POTENTIAL FFTF CARBIDE FUELS

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ABSTRACT

An ad hoc working group comprising knowledgeable fuel experts in the United States was convened at Pacific Northwest Laboratory (PNL) also known as Battelle-Northwest (BNW) to assist PNL in evaluating the potential of UC-PuC or UC driver fuels for the Fast Test Reactor (FTR), an integral part of the Fast Flux Test Facility (FFTF). The members assembled at PNL during June 27 - July 1, 1966, to delineate detailed research and development programs that would be required to assure adequate carbide fuel elements at the time of the first FTR core loading. Reactor core designs and fuel cycle parameters were investigated with a PNL adaptation of the MIDAS computer code and various carbide fuel combinations and operating conditions. Analyses of the various carbide fueled FTR's demonstrated that UC-PuC offers the possibility of an excellent FTR driver fuel from several aspects including cycle cost, flux, performance, fabrication requirements, and safety.

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INTRODUCTION

An ad hoc working group, comprising knowledgeable fuel experts in the United States, was convened at Pacific Northwest Laboratory (PNL), also known as Battelle-Northwest (BNW), to assist PNL in evaluating the potential of UC-PuC and UC driver fuels⁽¹⁾ for the Fast Test Reactor (FTR), an integral part of the Fast Flux Test Facility (FFTF).⁽²⁻⁷⁾ This report documents the work performed and includes further computer data generated after adjournment.

The participants contributed their experience in fuel evaluation; in fuel element design, fabrication, and testing; in research and development cost scheduling; and delineation of detailed research and development (R&D) programs needed to assure an adequate carbide fuel element in time for the first FTR core loading. The meetings took place in the Federal Building in Richland, Washington, during the week of June 27 - July 1, 1966. Those participating were:

W. J. Bailey	Battelle-Northwest	(BNW)
P. D. Cohn	Battelle-Northwest	(BNW)
F. M. Heck	Westinghouse Electric Corporation, Atomic Power Division	(WAPD)
J. H. Kittel	Argonne National Laboratory	(ANL)
J. H. Sako	United States Atomic Energy Commission	(AEC-RL00)
A. A. Strasser	United Nuclear Corporation	(UNC)
C. W. Wheelock	Atomics International	(AI)

Also, D. R. de Halas (BNW), W. E. Roake (BNW), and R. J. Myjak (AEC-RL00) were present for limited times.

The first 1 1/2 days was spent determining the proper input data for the analog computer runs. An updated version of the MIDAS computer code application⁽³⁾, developed by BNW for both the design analog and the economic evaluation and fuel cycle studies, was employed. These data included economic and technical restraints such as pin manufacturing cost, reprocessing and conversion costs, maximum allowable fuel centerline temperature, maximum allowable fuel surface temperature, fuel bulk (smear) density, overpower factor, and burnup for each fuel case. Two subsequent days were required to discuss and outline the detailed research and development program which would be needed to assure that adequate FTR fuel elements of UC-PuC or UC would be available by June 1, 1971. Following this, about 1 1/2 days were devoted to running specific fuel cases on the BNW analog computer to determine core operating and economic parameters. Items analyzed included fuel operating conditions (maximum fuel centerline and surface temperatures, linear power, etc.), fuel element design characteristics (pin diameter, number of pins per subassembly, etc.), fuel economics (fuel cycle cost, throughput, pin manufacturing cost, etc.), and reactor data (power, critical mass, core volume, flux, etc.).

SUMMARY

The ad hoc working group recommended that the seven fuel cases (Table I) be evaluated for potential use in a carbide-fueled FTR.

TABLE I. *Proposed Fuel Cases*
(all have stainless steel cladding)

<u>Case No.</u>	<u>Fuel</u>	<u>Structural Stoichiometry</u>	<u>Fuel-to-Cladding Bond</u>	<u>Form and Fabrication Method</u>
1	UC	hypostiochiometric to stoichiometric	sodium	cast slugs or high density sintered pellets
2	UC-PuC	hypostoichiometric to stoichiometric	sodium	cast slugs or high density sintered pellets
3	UC-PuC	stoichiometric to slightly hyperstoichiometric	sodium	cast slugs or high density sintered pellets
4	UC-PuC	hyperstoichiometric	helium	high density sintered pellets
5	UC-PuC	hyperstoichiometric	helium	packed powder or low density pellets
6	UC	hyperstoichiometric	helium	packed powder or low density pellets
7	UC-PuC	hyperstoichiometric	helium	high density sintered annular pellets

The current relative status of the fuels, from a known technology viewpoint is as follows:

Best Known	Cases 1, 4*
↓	Case 6
	Case 5
Least Known	Cases 2, 3, 7**

The research and development program proposed for UC-PuC and UC fuels generally covers the period FY-1967 through FY-1972.

* The consensus is that the most accurate predictions can be made for Case 4.

** The consensus is that Case 7 is the most difficult one for which to predict performance.

Manpower (engineers and scientists) required for each program task were estimated for each of the four designated fuel R&D categories* (first category-Case 1; second category-Cases 4, 6, and 7; third category-Case 5; and fourth category-Cases 2 and 3). The total manpower estimates for each category are shown in Table II.

