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FARET CORE I FUEL IRRADIATION PROGRAM
AND REFERENCE DESIGN

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I. INTRODUCTION

The primary objective of the first experiment in the Fast Reactor Test Facility (FARET)¹ is the testing and verification of fast-reactor fuel elements typical of those likely to be used in the large, fast, prototype power reactor which is part of the AEC's fast reactor development program.

Since the core and fuel specifications must be developed for FARET prior to the design of the prototype reactor, the FARET Core I design and operating conditions are being planned so that they can bracket the corresponding design specifications of the power reactor. Core I will test several different fuel-element designs under instrumented conditions, in order to provide the maximum of information which can be used to evaluate, define, or verify operating limits for the first core of the prototype reactor.

FARET Core I contains oxide, carbide, and metal fuel subassemblies driving each other. The several different designs of oxide, carbide, and metal reflect current judgment on the most likely operating conditions for the prototype. Specific design choices as to clad material, fuel density, clad temperatures, etc., have been made for the reference core. The research and development program for FARET Core I will test and evaluate these design choices, as well as some alternate design possibilities, prior to final specification of the core.

It is planned to study the performance of oxide, carbide, and metal fuel subassemblies simultaneously in a small, 70-liter core with a power-density distribution that will allow up to 15% heavy atom burnup in about 3 years of operation. With the small core, the design and the geometry of the prototype reactor can be simulated, but the fuel enrichment will be different. The Pu^{239} content is made similar to that of the large reactor by the replacement of U^{238} with U^{235} to achieve criticality.² Therefore, by selection of reactor position and/or enrichment in U^{235} , performance ratings will be achieved that bracket the anticipated operating conditions appropriate to the prototype power reactor.

The basis for utilizing this enrichment technique is that the substitution of U^{235} for U^{238} would not alter the metallurgical or physical properties of the fuels, at least to a first-order approximation. The U^{235} -enriched, uranium-plutonium system is chemically similar to a nonenriched, uranium-plutonium system.

This technique is compared with the method in which a nonenriched, uranium-plutonium system is irradiated in a higher-neutron-flux environment to achieve the same rate of heavy atom burnup. The fission-reaction rate, which is related to the index of fuel performance (burnup of heavy atoms) is the parameter common to both techniques. In one method, a given product of the neutron flux and macroscopic fission cross section is achieved by varying the atomic density of the fissile material; the other method requires the production of an external source of a higher neutron flux.

Phenomena such as diffusion, or plutonium segregation, which depend on force fields arising from temperature gradients, are expected to differ, if at all, only to second order. The reasons for this are as follows: First, for a given fission-reaction rate, the fission distribution, and hence the heat generation (and temperature gradient), in a fuel element should be the same for both systems. This is the case in a fast neutron environment since the fast flux would have a flat distribution in a small-diameter (0.5- to 1.0-cm) fuel pin. Second, in the case of the diffusion of atoms, a difference in the diffusion coefficient would imply an isotopic mass dependence.³ Because of the heavy masses involved, the diffusion coefficient and the resulting forcing function for the two mixed fuels is expected to differ only in second order.

The dimensional stability of a fuel pin and the design of the cladding to withstand internal gas pressures are dependent on the generation of gaseous fission products. A study of the fission yields from U^{235} and Pu^{239} for thermal neutrons⁴ and of estimates for fast neutrons⁵ indicates that the total yield of the more prevalent gaseous fission products (krypton, xenon, iodine, and cesium) are similar to within 10%. When the yields for a mixed, U^{235} -enriched, uranium-plutonium fuel are compared with a nonenriched, uranium-plutonium mixture, the difference becomes much less than 10%. For a total uranium-to-plutonium atom ratio of about 5/1, a 50 to 60% U^{235} enrichment, and approximating a U^{235} -to- Pu^{239} fissioning ratio of 1/2, the difference in the yields would be about 6%.

A difference in the fission-gas release rates and fission-product migration might exist if a fuel is used in which the plutonium is not in solid solution with the uranium. The existence and the extent of these differences have not yet been established. Even if these differences are observed, the effect is expected to become minimal at the high operating fuel temperatures of FARET and larger power reactors.

