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INTERIM REPORT: FARET EXPERIMENTAL PROGRAM

by

The FARET Program Group

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I. INTRODUCTION

A. Background Considerations

The Fast Reactor Test (FARET) program is part of the United States Fast Reactor Program directed toward the demonstration of breeder reactors essential to the long-range goal of utilizing fertile fuels.⁽¹⁾ Within this program is the current thinking that the early construction and operation of two or three prototype nuclear power plants of the order of 250 Mw(e) will lead eventually to the construction of practical and economic full scale breeder reactors by the early 1980's.⁽²⁾

The development of competitive fast power-breeder systems will depend on fuel stability under irradiation, high power densities, performance at high temperature, high fissile atom burnup and, for nonmetallic fuels, the safe operational reactivity feedbacks. The long-range purpose of the FARET program is to investigate these aspects, experimentally and theoretically, for a variety of fuel types and designs.

The initial effort in the program is being directed toward ceramic fuels since these have good high-temperature properties. Experiments to date have indicated that the ceramics possess a potential for high atom burnups at temperatures much greater than those employable for metal fuels. Further, it is expected that in large dilute systems, the light atom content present in the ceramics would degrade the gross structure of the neutron spectrum to an extent which would enhance the prompt negative feedback effect due to Doppler broadening.

An uncertainty with the ceramic fuels is the expansional characteristic. The thermal expansion of metallic fuels and corresponding reactivity effects under normal conditions should provide an inherent negative prompt coefficient due to the axial expansion of the fuel elements. However, experimental information on the dimensional behavior of ceramic fuels due to temperature changes, rate of temperature changes, history of the fuel, and degree of irradiation is at present limited. This behavior must be established under conditions which would allow a high degree of understanding and predictability.

Large systems, for a given composition and shape, result in a decrease of neutron leakage. Reactivity feedback having leakage-dependent

components, such as the sodium density effect, could in these large systems be nullified and eventually reversed in sign. Analysis indicates that the sodium coefficient should be expected to become positive for core sizes being contemplated as future power systems. The positive coefficient results mainly from changing the competition between fission and capture reaction rates caused by the spectral hardening. The coefficient may be maintained negative if the volume fraction of sodium is very high. This, however, would result in a loss of breeding gain and power densities.

The sodium coefficient could then be considered as introducing an upper limit to the size of reactor systems. To increase the total power output with this limitation, reactor fuels and designs must be directed toward higher power densities.

Engineering and physics data will have to be developed to allow an optimum safe design of large systems which properly takes into account the competition between the various prompt and delayed shutdown mechanisms. Although, to a great extent, physics measurements may be obtained from critical experiments, these do not simulate actual conditions of reactor operating temperatures, flow rates, and reactor coolant for which FARET is urgently needed. The facility will be directed to: (1) provide appropriate environment for fast reactor development such as required for high power densities and high temperature; and (2) resolve major unknown power-reactivity parameters of reference United States reactor designs.

The FARET work will be supplemental to that conducted in the EBR-I, an experiment in which feasibility of breeding was established, and the EBR-II, which is currently committed to pyrometallurgic metallic fuel-cycle studies.

B. Description of FARET Capability

The FARET facility will be located at the National Reactor Testing Station (NRTS) on a site approximately 4000 ft (1200 m) southeast of EBR-II. This facility is capable of accepting reactor experiments with a heat generation of 50 Mwt and a maximum mean outlet temperature for the reactor coolant of up to 1200°F (650°C). It is designed to accommodate a variety of reactor configurations and sizes operating over the full range of power levels (0 to 50 Mw) and at a variety of temperatures. The salient features include adaptability to a variety of core loadings, experimental access to the core for measuring core parameters, and operational flexibility.

FARET will be used to test small, compact cores at high temperature and high power density to assess the performance capabilities of fuel systems of interest. FARET will also be used for the determination of the neutronics of large, dilute reactor systems. Of major interest are the reactivity coefficients of such systems, particularly of fuel systems with

unknown or unreliable thermal-mechanical properties (such as ceramics and, possibly, highly irradiated metallic alloys and cermets). These systems may be tested to failure (with extensive instrumentation) in order to establish their specific operational limitations. It is also planned to investigate advanced fuel concepts which present too great a degree of risk to be considered initially in existing reactors, including those which present a high probability of contaminating the primary coolant system (such as vented fuel elements). In addition to the capability for reactor operation at temperatures up to 1200°F (650°C), provisions will be made for operation of individual fuel assemblies or small zones of the reactor at higher temperatures. The reactor system consists of a primary loop including reactor vessel, pump, and primary heat exchanger containing sodium as primary coolant. The secondary system is a liquid metal loop for removing heat from the primary heat exchanger and rejecting this heat to the atmosphere. Associated systems and laboratory space will be available for measurement of coolant quality. Equipment for handling fuel in and out of the reactor vessel will be provided. There will be a control and instrument room for central control of the reactor, and an office annex for support of technical office work and administrative functions.

The development of the design and construction of the facility is a separate phase of the FARET objectives, which is carried out by an ANL project group and architect engineer, the work of which is reported in preliminary design reports. A more detailed description of the facility is reported in Ref. 3.

C. Program Objectives

The fast reactor test facility is designed for conduction of experimental work which may be separated into two general categories:

- (1) engineering performance; and
- (2) reactor dynamics and safety.

The function of the engineering-performance phase of the program is the advancement of the engineering technology required for the development of prototype fast power breeders. The areas of investigation will be fuel and fuel-clad concepts and designs, high power densities and high temperatures, high fissile atom burnup, and component- and system-operating experience, all under conditions approaching those expected in an actual power breeder.

The dynamics part of the program is directed toward determining nuclear characteristics which are of vital interest in connection with the stability and safety of large dilute systems. Particular importance will be assigned to the observation of reactivity effects with respect to changes in temperature of fuel, sodium, clad, and structure as these temperatures respond to simulated actual operating conditions.

Currently, the above two phases of the FARET program are being pursued with equal effort which will allow a realistic evaluation of scheduling times for the development work activities. It may be desirable to conduct reactivity coefficient experiments initially because of the longer test duration required for the fuel-irradiation phase of the engineering program. Moreover, since reactivity experiments will be performed at lower irradiation levels, the system-irradiation problem would be lessened in the early phases.

1. Engineering

One of the requirements of reactor cores for economical, large prototype reactors is that the fuel should remain in the reactor for a long period unless fuel-handling and refabrication costs can be reduced appreciably. Currently, metallic fuels are limited to 1-2% burnup of heavy atoms at high temperature. Ceramic materials hold promise for increased burnup of the order of 10%. For this reason, the potential of these materials must be investigated and their performance limits should be established. It will be necessary to test different fuel and clad combinations on the basis of specific variables, such as fuel temperatures and composition of the fuel, in an environment of liquid metal and a fast neutron spectrum.

The procedure will be to pretest fuel and clad combinations in the form of samples such as pins, then to proof-test complete subassemblies, and, finally, to operate a completed core of the most promising materials. The development of the manufacturing processes would also be studied concurrently so that facilities for fabricating prototype cores would be available on time. The necessary experience for full-scale operation can then be adequately pursued by means of prototype reactors.

Other requirements are that the fuel should operate at highest possible power density and high thermal performance. Thus, it will be necessary to derive feasible power densities for application to the prototype program. Because the future prototype plants will be based on the use of plutonium fuels, the FARET program will require the investigation of the overall performance of the type of fuel particularly suitable to these plants. Since the carbide fuel has properties of good thermal conductivity, high-temperature stability, low vapor pressure, and high fissile atom density, the investigation of this fuel is given high priority in the FARET program.

Although the initial experiments will most likely use stainless steel as a cladding material, early work on high-strength refractory base alloys has indicated the desirability of including these materials in the FARET program.

To study the engineering performance characteristics, small high-power-density cores will be utilized. Small cores represent a lower fuel investment per experiment without compromising the desired information.

In the FARET facility with a total heat removal capacity of 50 Mwt (continuous), the PuUC-fueled, sodium-cooled system with an average specific power of the order of 0.64 Mw/kg Pu could be attained in core sizes (length-to-diameter ratio of 1) of approximately 42 liters. The power density is based on a core composition having a fuel/coolant/structure volume fraction of 30/52/18 with enrichments of 45%.

The theoretical maximum specific power is 1.05 Mw/kg Pu for the above system. This thermal output is based on a $v\Delta T$ for the coolant of about 10,000 (ft)(°F)/sec [1500 (m)(°C)/sec] and a radial form factor of 0.79, where v is the coolant velocity and ΔT the average rise of coolant temperature through the core. This is fairly consistent with current sodium velocities of approximately 33 ft/sec (10 m/sec) and a temperature differential of 300°F (150°C). An experiment of this type will allow heat-transport studies to be made with core power densities in the range of 1 to 1.2 Mw/liter with the use of equipment (pumps, etc.) in keeping with current technology. For higher power densities, a carbide core of 31 liters would allow core power densities up to 1.67 Mw/liter for the above composition with an enrichment of about 65%. The core arrangement and geometry of the prototype reactor will be represented to the maximum extent possible. In the case of fuel composition of a smaller core, the Pu²³⁹ content could be made identical with that of a large reactor by the replacement of U²³⁸ by U²³⁵ to achieve criticality. The substitution of U²³⁵ for U²³⁸ would not alter the metallurgical or physical properties of such systems.

The high power density is also desirable since irradiation studies on fuel (or fuel-clad system, or isotopes of interest) can be made at accelerated rates. With the above power density, a 1% burnup of heavy atoms can be obtained in approximately 30 days. The effects due to thermal stresses, fission gas release, and thermal conductivity changes of the ceramic fuel will be established within this burnup period. A one-percent burnup should also begin to give important information relating to long-term reactivity effects.

The high thermal performance of a carbide fuel may be compromised by the clad material used. An orientation survey of high-temperature properties and compatibilities of structural materials was made for the purpose of selecting a material having the minimum limitation on the physical conditions under which fuel elements may be studied. The materials included in the survey are the more common materials, such as the stainless steels; nickel-chromium alloys such as the Hastelloys, and the refractory metals. The materials are to be considered as representative of the type rather than as the specific choice. The relevant information available to date is compiled in Table XIV (see p. 63).

Another area of study is the small core loadings with part metal and part ceramic fuel. The advantage of these loadings is twofold. First, a known prompt shutdown feedback is introduced into the system through the expansion of the metal fuel. This initially assures a safety aspect while studying the dimensional behavior of the ceramics. Second, fuel-loading requirements per experiment are further reduced, so that the costs of investigating loadings are correspondingly reduced.

The development of reactor concepts also requires the testing of special features of design, such as fuel venting, and the testing of a complete integrated reactor system scaled to the prototype system.

The performance of a fuel element suitable for high rates of heat transfer and high burnup will be limited by the ability of the cladding material to withstand stresses and/or microstructural changes due to temperature, accumulation of fission product gases, and fuel swelling. One approach will be to utilize improved cladding materials, as has been already proposed. These are the refractory type and super alloys which have relatively high cross section for absorption, affecting the neutron economics so that for large reactors the use of these would depress the breeding gain and make for an autocatalytic sodium temperature effect.

If the requirements of fission gas containment by means of the clad are waived, then the clad thickness can be reduced. This will then result in a reduction of the volume fraction of this structural material and improvement of neutron economy, heat transfer, and burnup. The relaxation of requirements for fission gas containment in the fuel cycle will result in lower fabrication costs as well.

The problems in connection with vented fuel assemblies need considerable prooftesting. The purposes of the venting test would be to determine the spread of fission products throughout the system and to develop the means for controlling acceptable equilibrium conditions. The fuel assemblies will be developed to arrive at the most satisfactory means for venting either directly into the coolant system or into a reservoir.

2. Reactor Dynamics and Safety

The determination of the individual physics parameters important to the fast breeder reactor program may be divided into three general areas:

- (1) measurement of the reactivity coefficients which are involved in the total dynamic response;
- (2) determination of relative rates of neutron reaction by irradiation;

- (3) determination of the dynamic response of the reactor parameters (temperature, power, and flow rate) to each other and to external conditions.

a. Reactivity Effects

For the construction of large prototype reactors it must be shown that a design can be achieved which is simultaneously economically feasible and safe by virtue of a negative reactivity coefficient derived from: (1) structural expansion; (2) fuel heating; and (3) changes of the sodium density.

Metallic fuels have demonstrated inherent negative prompt reactivity coefficients due to fuel expansion. For the ceramic fuels of interest, the expansional effects are uncertain at this time so that there is a primary need and justification to determine these. On the other hand, for large, ceramic-fueled dilute systems, the light atom content present will degrade the gross structure of the neutron spectrum to an extent that would enhance the Doppler effect.

Therefore, the several components of reactivity effects of interest to large ceramic reactors are:

- (1) Doppler effect;
- (2) thermal expansion of fuel elements;
- (3) sodium expansion due to heating and vaporization; and
- (4) rod bowing and movement.

(1) Doppler Effect

The Doppler effect in a reactor is the effect on the reactivity caused by the variation of absorption and fission rates with the temperature of the fuel. This variation is due to the fact that most of the absorptions and fissions occur in narrow energy bands, or resonances. As the fuel is heated, the thermal motion of the fuel atoms allow neutrons, which originally were not of resonance energy, to have a relative motion which puts the neutrons into "resonance" with the atoms. The effect is an increase in the width of the resonance, that is, the resonances are broadened.

Because of the high cross sections at a resonance energy, the neutron flux is depleted at this energy. The broadening of the resonance results in decreasing the depletion of the flux. A particular reaction rate over the energy interval is the integral of the product of the neutron flux and the reaction cross section. The maximum increase in the reaction rate due to resonance broadening occurs when the neutron flux

depletion is strongly related to the resonance. If the fine structure of the neutron spectrum in this energy interval is determined mostly from other sources, the increase in the reaction rate is lessened. An example of this is the case of an absorber in an infinitely dilute system in which the resonance integral is temperature insensitive. Another case arises when a heated small sample is placed in a nonheated reactor spectrum.

The increase in the reaction rate is also lessened if the resonances are closely spaced (overlapping effects). At very high neutron energies, for which the level density in nuclei is high, the room-temperature motion of the atoms suffices to smear out the resonances completely. Therefore, hardened-spectrum reactors (which include virtually all small fast reactors) have a negligible Doppler effect.

Estimates indicate that the part of the reactor spectrum which contributes significantly to Doppler effect is the region below about 10 kev, and only very large (and, therefore, less enriched) fast reactors have an appreciable contribution from this group. This is a gross structure spectrum effect, and can be considered as a normalizing factor.

The Doppler effect depends on the relative contributions of U^{238} and Pu^{239} (or U^{235}), which give, respectively, negative and positive effects. These contributions can only be assumed, in turn, from detailed knowledge of the parameters of the individual resonances. In the energy range of interest, only the statistics of the parameters are known; although good data on U^{238} are now being procured, the fissionable isotopes do not appear to be measurable to the required precision by known techniques. The reported effects indicate that 60% of the Doppler effect may be contributed in the energy range from 9 kev to 1 kev. This region involves unresolved levels for both fertile and fissile materials. Another 35% may be contributed from the region from 1 kev to 300 kev. In this range, only the U^{238} resonances are resolved. The levels of Pu^{239} are resolved up to 52 ev and those of U^{238} up to 1 kev. On theoretical grounds, sufficient information now exists such that predictions of the Doppler effect with 30 to 50 percent accuracy are the state of the art. This precision depends largely on the U^{238} effect being dominant for low plutonium enrichments.

A comprehensive preparation of experiments oriented toward isolating the Doppler coefficient from all other reactivity effects will be underway for the ZPR-VI, a split-table critical assembly. This program, planned in a flexible zero-power critical assembly, spans a range of experiments from the oscillation of hot and cold samples in a system at room temperature to the temperature oscillation of a test zone containing molten sodium in a heated region. Although the sufficiency of critical-assembly work is anticipated, experiments in an actual reactor environment would augment the zero-power experiments. Actual power experiments would also make it possible to extend fuel temperatures up to the melting point of the fuel because of the use of nuclear heating.

