is the largest public power district supplying electric service in the eastern two-thirds of Nebraska. Consumers Public Power District also has a Western System. However, this is not electrically interconnected with the Eastern System and need not be discussed further.

Consumers Public Power District is 100 per cent financed through the sale of revenue bonds. All its revenue bonds are sold under public bidding to the low bidder (a bond firm or a syndicate) who in turn sells the bonds to insurance companies and others, including the general public. As a result private capital is used to pay for all CPPD capital additions to the plant.

The status of the Eastern System of CPPD as of Dec. 31, 1960, is given in the following list:

Bonds outstanding: \$55,174,000	Cities and communities served
Electric revenue: \$17,879,384	Retail: 268; Wholesale: 54
Customers served: 106,770	System demand: 209,177 kw
	Sales: 867,659,677 kw-hr

Since its inception, CPPD has been purchasing over 95 per cent of its power requirements for its Eastern System from the high-voltage transmission grid operated by the Nebraska Public Power System; but, prior to entering into negotiations with the AEC for the construction and operation of the HNPF, CPPD made a decision to construct a power plant to supply part of its 1961 loads and to pick up its load growth thereafter. System studies have established that more generating capacity will be needed by the summer of 1961 to carry part of the summer loads. The study also established that that generating capacity should be installed south of Lincoln to regulate transmission voltage in southeast Nebraska.

In view of the results of the studies and in view of the fact that the heaviest load was at Lincoln, CPPD began investigations for a plant location south of Lincoln.

We had determined the following factors and course of action

1. We would install a conventional turbogenerating plant to produce 108,000 kw(net) operating at 1800 psig with a reheat steam cycle of 1000/1000 F
2. The location would be in the vicinity of Hallam, Nebraska, as this location matched our plant-location criteria, which was as follows:
  - a. Have ample underground water with minimum salt contamination
  - b. Be near a railroad for hauling coal and construction material.
  - c. Be within a short distance of existing 115-kv transmission lines.
  - d. Be close enough to back up the Lincoln load with minimum transmission investment
  - e. Be within a reasonable distance of natural-gas transmission lines.

However, after entering into the AEC Contract, we had to resurvey the area because we had to add the following factors to our plant-location criteria

- a. Population density.
- b. Wind relation to existing communities.
- c. Drainage area.

The present location of HNPF was selected as it met almost 100 per cent of our location criteria. (See Fig. 1.)

Steam Source

The use of the reactor as a steam source to operate a turbine under utility production conditions introduced the following major problems

1. Would the reactor operate as well as a fossil-fired boiler.
2. Would the reactor be in operation by the time we needed new capacity (in the spring of 1961).
3. What would it cost to produce steam by the reactor.
4. Would we need a special turbine to meet the steam conditions of the reactor (800 psig, 825°F).

After months of study and investigations, we decided on the following program, which, in our opinion, would best answer our present and future production needs:

1. We should not depend on the reactor as we would on a fossil-fired-boiler steam source. It was recognized that HNPF was an experimental program built to collect operating data for future programs.

2. We would have to proceed with the construction of the conventional facility in order to have the necessary generating capacity available early in 1961.

3. After many conferences with turbine builders, it was decided that the best answer to our turbine problem would be to modify a standard 1450-psig 1000/1000°F reheat turbine for operation at 850 psig and 900°F

4. The steam generator (boiler) was selected on the basis of present and future need. Our load growth projections indicated that a second 100,000-kw unit would be needed on the system by 1965. Also our studies had determined that a 100,000-kw turbine operating at 1800 psig with a reheat cycle of 1000/1000°F was the most economical unit for the Sheldon Station location. Therefore, an 1800-psig boiler with reheat of 1000/1000 F to deliver 790,000 lb of steam was selected as the base unit. However, in order to meet the steam conditions of the modified turbine conditions of 850 psig and 900°F, the boiler was also modified to deliver steam at 850 psig and 900°F. This was done by using the reheat section as an economizer section and leaving off the reheat circulating fans. This arrangement means that the boiler can be changed back to its design conditions in a short time. Further, the CPPD and AEC contract covering the operation of HNPF would be nearing the end of the five-year contract period. If the reactor proves a reliable and an economical heat source during the contract period, CPPD can exercise its option to purchase HNPF and continue to use it to provide steam to the modified turbine, thus releasing the fossil-fuel boiler to supply steam to a second turbine rated at 128,000 kva which would operate under steam conditions of 1800 psig and 1000/1000°F reheat.

Fig. 1—Location of Sheldon Station. Hallam Nuclear Power Facility.

5. In order to meet our capacity requirements in 1961, we decided to later use HNPF as a steam source and to provide for additional capacity around 1965. Design and construction were carried out in the following manner:

a. The conventional facility was constructed by using a modified 1800-psig boiler and a modified 1400-psig turbine, and it is now in operation on the modified steam conditions of 850 psig and 900°F. (See Fig. 2.)

Figure 2

b. Provisions were incorporated in the design to make it possible to connect the conventional facility to HNPF. The turbine facility can then be operated from the reactor and backed up by the boiler. (See Fig. 3.)

c. If the reactor has proved to be a reliable heat source by the time the second 100,000 kw is needed at Sheldon Station, the boiler will be changed to operate at its design condition of 1800 psig and reheat cycle of 1000/1000°F since the second turbine will be purchased to operate at 1800 psig and on a reheat cycle of 1000/1000°F. (See Fig. 4.)

d. If for any reason HNPF does not fulfill the requirements required for providing steam under utility operations, we can replace the reactor heat source by installing a second boiler to operate at 1450 psig, 1000/1000°F reheat cycle and change the modified turbine back to its maximum design conditions of 128,000 kva, 1450 psig, and a reheat cycle to 1000/1000°F.

Integration into CPPD Operation

Consumers Public Power District transmission system covers a large part of the eastern two-thirds of Nebraska and supplies service to 268 communities at retail and 54 at wholesale, totaling 106,770 customers. In order to supply the widespread area, the CPPD transmission system is completely integrated with that of the Nebraska Public Power System, which in turn is integrated with the Omaha Public Power District and with outstate power facilities such as the U. S. Bureau of Reclamation Missouri River Hydro System in South Dakota, North Dakota, and Montana and the Southwest Power Pool through a connection to Kansas. (See Fig. 5.) As a result, the CPPD system is a part of a very large power pool. Nebraska is represented in

Figure 3

Figure 4

Fig. 6—Plot plan.

the pool by the Nebraska Public Power System (including HNPF capacity) and the Omaha Public Power District. The following table outlines their contributions.

Table 1 — 1961 CAPACITY (GENERATION AND PURCHASES
FROM THE U S BUREAU OF RECLAMATION)

Nebraska Public Power System and CPPD	646,000 kw
Omaha Public Power District	536 000 kw
Total	1,182,000 kw
Estimate of 1961 loads	
Nebraska Public Power System and CPPD	494,000 kw
Omaha Public Power District	410,000 kw
Total	904,000 kw

The CPPD-AEC contract provides among other things that CPPD will operate HNPF in the same manner that it would operate a fossil-fuel boiler steam source and that the cost of power from HNPF will be comparable to the cost that CPPD anticipated from its yardstick conventional plant operating at 1800 psig.

Since HNPF is integrated into a large system, it can be base-loaded provided the fuel costs are equal to or less than older plants connected to the statewide system.

We anticipate that we shall be able to sell economically energy produced by HNPF and thus be able to operate the reactor at high loads, thereby accumulating operation data that will be helpful in determining (1) whether or not a reactor is an economical and reliable heat source for utility service and (2) data for guidance in the design of other reactors.

In closing, as a guide to those of you who are planning to tour Sheldon Station and HNPF, the plot plan of the site is attached as Fig. 6.

Liquid-metal Reactor Developments

R. W. DICKINSON
Atoms International*

The sodium-cooled-reactor program at Atomics International (AI) is an integrated effort, covering core design, reactor configurations, system development, component engineering and testing, and experimental operation of a 20-Mw(t) sodium-graphite reactor experiment that generates about 6 Mw(e) in a turbogenerator station supplied and operated by the Southern California Edison Company. (See above figure.) As an essential part of the sodium-graphite reactor program, development of fuels uniquely adaptable to the requirements of high-temperature sodium-cooled reactors and their incorporation into fuel elements suitable for the reactor environment is also proceeding.

An important, although relatively small, segment of program effort is devoted to establishing and continually reexamining sodium-cooled systems in order to assure that current reactor parameters are both realistic and optimum and that component development projects in progress are in fact justified. These studies also identify areas of the entire system that require additional development to decrease the cost, to increase the reliability of the over-all reactor power plant, and to guide these areas of improvement by setting and reexamining required

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*This paper reports work performed for the U. S. Atomic Energy Commission.

conditions of material strength, flow, pressure drop, and temperature as applicable to the particular developmental area.

These studies have converged on a 360 Mw(e) system as a referenced design, producing 2400-psi 1050°F steam in a single reheat cycle with power-generation costs currently predicted in the range of 6 to 6½ mills/kw-hr. While this reference plant is, of course, being continually reexamined in the light of successful development efforts and improvements in those factors which affect plant and fuel-cycle economics, the conceptual design is maintained up to date in order to provide equitable comparison with competing reactor systems. Changes in system layouts, elimination of system components when technically justified, and review of core configuration as development proceeds are incorporated to determine the economic rewards associated with advanced concepts. Optimization of fuel-cycle costs through fuel management studies are carried out in order to provide the lowest electrical-power cost. Long-range studies incorporate methods of improving theoretical methods of core calculation. Critical masses of unfamiliar configurations, not backed by a succession of power reactors and critical experiments from which empirical information may be derived by short extrapolations, may err by 50 per cent or more. Applications of Monte Carlo techniques to intracell flux distribution in multirod heterogeneous geometries have reduced this error significantly and show promise of further reduction to the order of 1 per cent. To supplement theoretical studies and confirm their accuracy, a graphite-moderated critical assembly of great experimental flexibility has been constructed. An experimental program to derive basic physics parameters of complex, low-enrichment lattices has been in progress for over a year, contributing significant information to an enrichment region not previously covered. This region is in the range of 2 to 4 per cent U²³⁵. Dynamic experiments to determine effects not easily calculated, such as control-rod interaction and shadowing, effect of small geometry changes, and effective control-rod worth can also be measured. The assembly can be heated to 400°F to obtain early trends of temperature-dependent reactor-physics parameters.

Several engineering studies not specifically directed to a particular reactor design are in progress. The effects of thermal fatigue on materials are of great interest to sodium-cooled-reactor designers owing to the rapidity with which thermal stresses can be introduced into construction materials by liquid-metal heat transfer. Studies to date have shown that carburization to depths of 1 to 2 mils does not have a significant effect; data on long-term effects on austenitic and low-alloy ferritic steels are being acquired.

Decarburization and carburization is being studied; it has been demonstrated that high-temperature sodium apparently removes carbon from unstabilized ferritic steels up to 5 per cent chromium content and promotes absorption of carbon by the high nickel-bearing austenitic alloys, which must occasionally be used in the highest temperature zones of a reactor.

It is apparent that the engineering properties of materials in the high-temperature sodium environment must be well understood in order to properly select materials and operating conditions for satisfactory long-term life. Thermal stress and its prevention are perhaps more important in the sodium system than in any other coolant system. The same superb heat-transfer properties that make sodium valuable as a high-temperature high-power density coolant enhances its susceptibility to rapid temperature changes in structural materials. Techniques of dealing with normal operating temperature changes are well developed; it is the abnormal casualty conditions that can be postulated although seldom realized, which may impose temperature transients up to hundreds of degrees per minute. Theoretical predictions often are inadequate to deal with complicated structures under these conditions. Tests of full-scale equipment, carefully instrumented for temperature and strain, are often the only reliable way of determining weak spots in a design, from both short- and long-term-life standpoints. This technique has been successfully employed in tests of major core components for the Hallam reactor; now that these tests have been completed, the large equipment available at AI is being employed for experimental tests of advanced reactor core and system components.

All reactor systems are necessarily made up of a variety of devices. The coolant is a fluid; therefore pumps, valves, heat exchangers, purification systems, and the sub-components that make up these major devices must give assurance of long-life reliability and have a rea-

sonable cost. Our philosophy has been to perform full-scale tests wherever possible. Sodium pumps have passed through several generations of development commencing with the shaft freeze-sealed pumps employed in the SRE and early test loops. This type of pump, although relatively inexpensive and completely serviceable for test operations, has proved disappointing in its reliability for reactor operation. Consequently, a free-surface type pump, originally developed under the sponsorship of Argonne National Laboratory, is being employed in Hallam and advanced designs. A full-size unit of the Hallam type has operated very successfully at 1100°F with a flow rate up to 9000 gal/min. Electromagnetic pumps are also available, but they are inherently less efficient than mechanical pumps. In certain applications, they are quite valuable; but they are not susceptible to easy repair when employed in radioactive service. A development seeking to exploit the reliability of a mechanical pump with the hermetically sealed features of the electromagnetic pumps, plus offering the possibility of direct-contact maintenance of all working parts commonly subject to failure, has been tested. Refinement to this design, in which a magnetic field progresses mechanically with flow control effected by rheostat variation of a d-c field, is now being completed based on operational results of the first prototype.

Sodium valves have been expensive because of zero stem leakage and rigid seat leakage requirements. Traditionally an expensive bellows has been employed for stem sealing and hardened materials and complicated sealing designs have been used in the seat. The true function of most valves is casualty control; something substantially less than complete sealing may be utilized for immediate control purposes. Frozen sodium can be employed as a complete seal in the rare occasions when absolute leak tightness is required. Valve patterns based on this principle have been developed and show a substantial cost improvement over earlier types. Stem-sealing problems associated with bellows failure have been overcome by the use of stem freeze seals cooled by a secondary fluid in high ambient temperature areas and by ambient air in open areas. This scheme makes possible shaft sealing of very large valves, the stem travel of which makes bellows sealing almost infeasible. Freeze stem seals are well understood and are able to function very satisfactorily in sodium temperatures up to 1200°F.

Sodium, an elemental fluid, is not subject to breakdown by irradiation and does not easily dissolve more than minute quantities of materials commonly present in a reactor system. Purification is, however, necessary to remove common contaminants such as oxide, which may plug small passages; carbon, which may carburize thin high-temperature steel members; and fission products, which may be introduced into the primary system through cladding failures. Oxygen contamination can be reduced simply by cold traps; a design that permits substantially unlimited capacity has been developed. This design utilizes the entire capacity of the fill and drain tank. The tank is used for its primary purpose infrequently and thus is available to act as a sink for oxides if properly outfitted. Carbon contamination has been reduced in experimental loops by hot traps offering selective carburization at a predetermined system position; a rather large practical application of this principle is now being made in the SRE, in which about 7500 gal of sodium are being cleaned of possibly 30 lb of carbon inadvertently introduced through an auxiliary coolant-supply-passage failure. Removal of fission products from sodium is also being investigated through techniques such as zone refining, activated charcoal trapping, and cold trapping. The efficacy of these schemes and their relative economics is still to be determined, although there is no doubt that fission-product removal by one or more of these schemes is practical.

Liquid-metal heat exchangers have been under experimental development for more than 10 years. The heat-transfer relations on the tube side are rather well defined. A constant source of uncertainty has been the heat-transfer coefficients associated with the shell side. A statistically designed program to more closely identify shell-side heat-transfer coefficients has just been initiated. It is expected that this program will reduce the uncertainty, which is now often 50 per cent to the order of 10 per cent in calculating liquid-metal heat-transfer surface. This reduction in uncertainty is expected to reduce the amount of heat-transfer surface incorporated to cover ignorance and consequently to reduce equipment cost.

A significant contribution to cost in high-temperature sodium systems has been the necessity for using highly alloyed steels to obtain high-temperature strength. Traditionally the 300

series austenitic type has been employed. As part of the integrated program to reduce sodium-system cost, a program was initiated in cooperation with the U. S. Steel Company to develop a low-alloy steel that would retain acceptable strength characteristics up to 1200°F. This program has achieved its short-term goals in devising a modified 2¼% chromium-molybdenum alloy steel whose properties up to 5000 hr appear completely acceptable and whose compatibility with high-temperature sodium appears excellent. Long-term creep-rupture tests of this material are underway to provide information for acceptance as a material of construction. By definition long-term data require a long time to obtain. The program has been in effect about three years.

Application of zirconium alloys to sodium-cooled systems is desirable for maximum neutron economy. The Zircaloy series, designed to provide corrosion resistance to water at the modest temperatures associated with water systems, have inadequate strength at the higher temperatures necessary to produce 1050°F steam from a sodium system. A search was instituted for a zirconium alloy that would be compatible with sodium and still retain its strength at temperatures of interest. An aluminum-bearing zirconium alloy has been developed which approaches the high-temperature characteristics of austenitic steel in all properties except creep. Design modifications have been incorporated to ameliorate the relatively high creep rate; these modifications have apparently been satisfactory. It should be noted that this zirconium alloy is not suitable for water systems owing to its very high corrosion rate in water. Its corrosion rate in sodium, however, is negligible.

The high sodium-system operating temperatures also demand a larger allowance for thermal expansion in vessels and process systems than in systems of lower temperature. Classically these expansions are accommodated in piping systems by expansion loops. When these loops must also be shielded, the expense associated with them is magnified many times. Bellows are an obvious solution. The history of bellows performance has, however, not been one to inspire confidence when used in radioactive-liquid-metal service. There was no apparent fundamental reason why bellows could not be used; it appeared that their rather complicated stress patterns were not well understood and they were thus often misapplied. An analytical and experimental program was launched to more perfectly understand bellows behavior from a theoretical and practical standpoint. We are now confident of our ability to utilize bellows in the primary system, providing proper analyses and test confirmation have preceded the application. Bellows of all sizes from 1 to 44 in. in diameter have been and are now undergoing tests, and capabilities of various bellows configurations offered by vendors have been established in many cases.

Safety aspects of sodium-cooled reactors have recently been accorded increasingly favorable attention. It was always well known that the low pressure of sodium systems eliminated the possibility of mechanical-energy release. The chemical compatibility of sodium with fuels and materials of construction similarly precluded chemical-energy release such as metal-water and some gas-metal reactions. It has lately become apparent that fission products are retained by sodium to a remarkable degree. Apparently the only fission products released in significant amounts from sodium are xenon and krypton. The halogens appear to be completely contained by combination with the sodium. Other fission products appear to be retained as alloys or chemical compounds. This unexpectedly high retention is most attractive from a safety standpoint. Carefully controlled experiments are now in progress to verify the degree of fission-product retention and the release of fission products from sodium should the sodium ignite in some unforeseen fashion. Preliminary results are most encouraging. It appears that an almost insignificant fraction of fission products are contained in the smoke of burning sodium. This rather exciting program will, of course, be pressed and reported as additional quantitative results are obtained.

Sodium-cooled-reactor fuels have proved in the past to present difficult technical problems. In order to properly utilize the heat-transfer qualities of sodium, a fuel capable of yielding a surface heat flux up to one million Btu/(hr)(sq ft) is required. This in turn implies a high thermal conductivity to avoid central temperatures that are literally astronomical. Uranium metal and its alloys have proved to be disappointing from a long-term burnup standpoint. Uranium oxide suffers from the poor thermal conductivity associated with its ceramic

nature. Consequently a radical innovation in fuel technology was required; this has been realized with uranium monocarbide. Uranium monocarbide, which has a uranium density approaching the metal and which has a thermal conductivity 14 times that of the oxide, appears to have the irradiation stability associated with UO_2 . Irradiations up to 25,000 Mwd/ton have been achieved with swellings on the order of 1 to 2 per cent; high-temperature irradiations in the 40,000 to 50,000 Mwd/ton class are now in test reactors. Their examinations are eagerly awaited. Studies designed to establish the ultimate performance of this material and its physical properties are proceeding in parallel with the irradiation experiments. Possibly the most important of these studies is devoted to determination of the degree of nonstoichiometry which may be accepted. Permissible deviation from the exact composition of uranium-4.8% carbon will have a significant effect on fuel fabrication costs; it is necessary to demonstrate the degree to which composition control may be relaxed. These studies are accompanied by experiments in ternary additions to either strengthen the uranium matrix in hypostoichiometric compositions or to prevent the formation of UC_2 in hyperstoichiometric compositions. These ternary additions will be useful in the event that economics shows they are desirable in relaxing composition control.

The program to reduce cost and increase the reliability of sodium-cooled reactor systems, in progress on a large scale for about four years, has thus produced important technical advances in sodium technology at temperatures up to 1050°F as current status. The continuing research efforts are rapidly bringing on the time when routine system operation at 1200°F, producing 1050°F steam, will be an everyday occurrence.

Table 1—STRESS-RUPTURE DATA ON 4S4 AND 4S8,* PSI

Alloy	1000 hr		10,000 hr	
	1100°F	1200°F	1100°F	1200°F
4S4	32,000	17,800	28,000	12,000
4S8	31,500	19,800	26,500	17,500
Type 304 stainless steel	22,600	14,400	17,000	9,700

*4S4 is 2.25% chromium-1% molybdenum-0.14% chromium-0.6% vanadium and 4S8 is 2.25% chromium-1% molybdenum-0.14% chromium-0.6% vanadium-0.1% niobium.

Fig. 1—Bellows test machine showing 44-in. bellows in test fixture prior to insulation. Tested with 1000°F sodium on interior of bellows. (Hallam IHY bellows.)

Fig. 2—Large component test loop. Thermal shock apparatus to test full-size core and system components. Shocks to 140°F/sec possible from 1200°F to 700°F.

Fig. 3— Pump test loops; 14 in. pipe operable up to 1150°F. Hallam pumps tested to 9000 gal/min. Steady state only; no facility for transient or gradient tests. Loops also used for flowmeter calibration.

Fig. 4— Control- and safety-rod test tower. Temperature gradient identical to reactor conditions can be maintained by "profile furnaces." Distance from base of top structure to furnace bottom is same as Hallam core-tank depth.

Fig. 5—Uranium carbide irradiated to 25,000 Mwd/ton. Surface markings caused by wire mesh "basket" containing slugs. Note satisfactory appearance and dimensional symmetry. 6X.

Status of Work on the Fuel Cycle at Argonne National Laboratory

S. LAWROSKI

Argonne National Laboratory

I shall attempt to quickly and briefly review the work being done at Argonne National Laboratory (ANL) on the fuel cycle, particularly as it pertains to sodium reactors of the fast breeder type. The work on the fuel cycle has consisted of fuel-development work, as well as that of fuel reprocessing. The work on fuel development has consisted in an emphasis on metallic fuels of the uranium-fissium or the uranium-plutonium-fissium types. We are particularly interested in the latter since, in the long run, we wish to use plutonium instead of U^{235} as the fuel.

Because of our desire to incorporate some versions of very rapid processes for recovery and purification of fuel, it has been necessary to include along with our fuel-development work the development of processes and equipment that lends itself to remote operation so that we can work with fuels that have been only partially decontaminated in the course of reprocessing. These factors have meant a need for close cooperation and liaison between the reactor engineering people, metallurgists, and chemical engineers. May it suffice to say that particularly in the case of the EBR-2 project we have had this cooperation and liaison, and from this has resulted, for example, the design of the fuel element that is shown in Fig. 1, which we believe can be remotely fabricated and can be reprocessed for adequate removal of undesirable fission products and still produce a fuel that will give high performance in reactor operation.

Such a subassembly as is shown in Fig. 1 will be used in the EBR-2. The active section of the subassembly is the portion in the center. On either end there is a blanket section consisting of natural uranium. Also shown is a magnified view of the fuel section, which consists of 91 fuel elements, each containing a fuel pin along with some sodium put into a 304 stainless-steel tube in which an upper restraining cap is added. A remotely made weld is made to enclose the fuel, which later, along with the sodium, is vibrated at a temperature of from 430°F for about $2\frac{1}{2}$ hr to arrange a good sodium bond between the uranium and the stainless steel.

Table 1 gives the composition that has resulted from a compromise between the metallurgists, the reactor engineers, and the chemical engineers. In this particular case, the fuel is 95 per cent uranium, of which 48.1 per cent is enriched U^{235} . The remaining 5 per cent is a group of what we call fissium elements, and it represents those fission products which the simple process of melt refining does not remove.

Work is now under way in which plutonium will be used to replace a part of the uranium, particularly the U^{235} . The amount of plutonium we expect to use is about 20 per cent. This percentage, we find, will require somewhat larger percentages of fissium elements in order to get a satisfactory fuel material from the standpoint of physical properties and radiation-damage behavior.

Figure 2 shows the equipment which has been developed and which we believe from the experience we have had with it in preparing the first loading for EBR-2 will be capable of be-

Fig. 1—EBR-2 fuel subassembly.

Table 1—COMPOSITION OF FISSIUM FOR
FIRST EBR-2 FUEL LOADING, %

Uranium*	95	Ruthenium	1.96
Zirconium	0.1	Rhodium	0.28
Niobium	0.01	Palladium	0.19
Molybdenum	2.46		

*Uranium is 48.1% enriched in U^{235} .

Fig. 2—Injection casting furnace for first EBR-2 core.

ing operated by remote control. This equipment is an injection casting furnace with the bell-jar cover removed. Shown alongside the furnace is a group of 160 Vycor molds that are precision made so that fuel in the course of being injected into the mold and frozen devises a very uniform dimension in diameter and usually also in length. Thus the only step required later to get suitable dimension fuels is simply to crop the declad, that is, crop the material removed from the Vycor molds to a length of 14.2 in.

Figure 3 is a photograph of equipment that has been developed for the remainder of the fuel-pin processing and inspection. This processing precedes the assembling of pins into the

Fig. 3—Fuel-pin processing and inspection equipment for first EBR-2 core.

stainless-steel can, which is later made into a subassembly. The equipment shown in this figure is for removing the pins and breaking away the Vycor, for getting the pins cut to length, and for the inspection in which it is determined whether or not the fuel has the proper weight, length, diameter, and density and whether or not any voids are present.

Figure 4 shows a piece of equipment that we have found can be operated remotely through the use of the master-slave manipulators also shown in the figure to put together 91 of the fuel pins to make up the subassembly that was shown in the lower part of the Fig. 1.

In the course of manufacturing the first core loading for EBR-2, we have learned that our experience has been reasonably satisfactory; and we feel that only with additional modifications of some of this equipment it can be satisfactorily remotely operated, as will be necessary in the plant that is under construction in Idaho. I shall show some pictures of that plant later.

In regard to the development of the fissium alloys, which consists in a wide variety of material, the particular one we have chosen for the EBR-2 loading is shown in Fig. 5. It has

Fig. 4—Remotely controlled fuel-element assembly equipment for first EBR-2 core.

5 wt.% fissium. In this case the fuel pin has been subjected to a burnup of 1 at.%. This value may also be expressed as 10,000 Mwd/ton. This is a reasonably good looking pin, taking into consideration that it has been irradiated for this burnup at a maximum fuel temperature of 600°C. Other variations of fissium, also shown in Fig. 5, with the addition of 2½ per cent more zirconium also gives a fairly good looking pin.