TABLE II. Summary of Fuel Development Manpower Requirements

<u>Fuel R&D Category No.</u>	<u>Fuel Case No.</u>	<u>Engineers and Scientists, man-years</u>
1	1	93.2 - 100.7
2	4, 6, 7	116.0
3	5	117.0
4	2, 3	132.2

For all seven fuel cases, fuel elements for the first core with a burnup capability of 45,000 MWd/tonne (average) could be built by June 1, 1971, if the proposed research and development program is followed. On a short-term basis, the best fuel element possibilities are helium-bonded UC-PuC or UC, and sodium-bonded UC. On a long-term basis, a sodium-bonded, mixed carbide fuel element looks technically and economically attractive.

Serious safety questions may need to be answered if one considers the use of sodium-bonded fuel pins (e.g., what is the effect of sodium expulsion if a closure fails on the lower end of the pin). Equally serious safety questions may need to be answered if one considers the use of gas bonded fuel pins (e.g., if the closure fails on the lower end of the pin what is the effect of high pressure gas being blown into the fuel assembly and consequently voiding the assembly of sodium).

Mixed carbide fuel in the fissile concentrations required for FTR should provide a sufficiently negative value for the Doppler coefficient of reactivity. Complete agreement was not reached on whether or not a moderator would have to be added to the UC-fueled FTR for safety reasons. Physical calculations at PNL

* Not to be confused with technical status listing shown on Page 3.

indicate that with UC fuel, one would need to add some moderator (BeO or other) to obtain a safe system, because of the high fissile fraction. On the other hand the need for using BeO to improve the safety of the ^{235}U -fueled cases is questioned by AI, since they expect the Doppler coefficient to be significantly negative for all cases considered.

With a suitable research and development effort extending beyond June 1971, any of the seven fuel element cases could be commercially fabricated with a potential burnup capability of 100,000 MWd/tonne (maximum).

For the FFTF fuel and core parametric studies, fuel centerline temperatures were limited to the values shown in Table III.

TABLE III. *Fuel Temperature Limits*

<u>Fuel Case No.</u>	<u>Centerline Temp. (a)</u>
1	<2000 °F (<1100 °C)
2	≤1800 °F (≤1000 °C)
3	≤2700 °F, also <3600 °F (≤1500 °C, also <2000 °C)
4	<3600 °F (<2000 °C)
5	<3600 °F (<2000 °C)
6	<3600 °F (<2000 °C)
7	<3600 °F (<2000 °C)

Centerline temperatures were established as the limiting fuel criterion; however, heat flux considerations must be included in any detailed design study.

For the computer studies, the chosen basis for comparison was an 800-liter, mixed-oxide-fueled core. It was assumed that smaller cores could be fitted into the reactor structure. Analog computer results for a carbide-fueled, 600-liter core were

(a) *There was a fairly wide difference of opinion among the group members as to permissible maximum centerline temperatures, allowable burnup at these temperatures, and the use of sublimation or melting as the limiting criterion.*

compared with those for a PuO_2 -SS, cermet-fueled, 600-liter core and the base case, the mixed-oxide-fueled core.

The analog computer studies indicated that FFTF reactor performance with carbide fuel will be nearly equivalent to that achieved with PuO_2 -SS cermet fuel (e.g., 300 to 400 MW with a 600-liter core and a fast flux of 1 to 1.35×10^{16} nv). In addition, the mixed carbide fuels show predicted fuel cycle costs which are roughly equivalent to or less than those predicted for mixed oxide fuel. UC performance was similar to that of mixed oxide, but with higher cycle costs. Use of mixed oxide fuel results in an FFTF with a lower flux ($\sim 0.83 \times 10^{16}$ nv); therefore, to meet the same test space requirements (i.e., ability to provide simultaneously 10 to 13 experimental fuel spaces), an 800-liter core is needed.

The more reasonable performance data for each of the fuel cases considered are in Table IV. The economic parameters for the mixed-oxide-fueled FFTF were reviewed and compared with the parameters for mixed and uranium carbide fuel. The consensus was that the mixed oxide costs used by PNL were quite representative of a typical fuels process using mixed oxide; however, the cost estimates may be low for pin diameters less than 0.25 in. In specific areas the carbide fuel costs were modified to reflect the fuels' differences. The unit reprocessing, conversion, and pin manufacturing costs were raised 15% above that of the oxide to account for the more complex carbide fuel. The mixed carbide costs were reduced by 20% for the uranium carbide case. The rest of the economic data already generated for the mixed oxide were then utilized for the carbide studies. These data will be discussed in detail throughout this report. The effect of the economic data changes was such that, in general for the same performance requirements, the predicted mixed carbide cycle costs are lower than the mixed oxide costs. This is because larger diameter fuel pins can be used with the mixed carbide; thus the

TABLE IV. Summary of the Most Interesting Cases for Each Carbide Fuel Type (a,g)