The difficulty in measuring the Doppler coefficient directly arises from the fact that the simultaneous fuel-expansion coefficient is usually of the same sign and magnitude as the computed Doppler coefficient. An experiment in which the reactivity effect is produced only by changing the fuel temperature yields directly the "fuel temperature coefficient." Ideally, if this experiment is conducted in a reactor environment of interest (hot sample in a hot reactor), this quantity could be used directly as a design parameter without attempting to separate the two effects. This measurement is most valuable since the fuel temperature coefficient will depend on the details of the fuel configuration and rod size, the cladding material and coolant content, and the temperature distribution in the fuel.

(2) Thermal Expansion of Fuel

Thermal expansion of fuel material can be directly and reliably estimated for metallic fuels, and the corresponding reactivity effect may be readily measured in critical assemblies. The reactivity effect is negative and is due to axial expansion of the fuel elements. It dominates all prompt reactivity effects for small cores. For large cores it can be of the same magnitude as the Doppler effect, which is expected to be the major prompt negative effect.

For highly irradiated metal-fueled reactors and for ceramic-fueled reactors, the actual dimensional response of the fuel to thermal changes is not well known at present and is especially uncertain for ceramic fuel; consequently, the corresponding reactivity coefficient cannot be predicted. The dimensional response is sensitive to temperature, metallurgy of the fuel, irradiation, and rate of temperature change. The expansional problem may be eased through proper design of the fuel element. A tightly clad fuel element with the ceramic very close to the theoretical density may ensure an axial expansion upon heating. The requirements of radial constraint of the ceramic would not necessarily be very high.

FARET can be used to determine the practical fuel-expansion coefficient of reactor cores. For small cores (50- to 100-liter size), enrichment is high, the spectrum is hard, and the Doppler coefficient is negligible. Prompt, or short-delay, reactivity effects are due to fuel expansion, sodium expansion, or structural distortion. The last of these effects may be eliminated by using a rigid core structure. The fuel coefficient may be determined from transfer-function measurements; by varying the sodium flow rate, and extrapolating the prompt coefficient vs flow rate curve to infinite flow rate; or by programming simultaneous power and flow-rate variations of the slow transient power type.

An important point is that these measurements can be repeated several times on a core which has undergone considerable burnup and thermal cycling. Thus the fuel coefficients of reactivity can be examined under truly realistic conditions.

(3) Sodium Expansion

The thermal expansion of the sodium coolant flowing through a reactor core can be predicted reliably from knowledge of the power input to the coolant. The corresponding reactivity effect can be measured fairly directly in a critical assembly. Therefore, in principle, the sodium coefficient of reactivity per se need not be measured in FARET. However, because of the possibility of large variation in magnitude or even change in sign of the local sodium coefficient through some cores of possible interest, direct observation under power conditions may be sought.

In measuring any power-reactivity effect, it will be necessary to heat sodium. The sodium coefficient can be determined by varying the coolant flow rate, which will affect the sodium temperature as well as changing the temperature level of the coolant (inlet temperature, flow rate vs power level). The effect of fuel temperature having been determined, the effect of the sodium can be isolated.

The determination of Doppler coefficient and sodium coefficient will be somewhat complementary because the determination of one is of assistance in determining the other. In some of the proposed experiments, the success of one depends upon the isolation of the other. In FARET, the large coolant flow available will be of great value in isolating the effect of the sodium because of the capability to "over-cool" during some of the investigations.

The fuel-temperature effect and, in particular, the Doppler effect further depend on the sodium-density effect. Again, in FARET the ability to vary coolant temperature at the inlet allows this interdependence to be studied.

(4) Rod Bowing and Movement

In this category are included distortions of the core which are induced by differential heating and cooling of the fuel elements or by thermal expansions of the core support structure. For example, the prompt positive power coefficient observed in the EBR-I (Mark II) core was found to derive from inward bowing of the unsupported fuel rods in the core. The addition of restraint to the fuel to prevent inward bowing completely removed the positive power coefficient. Thus, it is believed currently that the origins of these effects can be eliminated by proper mechanical design. The reactivity effects of core distortion can be measured readily in critical assemblies, and these are usually small for large reactors.

b. Determination of Reaction Rates by Irradiation

These measurements include ratios of capture to fission cross sections of U^{233} , U^{235} , Pu^{239} , and Pu^{241} , and capture cross sections for U^{236} , U^{238} , Pu^{240} , and possibly Pu^{242} . It may also be possible to determine the capture cross section of Np^{239} and Pa^{233} , which have the relatively short half-lives (2.3 and 27 days, respectively); these are increasingly important as neutron flux increases and are quite significant at power levels of 1 Mw/liter. These neutron reactions cannot be measured in critical assemblies. The measurement of these cross sections will be essential to the accurate analysis of reactor performance and, in particular, the breeding of projected prototype fast breeder reactors. Furthermore, these measurements in FARET can complete the comparison between theoretical predictions and performance measurements of zero-power critical assembly experiments.

c. Reactor Dynamics

A reactor must be stable in the range of its rated power; if a small excursion is initiated in a reactor, the reactivity of the system must compensate and stop the excursion. To satisfy these conditions, a number of reactivity coefficients, such as those for sodium expansion and expulsion, fuel-cladding and support-structure expansion and distortion, fuel expansion, and Doppler effect must be explored. In addition to the magnitudes of these effects, the time delays associated with them must be examined. The analysis of these effects involves an extended program, and since it is uncertain that all effects of importance have been included, it is common practice to examine problems of stability and safety by an integral approach. A system is assembled which is a physical model of the proposed reactor, and its dynamic performance is evaluated. The two basic experiments for observing dynamic behavior are transfer-function measurements and excursion tests.

(1) Transfer Functions

Transfer-function measurements are made by superimposing an oscillatory reactivity input on the reactor operating at power, and then mapping the response of the reactor power to this input. When possible, other variables, such as fuel and coolant temperature, are also correlated with the input signal. The result of these measurements is the transfer function of the reactor, which displays incipient resonances, or signal frequencies at which potential instabilities might occur. The absence of such resonances is proof of stability; when they are observed, they may be investigated by varying operating conditions to see under what conditions they are enhanced. A mathematical model can be constructed to correlate these observations, and a prediction of stability can be made by applying this model to extrapolated operating conditions.

Thus, the stability of reactors can be determined by purely empirical means. In addition, the transfer function provides reassurance that unknown types of reactor feedback are not present (or its converse).

In FARET, transfer-function measurements can be made with both uniform and zoned core loadings. Uniform loadings ranging in size from 50 to 500 liters can be investigated at variable specific powers; virtually constant thermal conditions can be maintained over this range. Aspects of design which are primarily mechanical (fuel-element distortion and expansion, for example) are emphasized in this range in which stability can be examined with high precision.

Certain reactivity coefficients are predictably variable with reactor size, either directly or because critical enrichments decrease and reactor spectra soften as size is increased. Therefore, it is desirable to extend the transfer-function measurements to zoned systems. The concept is to provide a central region, of about 100-liter volume, which would be a representative portion of a large reactor. This zone would be embedded in a standard region of high reactivity and carefully mapped feedback. The experimental configuration would be calibrated in a critical assembly to determine the relative contributions of perturbations in the two zones to reactivity. Careful transfer-function measurements with FARET at power could then, in principle, be extended to determine the transfer function of a large reactor whose composition is that of the FARET central zone.

It should be emphasized that such "zoned" transfer-function measurements have not been performed previously, and there is no assurance that significant results can be obtained. If the concept proves attractive, considerable study and development of techniques will be required.

(2) Excursion Tests

Reactivity effects associated with short delay times can be studied more conveniently with excursion tests involving small reactivity. By using a short experimental time scale, the effects occurring with appreciable delay are effectively eliminated. These experiments can be performed with the same systems that are used for transfer-function measurements. Here, again, no experience with experimental zoned systems exists, and considerable evaluation and development will be required. Coupling between the test zone and driver must be determined.

II. DESIGN CONSIDERATIONS

A. Basic Core Plan

The basic plan of the reactor core (see Fig. 1) consists of an hexagonal array of fuel assemblies in a reactor vessel of approximately 6-ft (180-cm) diameter.

Fig. 1. Core Section

The area occupied by each hexagonal fuel assembly is determined by a lattice spacing of 2.320 in. (5.90 cm), the same as used for EBR-II. Table I illustrates the extent of the range of core sizes which can be assembled in the FARET reactor vessel.

Table I

POSSIBLE CORE SIZES (Excluding Reflector)

Core Parameter	Small Core	Medium Core	Maximum Core
Diameter (cm)	44	95	150
Length (cm)	33	71	120
Volume (liters)	50	500	2000

At present, three separate cores have been selected for the experimental program. They are: (1) a zoned core, (2) a medium core, and (3) a small core.

For each core, there are three basic types of subassemblies comprising a reactor loading: fuel, control, and reflector (see Fig. 2). Also, for each core, different fuel elements will be required. In the zoned-core experiment, fuel elements are also used in the driver and buffer zones. Table II gives various core parameters and detailed specifications of each fuel element.

The design of the reference subassembly selected employs the EBR-II hexagonal shape. This was established in order to utilize the existing engineering development and reduce the cost that would be required for tooling and fixtures. A representative fuel assembly is shown in Fig. 2. Although each fuel element has its own characteristics, certain features (at present) are the same.

A single subassembly size is employed throughout the reactor, which is constructed on the basis of a close-packed geometry. The hexagonal assembly outer dimensions are 2.290 in. (5.82 cm) across flats with a wall that is 0.040 in. (0.102 cm) thick. This size appears to allow all of the various fuel-pin dimensions of the planned experiments.

The design of the reference subassembly utilizes some of the features of the EBR-I, Mark III design. For example, the top 5 ft (150 cm) of the subassembly is a shielding extension providing for attenuation of fission product and sodium activity during shutdown. The handling extensions protrude above the sodium level, so as to reduce the complexity of fuel handling. The lower ends of the extensions serve to define the top of the reactor coolant outlet region or plenum. Since there must be clearances between the extensions, to permit subassembly insertion or withdrawal, primary coolant can leak between the extensions, from the reactor outlet region to an atmospheric pressure over flow region near the top ends of the extensions. Since that leakage should amount to only a small percentage of the nominal primary flow rate, calculations have been made to determine just how much leakage can be expected. It was found that with the 0.030-in. (0.075-cm) clearances planned between extensions offering the only resistance to flow, a 3-psi (0.2-atm) pressure drop across the extensions will cause leakage amounting to about 20 percent of the rated primary flow. [The maximum pressure drop expected across the extensions is 3 psi (0.2 atm).] To reduce the leakage, it is planned to enlarge the extensions over a short axial distance near their bottom ends, so that an additional resistance to flow is provided. If these enlargements are sized so that they leave a 0.004-in. (0.010-cm) clearance between subassemblies, the leakage is calculated to be reduced to about 9 percent of rated primary flow with the 3-psi (0.2-atm) drop; a local 0.002-in. (0.005-cm) clearance should reduce the

Fig. 2. Core Subassemblies

Table II
TENTATIVE TEST CORE PARAMETERS

Parameters	Zoned Core					Medium Core	Small Core	
	Test Zone			Buffer	Driver		1	2
	1	2	3					
Core								
Equivalent Diameters in (cm)	14 8 (37 6)	14 8 (37 6)	14 8 (37 6)	27 4 (69 6)	55 0 (139 7)	50 6 (128 5)	14 8 (37 6)	14 8 (37 6)
Height, in (cm)	36 0 (91 4)	36 0 (91 4)	36 0 (91 4)	36 0 (91 4)	36 0 (91 4)	12 65 (32 1)	14 8 (37 6)	14 8 (37 6)
Total Volume liters	102 0	102 0	102 0	248 0	1058 0	419	41 7	41 7
Composition v/o								
Fuel	32 0	32 0	32 0	32 0	32 0	32 0	32 0	30 0
Sodium	45 0	45 0	45 0	52 0	52 0	45 0	45 0	52 0
Structural Metal	16 0	16 0	16 0	16 0	16 0	16 0	16 0	18 0
Gas Void	7 0	7 0	7 0	-	-	7 0	7 0	-
Hexagonal Subassemblies*								
Number Required	37	37	37	90	384	433	37	37
Fuel Elements (Pin Type)								
No Elements/Subassembly	7	7	7	19	19	7	7	91
No Elements/Core	259	259	259	1710	7296	3031	259	3367
Fuel Diameter in (cm)	0 524 (1 331)	0 524 (1 331)	0 524 (1 331)	0 316 (0 803)	0 316 (0 803)	0 524 (1 331)	0 524 (1 331)	0 140 (0 356)
Clad OD in (cm)	0 640 (1 625)	0 640 (1 625)	0 640 (1 625)	0 354 (0 899)	0 354 (0 899)	0 640 (1 625)	0 640 (1 625)	0 154 (0 391)
Clad Thickness in (cm)	0 033 (0 084)	0 033 (0 084)	0 033 (0 084)	0 012 (0 031)	0 012 (0 031)	0 033 (0 084)	0 033 (0 084)	0 007 (0 018)
Bond Annulus in (cm)	0 025 (0 063)	0 025 (0 063)	0 025 (0 063)	0 007 (0 018)	0 007 (0 018)	0 025 (0 063)	0 025 (0 063)	-
Bond Material	Gas	Gas	Gas	Sodium	Sodium	Argon	Argon	-
Spacing in (cm)	-	-	-	0 114 (0 289) D Wire	0 114 (0 289) D Wire	-	-	0 058 (0 147) D Wire
Fuel								
Material	Pu ²³⁹ U ²³⁸ O ₂	Pu ²³⁹ U ²³⁸ O ₂	Pu ²³⁹ Pu ²⁴⁰ U ²³⁸ O ₂	Pu ²³⁹ U ²³⁸ O ₂	U Metal	Pu O ₂ -UO ₂	Pu ²³⁹ U ²³⁸ O ₂	PuC - UC
Enrichment %	12 5	10 0	12 5	12 5	8 12	24 2 *	55 3 *	44 9**
Quantity, kg								
U ²³⁵	-	-	-	-	515 772	-	-	-
U ²³⁸	299 3	307 8	288 6	721	5920-5663	1069	63	96
Pu ²³⁹	42 7	34 2	42 7	103	-	341	78	78
Pu ²⁴⁰	-	-	10 7	-	-	-	-	-

*Dimension across flats in (cm) 2 290 (5 817)

Tube thickness in (cm) 0 040 (0 102)

Lattice pitch in (cm) 2 320 (5 893)

**U²³⁵ Enrichments can replace Pu²³⁹ in part

leakage still further to about 5 per cent. It should be noted that small clearances of these magnitudes are desirable at that location anyway, so as to add dimensional stability to the core.

At the reactor outlet, tubular connectors join each subassembly proper to its shielding extension. The region enclosing the connectors constitutes the outlet plenum of the reactor, that is, the height of the outlet plenum is equal to the length of the connectors. The connectors have been sized by using the criteria that (a) local velocities should nowhere exceed core velocities; and (b) subassembly stiffness should not be degraded. It was found that the tubular connectors accordingly must (a) be about 1.81 in. (4.60 cm) OD x 1.56 in. (3.97 cm) ID; (b) be perforated with about the equivalent of twelve holes of $\frac{1}{2}$ -in. (1.27-cm) diameter; and (c) be about 12 in (30 cm) long

It is expected that the final design of the fuel element will permit easy and simple disassembly techniques. Techniques will be used that will produce a minimum of solid waste material (such as machine chips) and complicated manipulations of the fuel and/or hex container.

The control rods consist of an absorber section, which slightly overlaps the core in its position of full effectiveness, and a shielding portion, whose function is the same as that of a fuel assembly. The control rods are driven by the drive mechanisms, which are located above the core, mounted to the top reactor vessel head. The rods are guided by hexagonal tubes which have the same overall dimensions as the fuel assemblies. The lower portion of the rods contain stainless steel reflector material, which replaces the absorber section when the rod is withdrawn from the core.

The reflector assemblies are used primarily for control of coolant flow between the upper and lower plenums. The reflector assembly is essentially a slotted hex in the core zone. Its sodium flow area is about 18 percent. It allows coolant to flow unrestricted from inlet to outlet. The upper portion also contains shielding material similar to that of the fuel assemblies.

Fig. 3. Removable Grid Section
(dimensions in in.)

It is expected that loading into the reactor, removal from the reactor, and disassembly for analysis or shipping of the reflector assembly and control rods will be accomplished without the use of any other devices except those required for normal fuel handling.

The core support grid (Fig. 3) serves as the structural base for supporting the reactor core. The grid maintains the required spacing of the subassemblies.