Our problem then with cast uranium-fissium has become one of obtaining a better cladding material, something that can restrain the growth of the pin when the fuel is subjected to larger burnups than 1 at.% and to higher central metal temperatures.

Composition, wt.% (Balance is uranium)	3.7 Fs	3.3 Fs-1.7 Zr	5 Fs	5 Fs-2.5 Zr	6.7 Fs-3.3 Zr	10 Fs-5
Maximum irradiation temperature, °C	600	600	600	630	680	680
Burnup, at. %	1.3	0.6	1.0	1.0	1.2	1.0
%Δ vol./at. % burnup	3.84	7.93	2.30	3.00	2.98	4.90

Fig. 5 — Irradiation behavior of cast uranium-fissium alloys.

I might say that in the case of EBR-2 the design of central metal temperature is for temperatures on the order of a maximum of 1200°F with a fuel-surface temperature of approximately 1000°F.

The previous speaker has mentioned Atomics International is interested in fuels other than metallic ones. We, too, have been interested in the development of fuels that might permit higher operating temperatures. For the same reasons that he mentioned, the higher thermal conductivity and the high density of materials such as uranium carbide and plutonium carbide systems have received attention in our developmental work. We are very anxious to get over to the use of plutonium and a fast reactor fuel.

Up to the present time, people have used the classical ways of making uranium or plutonium carbides. These ways usually involve three different types of reactions: (1) reacting UO_2 with carbon at some temperature around 1400 to 1800°C, (2) reacting hydrided uranium or plutonium with carbon, and (3) reacting uranium-plutonium metal with carbon.

Table 2 — PREPARATION OF URANIUM
AND PLUTONIUM CARBIDES

$\text{U}(\text{Zn} - 10 \text{ wt.}\% \text{ Mg}) + \text{C} \xrightarrow{800^\circ\text{C}} \text{UC} \downarrow + (\text{Mg}, \text{Zn})$
$\text{Pu}(\text{Mg}) + \text{C} \xrightarrow{800^\circ\text{C}} \text{Pu}_2\text{C}_3 \downarrow + \text{Mg}$
Mg and Zn subsequently vaporized at 600 to 750°C and 0.1 to 4 mm Hg

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We have recently discovered what appears to be a new way of making these carbides. (See Table 2.) This method requires somewhat lower temperatures. Although the work is only in the preliminary stage, we have had some very good success in the case of making uranium monocarbides. The particular method with which we have been working is one in which zinc with approximately 10 wt.% magnesium has added to it about 15 to 20 wt.% uranium, making a metal solvent. This solvent is subsequently treated with activated carbon at a temperature of about 800°C. This is roughly 1000°C less than some of the present methods in use. We have obtained what is shown to be under X-ray examination pure uranium monocarbide. The reaction takes place very rapidly in this liquid metal solvent. We simply remove the metal solvent by a fractional distillation and a retorting. We very recently arranged to have a metallurgist take some of the resulting powder that we obtained after retorting and arc-melt it. This material had a density of approximately 13.6, very near that for the theoretical value of crystalline uranium monocarbide.

We have also tried to make plutonium carbide. However, for this case we changed solvents and used magnesium, which is a good solvent for plutonium; and, under approximately the same temperature conditions as we used for the uranium reaction, we reacted this with activated carbon. In our first preparation we found, at least to our tentative disappointment, that instead of obtaining something corresponding to PuC we obtained Pu_2C_3 . We have made only this one preparation, but we believe that the results are quite reproduceable because in the use of cerium in an earlier experiment, we found that we obtained Ce_2C_3 . Although these procedures are very preliminary, we feel that they may constitute a very promising way of making these carbides, at least uranium monocarbide. Perhaps by a choice of solvents other than magnesium we might make compositions of plutonium carbide other than Pu_2C_3 . These results have suggested to us high-temperature processing methods if carbide type fuels are used in sodium reactors instead of metallic fuels.

It was earlier mentioned that cladding materials with high-temperature stress were being studied. The influence of cladding on the restraint of growth is shown in Fig. 6. There has resulted a 26 vol.% for each atomic per cent burnup. However, in the case of a similar fuel pin having previously been encased in a zirconium foil and then put inside a 10-mil-thick stainless-steel can, there has been virtually a negligible growth even though the central fuel temperature has been 960°C and the burnup was also somewhat higher than on the unclad can. Even at a temperature of 1100°C, the cladding still restrained the change in volume even at a 2.1 at.% per cent burnup.

Fuel center temperature, °C	620	960	1100
Clad surface temperature, °C	unclad	480	590
Average burnup, at. %	1.4	1.8	2.1
%Δ vol./at. % burnup	26	0	0

Fig. 6—Effect of 0.010-in.-thick stainless-steel cladding in restraint of the irradiation behavior of uranium-2 wt. % zirconium alloy.

This sort of information illustrates the reason for concurrent work at ANL to develop cladding materials that will be compatible in the environment encountered with the fuel in the fast reactor, namely, compatibility with sodium and with a particular fuel composition at high temperatures. Such cladding materials must have sufficiently high strength at high temperature to permit restraint of growth of the fuel upon irradiation at fairly substantial temperatures.

Figure 7 shows some of the results which have been obtained in experiments to determine the potential clad materials that might be of use for, say, temperatures on the order of 600 to 700°C on the fuel surface. In this figure is shown the time in number of days required to get a penetration of 1 mil when a couple consisting of a fuel composition of uranium, 20 wt. % plutonium and 10 wt. % fissionium is put alongside titanium, molybdenum, and tantalum, which are some of the more promising high-strength materials for high-temperature operation. You will note that this is a long scale. A number of materials are at a temperature of 600°C, for example, and do not show a diffusion penetration of as much as 1 mil in that time. You will note stainless steel gets at least a penetration of 1 mil even at 550°C after only 10 days.

Figure 8 shows diffusion penetrations in relation to the time required to have a 9-mil penetration. We use the numeral 9 here rather arbitrarily. It perhaps is suggested by the 10-mil-thick stainless-steel cladding that we are using in the EBR-2; but we must note that the available data are by no means sufficient to now select the materials that might be ade-

Fig. 7—A 1-mil diffusion penetration of potential clad materials by uranium-20 wt.% plutonium-10 wt.% fissium.

Fig. 8—A 9-mil diffusion penetration of potential clad materials by uranium-20 wt.% plutonium-10 wt.% fissium.

Fig. 9—EBR-2 fuel cycle.

quate for developing techniques by which to fabricate tubing of suitable dimensions for cladding purposes. There are a number that look fairly promising. Molybdenum, vanadium, and niobium show fairly good possibilities. Of course, the big problem now is to find out whether or not we or somebody else can economically fabricate tubing of some of these materials that might be promising to clad fuel.

During the remainder of my talk, I shall discuss the development work and studies on the pyrometallurgical processes that we feel have considerable promise for at least fast-reactor fuel-reprocessing needs and possibly later for other types of reactors. These methods are characterized by the fact that they generally are high-temperature methods and that there is usually a minimum of chemical conversion involved during the course of purification of the fuel. Some of the economic factors supporting pyrometallurgical processes are reduction of fuel inventory, minimization of chemical conversion tests, recovery of expensive alloying agents, production of concentrated fission-product wastes, and abatement of critical-mass problems. Also they are suitable for small compact plants at reactor sites. Because of the fact that one can commence processing after about 15 days of cooling an opportunity is presented to reduce the amount of fuel in inventory. As I have already mentioned, the methods are characterized by minimization.

We do retain, as is illustrated in the case of the melt-refining process, which is a version of pyrometallurgy, some of the alloying agents; that is, we do not necessarily lose them as is the case in aqueous methods of fuel repurification.

Because of the absence of moderating materials or reflecting materials we have somewhat less of a critical-mass problem in these processes. These processes are applicable to very compact plants, perhaps so sufficiently that one can consider sizes that would be economic for single-reactor station sites.

Figure 9 is a scheme which we have evolved and which will be put into operation for the recycle of fuel from EBR-2. After some 15 days we shall arrange to declad the fuel and put it through a melt-refining process and following this through a remote fuel-element fabrication. The finished fuel subassemblies can be returned to the reactor for irradiation. We shall, in order to prevent continual buildup of some of those fission elements, subject a portion of the

fuel through what we call a "dragout" process so that those things such as zirconium, niobium, rhodium, palladium, and ruthenium do not build up indefinitely and interfere with the reactivity of the fuel.

Fig. 10—Melt-refining furnace with bell-jar cover removed.

In Fig. 10 is shown the piece of equipment that conducts the melt-refining step. This photograph depicts the process of the pouring operation. Normally, a bell jar with a fusible seal is used for this operation. To obtain purification of the sort indicated in Table 3, we would put in approximately 10 kg of cut-up irradiated fuel that had been discharged from the EBR-2 and then cooled for about two weeks and for about 3 hr hold at a temperature of about 1400°C in a zirconia crucible.

Table 3—FISSION-PRODUCT REMOVALS EFFECTED
BY MELT REFINING IRRADIATED URANIUM-FISSIUM ALLOY

Element	Percentage removed	Mechanism of removal
Rare earths plus yttrium	99.4	Selective oxidation
Barium and strontium	>99.9	Selective oxidation
Cesium	>99	Volatilization
Tellurium	95.4	Drossing reaction
Iodine	>99.9	Volatilization
Xenon and krypton	>99	Gas release
Zirconium	9	?
Ruthenium, molybdenum, palladium, and rhodium	0	

Charge: 0.4 kg of 5% fissium irradiated to 0.6% burnup.

Conditions: liquation for 3 hr at 1400°C in a zirconia crucible.

Table 3 indicates some results obtained very recently. We have been waiting quite some time to get fissium fuel irradiated to burnups to at least a reasonable fraction of that anticipated in the EBR-2. In this case we have irradiated some fissium to $\frac{1}{6}$ at.% burnup. We have done some work up to a little over 1 at.% burnup. For a melt-refining operation of 3 hr at a 1400°C in a zirconia crucible, we have effected the removal of a rather large number of the fission products, leaving essentially unremoved five elements.

Fig 11—Individual solubilities of uranium and noble-metal fission-product elements in zinc as a function of temperature

Fig 12—Effect of magnesium on the solubility of uranium in liquid zinc and cadmium

The mechanisms of removal are indicated in this table. These removal mechanisms have been very satisfactory in that they confirmed very well our experience of the last three years with only tracer radiated material, which we had available, and the use of synthetic inactive mixtures of fission. Experience now has been gained with something essentially closer to the real case, and we are still obtaining very gratifying results.

In connection with the dragout process that would be used to remove those things not handled in the melt refining and to permit reprocessing of the blanket material with a version of pyrometallurgy, we have taken advantage of the fact that there are very large differences in the solubility of the fission-product elements and uranium and plutonium in such solvents as zinc. Figure 11 represents only a very small fraction of the data we have obtained. We now have very extensive data for solubilities of various fertile and fissionable materials and fission products in zinc and cadmium, as well as in zinc and cadmium to which magnesium has been added for reasons that I shall mention in just a moment.

Fig 13—Liquid-metal process for reclamation of melt-refining skulls

If plutonium had been included in this figure, it would have shown a slightly higher solubility than uranium. Zinc has a fairly large liquid range so that we are able to perform liquid fractional crystallizations at different temperatures to take advantage, not only of the difference in solubility, but also the fact that the fission-product elements are present only in very small fractions compared to that of uranium.

We have learned that by the addition of a certain amount of magnesium to either zinc or cadmium that one can very substantially increase the solubility of uranium as well as plutonium.

Figure 12 illustrates, at the temperatures of 676°C for zinc and 448°C for cadmium, the somewhat larger solubility that one obtains, at least after the indicated amount of magnesium has been added compared to the case in which no magnesium is added. The values are given in atomic percentage. To obtain the result in weight percentage, one should multiply by about

four. At somewhat higher temperatures there is a somewhat more pronounced effect of magnesium. However, as the addition of magnesium is continued, there is a very precipitous and rapid decrease in solubility. In order to get uranium to precipitate out, say, to permit a separation between uranium and plutonium, we merely just continue to add further magnesium and then precipitate uranium out directly rather than as an intermetallic of U_2Zn_{17} .

Fig. 14—Liquid-metal process for uranium blanket metal.

From this method has evolved the suggested flow sheet given as Fig. 13 and another very similar one for blanket processing, which is given as Fig. 14. Figure 13 shows the dragout process or reclamation of melt-refining skulls and any dregs from the fuel-fabrication operation. We simply take the skull from the melt-refining output, oxidize it right into the zirconia crucible with a dilute oxygen and argon gas mixture at something like 700°C. This permits the material to be reduced into a form that permits the residue to be poured out in the crucible and enables the leaching out of the noble metals in zinc. The use of zinc allows us to take almost all the noble metals out in zinc, and the flux then bears the uranium oxide and those fission products whose oxides have not been reduced by zinc. Later, by the addition of magnesium, we arranged to reduce all the oxides, including those of uranium and plutonium. Some of the steps in the procedure can be eliminated, but the important step is one in which, by the use of additional magnesium, we can precipitate uranium, with the magnesium rate supernate bearing the fission products that were not removed with the zinc.

Figure 14 shows the liquid-metal process for the uranium blanket, which is almost completely developed. It is a very direct scheme for making the separation of uranium and plutonium from blanket materials to obtain a plutonium that might be used for recycling into a core.

There has been one change made in this flow sheet since the described preparation was effected. In the first step, instead of using pure zinc and cadmium, we are now incorporating the addition of magnesium to take advantage of the fact that a larger solubility of fuel and a more rapid solubility of irradiated blank material can be effected by the addition of the magnesium at this step. Also an arrangement is made for the later precipitation of uranium by any one of the three methods given in Fig. 14. The results of further additions of magnesium are shown in the figure, giving the effect of magnesium on zinc or cadmium solubilities for the uranium.

The use of excessive magnesium on zinc and cadmium results in a solvent that has fairly high solubilities for plutonium. It is indeed a very good solvent for plutonium, but it has very little solubility for uranium.

Fig. 15 — Pyrometallurgical reprocessing of uranium carbide fuel.

The fact that the uranium is precipitated through the use of excess magnesium allows uranium to be recovered, which either may be discarded or used in the retorting and reuse recycle. By simply removing the magnesium, one obtains a plutonium-metal product. We have taken material obtained in the use of all these methods and consolidated it simply by heating it a very short time to about the melting point of the materials.

Figure 15 is a very tentative flow sheet for what we think might be a pyrometallurgical method of reprocessing uranium carbide fuel. It takes advantage of the information that I noted earlier, that we would perform a carbide synthesis simply by the addition of something such as activated charcoal or methane, which might be composed at about $800^\circ C$.

We are now approaching the point where we are going to find out how well our thoughts on these versions of the fuel-cycle incorporating the remote fuel fabrication together with a pyrometallurgical repurification process will work. A fuel-recycle plant is under construction in Idaho. Figure 16 is a picture showing a cutaway of the fuel-recycle plant whose construction should be completed by Aug. 1, 1961. The reactor and an air cell where we plan to disassemble the fuel up to the point that the uranium is almost exposed are shown. At the point it is brought into this pure argon cell, it is in individual fuel pins. Through the use of manipulators and cranes located in this cell we plan to perform the remainder of the fuel decladding,

the purification, and then the remote refabrication of the fuel pins, cladding, etc., and then the reassembly into a subassembly.

Figure 17 is a plan view of this plant showing that the working annulus has a 16-ft width and that there are windows through which we shall make our observations for the manipulations involved in the air cell.

Fig. 16 — Argonne National Laboratory EBR-2 fuel-cycle facility.

Figure 18 shows a section view of a central control room, the annulus in which the equipment is located, and the manipulators. This equipment is powered from the center. The plant, as I said, is nearing completion in relation to its construction; and we hope about sometime in July to take over the plant and begin installation of the reprocessing equipment for melt refining and the remote fuel fabrication equipment and, perhaps, finish the shakedown of both the processing and fabrication equipment one year from this coming fall, in other words, about October 1962. This will permit processing and recycling of irradiated EBR-2 fuel.

We shall later add equipment for demonstrating the blanket process outlined on the flow sheet previously discussed and for performing the dragout process operations. Later we may wish to test other versions of processes that might evolve from further development work that we have in process at ANL and work that might evolve from other sites.

Fig. 17 — Main floor of EBR-2 process building.

Fig. 18 — EBR-2 process building. Section view of argon cell.

Development Work for the Enrico Fermi Atomic Power Plant

A. AMOROSI

Atomic Power Development Associates, Inc.

Experience with sodium as a coolant is relatively new. With the advent of atomic energy, it was recognized as an ideal coolant for high-power-density reactors. The initial applications were for the reactor for the submarine *Seawolf* and for the EBR-1. Since that time the use of sodium, as well as lithium, has received more extensive application in atomic power. These metals are now being considered as heat-transfer mediums for nuclear applications, such as for mobile power plants, space auxiliary power sources, nuclear rockets, and central-station power plants. Their characteristics and potential have already been discussed by Dr. Glen Wensch and others. I shall limit my presentation primarily to Atomic Power Development Associates, Inc. (APDA), experience with sodium as it applies to the Enrico Fermi reactor.

Test Facility Operation

The primary coolant system of the Enrico Fermi plant was filled with sodium on Dec. 1, 1960, and since that date the components of the system have been operating with excellent results. From 9:30 p.m. on November 30 to 6:30 a.m. on December 1, a total of 9 hr, 345,000 lb of sodium at 400°F were transferred from the storage tanks into the primary system by evacuating the primary system and by applying an inert-gas (argon) pressure to the storage tanks. The oxygen content before entering the primary system was 12 ppm. The plugging temperature was 230°F. The system was slowly heated to 500°F in a two-week period, during which time the sodium was not circulated or cold-trapped. At the end of that time, the plugging temperature was 350°F. The sodium was cold-trapped to a plugging temperature of 220°F in one day, which corresponds to the small amount of oxygen that was introduced during the filling procedure. The temperature of the system was held to 500°F until May 1, 1961. No change in the oxygen content was noted during this time. However, the cold-trap had to be operated when taking plugging readings so that some purification may have taken place. During this time the fuel-handling equipment was calibrated, and the pump tests were run. From May 1 to May 8, the system was heated to 1000°F and was above 800°F for four days, which is long enough to assume wetting of all surfaces. Since the plugging temperature only changed to approximately 231°F, the form of the stainless-steel oxide probably changed very little, and it would appear that there was little hydrogen in the system. The plot of plugging temperature as a function of time is shown in Fig. 1.

The core section was loaded with dummy subassemblies containing filters. An examination of representative filter elements removed from the system indicated a total of solid impurities or foreign matter of only about 2 lb. The material taken from one filter element is shown in

Fig. 1—System and plugging temperatures in the primary system of the Fermi reactor during test facility operation.

Fig. 2—Material taken from filter element.

Fig. 2. The nonmagnetic material is representative of materials of construction and the magnetic material of tools of construction. The light foreign matter that found its way to the free surface of the sodium in the reactor tank in the initial fill was skimmed off through the overflow pipe. A picture of the sodium surface in the reactor taken through the 6-in. hole in the plug is shown in Fig. 3.

Fig. 3—Three access ports located near the center of the rotating shield plug above the reactor are shown in the photograph at the left. A view looking down into the glass-covered port is shown at the right. Note the clean, bright, silvery surface of the sodium in the reactor vessel below the plug. No difficulty has been experienced in maintaining the high purity, as evidenced by the photograph, of the sodium during operation. The nearly semicircular dark area in the bottom center of the photograph at the right is the light source used to make the picture.

On May 9, after the system had been running above 1000°F for one day, a sharp increase in the plugging temperature was noted. At the same time carbon dioxide was noticed in the argon cover gas. (See Table 1.) It was found that this carbon dioxide came from the canned but vented graphite located in the argon-gas space of the rotating plug. Since carbon dioxide reacts with

Table 1

Date	System temp., °F	Plugging temp., °F	% CO ₂ in cover gas, %
May 9	1000	231	
May 10	1000	555	
May 11	995	418	15.4
May 12	1000	370	10.6
May 13	1000	300	4.4
May 14	1000	295	3.5
May 16	1000	312	2.5
May 18	800	240	1.0
May 20	800	<220	<0.1

sodium, forming sodium carbonate and carbon, the cover gas was purged to sweep out the carbon dioxide. Both the sodium carbonate and carbon are soluble in sodium, although to a lower concentration than sodium oxide. These impurities were cold-trapped from the system in about a week. Some of the material was removed from the sodium surface by shimming out the overflow. While these impurities were present in the system, care was taken to limit the operation

of equipment to prevent damage. Since these impurities were removed, the safety rods and drives, the fuel-handling mechanism; and the hold-down mechanism have been operating satisfactorily.

From the results of operation, it is concluded that clean systems can be built and can be maintained with proper precautions. Because sodium does not form compounds with common metals and because the solubility of common metals is very small in sodium at operating temperatures, virtually no corrosion or material transport takes place if proper sodium purity is maintained.

Fig. 4— Comparison of unused subassembly to one vacuum distilled cleaned at 950°F after exposure to 500°F sodium for three months. The unused subassembly is on the left.

Vacuum Distillation

Those of you who have had to clean sodium from large components without changing the character of the surface will recognize the difficulty of doing so with water or steam cleaning or by burning and brushing. Although sodium boils at 1608°F, it is rather interesting that it still has a 1-mm partial pressure at 825°F. This fact makes it relatively easy to remove sodium by boiling at temperatures that ordinary materials can withstand. The parts cleaned by this method look almost as bright as when they were placed into the sodium. This attests to the merit of the cleaning process and to the complete freedom from corrosion in a high-purity sodium system. Figure 4 shows a comparison of an unused dummy fuel subassembly and one cleaned by vacuum distillation.

Fig. 5 — Once-through steam generator.

Steam Generators

Mr. R. W. Hartwell showed you an installed view of the steam generators for the Enrico Fermi plant. They are once-through single-walled-tube units. Figure 5 shows a cross section and a model of these units. The three units cost \$1,300,000, which is equivalent to \$8/kw(e) based on a plant output of 150 Mw(e). This type of economics is brought about by several factors. The use of sodium provides high heat-transfer rates, which minimizes the total surface needed. The freedom from tube burnout and the use of low-pressure sodium on the shell side are very favorable to a once-through steam generator. The use of single-walled tubes reduces the cost one-half compared to double-walled tubes.

Because of the choice of a single-walled tube, the steam generator had to be designed for a full rupture of a tube and the consequent sodium-water reaction. We recognized that, regardless of how rapid the reaction of sodium with water is, it is still a surface reaction and is

Fig. 6 — Equipment for sodium-water reaction test.

Fig. 7 — Total deflection between tubes 1 and 2 after three runs in sodium-water reaction tests.

Fig. 8—Test equipment.

Fig. 9—Release of fission products from the relief diaphragm.

limited by the ability of the reactants to get together. Tests were run simulating the rupture of a $1\frac{3}{16}$ -in.-OD tube, allowing high-pressure water to run into the sodium as shown in Fig. 6. On triggering a valve the rate of flow of the water was as high as 300 gal/min into the sodium. The tests were conducted to establish the relief system required to vent the reaction products without generating high pressure on the sodium side. They were also conducted to determine whether failure of one tube might cause failure of adjacent tubes. After running three tests on the same bundle, the tubes were in good condition and bowed only $\frac{1}{2}$ in. The amount of distortion of the tubes is shown in Fig. 7. Pictures of the test equipment and of a test are shown in Figs. 8 and 9, respectively. The latter shows the release of reaction products from the relief diaphragm. The relief diaphragms were established at 24 in. in diameter for the Enrico Fermi plant.

Fuel Development

The fuel for the first loading of the Enrico Fermi reactor is uranium-10% molybdenum, the same alloy as that used in the first loading for the Hallam reactor. These elements were described in the Fast Reactor Information Meeting held in Chicago on Nov. 20 and 21, 1957, and were discussed earlier by Mr. Hartwell. The second loading will be cermet elements of UO_2 in a stainless-steel matrix. This type of element has had extensive use in high-power density reactors. It is fortunate that a large amount of structural steel can be used in a fast reactor because the use of this steel permits high burnup to be achieved. Because there is no fertile material in the core, the conversion ratio with U^{235} fuel is only 0.8, which is still as good as any solid-fueled thermal reactor. The fuel is a good fuel to operate any fast reactor and will be used until a more economical fuel with a high breeding ratio is developed.

The fuel-cycle cost for this type of fuel element is competitive with fuel-cycle costs for other types of reactor fuel elements, mainly because of the high achievable burnup for a cermet-type fuel element and because plate elements are less expensive than pin elements to fabricate. The fuel-cycle costs for the Enrico Fermi plant are given in Table 2.

Table 2—CORE B FUEL-CYCLE COSTS,
300 Mw AND 75 PER CENT PLANT FACTOR

	\$30/g of Pu		\$12/g of Pu	
	UO_2	PuO_2^*	UO_2	PuO_2^*
Cost, mills/kw-hr				
Fabrication				
Core	1.2	1.5	1.2	1.5
Blanket	0.5	0.5	0.5	0.5
U-235 destroyed	2.4	-	2.4	-
Reprocessing				
Core	0.3	0.2	0.3	0.2
Blanket	0.4	0.4	0.4	0.4
Conversion ($\text{UNH} \rightarrow \text{UF}_6$)	<0.1	-	<0.1	-
Reduction of Pu to metal	0.2	<0.1	0.2	<0.1
Losses (1.3% U + 2.0% Pu)	0.2	0.1	0.1	<0.1
Shipping of irradiated fuel				
Core	0.1	0.1	0.1	0.1
Blanket	0.1	0.1	0.1	0.1
Total	5.4	2.9	5.3	2.8
Plutonium revenue	3.4	0.3	1.4	0.1
Net fuel-cycle cost	2.0	2.6	3.9	2.7
Use charge, at 4.0%	1.09	1.3	1.0	0.5
Net fuel-cycle cost with use charge	3.0	3.9	4.9	3.2

* PuO_2 is not contemplated for the Enrico Fermi plant. It is listed only to show the relative economics of plutonium vs. U^{235} .

Fatigue Tests of Sine-wave Tubes

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Alco Products, Inc.

To begin this report, I should like to give acknowledgment to Professor M. F. Sayre, who did the theoretical stress work included in it.

In the design of intermediate heat exchangers and steam generators for high-temperature sodium service, there are many unique problems that confront the design engineer. Of no small consequence is the task of performing a detailed stress analysis that incorporates thermal transients as a result of large temperature changes, as well as the temperature gradients experienced in the normal, steady-state operating conditions.