Carbide Fuel	UC	UC-PuC	UC-PuC	UC-PuC ^(b)	UC-PuC	UC-PuC	UC	UC-PuC
A. Structural Stoichiometry, Fuel Form	hypo. to stoich., c. slugs or h.d.s. pellets	hypo. to stoich., c. slugs or h.d.s. pellets	stoich. to sl. hyper., c. slugs or h.d.s. pellets	stoich. to sl. hyper., c. slugs or h.d.s. pellets	hyper., h.d.s. pellets	hyper., p.powder or l.d.s. pellets	hyper., p.powder or l.d.s. pellets	hyper., h.d.s.a. pellets
B. Fuel-to-Cladding Bond	sodium	sodium	sodium	sodium	helium	helium	helium	helium
C. Linear Power, kW/ft	37.1	31.0	57.6	56.0	26.0	36.4	28.1	28.5
D. Fuel Centerline Temp., ^(c) °F(°C)	1900(1040)	1800(980)	2700(1480)	2700(1480)	3600(1980)	3370(1850)	3800(2090)	3770(2080)
E. Cladding Surface Heat Flux, 10 ⁶ Btu/hr-ft ²	2.3	1.6	3.2	2.9	1.6	1.8	1.7	1.5
F. Fuel Pin Cladding ^(d) OD, in.	0.207	0.246	0.233	0.254	0.210	0.253	0.213	0.246
G. Reactor Power, MW	513(e)	397	398	593(f)	398	398	408	398
H. Throughput, pins/yr	19,910	10,450	5,600	17,890	13,240	9,870	14,980	10,620
I. Centerpoint Flux, 10 ¹⁶ nv	1.0	1.0	1.0	1.49	1.0	1.0	0.8	1.0
J. Predicted Fuel Cycle Cost, \$10 ⁶ /yr	4.99	3.18	2.69	5.33	3.21	3.13	3.72	3.12
K. Identification Case No. (Run no.)	1(2)	2(1)	3(1)	3(5)	4(3)	5(1)	6(2)	7(2)

Notes: (a) Requirements for each core include those for test space (i.e., ability to handle simultaneously 10 to 13 experimental fuel elements in a vertical, 61-tube, skewed array), fast flux (centerpoint flux $\geq 0.7 \times 10^{16}$ nv), and core power (power < 400 MW). Assumption used: average burnup, 45,000 MWd/tonne. Abbreviations used: a. = annular, c. = cast, d. = density, h. = high, l. = low, p. = packed, s. = sintered, and sl. = slightly. For comparison purposes, values for items C-J are: for cermet case; 16.3; 1280(690); 1.1; 0.195; 300; 10,000; 1.0; and 4.2; respectively; for mixed oxide case; 16.9; 4850(2680); 1.0; 0.200; 360; 19,500; 0.83; and 3.4; and for mixed oxide case; 16.6; 4850(2680); 437; 36,000; 1.0; and 4.65; respectively. Homogeneous cores used in cermet case (4 zones).

(b) Third case but with high flux.

(c) Values are maximum at beginning of life. They will decrease rapidly for helium-bonded pins as the fuel swells and the gap is decreased. The probable maximum centerline temp. for helium-bonded pins should be 2500 to 2700 °F (1400 to 1500 °C) based on current technology, but might be extended by the time FFTF is operational.

(d) The cladding in all carbide fuel cases was austenitic stainless steel.

(e) Core power limit (a) exceeded. At 400 MW, centerpoint flux = 0.78×10^{16} nv.

(f) Core power limit (a) exceeded. One could hold 400 MW and high flux (1.49×10^{16} nv) by reducing core volume to ~405 liters and still offer adequate test space.

(g) Other notes:

- All cases in this study used these parameter values:

Parameter	Assigned Value	Parameter	Assigned Value
Core Volume, liters	600	Core Pressure Drop, psi	60
Core Height, ft.	2.83	Total Number of Subassembly Positions (W)	61
Lattice Dimension (distance across the flats of the hexagon), in.	4.6	Number of Fueled Tests (X)	10 (unless noted otherwise)
Core Height-to-Diameter Ratio	0.9	Number of Package Loops (Y)	1
Overpower Factor	1.1	Number of Control Positions (Z)	6
Coolant Temperature Rise, °F	300	Number of Driver Subassemblies (V)	44
Inlet Sodium Temperature, °F	500	(V = W - X - Y - Z = 44)	
Outlet Sodium Temperature, °F	800	Plant Efficiency Factor (operating availability)	0.75

- Since 45,000 MWd/metric ton of heavy atoms (average) was the limit used by PNL on all other FFTF studies, the participants recommended using it on all the carbide fuel cases.
- It was believed that the core designs (all with up-flowing coolant and a vertical, skewed-tube array) allowed adequate space for fuel pin plenums above the fuel to accommodate any specified burnup within reasonable limits.

fabrication costs (on a dollars/kg basis) are reduced. Sodium-bonded mixed carbide fuel showed the best performance in satisfying FTR testing requirements, followed by helium-bonded mixed carbide, sodium-bonded uranium carbide, and helium-bonded uranium carbide fuel. The state of the art of the carbide fuels is about the reverse of the performance potential. The best understood carbide is the sodium-bonded hypostoichiometric uranium carbide, followed by helium-bonded hyperstoichiometric mixed carbide; and the least understood is the sodium-bonded mixed carbide.

The major result of this study is to establish analytically the many attractive aspects of carbide (mixed carbide in particular) driver fuel for FTR. Some of these aspects are flux level, safety characteristics, and fuel cycle costs. For example, for the same peak fast flux (1.0×10^{16} nv), the order of reactor powers required with various fuels in the comparison core is: ^{235}UC > mixed oxide > mixed carbide > cermet. For 45,000 MWd/tonne (average) burnup and the same fast flux, the order of associated predicted fuel cycle costs is: ^{235}UC > mixed oxide > cermet > mixed carbide. Specific cases are discussed in the Analog Computer Studies Section.