The grid is supported within the reactor pressure vessel by a cylindrical liner (core support cylinder) which is itself supported by means of a flange at the reactor vessel wall (see Fig. 4).

The core support grid is comprised of two coolant plenums. These are formed by three parallel plates spaced to form upper and lower plenum chambers for coolant. The primary coolant enters the reactor vessel at an elevation opposite the active core. It flows downward through the annular space between the vessel and the core support cylinder into the lower inlet

Fig. 4. Reactor Vessel

plenum. The lower plenum receives the primary coolant at full pressure and serves as the high-pressure plenum. The upper plenum (at low pressure) receives coolant from the lower plenum through orificed passages in the lower adaptors of the outermost reflector subassemblies. The orifices can be sized to provide a variety of pressures for the upper plenum. Coolant inlet holes in the various subassemblies open into the appropriate plenum chamber, and are sized to permit coolant flow to be consistent with the heat generation expected in each subassembly.

A prime advantage of the two plenums is that they provide a convenient method of lowering the coolant pressure at the inlets of the low-power-density subassemblies; thus, it is unnecessary to use extremely small coolant inlet holes in those subassemblies in order to achieve consistency between heat generation and coolant flow.

Another important feature of the double-inlet plenum design is that it insures that the subassemblies will remain in place (will have a net hold-down force which is strongly downward) even though the coolant enters the reactor at very high pressure. This situation results from seating the subassemblies at the top of the secondary, low-pressure plenum, so that the hydraulic forces of the high-pressure coolant on the subassembly lower adapters are identical in both the up and down directions and thus produce no net upward force.

The core support grid, core support cylinder, and a full complement of subassemblies (595 elements) have a total dead weight of about 150,000 lb (70,000 kg). The grid and cylinder will be fabricated from stainless steel. It is intended that both the core grid and support cylinder can be separately replaced. Both are large structures and will, after many experiments and repeated exposure to radiation, become radioactive. The degree of radioactivity will depend on the time of exposure and the power level of reactor operation. Replacement of the structures, early in reactor life, may be fairly straight-forward. However, after exposure to radiation from 50-Mwt operation for one year, the dose rate at the core support grid may be 50-100 r/hr. Therefore, to remove either unit, it might require difficult procedures and handling methods (e.g., thick-walled, large, lead-shielded coffins).

To minimize any undesirable reactivity effects, mechanical stability of the entire core structure will be necessary. For example, bowing effects could invalidate experimental results of detailed measurements. For the same reason, hydraulic stability of the coolant system will be necessary, although refinements in the system design will be made to reduce any anticipated limitations.

A discussion of the separate cores follows.

1. Zoned Loading*

A core arrangement has evolved which is primarily directed toward measuring the fuel-temperature coefficient of a typical large reactor.

The DBS experiment has the flexibility of investigating a variety of compositions typical of large, dilute reactors. The advantage of this configuration is that the characteristic spectra can be reasonably reproduced in the test region.

This core consists of a driver zone, buffer zone, and test zone. Its use is predicted by the following considerations:

- (1) The cost per experiment of a very large reactor is reduced because of smaller size of the test zone.
- (2) The fuel concentration for the test zone may be varied without changing the complete core assembly.
- (3) A neutron spectrum representative of large fast reactors may be achieved.
- (4) For a ceramic test zone with reactivity effects to be measured, a metallic driver core would assure safety of the experiment by virtue of its reliable and prompt negative reactivity coefficient due to fuel expansion.
- (5) The fuel temperature of the test zone only may be varied to obtain a static measurement of the corresponding change in reactivity.

The reference reactor whose reactivity effects are to be studied in the zoned experiment is a 2200-liter system, including blanket, having a composition of $\text{PuO}_2/\text{Na}/\text{SS}$ as 32/52/16 by volume percent and an enrichment of 12.5% Pu^{239} ($\text{U}^{238}/\text{Pu}^{239} = 7$). The reactor is surrounded by a 35-cm depleted uranium (0.4 percent U^{235}) blanket with a volume fraction composition of 60/22.4/17.6.

The experimental system consists of a test zone 6 in. (15 cm) in radius and a 6-in. (15-cm) buffer zone surrounded by a 12-in. (30-cm) driver section. The composition and enrichment of the test and buffer zones are those of the reference reactor (or of any other core to be investigated). The buffer zone serves to equilibrate the neutron spectrum, in both fine and gross structure, in the test zone to that of the corresponding zone in the reference core. The driver section is metal fueled, having a composition of $\text{U}/\text{Na}/\text{SS}$, as 32/52/16, with an enrichment of 10% U^{235} . The driver zone furnishes whatever additional reactivity is required to achieve criticality.

*Also referred to as Driver Buffer System (DBS).

2. Uniform Loading (medium core, approximately 400 liters)

This core has a v/o composition of $\text{PuUO}_2/\text{Na}/\text{SS}$ as 32/52/16, for the structure, and an L/D ratio of 0.25 ($\text{U}^{238}/\text{Pu}^{239} \simeq 3/1$). The composition is chosen as that of a typical large-sized (2000- to 3000-liter) fast reactor except that the large reactor would have an enrichment of approximately 10 to 12 percent. The core is surrounded by a stainless steel reflector, approximately 30 cm thick, having a composition of 18/82 for sodium/structure volume fraction, respectively.

3. Uniform Loading (small core, approximately 40 liters)

This experiment is primarily directed to measuring or investigating the effect of thermal expansion of the fuel. The experiment to be performed on the small core is the varying power level-constant coolant temperature type.

The core composition and fuel design are to be the same as those of the medium uniform core or in the test section of the nonuniform core. Solid fuel pins are being considered for the initial test loading. For the composition and the size of the core, approximately 40 liters with an $L/D = 1$, the Doppler effect is much less than $10^{-7} \Delta k/\Delta T$ (although probably still negative). The fuel expansion and the sodium coefficient of reactivity are negative and of the same order of magnitude, approximately $10^{-5} \Delta k/\Delta T$. (This fuel-expansion coefficient is based on the assumption that the fuel expands in a normal manner.) The required sensitivity to measure these effects can be achieved, with control of the sodium-temperature variation not being restrictive.

To be assured that a prompt negative coefficient exists in the system, the initial small core will be loaded in part with metal fuel pins. The metal fuel rods will be removed as confidence in understanding the expansional behavior of the fuel is established.

This experiment is considered most important since the separation of the Doppler effect from the fuel-temperature measurement is greatly dependent on the quantitative determination of the expansional effects. Further, this expansion effect or behavior can be studied at various burnup stages in an environment which is most typical of a fast reactor system and at appreciable power densities.

B. Core Parameters

The arrangement of experimental cores must be determined early in the program so that experimental work can be anticipated and schedules can be prepared. To this extent, an early estimate of the core sizes and arrangements of fuel assemblies has been made so that preliminary physics, heat

transfer, and other calculations can be made. From this work a set of basic parameters has been established. These are shown in Table II. Three types of cores are indicated, classified according to size as follows:

- (1) a zoned core for duplicating large-core spectra, and for statically measuring reactivity coefficients due to test sections;
- (2) a medium core for measurement of reactivity coefficients; and
- (3) a small core for engineering performance and core expansional tests.

Currently, these cores are considered fundamental to the program. The parameters are useful as reference for estimating manufacturing procedures, procurements costs, fabrication costs, and scheduling, and for the development of further analytical and test work.

C. Operating Data

Preliminary operating data pertinent to the various experiments has been developed and is summarized as follows:

Figure 5 illustrates the radial distribution of power density at the horizontal centerline for the Zoned Core, Medium Core, and Small Core. Figure 6 shows the axial distribution of power density at the vertical centerline for each of the three cores. The required coolant flow rate per subassembly vs subassembly row number is given in Fig. 7 for each of the three cores. These flow rates are based on the requirement that with perfect flow distribution of the coolant within a subassembly, a coolant temperature rise of 170°C (300°F) will occur at the hottest point in the hottest subassembly in each row.

D. Instrumentation

It is anticipated that the experimental program will require special in-core instruments, particularly for the measurement of temperatures of fuel pins and coolant, rates of coolant flow, pressure differences in the system or fuel assembly, and neutron and gamma fluxes. These instruments will be in addition to the instruments required for plant operation.

1. Fuel-pin Temperatures

The accurate measurement of fuel-pin temperatures as high as 2760°C , which is the melting point of UO_2 , is the goal. A small sensor is desirable for several reasons. First, only a small amount of fuel should be removed from a pin. This, in turn, will minimize the flux perturbation if highly absorbing materials are required for the sensor. Secondly,

Fig. 5. Radial Power Density Distribution at Horizontal Centerline
Total Reactor Power = 50 Mwt

Fig. 6. Axial Power Density Distribution at Vertical Centerline
Total Reactor Power = 50 Mwt

Fig. 7. Coolant Flow Rate per Subassembly vs Subassembly Row Number
Total Reactor Power = 50 Mwt

the small size and, hence, mass, of the sensor will contribute to a better time response, which would be important for transfer-function experiments.

Thermocouples rather than other devices will be used to measure the fuel-pin temperatures for the following reasons. Such devices take up little space and can be installed in restricted locations. In compatible atmospheres, the thermal emf is a direct measure of the temperature and is not affected by environmental conditions. Barring contamination of the thermocouple metals, the expected error of a temperature reading can be stated with reasonable assurance.

High-temperature thermocouples are limited for use in surroundings that will not affect their thermoelectric output. For most applications, a sheathed construction is needed. The sheath metal must be compatible with the fuel material and the insulator over the thermocouple wire. Otherwise, solid-phase reactions can proceed at extremely rapid rates, thereby destroying the sensor. The thermocouple wires must also be compatible with their electrical insulator for the same reason. The refractory oxide which is used as an electrical insulator must have a high electrical resistivity at the operating temperature. Otherwise, the insulator will act as a shunt across the thermocouple hot junction causing an erroneous reading. The effects of irradiation must be negligible during the thermocouple lifetime.

Based on the work by E. J. Brooks et al.,⁽⁴⁾ a preliminary design for the reference fuel-pin thermocouple has been selected that is expected to perform satisfactorily up to 2760°C. The sheath metal will be tungsten-26% rhenium. The insulator material will be thorium oxide. The thermocouple metals will consist of tungsten-5% rhenium vs tungsten-26% rhenium.

2. Coolant Temperatures

The coolant temperatures at the inlet and outlet of the fuel assemblies must also be measured. Thermocouples have also been selected for this measurement. The reference design for this thermocouple provides that the sheath material be of stainless steel, the insulator material of magnesium oxide, and the thermocouple metals will consist of chromel vs alumel.

3. Coolant Flow Rates

Turbine-type flowmeters have been used to measure liquid metal flow rate in out-of-pile loops operating at temperatures up to 800°C.⁽⁵⁾ Several problems have resulted from this high-temperature, liquid metal operation. The selection of suitable bearings appears to be the biggest problem. The selection of the magnetic material used for signal generation is also a problem. The time response of the turbine flowmeter

must also be improved. The turbine flowmeter installed as integral part of the fuel assembly has been selected as a reference design if measurements of coolant flow become necessary.

4. Pressure Sensors

Pressure and differential-pressure sensors are needed for in-vessel FARET measurements. Liquid-metal-filled pressure transducers are a relatively new development. This device consists of a pressure transmitter (with electrical output) connected by means of long capillary tubes to diaphragm seals in contact with the in-pile liquid metal. Pressures developed at the diaphragms are transmitted by means of the filling fluid (NaK) back to the transmitter. The transmitter must be kept below 75°C, although the diaphragm can operate at temperatures up to 650°C. The disadvantages of this system are the large diaphragm size, the poor time response, and the great difficulty in bringing the transmitter through a vessel penetration.

The advantages of a directly immersed transducer with an electrical signal output are obvious. Consideration will be given to the development of such a sensor. The liquid-metal-filled system has been selected as the reference design.

5. Neutron- and Gamma-flux Sensors

The development of fission counters and chambers capable of operating at temperatures up to 820°C is progressing on a commercial basis. Such sensors may be available within the next year.

The selection of wires or foils for neutron measurements and the insertion of these into the reactor core should be feasible. The problem of handling the wires or foils after irradiation and the data reduction should also be straightforward and present no problems.

6. In-vessel Handling of Instrument Signals and Leads

The fabrication, assembly, shipping, and installation of fully instrumented fuel assemblies are complicated by the multitude and length of the protruding instrument leads. The leads are normally made long enough to be continuous from the sensor to the pressure seal on the reactor vessel. A fuel element of this type is loaded into the core with a special fuel-handling tool. Once the fuel element is in the core, the task of removing the instrument leads from the reactor vessel is very difficult. Two methods have been proposed to remove instrument leads from the reactor vessel that eliminate the above-mentioned problem.

The first method consists of coiling each sensor lead on a spool which is smaller in diameter than the fuel subassembly. The spools are stacked on the top of the element during final fabrication. The design is

such that the spools do not interfere with the standard fuel-handling tool. This arrangement simplifies the shipping and installation of each fuel assembly. After the element is installed in the core, the spools are withdrawn remotely through instrument-lead penetrations in the vessel, thereby uncoiling and extracting the leads. The disadvantages of this method of extraction from the vessel are that the sheath integrity may be destroyed, the insulation may be damaged, or the sensor calibration, in the case of thermocouples, may be altered. Some of these disadvantages may be overcome if coils of larger diameter are used, but the handling of the assembly then becomes a problem.

The second method consists of terminating the sensor leads in an electrical connector at the top of the fuel element. The connector, however, must be capable of withstanding the high temperature (650°C) and nuclear radiation environment of the gas space above the core, and must be sealed to prevent short circuits from condensing sodium vapor. This addition to the instrumented fuel-assembly design has many advantages. The fabrication, assembly, shipping, and installation of such a fuel element are simplified because long instrument leads are not protruding. Permanent instrument leads can be installed through penetrations in the reactor vessel. A relatively simple mechanism can be designed which will remotely make the connection between the fuel element and the permanent leads. A single fuel element could be used more than once in the core and could, if necessary, be moved to different core positions.

Electrical connectors of the type needed are commercially available for operation up to 430°C. FARET requires a connector capable of operating at up to 650°C. Some development work will, therefore, be done to design and test a suitable connector. The development of a connector capable of operating at up to 650°C will be a significant improvement leading to simplified handling of the instrumented fuel assembly.

7. Data Collection and Reduction

Because it is assumed that a rather large number of in-core sensors will be needed for the FARET experimental programs, a large number of data-display devices will be required. To simplify the data collection and display, all in-core signals will be sampled, digitized, and recorded either on magnetic or paper tape or through the use of a data-logging system. For many of the experiments and, in particular, for the transfer-function measurements, it is desirable to include a small digital computer in the system that would partially process the incoming data.

E. Coolant and Blanket

1. Coolant

One of the requirements for minimizing problems of corrosion and mass transfer is that the coolant must be maintained at high purity.

Maintaining high purity will minimize the problems associated with long-lived circulating activity. These problems involve considerations of special shielding for coolant lines and might permit only limited access to those areas of the system where the activated impurities tend to deposit.

In view of the operating parameters stated above [1200°F (650°C), with provisions for single assemblies or small zones in the core to be at higher temperatures], the estimated maximum tolerable impurity level, in terms of oxygen content of the sodium, is expected to be in the range from 5 to 10 ppm.

To meet this requirement, reactor-grade sodium along with cold and hot trapping will be employed in FARET.⁽³⁾ In addition, provisions will be made in the coolant system for installing and testing advanced purification concepts under actual reactor operating conditions.

A plugging indicator is to be provided in the system, and it will be employed for coarse impurity measurements that are likely to be made at the start of a run after reloading the core. After sodium cleanup has progressed to the range below the practical limits of detection by the plugging meter (namely, 35-50 ppm) the vacuum cup sampling technique will be used.

2. Blanket

A low-pressure, high-purity blanket of argon gas will be maintained over the primary coolant system. The system will contain provisions for releasing excess gas to a retention tank where it can be held for decay of activity prior to releasing it to the stack.

In spite of the low oxygen content acclaimed for commercially available argon (0.5-1.0 ppm oxygen), vapor traps and NaK bubblers will be used for final cleanup of all gas entering the system.