Before stress analysis begins, the heat exchangers are sized according to the required heat-transfer surface area and flow conditions. After the thermal requirements have been established, the exchangers are designed according to the ASME Boiler and Pressure Vessel Code, TEMA Standards, and/or other suitable codes. Upon completion of the mechanical design, which has established basic sizes and dimensions in accordance with these codes, the units are ready for a detailed stress analysis. Since the above codes do not consider thermal stresses, the U. S. Navy code, Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components, dated Apr. 1, 1958, may be selected as a method for handling stress analysis of combined mechanical, hydrostatic, and thermal stresses and for determining conditions for using creep- and stress-rupture properties. In the Navy code allowable values for thermal stress and stress concentrations are given only up to 700°F. Current designs include sodium temperatures up to 1200°F, a factor which necessitates an extrapolation of the Navy code in order to establish allowable values for these higher temperatures. With little or no test data available above 700°F, the allowable values for thermal stress, plus stress concentration, can be assumed to vary with the ultimate strength.

The Alco design for the intermediate heat exchanger and the steam generator incorporates a sine-wave tube to care for differential expansion between the tubes and shell. The tubes are prebent to a sine-wave shape, and the wave is given a second bend in order to fit or conform to the concentric-tube-row layout. This permits a nesting of the tubes in each tube row, thus keeping the shell diameter to a minimum and still maintaining maximum flexibility. As part of the work under this contract, a program was undertaken to study the mechanical behavior of the sine-wave tube as it was flexed plus and minus a predetermined distance from the neutral position. A check on the allowable values for thermal stress, based on assumed variation with ultimate strength, became possible as a by-product of this program.

The test rig, as shown in Fig. 1, was designed to simulate the desired differential expansion by supporting vertically a tube containing a sine-wave section and forcing the lower end up and down using an air piston. The air-piston travel was adjusted to vary the magnitude of flexure for the various tests conducted. The test tube was electrically heated and controlled by a thermocouple at the midpoint of the sine wave. The tube was pressurized with argon gas, which stopped the cycling when failure of the tube occurred.

All tubes tested were $\frac{1}{2}$ -in.-OD type 316 stainless steel. Tubes for the intermediate heat exchanger were 0.049-in. wall tubes, and tubes for the steam generator were 0.055-in. wall

Fig. 1—Sine-wave-tube test stand.

Fig. 2—Fatigue stress at failure. Δ, test results at room temperature. ○, test results at 1150°F. ×, test results at 1400°F.

tubes with Inconel linings. Because of the thicker wall tubes and the possibility of more extreme thermal transients in the steam generator, the sine wave was 60 in. long, as compared to a 48-in.-long sine wave for the intermediate heat exchanger.

Table 1 shows the results of the initial series of tests made at 1100 to 1200°F. These tests indicated that the length of the sine wave for the steam generator could be reduced from 60 to 48 in. since the maximum differential expansion between tube and shell under extreme thermal transients, which is not expected to exceed 25 times in the life of the unit, is not over 1 in. in either direction. These data further indicate the extreme flexibility of the sine-wave tube and the relatively large margin of safety, based on a maximum movement of 2 in. for 25 cycles.

Table 1—NUMBER OF CYCLES TO FAILURE

Stroke in inches	48-in wave, 9-in offset	60-in wave*, 11-in offset
1 5 to -1 5	18,422	
2 0 to -2 0	3,667	10,670
	4,999	17,301
	2,194	
2 5 to -2 5	1,037	8,976
3 0 to -3 0		4,446
		2,259
4 0 to -4 0		2,510

*Inconel-lined tubing.

Failure in all cases was of the typical fatigue type at right angles to the length of the tube, starting at the point of expected highest tensile stress, on the inside of the curve at slightly below the midpoint of the wave. Six of the tubes were Inconel lined, but this lining appeared to have no effect on their behavior.

Knowing the geometry of the sine wave and making allowance for the changes in geometry during the up and down stroke, equations were developed for the strain, in inches per inch, present in the extreme fibers of the tube at the critical section resulting from various up and down strokes. In all cases these exceeded the elastic limit of the material. The results could have been stated as strains, but as a matter of convenience the strains on the up and down strokes were averaged and the result multiplied by the modulus of elasticity to give a figure corresponding to the alternating stress intensity as defined by the Navy code. A log-log plot of alternating stress intensity against the number of cycles to failure is shown in Fig. 2.

On the basis of these preliminary results, additional work was undertaken to permit tests aimed specifically at the study of fatigue behavior of Type 316 stainless steel at various temperatures. Under this program, 20 specimens were tested at temperatures ranging from room temperature to 1400°F.

This program differed from the earlier one as follows:

1. The sine waves were given a curvature in one place only, the second cross-curvature being omitted in order to make the stress computations more accurate.
2. For a number of tubes, the internal pressure was raised from 100 to 1200 or 1700 psi in order to determine the possible effect of a continuing circumferential tension upon fatigue behavior under longitudinal stresses.
3. The normal speed of operation during test was 16 cycles/min. Two tests were run to gage the possible effect of slower speeds. One specimen was operated at 1100°F for 51 cycles at 1 cycle/hr and then speeded up until failure occurred. A second specimen was operated at 1400°F for 120 cycles at 2 cycles/hr and then speeded up until failure occurred. The first one showed slightly longer life and the second slightly shorter life than the mean for these temperatures.

Figure 2 is a log-log plot showing the results of these tests. It includes data on tests of 1/2-in.-OD Type 316 stainless-steel tubes at room temperature, at 1150°F, and at 1400°F. The individual points show the usually expected scatter in the results, accentuated possibly by some difficulty in accurately controlling and measuring the temperature of the tubes. Lines are shown to indicate the trend of the results at the three temperatures. As before, all failures were of the characteristic fatigue type. In the pressurized tubes, if the internal pressure used approximated normal working values for the temperature, there seemed to be no effect on the fatigue life. Some points plotted above and some plotted below the normal line. At 1400°F one tube pressurized to a circumferential stress of 14,000 psi showed, as would be expected, a reduced life.

The room-temperature-test results compare as follows with the allowable alternating stress values as given by the Navy code:

	500 cycles	2500 cycles	100,000 cycles
Tests at room temperature	185,000	132,000	63,000
Navy code, room temperature to 700°F	66,000	45,000	24,000
Ratio	2.8	2.93	2.62

On the basis of these tests, the following allowable alternating stress values at 1150°F are suggested:

	500 cycles	2500 cycles	100,000 cycles
Tests at 1150°F	110,000	80,000	39,000
Suggested allowable values	42,000	30,000	16,000
Ratio	2.62	2.67	2.44

These tests have definitely confirmed the applicability of high allowable fatigue stresses at elevated temperatures. The table above represents suggested allowable values to be used for fatigue stresses at 1150°F, which are not ultraconservative and yet which are in general agreement with the ratio of actual to allowable values used in the Navy code up to and including 700°F. If additional test data are lacking, it should be possible to interpolate allowable values for temperatures between 700 and 1200°F.

Status of Fast Oxide Reactor and Sodium Mass Transfer Projects

ROBERT W. LOCKHART
General Electric Company

For over fourteen years the General Electric Company has been designing, developing, and operating nuclear reactors which contain oxide or other ceramic fuels and which are cooled with sodium. This presentation is limited to a brief description of recent work in the development area.

About two years ago two projects sponsored by the Sodium Reactor Branch of the U. S. Atomic Energy Commission were established to study the fast oxide reactor and to investigate mass-transfer conditions in steel systems containing sodium at elevated temperatures.

Fast Oxide Reactor Project

Our interest in the application of ceramic fuels for fast power reactors stems from the potential of these fuels for very long burnup, and consequently low fuel-cycle costs, and core designs with a large overriding Doppler effect for safety.

The major results from the Fast Oxide Reactor program show that (1) high fuel burnup has been demonstrated, (2) adequate safety characteristics have been calculated, and (3) a technically feasible reactor with a low-cost fuel cycle has been designed.

The following tables present some of the more pertinent details that led to these conclusions.

Table 1 is a summary of the fuel irradiation program. Forty fuel specimens, each of them 0.15 in. in diameter and about 1.5 in. long were irradiated in 10 capsules and have been partially examined. The composition of the irradiated fuel in the experimental capsules is given in parentheses in the title of Table 1. Note that there was a mixed combination of plutonium oxide and uranium oxide. About one-half of the elements were swaged, and the other half were pelleted. The swaged fuel was about 75 per cent of theoretical density. The pelleted fuel was between 84 and 96 per cent of theoretical density. Improvements in these areas are certainly possible. Under the irradiation conditions for both types of fuel, the linear power was on the order of about 60,000 Btu/per (hr)(ft) of the length along the fuel element, and the clad surface temperatures were 1000°F on the average and about 1200°F maximum. The burnup is somewhere between 5000 and close to 100,000 Mwd/ton of metal (uranium plus plutonium).

There has been no leakage detected for any of these capsules. The fission gas released in the fuel capsules varied over a wide range in both cases, averaging about 52 per cent in the swaged case and about 40 per cent in the pelleted case.

Nearly all the capsules showed a central void when sectioned. The void was about 20 per cent of the area for the swaged case and about 9 per cent for the pelleted case. When these percentages are considered in relation to the initial densities, the remaining material must be

Table 1—DATA FOR 40 0.15-IN.-DIAMETER 1.5-IN.-LONG FUEL SPECIMENS
(20% PLUTONIUM—40% U²³⁵—40% U²³⁸ MIXED OXIDE) IRRADIATED IN 10 CAPSULES

Fabrication method	Swaged	Pelleted
Density, % theoretical	75	84 to 96
Number irradiated	18	22
Irradiation conditions		
Heat flux, 10 ⁶ Btu/(hr)(ft ²)		
Average	0.36 to 1.31	0.40 to 1.32
Maximum	0.50 to 1.52	0.62 to 1.58
Linear power, 10 ³ Btu/(hr)(ft)		
Average	18 to 65	20 to 65
Maximum	25 to 76	31 to 78
Surface temperature, °F		
Average	500 to 1055	650 to 1015
Maximum	705 to 1255	785 to 1215
Burnup, 10 ³ Mwd/ton (U plus Pu)	6.6 to 99.0	4.7 to 77.4
Postirradiation examination results		
Fission gas release, %		
Average	52.4	39.9
Range	27.8 to 71.7	11.5 to 56.1
Central void, %		
Area	0 to 21.3	0 to 9.7
Length	0 to 93	0 to 100

nearly at theoretical densities. The central void existed essentially throughout the whole length of the capsules.

The irradiation tests were based on our reference core design shown in Table 2. A target plant for 525-Mw(e) power requires about a 1420-Mw(t) power reactor. The core is composed of 34 vol.% fuel, 50 vol.% sodium, and 16 vol.% structure. The fissionable plutonium weighed about 12.8 wt.%. This design is based on a burnup of 100,000 Mwd/ton of metal, requiring an average specific power of 1000 kw/kg of the fissionable plutonium and an integral $k d \theta$ of about 5000 Btu/(hr)(ft). The coolant temperature of the core is 785°F inlet and 1050°F outlet with a 70-psi pressure drop across the reactor. The fueling interval is calculated at 10 weeks, and the fuel cycle costs 1.6 mills/kw-hr.

Table 2—FAST OXIDE REACTOR REFERENCE DESIGN

Reactor power, Mw	
Gross thermal	1,420
Net electric	525
Core composition, vol.%	
Fuel	34
Sodium	50
Structure	16
Plutonium concentration, 239/241, wt.%	12.8
Average fuel exposure, Mwd/ton	100,000
Average specific power, kw/kg of Pu ^{239/241}	1,000
Maximum $\int K d \theta$, Btu/(hr)(ft)	5,000
Coolant temperature, °F	
Inlet	785
Outlet	1,050
Coolant pressure loss, psi	70
Refueling interval, weeks	10
Fuel-cycle cost, mills/kw-hr	1.56

Some details of the cost estimations are shown in Table 3. We considered the fabrication, reprocessing, plutonium credit, in-pile and out-of-pile plutonium inventory, and also included

an item for the first core capitalization. If the interest-rate charge on the inventory is 10 per cent, this will provide the fuel cost (1.6 mills/kw-hr) I stated. If the interest-rate charge is 4 per cent, then the cost is considerably lower. Our basis of costing, as shown in Table 3, may be of interest. Note the plutonium value of \$12/g, the plutonium use charge of 10 per cent, the first core capitalization of 14 per cent and a plant load factor of 80 per cent.

Table 3—BREAKDOWN OF FUEL-CYCLE COST, MILLS/KW(e)-HR

	Core	Ax. bl.	Rad. bl.	Total
Fabrication	0.236	0.182	0.072	0.490
Reprocessing	0.060	0.046	0.133	0.239
Plutonium credit	-0.072	-0.026	-0.021	-0.119
Plutonium inventory				
In-pile	0.400	0.041	0.059	0.675
Out-of-pile	0.140	0.014	0.021	
First core capital	0.134	0.104	0.040	0.278
Total inventory				
Interest rate charge of 10%				1.56
Interest rate charge of 4%				1.18
Basis of costing				
Fabrication, \$/kg				
Core and ax. bl.			270	
Radial bl.			34	
Reprocessing \$/kg				
Core and ax. bl.			68	
Radial bl.			63	
Plutonium value, \$/g			12	
Plutonium use charge, %			10	
First core capitalization, %			14	
Plant load factor, %			80	

Table 4 presents a summary of some of the nuclear data with reference to the safety parameters. Note the large negative Doppler coefficient (in excess of $9 \Delta K/^{\circ}C$) in this core design. Sodium expansion and steel clad expansion are both positive. Excluding fuel axial expansion (we are not certain of this value at this time) there is a total negative reactivity coefficient of nearly $8 \Delta K/^{\circ}C$.

To consider some extreme cases of control that may be needed, if there were an instantaneous loss of all sodium from the core, essentially there would be a zero reactivity effect with the reference core design that was presented. If all the control rods were blown out instantaneously, there would be 1.8 dollars worth of reactivity.

The third item in Table 4 is an equation indicating the condition in which all the fuel might melt inside the cladding. In other words, it appears as though there is enough negative control to contain an incident of 1.9 dollars worth of reactivity.

Two other items of interest are (1) the maximum reactivity worth of the subassembly of one dollar, i.e., the fuel subassembly inserted in the reactor in the most critical location, and (2) 100 per cent fuel slump reactivity. This would result in a 7 dollars worth of reactivity accident; however, it is an event that is relatively unlikely to occur.

This particular reactor, whether it has uranium or plutonium in the core, has a 0.004 delayed neutron fraction, with the neutron lifetime on the order of 5×10^7 sec. The plutonium enrichment of the fissionable material is 12.8 per cent, and the breeding ratio is just over 1 per cent.

An experimental fast oxide reactor, termed the EFOR, has been proposed in order to demonstrate the key safety characteristics developed during the Phase I program. The purpose of this experiment is (1) to demonstrate the operating characteristics of the large fast oxide reactor by using representative core composition, neutron leakage, and fuel and coolant tem-

perature profiles; (2) to check ability to calculate reactor coefficients and core physics parameters and to predict behavior; (3) to demonstrate the effectiveness of shutdown mechanisms in transient tests; (4) to gain operating experience in general and, in particular, with plutonium fuel under this type heat flux operation; and (5) to identify possible problems such as the effect of fuel slumping and plutonium migration.

Table 4—SAFETY PARAMETERS OF REFERENCE CORE

Reactivity coefficients, $\Delta K/^{\circ}C$	
Doppler	-9.3×10^{-6}
Sodium expansion	1.0×10^{-6}
Steel clad axial expansion	0.38×10^{-6}
Fuel axial expansion	-1.27×10^{-6}
Total (excluding fuel expansion)	-7.92×10^{-6}
Coolant and control-rod loss, dollars	
(ΔK) sodium	0
(ΔK) rod	1.8
$[1 + \frac{1}{2} (\Delta K) \text{ doppler}]$	-1.9
Maximum reactivity worth of subassembly	1
100% fuel slump reactivity	7
Delayed neutron fraction	0.004
Neutron lifetime, sec	$\sim 5 \times 10^{-7}$
Pu ²³⁹ enrichment, %	12.8
Pu ²³⁹ breeding ratio	1.08

Some of the experiments we hope to evaluate during this test program are as follows. There is a series of critical tests at low power to control calibration, flux distributions, and temperature coefficients; some power oscillation tests (small signal, 0.003 to cps, increasing power steps); coolant transients in large load following disturbances (inlet temperature, flow transients, and power increases in short periods); and power excursions tests (graduated severity in initial power, reactivity, and insertion rate). This experimental program is designed to provide information pertinent to the evaluation of reactivity effects due to Doppler effect with long fuel time constant; thermal expansion of fuel and other core components; possible slumping or clad distortion during severe transients; and nonlinear reactivity changes due to possible fuel bowing or impediments to mechanical motion of core components under severe temperature gradients.

The power excursion series of tests will demonstrate the capability of the inherent shutdown mechanisms. Reactivity ramp insertions will be gradually increased in severity in successive tests, with an upper limit of about 2 dollars reactivity insertion in a period somewhere between 50 and 200 msec.

In parallel with our reactor development work centered on EFOR, we are continuing a related fuel development program that includes a number of tasks such as testing fuel pins in TREAT to overpower conditions, additional fuel irradiations in the fast flux of EBR-2, the investigation of a vented fuel concept, and a study of the use of mixed uranium-plutonium carbide fuel in a fast reactor. In addition to the foregoing programs, we shall continue analytical evaluations of both the fuel and the reactor. These studies are intended to ensure that the experimental programs continue to be oriented in the most fruitful directions.

In conclusion, the potential for low fuel-cycle cost represents one of the prime incentives for the development of fast reactors. However, the well-recognized capabilities of these units to operate with a net gain of fissile material is in itself insufficient. Total fuel-cycle costs are made up of a number of components, most of which are a function of fuel burnup. In consequence, the high burnup capabilities of ceramic fuels appear necessary for the achievement of low fuel-cycle cost fast reactors. Recent in-pile tests performed by the Atomic Power Equipment Department for the AEC have shown that PuO₂-UO₂ fuels clad in stainless steel can withstand burnups of up to 100,000 Mwd/ton without mechanical damage to the clad. To use this high burnup capability, two fast reactor conditions have to be met. These are (1) a reliable

over-riding prompt negative power coefficient of reactivity must be found and (2) operational swings of reactivity in the reactor core must be limited so that excessive amounts of reactivity are not held by the control rods.

The results of our preliminary work make it appear that ceramic-fueled fast reactors can be designed to have both an attractive economic performance and a reliable prompt negative power coefficient. We have proposed that an experimental fast oxide reactor be operated to confirm these calculations.

Sodium Mass Transfer Project

The remainder of my presentation will concern the investigation of mass-transfer conditions in steel systems containing sodium at elevated temperatures. Data from this project should be applicable to both primary and secondary sodium coolant systems for reactors with any neutron spectrum.

This development program is sponsored by the Sodium Component Development Program of the AEC, and as part of the AEC objective consists in designing, fabricating, operating, and evaluating data from six experimental sodium test loops to determine rates of metallic mass transfer for a selected group of steels. The scope and loop design conditions are summarized in Table 5.

Table 5—MASS-TRANSFER INVESTIGATIONS IN LIQUID-METAL SYSTEMS
AT MODERATELY HIGH TEMPERATURES

Scope		
Design, fabricate, and operate six test loops, analyze test data to determine mass transfer or corrosion of three types of steel that could be used to fabricate liquid-metal heat-transfer systems		
Design conditions		
Liquid metal,		sodium
Temperature, °F		
Maximum		1300
Minimum		600
ΔT, °F		200 to 500
Pressure, psig		100
Velocity, ft/sec		up to 30
Purity, ppm of oxygen		50
Flow, gal/min		1

The steels to be tested include type 316 stainless steel, 2¼% chromium–1% molybdenum and 5% chromium–½% molybdenum–½% titanium alloy steels, and bimetallic systems containing stainless and one of the low-alloy steels. Data are obtained from removable corrosion samples located in various temperature regions around the loop, as well as from segments of the loop. A radioactive-tracer method has been utilized to obtain additional data by placing an activated steel foil sample into a biologically shielded sample holder in the maximum temperature region. The six loops are identical except for material composition. As required by our contract, the steels and the sodium (not reactor grade) were purchased with commercial specifications. Our six test loops were installed, charged with sodium, and had some operation by the end of August 1960.

Figure 1 shows a typical hot-leg sample holder. Normally samples of the same material as the sample holder are used. Rectangular cross-section samples are located in rectangular flow passages or slots in order to get a maximum amount of flow variation across the sample, obtaining the 3 to 1 (high to low) velocity range which will be mentioned later and which should be obvious from the three-slot and single-slot insert design.

The number of samples loaded in each holder is described in Fig. 2. This figure shows that all sample slots were charged with samples during the initial run of each loop. Other patterns are possible for future testing.

Fig 1—Typical hot-leg sample holder.

Fig. 2—Run 1-1 sample loadings. Regular, 25, 6-mil foil, 1

Figure 3 shows the flow sheet for the corrosion samples. All low-alloy-steel materials for corrosion samples were prepared by splitting 2-in. schedule-40 pipes and rolling them into $\frac{1}{16}$ -in.-thick sheet. The stainless-steel samples were prepared from as received $\frac{1}{16}$ -in.-thick sheet. The dashed boxes in Fig. 3 indicate that some samples proceed through this step and others do not. For instance, the low-alloy-steel samples are heat treated, whereas the stainless-steel samples are used without further heat treatment.

Table 6— LOOP OPERATING TIME TO MAY 15, 1961

Material	Run number	Run hours	Total hours
316 stainless steel	1-P	394	
	1-1*	2,964	3,688
316 stainless steel-2 $\frac{1}{4}$ % chromium	2-1	2,782	
	2-2*	183	3,656
316 stainless steel-2 $\frac{1}{4}$ % chromium (4S8)	3-1	2,729	
	3-2	704	
	3-3	381	4,796
316 stainless steel-5% chromium	4-1	2,802	
	4-2*	0	4,045
2 $\frac{1}{4}$ % chromium	5-1	2,697	
	5-2	0	2,730
5% chromium	6-1	2,792	
	6-2	0	3,745
Total			22,660

*Activated foil.

In Table 6 the hyphenated run numbers describe first the loop number and second the chronological test number. The asterisk means that an activated foil sample has been inserted in the H3R sample holder. Transfer of activated isotope data will be collected on all corrosion samples. The total number of loop hours indicates a considerable amount of operating time, with 24 gasketed flanges operating over 1100°F and two operating at 1300°F.

Figure 4 presents the available corrosion-rate data. It should be reemphasized that these data are a composite from the four loops containing 316 stainless steel. It also should be noted that the data presented are at the low-velocity (or three-slot) conditions. The hot-leg corrosion data at high velocity can be obtained by doubling the low-velocity corrosion rate. This factor of 2 in corrosion rate for a factor of 3 in velocity is reasonably consistent with a Reynolds number to the 0.8 power variation. To date, there is no high-velocity data for the cold leg.

Both low-chrome-alloy loops (numbers 5 and 6) have a slightly lower slope on the hot-leg corrosion samples. It is conceivable that carbon loss and gain corrections may normalize this data with the 316 stainless. In comparing the loop 2 and loop 3 data, the low oxygen data from loop 2 indicates a slightly lower corrosion rate for both the hot and cold legs.

"Down-stream" samples in H3R had a lower corrosion rate of about a factor 6. Metallographic results can be summarized by examining the following figures:

Figure 5 shows 316 stainless steel before test.

Figure 6 shows 316 stainless steel after test.

Figure 7 shows the 316 stainless-steel ferritic layer

Figure 8 shows 2 $\frac{1}{4}$ % chrome before test.

Figure 9 shows 2 $\frac{1}{4}$ % chrome after test.

In addition to the corrosion and metallic exposure effects that have been presented, additional information is becoming available from the preliminary radioactive tracer data from the 316 stainless-steel loop number 1. This test run was completed last week, therefore end-of-run test data are not available at this time. The data obtained with a 256 channel analyzer from the corrosion samples removed from the loop after 700 hr of operation with the activated foil

(Text continues on page 148)

Fig. 3—Flow sheet for corrosion samples.

Fig. 4—Run 1 corrosion data.

Fig. 5—316 stainless steel before test. A10M. Carbon, 0.045 per cent.

Fig. 6—316 stainless steel after test. 2476. 1200°F for 1438 hr. Carbon, 0.075 per cent.

Fig. 7—316 stainless-steel ferritic layer. 1401. 1300°F for 394 hr. Cold worked.

Fig. 8— $2\frac{1}{4}\%$ chrome before test. B11M. Carbon, 0.105 per cent.

Fig. 9— $2\frac{1}{4}\%$ chrome after test. 5476. 1200°F for 1251 hr. Carbon, 0.040 per cent.

indicated that (1) gamma activity was observed on all samples removed, whether they gained or lost weight, (2) manganese was transferred at a very high rate, (3) iron was transferred at a high rate, (4) chromium was transferred at a low rate, and (5) cobalt was transferred at a very low rate.

Several other observations during the test operation should be noted. They are as follows:

1. During the initial runs, when the plugging temperature was about 450°F or 50 ppm of oxygen, it was necessary to add oxygen to the system to maintain the correct level. The loss rate was as high as 1 ppm/day.

2. During the final two months of the four-month test run, the flow rate decreased with time. Subsequent tests indicated that there was some slow fouling occurring on the 12 cold-leg corrosion samples in four of the loops. The decrease in flow was relatively negligible in the low oxygen run (two to one). The 2 $\frac{1}{4}$ % chromium loop demonstrated the most rapid fouling of about 5 per cent loss in flow per day. The second most troublesome loop was the 316 stainless-steel loop where the fouling was in the order of 2 per cent loss in flow per day. This flow fouling was temporarily corrected by cleaning all the cold-leg samples and returning them to the loop. However, the fouling rate continued after each cleaning.

3. Several flange leaks were experienced on the 316 stainless steel and the 2 $\frac{1}{4}$ % chromium sample holders. In each case, if the temperature exceeded 1100°F, there was considerable corrosion of the sample holder on the air side. We have proposed that additional work be initiated to determine how to minimize or prevent this air-side corrosion.

In summary, the initial data from the mass-transfer program have indicated that tolerable corrosion rates are apparent when the test loops have been operated under severe, screening type operation of 1200°F, $T_{\max}$; 500°F, ΔT ; and at a 450°F plugging temperature (50 ppm of oxygen).

Apparently, there is no doubt that the 2 $\frac{1}{4}$ % chromium steels will be depleted of most of their carbon in a relatively short time and that the 5% chromium alloy has not exhibited this change. The 316 stainless steel gained carbon and a ferritic surface layer was formed.

It appears to be significant for the reactor and coolant system design to take into account the changes in the properties of the steels under these conditions. It is anticipated that the corrosion rate will be lower where there are fewer impurities in the sodium.

The preliminary data from the radioactive-tracer technique indicates that its use is warranted in order to determine some of the mass-transfer and corrosion mechanisms that are operating within such a complex system.

For those considering operation of major plants with temperatures above 1000°F, some attention should be given to the prevention of rapid air-side corrosion after a sodium leak.

Los Alamos Molten Plutonium Reactor Experiment Development Work

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The LAMPRE Program

My topic will cover the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) program as differentiated from the LAMPRE-1 reactor that you heard described this morning. The reactor is only one phase of an over-all program whose goal is the development of proved technology for efficient utilization of plutonium fuels for reactor applications.