Future work on FFTF carbide fuel studies is planned. Analytical studies are being continued regarding the use of carbide fuels for FTR under the Full Performance Fuel Design Analysis Task which is part of the FFTF Fuels Development Program. This task involves the examination of various fuels as potential candidates for FTR full performance fuel. These follow-up analyses will reflect design improvements both in the reactor and the final FFTF core design. These design improvements include using a radially zoned (two equal-volume zones) core which, in the case of the mixed oxide fuel, increases the allowable centerpoint flux (due to the reduction of the radial peaking factor) about 15%, while maintaining the same maximum heat rate in the fuel element. Increasing the core coolant

temperature drop from 300 to 400 °F would also provide design improvement by permitting larger fuel pins with associated lower fissile fractions because of the resultant lower pressure drop for the same power. The carbide-fueled FTR will be examined to explore the effects of increased core outlet temperature. Trade-off studies will be made between flux, test locations, cycle cost, control positions, pressure drop, etc.

DISCUSSION

R&D PROGRAM FOR PROPOSED FFTF CARBIDE FUELS

Assumptions

The following assumptions apply to this study:

- Fabrication of first core fuel elements of UC-PuC or UC is to be done by the site (AEC or commercial) which conducts the fuel fabrication development work associated with the first core,*
- The proposed research and development program must assure that adequate FTR fuel elements of UC-PuC or UC would be available by June 1, 1971.

Fuel Cases Considered

See Summary Table 1.

Current State of Technology

By consensus the current relative technical status of the seven fuel cases was established as:

Best Known	Case 1, 4**
↓	Case 6
	Case 5
Least Known	Cases 2, 3, 7†

* This same assumption was employed in the planning and preparing of the Schedule 189a⁽⁸⁾ for FFTF Fuels Development.

** The consensus is that the most accurate predictions can be made for Case 4.

† Of the seven cases, Case 7 is the most difficult one in which to predict performance.

By consensus, the linear power generation in FFTF pins was limited to a maximum of about 40 kW/ft. Although higher performance requirements were predicted for several cases from the analog study which followed, current carbide tests are generally operating at linear powers in the 20 to 30 kW/ft range; however, tests to operate at linear powers to 45 kW/ft are presently being designed. C. W. Wheelock mentioned that sodium-bonded, hypostoichiometric UC has reached a burnup of at least 30,000 MWd/tonne at linear powers to 40 kW/ft and at higher temperatures than required for FFTF. The group consensus was that major emphasis should be placed on helium-bonded UC-PuC and UC, and that there should be a strong backup effort exerted on sodium-bonded UC-PuC and UC fuel. With all four, the group felt that a burnup capability of 45,000 MWd/tonne (average) was a reasonable assumption.

Currently, sodium-bonded, UC-PuC fuel specimens have yet to be irradiated. Helium-bonded, high density pelletized UC-PuC fuel has achieved an average burnup of 114,000 MWd/tonne with niobium cladding, but the stainless steel-clad specimens failed. However, lower density vibrationally compacted UC-PuC has reached 70,000 MWd/tonne (average) successfully with stainless steel cladding in an ANL Test. The most recent British data available to the participants show that stainless steel-clad, low density UC-PuC powder can reach a burnup of 100,000 MWd/tonne at lower operating temperatures.

The helium bond appears to be better than a sodium bond for small diameter pins. Small pins are of interest because in using them it is possible to obtain higher fluxes without exceeding fuel temperature limits. On the other hand, sodium bonding eliminates the need to go to small pin diameters.

The significant unknown is the highest centerline temperature at which one can expect to operate fuel pins successfully.

With helium-bonded carbide fuel there has been some experience with temperatures to 1800 °C (3272 °F). For the fuel cases, a maximum beginning-of-life temperature of 3600 °F (~2000 °C) was assumed (operating temperatures are expected to be much lower). With sodium-bonded stoichiometric or stabilized fuel the probable maximum centerline temperature which should be allowed is 1400 to 1500 °C (2552 to 2732 °F).

Carbide fuels show a volume increase of 0.7 to 1% per 10,000 MWd/tonne. By consensus the fuel bulk (smear) density should be about 80% of theoretical to provide more room for fuel swelling.

Anticipated Performance of Fuel Fabricated by June 1971

With all seven fuel cases, it was estimated that fuel elements with a burnup capability of 45,000 MWd/tonne (average) could be built for the first core loading by June 1, 1971, if the proposed research and development program is followed.

Probability of Success

With the proposed research and development program, the group believed that a first-core loading of any of the seven cases could be completed by June 1, 1971. On a short-term development basis, the best fuel element possibilities are helium-bonded UC-PuC or UC and sodium-bonded UC. A. A. Strasser furnished the information in Table V as typical data for helium-bonded UC-PuC pellet specimens. On a long-term basis, a sodium-bonded mixed carbide fuel element appears attractive, both technically and economically.

Stretch Capability

It was the opinion of the group that, with a suitable R&D effort beyond June 1971, any of the seven cases could be fabricated into fuel elements with an expected burnup capability of 100,000 MWd/tonne (maximum).