Therefore, the performance ratings of fuel-element designs, obtained by using the enrichment technique as in FARET, can be assigned to prototype reactor elements with a good degree of confidence.

If it becomes necessary to establish this correlation to a higher-order approximation, one or more control subassemblies can be readily included at any stage of the experimental program.

The advantage of the above technique is that many different fuel-element designs may be tested simultaneously in a flexible and accessible small-core loading.

The hard neutron spectrum and the magnitude of the flux in FARET would subject the clad material to neutron dose rates comparable to those that would be obtained in the typical softer spectrum of a large, fast, power reactor at the same fuel ratings. The frequency of adding or replacing fuel elements in order to maintain criticality is determined only by the experimental requirements of these same test elements. The fuel inventory is determined only by the number of test assemblies required in the experiment. The flexibility in core size allows the generation of experimental data at accelerated rates of heavy atom burnup. Along with these low fuel loadings is the advantage of needing only a low heat removal system.

The FARET Core I reference design assumes that as the subassemblies are removed for inspection and destructive testing, they will be replaced with fresh subassemblies of the same design. This need not be the case. Improved fuel-element designs resulting from the continuing fast-reactor fuels-development program can be inserted into the core whenever available, replacing types showing a reduced performance capability. Further, should some originally fabricated designs prove to be of less interest in the early stages of the FARET experiment, they would be replaced by additional subassemblies of other designs having greater potential. Extra subassemblies of several types will be provided in the original fuel inventory. The physics of the core permits such flexibility.

II. REFERENCE DESIGN, CORE I

A. Basic Core Plan

The FARET Core I reference design will consist of carbide, metal, and oxide fuels and will test simultaneously ten different fuel-element designs with good statistics. The Core I data are presented in Table I. Two of the oxide designs will be run at two different linear heat-power ratings, and hence different maximum cladding temperature. The fuel-element types and designs and the tentative conditions are presented in Tables II and III. The elements are typed according to fuel material, fabrication method, clad material, and/or thermal conditions.

TABLE I. Core I Data

<u>Entire core</u>		<u>Radial reflector</u>	
Equivalent diameter, in.	17.5	Height, in.	80
Height, in.	18	Equivalent outside diameter, in.	42.3
Volume, liters	70	Radial thickness, in.	13.1
<u>Subassemblies</u>		<u>Radial reflector composition, v/o</u>	
Mixed oxide	24	Stainless steel	82
Mixed carbide	16	Sodium	18
Pu, U alloy	8	<u>Heat output, MW</u>	
Special subassemblies	3	50	
Control rods	12	<u>Primary sodium temperature, °F</u>	
Radial reflector (including loops)	238	To core	810
Total	301	From core (nominal)	1050
Configuration	Hexagonal	<u>Primary sodium flow rate</u>	
Dimension across flats, in.	2.290	Through core, gal/min	6000
Lattice spacing, in.	2.320	Core pressure drop, psi	80
Tube-wall thickness, in.	0.040		
<u>Control-rod composition, v/o</u>			
Boron carbide	55		
Stainless steel	20		
Sodium	25		

TABLE II. Fuel-element Data

Fuel Material	UO ₂ -15 w/o PuO ₂	UC-15 w/o PuC	Metal ^b
Elements per subassembly	37	37	61
Fuel height, in.	18	18	18
Clad outside diam, in.	0.290	0.290	0.220
Clad thickness, in.	0.020	0.020 ^a	0.016
Spacer wire diam, in.	0.057	0.057	0.052
Axial blanket height (each end), in.	6	6	6
Gas reservoir, in.	25	25	25
Element height, in.	55	55	55
Effective fuel density, %	80	80	70
Bond material	Helium	Helium	Sodium
Fuel U/Pu atom ratio	6/1	6/1	5/1
Fuel details	Coprecipitated; O/M 1.97-2.00	Single Phase; Solid Solution	-

^a0.015 in C4 subassemblies.^bU-15 w/o Pu-6.5 w/o Ti; U-15 w/o Pu-10 w/o Zr.