Temperature conditions in FARET will be such that the quantity of sodium vapor held up in the blanket gas region will be higher than has been encountered in previous reactors. With an average coolant temperature of 1200°F (650°C), holdup of sodium vapor will be approximately 23 g/m³. This condition presents no unusual problems at the present time. It is necessary to take special precautions to insure that all instrumentation connections, located in the blanket gas zone, are adequately sealed.

III. EXPERIMENTS

A. Reactivity Effects of Fuel Temperatures

The objective of this phase of the program is the measurement of reactivity effects with respect to changes in temperatures of fuel, sodium, and clad structure due to nuclear heating. Of these effects, the fuel temperature coefficient is comprised of two components: the fuel expansional effect and the Doppler effect. Experiments, or a sequence of experiments, are being planned to obtain these coefficients. In addition, attempts will be made to separate the fuel temperature coefficient into its two components with a good degree of confidence, and to establish the principal cause of the prompt reactivity feedback.

Even if the objective is to isolate the reactivity effect of the Doppler coefficient, it will be necessary to correct the measured reactivity change by subtracting reactivity contributions due to axial expansion of fuel, and to temperature changes of coolant and structure. Table III indicates the magnitudes of the contributions to be subtracted for the zoned and medium cores for a fuel temperature change of 500°C, and indicates the maximum errors which should be encountered in the subtracted terms. It should be noted that the total errors listed are based on the assumption that all individual errors act in the same direction, something that is highly improbable. In other words, the standard deviation of a Doppler coefficient measurement obtained by applying these corrections to a total reactivity

Table III
EFFECTS OF ERRORS IN DOPPLER COEFFICIENT MEASUREMENTS*

Reactivity Contribution	Nominal $\Delta k/k \Delta T$ $\times 10^6$	Nominal $\Delta T, ^\circ\text{C}$	Nominal $\Delta k/k$ $\times 10^6$	Maximum Possible $\Delta k/k \Delta T$ $\times 10^6$	Maximum Possible $\Delta T, ^\circ\text{C}$	Maximum Possible $\Delta k/k$ $\times 10^6$	Maximum Possible Error in $\Delta k/k$ $\times 10^6$
ZONED CORE							
Doppler Coefficient, B and D	-6	6	-36	-8	8	-64	-28
Fuel Expansion, TZ	-0.25	500	-125	-0.4	530	-212	-87
Fuel Expansion, B and D	-2.5	6	-15	-4	8	-32	-17
Na and Structure, TZ, B and D	-4	0	0	-6	2	-12	-12
TOTAL			-176			-320	-144
MEDIUM (UNIFORM) CORE							
Fuel Expansion	-0.6	500	-300	-0.8	530	-424	-124
Na and Structure	-10	0	0	-13	2	-26	-26
TOTAL			-300			-450	-150

*500°C change in fuel temperature

measurement should be substantially less than indicated by the maximum errors given in Table III. For the zoned core, the expected Doppler coefficient $\Delta k/k$ for a 500°C fuel temperature in the test zone is about -300×10^{-6} , so the "maximum error" amounts to about 50% of the sought Doppler effect. For the medium core, the expected Doppler $\Delta k/k$ is about -850×10^{-6} , so the "maximum error" amounts to less than 20% of the sought Doppler effect. The general experimental procedure will be to measure temperature changes in conjunction with positions of control rods. Errors in control-rod calibration or position determination are negligible with respect to the other errors noted above. This preliminary and admittedly crude error analysis is encouraging in that it shows that the Doppler effect is large enough to be seen clearly even in spite of assumed generous errors.

The requirements for control of coolant flow are not very severe. Pumps can readily be controlled, over a range of at least 10 to 1, within 0.1% of rated flow. With the reactor operating at 10 Mwt and with the rated flow, the total rise of coolant temperature through the reactor is about 30°C. Calculations show that a 0.1% change in flow will change the average reactor temperatures by 0.015°C (0.03/2). This corresponds to a $\Delta k/k$ of about 13×10^{-6} (0.015) or 0.2×10^{-6} , which is small compared with the other effects involved. With the reactor at 2.5 Mwt and one-fourth of rated flow, to give the same temperature rise of the coolant the effect of a flow change amounting to 0.1% of rated flow will cause 4 times as large a reactivity change, but the effect is still relatively small.

Several alternative methods are being pursued concurrently for achieving the desired temperature changes in the fuel. Although these methods are somewhat detailed in nature, they are basic to the design of these experiments, and for that reason they have been given considerable effort. The methods may be classified according to the following random order of design schemes:

- (1) variable gas gap;
- (2) variable power with gas gap;
- (3) variable power without gap;
- (4) transfer functions.

1. Variable Gas Gap

One way to obtain large changes of fuel temperature, without at the same time changing cladding or coolant temperatures excessively, is to use fuel elements which employ an insulating gap between the fuel and cladding. Tubing connects each such element in the reactor to a station outside the reactor, thus making it possible to vary the pressure and/or composition of the gas in the gap. Such variations change the thermal conductance

of the gap and, as a result, change fuel temperatures even though the reactor power, flow rate, and coolant and cladding temperatures remain unchanged. The first two cases of Table IV indicate how the temperatures of the fuel surface are calculated to change, for the stated conditions, if the gap gas is changed from helium at atmospheric pressure to argon at atmospheric pressure, or if helium at atmospheric pressure is evacuated to a high vacuum. Since the values given are for a total reactor power of 10 Mwt and since FARET is a 50-Mwt facility, the large temperature changes shown are still not the maximum values obtainable with the facility.

Table IV
CALCULATED CONDITIONS FOR ZONED-CORE LOADING

Average Coolant Temperature (°C)	Initial Reactor Power (Mwt)	Final Reactor Power (Mwt)	Initial Gas in 0.020-in. Gap	Final Gas in 0.020-in. Gap	Maximum Increase in Fuel Surface Temperature (°C)	Maximum Increase in Cladding Temperature (°C)
593	10	10 (no change)	Helium at 760 mm Hg	Argon at 760 mm Hg	327	0
593	10	10 (no change)	Helium at 760 mm Hg	Helium at $<10^{-3}$ mm Hg	505	0
593	2.5	10	Argon at $<10^{-3}$ mm Hg	Argon at $<10^{-3}$ mm Hg (no change)	397	10
427	2.5	10	Argon at $<10^{-3}$ mm Hg	Argon at $<10^{-3}$ mm Hg (no change)	444	10
593	2.5	10	Argon at 760 mm Hg	Argon at 760 mm Hg (no change)	336	10

Notes: (1) 0.020-in. (0.051-cm) gap between fuel and clad

(2) Fuel surface emissivity = 0.7

(3) Clad surface emissivity = 0.5

(4) Coolant temperatures shown are used only for the purpose of calculations.
Higher fuel temperature changes will result with lower coolant temperatures.

The problems connected with the variable gas-gap experiment are these: (a) Thermal conductances of the gap cannot now be calculated precisely, due to uncertainty in such things as surface emissivities and the effects of fuel support or spacing devices. Therefore, it is not certain that the temperature changes given in the table are valid; in addition, the problem of translating a finite number of measured fuel temperatures into an average fuel temperature for the entire sensitive zone in the reactor is made more severe. (b) The fuel integrity and desired gap width must be maintained, that is, if the fuel should crumble and/or touch the cladding, the problem of determining the average fuel temperature becomes ambiguous. (c) Changing the gap gas, or evacuating the gap, by using tubing of small diameter may be difficult. (The gas-changing method appeals since it is thought to be easier to change the gas than to maintain a vacuum through small lines to the fuel elements. However, both the gas-change and the vacuum conditions are still receiving equal emphasis.)

To resolve these problems, a related experimental program has been begun. The first experiment conducted consisted of heating a test element electrically. The components used are illustrated in Fig. 8. The fuel pellet was made of lava, a material with a thermal conductivity similar to that of uranium dioxide. Small projections on the pellet maintained a 0.015-in. (0.038-cm) gap between pellet and cladding tube. Typical temperatures obtained with a high vacuum in the gap are given in Table V. These temperatures are close to what would be predicted by using conventional methods of calculation. The linear heat flux for the test was 6.9 w/cm, which corresponds to a reactor power of about 2 Mwt for a 1670-liter zoned core. Other measurements were made with helium and argon, respectively, in the gap, and again the results agreed well with calculated values. Thus, the values of Table IV have received strong support from the initial experiment.

Fig. 8. Test Element No. 1

Table V

TEMPERATURES OBTAINED IN MODEL FUEL ELEMENT NO. 1

(0.5325 in. in diameter x 7 1/2 in. long)

	°F	°C		°F	°C
Heating wire	2620	1440	Metal sheath	500	260
Center of pellet	1575	860	Gradient across		
Surface of pellet	1315	715	0.015-in. gap	815	435

Currently, efforts are being made to conduct a similar, but more elaborate, experiment on a second test fuel element which has the fuel divided into relatively small pieces, each supported independently by the cladding. This axial fuel segmentation is carried out because there is additional interest in eliminating the axial fuel-expansion effect so that only the effect of the Doppler fuel coefficient is isolated as much as possible. There is a problem of anchoring each high-temperature pellet to the clad, which may be eliminated for modest fuel temperatures $\lesssim 1000^{\circ}\text{C}$. The second test element is illustrated in Fig. 9.

Fig. 9. Test Element No. 2

Recently, pellets of lava and a tubular platinum heater, both procured for use in the second test fuel element, were inserted in a quartz tube (simulating the fuel-element cladding) to permit visual observation of the internal behavior of the model fuel element. Maximum heater temperature used in this test was about 1460°C ; heater temperatures produced pellet surface temperatures of about 900°C (as indicated by an optical pyrometer), a linear heat flux of about 21.4 w/cm , and a temperature drop through the pellets of about 400°C . Replacement of air in the quartz tube with helium, by using a $\frac{1}{8}$ -in. (0.32-cm) OD tube, 20 ft (6.1 m) long, to simulate reactor conditions, caused the heater temperature to drop from about 1260°C to about 970°C , with the linear heat flux being held constant at about 13.3 w/cm ; although the conditions in quartz tube were obviously different from those in the actual fuel element, this gas-replacement experiment did serve to illustrate how fuel temperatures can be changed in the FARET without changing reactor power. Figure 10 shows the pellets inside the quartz tube, and Fig. 11 shows the heater while being calibrated

Fig. 10. Electrically Heated Test
Pellets in Quartz Tube

Fig. 11. Heater Calibration

for resistance vs temperature with an optical pyrometer. High-temperature, refractory metal heaters will be calibrated similarly, that is, inside the quartz tube where they can be protected by an inert atmosphere; the effect of the quartz tube on pyrometer readings is being determined by recalibration of the platinum heater inside the quartz tube. Figure 12 shows the second test element and its components. External cooling fins, not present in

Fig. 12. Components Test Element No. 2. Gas Gap Bench Test - Exploded View

the prototype element, make it possible to keep the cladding temperature below 550°C even with linear heat fluxes of as much as 45 w/cm . The external heater can be used to raise the clad temperature to 550°C even when test element heat fluxes are quite low.

When attempts were made to assemble the second test element, the ceramic spacing washers fell out of position during element handling. Therefore, redesigned washers, which will permit experimentation with the second test element shortly, are being procured. In addition, a third test element, which uses dimples in the clad instead of projections on the fuel to maintain the insulating gap, and holds the spacing washers with dimples, rather than with grooves, in the clad, is being procured. An element of the third type should be much cheaper than an element of the second type. When experimentation with the second and third test elements is completed, as scheduled for the fourth quarter of 1963, the problems listed above should be under control. The final prooftesting of

selected schemes will be done in an in-pile experiment (probably in CP-5) to insure adequate performance in the reactor.

2. Variable Power with Gas Gap

A second way to obtain large changes of fuel temperature without at the same time changing cladding or coolant temperatures excessively,

is to use fuel elements with an insulating gap, but not to vary the conductance of the gas (argon) in the gap. Instead, the reactor power is changed while concurrently adjusting coolant flow rate to maintain coolant and clad temperatures substantially constant. The principal advantage of this scheme is that larger changes in fuel temperatures will be attainable than in the previous method. Because of the gas gap, the change in clad temperature gradient per power rise is held to a minimum, so that reactivity changes due to clad changes are minimized. Since it will not be necessary to change the composition or pressure of the gas in the insulating gap, the required development work is reduced, and it would be possible to do this experiment at an earlier date.

The last three cases of Table IV indicate how the fuel surface temperatures would change, for the stated conditions, if the reactor power is increased from 2.5 to 10 Mwt. Problems connected with the variable-power (with gap) experiments include Problems (a) and (b) listed above for the variable gas-gap experiment. Because of the changing of power level, the additional considerations that must be taken into account include: (a) Fuel temperatures outside the sensitive zone of the reactor, and coolant and cladding temperatures everywhere, must remain substantially constant during the experiment, to keep reactivity contributions from these sources relatively small. Therefore, the fuel elements involved must be designed so that this requirement is met, primarily by keeping the thermal conductances of the element high. For this reason, in the zoned core, metallic fuel elements have been selected for the driver zone and sodium bonding for the buffer zone. The element sizes in those zones were also dictated by this consideration. (b) The coolant flow rate must be controllable accurately at each extreme of the range required by the magnitude of the power change. The relevant values given in Table IV are based on power (and flow) changing by a factor of four, which is considered a conservative approach. Actually, engineering work on the coolant-circulating system is directed toward accurate control of flow rate for flow-rate change ratios of the order of ten. Then it should be possible to obtain greater changes of fuel temperature than indicated in Table IV (see Table VI). The desired fuel temperature will be obtained by operating the reactor at two power levels and adjusting the coolant flow rate for each power level to maintain isothermal conditions of coolant and clad as nearly as possible. The reactivity loss due to the temperature change is then compensated by a calibrated control rod.

With regard to obtaining average fuel temperature changes of around 1000°C , Table VI indicates changes which can be obtained by varying the reactor power and coolant flow rate by a factor of 10 to 1, a procedure which currently appears feasible by means of conventional equipment. Changing the power from 5 Mwt to 50 Mwt in the zoned core (1.25 to 12.5 Mwt in the medium core), while holding the average coolant temperature constant at 200°C by a corresponding flow rate change, will produce an increase in average fuel temperature of about 950°C .

Table VI

EFFECTS OF POWER AND FLOW RATE CHANGES BY FACTOR OF 10 to 1

Zoned Core		Medium Core		Increase in Average Fuel Temp (°C)	Maximum Fuel Temp at Final Power (°C)	Maximum Fuel Temp Drop at Final Power (°C)
Initial Power (Mwt)	Final Power (Mwt)	Initial Power (Mwt)	Final Power (Mwt)			
1	10	0.25	2.5	460	1120	100
2.5	25	0.62	6.2	700	1740	260
5	50	1.25	12.5	950	2440	510

Notes: (1) Fuel-element dimensions are given in Table II.

(2) Fuel surface emissivity = 0.7

(3) Clad surface emissivity = 0.5

(4) Average coolant temperature = 200°C

(5) Fuel-element thermal conductivity = 0.0346 w/(cm)(°C). (A value of 0.0208 w/(cm)(°C) was also used, with the most significant change found being that fuel temperature drops were higher.)

The values given in Table VI are based on maintaining the temperatures of the reactor coolant at inlet and outlet constant, as well as the average coolant temperature. Another possibility is to hold only the average coolant temperature constant. Then, an initial condition consists of having the reactor at very low power, substantially isothermal at the reference average coolant temperature of 200°C. Increasing the reactor power then to 10 Mwt (Zoned Core), while still maintaining an average coolant temperature of 200°C by coolant flow and/or heat sink control, now produces an increase in average fuel temperature of about 560°C. Increasing the power, likewise, to 25 Mwt or 50 Mwt produces an increase in average fuel temperature of about 920°C or 1290°C, respectively. In other words, this latter scheme is capable of producing a given average change of fuel temperature with a lower radial fuel temperature drop than the method in which coolant inlet, outlet, and average temperature are all held constant.

3. Variable Power without Gap

This scheme is a further simplification of the previous two in that development work on fuel-element design is eliminated entirely. The basis for this scheme is in making use of the low thermal conductivity of the fuel to obtain a reasonable temperature change for different power levels. Large-diameter fuel elements may be used to reduce the thermal conductance and enhance the temperature rise per power level change while avoiding the use of an insulating gap. An advantage of this experiment is that the fuel integrity is not as important as it is in the experiments using an insulated gap. In addition, fewer individual fuel elements

are needed. The principal disadvantage of this scheme is that the individual reactivity effects will be most difficult to isolate. Problems associated with this experiment are the same as Problems (a) and (b) listed for the variable-power-with-gap experiment and are handled in the same way.