TABLE V. Typical Data from UNC for Helium-Bonded (U, Pu)C Pellet Specimens

Fuel Diam, in.	Pin Diam, in.	Linear Power, kW/ft	Heat Flux, 10 ⁶ Btu/hr-ft ²	Temperatures at Beginning of Life ^(a,b)		$\int_s^L k dt,^{(b)}$ W/cm	Cold Radial Gap, mils
				Fuel Centerline, °F	Fuel Surface, °F		
0.200	0.236 ^(c)	20	1.4	2810	2340	52.2	3.25
0.200	0.236	30	1.9	3510	2810	78.3	
0.200	0.236	36		3880	3040	94	
0.250	0.287 ^(d)	20	0.9	2700	2230	52.2	3.75
0.250	0.287	30	1.7	3350	2650	78.3	
0.250	0.287	36		3710	2860	94	

(a) These are maximum temperatures which will decrease with burnup (i.e. fuel swelling) and improved fuel-cladding contact.

(b) Thermal conductivity of helium is taken as a linear function of temperature. $\bar{k}$ = centerline temp. and s = surface temp.

(c) 169 pins/hexagonal subassembly.

(d) 127 pins/hexagonal subassembly.

Safety Problems

By consensus an overpower factor* of 1.1 was employed in the fuel element performance and safety calculations. The melting point basis used for the overpower factor (1.25) for oxide fuel is not valid for carbide. The bond loss problem in both the sodium- and gas-bonded fuel systems should be thoroughly researched to establish its accident potential. However, it is not readily evident that loss of bond results in a catastrophic occurrence. In the case of sodium bond loss, an element operating at 40 kW/ft will run about 800 °F higher after sodium loss. Consequently, fuel swelling will accelerate and swelling damage may occur within a few days after bond loss. However, the fission products which escape with the sodium bond should be detected within minutes after the cladding is breached. Heat rates are such that neither fuel nor cladding could be melted in this occurrence; consequently, no sudden reactor incident is envisioned. In the case of a gas-bonded element and an end-of-life cladding failure, it would be expected that the total gas volume within an element or pin would equal about 10 to 15 times the element's volume at the core inlet pressure. The core coolant resident time in most FBR designs is on the order of 0.2 sec, so it is not believed likely that a brief blanketing of about 10 to 15 elements in one location would lead to their melting, especially since the gas should be swept out rather quickly as it expands. The criticality problems associated with sodium expulsion in the coolant channel must also be explored.

Mixed carbide fuel for FFTF should provide a sufficiently negative value for the Doppler coefficient of reactivity in the fissile fraction range required. Complete agreement was not

* The carbide fuel pin overpower factor is defined as the ratio of the maximum linear power under transient conditions to the maximum linear power under steady state conditions.

reached on the answer to the question of whether or not a moderator would have to be added to the UC-fueled FTR for safety reasons. Physics calculations at PNL show that with UC fuel, some moderator (BeO or other) would be needed to obtain a safe system because of the high fissile fraction. On the other hand, the need for using BeO to improve the safety of the ^{235}U -fueled cases is questioned by AI, since they expect the Doppler coefficient to be significantly negative for all cases considered.

R&D Program - Tasks

Research and Development Program for Proposed FTR Carbide Fuels - Major Tasks

The list of major task headings is shown below.

- Fuel Process Development
- Fuel Element Hardware Design and Evaluation
- Sodium Testing of Fuel Elements and Components
- Quality Certification Procedures
- Nondestructive Test Development
- Irradiation Testing
- First Core Fabrication Development
- First Core Fabrication
- Fuel Element Inspection and Installation
- Advanced Fuel Development

The task headings and descriptions (see following section) were made the same as those employed by PNL in the Fuels Development Schedule 189a⁽⁸⁾ for case of cost and schedule comparison.

R&D Program - Task Descriptions, General

Before outlining the Fuels R&D Program, the participants reviewed all the FFTF 189 submittals. They were fully aware of the fuels-related tasks not included in the Fuels Program presented in this report. These include such items as materials development and irradiations, fuel washout studies, fuel bundle

hydraulics tests, physics, core design, fuel cycle parametrics, reprocessing, etc. With these tasks assumed to be part of any FFTF development program, the group then addressed itself to the specific program involving developing, designing, and manufacturing a carbide fuel for FFTF. The review of the fuels-related FFTF programs indicated that these programs would be adequate for the mixed, or uranium-only carbide fuels as well as for the other fuels considered for FFTF. The task descriptions are as follows:

(a) Fuel Process Development

The over-all objective of this task is definition of a feasible process for producing the driver fuel pin for the FFTF. The primary consideration is the development of a reliable, chemically and mechanically compatible fuel-cladding system that is compatible with the sodium coolant. The technology for fast reactors requires the investigation of several candidate fuel systems (e.g. pellets, powder, cast slugs). Development of the driver fuel system will include three phases:

- A Process Evolution period for selecting the leading process of a given system.
- A Fabrication Demonstration period in which pilot-scale operation will be initiated for prototypic fuel fabrication.
- A Fabrication period for manufacturing the first core loading. The scope of this task includes Phase 1, above.