TABLE III. Subassembly Data

Fuel Material	Subassembly Designation	Fuel Type	Clad Material	No. of Subassemblies	Instrumented Subassembly	Maximum Linear Heat Flux (Design), kW/ft	Maximum Clad Temp, °F (°C)	Maximum Fuel Temp, °F (°C)	Burnups* at Removal, a/o	Maximum Clad Thermal Stress, psi	Coolant Velocity, ft/sec
Oxide	O1	Pellets	304	4	(1)	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O2	Pellets	304	4	(1)	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O3	Vipak	304	4	(1)	15	1250 (675)	3220 (1770)	5, 9, 12, 14	13,300	16
	O4	Vipak	304	4	(1)	22	1290 (700)	4220 (2330)	5, 8, 10, 12	19,400	23
	O5	Pellets	HSA	4	(1)	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
	O6	Vipak	HSA	4	(1)	22	1340 (725)	4250 (2340)	5, 9, 12, 15	-	20
Carbide	C1	Pellets	304	4	(1)	15	1250 (675)	2030 (1110)	5, 9, 12, 14	13,300	16
	C2	Vipak	304	4	(1)	22	1290 (700)	2520 (1380)	5, 9, 12, 14	19,400	23
	C3	Pellets	HSA	4	(1)	29	1340 (725)	3050 (1680)	5, 9, 12, 15	-	30
	C4	Vipak	HSA	4	(1)	36	1340 (725)	3390 (1870)	5, 9, 12, 15	-	39
Metal	M1	U-Pu-Zr	HSA	4	(1)	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
	M2	U-Pu-Ti	V-Ti	4	(1)	14	1250 (675)	1580 (860)	5, 9, 12, 14	-	20
Advanced	C4S	-	-	3	(3)						
				51	(15)						

NOTES: 1. Vipak means "vibratory compacted;" HSA means "high-strength alloy," not necessarily the same for oxide as for carbide, etc.

2. Coolant inlet temperature is 810°F. Average outlet temperature from each subassembly is 1050°F. (Exception: outlet temperature from O5 and O6 subassemblies is 1085.) Axial peaking factor = 1.2.

3. Maximum clad temperature include both a 1.33 factor for flow and power maldistribution effects on coolant temperature rise, and a 1.10 uncertainty factor. Maximum fuel temperatures include the 1.33 factor but no uncertainty factor.

*Approximate burnup intervals.

The reference core contains six mixed-oxide types, for a total of 24 oxide subassemblies; four mixed-carbide types, for a total of 16 carbide subassemblies; two metal alloy types for a total of eight subassemblies; and three subassemblies for special experiments, making an overall total of 51 (see Table III). Two loops at the core edge and one loop in the reflector are also planned; details of the experimental program in the loops and the special subassemblies will be provided later.

Each subassembly type is to be tested up to about 15 heavy a/o burnup, if the design permits. The elements are to be destructively tested at predetermined subintervals of burnup. The suggested periods reflect the current confidence of the metallurgists that most of these elements can achieve at least the first two burnup periods. Also reflected in the frequency of inspection is the attempt to maximize reactor full-power operating time. As will be subsequently discussed, other important constraints on the frequency of testing and irradiation intervals are the maintaining of criticality and the worth of the control system.

B. Thermal Considerations

An average coolant outlet temperature from each subassembly of 1050°F has been chosen because it is believed that reactor plant temperature of that order will be of interest for some years to come, due to overall economic considerations. In those subassemblies in which the high 1.33 factor applied to the average coolant-temperature rise proves to be too large, the average outlet temperature will be raised to produce the maximum clad temperatures given in Table III.

With regard to the maximum fuel temperatures, uncertainty factors are not included in the fuel temperatures given in Table III. It is believed that a factor of not more than 1.3 will eventually prove to be appropriate. (This uncertainty factor is applied to the difference between the maximum fuel temperature and the coolant inlet temperature.) If the factor is 1.3, however, the maximum oxide temperature becomes 2910°C and the maximum metal temperature becomes 990°C. Such temperatures appear to be high. If the uncertainty factor proves to be as high as 1.3, the power in some of the oxide and metal fuel elements may have to be reduced. In that event, the coolant outlet temperatures can be increased to keep the maximum clad temperatures as high as those in Table III. Table IV lists the assumptions used in the thermal calculations.