The LAMPRE program has evolved from considerations of the most rewarding manner in which plutonium might be used as a reactor fuel. The important conclusions from these studies may be summarized as follows:

1. Plutonium, as the conversion product of the abundant U^{238} , at some future date must assume the position of the main source of fission energy for a nuclear industry.
2. Plutonium fuel is most efficiently used in a fast breeder reactor.
3. The fuel-cycling costs and fabrication charges must be reduced to minimum values if economical nuclear power is to be achieved.

These considerations have led to a conceptual reactor fuel of molten metal having diluents to improve the physical and thermal properties of plutonium. The first alloy in this category which has been studied extensively is the eutectic composition containing approximately $9\frac{1}{2}$ at. % iron. The use of the fuel in a molten form eliminates the need for expensive fuel fabrication and, in principle, allows extremely high burnup. Thus one of the principle disadvantages of plutonium as a reactor fuel, i.e., its low melting point, is turned into a potential advantage.

Calculations based on many different reactor concepts using molten-plutonium fuel indicate that systems of this type should ultimately be able to breed efficiently with a doubling time of less than ten years.

The most important and formidable problem in the program to establish the feasibility of the reactor concept is that of finding a suitable container and determining the appropriate specifications for the container and fuel materials. This phase investigation has dominated the program for the past several years.

Substantial effort has gone into the study of cycles that might achieve optimum over-all efficiency in a fluid-fuel reactor system. The LAMPRE-1 design, with a specific power of about 40 watts/g, offers little promise of leading to economical nuclear power. Indeed the

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*Work done under the auspices of the U. S. Atomic Energy Commission.

encapsulated fuel design* defeats the purpose of the mobile fuel system. To be of economical interest, the reactor system should be capable of producing about 500 watts/g. The systems that seem to show promise for use in competitive power plants fall into two general categories; contained cores and direct-contact cores.

Contained cores are more conventional designs in which the fuel is separated from the coolant by a metal wall through which the heat must be transferred. To keep heat flux and fuel temperature within reasonable bounds, the fuel ligament must be kept very small, of the order of 100 mils, and even then heat fluxes of 2 million Btu/(hr)(sq ft)(°F) will be encountered. To keep nuclear reactivity within reasonable bounds, it is necessary to process the fuel continuously to remove gaseous fission products. Although this requires only a low fuel-circulation rate (several core volumes per hour), it does require additional equipment and adds complication to the core design. One may then ask why not circulate the fuel faster and take advantage of the process for heat transfer, eliminating the many small fuel passages.

Fig. 1—Direct-contact reactor system. Schematic.

The direct-contact core system, as evolved from the above considerations, is shown schematically in Fig. 1. In this arrangement, the fuel is circulated in direct contact with the immiscible sodium coolant to effect heat transfer as well as to purge fission gases. The system is comprised of a containment vessel of simple geometry always completely full of fuel in which the nuclear reaction occurs. Fuel, heated in the reacting region, is pumped from the core by a sodium jet pump that intimately mixes the fuel and coolant. After heat transfer and some fission-product extraction have occurred, the mixed stream passes into a separator where the coolant and gases are separated from the fuel. The fuel is then returned to the bottom of the core and the heated sodium is removed to a heat exchanger. The flow rates

*See paper by R. E. Peterson on pages 52 to 62 of this report.

necessary for such a system are of the order of core volumes per second, indicating rather formidable problems in fluid dynamics.

It would be irresponsible at this date to say that direct-contact cores are feasible and will be operated in the near future. However, there are many attractive features of the concept which ease the problem of transferring 2 or 3 million Btu/(hr)(sq ft)(°F) and tease the imagination with schemes for economical in-pile fuel processing. However, much experimental work on the fluid dynamics, materials, and neutronics problems must be accomplished before we can say that these schemes are practical. The LAMPRE program at LASL is just such a development effort.

Container Development

The most significant problem in the program to establish the feasibility of the LAMPRE concept is that of finding a suitable container and determining the appropriate specifications for the container and fuel materials.

Early in the establishment of the molten-plutonium concept it was demonstrated that the plutonium-iron eutectic fuel could be contained in tantalum; however, over the past four years, numerous difficulties with tantalum have been uncovered. Originally the available material was very high in impurity content and in many cases was difficult to weld due to gas evolution. Although structurally satisfactory, the heat-affected zone adjacent to a weld would deteriorate due to migration of impurities during welding and be less resistant to corrosive attack by the fuel than either the parent metal or the actual weld. In order to provide a container for LAMPRE-1, two approaches were taken: (1) efforts were initiated to improve the quality of tantalum and (2) a container was developed which did not contain any welds in direct contact with the fuel.

The program to improve the quality of tantalum has been eminently successful, owing largely to the cooperation of the tantalum industry. Improvements in vacuum arc-melting techniques and the introduction of electron-beam melting furnaces have made tantalum ingots of extreme purity readily available at reasonable cost. Table 1 lists the typical composition of a commercial 3-in.-diameter electron-beam-melted ingot.

To satisfy the second requirement, it was necessary to develop seamless capsules about 0.43 in. in diameter and 8.3 in. long with a 0.026-in. wall thickness. Two methods of fabrication have been successfully used to date, both starting from 3- to 4-in.-diameter ingots. However, satisfactory fabrication of seamless capsules by either method depends upon breaking the extremely coarse-grain structure of the as-cast ingot into uniformly fine-grain material.

The deep-drawing process requires a 4-in.-diameter disc of 30-mil sheet with a uniform equiaxed fine-grain structure. The 13 stages necessary to go from the sheet to the finished capsule are shown in Fig. 2. The success of this method depends on the quality of the starting sheet material. Excessive directionality in the grain structure usually causes earring in the early cupping stages which lead to folds and seams in the final product. In addition, the phenomenon of critical strain was observed, which caused excessive grain growth during annealing of the material containing between 10 and 20 per cent cold work. In the deep-drawing process, some part of the cup is inevitably within this range; and these areas result in a rough surface on the finished capsule. However, with proper attention to rolling procedure, die design, die materials, heat treatment, and lubrication acceptable capsules can be made by this method.

The second, and generally preferred, method of capsule fabrication is referred to as impacting and ironing. In this process the large ingot is rolled and swaged into a $\frac{5}{8}$ -in.-diameter rod. This rod is cut into pieces about $1\frac{1}{4}$ in. long and is then formed into a heavy walled cup by impact extrusion. The capsule is finished by successively thinning the wall in a series of drawing operations. Figure 3 shows the various stages of this process.

The impacting and ironing process is preferred at the moment for production because the resulting capsules seem to have a better structure and better surface finish, are stronger, and seem to be more resistant to corrosive attack by the fuel. An added bonus is that there is

Table 1—CHEMICAL ANALYSIS OF ELECTRON-BEAM-MELTED TANTALUM

Element	Limit of detection, ppm	Typical composition of ingot, ppm
Oxygen	10 ± 5	15–20
Nitrogen	5 ± 3	10–20
Hydrogen	10	<10
Carbon	25	<25
Aluminum	10	<10
Silicon	3	<3
Titanium	5	<5
Vanadium	20	<20
Chromium	10	<10
Manganese	3	<3
Iron	10	<10
Cobalt	20	<20
Nickel	10	<10
Copper	2	<2
Zinc	100	<100
Zirconium	30	<30
Niobium	200	<200
Molybdenum	30	<30
Tungsten	100	<100
Lead	30	<30
Bismuth	10	<10

Fig. 2—Deep drawing stages for LAMPRE-1. Pin type.

Fig. 3—Impact extruded and ironed LAMPRE-1 thimbles

less scrap material produced by this method. At a cost of \$51.50/lb for the material, this is a significant item.

Fuel Development

The fuel with which we have had the most experience is the eutectic binary alloy of plutonium and iron containing $9\frac{1}{2}$ at. % iron. This fuel has a melting point of 770°F, a density of about 16 g/cc, and a thermal conductivity about the same as stainless steel. This fuel has been used in the largest part of our work to date, and it is the only fuel that has been used in LAMPRE-1.

For many of the reactor concepts being studied for power-plant use, it is desirable to have a fuel in which the plutonium concentration can be reduced without greatly changing the melting point of the fuel. Studies of the direct-contact core system indicate that a fuel with a plutonium concentration of 20 at. % is preferred to the plutonium-iron fuel with 90 per cent plutonium.

Plutonium-cobalt, plutonium-nickel, and plutonium-copper form low-melting alloys and, in addition, these systems can be diluted with cerium to form a variety of ternary alloys covering the range of plutonium concentrations from 20 to 90 at. % with melting points in the range of 770 to 840°F. Work is now in progress to establish phase diagrams and necessary physical properties for some of these alloys. Some very preliminary corrosion tests have been run which indicate that the containment problem for these alloys is about like that for plutonium-iron. However, the alloys that are very dilute in plutonium seem to be slightly less corrosive.

Corrosion Testing

Early corrosion tests were designed to determine temperature-gradient mass-transfer effects.¹ The test consisted of a sealed tantalum capsule containing fuel alloy positioned in a furnace that maintained one end of the capsule at about 1000°F and the other end at 1270°F.

The furnace was then tipped end for end on a 6-min cycle over a period of several hundred hours. By the use of extremely sensitive radioactive tracer techniques, it was shown that there did not appear to be a mass-transfer effect so long as trace amounts of carbon were present in the fuel. However, a severe intergranular type of attack was observed. The development and evaluation of methods to eliminate this intergranular penetration have dominated the corrosion test program during the past several years. The basic approach has been to improve the purity of the tantalum, thus removing the tramp material from the grain boundaries; to add alloying agents to the tantalum such as tungsten and yttrium; and to study the effect of various impurities and additives in the fuel. The basic corrosion test consists of an isothermal circulating sodium loop that contains 12 to 18 full-sized LAMPRE-1 capsules. The sodium is hot-trapped with zirconium to provide a chemical environment as close as possible to that found in the reactor. These capsules are exposed to the flowing sodium stream at temperatures of 975°F and up, depending upon the purpose of the test. Usually after 700 hr the capsules are removed and checked for fuel penetration by using autoradiographic techniques. When failures occur they are usually sectioned and studied metallographically.

The fuel elements now in LAMPRE-1 have been tested in the manner just described and have demonstrated a capability of withstanding a temperature of 975°F for more than 2500 hr. These same fuel elements are now undergoing further testing at 1065°F. Some capsule and fuel combinations now in test show promise of reliably withstanding fuel temperatures of 1200°F for 3000 hr or more. The current test program is aimed at improving the fuel alloy and developing a container alloy that will permit higher fuel temperatures to be used in the reactor.

Numerous binary alloys of tantalum have been made and tested in an effort to improve the corrosion resistance of this material. The following general statements may be made:

1. Additions of tungsten, rhenium, and yttrium have shown beneficial effects.
2. Additions of titanium, thorium, and iron have been ineffective.
3. Additions of zirconium, hafnium, and carbon have been detrimental.
4. Alloys containing aluminum, beryllium, cerium, lanthanum, scandium, and uranium are still under investigation.

Direct-contact Core Experiment

A very preliminary effort to demonstrate the feasibility of the direct-contact type core described earlier has been successfully completed. The limited purpose of this experiment was to show that molten plutonium could be pumped around a loop by a stream of molten sodium and then separated and returned to a reservoir. The loop was very small, containing about 200 g of fuel. In this case the fuel was one of the dilute ternaries mentioned previously: 48 at. % cerium-23 at. % cobalt-29 at. % plutonium which melts at 825°F.

The test geometry is shown in Fig. 4. The fuel is contained in a 1-in.-diameter reservoir. Sodium enters through the tube leading into the bottom of the reservoir and then up through the pumping section. The jet pump action occurs at the discharge of the sodium inlet tube and the fuel-sodium mixture then flows up through the $\frac{1}{4}$ -in.-diameter chimney. It is this mixed-flow region that schematically represents the heat exchanger in the direct-contact concept. At the discharge of the chimney, the mixture separates, the fuel falling off to the side and the sodium going on up.

This apparatus operated quite satisfactorily for several days. No carry-over of fuel into the rest of the system was observed; however, under some conditions, sodium did entrain with the fuel. Much better separation could certainly be achieved by employing change of direction to separate the two liquids.

Figure 5 is an X-ray picture of the system in actual operation. The low-density sodium stream can be seen discharging from the top of the chimney. The increasing density in the annular region indicates fuel settling into the reservoir, and toward the bottom the complete separation of fuel from the sodium may be observed.

Fig. 4—Direct-contact pumping experiment.

Fig. 5—Direct-contact core experiment. Pumping at 870°F.

The 2-Mw Steam-generator Test Facility

We are now operating a 2-Mw gas-fired sodium component test facility and steam generator that produces steam at 925°F and 900 psi from sodium at 1100°F. These are the minimum conditions that can efficiently be used with a reactor operating with molten-plutonium fuel. The system is a scaled-down version of an actual reactor power plant containing both a primary and a secondary sodium system. Insofar as possible all test components are $\frac{1}{10}$ scale models of actual equipment for a molten-plutonium power plant having a thermal capacity of 20 Mw.

The objective of this particular steam-generator design is to demonstrate reliability as a result of simple design and rigorous application of quality control measures during construction. Thoroughly inspected single-wall tubing was used to avoid the complex fabrication procedures and resulting high cost required for duplex-tube third-fluid designs. Thermal expansion in tubes and in shells is accommodated through a U type configuration. All tube sheets are isolated from the sodium by a gas space to prevent high thermal stress and shock to these components. A complete description of the plant is given in reference 2.

Figure 6 shows the general arrangement of the system and gives the full-load design conditions. The purpose of the economizer type of tempering sodium heater is to cool the sodium leaving the superheater prior to entering the evaporator and then to reheat the sodium on its way back to the intermediate heat exchanger. This piece of equipment is necessary in order to reduce thermal stresses in the tubes and shell at the evaporator inlet. It is very effective in controlling tube-wall temperatures in the evaporator.

The steam-generating unit is depicted in Fig. 7, showing how the evaporator, tempering heater, and superheater are formed into a single horizontal U-tube unit. Sodium makes a

Fig. 6—Steam-generator flow diagram. Full load.

single pass through the shell side. The vertical risers are arranged as shown in order to control the sodium level a minimum of 2 ft from the tube sheets at all times.

Materials used in this unit are $1\frac{1}{4}\%$ chromium- $\frac{1}{2}\%$ molybdenum steel in the evaporator and the tempering heater and type 316 stainless steel in the superheater and the intermediate heat exchanger. Croloy was used where possible in order to reduce thermal stresses and because it is more resistant to chloride stress corrosion. Where pressure and temperature conditions would not permit the use of Croloy, it was necessary to use stainless steel.

The tube-to-tube sheet joints have been designed to eliminate all crevices in order to prevent corrosion due to build-up of chloride concentration in the crevice. The joint was made by butt welding the tube to a boss machined on the tube sheet. The welding is performed from the inside of the tube by using a special heliarc torch in an automatic welding jig without the addition of a filler metal.* Figure 8 shows sample welds prepared in this manner. It is possible to obtain reliable full-penetration welds with a smooth surface and high strength in this manner. The technique has been successfully applied to tubing as small as 0.525 in. in inside diameter.

In this header design it was possible to inspect radiographically all tube-to-tube sheet welds with a sensitivity capable of detecting spherical voids 5 mils in diameter. In the four units constructed, the two stainless-steel units were completed with no detectable flaws in the welds. The Croloy welds, however, consistently had weld defects of the order of half the tube wall in size. Numerous unsuccessful attempts were made to eliminate these voids, even to the extent of trying vacuum-cast material for the tube sheets. Thus the tempering heater was assembled containing many such weld defects. Since this unit has sodium from the same

*This joint involves the use of patents controlled by the Griscom-Russell Company.

SUPERHEATER - 316 SS

TEMPERING HEATER- 1 1/4 GR.-1/2 MO STEEL

EVAPORATOR-1 1/4 CR- 1/2 MO STEEL

Fig. 8—Tube-joint weld preparation and weld specimen.

system on both sides of the welds, a rather extensive weld failure would not seriously affect its performance.

In the evaporator, however, no such risk could be tolerated. In this unit it was necessary to use a more conventional rolled joint with a seal weld at the top of the tube sheet. These welds were not inspectable and the joint provided a deep crevice on the sodium side. However, chloride crevice corrosion has not proved to be a problem on the sodium side; and, since in this system the sodium level is kept at least 2 ft from the tube sheet by a gas buffer, it was felt that the unit would perform satisfactorily.

Figure 9 shows the header of the intermediate heat exchanger during construction. Figure 10 shows the same unit installed in the loop. Here the by-pass legs for the sodium-level detectors can be seen. A variable-reluctance type of level detector is used to monitor the liquid level throughout the system. This consists of a small transducer coil that operates inside a thimble installed in the sodium system. A ball-screw mechanism controlled by a servo sensing mechanism causes the transducer to seek the liquid level, and the level is then read out to the nearest one-tenth of an inch on the control panel. These instruments have given satisfactory service in sodium at 1100°F for many months.

Operating History

The test facility was placed in steaming operation on Mar. 22, 1960. Since that time the following operating hours have been logged: steam generator, 7004; intermediate heat exchanger, 9165; primary sodium pump, 9165; secondary sodium pump, 10,324; and gas-fired sodium heater, 8124.

The operation of the sodium system and the steam generator has been more than satisfactory. All the down time and operating problems have been associated with the more conventional equipment such as the feedwater pump, the gas-fired heater, the feedwater control valve, etc. It is significant to note that since the secondary sodium system was placed in operation over 14 months ago it has not been shut down or serviced for any reason whatsoever. The primary system has been shut down several times primarily to service the gas-fired

Fig. 9—Intermediate-heat-exchanger tube-to-tube sheet welds.

Fig. 10—Intermediate heat exchanger erected in place.

heater. The only service required on any part of the sodium system was the replacement of a defective seal in the primary sodium pump early in the operation.

Operating characteristics of the steam generator have been quite satisfactory and over-all performance of the unit has been reasonably as predicted. It has been observed that allowing tube-wall temperature in the evaporator to rise too far above saturation temperature will reduce the heat-transfer coefficient and lead to operating instabilities. We are not prepared to elaborate on this point until more data have been obtained and analyzed.

The plans for this unit are to complete 9000 hr of steaming operation and then metallographically examine the entire system. Upon completion of this work, a complete report will be written including all significant operating data.

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Analytical Method for the Determination of Oxygen in Sodium

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In the development of improved sodium systems, progress in reducing the oxygen content has outstripped the methods for measuring it. At least five such methods are in active use and have proven to be quite effective in the region of 30 to 100 ppm or higher of oxygen in sodium.

As the oxygen content approaches the lower limit obtainable by cold-trapping, approximately 15 ppm, all these methods encounter difficulties that produce scatter in results equivalent to the quantity being measured. On the other hand, corrosion tests have indicated that hot-trapping can reduce the oxygen content below 10 ppm and is desirable for sodium systems containing zirconium, niobium, or tantalum and for reduced corrosion of steel at high temperatures.

The program described herein is aimed at developing an analytical method capable of measuring the low oxygen contents that hot-traps are believed to produce in sodium, i.e., 1 to 10 ppm. This program is sponsored by the U. S. Atomic Energy Commission under contract AT (30-1)-2303(XV), Scope 2, with Nuclear Development Associates, Inc. H. Steinmetz is the responsible scientist. Work is still in progress; this report covers approximately the first nine months of work.

The intent of the method being developed is to combine two present chemical techniques, the mercury amalgamation method and the Karl Fischer titration method, in such a way to improve the sensitivity and to avoid the major sources of error heretofore encountered. The standard mercury amalgamation method involves separating the sodium oxide from the sodium by repeated extractions with large volumes of mercury, during which the preservation of microgram quantities of sodium oxide is mechanically difficult.

The standard Karl Fischer titration method avoids separating the oxide from the sodium; but it requires large volumes of alcohol, for which the correction due to moisture content is much larger than the value being measured. The main advantage of the new approach is that a complete separation of sodium from the sodium oxide is not required.

The sodium sample is amalgamated with mercury in a glass flask. The bulk of the amalgam is withdrawn through a stopcock at the base of the flask, leaving behind a small amount of amalgam in the flask. Any sodium oxide present in the sodium sample also remains behind in the flask, floating on top of the amalgam. The amalgamation process may be repeated a second time.

Up to this point the Pepkowitz amalgamation method for the determination of oxygen in sodium has been followed. A measured volume of an alcohol, such as methanol or ethyl cellosolve, is then added to the flask to dissolve the sodium oxide as well as some of the sodium remaining in the flask. In a series of reactions, an amount of water equivalent to the sodium oxide is formed. This water is then determined by titration with the Karl Fischer reagent.

Sodium metal does not interfere with this determination. Although the entire original sample of sodium could be dissolved in alcohol, thus avoiding the mercury extraction step, this method would require a large volume of alcohol. Since the alcohol has some moisture in it, which is determined in advance and for which allowances as a blank are made, such a direct solution of the entire sample in alcohol would decrease the sensitivity of the method.

The method can be summarized by the following pertinent chemical equations.

1. $\text{Na} + \text{Hg} \rightarrow \text{Na(Hg)}$
2. $2 \text{Na(Hg)} + 2 \text{ROH} \rightarrow 2 \text{NaOR} + \text{H}_2 + \text{Hg}$
3. $\text{Na}_2\text{O} + \text{ROH} \rightarrow \text{NaOH} + \text{NaOR}$
4. $\text{NaOH} + \text{HA} \rightarrow \text{NaA} + \text{H}_2\text{O}$
5. $\text{NaOR} + \text{HA} \rightarrow \text{NaA} + \text{ROH}$
6. $\text{SO}_2 + \text{I}_2 + 3 \text{C}_5\text{H}_5\text{N} + \text{CH}_3\text{OH} + \text{H}_2\text{O} \rightarrow \text{C}_5\text{H}_5\text{NH} \cdot \text{SO}_4 \cdot \text{CH}_3$ (pyridinium hydrogen methyl sulfate) + $2 \text{C}_5\text{H}_5\text{N} \cdot \text{HI}$ (pyridinium hydroiodide)

High-purity sodium is prepared by gettering sodium with a zirconium foil at 1200°F. The sodium is first aged in a stainless-steel tank, where it is held just above its melting point to permit excess oxide to precipitate and grow. The sodium is then transferred through a porous stainless-steel filter with a mean pore opening of 5 microns into the gettering tank. In the gettering tank the sodium contacts a large area of zirconium-foil surface plus the 304 stainless-steel surface of the tank. This treatment is effective in reducing the oxygen (and carbon) content of the sodium to extremely low levels.

Table 1—MEASURED OXYGEN CONTENT OF
ZIRCONIUM-GETTERED SODIUM

Sample size, g	Comment	Oxygen content, ppm
1 76	First three samples were taken from same extruded fill	2
2 07		3
1 72		6
2 38	Last three samples were taken from a different fill	3
2 02		1
1 78		3

A supply of a high-purity inert gas, helium, is used as a cover gas and to facilitate transfer of sodium. The purified sodium from the gettering tank is transferred at low temperature into an extrusion cylinder and allowed to freeze. Samples for analysis are subsequently extruded from this cylinder through a die, and the desired length of extruded sodium is cut off by means of a special cut-off device and then allowed to fall into a sample holder. The sample is transferred into the amalgamation portion of the analytical apparatus in a manner that ensures that no oxidation of the sodium from gettering to amalgamation occurs.

In the first nine months of this program, several experimental problems have been encountered and solved. These problems include those concerning the water content of the alcohol used to dissolve the sodium oxide, the Karl Fischer titration, the possibility of interference from zirconium, zirconium oxide or mercury in the Karl Fischer titration, and the surface oxidation of the low-oxygen-content sodium sample.

The water content of the alcohol used is reduced to a minimum by the use of a column of Linde Molecular Sieves. The Karl Fischer titration has given a sharp and satisfactory end point. It has been found that the presence of zirconium, zirconium oxide, or mercury does not interfere with the Karl Fischer titration. The surface oxidation of the sodium sample can be overcome by proper sampling-transfer-tube design.

The most recent results obtained with zirconium-gettered sodium are given in the Table 1

Work is continuing to improve reproducibility although the above results represent a significant improvement over earlier results varying from 0 to 32 ppm of oxygen in a given batch of hot-gettered sodium.

Another object of the current program is to add known amounts of oxygen to the sodium samples and measure the completeness of recovery by the method under development. Table 2 indicates the results achieved to date in this recovery phase of the program.

Table 2—PRELIMINARY RESULTS ON
OXYGEN RECOVERY

Sample Size, g	Oxygen in sodium,* μg	O ₂ added, μg	Calculated total oxygen, μg	Oxygen found, μg
1.88	5	30	35	39
2.05	5	60	65	47
2.00	5	65	70	63

*Assume sodium contains 2.5 μg of O₂/g.

Additional work is being done on improving the efficiency of recovery by improving the apparatus.

It is too early to fully evaluate the method under development for the determination of oxygen in sodium. It has already been demonstrated that the method can be used to analyze for oxygen in the range of 0 to 10 ppm although the variation between supposedly duplicate samples appears to be greater than desirable. It is also evident that further work is needed on improving the calibration of the method by adding known amounts of oxygen to sodium samples. Additional data and experience with this method are needed, and improvements should and can be made on simplifying the apparatus. With some additional work, the proposed method appears to be capable of fully achieving the objectives originally described.

Fig. 1—Analytical apparatus for the determination of oxygen in sodium.

Fig. 2—Analytical apparatus for the determination of oxygen in sodium.

Fig. 3—System for purification of sodium by zirconium gettering.

Fig. 4—Sodium extruder and cut-off device.

The Rhapsodie Project

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The French sodium-cooled-reactor program is focused on the Rapsodie project. The Rapsodie reactor will be a plutonium-fueled fast breeder reactor. This type of reactor was naturally included in the French atomic-energy program because the large thermal reactors built by Electricité de France, the French state-owned electricity company, will produce plutonium.

The trend to increase the fuel burnup in these thermal reactors will also increase the Pu^{240} isotope content, probably beyond the limit admissible for recycle in thermal reactors.

Alternatively this plutonium with a high Pu^{240} isotope content can be used efficiently as fuel for fast reactors. The fast-reactor blanket, in turn, will produce high-quality plutonium, which is usable as fuel in the thermal reactors. Since the fast reactor can produce more fissile material than it uses, there is an additional reason to encourage their use for a long-range nuclear program.

The high power densities and attendant high heat fluxes lead to the necessity of liquid-metal cooling. Although the Rapsodie reactor will be the first French sodium-cooled reactor, it will use technology developed in France during the last 10 years. Several experimental sodium and NaK loops were used to develop sodium equipment.

The Rapsodie reactor characteristics were fixed in order to answer three purposes. These are as follows:

1. The reactor will be an experimental reactor. The study of its behavior at different power levels and in varied thermal and hydraulic conditions is the main aim.
2. It is hoped that the reactor can supply basic information for future fast breeder power reactors.
3. The neutron flux will be enough to test new fast reactor fuels.