(b) Fuel Element Hardware Design and Evaluation

The over-all objective of this task is the analysis, design, fabrication, and testing of the structural members of the FFTF driver fuel elements. The task will consist of evaluating and improving the conceptual design of the driver fuel element hardware, a program of material selection and testing to establish the final design, and establishment of the necessary fabrication techniques.

(c) Sodium Testing of Fuel Elements and Components

The objective of this task is to evaluate corrosion, fretting, wear, thermal expansion, deformation, pressure drop, flow streaming, and structural integrity of prototypic FTR driver fuel elements and components.

Driver fuel element fuel pins and prototypic fuel element subassemblies will be flow tested in out-of-reactor sodium loops. Information from prototypic subassembly tests will be iterated into fabrication and design studies so that the first-core FTR subassemblies can be designed and built. Material performance data will be collected during testing and iterated into design and fabrication studies.

(d) Quality Certification Procedures

The over-all objective of this task is the definition of a quality control system of procedures designed to deliver an adequately high quality fuel element to the FTR.

The Quality Certification system extends broadly throughout the entire fuel design and development work. Activities include: defining the quality which the reactor system requires; analyzing the product and processes so the desired quality will be produced with a minimum of difficulty and cost; planning the quality controls for the fabrication process; planning the inspection and test equipment to measure the product and to adjust the process; working with vendors to assure their delivery of material of the required quality; controlling incoming material quality; solving process quality problems as they arise; auditing final product quality going to the reactor; assuring that packaging and shipping methods are capable of delivering goods to the reactor without loss of quality; and seeing that the fuel element and control system perform satisfactorily in the reactor.

The scope also includes close coordination with vendors in establishing certified suppliers of fuel element materials, control rod materials and components, and fuel elements and components.

(e) Nondestructive Test Development

The first objective of this task is the development of non-destructive testing methods and equipment to guarantee the integrity of the fuel pin end closure, to verify the integrity of fuel pin internal hardware components (plenum parts, etc.), to assure cladding integrity on all FTR fuel pin designs, and to assure good bonding between cladding and core materials in fuel pins. A second objective is to demonstrate model production testers to industry.

This task includes: (1) the feasibility study required to determine the test methods to be used, (2) application of the best available tests to experimental fuel pins, (3) design and fabrication of test units for industry demonstration, (4) vendor certification for fuel element testing, and (5) providing for sample testing of fuel pins received from vendors.

(f) Irradiation Testing

The objective of this task is to provide the additional irradiation experience with prospective FTR driver fuels needed to proceed with the design of a safe, reliable, efficient, and economical FTR driver fuel element.

The scope of this task includes continued irradiation testing of prospective FTR driver fuel forms to provide the following information:

- Propensity of fuel to relocate axially during steady-state and transient conditions,
- Fuel pin swelling characteristics,
- Fission gas release characteristics,

- Chemical and mechanical interactions between fuel and cladding,
- Plutonium and fission product migration characteristics,
- Failure behavior,
- Fuel pin stability throughout high burnup and under transient conditions,
- Effects of burnup on physics (e.g., Doppler), chemical, and mechanical (e.g., thermal expansion) characteristics of the fuel pin,
- Characteristics and desirability of various core-cladding bonds,
- Fuel structural and property changes as a function of burnup,
- Mechanical stability of pins and subassemblies, and
- Final design verification.

(g) First Core Fabrication Development

The over-all objective of this task is the demonstration of a feasible process for the fabrication of driver fuel elements for the FTR.

The scope of this task will include selection of one process for the driver fuel and the initiation of pilot-scale operations for the fabrication of prototypic fuel elements.

(h) First Core Fabrication

The over-all objective of this task is fabrication of the first-core driver fuel elements for the FFTF.

The scope of this task will include the fabrication operations necessary for manufacturing the required driver fuel elements. This fabrication work will be conducted initially in the same pilot plant facilities in which the first core fabrication process demonstration (Task g., above) was performed. Assurance of reactor operating continuity will require continued production until additional vendors can be certified. During the

fabrication of the first core fuel elements, every effort will be made to certify commercial vendor production of the fuel elements at the earliest possible date.

(i) Fuel Element Inspection and Installation

The principal objective of this task is to complete pre-installation and postinstallation inspections of driver fuel elements to validate compliance with applicable requirements.

The scope of this task includes formulation of plans and definitions of procedures for the inspection of fuel elements before and after installation of the first core in the FTR. The plans and procedures will be established by studying the requirements, analyzing candidate inspection systems, and selecting the best concepts. After selecting the most suitable concepts, a more refined study will be conducted to select the best inspection concept. Studies will be conducted to establish how fuel elements (elements for first core, Nuclear Proof Test Facility, critical experiments, thermal-hydraulic tests, etc.) are to be received, inspected (prior to installation in the FTR and to sodium flow and nuclear proof tests), shipped, stored, and handled. The latter four items are for work on a fuel element after initial receipt. Methods will be established for the pre-installation inspection of fuel elements after sodium flow tests (if required) and nuclear proof tests have been conducted.

Studies will be initiated to determine the manner in which fuel elements are to be accepted by and transferred to FTR Operations and to be installed in the FTR by the FTR Operations. If required, techniques will be established for the inspection of installed fuel elements after FTR flow and critical tests have been performed and before the FTR begins fuel and material testing in 1972.