TABLE IV. Assumptions for Thermal Calculations

Fuel Material	Sodium Film Coefficient, Btu hr ⁻¹ ft ⁻² °F ⁻¹	Clad Thermal Conductivity, Btu hr ⁻¹ ft ⁻¹ °F ⁻¹	Fuel-clad Interface Conductance, Btu hr ⁻¹ ft ⁻² °F ⁻¹	Fuel Thermal Conductivity, Btu hr ⁻¹ ft ⁻¹ °F ⁻¹
Oxide	20,000	13.5	1,500	1.7 ^a
Carbide	20,000	13.5	1,500	10
Metal	20,000	14.5	40,000 ^b	9 ^b

^aThe fuel is assumed to have a central hole large enough to produce an "effective" fuel density of 85%, provided that the remainder of the jacket interior is filled with fuel of 100% density. This condition reduces the temperature rise through the fuel to 67% of that through a solid fuel pellet. The central hole is expected to form in oxide at centerline temperatures above 1600-1800°C.

^bThe fuel is considered to have expanded against the jacket, displacing the bond sodium, and to have an accompanying reduction in thermal conductivity.

C. Instrumented Fuel Subassemblies and Fuel-element Design

The design of the reference core subassembly selected for FARET employs the EBR-II hexagonal shape. The core is constructed using a close packed geometry (for dimensions see Table I) of 301 total subassemblies. There are 51 fuel elements and 12 control rods, the remaining locations being occupied by the loop and the stainless-steel reflectors. A conceptual instrumented fuel subassembly is illustrated in Fig. 1.

The design of the fuel-element fission-gas reservoir assumed an estimated fission-gas generation of 2.4 and 2.9 cc/(cc of fuel) (a/o burnup) for the oxide and carbide fuels, respectively. It was further assumed that a 100% fission-gas release could occur in the oxide fuel and that a 75% fission-gas release would occur in the carbide fuel. The highest clad stress, and hence the most pessimistic case, occurs in the oxide fuel elements with 14% burnup, 304 SS clad, and a maximum clad temperature of 675°C. The maximum temperature of the gas reservoir is 610°C. Under these conditions, the pressure stress in the cladding is about 6000 psi. Recent data indicate that a stress of 8000 psi will produce 1% creep in 10,000 hr in Type 304 stainless steel at 675°C. Therefore, from the point of view of cladding stress, it appears that the burnup goals listed in Table III can be realizable in FARET.

The approach adopted in the design of the fuel elements is toward a thick-wall cladding. This is better able to withstand fission-gas and fuel-swelling pressures than a thin-wall cladding. The thick-wall cladding will be subjected to higher thermal stresses than thin-wall cladding, but Case 1331 of the Nuclear Vessels Code indicates that many thermal stress cycles are permissible at the stress levels anticipated. On this basis, it then appears desirable to concentrate on reducing pressure stresses rather than thermal stresses.

D. Core Composition and Configuration

The core composition and configuration designed to approach the conditions outlined above are presented in Fig. 2 and Table V. To achieve the desired linear heat fluxes, the uranium enrichment and subassembly positions were varied. A condition imposed on the obvious flexibility of enrichment was that the uranium enrichment be specified and held fixed for each specific technique of fuel fabrication. Thus, the C1 and C3 pellets have the same enrichment, although the clad materials are different.

Several conditions were considered in the positioning of the subassemblies. Symmetry was maintained as closely as possible to simplify analysis and to subject the several elements of each type to similar power conditions throughout the experiment.

Fig. 1. Instrumented Core Subassembly

5TH IRRADIATION INTERVAL

6TH IRRADIATION INTERVAL

+

7TH IRRADIATION INTERVAL

S8

S7

hS

hS

+

8TH IRRADIATION INTERVAL

TABLE V. Performance Parameters of FARET Core I

Reactor Power: 50 MW(t)
 Core Composition (v/o): Ceramic Fuel 37/Sodium 40/Structure 22
 U/Pu Atom Ratio 6/1
 Metallic Fuel 30/Sodium 48/Structure 22
 U/Pu Atom Ratio 5/1
 Core Volume: 70 liters; Height: 45.7 cm (18 in.)