4. Physics Analysis

a. Zoned Loading

To measure the reactivity effects in large dilute systems having U^{238}/Pu^{239} atomic ratios of 7/1 and 9/1, a series of multiregion core configurations were studied. The above enrichments in reactor systems having compositions of Pu- UO_2 /Na/SS of 32/52/16 by volume percent were taken as being typical of large systems of approximately 2000 to 3000 liters. The Doppler components of the fuel temperature coefficient for these systems were computed to be -7.6×10^{-6} and $-1.1 \times 10^{-5} \Delta k/\Delta T$ at 750°K for the 7/1 and 9/1 atom ratios respectively. These are considered to be of significant magnitude to generate a measurable reactivity effect in the DBS system.

Multiregion core configurations are being studied in an attempt to select a system which is most sensitive to a rise of fuel temperature in the test zone. The central and buffer zones are to contain the composition and enrichment of the reactor system to be investigated. The driver section is metal fueled and serves, among other reasons, to furnish whatever additional reactivity is required. The buffer zone serves to equilibrate the neutron spectrum (both fine and gross structure) in the test zone to that of the corresponding zone in the reference reactor.

In studying the various experimental configurations in the DBS system, a criterion used was that a reactivity change arising from a change of fuel temperature in the test zone be within some acceptable error of the reactivity change in the corresponding zone of the reference reactor subjected to the same perturbation. Reactivity changes were compared for other arbitrary perturbations having different energy dependence as well as different magnitudes. This would assure one that, not only is the neutron spectrum reasonably reproduced, but also the importance spectrum. The reproducibility of the reactivity effect appears to be sensitive to the type of perturbation applied.

In the case of the Doppler effect, the spectrum and importance function were maintained to within 2 to 10 percent for the energy groups below 10 kev. In particular, the groups between 10 and 1 kev were kept to within 2 and 3 percent since, it is estimated that 60 percent of the Doppler effect for the reference reactor results from this energy range.

In Table VII are listed some results of the averaged Doppler coefficient for a temperature change of 750°K at 750°K for oxide and carbide systems.

Table VII

DOPPLER COEFFICIENTS FOR REFERENCE AND TEST CORES

Composition (Volume Fraction) Fuel/Coolant/Structure	Doppler Coefficient $\Delta k/\Delta T \times 10^6$			
	Reactor Core	Core Test Zone Radius - 6 in. (15 cm)	Test Zone/Buffer/Driver 6/9/15 in. 15/23/38 cm	Test Zone/Buffer/Driver 6/6/12 in. 15/23/38 cm
PuUO ₂ /Na/SS 32/52/16			Metal Driver (U ²³⁵ enrichment)	Metal Driver (U ²³⁵ enrichment)
U ²³⁸ /Pu ²³⁹ = 7/1 (~2200 liters)	-7.6	-0.72	-0.63 (9.25% U ²³⁵)	-0.60 (10.16% U ²³⁵)
U ²³⁸ /Pu ²³⁹ = 9/1 (~3000 liters)	-11	-0.45	-0.48 (10.46% U ²³⁵)	-0.47 (11.02% U ²³⁵)
PuUC/Na/SS 30/52/18				
U ²³⁸ /Pu ²³⁹ = 7/1 (~2000 liters)	-4.75	-0.52	-0.61 (7.78% U ²³⁵)	-0.59 (9.2% U ²³⁵)
U ²³⁸ /Pu ²³⁹ = 9/1 (~2700 liters)	-7.64	~0.40	-0.46 (9% U ²³⁵)	-0.46 (~10% U ²³⁵)

The results indicate that either of the two DBS systems considered would give approximately the same reactivity change. The greatest reactivity change, however, is expected from the system having a 7/1, U²³⁸/Pu²³⁹ atom ratio. This is so since the full-scale 7/1 reactor is of smaller volume and a 6-in. (15-cm) test zone represents a larger sampling. Another way of looking at this is to observe that the 7/1 test zone contributes more to the reactivity of the DBS. The actual values generated are subject to change, since these depend strongly on the cross-section set used. However, the general trends will hold. Most of the computations were performed with an 18-group cross-section set (Elmoe corrected). This set is being continually revised, and the final choice of these numbers will be made after a check of the cross-section set on the actual system in a critical assembly.

Of the two DBS systems, it appears that the experiment having a 6-in. buffer zone and a 12-in. driver section is preferable, since this involves a lower fuel loading. Studies are being continued on the zoned systems in which depleted uranium or some other good scattering material for the buffer zone is utilized. Higher enrichments and smaller driver sections were considered, but the computations indicated that the gross structure of the spectrum in the energy region of interest (that below 10 kev) was not satisfactorily reproduced. The low enrichments in a larger driver section are preferred so as to minimize the cooling requirements for the varying power-level type of experiment. If the power density were too high, it would impose an upper temperature limit to this type of experiment.

Another advantage of the metallic fuel is that it introduces a definite, negative, prompt temperature coefficient into the system. This is desirable since the dimensional behavior of the ceramic fuel is uncertain

The DBS experiment involves heating the fuel in the test zone while simultaneously keeping the buffer and driver section at constant temperature. Then the reactivity effect observed, while maintaining the sodium and clad temperatures constant, is the fuel-temperature coefficient, which is comprised of the Doppler coefficient and fuel-expansion effect. If the oxide fuel were to expand in a normal manner, the fuel-expansion coefficient of the oxide calculates out to be approximately $-0.25 \times 10^{-6} \Delta k / \Delta T$ for the DBS experiment with the 6-in. (15-cm) buffer region and a U^{238}/Pu^{239} atom ratio of 7/1. The total fuel temperature coefficient is then expected to be of the order of $-0.9 \times 10^{-6} \Delta k / \Delta T$. A $1000^{\circ}C$ temperature rise should result in a reactivity change of -0.9×10^{-3} or -0.09% k.

The computed Doppler effect and, in turn, the fuel temperature coefficient quoted above should be considered as an upper limit. This arises from the neglect of interference or overlapping effects in the resonance integral formulation. Assuming an error of 50%, the reactivity change is measurable.

The sodium temperature coefficients for the reference reactor systems have been computed to be $-1.0 \times 10^{-6} \Delta k / \Delta T$ and $+3.5 \times 10^{-6} \Delta k / \Delta T$ for the U^{238}/Pu^{239} ratios of 7/1 and 9/1, respectively. For the 7/1 ratio system the sodium coefficient in the center zone could be positive, and this may also be the case for the DBS test zone. This problem has not as yet been investigated. In either case, based on a volume and statistical weight ratio of 0.5, which is pessimistic, the sodium coefficient for the 7/1 atom ratio test may be of the order of $\pm 0.5 \times 10^{-6} \Delta k / \Delta T$. Allowing a 5% contribution from this effect to the fuel temperature reactivity change requires that the coolant temperature changes be maintained within 10% of the temperature change in the fuel.

The coefficients to be expected in the DBS experiment are currently the subject of a study program which will involve critical experiments in the ZPR-VI facility.

To obtain the fuel temperature rises in the test zone, all the techniques described above may be used singularly or in some combination. The variable-gas-gap technique could be applied to the test zone only, in which case fuel elements of the same size may be used in the buffer region. At constant power, reactivity effects due to coolant and clad in the buffer and driver sections should be minimal. The temperature rise attainable in this case may be at least $500^{\circ}C$. This temperature rise may still be adequate depending on the magnitude of the actual fuel temperature coefficient.

The alternative technique of varying power level and coolant flow for a gas-bonded fuel element would result in much higher temperature rises if the power level were varied in a ratio of 10/1 instead of 4/1 as

shown in Table VI. For thermal conduction the average fuel temperature rise is proportional to $q_H[1-(q_L/q_H)]$, where q_H and q_L are the heats generated in the rod at the high and low power level, respectively. If radiation losses are neglected, a two-fold increase in the temperature change could be realized. This technique, however, requires rods of smaller size in the buffer zone in order to isolate the fuel temperature rise in the test zone. The temperature rise of the driver section is kept to a minimum, first by the low enrichment resulting in low power density and second by good thermal conductance of the metallic fuel. The errors introduced into the reactivity measurements caused by possible temperature changes in the buffer and driver sections are currently being investigated.

b. Medium-sized Core

An experiment which will give a greater reactivity change for a given change of fuel temperature is that of a uniformly loaded system. The core is a 400-liter oxide core having a composition of 32/52/16, fuel/sodium/SS volume fraction, respectively, an L/D ratio of 0.25, and a plutonium enrichment of 24.2 percent ($U^{238}/Pu^{239} = 3/1$). The composition again was chosen as that of a typical large-sized (2000- to 3000-liter) fast reactor, which has, however, an enrichment of approximately 10 to 12 percent.

The Doppler coefficients computed for a variety of fuel temperature changes are presented in Table VIII.

Table VIII

AVERAGE DOPPLER COEFFICIENT AS A FUNCTION OF TEMPERATURE

Temperature (°K)	k_{eff}	$-\Delta k_{eff}/\Delta T \times 10^6$ (T - 300°K)	$-\Delta k_{eff}/\Delta T \times 10^6$ (T - 750°K)	$-\Delta k_{eff}/\Delta T \times 10^6$ (T - 1500°K)
300	1.000001			
750	0.998331	3.71		
1500	0.997075	2.44	1.67	
2500	0.995815	1.90	1.44	1.26

An immediate advantage of this medium-sized, uniform-core experiment is that the magnitude of the Doppler coefficient and the sensitivity of the system are greater by a factor of about two to three than that obtained in the DBS experiment with a U^{238}/Pu^{239} ratio of 7/1 ($\Delta k/\Delta T = -0.63 \times 10^{-6}$ at 750°K). The fuel loading is only 10% greater than in the DBS system.

The sodium coefficient and fuel-expansion coefficient computed for L/D ratios of 1, 0.5, 0.25 are presented in Table IX.

Table IX

SODIUM AND FUEL-EXPANSION COEFFICIENTS FOR VARIOUS
L/D RATIOS AT A CONSTANT VOLUME OF 400 LITERS

L/D	Sodium Coefficient, $\Delta k/\Delta T \times 10^6$	Fuel-expansion Coefficient, $\Delta k/\Delta T \times 10^6$
1.0	-3.4	-2.2
0.5	-5.0	-1.2
0.25	-9.1	-0.6

The fuel-expansion coefficient is sensitive to the L/D ratio. The expansional effects can be reduced by factors of 2 and 3 in going from an L/D ratio of 1.0 to 0.25. A neutron-inventory study for a variety of oxide systems shows that the reduction results in part from the decrease in the radial leakage component of the expansion effect. In Fig. 13 is plotted the fuel-expansion reactivity as a function of radial leakage. Spectral effects also contribute, but more detailed study is required and is being planned in order to understand their contribution. Coupled with the reduction in the expansional effect is the increase in the sodium coefficient as the L/D ratio is decreased. This is mostly due to the increase in the surface-to-volume ratio. For these small systems, the sodium coefficient is predominantly due to leakage effects, with the spectral effects being of secondary importance. This is another area of study which is currently being pursued.

Composition Volume Fractions Pu-UO ₂ /Na/Nb	Pu Enrichment (%)	Volume (liters)	L/D
1. 40/50/10	23.2	234	1.26
2. 40/50/10	13.6	1300	1.1
3. 40/50/10	13.6	1520	0.47
4. 50/37.5/12.5	17.9	263	1.2
5. 50/37.5/12.5	13.6	673	1.12
6. 66/18/16	13.6	304	0.88
7. 66/18/16	13.6	310	1.09

Fig. 13. Fuel Expansion Reactivity Coefficient
as a Function of Radial Leakage

In an attempt to extract the Doppler effect from the fuel temperature coefficient, it is desirable to minimize the expansional effect. This means that measurements of fuel temperature coefficient should be made with systems having low L/D ratios.

For the experimental system suggested, namely, that with $L/D = 0.25$, the expansional effect is some 50 percent of the fuel coefficient (if a Doppler coefficient of -1.67×10^{-6} is used). The fact that the sodium coefficient should be greater for this system is not troublesome since the technique of constant coolant temperature will control this component. An acceptable error of 10% in the total reactivity change arising from the sodium effect requires that the coolant temperature change be kept to 2 percent or less of the temperature rise in the fuel. Since temperature rises in the fuel will range between 300 and 1000°C, or more, this imposes a control capability of 6 to 20°C on sodium inlet and outlet temperatures. This is not difficult to achieve and represents a design specification.

Another experiment that may be attempted with this 400-liter system is directed toward studying scaling laws, that is, the relation between perturbations in a sampling of the reactor core to the same perturbations for the whole core. This experiment is essentially a DBS type of experiment without a driver region. The gas-bond, varying power level-constant coolant temperature technique may be applied. This may be done by overcooling the system except for a 2-ft-diameter section of the core. Use of a volume and statistical weight factor of 0.6 gives a fuel temperature effect of approximately $-1.4 \times 10^{-6} \Delta k / \Delta T$. This is based on the assumption that the fuel-expansion effect scales as the Doppler effect, which is probably not a bad approximation. The size of the zone was selected for the purpose of minimizing the "hot" and "cold" boundary effect on the Doppler component of the fuel temperature coefficient.

Analysis of the medium-core experiment is continuing with emphasis on the study of the sodium effect in the central zone as well as the structural and coolant effects outside the central zone.

Dynamic tests involving controlled moderate transients will be studied for the 400-liter, uniformly loaded core. The details of the study, which has not yet been done, will establish the feasibility of using such measurements to separate delayed and prompt effects.

c. Small Core

An experiment being planned involves the direct measurement of the thermal expansion effect and/or investigating the dimensional behavior for fuel-element designs such as a tightly clad element which would insure an axial expansion.

For the oxide fuel, the core composition is the same as that in the medium uniform core. Solid fuel pins are being considered. For the composition and size of core, approximately 40 liters, with an $L/D = 1$, the Doppler effect is negligible (less than $10^{-7} \Delta k/\Delta T$). The total fuel-expansion and the sodium coefficients of reactivity are of the same order of magnitude, which is approximately $10^{-5} \Delta k/\Delta T$. The fuel-expansion coefficient is based on the assumption that the fuel element expands axially without restraint. This large reactivity change makes for a sensitive experiment. The control of the sodium-temperature variation is restricted only by the acceptable error on the fuel-expansion effect.

Calculations made for a carbide-fueled system having a composition of 30/52/18 of fuel/coolant/structure volume fractions, respectively, for a 50-liter and a 25-liter volume, are presented in Table X.

Table X

SODIUM AND FUEL-EXPANSION COEFFICIENTS
FOR SMALL CORES

Core Volume (liters)	Sodium Coefficient, $\Delta k/\Delta T \times 10^6$	Fuel-expansion Coefficient, $\Delta k/\Delta T \times 10^6$
50	-5.4	-4.9
25	-7.3	-5.1

To be assured that a prompt negative coefficient exists in the system, the initial small core will be loaded in part with metal fuel pins. The metal fuel rods will be replaced as confidence in understanding the expansional behavior of the fuel is established.

This will probably be one of the more important experiments in the series, since the expansional effect from the fuel temperature measurement or the fuel-design behavior will be studied at various burnup stages.

The changes of fuel temperature may be varied by any of the techniques previously described. Studies involving transient or transfer-function types of experiment will be considered for the small core as well as for the medium-sized core.

d. Structural Materials

An investigation of the effect of structural materials on the sodium and fuel-expansion coefficients has been underway. The structural materials considered were niobium, vanadium, and stainless steel.

Niobium and vanadium were used in this study since they are being considered as possible cladding materials for the fuel element for high temperatures. Niobium and vanadium have similar good high-strength properties and similar energy dependency of the capture cross section. The magnitude of the vanadium capture cross section for a typical hard spectrum in a carbide, large, fast reactor is of the same order as that of stainless steel. This makes it attractive with respect to neutron economy. Preliminary results of the calculations, involving a 30-cm layer of depleted uranium and the structural material in the core, for a variety of compositions are given in Table XI.