(j) Advanced Development

The objective of this task is to select and develop advanced carbide driver fuels, to fabricate one-third to one-half of a core loading of the selected fuel, to prepare industry to take over the fabrication task and complete the core loading, and to provide industry with the final recommendations on the advanced driver fuel.

There is a need to define the requirements for an advanced carbide driver fuel. The requirements will be determined by the FTR service objectives, the existing technology, the economic restraints, and the FTR environment. After defining the requirements, the advanced driver fuel objectives and characteristics will be specified. Process and hardware design studies for the advanced driver fuel will be initiated, followed by fuel fabrication studies and sodium testing of fuel elements and components, all on a continuing basis. Irradiation testing of the fuel will be initiated. The performance of the fuel will be analyzed and recommendations made on a continuing basis. Based on the recommendations, fabrication of the advanced driver fuel will be initiated and a simultaneous demonstration of the fabrication work for industry will be started.

The advanced driver fuel development work needs to be done for the following reasons:

- The first core fuel element might turn out to be unsafe, unreliable, or too costly.
- Significant improvement in the technology of advanced fuel concepts may occur. Advanced fuel elements may exhibit greater safety, greater reliability, lower cost, or better fuel and/or reactor performance than the first core elements.

R&D Program - Task Descriptions, Specific

The participants stated the following specific R&D needs:

a. Task - Fuel Process Development

UC-PuC Fuel Cases*

- There is a need to develop a fabrication process for Case 2 fuel pins. Processes for fuel pins in Cases 3, 4, and 7 need to be scaled up from the existing laboratory to the pilot plant scale.
- A technique is needed for preparing high-density feed material for Case 5 packed-powder fuel pins.
- Specifications and quality control requirements need to be established. The cost of material quality control is still high and opportunities to reduce the process cost exist.⁽⁹⁾

UC Fuel Cases*

- There is a need to study the fabrication problems with the small pin diameters in Case 1.
- Same as entry for mixed carbides but a slightly lower effort level would be required.

* The fuel cases are repeated below for the reader's convenience:

<u>Fuel Case No.</u>		<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>
Fuel:	UC-PuC		X	X	X	X		X
	UC	X					X	
Stoichiometry:	Hypo.	X	X					
	Stoichiometric	X	X	X				
	Slightly Hyper.			X				
	Hyper.				X	X	X	X
Bond:	Sodium	X	X	X				
	Helium				X	X	X	X
Form:	Cast Slugs	X	X	X				
	Sintered Pellets	X	X	X	X			
	Low Density Pellets					X	X	
	Packed Powder					X	X	
	Annular Pellets							X

- All the engineering properties such as thermal conductivity and diffusivity, melting point, coefficient of expansion, vapor pressure, U-Pu-C phase diagram, and compatibility with claddings require confirmation with improved precision and measurement to much higher temperatures. This applies particularly to the thermal conductivity and the U-Pu-C phase diagram with and without oxygen and nitrogen impurities. The effect of fuel structure and composition on properties is also required as the first step toward optimization of the fuel cycle economics.
- Same as mixed carbide entry, but only a small effort would be required.

Some important properties have not been measured to date and include:

High temperature mechanical properties (compressive creep, modulus of elasticity)

Thermodynamic properties

Properties of fuel containing fission products.⁽⁹⁾

Physical and mechanical property determinations need to be made. Some work is going on now (two items mentioned were creep studies and strength measurements). Thermal conductivity values need to be determined. There is a need to establish compositional (e.g., carbon, oxygen, and nitrogen contents) control measures.

Stoichiometry control studies are needed; however, stoichiometry control is not as much of a problem with pellets made by the carbothermic process.

- There is a need to conduct compatibility tests (fuel-cladding-coolant) with Case 2 fuel. More work is needed on Case 3 than on Case 2. Some of this work is in progress now. The fuel-cladding compatibility work needs more emphasis - it is currently at a low level.
- Sodium-bonded UC-PuC will require more development effort than the helium-bonded UC-PuC fuels cases. The analog computer runs indicate that sodium-bonded fuel may be quite favorable from a fuel cycle cost viewpoint. While there may be some common requirements in the development of helium- and sodium-bonded carbide fuel systems in the fuel material, most of the requirements are different.
- The packed powder fuel cases will require more development work than the pellet or cast slug cases. In particular, economic preparation of high density feed material for packed powder pins will be a problem.
- Laboratory-scale use of arc melting and casting techniques has been successful. Arc melting requires the development of methods to minimize plutonium losses, as well as casting long slender pins (inherent wide tolerances would limit their use to sodium-bonded fuel pins).⁽⁹⁾

- Development work needs to continue on the process⁽⁹⁾ to produce mixed carbide powder (pellets have been successfully cold pressed and sintered) continuously, starting with uranium and plutonium metal.

b. Task - Fuel Element Hardware Design and Evaluation

UC-PuC Fuel Cases

- In comparison to the helium-bonded fuel cases, the sodium-bonded fuel cases require more development problems to be solved. There is a lack of data on the behavior of both helium- and sodium-bonded fuel during transient or nonstandard operating conditions and work needs to be initiated (see ninth item under UC-PuC in Task f). Data are needed on fuel elements operating with high heat fluxes. High heat fluxes pose a possible burnout problem, a cladding thermal stress problem, and an increased mass transfer problem. With sodium-bonded fuel, studies are needed to establish the effect of bubbles and loss of sodium bond on heat transfer. Bubbles, formed from released fission gases, could become trapped and cause burnout; however, the effects of bond loss

UC Fuel Cases

- Same as for mixed carbide.

from a sodium-bonded fuel element are believed to be noncatastrophic and, as indicated by experience,* are generally predictable.