Row and No. of Subassembly	Fuel Type	Approx. ^{235}U Enriched	No. of Elements per Subassembly, (diam, in.)	Average Power per Subassembly, MW	Power Density per Subassembly, MW/liter	Avg. (Max) Linear Power, kW/ft	% Heavy Atom Burnup (180 days at Full Power)
(1) 1	C4	0.67	37 (0.29)	1.5	1.1	27 (31)	5
(2) 6	C4	0.67	37	1.45	1.06	26 (31)	4.9
(3) 4	O4	0.60	37	0.88	0.64	15 (19)	3.9
4	O6	0.60	37	0.88	0.64	15 (20)	3.9
4	C3	0.60	37	1.18	0.88	21 (27)	3.9
(4) 4	O2	0.65	37	0.84	0.61	15 (20)	3.6
4	O5	0.65	37	0.84	0.61	15 (20)	3.6
4	C2	0.67	37	1.01	0.73	18 (23)	3.6
4	M1	0.57	61 (0.22)	0.97	0.71	10 (13)	2.7
2	M2	0.57	61	1.0	0.73	10 (13)	3
(5) 4	O1	0.65	37 (0.29)	0.72	0.53	13 (16)	3.3
4	O3	0.60	37	0.69	0.50	12 (15)	3.2
4	C1	0.60	37	0.85	0.62	15 (19)	2.7
2	M2	0.57	61 (0.22)	0.84	0.61	9 (12)	2.4

The positioning was also influenced by the sequence of removal and replacement. Instrumented subassemblies are usually located so that the loading and unloading of other subassemblies are not made cumbersome by having instrumentation leads located over subassemblies to be removed. It is planned to remove the instrumented subassemblies in the latter stages of burnup for each type element.

This makes possible the continuous recording of fuel and coolant temperatures, and possibly of coolant flow rates, neutron flux, and fission-gas release throughout most of the core life of each fuel-element design.

A preliminary placement of the instrumented subassemblies and the manner in which the cables may be laid out are shown in the overlays in Fig. 2. The sequence of removal coincides with the level of the overlays. When several assemblies are to be removed at the end of a period, the pattern will be to remove the innermost first and then continue radially outward.

If instrumented subassemblies cannot be placed adjacent to each other as shown in Fig. 2, an alternate arrangement will be made with at least one uninstrumented subassembly separating each instrumented one.

Several alternate layouts are possible. For example, the C45I*, C35, and C27I subassemblies can be interchanged with the subassemblies C41, C34, and C22. As another example of this flexibility in the placement

*Carbide subassembly, Type No. 4, removal at end of fifth irradiation period, instrumented.

of instrumented subassemblies, the O37I and M18I assemblies can be interchanged with the O34 and M17 subassemblies, either separately or as a pair.

This flexibility will allow utilizing all the vessel penetrations (or nozzles) and will thus avoid congesting any single outlet. The penetrations S1, S2, and S3 are planned to be used exclusively for the core-edge loops and the reflector loop.

The overlays include several possible layouts for removal of the special C4S subassemblies. Optional instrumentation schemes for instrumenting the special subassemblies have been indicated, depending on the time of removal. Less leads are required than shown, as a consequence. The final arrangement of the instrumented subassemblies will depend on the capability of handling connectors, cables, and other related functions inside the reactor.

The advanced types of subassemblies referred to in Table II have not been specified. They are presently designated as C4S.

The radial power distribution within a subassembly varies about 10% from the average (see Fig. 3). With the axial maximum-to-average ratio of 1.16, the overall maximum-to-average power density is about 1.28. This factor was used in the approach to the design heat fluxes listed in Table III. The uranium enrichments also include the effect of control rods on the power-density distribution. Partially inserted control rods could reduce the average power distribution in the outer rows by 10%. Therefore, the subassemblies in the outer row, when all factors are applied, approach the design heat flux.

Fig. 3. Relative Radial Power Density Distribution at Core I Center Plane

dimensional computations indicate that a 5% variation in the power densities can be expected between these two positions.