Plans call for successive operation at 1 and 10 Mw and possibly at 20 Mw. A 1-Mw cooling system can be replaced by a second 10-Mw system and then by a third 20-Mw stage if it seems desirable. Every cooling system includes a radioactive sodium loop yielding its heat to a secondary nonradioactive NaK loop.

Plutonium will be used in the first core loading. Fabrication and irradiation stability limit the plutonium fraction to about 20 per cent of the fuel. Enriched uranium will be used in conjunction with plutonium to control the critical dimensions of the core.

Several fuels were investigated. A 70 wt% uranium–20 wt% plutonium–10 wt% molybdenum alloy is the most highly developed one. Slugs were fabricated and are now being irradiation tested in the EL-2 and EL-3 reactors. With this fuel a double cladding will be used; an inner niobium layer will separate the fuel from the outer stainless-steel clad to avoid the possible formation of a uranium–plutonium–iron eutectic during a high-temperature transient. A mixed

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*Assigned to Atomic Power Development Associates, Inc.

UO₂-PuO₂ fuel is also being developed. Also preliminary studies of a mixed uranium carbide-plutonium carbide have been made.

The maximum neutron flux will be 1.1×10^{15} neutrons/cm²/sec at 10-Mw power. At the same output the average core power density will be 200 kw/liter.

The reactor design work started in 1957, and a preliminary design was achieved in December 1958. The basic neutronic, thermal, and mechanical studies were made in 1959. It was decided to build a full-scale mockup of the reactor vessel and loops prior to building the reactor itself. This mockup was constructed in 1960 and will be tested by the end of 1961. The reactor-facility construction will start in the summer of this year.

Ten million dollars have been allocated for the research and development work, including the full-scale mockup.

The reactor facility encompasses the reactor and primary-loop containment, the secondary-loop building, the control building, the fuel-element and repair building, and the electrical-power-supply building.

The reactor containment is a steel pressure vessel designed to withstand 32-psi internal pressure and to contain all the fission products and radioactive sodium that might be released by a reactor incident or rupture of the primary system.

A 316 stainless-steel reactor vessel contains the core blanket and stainless-steel shielding. The vessel is supported at the bottom and is closed at the top by one fixed and two rotating plugs that are equipped with frozen seals. The closure is linked to the vessels by means of two large bellows and is supported by an independent structure.

The reactor vessel contains about 50 core subassemblies surrounded by 500 blanket subassemblies. The fuel bundle of each core subassembly consists of 37 fuel pins separated by spiral wire spacers. The upper and lower blanket bundles are made of seven blanket pins. The three bundles are contained in a hexagonal wrapper can. A blanket subassembly consists of a single long blanket bundle in a hexagonal wrapper can. Attached to the lower end of the subassembly can is a nozzle section that is inserted into two support plates. Although the sodium flows upward through the subassembly, no upper restraint is used since the hydraulic lifting force is inferior to the gravity force. The lifting force is inherently relatively small because of the low coolant velocity and pressure drop, and it is further reduced by the effect of a venturi tube located at the inlet of the lower support section. The subassemblies therefore expand and bow freely under their temperature gradients.

The sodium inlet and outlet temperatures will be 480 and 640°F, respectively, for the first core. The design, however, will permit a maximum outlet temperature of 1020°F when suitable fuels become available.

The sodium is supplied by a single nozzle that penetrates the bottom of the vessel on the vertical axis of the reactor. An annular orifice divides the sodium flow into a high-pressure flow that cools the core and a low-pressure flow that cools the blanket. Other circular orifices located on the lower support plate distribute the sodium through the subassemblies in order to have an approximatively equal sodium temperature at the subassembly outlet.

The operating sodium level exceeds the top of the subassemblies by about 3 ft, with an argon atmosphere above the sodium. Two natural-boron rods provide regulating control, and four enriched-boron rods provide safety and burnup-compensation control.

The primary sodium flows by gravity from the reactor vessel to the shell side of intermediate heat exchangers and then to the pumps. From the pumps the sodium flows back to the reactor. Two counterflow floating-head shell-and-tube type heat exchangers are used.

The secondary coolant, NaK, will be air cooled. However, it will be possible to replace the NaK-air heat exchanger by a steam generator feeding a turbine and electric generator if it appears worthy to study the control of a complete plant system.

The primary and secondary pumps are electric-motor-driven vertical-shaft centrifugal type pumps. Two electromagnetic pumps can replace the mechanical pumps for the 1-Mw loops if they are required. The primary sodium service system includes cold-traps, plugging indicators, and an overflow tank.

It has been decided to locate all fuel-handling mechanisms outside the reactor vessel. To initiate reloading the plug seals are melted, and the refueling port in the small eccentric ro-

tating plug is located above the chosen subassembly. A handling machine picks up the subassembly through the hole and transfers it into an intermediate storage cell located adjacent to the reactor vessel. There the subassembly is put into a pot containing molten lead. The pot is closed and the subassembly is carried from the reactor building to the fuel building by a second, smaller handling machine.

Another intermediate storage cell is provided for new subassemblies during the inverse loading operations. The fuel building includes a fresh-subassembly storage area, an irradiated-subassembly decay storage pool, and a radioactive-equipment repair area.

Panel Discussion

Present and Future Reactors

Panel Members

G W WENSCH, *Chairman, U. S. Atomic Energy Commission*, S SIEGEL, *Atomics International*, B I SPINRAD, *Argonne National Laboratory*, J YEVICK, *Atomic Power Development Associates, Inc.*, E JONES, *Consumers Public Power District, Nebraska*, and K P COHEN, *General Electric Co.*

WENSCH: We propose a panel discussion of the sodium-cooled-reactor development programs. We hope that if you have questions and you would like to have them discussed please fill out the cards which Mr. Tom Heckman is passing out now. Put your name and your company affiliation on these cards so that in the written discussion you will be given credit for your remarks. If we cannot answer them here, answers will be sent to you in writing.

Panel discussions are interesting mainly since they develop differences in viewpoint and healthy, honest controversy. Based upon my association with each of the panel members, I am well aware of a difference of opinion which each of them has.

We have heard discussed in this symposium the merits of sodium, and we are all in agreement that sodium reactors are very good.

However, I think that these different beliefs we have should be discussed, and I, for this session, rather than to be a proponent of sodium-cooled reactors am going to take the opposite standpoint. After each panel member has presented his opening statements, questions that I have and questions coming in from the floor will be asked the panel. We hope to put the panel on the defensive.

Just to keep the record clear so that you will know whether this is a planned panel or not, I must say that last night after the banquet I did give each member of the committee a list of the questions that I propose to ask this morning.

Dr. Siegel, would you like to start off with your opening remarks?

SIEGEL: We heard yesterday of the substantial progress on the development and engineering of large sodium-graphite reactors and other sodium-cooled systems which has occurred in the last few years. The operation of the Sodium Reactor Experiment (SRE) has provided experience on the practicability of working with large sodium-heat-transfer systems and with their maintenance and repair under normal and unusual conditions. The stability and the load-following capability of the reactor has been demonstrated. Superheated steam above 1000°F has been produced. The design and construction of the Hallam plant, the development and test of its components, and the work that has proceeded on the fast reactor programs have provided importance guidance to the design of the second-generation large sodium reactors.

Based on this information, it is possible today to construct and guarantee the performance of a current-status plant that would use uranium-10% molybdenum that produces fuel steam at

1250 psi and 850°F. It can be rather confidently estimated that the energy cost in such a plant would be about 7.5 to 8 mills/kw-hr.

In part this progress reflects increased knowledge, and in part it reflects the increase in the size of the plant about which we are talking compared to earlier plants in which the costs per kilowatt were naturally higher than those associated with a 300-Mw current-status plant. Incidentally the cost has been determined to be about \$225/kw; and, by adding the customer cost, it goes up to about \$250/kw. These costs are quite closely estimated prices upon which commercial proposals can and have been made.

Important developments continue in progress on fuels, materials, and heat-transfer components. Uranium monocarbide has shown very promising radiation stability, beyond 25,000 Mwd/ton. In the first tests these burnups were at moderate temperatures, in the neighborhood of perhaps 1400°F. Continuing work is in progress to perhaps establish the limits of the usable operation of this fuel.

The properties of uranium carbide are well matched to the requirements of a high-performance sodium-heat-removal system. Most particularly these properties relate to the good thermal conductivity of uranium carbide and its high melting point. The power density that can be achieved in a given slug of uranium carbide is better than that which can be obtained in either uranium metal, which, although it has a higher thermal conductivity, is usually limited for radiation-damage reasons to lower central temperatures, or, similarly, uranium oxide, in which, although a high temperature can be achieved in the center of the fuel, the thermal conductivity is poor. Uranium carbide appears to have an optimum combination of thermal properties, and it also appears to have the desirable long-time radiation stability.

Zirconium alloys having good high-temperature strength can be used for core components at 1100 to 1200°F, resulting in significant neutron economies.

Ferritic steels usable at 1200°F have been developed through commercial proof tests, and they can result in substantial system cost savings. New core configurations have been designed and will be subjected to test within the coming year in existing large facilities that the AEC has built at our laboratory and at other places.

The construction of Hallam and the development and utilization of large stainless-steel bellows that are incorporated in some of the equipment in that plant can permit substantial savings in the layout of large sodium-heat-transfer systems. Advanced boiler models are being constructed for test in a new test facility that is approaching the construction phase now. Furthermore, modifications of the SRE are now being designed which will permit raising its power to a higher level of about 45 Mw and, possibly, its temperature to about 1200°F, rather than the 1000°F which has been experienced in the past so that in the next few years it will provide reactor-operation experience at this advanced temperature. I might comment that a great deal of loop work has been done at temperatures approaching 1200°F. I also might call attention to the fact that in another of the AEC programs for small compact reactors, a small reactor has been run with the previously mentioned materials at 1200°F for a period of over a year with no concern having arisen about material properties and behavior.

Based on this work in progress, preliminary designs have been prepared for a large 360-Mw(e) plant, which we believe could be in operation in about 1968 provided the necessary work is pursued with vigor. This plant incorporates the developments that I have mentioned. It would use uranium carbide fuel and could produce steam at 2400 psi and 1050°F with reheat. Because of these good steam conditions and because of the absence of stack losses, which are appreciable in a conventional thermal plant, its heat rate would be under 8000 Btu/kw-hr. The energy costs from such a station would be about 6.1 to 6.5 mills/kw-hr, depending on the details of utility costs. The equilibrium fuel cost is estimated at 1.9 mills/kw-hr; and with reductions in cost expected during the life of the plant from technological advances in fuel technology and in fuel fabrication, it is expected the fuel cycle could be reduced to something like 1.5 mills/kw-hr.

This plant would be competitive when measured during its life against a conventional plant that would be built at the same initial time if it is assumed that the conventional plant would have a slightly rising fuel cost.

Even if we assume constant fuel costs for the conventional plants and accept the reasonable expectation that there will be some decrease in nuclear-fuel costs, the plant that I described, which is within our grasp if the technical work in progress continues with the expected success, can definitely have an economic advantage over the conventional plant with which it is compared at the same time. The economic advantage would be some \$20 million, cumulative over its life.

Projecting further into the future, we believe that the longer range advances in the technology and performance of sodium-cooled reactors appear to be associated with nonthermal systems. We have work in progress on the neutron physics of fast-reactor systems based on the U^{233} cycle, which is being supported by the group of utilities known as the Southwest Atomic Energy Associates. In this symposium we have heard a great deal about the work in progress on fast reactors based on the plutonium- U^{238} cycle.

We are conducting an extensive series of critical experiments using U^{233} , covering a broad range of nuclear fission energies. The initial information we have obtained confirms the promise of this system to achieve high neutron economy; fuel costs ultimately will approach 1 mill/kw-hr. Preliminary designs for a high-power-density core are being evaluated, and initial work is in progress on novel fuel configurations based on thorium and uranium carbides. However, I think it is clear that for this system, as well as for any of the fast reactors, substantial additional work on the whole problem of closing the fuel cycle and on fuel reprocessing has yet to be done to assure us that fuel costs as low as 1 mill/kw-hr can be obtained.

These advanced systems, if the promise is truly fulfilled, appear capable of becoming practical in the 1975 period and of producing power in the neighborhood of 5 mills/kw-hr. This attractive end result of the program now in progress will result largely from advances in neutron economy, reduction in capital costs, and especially from the exceptional thermal performance of sodium-cooled reactor systems.

WENSCH: Thank you, Dr. Siegel.

Mr. Yevick.

YEVICK: To understand the present and future programs of Atomic Power Development Associates, Inc (APDA) as they relate to sodium-cooled fast reactors, I shall present a review of the past programs of APDA.

As set forth by the Atomic Energy Act of 1954, it is the declared policy of the United States that the development, use, and control of atomic energy shall be directed to promote world peace, improve the general welfare, increase the standard of living, and strengthen free competition in private enterprise.

To further this policy, APDA was established on Mar. 10, 1955, as a nonprofit membership corporation. Currently 30 electric-utility companies and 12 industrial and engineering organizations comprise the membership of the corporation.

The basic objectives of APDA, as stated by its by-laws, are (1) to study the potential of economic production and utilization of atomic energy, (2) to engage in research and development activities leading to the realization of the potential, and (3) to contract for work to be done by others.

To further the above objectives APDA is required by its by-laws to educate its members, employees, and other interested people; acquire and provide information; advance knowledge and learning; stimulate research and development; and maintain and operate laboratories and other facilities required to implement the basic objectives. In keeping with the furtherance of the objectives of APDA, the present policy on patents, as set forth by the by-laws states: "Any person or corporation within the United States shall be entitled upon application and to the extent permitted by law to be granted an appropriate license on a nonexclusive and nondiscriminatory basis as to royalties or otherwise to make, use, and sell any such invention."

The past, present, and future programs of APDA have been based on the preceding stated principles. The member companies have provided the necessary funds for research and development expenditures of \$21 million from the date of incorporation through Dec. 31, 1960, and have committed themselves to an expenditure of over \$3 million in 1961.

The APDA program to date has placed major emphasis on research and development for the Enrico Fermi Atomic Power Plant, which incorporates a sodium-cooled fast breeder reactor having a capability of 150 Mw(e) generated from 430 Mw of heat.

The construction and operation of the Enrico Fermi plant is the first step in the program to the economic production and utilization of atomic energy. Additional steps have been taken as part of the program to reach this objective. These include research and development in the following areas: (1) nuclear physics, (2) fuel, (3) structural and shield materials, (4) sodium and sodium components, (5) the sodium-cooled solid-fueled Advanced Fast Reactor, and (6) the sodium-cooled Paste Fuel Fast Reactor.

Atomic Power Development Associates, Inc., has had two major tasks in the development of the Enrico Fermi plant: (1) the design, development, and erection of the major-components test facility at the plant site and (2) the research and development for the equipment and systems in the reactor plant, including their satisfactory operation. The major-components test facility is that portion of the nuclear plant which includes the reactor and one primary coolant loop. The purpose of the facility is to test the major nuclear components. Research and development work in support of the Fermi project has included fuels, cladding and other materials, refueling, sodium, sodium components, core and blanket physics and design, heat-transport system, auxiliary systems, and service systems.

A comprehensive program to develop improved fuels for the Advanced Fast Reactor was started in 1956. Evaluation of metal alloys, ceramics, cermets, and cladding materials included fabrication processes; mechanical design; and thermal, hydraulic, nuclear, and economic analyses. A fast reactor core design parameter study for the AEC, correlating nuclear, hydraulic, thermal, and throughput characteristics for metal alloy, ceramic, and cermet fuels concluded that the optimization of design parameters can measurably reduce the fuel-cycle cost.

Atomic Power Development Associates, Inc., completed a study in April 1959 reviewing the status of the fast reactor for power production. By utilizing the design and construction experience from the Fermi plant and other reactors and applying technology developed in recent research and development programs, it was determined that a fast breeder power plant could be designed which would operate at substantially less cost than the Fermi plant. The plutonium-fueled plant with a gross output of 300 Mw(e) generated from 775 Mw of heat could be constructed at an estimated cost of \$69 million or \$230 per gross kilowatt, based on 1959 prices. The total cost of energy was estimated to be 9 mills/kw-hr (net), including a fuel-cycle cost of around 3 mills/kw-hr, based on private ownership of fuel.

A subsequent study, completed in March 1960, applied later developments to a 150-Mw(e) atomic power plant incorporating a plutonium-fueled fast reactor. This plant can be built for an estimated capital cost of \$36 million or \$240 per gross kilowatt, based on 1959 costs. The total cost of energy is 9.4 mills/kw-hr (net), including 5.9 mills for capital charges, 2.1 mills for fuel-cycle costs, and 1.4 mills for operation and maintenance, based on private ownership of fuel with a 10 per cent annual rental charge. The costs are based on a value of \$12/g of plutonium. The total cost of energy is reduced to 8.7 mills/kw-hr if the fuel is owned by the government and rented at a 4 per cent annual charge.

Work on a paste-fueled reactor was started in 1955. Fuel-cycle cost studies indicated that a sodium-cooled fast breeder reactor having a mobile fuel and a mobile blanket that continuously move into and out of the reactor, together with on-site fuel reprocessing and fabrication, would produce low-cost power.

In principle, small spherical particles containing fuel are fed through the core as a paste of 60 vol.% solids in sodium. On leaving the core the paste is made into a sodium slurry that is circulated at low velocity to the reprocessing plant. A separate cooling system is used to remove the heat from the reactor because transporting the mobile fuel material through an external heat exchanger to remove all the reactor heat requires large fuel inventories. Supplementing the advantages of high burnup, low fabrication cost, refueling simplification, and on-site reprocessing inherent in mobile-fueled reactors, the paste-fueled reactor has the additional advantages of the fuel being handled in small volumes; the use of sodium as a noncorrosive carrier; low fuel velocities, which reduce erosion and particle attrition; and fuel whose concentration is not affected by loss of pump power.

Six years of research and development have demonstrated the feasibility of a settled density bed and have established that paste can flow without a significant change in the concentration of solids under steady-state and transient conditions and that paste flow rates can be controlled. Plans have been made and are being extended for three basic paste reactor programs: (1) simulated core tests, (2) fuel irradiation, and (3) in-pile paste-fuel system testing.

The present APDA program in support of the Enrico Fermi plant includes (1) the completion by the summer of 1961 of the major components tests, (2) responsibility for the satisfactory operation of the reactor and associated mechanisms and the heat transport and service systems, (3) research and development for advanced types of fuel loadings, and (4) plans for a program of postirradiation examination of fuel.

The present advanced solid-fuel technology work consists of an evaluation of fuel development for the Enrico Fermi reactor and other fuel development programs at Oak Ridge National Laboratory, Argonne National Laboratory, General Electric Company's Atomic Power Equipment Department, Hanford Atomic Products Operation, Atomics International, Nuclear Materials and Equipment Corporation, and Nuclear Development Associates, Inc. The results will be used to determine the best fuel system for the Advanced Fast Reactor.

The Advanced Fast Reactor studies will include optimization of core, blanket, and thermal cycles; simplification of refueling and decay storage schemes; use of improved heat exchanger and boiler designs; and the determination of the safety of fast reactors, with particular emphasis given to reactivity coefficients, reactor kinetics and stability, and fuel meltdown.

The objective of economic nuclear-power production is to provide electric power to the home, office, and industry at a cost per kilowatt hour sufficiently attractive to induce the user to install and operate such items as mechanisms for space cooling, space heating, dust collecting, air purifying, humidity control, germicidal control, water heating, and improved lighting, all powered by electricity. In 1959 the average residence in the United States consumed 3500 kw-hr of electricity. Lower rate structures invariably result in increased utilization with a consequent rise in the average consumption per residence, conceivably to as high as 15,000 kw-hr of electricity per year in the near distant future. A lower rate structure can be accomplished by a reduction in the cost of producing electricity and a reduction in the cost of electrical-power distribution. Relatively few years of effort have resulted in substantial reductions in the cost of producing nuclear power. A few more years of effort should result in further substantial reductions, which would provide the cornerstone in the lower rate structure. The other task, that of reducing the cost of distributing electricity, will be accomplished as a result of improving annual use factors and innovations in the design and use of distribution equipment.

The stairway to the main goal has three steps. First, we must prove the technical feasibility of nuclear power in experimental and prototype reactors. Second, intensive research and development is necessary, using the initial plants as the basis to develop nuclear power plants whose costs are competitive with fossil-fueled plants. The third and most important step is to reduce power-production costs by bold approaches to the design and utilization of the heat source and the means of transmitting the heat to the power-conversion apparatus. The third step must be combined with changes in the concept of distributing electrical energy from central sources. One-half to two-thirds of the cost of delivering power to the site of utilization is in its distribution.

The first step has been and is being accomplished. The second step for fast reactors should have the period 1968 to 1970 as its goal. The planning for this step has been delayed too long. The third step, for which planning should be under way, should have as its time goal the period 1975 to 1980.

WENSCH: Thank you, Mr. Yevick.

Dr. Jones.

JONES: Our organization, an operating utility must, of course, examine another aspect of the program, that is, the customer of these systems.

As we have been progressing in the field we have been asked from time to time just why we were participating in the atomic-energy program, and, in particular, why we picked the

sodium-graphite system. Depending upon the occasion, I sometimes plead temporary insanity; but seriously I think the program as a whole in which we are participating has some real value.

As our Superintendent of Power Supply, Mr. Bill Venable, pointed out yesterday in his discussion, we are a relatively small electric utility, and therefore we do not have funds available for anything other than operating power-producing equipment. Nevertheless, by participating in the atomic-energy program with an adequate back-up system, we felt that we could contribute to the program development; and, from our own standpoint, we felt that we would be able to establish an experienced staff capable of deciding whether the next power plant should be based on the atomic-energy approach or whether we should again build another conventional power plant.

In examining the balance between plants, we must be able to design high-efficiency systems. This factor leads then to the liquid metals. That was pointed out by Dr. Hilberry in his banquet address last night. The superheated steam available from sodium systems also permits the use of relatively standard power-plant equipment; this is an advantage in itself. More importantly we have found in the Hallam program that it also permits a convenient and relatively inexpensive system for backing up the nuclear facility with conventional steam sources during the years when the development work has to be done.

We feel that the major improvements in power costs in the atomic-energy field are going to come through fuel improvements, and there are probably fewer restrictions on using sodium as a coolant for the variety of fuels that might be feasible because of the chemical compatibility of sodium with the majority of the proposed fuels. Therefore we feel that we have a broader field in which to realize improvements in the fuel portion of the program.

We continue to be optimistic about the ultimate outcome of sodium-cooled reactors in the race for economic nuclear power, but we find difficulties on nontechnical issues that make the use of sodium somewhat more complicated. We have been working with the insurance companies since 1955, and I think that probably in the insurance field we have had less cooperation and less understanding concerning sodium systems than in any other phase of our program. At this stage in the development of sodium-cooled systems, in comparing notes with some of the others, we have the general feeling that the insurance pools are penalizing the sodium system simply because it is sodium.

I think that from an operating utility standpoint we have increased problems with permits and licenses associated with sodium because it has a nasty reputation. Again this is a side issue in which permits and licenses are already becoming a major part of power-production costs for atomic-energy plants.

There also is the question of spent-fuel reprocessing, which has been previously mentioned. With the reprocessing costs artificially described by regulations in the AEC manual, one has little confidence that these regulations represent an accurate yardstick for costs in the future. As a result anyone building a plant today who is interested in operating that plant 20 or 25 years from now has little confidence that those pages in the book will not be changed. The difficulty of getting fuels reprocessed cannot be overcome until it is done commercially so that some kind of a cost comparison can be made on a sound basis.

Again, in the over-all view of the program, we feel that we are participating in an excellent portion of the AEC Power Reactor Program, and we are anticipating continued confidence in it.

WENSCH: Thank you, Dr. Jones.

Dr. Spinrad.

SPINRAD: In view of the fact that I have had an opportunity to hear Dr. Hilberry's speech last night, I shall depart a little from the prepared remarks I had. If you notice some similarity between some of the things that I have to say and Dr. Hilberry's remarks, chalk it up to excellent rapport within Argonne's organization.

I cannot miss this opportunity to be controversial. In spite of the fact that I am sure that both Dr. Siegel and Mr. Yevick quoted very conscientiously derived costs, I still have the tendency to believe that costs for reactors are artificial before they are built. Such beliefs are strengthened by a recent announcement that out of 10 superheating reactor concepts studied seven of them gave power costs of precisely 7 mills/kw-hr. This similarity is more than

coincidence to me. Therefore I think we have to change the rules for scoring, and I say that we not only need to figure out how many mills the power will cost but what is the credibility of these mills.

As befits the theme of this program I shall limit myself to sodium reactors for central-station power generation. Although sodium reactors have application as remote package systems, particularly as high-temperature systems in space, I shall not discuss these uses in detail. Some of the lead technology for central-station power application does appear, however, to be logically derived from development of these futuristic applications.

It appears that sodium reactors of the current generation have higher capital costs than conventional systems. These costs are all directly traced to our inexperience with sodium components, particularly at high temperatures. Thus future economies in capital costs will come either from increased knowledge and improved technology of sodium components at high temperatures or from accepting limitations on temperature. A third possibility exists if we can become more imaginative in our choice of electrical generating concepts.

Sodium-reactor fuels must be less expensive than water-reactor fuels if the sodium reactors are to be competitive. Continued development of carbide fuels is the approach indicated by a concentration on higher temperatures. Continued use of metal fuels can be contemplated if the targets on sodium temperatures are relaxed. Of all the sodium reactors, the fast reactor has the greatest potential for ultimate economy of fuel; but to realize this economy it must run at very high specific power. I must hasten to state that I believe that this development will ultimately come to pass. However, I shall leave further remarks on fast vs. thermal reactors to the question period and consider other conceptual directions.

The first of these is the concept of lowering the sodium temperature to that point at which uranium-metal fuels and low-alloy steel components have sufficient strength and integrity to meet operational requirements. These temperatures are in the range of 300 to 500°F. Reactors with temperatures in this range do not give exciting thermodynamic efficiency. However, they are cheap to build and to fuel; and, indeed, this is the temperature range of most of our water reactors. Therefore sodium reactors do not compare too unfavorably with their competition.

This still leaves us with the headache of transferring heat from sodium to a working fluid. The fact that sodium and water are violently reactive imposes difficulties that probably end up as expenses in any scheme that involves transferring heat from sodium to water. However, other working fluids are possible and should be examined in this connection. Benzene is a working fluid that has had appreciable development, and it is compatible with sodium. In spite of its low thermodynamic efficiency, I therefore think that a sodium-cooled reactor with relatively mild thermal conditions and which is used as a heat source for a benzene turbine might have interesting possibilities.

At the other extreme, investigation of the possibility of handling sodium at high temperatures above 1200°F might also serve to eliminate the difficulty of coupling sodium to the steam cycle.