Table XI
THE EFFECT OF STRUCTURAL MATERIALS ON SODIUM AND FUEL-EXPANSION COEFFICIENTS

Core Sizes (liters)	Composition Volume Fraction PuUC/Na/Structure	L/D Ratios	Sodium Coefficient (Equivalent Sphere Geometry Problem) ($\Delta k/\Delta T \times 10^6$)			Fuel-expansion Coefficient Cylindrical Geometry ($\Delta k/\Delta T \times 10^6$)		
			Nb	V	SS	Nb	V	SS
3000	25/59/16	1	+8.9	+7.4	+4.4	-4.5	-3.5	-3.4
		1/4	+5.3	+3.8	+0.4	-2.2	-0.8	-0.9
400	30/52/18	1	-1	-2.7	-3.8	-4.8	-3.8	-4.3
		1/4	-2.7	-4.7	-5.6	-1.3	-0.3	-0.6
50	25/59/16	1	-6.7	-8.4	-7.6	-4.7	-4.2	-4.5
		1/4	-6.7	-8.4	-7.6	-1.1	-0.4	-0.9

The calculations were made for spherical geometry and the enrichments computed for cylindrical geometries with L/D ratios of 1 and 0.25. This is adequate for a comparison type of study. The computations indicate that, for large cores, the sodium temperature coefficient is more negative for the stainless steel structure than for either of the two refractory metals. The fuel-expansion coefficient, on the other hand, is slightly lower if not insensitive to the structural materials.

What is interesting is that the preliminary summary indicates that the spectral effects of vanadium in a large system are similar to those of niobium, even though the magnitude of the vanadium cross section is closer to that of stainless steel. A more detailed analysis is underway in order to establish the results more firmly and to investigate the detailed competition between capture, fission, spectral, and leakage effects of these structural materials on the sodium void coefficient.

To investigate experimentally the effects of materials on the reactivity coefficients, a medium-sized core would be adequate to emphasize the spectral components. A similar study of a zoned system is in the planning stage.

5. Transfer Functions

For both of the uniform-core types (small and medium) transfer functions will be determined. The oscillating-absorber method is reasonably standard, and suitable equipment will be designed and furnished.

The transfer-function determination is made through observation of the amplitude and phase response of reactor power (and any other reactor parameters) to a sinusoidal disturbance of reactivity. Crudely, reactivity feedbacks due to changing power level forced by the original disturbance can be divided into three parts:

- a. If the delay time of the effect is long compared with the period of the disturbance, the effect is observed as its time average, which is zero.
- b. If the delay time of the effect is short compared with the period of the disturbance, it will be seen as a frequency-independent prompt effect.
- c. If the period of the disturbance is of the same order as the time lag of the feedback effect, then as frequency is increased, the response goes from "pseudo-prompt" (type b) to essentially zero (type a).

Thus, observation of the frequency response can, in principle, show what feedback times exist in the system. These may then be identified with suitable known effects. Beat frequencies, corresponding to differences in time lags, will also be observed.

At low power, only effects due to delayed neutrons are observed, since the power variations are not large enough to produce significant thermal feedbacks. As power increases, the thermal feedbacks become observable. The ability to vary flow adds still another dimension to the flexibility of the technique.

Unfortunately, in this case, fuel temperature effects will all be virtually prompt, so that oscillator response will not be able to separate, for example, Doppler and expansion coefficients. The major problem that exists, however, is to find a means of getting sufficient forcing amplitude at high enough frequencies (in the sonic range) to be able to see other effects due to heat conduction and convection. Additionally, the nature of the fuel is such that the characteristic time lag for heat transfer will probably be a statistical function depending on the surface contacts of the individual fuel elements. Also, the heat-storage capabilities of oxide fuels, particularly those with insulating gaps, are such that the temperature response to power fluctuations is appreciably damped.

Nevertheless, the transfer-function determination is still a useful piece of information, providing an empirical test of whether specific power-flow-heat transfer configurations are well below, or near, an instability point.

The problems mentioned above are included in the research tasks of the program, to obtain measures of feasibility of measurement and of interpretive schemes. Finally, the transfer function of zoned cores will be examined analytically for synthetic models, to determine whether useful information can be extracted from such configurations. If this turns out to be the case, the mechanics of equipping the zoned core are the same as for the uniform cores, and the measurement will be performed.

6. Bowing and Stiffening

The effect of bowing of fuel elements on changes in effective core radius and resulting change in reactivity is essentially caused by non-uniform radial distribution of heat generation within the fuel elements and assemblies.

The rod-bowing effect on reactivity is also a function of the height of the core. For a low ratio of L/D , this effect is minimized.

Details for obtaining characteristic core measurements have not been worked out; rather, these will be an important consideration of the design of the experiments. In the case of the measurement of the fuel temperature effect, the bowing effect could be eliminated by clamping the core as rigidly as possible.

B. Physics Irradiation

Irradiation studies can be made in the center of the zoned core in which the neutron spectrum is close to that of a reactor system of interest. For the oxide loading with a plutonium enrichment of 12.5%, the power density at 50 Mwt total power should be approximately 0.037 Mw/liter of core. The macroscopic fission cross section is 0.0024 cm^{-1} . This yields a flux of $4.8 \times 10^{14} \text{ n}/(\text{cm}^2)(\text{sec})$, which, for a reaction cross section of one barn, gives a reaction rate of 0.12% per month. A one- to two-month irradiation should be sufficient to determine average capture cross sections for isotopes having cross sections in the range from 0.5 to 1.0 b.

The significance of the measurements of α and the average cross sections will depend on the reproducibility of the entire neutron-energy spectrum. In the zoned systems studied, the spectrum was reproduced acceptably in the energy region below 10 kev. However, there were discrepancies in the higher-energy range. This implies that a zoned system which may be adequate for the Doppler-type measurements may not be adequate for the measurements of α or the average cross section.

The errors in the values of α and the average cross section will be evaluated as an aside to the problem of reproducing the characteristic spectrum for a reasonably small-sized test zone. This is a continuing problem and will be subject to change depending on cross-section revisions and improvements.

The theoretical effort has been in generating a consistent set of 18-, 4-, and 1-group capture, fission, and transport cross sections. The 18-group and the 4-group cross-section sets have been utilized in a consistency study of one- and two-dimensional calculations involving the sodium and fuel-expansion reactivities for systems of cylindrical geometry having L/D ratios of 1, 0.5, and 0.25. Studies of reaction rate will be done with the CYCLE code.

C. Engineering Irradiation

At full reactor power, it will be possible to obtain, in several weeks, significant information about materials. The ability to instrument fuel assemblies, which will be designed into FARET, is equally important to the conduct of an experimental program. The general objective of the experiments will be the establishment of limiting conditions of temperature and burnup of various promising types of fast reactor fuel elements. Such a program will be concentrated on the advancement of the fertile-based, plutonium-bearing fuels and the cladding combinations necessary for their containment.

The experimental program for fuel development is envisioned to start with the performance evaluation of a complete, small core of high power density. The selection of this core will be based on existing technology, with the objective of determining the characteristics of a complete core under high power density, fast flux, and high-temperature conditions. Uranium-plutonium monocarbide is proposed as the reference fuel material for the initial core. As an alternative, uranium-plutonium dioxide will be considered even though the power density is lower. Stainless steel or a nickel-base alloy will be used as the jacket material if proven satisfactory. If not, a refractory metal alloy will be considered.

Accompanying or following initial operation of the small core, experimental fuel elements of advanced design and composition will be irradiated. If the risk of failure is high, the fuel specimen will be contained in a secondary jacket during initial experimentation. Satisfactory performance of a number of such specimens will then be followed by testing of uncontained elements. Final experiments will be conducted on full core loadings of these advanced fuels.

1. Fuels

a. Carbides

The main interest is in the uranium carbide-plutonium carbide system (10 to 20 w/o PuC). Fuel elements containing these compounds prepared by arc melting and casting, pressing and sintering, and mechanical compaction will be included. The experiments will be designed to produce information on the swelling behavior, release of fission gas, clad compatibility, burnup and temperature capability, and effective thermal conductivity. These experiments will be made with stoichiometric materials and with those of varying carbon contents to establish the degree of carbon control necessary in a reactor fuel. Extensive use will be made of devices for temperature and flux measurement so that appropriate evaluations can be made.

Experiments will be made with various jacket materials starting with the stainless steel and nickel-base alloys, and extending to refractory alloy jackets based on niobium, vanadium and molybdenum.

b. Oxides

Although oxide fuel suffers from the disadvantages of low density and low thermal conductivity, it has good compatibility with most cladding materials. Some irradiations of these fuel elements are planned.

c. Sulphides, Phosphides, and Nitrides

The monosulphides, monophosphides, and mononitrides of uranium and plutonium are potentially attractive as fast reactor fuels. Irradiation performance of elements containing these fuels is practically unknown. The irradiation performance will be determined, first, for the uranium compounds and then for the plutonium and mixed uranium-plutonium compounds.

d. Alloys

Plutonium-bearing alloys developed as potential high-performance fast reactor fuels will be included in the FARET irradiation program. Among these alloys will be included those based upon the fertile elements uranium and thorium and those based on inert elements, such as zirconium and iron. Of primary interest in these metallic fuels are the parameters of composition, microstructural features, temperature, and burnup which effect swelling.

e. Cermets

Very strong fuels can be made by incorporating refractory fuel compounds in inert metals as cermets. An example is the comparatively well-developed UO_2 -stainless steel system. Such cermets generally are not favorable to good breeding or to high fuel burnup without a large amount of fuel handling, so that their economic promise is doubtful. If economic analysis shows favorable characteristics, they would of course be brought into the FARET program.

Of interest are cermets involving plutonium compounds and uranium metal. This would be one way of incorporating plutonium into uranium without encountering the deleterious effects of plutonium on the metallic phase. The first phases of this effort would involve the development of these dispersions, and the determination of stability of plutonium compounds in a uranium matrix, the determination of radiation effects, and the analysis of fabrication and process concepts.

2. Cladding Materials

In addition to the cladding materials that are involved in the irradiation of the fuel materials discussed above, determination of irradiation effects on these materials alone in fast neutron flux under reactor operating conditions is needed. Both in-pile and postirradiation testing of refractory and commercial alloys will be performed to determine changes in the mechanical and physical properties of these materials. The cladding materials of interest are the stainless steels, super alloys, and refractory alloys. In addition duplex and coated tubes require study.

Irradiations will be made to determine the ability of various promising cladding materials to restrain high temperature fuel swelling. Some success has been obtained in the control of swelling of plutonium alloys in thermal reactor irradiations. FARET, with its fast flux environment, will be a much better place to evaluate restraint capabilities.

3. Fuel Design and Operation

Fuel-element design and operating conditions which affect the performance of these fuels will also be studied. Irradiation of intentionally defected fuel elements will be performed to determine fission product release from the fuels discussed previously. Sodium logging will also be studied in defected, unbonded fuel rods. Metallic fuels will be irradiated to determine the conditions which affect the clad compatibility.

4. Vented Fuel

Single-element vented-fuel experiments are planned for FARET to determine the quantity and distribution of fission products that escape

into the system. If the release of activity into the system is found to be moderate or can be held down to acceptable levels by suitable gas or coolant cleanup, then the containment requirements of the cladding material could be relaxed or the clad thickness could be reduced. Then, under these conditions some of the high-temperature metals, which have relatively high absorption cross sections, might be used as cladding without significantly depressing the breeding gain.

The assembly will be instrumented to obtain central fuel temperature and cladding temperature at two or three axial levels, and the assembly will be vented to a chamber, outside the vessel, where samples of the gas can be withdrawn for analysis. Subsequent experiments will employ complete subassemblies.

These experiments are expected to yield valuable information applicable to the design and installation of a future full-scale, small-core, vented-fuel experiment. In particular, they will indicate the problems connected with cleanup of the blanket gas.

In addition to the concern with the release rates of xenon and krypton, there is concern about other nuclides that might be released. It will be particularly important to observe the behavior of cesium during experiments. Cesium activity could build up in the primary coolant via xenon decay or by direct release from the fuel as cesium. It would become a serious problem, not only because of its 30-year half-life, but because it is not removed by conventional coolant-cleanup operations (cold trapping or hot trapping).

D. Coolant Chemistry

Purification of the sodium coolant is limited by existing technology to two basic methods, namely, cold trapping and hot trapping. Under ideal design and flow conditions, the efficiency of the former method for removing impurities is limited to the solubility of the impurity in sodium at the lowest interfacial temperature in the cold trap. Hot trapping can remove oxygen, nitrogen, and carbon; its efficiency is related to the available surface area of the getter material, temperature of operation, and the thermodynamics of the getter metal-impurity-sodium system. The selection of getter material, its physical form and the mass of getter employed become controlling factors.

It is anticipated that the advanced operating conditions of FARET will demand maximum or greater purity than is currently available by existing purification methods. Experimental work will be directed toward possible improvements in the existing purification methods and in the evaluation and testing of new methods.

In order to permit improvement of existing methods, the effects of flow rate on trapping, improvements of trap design, and the use of some unconventional getter materials capable of reducing sodium oxide will be studied.

In the realm of new methods, the possibility of testing a continuous-flow, in-line distillation or evaporator type system will be considered. Extrapolation of data from the literature indicates evaporation rates of ~ 100 kg/hr can be obtained from a 36-in. (90-cm) diameter still or evaporator operating at ~ 30 -50 mm Hg. It is unlikely that this method of purification would replace either of the other two. It would function as a supplementary system, and, being independent of the limitations of cold trapping and hot trapping, it would be expected to achieve extremely low levels of gross impurities. From a consideration of the vapor pressure and dissociation characteristics of the common impurities found in sodium, effective separation from all contaminants except potassium and cesium should be possible.

Provisions have been made in the FARET design to permit installation and testing of an advanced type of purification system.

An experiment is planned to study continuously buildup of corrosion product activity in the primary coolant. Periodically, samples of sodium will be withdrawn from the system. These will then be separated by a rapid and convenient wet chemical technique into three or four basic groups of elements. Each group will be analyzed by gamma-ray-scintillation counting to identify and obtain a relative measure of the active nuclides.

Changes in activity levels induced by the variation of other reactor parameters, such as operating temperature and oxide concentration of the coolant, and power level, will be observed. Of significance will be the observation of possible effects due to the introduction of new or different cladding materials into the system.

It is also planned to determine corrosion rates and to study mass-transfer effects. Metal coupons will be located at various positions in the vessel for this purpose. The blanket-gas zone in FARET operating at 1200°F (650°C) will be rich in sodium vapor (~ 23 g/m³ at 1200°F) and therefore provides an ideal region for investigating carryover of active materials into the blanket-gas system. Information gained from the coupons by radiochemical and metallurgical analysis will supplement that obtained by analysis of coolant samples. Together, the two sets of data will give an understanding of the formation, transport, and deposition of corrosion product activity.

IV. MATERIALS DEVELOPMENT

A. Cladding Materials

1. Stainless Steel

The most widely used materials for cladding are the stainless steel alloys. These alloys are readily fabricable, and the experience associated with them is both extensive and well established among the numerous commercial suppliers. The short-time and long-time mechanical properties of these alloys are extensively documented over wide temperature ranges and in a variety of environments. Work is currently in progress at Mine Safety Appliance Research (MSAR) to study the effect of a liquid sodium environment on the mechanical properties of these alloys at temperatures up to 1200°F (650°C). Preliminary results suggest a negligible effect, but this conclusion should be substantiated by a greater number of tests.

The austenitic stainless steels have been shown to have good corrosion resistance at 1200°F (650°C) if the impurity content of the sodium is maintained at a low level. The stainless steels are sensitive to the level of oxygen impurity in the sodium. Tests have shown that the corrosion rate is markedly increased if the oxygen concentration in the sodium exceeds 200 ppm. With extremely high concentrations of oxygen, sodium oxide causes intergranular attack, with a corresponding deterioration of mechanical properties.

The problem of mass transfer of carbon in liquid sodium and its effects on Types 304 and 316 stainless steels is currently being investigated by the Atomic Power Equipment Division of General Electric at San Jose, California. To date, the mechanism of transfer and a correlation of the data regarding effects on materials are yet to be determined. Carbon contamination in sodium also adversely affects the mechanical properties of austenitic stainless steels. Excess amounts of carbon in sodium produce carbonization at the surface resulting in a brittle surface layer.