The power density in the carbide subassembly C1 may be considered too high. If this is the case, the C1 elements may be moved into the "corner" positions of row 5. Two-

E. Considerations in Planning the Experimental Program

In planning the experimental program, an important factor in establishing the fuel removal and replacement sequence (see Table VI) was the

total worth of the control system. The optimum experiment with respect to time, would be one in which the excess reactivity needed for the maximum per cent burnup is accommodated by the initial fuel enrichment. This would minimize the number of fueled assemblies required in the outer core region to maintain criticality. The addition of these assemblies reduces the overall power density and thereby lengthens the time to attain a given burnup. The approach to the optimum experiment is to design the control system with maximum reactivity worth.

TABLE VI. Calculated Nuclear and Control Parameters
for FARET Core I Reference Design

<u>Nuclear Data</u>			
Neutron Flux, $n/cm^2\text{-sec}$			
Energy Range	Radial Position		
	Core Center	Core Edge Loop	Reflector Loop
>1.35 MeV	0.8×10^{15}	0.3×10^{15}	0.3×10^{14}
100 keV-1.35 MeV	2.4×10^{15}	1.5×10^{15}	5.7×10^{14}
1-100 keV	0.4×10^{15}	0.5×10^{15}	3.9×10^{14}
<1 keV	6.5×10^{11}	1.8×10^{13}	1.2×10^{14}
Total	3.6×10^{15}	2.3×10^{15}	1.1×10^{15}
Prompt neutron lifetime, μsec			0.21
Effective delayed neutron fraction, β_{eff}			0.0058
<u>Reactor Control</u> Average Reactivity Coefficients, $\Delta k/\Delta T^\circ\text{C}$			
Overall Doppler @ 750°K	Uranium		-0.45×10^{-6}
	Plutonium		$<+0.1 \times 10^{-6}$
Radial core expansion			-10×10^{-6}
Overall axial fuel expansion			-4×10^{-6}
Sodium expansion coefficient			
Overall core region avg			-10×10^{-6}
Core and reflector			-14×10^{-6}
<u>Estimated Range of Total Worth of Control Rods</u>			
Composition (v/o): B ₄ C 55/coolant 25/structure 20			
No. of rods			12
70% B ¹⁰ enrichment			0.075-0.085
95% B ¹⁰ enrichment			0.085-0.10
<u>Excess and Shutdown Reactivity</u>			
Heavy atom burnup			0.035-0.060*
Operational temperature			
Sodium $\Delta T_{\text{avg}} = 500^\circ\text{C}$ core and reflector	0.007		
Radial core expansion $\Delta T_{\text{avg}} = 500^\circ\text{C}$	0.005		
Axial fuel expansion $\Delta T_{\text{avg}} = 1500^\circ\text{C}$ oxide	0.004		
$\Delta T_{\text{avg}} = 1100^\circ\text{C}$ carbide			
$\Delta T_{\text{avg}} = 500^\circ\text{C}$ metal			
Total operational temperature			0.016
Excess reactivity			0.051-0.076
Shutdown reactivity			0.020
Design worth of control system			0.071-0.096

*Depending on manner of compensating for burnup.

Table VI lists the calculated nuclear and control parameters for the reference core. Computations, along with related ZPR-3 critical experiments (see Section F below) indicate that a 10% reactivity control system may be possible to attain, using 12 highly enriched boron-carbide rods. Also listed in Table VI are the reactivity effects due to the expansion of coolant and fuel system for the anticipated operating temperature changes. The total excess reactivity for the operational temperature change is estimated to be 1.6%. Using a range of excess reactivity from 3.5 to 6.0% for burnup, along with a 2% shutdown margin and the 1.6% margin for the operational swing, the required range of control reactivity, 7.1 to 9.6%, is within the estimated range of the total worth of the control rods.

F. FARET Critical Experiments

A series of FARET critical experiments was completed to confirm the physics analysis of multifueled, nonuniform core loadings. A primary interest was to establish experimentally the predicted control-rod reactivity worths of FARET core systems. Of equal importance was to substantiate the neutronic behavior of a mixed core when subassemblies of different type fuels are interchanged or replaced. Other measurements included the distributed worths of fuel and structural materials, reaction-rate traverses, and Rossi-alpha measurements.