Three possibilities can be suggested. One is the rejuvenation of the mercury turbine in central-station applications. This is a development that already exists. It is simply required to be connected to the sodium reactor. The second, which obviously is more advanced and with more chance involved, is the use of direct-cycle boiling with sodium or NaK as the working fluid. The third, which has the element of chance even more involved, is the production of electricity by means of thermally regenerative fuel cells.

It is also worth noting that if we decide to continue to utilize steam, the current 1200°F temperature limit asked for sodium reactors is probably too low. Many modern, and probably more future, power plants operate at the steam temperature of 1150°F. If one adds the various temperature drops in going from the primary working fluid to the steam and if, in addition, he considers the fact that the primary working fluid usually makes its heat transfer after mixing the fuel outlets, which vary rather widely in temperature, he can see that it is necessary to develop components, particularly fuel structure components that can go well beyond 1400°F, if modern turbine conditions are to be obtained.

For all these concepts it is clear that carbide fuels are preferable and that improvements in structural materials are required. However, I am unwilling to admit that the required developments are intrinsically more difficult than the development of steam generators to be both economical and in the long run reliable.

Mr. Yevick has touched upon one very interesting point that I think requires very much more careful examination; that point is the question of the use of fuel materials that may have the possibility of allowing some contamination of the working fluid. It appears that if we work with sodium we are working with a material that becomes violently radioactive. It is true that the induced radioactivity dies away; nevertheless, one of the greater penalties on allowing primary system contamination certainly seems to have disappeared when we use an activating coolant like sodium. Therefore I think that such materials as paste fuels allowed to come into contact with the major coolant and other concepts of vented fuels have a great deal of applicability in the future and might serve to minimize future fuel costs.

Dr. WENSCH: Thank you, Dr. Spinrad.

Dr. Cohen.

COHEN: I think it is perhaps significant that in this discussion that my colleagues on the panel have given there is a great deal of talk about different reactor concepts, new materials, new fuels, and new temperature ranges. This is symptomatic, I think, of the fact that we are at the beginning of a long development. Now, when one thinks of a reactor system being developed in which it turns out that he can use six different fuels, seven different temperature ranges, a cladding that can be one of three materials, and a coolant that can be sodium or maybe lithium, I think he can call it a not well-characterized system; and a system that is not well characterized is one that is faulty and immature.

I think that we shall be able to find how well we are doing in the sodium systems, which I believe have a place in our power-generating activity, when well-characterized systems are in evidence and when people begin talking about one instead of four fuels, etc.

I notice all my friends find very obvious advantages in particular things. One would be very dense indeed not to see that plutonium carbide, for example, has a higher conductivity than plutonium oxide and not to see, for example, that deuterium absorbs fewer neutrons than hydrogen and that lithium has a higher heat capacity than sodium; but does any of these things lead us in any useful direction. When one is trying to climb a mountain range what one wants to find out is the path to get from here to there. That is the first and most important question, and similarly in our program the first and most important question we have addressed to ourselves is how to get from here to there.

Some time ago, at the time of the Seawolf, we became interested in sodium systems; and, when I first came to the General Electric Company, I was indoctrinated by people who had had experience with the Seawolf. They regarded sodium as a thermal and a fast reactor coolant. However, after working for some time in the thermal reactor field on sodium-graphite reactors, we decided that attractive as this concept was it was not sufficiently different in its basic fuel economy or in its ultimate power costs to represent a real advantage over the water-reactor systems that we were currently developing.

We therefore decided to look further along the line for uses of sodium and other liquid-metal coolants. On this basis we abandoned work on the sodium-graphite concept and decided to concentrate all our effort on the sodium-cooled fast reactor.

Ultimately, perhaps a sodium-cooled fast reactor using the oxide fuel that we talked about will be developed in a later generation. It might be 12 years from now and be carbide fueled and lithium cooled. The thing that we are talking about now is a means to get from here to there; and, to get from here to there, we have to have things about which we know something, with which we can work with, and of which we can make, let us say, a ton. It is on these bases that we are talking about oxide-fueled sodium-cooled fast reactors.

The prime incentives for the development of fast reactors are the ability to utilize completely our fissionable-material resources and the potential for low fuel-cycle costs. In view of the slow development of nuclear power and the 30-fold increase in United States ore re-

serves of fissionable material since 1952, only the potential for low fuel-cycle costs is an important present incentive. Now, I think everybody knows this and everybody gives lip service to it, but I see very few people trying to design reactors around it.

The well-recognized capability of fast reactors to operate with a net gain of fissile material is insufficient to ensure low fuel costs, but it is so attractive to people that I believe most reactor designers spend their time trying to see whether the reactor will breed or not. Then they secondarily look to see what the fuel costs will be when they are all finished.

Total fuel-cycle costs are made up of a number of components in addition to depletion charges, and most of these components are a function of fuel burnup. There are two routes that can be taken: either to develop reprocessing and fabricating procedures so cheap that frequent reprocessing can be tolerated—and this implies on-site fuel fabricating and reprocessing facilities such as the very ingenious things that Dr. Lawroski showed you yesterday—or to develop very long-life fuel elements so that the costs of occasional reprocessing are divided by a very large energy output. This latter route permits off-site fuel fabricating and reprocessing. We have elected to pursue this route.

The kinds of fuels that appear necessary to us for very long burnup are ceramic fuels. Recent in-pile tests that we have performed for the AEC have shown that the plutonium oxide uranium oxide fuels clad in stainless steel can withstand burnup of up to about 100,000 Mwd/ton, or about 13 at.%, without mechanical damage. Therefore this route now appears feasible. If and when it turns out that one can prove the same things with carbides, I shall be prepared to state that that route also appears favorable.

To use this high burnup capability of ceramic fuel, two problems have to be met: a reliable negative temperature coefficient of reactivity must be obtained, and the operational swings of reactivity in the reactor core must be limited so that excessive amounts of reactivity are not held by control rods. These two problems are safety problems.

In the small metal-fueled fast reactors previously designed, such as EBR-1, EBR-2, and Fermi, the prompt negative power coefficient is furnished by thermal expansion of the metal fuel. This cannot be relied upon in ceramic-fueled reactors, particularly in the large reactor cores that are over the 3000 liters necessary both for large power outputs and to limit core reactivity changes.

We have found it possible to design large, low-leakage oxide-fueled fast reactors which have attractive economic performance and which override negative power coefficients due to the Doppler effect. The large Doppler coefficient occurs because of the degradation of the fast neutron spectrum and because of the high ratio of uranium-plutonium atoms. Both of these characteristics result mainly from the low neutron leakage of the large core. The moderating influence of the oxygen atoms is an important contributing factor.

In a typical large fast oxide reactor, about 20 per cent of the fissions occur in the resonance region, below 10,000 ev. This gives a power coefficient that we consider more reliable than those depending upon mechanical movement from thermal expansion of the fuel.

The major requirement necessary to bring these potential advantages to the point where they can be incorporated in future power systems is the adequate demonstration of the nuclear characteristics, particularly the effectiveness of the negative Doppler coefficient.

Since the fast oxide reactors differ markedly from existing experimental reactors in size, in nuclear and thermal characteristics, and in the use of plutonium oxide fuel of low plutonium-to-uranium ratio, it now appears that a new experimental facility is required.

I think that by the time we are going to want to incorporate these fast reactors in our power systems we ought to consider a power of about 500 Mw(e) and up. This is going to mean that the reactors are going to be very large and that they are going to have very large cores. I think that the metal-fueled-reactor concepts will also consider this kind of effect.

WENSCH: Thank you, Dr. Cohen.

Now, we shall discuss questions from the audience.

Audience Participation

WENSCH: The first question is addressed to Dr. Siegel.

QUESTION: Dr. Siegel, it is a little puzzling to a neophyte, but Dr. Spinrad implied that fast breeder reactors are superior to thermal reactors; and Dr. Cohen said that his company had undertaken an extensive evaluation and that they have concluded that sodium-graphite reactors did not seem to be feasible because of the economics. What are your comments concerning these statements?

SIEGEL: Well, I think the time scale is the essential aspect of this question. As I indicated in my own remarks, we agree that the long-term potential of the sodium-cooled or liquid-metal-cooled systems probably is associated with faster systems and that these, provided the whole fuel cycle is thoroughly examined, seemed to offer the prospects of low fuel-cycle costs; but, for matters that one can be reasonably definitive about today and in the next several years, the prospects for lower fuel-cycle costs appear to be associated with a reactor such as the thermal sodium-graphite one. The figures I gave for a plant that could be in operation in 1968, 6.5 mills/kw-hr energy cost, show a fuel cost of about 1.9 mills/kw-hr. There are not too many uncertainties associated with that.

If we demonstrate to our satisfaction that an average burnup of 25,000 Mwd/ton of the fuel can be achieved and we continue to learn how to fabricate an adequate uranium carbide fuel, then the fuel-cycle cost will have few uncertainties associated with it; and I think that, as a near-term goal, it is something we can accomplish with a few years of effort directed along a fairly well-defined course of action. I think it is a more realizable goal today, even more realizable than longer term courses. There will be further goals that we will have to achieve once we have achieved this first one.

My main point is that I think today we can be more definitive about relatively low fuel costs in the, for example, 2-mills/kw-hr region for a thermal reactor than we can for the other systems, that is, for the faster systems; but I concur with the belief that if we solve more of the problems and demonstrate the long fuel life, rapid fuel reprocessing, etc., that are necessary for the fast breeders the fast breeders do offer ultimately the possibility of a lower fuel-cost situation.

WENSCH: Thank you.

One thing sort of puzzles me in this fast reactor program. I listened to Dr. Cohen talking about the high burnup and the long-life fuels as being the objectives for which his group is striving; and, after I listened to Dr. Lawroski yesterday, it seemed to me that ANL was really looking for the short-time burnup fuels with rapid reprocessing. Are these two approaches contradictory?

Dr. Spinrad.

SPINRAD: I do not think that they are as contradictory as it would appear from your statement. There are real advantages to be gained in both directions. If one can make the reprocessing or turn-around time of the recycle fuel shorter, he can save a good deal of money on inventory costs in the chemical plant. If one can make the in-pile time long he can clearly save a good deal of money on the fabrication charges. Both of these are goals that one should wish to achieve essentially simultaneously. Additionally, I do not believe that one can claim a clear-cut advantage for one type of fuel over another at this point in time in terms of allowable burnup.

If I can recapitulate the fuel situation, it appears that as one goes to very long burnups all fuels become essentially broken down. When one goes beyond 50,000 Mwd/ton and into the region of 100,000 Mwd/ton certainly this is true. There is a good deal of fission gas released and a good deal of fuel growth; and in each case oxide metal and probably carbide are formed. We end up having to rely upon the integrity of a very stout jacket. In a word, fuel of any sort ends up being bad in its own pressure tube.

The point of all this is that if one can find the right pressure tube for any given type of fuel, then one has to look at other advantages and disadvantages of the particular fuel species in order to determine what is the best to use. At the moment we at ANL are still considering

metal fuel elements because, of course, we are hopeful that we can find the right pressure tube in which to put metal. It has nuclear advantages that none of the ceramic fuels have.

The carbide appears to be an intermediate case between metal and oxide, both in terms of nuclear properties and in terms of thermal conductivity; and it might also be intermediate in terms of retention of fission products. It seems to allow jacketing in more conventional materials than the metal does, and therefore it looks like a convenient second choice for the long run. The oxide certainly allows jacketing in stainless steel, and therefore one can say it is current status from that point of view. However, it has disadvantages in terms of breeding ratio—which means money, regardless—and in terms of process and fuel-fabrication costs; therefore we are really not very excited about it and consider it only as a last resort.

WENSCH: Dr. Cohen, do you agree with Dr. Spinrad?

COHEN: I think it is fairly obvious that I do not. The nuclear advantages that Dr. Spinrad discussed, he is counting twice. That is what I said before.

SPINRAD: How?

COHEN: Somehow or other you obtain your mills per kilowatt hour, and then you have a nuclear advantage. You are just adding those up twice. This is just a question of how many mills per kilowatt hour it turns out to be. I do not know how you figure out that there are cost disadvantages in fabricating and reprocessing oxide fuel. So far as I know, oxide fuel is the only one which has an established procedure for either of these two operations.

SPINRAD: May I be allowed to rebut on that?

I have recently tried to determine what the established procedures are for fabricating and processing oxide fuels. Dr. Cohen also has interest in these procedures for the water-reactor projects. If one examines the problem carefully, he discovers that there is only one type of fuel for which established fabrication and processing procedures are indeed available. These are the metal fuels. These procedures have been performed for almost 20 years since they had their origin, and they have been improved from time to time.

The various processes for making oxide fuels are very much in the development stage in the AEC program. I think it involves approximately \$10 million a year. They certainly can not be considered as an established cost sort of business. In addition, these processes do not seem to allow inexpensive utilization of plutonium since almost all of them require a form of human intervention at one stage or another in the process. This factor makes it very difficult to conceive of doing the job in a glove box, much less a shielded cell. In fact, if one does look into the possibility of automating the processes for manufacturing oxide, he soon comes to the conclusion that the sort of equipment needed represents an investment that is a very severe penalty on the fabrication, primarily because if it is used once for plutonium it becomes dirty and dedicated to that use and it is no longer a competitive method for making oxide fuels. Therefore one has a rather large and extensive facility that is reserved for a rather small fuel market.

WENSCH: Dr. Cohen, you certainly must have an answer to this because it seems to me that all our fuels really should contain plutonium in the fast reactor program, and that is one point I do not quite understand. What would you like to say?

COHEN: Well, it seems to me that it is astonishing how two people can look at the same set of facts and come to such different conclusions. Let us put it this way: Dr. Spinrad, I am perfectly satisfied to have you pursue the metal route; and I do not know that I have the missionary zeal to convince you to go on the other route.

SIEGEL: I should like to ask a question here. You were talking about the oxide fuels. If the capital costs can be reduced in a reactor plant by high-power densities—this means smaller systems—and liquid metals are used in order to get the highest heat transfer or the most effective heat-transfer coolant, how does this then fit in with the use of oxides, which have poorer thermal conductivity?

COHEN: I think the question comes in as to how much fuel subdivision there is. Like most other things in life, it is a balance. I could even conceive that if I had sufficient power density my course might go up because I have quite a problem getting my coolant to take this away. All I can say is that if 1000 kw/kg on the fuel is obtained, then that seems to me to be a fairly respectable power density. What I am trying to say is that I have no measure other than economic.

WENSCH: Thank you. John, I think you must have some opinions on these questions. I know that you were very interested in obtaining a high burnup fuel for Fermi, and here we have one already at hand.

YEVICK: Well, you remember, Glen, in 1959 the AEC, APDA—ANL, and LASL worked up a 10-year fast reactor program; and for the fuel development statements were made: we shall consider metals for X number of years to see if we can lick the problem of dimensional changes by things like canning or, possibly, by alloying. I think that ANL and the United Kingdom Atomic Energy Authority are considering strength cladding. We saw some figures yesterday on that uranium—2% zirconium which looked fairly good. So there is a good possibility that metal alloys still may come into being, and it does not take very much for metal alloys in relation to burnup to make it economic. Anything above two per cent begins to look good. At APDA we are in an industrial sort of a complex so in the meantime we have to look at economic power as soon as possible. We are in agreement with Karl about this, that we want to achieve economic power. If we get breeding with it, fine. It is a bonus.

As an interim measure at Fermi, we are going to utilize a uranium oxide, which Oak Ridge has developed for us, that is fully enriched in the core as a cermet. We are looking into the future for further cores and other reactors. We have considered the oxides, and we have an open mind. Whatever fuel is best developed at the particular moment, that will be our reference design. We certainly think that, at the present, the oxide has a very good potentiality because the Navy people, who essentially have done the most work on it, have come up with fairly good results. I think that the GE work in the past year on their capsules has indicated that they get exceptionally good burnup although there are some problems involved in it.

Our studies today have been based upon future reactors, that is, have been based upon the use of plutonium oxide and uranium oxide mixtures. Certainly the carbides appear to be very good, but at the present time the carbide program probably is in a stage where the oxide program was many years ago.

WENSCH: Thank you.

The next question is addressed to Dr. Jones.

QUESTION: Suppose that both water-cooled and sodium-cooled reactors are economical for power production, which of the two general concepts would you select for power generation, and what reasons do you use as the basis for your selection?

JONES: Well, assuming that the question was as you say, that they are feasible from the standpoint of economics, I think from the utilities standpoint it would not make any difference. However, I think that, as a general rule, we have a tendency to leave out many things that go into the economic study. I would suppose that if one were faced with such a decision, simply because he would have had some experience in one field or another would lead him to stay in that field because that is where his experience is. However, if they are truly economically competitive in all the costs, both direct and indirect, and are involved in the reactor program and if that should make them equal, then it would make no difference.

WENSCH: Thank you. Dr. Cohen, your company represents one that is fairly well known in the nuclear field, particularly in the field of water reactors. What is your interest in the sodium-reactor program? Your company probably has a large part of the market for water reactors already, and they seem to be doing very well.

COHEN: Well, that is certainly a very generous remark. I think I could talk a little about the general relation as we see it between the water reactors and the, let us say, less advanced—or the more advanced, depending upon from what viewpoint from which one looks at it—reactors.

We feel, and I think all our experience shows, that it takes a great deal of time for a concept to mature, that is, to have it proved out, to know what its economics are, to be able to warrant it to somebody, and to give them the faith that it will perform.

If economic nuclear power in an important fraction of the U. S. domestic market by plants that will go into operation in 1970 is to be achieved, in my opinion, it is going to have to be done by utilizing one of the hydrogen—moderated reactors because these are the reactor systems whose technology is familiar enough, whose components are well enough established, and for which the manufacturers are set up for sources of supply. We do not have sufficient knowl-

edge on chemical reprocessing, for example; but I think that we know how to do it. In my opinion, these reactors are what is going to make the nuclear industry

Now, in examining the long-range competition, which is with improvements in fossil fuel, as well as with further reactors, I have set up a working hypothesis that the advantages in water technology will sort of disappear exponentially on the grounds that eventually people will learn to do as cheaply with sodium what they now can do with water. The present water technology will certainly give a power station at less than \$225 to \$250/kw installed. Thus there is a difference in cost at the present time between the sodium reactor and the water reactor. It would be foolish, however, to rely on this lasting forever when one begins to look at the ultimate fuel-cycle potential. Now, it is a striking fact that the reactors that are best from the ultimate fuel-cycle potential, to wit, the fast breeder, at the present time has conspicuously the worse fuel costs, which shows that it would be premature to pursue this argument now; but, in the decade after 1970, I believe that the physics part of the fuel cost will, in fact, be the rock toward which everything else will sink. We are going to learn to do all the fabricating inexpensively enough and all the operations inexpensively enough so that it will come down against this bedrock of the physics cost. At that time we think a system with superior neutron economy will be worth talking about.

I want to make it very clear, this question is not worth talking about now; it is premature. It will be worth talking about in the decade from 1970 to 1980.

This is a basic relation that we see between the water reactors and the fast reactors

WENSCH: Discussion on this topic.

YEVICK: Yes, I should like to make a remark on this. I think that this business of faith stems back to when the Seawolf S2G sodium reactor was dismantled

Now, I shall take an example of a 150-Mw pressurized-water reactor and a 150-Mw plutonium-fueled sodium-cooled fast reactor and compare them in weight and cost. For example, data given in the Westinghouse Report WCAP-1196, in Report SL-1674, and in Report APDA-136 showed that the thermal powers were 555 Mw for the pressurized-water reactor and 434 for the plutonium-fueled reactor. The gross electric power was 147 Mw for the pressurized-water reactor and 150 Mw for the plutonium-fueled reactor. Thus they are about the same in this respect. The net efficiency was 24.3 for the pressurized-water reactor and 32.8 for the plutonium-fueled reactor.

In reference to the reactor vessel as an example of size and weight, the diameter is 11.8 ft for the pressurized-water reactor and 8 ft for the plutonium-fueled reactor. They have the same height, around 31 feet. The wall thicknesses are 7 in. for the pressurized-water reactor and 0.5 in. for the sodium-cooled reactor. The cost for the installed reactor is \$2,950,000 for the pressurized-water reactor and \$930,000 for the thin-walled plutonium-fueled reactor. The number of safety and control rods are 36 for the pressurized-water reactor and 10 for the plutonium-fuel reactors, with the cost for the rods and drives being nearly \$3 million and \$930,000 for the pressurized-water reactor and \$2 million and \$415,000 for the plutonium-fueled reactor, respectively.

As we progress further in our evaluation of these reactors, including the intermediate systems through the heat exchangers, we begin to see advantages even with the intermediate system that the sodium-cooled reactors are at the present time forced to have. The flow in pounds per hour is 36.6 times 10^6 for the pressurized-water reactor and 16 times 10^6 for the primary system of the sodium-cooled reactor. Adding the secondary flow brings the total to 32 times 10^6 for a sodium-cooled reactor. The pressurized water system does not have an intermediate system.

In relation to the pumps, the situation exists in which an up-to-date arrangement has not been developed, but I think eventually such an arrangement will be. I think somebody is going to try to use pumps with mechanical seals. We were one of the first, along with ANL, to lead in this new low-pressure reactor, and we find that there is no problem to use the type of pumps that we use, the centrifugal or mechanical pump with a free surface. I see that AI has found out that this is the kind of pump to use as well. It is very inexpensive, but we can even bring these costs down. We can get a 28,000-gal/min pump for about \$370,000 without the motor. That value is a quotation.

When these factors are considered, it can be seen that a sodium system can be utilized effectively if it is hermetically sealed—and the Seawolf reactor was hermetically sealed. It was in operation for two years—and this system is sealed and then no maintenance is required. There are no service systems. The oxygen is cold-trapped once, and it stays cold-trapped. Sodium is noncorrosive once the oxygen content is removed.

I just happened to pick the pressurized-water reactor. I think I can take a boiling-water system and make about the same comparisons.

A once-through steam generator has lent itself very well to the liquid-metal type of system. It is true that the thermal reactor such as AI is developing is also sodium-cooled, and it has its advantages. However, I am not going to discuss the thermal part. I basically think that sodium-cooled reactors have not only tremendous possibilities, but have shown them. Somewhere along the line some faith was lost because of that one particular problem with the Seawolf.

WENSCH: Thank you. Further discussion?

SIEGEL: I should like to add my comments. Mr. Yevick went far into detail and presented many of the merits and advantages of the sodium-cooled system. I am only going to comment on the costs for a current-status 300-Mw sodium-cooled thermal reactor. We made a fixed price proposal in the last six months of \$217/kw. These costs compare very favorably with the costs that Karl mentioned for the water reactors.

COHEN: I do not want to quarrel with my friends here who say that sodium-cooled reactors should be cheaper; and, as I recall, I said that eventually they will be. I think the fact of the matter is that they are not now.

WENSCH: I am glad that this question of cost came up because the AEC had two studies made on the costs of various systems, one being the sodium-graphite reactor. I just happen to have some figures here. These figures are for a 300-Mw(e) sodium-graphite reactor in whose program Mr. Yevick participated, as I recall, as a consultant to the AEC. The fuel was uranium-10% molybdenum for which the burnup was 8000 Mwd/ton; the central temperature was 1050°F; and the capital cost came out to be \$300/kw. The cost per kilowatt-hour was 6 mills for capital, 4.4 mills for fuel cycle, and 1 mill for operating insurance. If these are added together, a cost of 11.4 mills is the result.

We heard yesterday from Mr. Dickinson that something like 6 or 7 mills is involved; and earlier today I think the same figure was mentioned.

What has happened since the AEC made these studies, what has been the breakthrough that these costs are much lower than the AEC costs?

SIEGEL: You have mentioned the costs for a 300-Mw current-status plant. I just recounted to you the fact that we made a pretty detailed engineering design study a year ago; and, on the basis of that study, we prepared a commercial proposal that involved a cost of around \$225/kw. Thus there is an immediate difference of 1.5 mills/kw-hr in the capital-cost.

About some of the other more technical characteristics of the plant that were assumed in the earlier study to which Glen refers, the uranium-10% molybdenum fuel then was based on a burnup of 8000 Mwd/ton. The current-status plant that we described had a substantially longer burnup based on the additional information that has been developing through the work, in particular, by APDA. Thus for a maximum temperature of 1050°F it is possible to use a burnup figure of 18,000 Mwd/ton. This results in a substantial reduction in fuel-cycle costs. The fuel-cycle costs, instead of the 4.4 mills that Glen quoted, are about 3 mills. These are the two major differences. There is a difference in thermal efficiency over the figures that were used in the data that were quoted. The thermal efficiency of the current-status plant was 36.8 per cent, which is higher than we used earlier. The current-status plant that we described in a commercial proposal last summer had a total power cost of about 8.7 mills/kw-hr rather than the 11.4 that was quoted in Glen's question. In summary, this cost represents a reduction in capital costs that is due to better fuel performance and a slightly higher temperature (1000°F instead of 945°F) of the sodium so that there is better thermal efficiency. The reductions in capital costs themselves are based on experience—experience with APDA in the use of the cheaper boiler, experience at Hallam in the use of a bellows in the system, and experience in the operation of SRE itself.

Therefore the 1961 current-status plant has a technical basis for improved performance and lower power costs and has engineering and design and construction experience which leads to a lower capital cost.

I think I have already commented on the advances that are expected to be made to arrive at the 6.5 mills/kw-hr that both Robert Dickinson and I mentioned for an advanced plant that could be constructed in the later 1960's, but these include significant advances beyond the current status in the way of the new fuels, higher sodium temperatures, and, in particular, the associated thermal efficiencies.

WENSCH: Thank you.

SPINRAD: I suppose this audience is sophisticated enough for me to say what I really think, which is that there is no reason why the water-cooled reactors ought to be cheaper than the sodium-cooled reactors, except that there is a popular mood that they are; and, as soon as there is such a mood, they become so. The fact of the matter is that I do not believe that one can defend current costs. By the time he is through a defense of a sodium plant on the basis that it will be a cheaper power producer than a water reactor, it cannot be bought.

I think that for every quotation that Sid makes, there is to be found a quotation from some competitor on the water plant which will be a little lower. Indeed AI has an organic system right now on which they are trying to quote lower prices.

The real problem is to overcome the associated technological problems. Also a problem exists in that very few people are interested in sodium plants. Each job is a very special job, but no one is going to do anything about it for 10 years. I think that we ought to be building about as many sodium-cooled reactors as water-cooled ones. We are not. Until we are doing this, they will be more expensive. I do not know how to get around the circle except to keep quiet about our reservations that they would be more costly and just go out with a missionary zeal and say that they should not be more costly.

WENSCH: Any comment?

SIEGEL: Well, I shall only comment on the so-called speculation on the capital costs. There really need not be any speculation about that. Various organizations in the field are now making fixed price proposals.

The utilities know what their own additional costs are going to be; and, of all the aspects of the uncertainties here and there in the nuclear-power field, I think the capital costs today has less than all the rest of them because these can be determined from proposals made by commercial organizations to utility organizations.

WENSCH: Thank you. I should like to continue with some questions we have received from the audience at this time. If you have additional questions will you please send them forward. The next question is from Mr. Rhode of Niagara Mohawk Power Corp.