Fast-neutron exposures up to 10^{22} nvt tend to increase the tensile and yield strengths and decrease the ductility of austenitic alloys. Limited data are presently available regarding the effect of fast-neutron exposure in a sodium environment on the impact strength of these alloys.

2. Refractory Metals

These metals may be of interest as jacket materials because of their high melting points. All metals with melting points above 1800°C can be classified as refractory metals and these are shown in Table XII. Of the fourteen listed metals, tungsten, tantalum, molybdenum, niobium, and vanadium are the principal members of the refractory metal group. The balance

is primarily precious metals. A large number of refractory metal alloys have been tested for strength. However, material condition and conditions of testing have not been consistent, which complicates comparison of alloys. Most tests have been short-time tensile tests conducted at room temperature and at about 2000°F (1100°C). Interpolation to intermediate temperatures is inadequate.

Table XII

MELTING POINTS OF REFRACTORY METALS*

(above 1800°C)

Metal	Melting Temp		Metal	Melting Temp	
	°C	°F		°C	°F
Tungsten**	3410	6170	Niobium (Columbium)**	2368	4294
Rhenium	3180	5755	Hafnium	2222	4032
Tantalum**	2996	5425	Boron	2030	3686
Osmium	2700	4852	Rhodium	1966	3571
Molybdenum**	2610	4730	Vanadium**	1900	3452
Ruthenium	2500	4532	Chromium	1875	3407
Iridium	2454	4449	Zirconium	1852	3366

* Reference 13, 8th Edition, 1961, pg. 46.

** Principal members of the refractory metal classification.

Little information is available for long-time tensile tests. Creep tests have been conducted up to 100 and 1000 hr on many of the refractory metal alloys at temperatures at about 1100°C. Very little creep information below 1100°C is available. In many cases, short-time tensile properties do not adequately describe long-time strength, due to structural changes caused by stress, temperature, and environment. Time for these changes to occur is not provided in short-time tests. Therefore, there is a need for long-time testing which will simulate service stress, temperature, and environmental conditions. There is considerable evidence that refractory metal alloys lose ductility when exposed to impure liquid sodium.

In the temperature range from 600 to 700°C, an important factor of corrosion rates in the refractory metal systems is probably the oxygen impurity in the sodium. The corrosion rates of the refractory materials increase with oxygen content, and the tolerance levels for oxygen in the sodium are lower in these cases than for the stainless steels. There is evidence that many of the refractory alloys now available will not perform adequately at the impurity levels presently found in large-scale sodium systems.

The effect of irradiation of the refractory metals, though not thoroughly studied, has been shown to increase the tensile strength and decrease the ductility.

At this time only a few refractory metals or alloys are commercially available in tube form suitable for cladding of FARET fuel elements. These include Nb, Nb-1 Zr, Nb-33 Ta-1 Zr, V, Mo, Ta, and Ta-10 W. Highly specialized techniques are required for the production of tubing from many of the refractory metals. Such techniques are currently being studied. Most assuredly some time will pass before quality cladding materials are commercially available in all of the refractory metals of interest.

Alternative candidate cladding materials should be selected for "back-up" support in the event that the above commercially available tubing materials do not fulfill FARET cladding requirements. Preferable, the alternates should be in a highly refined state of development. Four possible alternate alloys worthy of continued interest are listed in Table XIII in the relative order of commercial availability and fabricability.

Table XIII

ALTERNATE REFRACTORY
METAL ALLOYS

Availability	Fabricability
Nb-10 Ti-5 Zr ⁽¹⁾	Nb-10 Ti-5 Zr
Nb-5 Zr	Nb-5 Zr
V-20 Nb-5 Ti	V-20 Nb-5 Ti
Nb-39 V-1 Ti	Nb-39 V-1 Ti

(1) Numerals indicate weight percent - w/o.

tubing could involve extensive development effort, depending upon the desired bond quality and integrity.

Duplex tubing should also be investigated. The concept would involve stainless steel on the outside with a thin refractory metal layer on the inside. The production of such

At contemplated higher temperatures of operation, stainless steels and the nickel-containing alloys are not compatible with metallic plutonium-bearing fuels. Refractory metals must therefore be considered. High-purity sodium and lithium are reasonably inert toward high-quality (i.e., low level of interstitial impurities) niobium, molybdenum, tantalum, and their alloys, up to at least 800°C.⁽⁶⁻⁸⁾ However, specifications for coolant chemistry and alloy purity necessary to the operation of practical impure systems remain to be established and require extensive research efforts. Such efforts are in progress at many laboratories, including Argonne.

A compilation of reported mechanical and physical properties are listed in Table XIV. The materials are considered to be representative of the types rather than of specific choices.

Table XIV

MECHANICAL AND PHYSICAL PROPERTIES*

Clad Temp, °F	SS-347 (7,9,10)	Hastelloy (X) (9,10,11)	Mo-alloy Low Strength (12,13)	Nb-alloy High Strength (12,14)
<u>E, Young's Modulus: psi x 10⁻⁶</u>				
Room Temperature	27.5	28.5	37.1	25
1000 (540°C)	22.8	24	33.7	23
1200 (650°C)	21.4	22	32.2	23
1400 (760°C)	20	21	28.5	22
<u>Coefficient of Linear Expansion: °C⁻¹ x 10⁶ (average from about 20°C)</u>				
1000 (540°C)	16.0	15	5.83	7.6
1200 (650°C)	16.5	15.5	5.9	7.6
1400 (760°C)	17.0	15.9	6.0	8.0
<u>Thermal Conductivity: cal/cm/°C/sec</u>				
1000 (540°C)	0.053	0.047	0.260	0.103
1200 (650°C)	0.0565	0.052	0.258	0.106
1400 (760°C)	0.0595	0.057	0.258	0.110
<u>Creep Strength - 1% creep in 10,000 hr: lb/in.²</u>				
1100 (590°C)	23,500	24,000	55,000	55,000
1200 (650°C)	15,000	19,000	50,000	50,000
1300 (700°C)	10,000	11,000	45,000	40,000
1400 (760°C)	2500-5000	8,500	40,000	30,000
<u>Performance No., fk/αE</u>				
1100 (590°C)	3.58	-	-	-
1200 (650°C)	2.40	2.9	68	30
1300 (700°C)	1.67	-	-	-
1400 (760°C)	0.44-0.88	1.45	60	19
<u>Max Specific Power Corresponding to the Performance No.: Mw/kg of UC</u>				
1100 (590°C)	1.16	-	-	-
1200 (650°C)	0.78	0.95	22	9.8
1300 (700°C)	0.54	-	-	-
1400 (760°C)	0.14-0.28	0.47	19	6.2

* Some values represent estimates when data not available.

Lithium has a very high boiling point and good thermal conductivity. Using lithium as a bond instead of sodium would allow higher fuel temperatures without internal pressurization to avoid boiling. Since the initial fuel elements will be designed with a fission gas reservoir, permitting thin-wall cladding, the lithium bonding concept may be considered in the FARET study program. The lower vapor pressure of the carbide fuels will also be considered as a factor when the choice of ceramics is made.

In addition to having good compatibility, the clad material should have a good thermal conductivity, low coefficient of linear expansion, and high-temperature strength. A dimensional number whose numerical value is a measure of performance is $fk/\alpha E$, where f is the permissible stress, k is the conductivity, α is the linear expansion coefficient, and E is Young's Modulus. This performance number is based on the assumption that the maximum specific power and temperature for the fuel-clad combination depends on the thermal stress in the clad. For the case of internal pressurization, the performance number is proportional to the stress. Large thicknesses cause thermal stresses to become the limiting factors.

For high-temperature service, the permissible stress was based on the creep strength of the material. The value selected in this study is the stress (extrapolated) required to produce a total creep of 1% in 10,000 hr.

Table XIV gives the values of the various quantities used in computing the performance number. The significance of this number is based on the fact that it is proportional to the theoretically attainable specific power for the indicated clad temperature. The computed powers for these temperatures are based on 0.174-in.-OD x 0.009-in.-wall (0.44 x 0.023-cm) tubing.

The survey and, in particular, the performance number indicate that, insofar as thermal stresses are concerned, there is a minimum restriction on the specific power when refractory alloys are used as cladding materials.

Preliminary experiments^(15,16) to determine irradiation effects on rupture stress show that the stresses are reduced by a factor of 2 and that this reduction appears to be independent of temperature. These preliminary tests were made with stainless steel Inconel, Inconel X, and Nb-1% Zr alloy.

B. Metallic Fuel

It has been proposed to use metallic fuel only for the driver zone of the zoned experiment. This fuel can be uranium-fissium of the type used in Core I of EBR-II. It is believed that minimal development will be required, since the uranium-fissium alloys have been shown to be relatively stable under the irradiations conditions effected in the driver core.

A preliminary flow diagram for production of the driver-zone fuel elements is shown in Fig. 14. The fuel pins are precision cast by procedures already used for EBR-I and EBR-II fuel rods. Although these methods are developed, they must be adapted to the production of rods that are of larger diameter and longer.

Fig. 14. Preliminary Flow Diagram for FARET Driver Fuel

The jacketing, sodium-filling, and bonding procedures are similar to those developed for the EBR-II fuel rods. The FARET driver fuel slugs [0.305-in. (0.768-cm) diameter] compare roughly with the EBR-II upper and lower blanket slugs [0.316 in. (0.796 cm) in diameter]. EBR-II sodium-loading assembly and bonding procedures and equipment designs should apply with only slight modifications.

Because of the difficulties in applying conventional fabrication procedures to these alloys, precision-injection casting was developed. This method has been applied satisfactorily to production of the uranium-fissium alloy core for EBR-II, to a plutonium core for EBR-I, and to the manufacture of a large variety of test specimens of uranium and uranium-plutonium alloy.

The current status of the gas pressure-injection casting process is as follows: castings 0.36 cm to 0.95 cm in diameter by 45 cm long have been made. Diametral precision is over 99%. New equipment would have to be designed and built if this method is to be used for the larger fuel rods. Gas pressure-injection casting has not been done commercially.

In order to proof test a fuel material which appears to be suitable for the driver zone, uranium-15 w/o fissium or other alloys will be irradiated. Three prototype fuel elements which have the same diameter as those proposed for the driver zone in the zoned-core test will be irradiated in the MTR in instrumented capsules. These irradiations will serve to substantiate the information obtained on specimens of uranium-fissium alloys of smaller diameter.

For the conventional fabrication of metallic fuels other than uranium-fissium alloys, there exists sufficient capacity at several commercial vendors. These vendors may have to rely on the Laboratory for technological assistance. For example, an alternate element in which the fuel is bonded to the cladding by co-extrusion has been produced commercially for the EBR-I Mark-III core.

C. Ceramic Fuel

A relatively small amount of irradiation information is available on UC-PuC. The Laboratory has irradiated both arc-melted and pressed-and-sintered specimens of UC-20 w/o PuC in EBR-I to a maximum burnup of 0.1 a/o (890 Mwd/MT) at center temperatures of the fuel ranging from 370 to 490°C. Fission gas release from the arc-melted specimens was low, whereas the release from the pressed-and-sintered pellets was considerably higher because of low pellet density.

Two small-diameter UC-10 w/o PuC arc-melted pins have been irradiated at Harwell in PLUTO to an estimated burnup of 22,800 Mwd/MT at a center temperature of 750°C. Very little fission gas was released, and

changes of the cladding dimensions were insignificant. A slightly higher gas release from one specimen was thought to be associated with observed grain-boundary cracking.

Preliminary work at Dounreay indicates that the addition of 20 w/o PuC to UC reduces the thermal conductivity of cast material by a factor of almost three at about 50°C. The alloy, however, was hyperstoichiometric.

Studies have been made or are being made of the preparation and fabrication of UC and PuC. The mixed carbides UC-PuC are being prepared, and the sintering characteristics of these mixtures are being evaluated. This program will be expanded to measure the pertinent properties of these mixed carbides, such as compatibility with jacket materials, the effect of stoichiometry, thermal properties, and phase relationships.

Although the technology for UC is further advanced than that for PuC or for mixtures of UC-PuC, complete data about all characteristics are not available. Details of preparation and fabrication of the mixed carbides, to include optimum sintering time, temperature, atmosphere, shrinkages, density, homogeneity, and stability, must be established.

The Metallurgy Division at Argonne National Laboratory has underway a program for the irradiation of nuclear carbides, including UC, PuC, and UC-PuC. The program covers the irradiation of vibratory-compacted, cast, and pressed-and-sintered specimens of these fuels in various cladding materials of interest. Data being gathered from the experiments include information on fission gas release, swelling behavior, clad compatibility, burnup and temperature limitations, and in-pile effective thermal conductivity. The effect of stoichiometry is also being investigated. The experiments are being conducted at cladding surface temperatures of 500, 600, 800, and 1100°C. Anticipated burnup levels are 2, 5, 10, 15, and 20 a/o of the metal atoms. The experiments are being carried out in the CP-5 and MTR reactors. Experiments will also be conducted in EBR-II when space in this reactor becomes available. Current emphasis is on Type 304 stainless steel and niobium 1 w/o zirconium claddings of 0.009-in. (0.023-cm) to 0.020-in. (0.051-cm) thickness.

The major portion of the data on UO_2 - PuO_2 fuel has been contributed by GE-VAL and GE-HAPO. In the program at GE-VAL,⁽¹⁷⁾ stainless steel-clad swaged and pellet rods of PuO_2 -80 w/o UO_2 solid solutions were irradiated to burnups ranging from 5,000 to 99,000 Mwd/T at heat fluxes ranging from 0.5×10^6 to 1.6×10^6 Btu/(hr)-(ft²). The cladding surface temperatures ranged from 370 to 650°C. Cladding thickness was 0.016 in. (0.04 cm) on the pellet rods and 0.020 in. (0.05 cm) on the swaged rods. No changes of cladding dimensions were noted, and no cladding failures occurred. Maximum length increase of the fuel zone was 13% for the swaged rods and 7% for the pellet rods. Fission gas release ranged from 10 to 65%, with

the average for swaged rods slightly higher than for pellet rods. It was noted that the gas release was higher than would be expected from UO_2 under similar conditions. Designing for 100% gas release appears to be a satisfactory method for accommodating the released gases. Increased columnar grain growth and central void formation were noted in areas of substantial fuel-clad gap. The pellet rods were assembled with a 6- to 8-mil (0.15- to 0.20-mm) diametral clearance.

Hanford has worked with UO_2 - PuO_2 fuels which generally contained less than 5 w/o PuO_2 . Sintered UO_2 - PuO_2 pellets and swaged mixtures of PuO_2 and UO_2 in Zircaloy-2 cladding were irradiated to maximum burnups of 10,000 Mwd/T. Gas release ranged from 3% for dense sintered pellets to 22% for low-density unsintered pellets. It was noted that the greatest gas release occurred in specimens in which the plutonium had been added as the coprecipitated mixed-crystal oxide.

The technology for fabricating UO_2 is well known, and the fabrication of PuO_2 and of the mixed oxides of UO_2 and PuO_2 is being studied at several laboratories. These studies have to some extent evaluated the various methods of forming, the sintering characteristics, and many of the pertinent properties of the mixed oxides. Although only limited studies have been made on PuO_2 at Argonne, no extensive development would be necessary to test the manufacture or the properties of mixed oxide compositions.

Commercial facilities have not yet produced any appreciable quantities of mixed plutonium-uranium oxide or carbide fuel materials. GE-HAFO has experience with mixed oxides for PRTR and Plutonium Recycle fuels. Their technology could be applied at other fabricators. One vendor has a pilot-plant facility capable of making plutonium carbide, but does not have plant capacity for producing a reactor fuel loading. However, there is likelihood that by January 1964, plutonium facilities will be constructed which will be capable of producing all the plutonium carbide needed for FARET. Additional time will be required for development of the mixed uranium-plutonium carbide.

Some intermetallic compounds with plutonium and uranium have melting temperatures well above the maximum desired operating fuel temperature, yet they do not melt so high that they cannot be fabricated by conventional melting and casting techniques. PuFe_2 and UFe_2 are especially interesting in that they melt at about 1240°C. Under consideration is a simple fuel assembly consisting of $(\text{U},\text{Pu})\text{Fe}_2$ (with uranium and plutonium in the ratio of most interest to fast reactors) within a stainless steel clad. Such a combination is promising because PuFe_2 and UFe_2 form relatively high-melting eutectics with iron (1080 and 1165°C) and because stainless steel tubing is both economical and proven in a sodium environment. Nickel-rich plutonium and uranium alloys show similar potential.