1. Control-rod Experimental Results

The preliminary results of the experiments have been published in a series of ANL Reactor Development Program progress reports.⁶ Some analytical results for the reactivity worths of 12 control rods in the 84-liter mixed oxide core are compared with the experimental values in Table VII. The reactivity effects were studied for two B¹⁰ enrichments of the boron-carbide control rods. The agreement between the computations and experiments establishes the range of control possible in the FARET reactor.

TABLE VII. Control-rod Worths for
FARET Critical Experiments on Oxide Loading

B ¹⁰ Enrichment	30%		70%	
	In hours	Δk	In hours	Δk
Experiment	-2544	0.051	-3897	0.077
Calculation	-2450	0.049	-3800	0.076

2. Establishing Concept of Mixed Core Systems

From Table VIII, the experimental data on the interchange of differently fueled assemblies in the oxide and carbide core loadings can

be predicted to within 10%. In the oxide core, the row indicated by 5/6 means that the fueled assembly location was better represented by placing the assembly partly in the fifth and sixth rows of the matrix in the critical facility.

TABLE VIII. Subassembly Replacements in the
FARET Critical Experiment

Replacement (1.5 liters put in)	Row	Reactivity Change, inhours	
		Experiment	Calculations
<u>A. Oxide Core (84 liters), 503 inhours = 1% Δk</u>			
Carbide for oxide	1	+258	+297
Carbide for oxide	3	+204	+239
Carbide for oxide	5	+110	+122
Carbide for reflector	6	+254	+242
Carbide for reflector	5/6	+348	+335
Oxide for reflector	6	+180	+166
<u>B. Carbide Core (68 liters), 545 inhours = 1% Δk</u>			
Oxide for carbide	1	-173	-156
	2	-161	-147
	3	-128	-112
	4	-81	-76
5 oxide assemblies in row 3	3	-618	-583

Pulsed neutron measurements made on the carbide and oxide core, were in reasonable agreement with the computed alpha (ratio of delayed-neutron fraction to neutron lifetime, β/ℓ) value of 2.0×10^4 and $2.7 \times 10^4 \text{ sec}^{-1}$ for the two cores, respectively.

The significance to be attached to the agreement between the experiment and the preliminary analysis, is that the behavior of a mixed core loading can be understood as well as that of a uniform core.

The maximum allowable excess reactivity for burnup studies in FARET Core I, will be determined from the critical experiments on the mockup of the core. The experiments have been planned to immediately follow the final fuel specifications.

G. The Experimental Program

The uranium enrichments quoted in this report include the 6% excess reactivity. If less than 10% total control worth is used, excess reactivity can

be introduced by the addition of fueled assemblies at the edge of the core, either in the initial loading or after the first burnup interval.

The atom per cent burnup as a function of full-power operating days is presented in Table IX and in Fig. 4. The estimated atom per cent burnup as a function of time is plotted for the 12 types of fuel elements under study. The types of subassemblies to be removed after each irradiation period are indicated by circles in Fig. 4. There are only two metal M2 assemblies in row 4, so that after 9 a/o burnup, the burnup experiment for this type of fuel is to be followed with the M2 assemblies in row 5. After 780 days of full-power operation, all but three types will have exceeded 14 a/o burnup (see Table IX). To expedite burnup of these types late in Core I life, the metal fuels M1 and M2 and the carbide fuel C1 could be moved to the third row enabling the additional 3 or 4 a/o burnup to be achieved during an additional period of 110 days. It is intended that the linear power in these elements be maintained as close as possible to the initial power rating.

TABLE IX. Experimental Program Removal and Replacement Sequence
(Heavy Atom Burnup versus Days at Full Power)

Row and Number of Subassemblies	Irradiation Period (days) a/o Burnup	Fuel Type	1	2	3	4	5	6	7	8
			(180)	(240)	(305)	(430)	(550)	(690)	(780)	(890)
(1) 1		C4								
(2) 6		C4	5		8.5	12	15.2			
(3) 4		O4		5.2		9.5	12	15		
4		O6	3.9			9.5	12	15		
4		C3		5.2		9.5	12	15		
(4) 4		O2	3.6			9	11	14.3		
4		O5		4.8		9	11	14.3		
4		C2		4.4		8		12.5	14.3	
4		M1			4.6		8.3		12	15
2		M2			5		9			
(5) 4		O1		4.4		8		12.5	14.3	
4		O3		4.4		8		12.5	14.3	
4		C1			4.6		8.3		12	15
2		M2							10.5	14
No. of subassemblies removed			3	6	4	9	9	8	6	3
No. of subassemblies replaced			3	6	4	15 ^(a)	9	8	6 ^(b)	
No. of subassemblies added in row 5					2					

(a) Replace six C4 subassemblies or others as required to maintain desired reactivity for burnup beyond 10 a/o.