QUESTION: It appears that sodium-cooled fast reactors in this country have been constructed with pressure type containment, whereas sodium-cooled thermal reactors have not incorporated this feature. How may this be explained, and are these trends expected to continue in the future?

WENSCH: I think I shall ask Mr. Yevick to answer this question.

YEVICK: I do not know exactly how to answer the question. As far as we are concerned, we are proceeding on the basis of pressure containment for a fast breeder sodium-cooled reactor. We do not intend anywhere in the near future to void that principle.

Mr. Rhode is talking about pressure containment for the building. I think that Mr. Siegel probably could answer that better.

WENSCH: Yes. Sid, would you pick up the second part?

SIEGEL: Well, I would want to restrict my comments to the requirements of the thermal reactor. As in any of the reactor systems for which we carry out a design, we know what the requirements of the site are and that requirements to satisfy the general public safety have to be met. We then analyze the design, seek out the characteristics of the plant under normal and abnormal conditions, and make an engineering judgment on what is the necessary means that have to be provided for taking care of potential accidents.

In the sodium-graphite reactor system we have demonstrated in two cases, both in the SRE and Hallam, that with the sequence of confinement that is provided for the low-pressure

radioactive system is, first, the fuel, second, the primary system, and beyond that vaults which contain the primary system and which can be kept closed. Beyond this containment sequence, a relatively conventional industrial building with some precautions on ventilation and enclosures is all that is necessary to satisfy ourselves and the review authorities.

WENSCH: Is there really a difference here, between the way these two different reactor types are contained? I know they look different, having the steel reactor container for the fast reactors as compared to the containment provided by concrete below grade in the sodium reactors. I really wonder if the purpose for the design is the same in both cases. One designer wants to make steel and the other one concrete.

I should like to ask this question of one of the senior designers in the audience. I am going to ask Mr. Amorosi of APDA.

AMOROSI: The same types of studies that Dr. Siegel has made at AI we have made at APDA-PRDC for the Fermi plant.

There is no accident called a maximum accident or even a hypothetical accident that would release sodium into the building in our case; and, as a consequence, we do not need the steel containment on the basic design principle applied by AI. I think that based upon this practice there should be no difference in containment requirements for a sodium-cooled thermal reactor and a sodium-cooled fast reactor.

I shall hasten to point out that we use as a hypothetical accident one that is far beyond any accident that can be called credible—a release of energy. In the thermal case no release of energy is assumed. The total release of energy is so small that we do contain it without resort to a steel containment building. I want to also point out that this energy is quite a bit smaller than the energy in a pressurized-water or boiling-water reactor.

I have refrained from giving a number. I think everyone recognizes that for Fermi we designed for a hypothetical release of energy equivalent to 1000 pounds of TNT; but, assuming somehow that the complete core started to run together at the speed of gravity, a condition that we cannot even readily visualize, the numbers come out on the order of 300-lb TNT, not 1000. I am only mentioning this to show that we have compounded a great many things in arriving at a hypothetical design condition.

There really is no great difference in requirements. There are differences in philosophy of safety of design that exists between various design agencies. It is hard for me to say whether or not our more conservative approach is due to the fact we have been living with the courts and hearings so long that we have grown more conservative. I do not like to think that, but it may be that this is the major reason.

WENSCH: Thank you. Any additional comments?

SIEGEL: I do not think there is any difference in the conservative attitude between us although there are some psychological differences in the two circumstances which really have no sound technical basis at all, just the sort that AI has been implying. Beyond those psychological differences, I suspect that there is no real technical difference and that each of us has been adequately conservative in the entire context of the background of the situation, both technical and otherwise.

WENSCH: Thank you. One question that frequently comes up is why does EBR-2 have a huge containment vessel. I should like to say a few words about that. Data on the reactor as originally designed are given in an Argonne report, Report LJK-10, going back to maybe 1949 or 1950 when construction was approved by AEC. The location for the reactor was then uncertain. Many of us thought it could be built at Lemont, Ill.; and we were prepared to have it built at Lemont; and in order to circumvent any hold-up with the state and federal authorities, we said that we would have containment. At the last minute management said no to the Lemont choice of location and that we had better place it out at National Reactor Test Site (NRTS) with the Idaho Division because many hundreds of kilograms of plutonium would be placed in the reactor. Since the drawings were far along and the costs were not drastically different between the containment and the conventional building, it was placed as designed at NRTS, where it is today.

Dr. Spinrad, do you have anything along those lines?

SPINRAD: I think that you have given the history. Certainly at ANL we are perhaps more sensitive on containment than at almost any other place in the world, but for a reason that is not obvious to everybody. Argonne is a general-purpose laboratory that includes quite a bit of extremely low-level counting equipment. One of our most famous experimental setups is a room where people can come to be counted for their body burden of radiation. Amounts of radiation that are completely innocuous to human populations foul up our laboratories to an incredible degree. Therefore we just bottle up everything we can and that is the reason why fears of such occurrences appear to be over the ANL site.

COHEN: I have a comment on this question. I believe that what has happened in the case of the fast reactor, of course everybody knows about it, is that there was an incident that happened in EBR-1. It turned out that it had the wrong kind of coefficient. Also the first Fermi core load, as it turned out, did not have the necessary thermal rigidity. In short, we are relying for controllability of these reactors on a very small effect, an effect which experience proves is not always correctly evaluated in the design basis. It seems to me only prudent that we recognize this.

The fact is that this is one of the things about fast reactors that in the design we should like to eliminate. We should like to have a more stable physics control mechanism rather than depend upon small mechanical motions. The larger the core is, the more degree of freedom of movement it has, and the more modes and harmonics it has, the more we worry.

Another thing that I think is extremely difficult to say is how are the fuel pins going to expand when the long life that everybody is saying is necessary with metal fuels in order to have good economics is reached.

JONES: I think that maybe from the operating standpoint it makes some difference because, as a general rule, the construction within the containment sphere leads to operations similar to building a boat in a bottle. Although with the containment as employed at Hallam, one can go to the ranch-style construction, which offers some operational maintenance advantages, it also introduces some problems. However, I think the over-all net advantage to the operator is good.

WENSCH: Thank you. Well, it is not true that all the fast reactors do have this steel containment vessel. As I recall, the LAMPRE-1 does not; and its containment is no different from any other reactor. It is in a laboratory building, and it is located adjacent to Los Alamos and its community. Neither does the EBR-1 have the steel containment.

Mr. Rhode, have we answered your question satisfactorily?

RHODE: Yes.

WENSCH: The next question is from Mr. Amorosi.

QUESTION: It is clear from the remarks of the speakers that economic power can be achieved but that it will take considerable research and development effort to bring this about. Can you indicate the AEC plans in sponsoring research and development?

WENSCH: Present plans are to support the sodium-cooled-reactor program in the foreseeable future, and we have long range plans through 1966. As you know, in the government method of budgeting ahead, although we may have plans, we still operate on a fiscal-year basis one or two years ahead of time. We are now working on our fiscal year 1962 budget, and the fiscal year 1963 plans are being firmed up. I do not know what more I can say than that.

Maybe we should ask if the levels of research and development are satisfactory. The sodium-reactor research and development program is not nearly as large as the research and development for the water reactors. As I recall, the Shippingport plant alone gets about \$20 million a year for research and development, whereas our total program, including the fast and the thermal reactors, is \$21 million. Our program has gone slowly but steadily. We have had good support from the AEC, and I expect that we will receive continued support.

Does this satisfy you, Mr. Amorosi?

AMOROSI: Thank you.

WENSCH: Here is another question from Mr. Rhode.

QUESTION: May one infer from the direction of current AEC and research and development efforts that AEC is more interested in sodium fast reactors than sodium thermal reactors? How does the AEC evaluation of potentials for the two concepts compare?

WENSCH: I shall answer the first part. The second part I am really not qualified to answer because this concerns another part of the AEC.

As far as the AEC is concerned, we do not make a distinction between the fast reactor part and the thermal reactor part of the program. One and one-half years ago management integrated both programs so that they would complement one another, particularly in the area of sodium technology. In that group we formed a sodium-component development program for all sodium-cooled reactors for civilian power. Therefore I can say that at this time there is really no difference, no bias, and no prejudice. Nothing except to go ahead; and that is to support this program and really get it moving.

I think it is only fair to say that there could be a change in support depending upon the performance of the reactors that we shall eventually be operating (EBR-1, EBR-2, Fermi, and Hallam). The successful operation and evaluation of these plants could well change the direction of AEC support.

The second question is how does AEC evaluate potentials for the two concepts? Well, the AEC does have an evaluation and planning group who are, I would say, almost continually evaluating different reactors. This group itself has come in from various parts of the United States, Mr. Yevick has been in one. I think Mr. Siegel has served in the group, and Mr. Amorosi has helped. Now, these consultants do not tell the AEC what they should do. What they do is tell the AEC the status of the technology—these are the facts; this is the way we think these reactors can be built; how they should perform; etc. Then the AEC does the evaluation.

From the recent studies I do not believe that there has been a big difference between sodium thermal reactors and fast reactors. They compare about the same. I believe there was one study which showed that the fast reactor was better under the potential of the very small system than when it became a plutonium burner. This, of course, surprised everyone. How could one build it for a real small power system? I shall not belabor this point because I might say things that are really not facts.

The next question is from Francis Green of the National Carbon Company.

QUESTION: Is there an economic incentive based on the level of neutron-economy improvement for developing an uncanned graphite moderator, a graphite-clad uranium carbide fuel element compatible with sodium?

WENSCH: I think this is a very good question because one of the characteristics of sodium-graphite reactors is in its very low conversion ratio.

Dr. Siegel, would you like to answer this question, please?

SIEGEL: There are undoubtedly neutron economies that could be profitable if the technology could be developed in working with graphite and, perhaps, uranium carbide in either direct contact with sodium or essentially separated from sodium by some low-neutron cross-sectional material. What is suggested in this question, I gather, is coated graphite. At the present time we cannot use graphite in direct contact with sodium because the porous nature of conventional reactor graphite absorbs so much sodium that the neutron economy would be ruined. Even if the graphite is coated with a carbonaceous material so that it becomes impermeable, there are still material-compatibility problems. At the temperatures that we are becoming interested in, in the neighborhood of 1200°F, there are carbon-transport problems that would substantially affect the characteristics of stainless steel. This is one of the possible drawbacks to using graphite coated with carbon. There are other ways of coating graphite, which we are exploring. For example, we are looking at coating graphite with thin sheets of zirconium, avoiding some of the mechanical problems of the approach that has been taken at Hallam. The sheet has to be a mechanical structure with its own integrity; but, if we use instead what amounts to thin adhering coats of zirconium on graphite, I think there are some substantial neutron economies that can be achieved that way in a practical sense.

There is also some benefit in using lower cross-section cladding on the fuel. However, it is not clear to me that uranium carbide lends itself very well to this approach.

WENSCH: Do your plans include research and development to improve the conversion ratio?

SIEGEL: Yes, we are pursuing alternate core designs with not so much emphasis on the conversion ratio as on fuel-cycle costs. I think Karl emphasized a little while ago that this is the matter that requires attention on optimizing fuel-cycle costs, in fact, optimizing energy costs in the plant as a whole and fuel-cycle costs in part, rather than merely paying attention solely to conversion as such. It can easily turn out that fuel-cycle costs are optimized at not necessarily the highest conversion ratio.

WENSCH: Does this satisfy you, Mr. Green?

GREEN: Yes, thank you.

WENSCH: The next questions are from Mr. Foster of Atomics International.

QUESTION: To what extent can quality control be relaxed for sodium components such as steam generators? What risks are involved? What economics can be gained?

WENSCH: I should like to add a question for him. Why is there a difference between the steam-generator designs for the reactors now being constructed? We have the single-wall tube for Fermi and double-wall tubes for EBR-2 and for Hallam. As you heard yesterday there seems to be a remarkable difference in the design philosophy.

I should like to direct these questions to a volunteer.

SPINRAD: I think I can start off simply by saying why EBR-2 has the double-wall tube. These projects that are being completed now were originally specified and set quite a while back, four or five years ago. At that time it was just a matter of engineering judgment, whether one could afford a single-wall tube or a double-wall tube. Thus we quite arbitrarily, I should say, picked the double-wall tube. It seemed to be a good idea; and, now that we have gone this far, we are finishing it up that way because, having done this much work, it is worth a competitive evaluation in service. I do not think there are any claims being made that this method of construction is the right way. This is one example of trying various approaches and putting them into a service test and seeing what the results are.

WENSCH: Mr. Yevick?

YEVICK: I shall talk about the quality-control question. We have had considerable experience in the Fermi program. Fermi is the result of a very complex plan, and without quality control we would probably be having considerable difficulties in alignment problems, in the fitting. As it is we have run into many problems. One of them is replacement of the graphite. It is not a too difficult task to replace it, but I think that if there had been some quality control on inspection there would not have been some of the difficulty experienced.

I think quality control pays for all the time spent on it. At Combustion Engineering, Inc., we have had resident inspectors to inspect the reactor vessel. We kept them there 16 hr along with the inspectors that Combustion Engineering, Inc., had. There were still a few items that were missed.

I think it is very important in this business that for the next few years, until we get these things on a production line basis, that quality control be kept high.

WENSCH: Dr. Siegel?

SIEGEL: Well, the question implies hope for a quantitative answer, which obviously I cannot give. To what extent can we relax quality control? I think that we are in an industry that has a very advanced technical nature; and in this industry, as in others of greater familiarity but which still are of a very advanced technical nature, quality control is going to continue to be a highly essential aspect of the manufacturing process. I am sure that when a 3500-psia conventional boiler is built for use to produce 1050°F steam that it will not be built with less inattention to quality control than is used in the nuclear plants. I believe, however, that as we gain experience in the design and manufacture of plants, we shall be able to have simpler designs, which will have perhaps less intricately quality-control problems associated with them. Also there will be more manufacturing experience so that quality control can be accomplished in perhaps a more routine and less expensive and laborious fashion. However I believe that we shall still want to build high-quality equipment.

WENSCH: Thank you. I think that, in addition to quality control, one must consider house-keeping, particularly housekeeping during the construction stage because it is rather stupid to spend premium costs to fabricate components and to keep them in a sterile environment and then to deliver them to a site at which all sorts of debris gets inside the sodium components.

I would suggest to anyone really interested in the subject of housekeeping and quality control that they see the movie called "Integrity Plus." The Navy has had a great deal of experience on this subject. Nevertheless, the Navy has found many strange things inside their reactors. I cannot recommend this film too highly; I can only say that it should be a must for anyone in the nuclear industry.

Does that satisfy you, Mr. Foster?

FOSTER: Well, I was referring to the cost of fabricating pieces of a component such as steam generators and pumps. If one buys pumps for oil refineries of a similar capacity to what we might need in a sodium reactor, these pumps would cost, including motors, somewhere in the neighborhood of \$50,000; but if one buys a sodium pump it will cost \$200,000 to \$300,000. I think the same parallel can be drawn with steam generators. I have been told by the manufacturers that one of the major reasons for this high cost is the specifications that are drawn up concerning quality control and welding. Now, it seems to me that there must be some parallel between the specifications that are drawn up and the risks one is willing to run in terms of a leak. I am not quite sure that one cannot assume a greater risk and thereby cut the cost.

WENSCH: May we have further discussion from the members?

COHEN: I should like to say that I think that rigid quality control in the nuclear industry is a way of life, and I think we are going to have it for a long time.

WENSCH: I should like to ask someone in the audience in reply to his question.

Mr. Koch, you have had a lot of experience in quality control. What are your judgments concerning this matter? Would you come forward, please?

SIEGEL: Well, while Mr. Koch is coming forward, I should like to comment that what has been emphasized is that perhaps the requirements are redundant, excessive, and that this is a matter really of engineering judgment. As we learn we shall reduce many of these requirements, as we learn that they are not necessary. Basically the equipment will have to have the quality necessary for its purpose, but it need not have excessive quality.

L. J. KOCH (Argonne National Laboratory): I think I should like to amplify Karl's remark on quality control being a way of life and, in particular, to be specific with respect to steam generators, which is one of the components mentioned. In the case of EBR-2, we were faced with a decision concerning the general design characteristics and improving, of course, quality and reliability that we had hoped to achieve. We also had to make this decision within a psychological framework, or environment, at a time when the Seawolf was having some fairly sad experiences with the specific pieces of equipment that were mentioned. The general attitude, I think, in the industry was that it was virtually impossible to construct a leak-tight unit, but that it was mandatory to achieve a leak-tight unit. Within that framework we carried quality control and design reliability to a point to which perhaps we would not in a different psychological environment. However, we attempted to use more than the simple matter of reliability alone. We also concentrated on design features, and we felt their applicability to higher performance. For example, we put in much concentration on the problem of stress corrosion, which seemed to be the major difficulty at that particular time on the water side. We used a material that was less sensitive to stress corrosion, namely, the 2 1/4% chromium-molybdenum alloy. We also designed the equipment on the basis of eliminating crevices, hopefully, with the potential of the same basic design features, once demonstrated, being applicable to higher strength materials for higher performance. By obviating the stress-corrosion problem along two routes, namely, the material and the lack of crevices, we hope to eliminate crevices in higher temperature units in the future.

However, I think if you just want a simple answer, Karl's answer is the only real one that can be given right now, and that is that we are in an industry which is new. Every failure of any kind, whether or not it is minor or major, has the effect of causing difficulty to the industry. One can look upon them as setbacks, minor or major setbacks.

We are striving essentially for perfection, and I think this is costing us something. Part of this attempt to achieve perfection is increasing the costs; it is also increasing the problems all along the line. I think that we shall come closer to perfection later with experience without necessarily assuming the same degree of costs, but this will come only with experience. I think

experience will help us in two ways: (1) operational experience will give us the confidence that we need because the situation will no longer be so new and novel and (2) fabrication experience will help us achieve a higher degree of reliability at a lower cost.

JOHN LeVAN (Aero Research Instrument Co., Inc.): As a manufacturer of nuclear components, I should like to amplify some of the cost considerations in these specifications. It appears that the specifications go one step beyond quality insurance and attempt to determine manufacturing processes and the way of doing things. This we know increases costs quite considerably. As a case in point, in one contract that we were just awarded, we went in at a quotation of \$100,000. The next quotation in line for the type of manufacture suggested in the specification was \$300,000. Now, we had to take an exception to the manufacturing process, but no exception to the intent or the quality insurance in the specification; and we were, in effect, awarded the contract. This is just one small indication where the design engineer was attempting to get such a high safety factor, plus imposing his own state of the art into the specification, which, in fact, added a lot of costs to the product. The manufacturer is not allowed to use his own research and development and progressive engineering to determine what will pass the quality insurance as set up in the specification.

WENSCH: Thank you, Mr. LeVan. I think we could probably belabor this subject on quality control because it is intangible and quite elusive. We all know that in this business at this time it should be good.

Continuing, we have a question from Mr. Augustine of the AEC, addressed to Dr. Jones.

QUESTION: What are the costs of generating and transmission and distribution in the CPPD system, and what is expected of the Hallam reactor in reference to the previously mentioned costs?

JONES: It would be better to let Bill answer this one. Actually, of course, in any utility system one ends up with a variety of power sources, each with their own power-production costs. The grid system that covers the eastern two-thirds of Nebraska is actually fed by steam plants; it is fed by fuel-reclamation power of the Missouri River basin and has interconnections into Kansas, Iowa, etc.

I think probably the best thing to do would be to talk about the power plant in one location, namely at Hallam, where it takes something in the neighborhood of 7 mills/kw-hr to produce the 1800-psi reheat conventional plant, and then compare the cost with nuclear-power costs. Depending on which system of books one elects to use, he probably would come up with something in the neighborhood of 12.5 mills/kw-hr for production costs; but this depends on how the books are kept and what methods of conversion are used.

When one considers transmission costs, he is considering an integrated operation. I think the cost is something in the neighborhood of 1 mill/kw-hr for transmission. Again this takes into consideration how far one goes in considering transmission costs.

SIEGEL: The question is a very generalized one, and I do not think we have any real firm answers to it. The only other way we are going to be able to compare the costs will be at one place with one specific kind of plant as compared with the best conventional plant that would be built in that same location with the fuels that are available in that one location. One cannot say that nuclear power is competitive with conventional power as a general rule because of the kind of plant that is built in one location with the fossil fuel. Despite the comments made earlier, we do have much variety in fuel in the conventional power field: coal, gas, and oil. These all go into the selection of the plant, the design, the efficiency of the plant, etc.

JONES: I think we are talking about a 50 to 60 per cent penalty in power-production costs—Hallam nuclear power as compared to the conventional power plant.

WENSCH: Thank you.

We have a question here from Mr. Stern of Nuclear Development Corp. of America.

QUESTION: Dr. Cohen spoke of a burnup of 100,000 Mwd/ton for a mixed oxide fuel. How can this be reconciled with the generally accepted, experimentally verified practical limit of about 50,000 Mwd/ton with straight UO₂ fuel?

COHEN: I do not know that there is any limit. It is placed in the reactor and then one lets it go. I suppose what Mr. Stern means is that usually one is relying on the oxide structure to hold the fission-product gases and that there is a limit to the burnup that will do this, which is

perhaps 30,000 Mwd/ton or something near that. The answer is that the fission products were not held in the lattice. Thirty or forty per cent were released, and we rely on a fuel-element design that just accommodates the release.

STERN: Thank you.

WENSCH: The next question is from Mr. Schlottenmier of Minneapolis-Honeywell Regulator Co.

QUESTION: I should think that one, perhaps, small problem in operating the high-temperature systems we have been discussing is adequate measurement of the temperatures, pressures, flows, and levels which allow proper control in our analysis of system behavior. Could we have some comments on this? Do measurements of this kind present a severe problem? Is present equipment adequate? Are there any new developments and new problems?

WENSCH: I should like to ask these questions to Mr. Yevick.

YEVICK: In reference to the sodium-cooled reactors, there are several instrumentation developmental problems for which we certainly would like to have the answers. One problem concerns a method whereby we can reach down into the sodium and locate things. Another one concerns a means of a fission-product detector or a means of locating leaky elements if we have a system in which we do not want to have any leaky elements. The existing types of instruments such as temperature measurements seem to be pretty adequate insofar as the detector is concerned in the receivers, but we do not have adequate means of getting into the reactor itself in most cases. In one case for the temperature measurements on the fuel elements, the thermocouples were disengaged for a very short period of time. Whether this was due to mechanical difficulty or radiation, I do not know; but they did lose some thermocouples and the fuel rapidly.

Very few reactors are adequately equipped in this respect to measure things such as flows, for example. It would be very desirable to be able to measure the flow on every single subassembly. In this way, then, one could program a flow through a core if it were needed. I think there is a lot of work still to be done on liquid-level detection.

We heard a report yesterday on oxygen analysis. It is in a stage at the present time where it is inadequate. In these low levels of detection of oxygen, we would certainly like to be able to develop some piece of apparatus, not such as was shown in a slide that was presented—that was a laboratory model—but something you can attach to the system because one certainly is not going to take the sodium out of the system if he can help it. That is why the plugging indicator has turned out to be such a valuable and useful instrument. However, it really does not indicate what the oxygen is; it is just a correlation between the plugging temperature and the oxygen content. There are things such as hydrogen detection in case one has a leak from the steam generator; and I do not know of a steam generator that has been built in which there is not going to be a leak from the water system into the sodium. We would like to be able to detect on the secondary side that we do have a leak. This is quite a problem. How does one inspect a steam generator after he has it in service? This is a very ticklish problem, and we do not have the answer to it yet. Thus I think there is a lot of work to be done.

WENSCH: Thank you.

We have two questions from Mr. Potter of the Rural Electrification Administration. I shall read the first one.

QUESTION: What is the full-load auxiliary power for Hallam on fossil fuel and on nuclear fuel?

JONES: On the conventional plant end, as I recall, it is about 5 per cent. I think that in the reactor end the last estimates that I saw were a little over 6 per cent for station use. Does this adequately answer the question?

I think, of course, these things enter in when one starts talking about production costs. Fundamentally why we are building the plant is to determine what correction costs are and what the various technical problems are going to be in our own organization. Of course, with the joint participation of the AEC, we are going to end up with production costs that would have been equal to what we would have with conventional power costs. I think you have to appreciate that in all these cost predictions the real reason we are building is to determine these factors, and every estimate of other factors is going to be a crude one.

WENSCH: We shall now discuss Mr. Potter's other question. This will be the last question to be discussed.

QUESTION: Does sodium offer any attractive possibilities for nuclear-fueled gas turbines, or is gas alone a better medium for both coolant and a working fluid?

SPINRAD: I think the answer to the first part of the question is probably yes. The point of using sodium in the heat-exchange medium to heat up the gas for a gas turbine has some value in minimizing the amount of heat-transfer service that is required in the reactor, in the sodium-to-gas heat exchanger. If it is decided for one reason or another that one must use an indirect cycle then a very good high-temperature material is needed. A good high-temperature heat-transfer fluid such as sodium would show up very well.

The real question is whether or not one is willing to run the gas turbines with the possibility of contamination. If he is willing to do so, then I think he would choose the straight-core gas system. If he is not willing to do so, I think the sodium system shows a lot of value.

WENSCH: We must now conclude this discussion. Following this panel session we are going to have a movie that has been kindly provided for us by ANL. This movie shows the construction of the EBR-2.

I should like to express my appreciation to the panel. I believe it would be in order to really express our appreciation to the Consumers Public Power District of Nebraska for organizing and running this very good conference. Let us do it by a rising vote of thanks.

Appendix

Postmeeting Questions and Answers*

QUESTION [G. W. Wensch (U. S. Atomic Energy Commission) to J. G. Yevick (Atomic Power Development Associates, Inc.)]: Let us assume that fissionable fuel is a commodity and that U^{233} , U^{235} , and Pu^{239} are each worth \$15,000/kg. If a fast breeder reactor requires 1000 kg of fissionable fuel as its inventory, this would represent \$15,000,000. Suppose that the reactor system generates a net 300 Mw(e) with an 0.80 load factor and the interest on the fuel commodity is 12.5 per cent. This represents approximately 1 mill/kw-hr, and we have not considered the fabrication and reprocessing portions of the fuel cycle cost. Do you think that fast breeder reactors would be economically feasible?