In addition to these metal fuels, several refractory compounds of plutonium are of interest. Currently, the preparation of plutonium sulphide and phosphide is being investigated. These compounds have high melting points and appear to have thermal properties intermediate between the metal and the oxide. Although the data concerning these compounds are meager, it also appears that property modifications pertinent to the individual compounds can be made by mixing more than one compound. There is evidence that at least a limited region of solid solution exists between the compounds uranium monocarbide and uranium monosulphide, and that other solid solutions may be possible between other compounds of uranium and plutonium. Thus, a wide variety of fuel materials could be compounded, which would have certain advantages for reprocessing.

D. Fuel for Reactivity Test

Uranium dioxide fuel has been selected for the proposed reactivity measurements. Several different shapes have been suggested, one of which is an extruded rod with a central hole for temperature measurement. The outer surface of the rod has small projections which will serve to center the rod in the jacket and will allow the passage of gas through the tube. These projections will also minimize the thermal conduction from the fuel to the jacket.

Techniques for extruding this shape are being developed. Some difficulty has been encountered because of cracks developing during drying and in machining the projections on the outer surface of the rod. As extruded, the projections are in the form of fins, which are partially ground from the rod to form the projections. During the fabrication of the first samples, some breakage occurred. It is anticipated that a further study of binder content will eliminate these difficulties.

V. FUTURE WORK

A. Physics

The study of optimizing core configurations for statics measurements will be continued, the objectives being to minimize the fuel loading and to maximize the reactivity sensitivity of the test. Techniques such as replacing the buffer zone with a filter-type section of depleted uranium will be examined.

A problem which arises in the zoned system is the boundary effect. This is the region in the vicinity of the test and buffer-zone interface. In general, the Doppler effect is dependent on the fine, as well as the gross, structure of the neutron spectrum. The fine structure is essentially due to the composite resonance structure of a reactor system. Since the resonance broadening and corresponding increased reaction rates occur on the heated side of the boundary (test zone), the reaction rates in the vicinity of the boundary would not be representative of the composition at temperatures on either side of the interface. For too small a test zone, this may lead to an appreciable error for the Doppler coefficient. The boundary effect may be accounted for experimentally by resorting to a series of measurements as a function of test zone size.

The analysis of the ZPR-III critical loading of a zoned uranium carbide system is in progress. The loading consists of a 25-cm-radius test zone with a composition of approximately uranium-carbon/sodium/SS fraction of 27.2/47.73/19.61, respectively. The composition of the surrounding 1.7-cm, depleted uranium-carbon/SS filter was 48.6/9.1. The driver zone contained a mixture of 46/36 volume fraction of uranium-graphite/SS, respectively. The computed fission rates will be compared with the experimental values which are currently being determined.

A program of theoretical investigations is being planned for transfer-function-type tests and moderate transient measurements. The objective of this study is to select a set of experiments which have the maximum possibility of yielding interpretative results.

The early phase of the critical experimental studies is being planned for ZPR-VI with U^{235} enrichments. This would be primarily aimed at understanding zoned loadings and refining sets of cross sections.

B. Instrumentation

1. In-vessel Connector

The advantages of terminating in-core sensor leads with connectors located in the reactor vessel have already been discussed in Section II, D.6. A program will be undertaken which will provide a connector suitable for use in the FARET facility.

The basic specifications for this type of connector are:

- (1) temperature: continuous operation up to 650°C;
- (2) insulation resistance: at least 100,000 megohms between pins at 650°C with 500 v(dc) applied;
- (3) construction: stainless steel with proper gasket seal between plug and receptacle to prevent the admittance of sodium vapor;
- (4) contacts: pins and sockets made of tungsten-5% rhenium, tungsten-26% rhenium, chromel, alumel, and stainless steel.

The major problems which must be solved are:

- (1) keeping the insulation resistance high (>100,000 megohms) at 650°C;
- (2) keeping the contact resistance low between mating pins and sockets at temperature, and
- (3) fabricating refractory alloy thermocouple contacts, both pins and sockets.

Connectors which meet most of the basic specifications are now commercially available. Test specimens will be purchased and evaluated. Modifications to these connectors will be made as the tests proceed.

2. Fuel-pin Thermocouples

The reference thermocouple for measuring temperatures of fuel pins has been selected (see Section II.D.1). Commercial suppliers of thermocouples will be contacted to determine the availability of these sensors. The development program will be finalized, and the testing and evaluation schedule will be determined.

C. Core Design

The initial designs of the various fuel assemblies, control rods, and reflector elements have been started. The current phase involves an analysis of design as it relates to heat transfer, fluid flow, and considerations of materials. The next phase would include methods of fabrication, assembly, disassembly, and installation of the core components. Model core components will be constructed as necessary for laboratory-scale tests.

The core support grid and the core support cylinder will require detailed stress analysis. Current considerations include effects of high levels of induced radioactivity on the feasibility of interchanging core components.

The problems, which appear to be critical, are: (1) insertion and removal of instrument leads attached to core assemblies; and (2) disassembly of fuel (or other elements) when instrument leads are attached to fuel pins.

D. Heat Transfer and Fluid Flow

The following material applies equally to either the Zoned Core, Medium Core, or Small Core, unless otherwise noted. To assure adequate designs of the experiments it will be necessary to obtain thermal data from out-of-pile model fuel elements. This subject is discussed, with respect to the fuel elements for fuel expansion reactivity measurements, in Section III.A. With respect to more conventional fuel elements, out-of-pile thermal measurements will be used only to determine interface resistance between fuel and cladding. The only new equipment needed for this latter work is a model fuel element for each prototype fuel element to be tested in the reactor.

It is also advisable to observe the thermal behavior of the fuel elements for Doppler coefficient measurements, with nuclear heating as well as with electrical heating. This is true mainly because, with electrical heating, the gaps between pellets have a heat source in them, namely, the heater, which the gaps do not have with nuclear heating. Therefore, it is planned to conduct a well-instrumented in-pile thermal test of such a prototype fuel element in the CP-5 reactor. The irradiation capsule will be designed so that fuel temperatures within 200°C of the fuel melting point may be attained. A variable thermal conductance between fuel element and reactor coolant will be included to permit variation of fuel temperatures.

For fuel elements of more advanced design, it will be necessary to conduct similar in-pile experiments to establish fuel- and clad-interface thermal conductances. It is also planned to design the subassemblies and reactor vessel for proper coolant flow distribution by means of the information and experience gained from the EBR-II model flow tests.

In order to insure that the subassemblies are sturdy enough to withstand actual operating conditions, it is desirable to test a cluster of subassemblies in a sodium loop operated at reactor temperatures. The characteristics of pressure drop vs flow rate of the subassemblies will be measured in the test, for use in the analysis. Use of a cluster of subassemblies will make it possible to verify that the subassemblies can be inserted and withdrawn under simulated reactor conditions. This test will require construction of a new loop.

A detailed thermal analysis will be made to determine the number and placement of thermocouples to be installed in the reactor for each experiment. The goal of this analysis will make it possible (1) to use as few thermocouples as possible consistent with acceptable accuracy, and (2) to assign a standard deviation to each of the temperatures used in analyzing the experimental data. This will aid in assigning a standard deviation to the overall experimental results.

E. Chemistry

Future work with respect to advances in knowledge of coolant chemistry required for the program will involve study and development of a low-level oxide-measuring technique and study and development of advanced systems for sodium purification. Advancement of coolant technology is necessary for the operation of sodium-cooled, nuclear reactor central power plants using sodium at 650°C.

Reference conditions for a sodium-cooled, stainless steel reactor system based on current technology are considered as operating (at 500°C) with or tolerating from 50 to 100 ppm of oxygen as sodium oxide. For the FARET condition (650°C) it is reasonable to expect the impurity tolerance in the sodium to be in the range from 5 to 10 ppm oxygen. However, the maximum tolerance for oxide impurity will be established by high-temperature corrosion and metallurgical testing of the proposed core materials.

The planned work will be directed toward the investigation of oxide-measuring techniques sensitive in the 0-10 ppm range and of advanced systems for sodium purification capable of maintaining low levels of gross impurity.

Although practical limits of sensitivity for measuring oxygen in sodium ranging from <10 ppm to 50 ppm are claimed, a comprehensive analysis of the most accepted methods in use to date⁽¹⁸⁾ leads to the conclusion that there is no proved method which is free from systemic interferences and which is capable of measuring oxygen in sodium in the range of 400 ppm or less.

The Laboratory is currently engaged in the study of a most promising approach to the problem of determining oxygen, namely, by the neutron activation technique. This method uses 14-Mev neutrons, from a low-cost Cockroft-Walton type accelerator, to react with O^{16} by an (n,p) reaction to produce N^{16} . The N^{16} emits 6-7-Mev gamma rays, with a half-life of 7.4 sec, which can be measured by scintillation spectrometry.

Two other techniques that will be investigated should provide sensitivities well below the 10-ppm range. These are (1) an adaptation of the neutron activation approach utilized for the determination of oxygen in lithium, and (2) a tracer technique employing tritium.

The activation technique requires the preparation of a sodium-lithium alloy containing low oxide content. Tritons produced by the reaction $\text{Li}^6(\text{n},\text{t})\text{He}^4$ activate the oxygen by the reaction, $\text{O}^{16}(\text{t},\text{n})\text{F}^{18}$. Oxygen content is then determined by measuring the annihilation radiation due to the β^+ decay of F^{18} , which has a half-life of 1.87 hr.

The tracer technique requires isolation of the sodium oxide from a sample of sodium. The oxide is then tagged by exposure to tritiated water vapor. After conditioning the tagged sample (now in the form of NaOH^2 and NaOH^3) to remove any excess or adsorbed tritiated water vapor, it is transferred to a liquid scintillation solution for counting the 19-keV β^- ; tritium has a half life of 12.26 yr.

The success of both of these techniques depends strongly on performing the sampling and sample-preparation steps in a vacuum or in an absolutely oxygen-free atmosphere.

The mechanism of carbon transport in liquid sodium is a program being set up by the Laboratory for the study of the carburization and decarburization of ferrous alloys and refractory metals in liquid sodium. Current plans are to employ carbon enriched with C^{14} in the examination of the kinetics of carburization with stainless steel.

It is also planned to develop a device for monitoring relative effects of changing oxide concentrations in the sodium for use in an operating reactor plant. Such a device would eliminate routine sampling and analysis, and require only periodic calibrating or standardizing. Future work in this area will be to examine effects that may interfere with the measurements. It is planned to construct a working device and test it under reactor operating conditions.

The items that will be examined continually are:

- (1) coolant additives to inhibit corrosion;
- (2) cold trapping and hot trapping;
- (3) basic investigations of the effects of other impurities, i.e., carbon on corrosion and mass transfer.

VI. PERT SCHEDULING

PERT (Program Evaluation and Review Technique) has been used by a large and diversified number of organizations and groups to plan and schedule engineering, construction, maintenance, process control, production, and also research and development projects. Improvement in time schedules has been the principal advantage as a result of the use of this new planning technique. The Laboratory's ZGS facility has benefited from the use of this technique, and it was found a valuable tool in aiding the progress of the construction. The proven success for planning and evaluating the progress of a major research and development program, such as the Fleet Ballistic Missile Program for the U.S. Navy, has brought the use of the PERT technique into the FARET program.

Basically, the plan of a research and development program is similar to that of any other type of program. A series of tasks are scheduled in a logical sequence that builds up to attainment of a final objective. Product performance is specified, resources are allocated, and time of achievement of each task is presented. However, there are factors which set research and development programming apart from this generalized approach. First, there is an attempt to schedule intellectual activity as well as the more easily measurable physical activity. Secondly, the unpredictability of specific research results inevitably require frequent change in program detail. These points are acknowledged by all experienced research people.

The identification of the programmatic objectives is the first step in the PERT program. In this process, the current end objective is toward the testing of three cores: a zoned core, a medium-sized core, and a small-sized core. There are many parallel paths that lead to the end objective, all of which must be completed. These paths are composed of various activity items. The subsequent division of these items into component parts creates a structure which serves as a framework for planning and controlling the program. Such a work breakdown structure is shown in Fig. 15.

This approach to planning and control of a developmental program assures that the total program is fully planned and that all derivative plans contribute directly to the desired objective. The level of detail that is desirable to apply to a PERT system is a matter of judgement and varies as a program progresses. The degree of detail required for planning and controlling during the execution of a program is naturally greater than that needed during the preparation stage for the program. The outline in Fig. 15 is an over-simplification of the more detailed diagrams that will be required during the program. The outline is developed only to the level of detail essential to achieve the degree of planning and control required. Later, there will be an expansion into more detailed levels as needed for effective planning and control of the work.

Fig. 15. PERT Basic Flow Chart

Scientists are generally cognizant of the fact that research and development projects are of a pioneering nature. Therefore, previous parallel experience upon which to base schedules of a new project is relatively unavailable. Also, as stated previously, the unpredictability of specific research results inevitably requires frequent change in program detail.

Probably the most significant uncertainty in the program schedule is whether the zoned core or medium core will be the first to be installed in FARET. The basis for this uncertainty is the development work and physics calculations associated with the respective cores. At best, the only solution is to estimate the design and fabrication times for both cores and work backwards from the core-installation date to the latest time for start of design. A decision must be made as to which core to fabricate, and this decision would be based on the status of the development program at that time.

Both medium and zoned cores utilize some of the same metallurgical technology that centers around the fabrication of the oxide-type fuel elements. The zoned core also uses metallic fuel in the driver zone, but the technology in this area is in a more advanced state. It is most probable that the research and development work that is being carried on in a parallel effort will be completed at the same time. At that time a decision must be made on the core to fabricate: the zoned or the medium core. Whichever core is chosen for the first core will allow more time on the other core for further study of flow effects, material testing, and instrumentation development. If the second core is the medium core, there may be sufficient time to develop a carbide core instead of the present oxide core.

A partial breakdown of activities is shown in Fig. 16. This breakdown illustrates the interdependency of the medium and zoned cores with the date indicated for the necessary decision on which core fabrication is to begin in order to meet the target completion date for the project. The estimated completion dates are computed from the duration time assigned to each activity. The duration times are shown in parenthesis above each activity.

The preliminary analysis of activity duration times indicated on Fig. 16 has shown that the zoned core has the shortest time duration to completion. This is why the zoned core is shown as the first core to be installed in the FARET facility.

The difference between the zoned and medium cores lies in the estimate of the fuel-fabrication time for the respective cores. There is a ten-week difference in fabrication time in favor of the zoned core, which makes the latter most likely to be completed on schedule. However, all of the research and development on both cores is performed on a parallel effort basis of the same activity time duration, and only after the completion of this work

Fig. 16. PERT Schedule Experimental Program

effort can a reasonable decision be made on which core to begin fabrication. The target date for review of both cores has been set for May 1964. There is always a possibility that certain advances in technology could foreshorten this date.

All of the activity duration times are estimates made by the scientists directly associated with the research and development activity. The target date for installation of the first core is January 1966. It is expected that the 8- to 12-month period after installation of the first core will be consumed by shakedown operation of the facility and critical experiments. Another 6 to 10 months is allocated for experimentation with the zoned core. This brings the target date of installation of the second core to July 1967. One and one-half years is estimated for installation and experiments with the medium core. The installation of the small core is scheduled for January 1969. This almost coincides with the estimated time required to complete the research and development and fabrication of the small, high-performance core which has been indicated for March 1969.

These dates represent only the effects due to the FARET program group efforts and do not show the effect of the facility design and construction. None of these dates are firm, and continual analysis of the network diagram during breakdown of the respective activities may indicate a method for accelerating the test program.

At present, work is progressing on the next level of work breakdown. After this is completed, reviewed by management, and approved, the work areas will be broken down further into as many subactivities as considered necessary for good management planning and control.

It is also expected that an attempt will be made to utilize the PRAM method of leveling manpower availability. This will aid in placing the manpower effort in the most critical areas to insure maintenance of the project schedule.

In order that delays not be incurred by late delivery of major items by vendors, it has been suggested that vendors be required to submit some sort of PERT diagram of their activities. This is believed to be necessary in order to maintain some measure of control over items of long-term delivery.

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