(b) Irradiated subassemblies M1, M2, and C1 remaining may be moved into row 3.

Fig. 4. Per Cent Heavy Atom Burnup versus Full-power Operating Days

The irradiation periods and the number of subassemblies removed and replaced are included in Table IX. The carbide C4 fuel assembly is expected to reach a 5 a/o burnup in 180 full-power operating days. At this time, two of the higher-performance oxide assemblies would also be removed for investigation. These two types of subassemblies would then be allowed to go directly to about 9 a/o burnup, should the investigation show satisfactory performance. An intermediate step, in which the other type oxides have burned to about 5 a/o, would require some 60 additional days. As shown in Fig. 5, which shows k_{eff} versus operating days, after the

Fig. 5. Variation of Reactivity versus Fuel-irradiation Periods (Days) at Full Power

third irradiation period (305 operating days), a reactivity loss of 5% is to be expected. If, at this time, two additional fuel assemblies are added in the fifth row (total worth of 1%), the reactivity loss could be reduced to 3.2%. The excess reactivity would now allow an additional irradiation period of 125 days, at the end of which most of the fuel types would have incurred between 8 and 12 a/o burnup. Included in Fig. 5 is an alternate plan which could be adopted for the case in which a 5% excess reactivity was allowed for the burnup experiment. The control system would have a total reactivity worth of 9%. The only difference in the removal and replacement sequence is the addition of two fresh assemblies in the first replacement step. The burnup interval became somewhat longer so that the 8 to 12 a/o burnup occurs after 465 days.

At the end of a 430-day operating period at full power (1.5 years, assuming a plant factor of 0.75), the performance of close to 1000 fuel elements (22 subassemblies) would have been investigated at advanced thermal and burnup conditions. During this irradiation period, the fuel-pin cladding material would have been subjected to a total neutron exposure ranging from 0.9 to 1.4×10^{23} nvt. The range of dosages resulting from neutrons above the (n, p) and (n, α) threshold energies would have been as follows: 1.1 to 3×10^{22} nvt for energies above 1.35 MeV, and 2 to 4.6×10^{22} nvt for energies above 1 MeV. These threshold reactions may prove to be a main factor in limiting the life of the fuel cladding. In this case, then, the hard neutron spectrum in FARET would subject the clad material to a dose rate comparable to that attained in the softer spectrum of a large power reactor having a total flux of 1×10^{16} neutrons/cm²-sec. For example, in some of the large power-reactor systems studied, the fraction of neutrons above the energy of 1.35 MeV ranges from about 0.08 to 0.10 for ceramic and metal-fueled elements, respectively. This should be compared with the fraction of 0.22 for the FARET core (see Table VI).

The performance limits of a fuel-element design, including the effect of radiation on the clad, as determined in the FARET experiment for a given fuel rating, are then directly applicable to a fuel-element design of a large power system.

To go to burnups of 15 a/o, the necessary reactivity could be obtained by replacing some partly burned-up subassemblies with new subassemblies of the same design or of a new design. The choice could depend on the fact that the replacement is of a type that has shown a greater dimensional stability than the others or has had a better performance history. If needed, a few additional subassemblies could be added at the core edge. The sequence of removal and replacement of subassemblies could continue until all types, or all those that have performed satisfactorily throughout the course of the experiment, reach a burnup of 15 heavy a/o.

The reactor downtime due to removal and replacement of the sub-assemblies will be maintained at a minimum, depending on the degree of confidence (obtained from previous research and development and from inspection of removed subassemblies) that the fuels will perform satisfactorily. The estimate of the time required for the usual loading and unloading operation is 2 weeks. Using an average plant factor of 75% over the duration of this experiment, the time to complete the program as outlined is expected to be 3 years.

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