ANSWER: The AEC reference data for status and potential reactors uses a value of \$12,000/kg of plutonium. For purposes of this discussion, I shall use this figure. For a 300-Mw electric plant, the critical mass of Pu^{239} is 674 kg with a total inventory of 1000 kg in the total fuel cycle. Assuming an oxide core with a 1.4 breeding ratio, an average burnup of 100,000 Mwd/metric ton of metal (plutonium plus uranium), and a plant factor of 0.8, 180 kg of plutonium per year will be consumed after including a bonus of 60 kg from U^{238} , Pu^{240} , and blanket burnup. The plutonium created is equal to $180 \times 1.4 = 252$ kg. The value of this plutonium is $\$3 \times 10^6$. The inventory is $10^6 \times \$12 = \$12 \times 10^6$. The inventory annual charge at 10.33 per cent amounts to $\$1.24 \times 10^6$ per year. An inventory charge of 10.33 per cent rather than 12.5 per cent is used as determined by studies at APDA and other power groups. Compare this with a thermal boiling-water reactor utilizing UO_2 . Burnup is 240 kg since there is no bonus from U^{238} or blanket burnup. The critical mass (see Reports SL-1674 and GNEC-100) is 1000 kg of U^{235} , and total inventory is 1600 kg of U^{235} at 1.7 per cent enrichment. Inventory is $1600 \times 10^3 \times \$7.00 = \$11,200,000$. The inventory charge at 10.33 per cent is $\$1.16 \times 10^6$. The plutonium credit is 4 g/kg of uranium thruput or 143 kg valued at $\$1.7 \times 10^6$. For the fabrication costs assume \$110/kg total uranium for the thermal reactor, \$370/kg of metal (uranium plus plutonium) in the fast oxide core, and \$50/kg of uranium metal in the fast breeder blanket. Charges include conversion, shipping, and insurance. Assume thruput for the fast reactor at 1750 kg/year of core based on a burnup of 100,000 Mwd/metric ton limited only by fuel damage and 4000 kg/year of one fast breeder blanket. Assume a thruput of 8000 kg/year for the thermal reactor based on a burnup of 30,000 Mwd/metric ton of uranium, limited by reactivity considerations. The 30,000 Mwd/metric ton of uranium is based on solving the excess reactivity problem. The fabrication cost is \$850,000 for the fast reactor and \$880,000 for the thermal reactor. The processing costs are \$160/kg for the fast reactor core (uranium plus plutonium), \$90/kg for the fast breeder blanket including the conversion of plutonium nitrate to metal, and \$65/kg of uranium for a thermal reactor including the conversion of plutonium nitrate to metal. Costs include shipping and losses. The processing costs are \$640,000 for the fast reactor and \$520,000 for the thermal reactor. The following table is a comparison of the costs per year:

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*The limitations of time for publication did not allow for a thorough exchange of questions and answers with the speakers, therefore some questions posed are unanswered.

	Fast breeder reactor		Thermal	
		Per net kw-hr	BWR	Per net kw-hr
Fabrication	\$ 850,000	0.41	\$ 880,000	0.42
Processing	640,000	0.30	520,000	0.25
Inventory charge	1,240,000	0.59	1,116,000	0.53
Depletion charge	*		1,680,000	0.80
	\$2,730,000	1.30	\$4,196,000	2.00
Plutonium credit	-864,000	0.41	-1,700,000	0.81
Net	\$1,866,000	0.89	\$2,496,000	1.19

*Recycle basis

I should like to correct two common errors regarding fast reactors:

1. That inventories are large. A review of Reports TID-8516 and SL-1674 shows initial U^{235} critical masses of 1000 to 2000 kg for 300 Mw(e) plants such as PWR, BWR, OMR, SGR, and GCR types as compared to a critical mass of 500 to 700 kg U^{235} or Pu^{239} for a fast breeder reactor. The total metal loading is even smaller. Fast breeder enrichments run 16 to 25 per cent, whereas thermal reactors run 1 to 3 per cent, giving metal core loadings of 30,000 to 100,000 kg or more for thermal reactors as compared to approximately 5000 kg for the fast breeder core.

2. That processing and fabrication costs are too high. This again is not in accordance with facts. The unit cost per kilogram processed or fabricated is high, but the quantity processed is small as noted by reviewing the example. The total cost per year comes out less than for thermal reactors. The burnup of 100,000 Mwd/metric ton used for the fast reactors is as realistic as the 30,000 Mwd/metric ton for thermal reactors.

There is no question that fast breeder reactors are as economically feasible as other reactors. There are capital charge reasons that have not been discussed which make liquid-metal-cooled reactors, particularly fast reactors, very attractive.

QUESTION [J. T. Jackman (Kaiser Engineers) to J. G. Yevick]: What do you believe is the future potential of the fast reactor as a breeder, that is, optimized for breeding primarily? Are breeding and power generation both achievable on an optimum basis from a single reactor design?

ANSWER: My opinion is based on confidence in the use of fast reactors as breeders and the need for breeding. There is no question in the minds of those who are in the business that fissile fuels will, in a period of 10 to 20 years, displace fossil fuels for large generations of power. There are a few of us who believe it is wasteful to utilize only a portion of the available fissile fuel and that a means must be found to economically utilize all the fuel possible. The approach at APDA is that breeding in conjunction with the production of electric power is economical. In my previous discussion the fabrication of a blanket is shown to cost \$50/kg and the processing of this blanket \$90/kg including conversion of plutonium nitrate to metal, or a total charge per kilogram of \$140. The blanket accounts for approximately 50 per cent of the plutonium that is bred, around 126,000 g or 31 g/kg of uranium. This is a return of \$370/kg of uranium blanket processed, a substantial net gain. Breeding is economical.

QUESTION [G. K. Rhode (Niagara Mohawk Power Corp.) to the panel]: May one infer from the direction of current AEC research and development efforts that AEC is more interested in sodium fast reactors than sodium thermal reactors? How do the AEC evaluated "potentials" for the two concepts compare? It appears that sodium fast reactors in the country have been constructed with pressure-tight containment whereas sodium thermal reactors have not incorporated this feature. How may this be explained and are these trends expected to continue in the future?

ANSWER [J. G. Yevick]: I can only point to the statements made by the AEC to the Joint Committee on Feb. 21, 1961, which had very little to say about either the SGR or the FBR. The

1959 reports indicated that both reactors had a good potential, slightly under 7.5 mills/kw-hr for the studies made in the summer of 1959 based on 300 Mw(e) reactors. Later studies have indicated that this potential can be reduced further.

ANSWER [B. I. Spinrad (Argonne National Laboratory)]: Speaking for a laboratory that is primarily interested in sodium fast reactors I must present this single point of view. We are convinced that sodium fast reactors, by virtue of their breeding potential particularly, have a brighter long range future than do sodium thermal reactors. In developing these systems, which are admittedly of longer range potential, we have further observed the simplicity in design made possible by use of sodium coolant. Therefore we must also applaud the use of sodium in thermal-reactor concepts. As to the evaluated potential, I should like to record my own feeling that mills/kw-hr of systems that we can design now is a poor clue to the ultimate cost potential of fast breeder systems. Without emphasizing breeding too strongly as a "be-all" and "end-all," I must admit that it seems as though it should be in the long run an attractive way of making money. I don't believe that the absence of containment vessels on sodium thermal reactors represents any very significant long term trend. Sodium thermal reactors have known safety features and have so far not been constructed too close to populated centers. Sodium fast reactors do not have safety features that are as well approved and have been sited in more sensitive areas; hence, the containment vessels. We have not yet taken the step at ANL of designing flow orifices adjustable during the reactor operation. The orifices in EBR-2 are fixed at the time the core is loaded. Therefore no safety analysis is being made of adjustable orifices.

QUESTION [F. A. Smith (Atomic Energy Commission) to the panel]: What is the ratio of effort (men $\cong$ \$) appropriate for reactor development (sodium reactors) projects, general research and development, and studies and evaluations?

ANSWER [J. G. Yevick]: The research and development for prototypes is 50 to over 100 per cent of the cost of the plant. For PWR the research and development was around 100 per cent of the plant cost. For Fermi the total plant cost including fuel fabrication is around \$59,000,000 and the research and development to date is approximately \$25,000,000 or approximately 42 per cent of the plant cost. The AEC estimates show that through 1959 research and development for sodium reactors was \$94,000,000, whereas the construction costs were \$41,000,000. With three sodium-cooled reactors nearing completion, this ratio will change abruptly. The amount of monies that has been spent on studies and evaluation has not been large in comparison to the research and development. A rough estimate indicates that 10 per cent of the research and development could readily be spent on studies and evaluations. It is doubtful that this amount is being spent.

QUESTION [F. A. Smith to industrial organizations]: It appears that the sodium reactor is not now economical. Technical problems are unsolved. What can industry contribute to the solution of the problems when there is now no market for the product?

ANSWER [J. G. Yevick]: It can be shown that the sodium reactor thermal or fast is as economical as reactors with other coolants. Industry in the form of combinations of utilities and industrial concerns has organized in the past 10 years to solve the problems and will keep organized until they have reached a solution of these problems.

QUESTION [F. A. Smith to the panel]: What are the technical (specific) problems remaining before sodium reactors are economical? Who is working on the solutions of these problems?

ANSWER [J. G. Yevick]: Groups such as APDA, the AEC, the laboratories, and individual industries are working on the solutions of these problems. The Sodium Components Committee is an example of how industry has joined with the AEC and the laboratories to determine what constitutes the problems and the methods for solution. References to the Civilian Power Reactor Program (Reports TID-8516 and -8517 and Books 1 and 6 of Report TID-8518) present a very good summary of the problems in the sodium-reactor field.

ANSWER [B. I. Spinrad]: For fast reactors the overriding problem concerns the feasibility of useful fuel cycles. Argonne is now working on the feasibility of a fast reactor fuel cycle incorporating metal fuel elements, pyrometallurgical reprocessing, and remote refabrication. I think that in all probability these remote processes are at the heart of the matter of sodium-

reactor feasibility. The laboratories are working on them and will doubtless continue until industry generates enough confidence in this approach. An alternate approach that would be useful, regardless of whether or not the greatest economies in remote processes can be achieved, is an achievement of very high burnup in reactor fuel. This problem for oxide fuels is already one requiring solution for thermal reactors that are well along the road to economic feasibility; thus this aspect is being considered by industry in the AEC fuel-cycle program. Steps for advancing this target through the use of metal fuels or of more advanced developmental fuels are being investigated by national laboratories. A major problem of fast reactors is considered to be their safety. I consider that this problem is mostly emotional in origin and that specific designs of fast reactors can be made extremely safe. The AEC has a program aimed at investigating particular postulated stumbling blocks to safety and this program will help some in the long run. However, I think that it will take the successful operation of many fast reactors to win approval of the safety of this type regardless of technical argument that can be mustered. I am afraid that I sound overly cynical, but such is my impression.

QUESTION [G. W. Wensch to B. I. Spinrad]: We have stated that sodium is the logical choice for a reactor coolant. However, when comparing its heat-transport capability with other proposed or tested coolants, some people are not convinced that sodium is the best coolant for economic nuclear power. What do you think?

ANSWER: Sodium has a number of advantages that have the capability of being capitalized for economic profit. One is that at high temperatures it can be handled in systems of low pressure to raise steam in the most modern cycle conditions. The materials that can contain it under these conditions are conventional. The only problem that sodium is corrosive when it contains large amounts of impurities appears solvable by restricting the impurity level through techniques that are already known. One can also make use of the excellent thermal conductivity of sodium to design compact boilers for less advanced conditions. Here compactness would seem to be the key word for future economic advantage. Since I am neither an economist nor an expert in investment, I cannot make a firm forecast that this is the route for economic nuclear power. However, it would appear to me that any fluid whose use allows simple and elegant technology is a good bet for future economic use. Sodium appears to have these advantages.

QUESTION [Thomas Schlottenmier (Minneapolis-Honeywell Regulator Co.) to the panel]: I would think that one perhaps small problem in operating the high-temperature systems we have been discussing is adequate measurement of the temperature pressure flows, levels, etc., which allow proper control and/or analysis of system behavior. Could we have some comments on this? Are measurements of this kind a severe problem? Is present equipment adequate? Are there any new developments and new problems?

ANSWER [B. I. Spinrad]: The problem referred to in this question is a rough one. A large number of the possible problems with regard to the system as a whole are obviated in the EBR-2 sodium pot type of design. The major process controls are actuated through the secondary system and the nuclear problems are thereby divorced from the process problems. Of course, we would like to have better information on the workings of our reactor core. We particularly like to have detailed flow and temperature information at the output of each of our subassemblies and good neutron flux mapping throughout our core. The basic problems in the fast reactor affecting our ability to measure are as follows:

1. Since the reactor core is so compact, the instrumentation for flow and temperature must also be compact. In addition, it must operate in a hot sodium environment. These are very difficult requirements for engineering equipment even when we know, as with these variables, what physical processes can be used for measurement.

2. Neutron flux monitoring equipment is generally delicate and if use is made of nuclear reaction monitoring one has heat-removal and material-depletion problems. Types of wire activation can be conceived for flux shape monitoring, but the major drawback to this is the purely mechanical one of getting wires into and out of a core reliably without affecting core performance and still allowing capability for servicing. I do not know enough about the instruments themselves to comment on the promise of new types of detectors.

QUESTION [F. A. Smith to Argonne National Laboratory]: How can the national laboratory participate in the development of the fast reactor from the feasibility era (now past) to the economic era (future)?

ANSWER [B. I. Spinrad]: The national laboratories are major national assets, and it would seem most appropriate that their participation in a program be conditioned to national need. These needs are usually for systems of special and unusual performance, and in the long run I would think that this is the type of reactor which the national laboratories will continue to develop. The case of economic nuclear power represents to some degree a departure from this ideal. I suppose that one could say that eventual achievement of nuclear power is a national achievement of some magnitude. Therefore the national laboratories have been utilized in the early stages of central-station power development to bridge the gap between the statement of a promising concept and a demonstration of technical feasibility of the concept. There are many reasons why the future progress toward economy should be made by organizations that stand to profit from such economies. In general terms, the fast reactor for central-station power still has so many technical problems associated with its forecast that it appears appropriate for ANL to continue its development program for at least several more years. I would hope that by the end of this decade fast reactor design would be as commercial as boiling reactor design is now. Of course, it is the primary objective of every researcher, in the national laboratories or elsewhere, to make a real contribution to the advancement of his profession, particularly in areas where the direction of progress is not yet well defined. As a general comment therefore (even though it may not be unusually helpful) I would say that the national laboratories efforts tend toward consideration of the next step beyond achievement of economic feasibility once it appears that this economic feasibility is reasonably justified.

QUESTION [to B. I. Spinrad]: Will you please comment on Dr. Cohen's statement concerning control and safety aspects of large fast reactors fueled with ceramics and metal fuel?

ANSWER: So far the major safety problems which have been uncovered in fast reactor design areas arise from mechanical motions of the core during thermal transients, and the apparent solution is to enhance safety by using the mechanical motion generated to achieve control. Since metal fuels expand on heating they are more conducive to safety. I think that I could invent an accident of powerful proportions for any reactor type including ceramic-fuel types. I also think that for any reactor type an economic and safe design can be worked out. I also think that current design practices for reactors protect quite adequately. Perhaps the best answer is to turn an argument around and thereby illustrate the vagueness of many of these general comments. The metal-fueled fast reactor is the safest of all because it is the only reactor type in which a core meltdown led to absolutely no massive chemical reaction and irradiated no people (EBR-1).

QUESTION [To the panel]: Discuss the advantages of sodium system reactors for reheat cycles with the reheat unit alongside the turbine and hence low-energy potential in event of turbine accident. I suggest that most power companies are now lowering steam temperature to 1000°F, thus reheat becomes more advantageous.

ANSWER [B. I. Spinrad]: Argonne has recently put out a report comparing systems for superheat and reheat application in which this point is discussed. It is entitled, An Evaluation of Reactor Concepts for External Superheat, by D. H. Lennox et al., and will soon be published.

QUESTION [To the panel]: It seems to me that when sodium technology breaks the bounds of 1300°F, stainless steels must necessarily be discarded (if not for corrosion then for strength reasons). On the basis of today's technology, the logical replacements would be refractory metals. Such metals, if they can be put to use at all, can be used in the temperature range 1600 to 2000°F, as well as at 1300 to 1600°F. For this reason the latter temperature range seems to me a black area in the development of sodium power stations. Should not future thinking about more modernistic working fluids and fuels take into account the bypassing of this temperature range? Further, with the NASA money being spent on 1600 to 2000°F liquid-metal systems, is it not reasonable to suppose a "great leap forward" could be in the cards for central stations in the not too distant future?

ANSWER [B. I. Spinrad]: Yes, I am all for it.

QUESTION [John LaVan (Aero Research Instrument Co., Inc.) to the panel]: Are the capital costs of reactors based on present costs for material and components that (a) are in limited production, (b) have an extreme safety factor, (c) are not commercially developed, or (d) are not yet incorporated into our private enterprise system. If so, are we comparing apples and oranges?

ANSWER [B. I. Spinrad]: Yes to all parts.

QUESTION [Francis Green (National Carbon Co.) to the panel]: Is there an economic incentive, based on the level of neutron economy improvement, for developing an uncanned graphite moderator or a graphite-clad uranium-carbon fuel element compatible with sodium? What is the present estimate of limiting temperature of compatibility between the two materials?

ANSWER [B. I. Spinrad]: I do not know of any economic incentive for the development suggested. However, I would say that there is technical incentive in that the fabrication of fuel claddings often adds problems to reactor operation. New materials compatibilities must be considered, and there are extra barriers to heat flow and extra neutron poisons. One primary residual reason for cladding ceramic fuels is their tendency to spall and crack under thermal shocks or irradiation. This characteristic must be overcome in any new ceramic fuel development before compatibility with sodium becomes limiting.

QUESTION [G. W. Wensch to S. Siegel (Atomics International)]: In the last two 10-year plan studies of the AEC in which AI participated, the status of sodium graphite reactors for 300 Mw(e) was as follows:

Reactor coolant inlet	615°F
Reactor coolant outlet	945°F
Fuel	Uranium-10% molybdenum
Burnup	8000 Mwd/ton
Maximum center fuel temperature	1050°F

The following economics were estimated:

Capital cost, \$300/kw	6.0 mills/kw-hr
Fuel-cycle cost	4.4 mills/kw-hr
Operating, insurance	1.0 mills/kw-hr
Total	11.4 mills/kw-hr

What data and information exist which indicate that these conditions have improved and that more attractive economics are possible?

QUESTION [G. W. Wensch to Emerson Jones (Consumers Public Power District)]: Suppose that both water-cooled and sodium-cooled reactors are economical for power generation. Which of these two general concepts would you select for power generation, and what reasons do you use as the basis for your selection?

QUESTION [G. W. Wensch to K. P. Cohen (General Electric Co.)]: It is cheaper to prepare uranium and plutonium in metallic forms than to convert these fuels to oxides and perform subsequent fabrication techniques required in order for oxides to be used in reactors. In view of the complicated fabrication processes needed, what are the prospects for economic nuclear power with mixed oxides?

QUESTION [G. W. Wensch to Mr. Werden]: You are familiar with the impact that the population growth of Southern California will have on electrical power sources there. Assuming that nuclear power is demonstrated to be economical in Southern California, what other criteria would you apply in selecting a reactor concept for power generation?

QUESTION [J. T. Jackman to K. P. Cohen]: What are the projected power costs and fuel cycle costs for the fast oxide reactor? What plant size is necessary to achieve such costs? (It is assumed that your design has been optimized for power generation, not for breeding. Is this true?)

QUESTION [A. Amorosi (Atomic Power Development Associates, Inc.) to G. W. Wensch]: It is clear from the remarks of the speakers that economic power can be achieved, but it will

take considerable research and development effort to bring this about. Can you indicate the AEC plans in sponsoring research and development?

QUESTION: Variable fuel element flow orifices have been referred to at this meeting. Do these automatically adjust by sodium temperature? If so, how important are they to the safety of the reactor and what experience exists to establish their reliability?

QUESTION [J. L. Allen (Philadelphia Electric Co.) to Emerson Jones or K. P. Cohen]: Much has been said about improving fuel cycles and equipment and the necessity of making nuclear-power generation competitive with conventional plants. Please comment on how the recently proposed amendment to the site criteria is assisting toward the goal of making nuclear power competitive, taking into consideration the fact that a conventional plant can be located in the center of a load growth wherever it develops.

QUESTION [C. R. Potter (Rural Electrification Administration)]: What is the final HNPF cost? How does it break down in terms of nuclear and nonnuclear portions?

QUESTION: What is the target date for 1000°F operation with reactor?

QUESTION: Does sodium offer any attractive possibilities for nuclear-fueled gas turbines, or is gas alone a better medium for both coolant and working fluid?

QUESTION: What is the full-load auxiliary power for HNPF on fossil fuel and on nuclear fuel?

QUESTION [M. Egleme (Belgonucleaire)]: How is the reactor inlet temperature controlled in the HNPF? Is the primary sodium flow constant or variable with power?

QUESTION [G. Stern (Nuclear Development Corp. of America)]: Dr. Cohen spoke of a burnup of > 100,000 Mwd/metric ton for UO_2 - PuO_2 fuel. How can this be reconciled with the generally accepted and experimentally verified practical limit of ~55,000 Mwd/metric ton for straight UO_2 fuel?

QUESTION: In view of Dr. Hilberry's enthusiasm for sodium reactors, will AEC initiate an active demonstration program for that type? Assuming a "small" amount of sodium leaks within the reactor vessel, what would be the maximum credible damage?

QUESTION [F. A. Smith to S. Siegel]: Define what an AI "guarantee" of 8-mill power means to the customer; i.e., if the reactor does not operate, will AI reimburse financially the customer for loss of income revenue during down time beyond 80 per cent.

QUESTION [F. A. Smith to Atomics International]: Please discuss whether technically the SRE should be considered a reactor experiment or a demonstration prototype for HNPF.

QUESTION [P. Augustine (U. S. Atomic Energy Commission)]: What are the costs of generating and transmission and distribution in the CPPD system, and what is expected of the Hallam reactor in reference to the above-mentioned costs?

QUESTION [K. W. Foster (Atomics International)]: To what extent can quality control be relaxed for sodium components such as steam generators? What risks are involved? What economics can be gained?

List of Attendees

AERO RESEARCH INSTRUMENT CO., INC: John LaVan

ALCO PRODUCTS, INC: C. S. Seabring, Roy D. Seifert

ALLIS-CHALMERS MFG. CO: Charles L. Teefer, Richard Higgins

AMERICAN MACHINE & FOUNDRY CO: C. A. Prendergast, Ivan F. Tilsley

ARGONNE NATIONAL LABORATORY: B. I. Spinrad, Norman Hilberry, F. W. Thalgott, S. Lawroski, L. J. Koch, Frank G. Foote, W. R. Simmons

U. S. ATOMIC ENERGY COMMISSION: Thomas P. Heckman, G. W. Wensch, Claude A. Pursel, Thomas A. Nemzek, Robert Aronstein, Leonard A. Redecke, G. V. Packer, R. E. Harris, F. A. Smith, George Y. Jordy, Ray D. Walton, Martin Wiener, Curtis A. Nelson, James F. Smith, Phil McGee, A. G. Lane, Joseph G. Gratton, John P. Holliday, E. K. Loop, R. E. Buntain, Pat Augustine

ATOMICS INTERNATIONAL: R. Clyde Gerber, Lyle E. Glasgow, Jim Owen, R. W. Dickinson, P. R. Keep, Ken W. Foster, John D. Williams, Sid Siegel, L. Newcomb, John Moore, L. R. Bivins

ATOMIC POWER DEVELOPMENT ASSOCIATES, INC: Alfred Amorosi, John G. Yevick, J. J. Morabito
 BABCOCK & WILCOX CO: Robert W. Carlson, J. Schlichting
 BRITISH EMBASSY: D. G. Avery
 CINCINNATI GAS AND ELECTRIC CO: Edward C. Pandorf
 COMBUSTION ENGINEERING, INC: R. Reimer, B. J. Tompkins, W. C. Novick, F. C. Hensel, N. D. Romanos
 COMMISSARIAT À L'ÉNERGIE ATOMIQUE, FRANCE: John R. Leduc*
 CONSUMERS PUBLIC POWER DISTRICT: Frank P. Grant, R. L. Schacht, W. P. Venable, Emerson Jones, Charles D. Sayre, Robert S. Kamber, J. Denny Cochran, L. John Cooper, Ivan O. Sunderman, Glen E. Schukei, Lloyd E. Keller, Richard L. Large
 DETROIT EDISON CO: Thanos Alexiou, Kozo Odajima
 E. I. DU PONT DE NEMOURS & CO., INC: Frank J. Holt
 FACTORY INSURANCE ASSOCIATION: F. J. Zeleny
 FOSTER WHEELER CORP: J. F. Schonberger
 FOXBORO CO: Leo P. Johnson
 GENERAL ATOMIC: Don B. Coburn
 GENERAL ELECTRIC CO: Karl P. Cohen, George Billuris, R. W. Lockhart, B. G. Voorhees
 GENERAL KINETICS CORP: Herman L. Paul
 GRISCOM-RUSSELL CO: R. W. Schroeder, J. R. Christman, P. S. Otten
 KAISER SERVICES: J. T. Jackman
 LOS ALAMOS SCIENTIFIC LABORATORY: W. R. Wykoff, R. E. Peterson
 MALLINCKRODT CHEMICAL WORKS: M. A. Thorpe, R. B. Clark
 MARTIN CO: Edward Scicchitano
 METALS AND CONTROL CORP: Joel F. Williams
 MINERAL CONCENTRATES & CHEMICAL CO., INC: R. L. Bender, R. S. Myre
 MINNEAPOLIS-HONEYWELL REGULATOR CO: Thomas Schlottenmier
 MONSANTO RESEARCH CORP: L. J. Wittenberg
 NATIONAL AERONAUTICS AND SPACE ADMINISTRATION: L. R. Nichols
 NATIONAL CARBON CO: Francis P. Green, David W. Sheddon
 NIAGARA MOHAWK POWER CORP: G. K. Rhode
 NUCLEAR DEVELOPMENT CORP. OF AMERICA: Kurt Goldmann, George Stern
 OAK RIDGE NATIONAL LABORATORY: H. Wesley Savage, Blynn E. Prince, Jack H. DeVan
 PHILADELPHIA ELECTRIC CO: John L. Allen
 PICKARD-WARREN-LOWE ASSOC: William W. Lowe
 POWER REACTOR DEVELOPMENT CO: Robert W. Hartwell
 PRATT AND WHITNEY AIRCRAFT: W. J. Lueckel
 PUBLIC HEALTH SERVICE: Mortin I. Goldman
 RURAL ELECTRIFICATION ADMINISTRATION: Wade M. Edmunds, Charles R. Potter
 SARGENT AND LUNDY: Ivar Nelson
 SPEER CARBON CO: Melvin B. Redmount
 STEARNS-ROGER MFG. CO: Traver E. Metcalf
 SWEDISH EMBASSY: Pelle Isberg
 UNITED AIRCRAFT CORP: Ned C. Rice
 UNITED SHOE MACHINERY CORP: Elliot R. Black
 WESTINGHOUSE ELECTRIC CORP: M. Shimizu, C. C. Randall, W. A. Bryant, S. N. Tower, H. Harty, Donald Nixon
 YANKEE ATOMIC ELECTRIC CO: Donald E. Vandenburg
 RECORDER: Albert Morwitz
 AFFILIATION NOT LISTED: B. Gleme, Louis Schustik

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* Assigned to Atomic Power Development Associates, Inc.