- There is a need to study various cladding materials, especially those which exhibit high strength and ductility and good chemical and/or mechanical compatibility characteristics, in an effort to further improve fuel pin performance. The group recommended a study of the medium-nickel alloys (e.g., Incoloy 800, Inconel 625).
- There is a need to study how the gap conductance varies as the gap changes. The gap decreases as irradiation progresses and the gap conductance increases but at an unknown rate and thus, uncertainty is introduced in the fuel temperature estimates.
- Proposed solutions⁽⁹⁾ to the fuel swelling problem need to be evaluated. The solutions proposed are:
- Same as for mixed carbide.
- Same as for mixed carbide.

* Fuels have operated in EBR-II, SRE (Cores I and II), and Hallam without evidence of this loss-of-sodium-bond problem. The effect of sodium bond loss has been demonstrated (see NAA-SR-10850 report) with a capsule (NAA-81-4) which developed a leak. The cladding was dimensionally unchanged and fuel melting did not occur.

provide space in the fuel pin to accommodate fuel expansion, strengthen the fuel to prevent swelling or weaken the fuel to make it easy to restrain, and, develop more ductile strong claddings.

c. Task - Sodium Testing of Fuel Elements and Components

UC-PuC Fuel Cases

- No specific comments were made; however, the participants agreed that there are limited hot sodium flow facilities available. The group suggested checking the applicability of the new WAPD facilities, the UNC facilities, and the extensive AI facilities.

UC Fuel Cases

- Same as for mixed carbide.

d. Task - Quality Certification Procedures

UC-PuC Fuel Cases

- No specific comments were made. See third item under UC-PuC in Task a.

UC Fuel Cases

- One may have lower demonstration costs with UC fuel than with UC-PuC fuel since highly-enriched UC can be simulated with low-enriched UC. In addition, less effort may be required to

establish specifications with UC fuel than with UC-PuC fuel.

e. Task - Nondestructive Test Development

UC-PuC Fuel Cases

- A bond test (ultrasonic or other) will be needed for sodium-bonded fuel pins to assure adequate in-reactor performance. Thorough inspection of the bond is difficult (though such a bond test was employed in making three core loadings for SRE and one for Hallam), but is important since undetected gas bubbles or nonbond areas can cause cladding burnout. No bond tester is needed for the helium-bonded fuel pins.
- For both helium-and sodium-bonded fuel pins, there is a need to concentrate more effort on weld and closure tests. In-reactor failures of welds or closures can result in the release of fission gas, which in turn may cause a more serious problem - voiding the fuel subassembly of sodium.

UC Fuel Cases

- Same as for mixed carbide.
- Same as for mixed carbide.

f. Task - Irradiation Testing

UC-PuC Fuel Cases

- Screening tests need to be conducted in ETR-MTR and be followed by additional tests to goal burnups

UC Fuel Cases

- Same as for mixed carbide.

of 3 to 4 at.% in ETR-MTR. The purpose of the screening and other tests is to facilitate the selection of the more promising fuel candidates, to determine the behavior of the selected candidates under thermal flux irradiation conditions, and to attempt to establish a correlation between thermal flux and fast flux irradiation data. The situation on screening tests is that pins with Case 1 fuel have been irradiated. Irradiation tests are currently in progress with Case 4 fuel. Limited irradiation tests are going on now with Case 3, Case 5, and Case 7 fuels. Pins from the other cases need to be tested.

- Fuel pins from Cases 1-7 should be irradiated in a fast flux under comparable conditions (e.g., in sodium coolant, at representative temperatures, and with typical heat outputs) since there are insufficient data available to predict adequately the ultimate capabilities of either sodium- or helium-bonded carbide fuel. All but the Case 7 type pins could be tested in a 37-pin bundle array. For Case 7, initial single pin screening tests should be conducted. Goal burnup in all cases should be set at 3 to 4 at.%.
 - Same as for mixed carbide.

Fuel pins from Cases 1, 4, and 5 could be irradiated in EBR-II almost immediately. Double encapsulation of these fuel pins would not be needed.

- Safe, reliable performance of the fuels needs to be demonstrated under normal operating conditions and to maximum goal burnup, under overpower conditions and under transient conditions (run a limited number of tests of the last type). The maximum heat fluxes and fuel temperatures both need to be tested in actual irradiation experiments (with fast and thermal flux tests). The heat flux is a limitation just like the maximum fuel temperature.
- Hot sodium tests should be considered with full-scale instrumented, prototypic fuel elements to evaluate design characteristics and subassembly and fuel pin fabrication parameters. Because of the limited ability of available fast flux facilities to accommodate instrumented specimens, the tests might be conducted in a thermal flux reactor. There may be heat flux limitations established by cladding thermal stresses, burnout potential, and mass transfer.
- Same as for mixed carbide.
- Same as for mixed carbide.

- To assure a safe, reliable, and economic FTR fuel, there is a need to establish a comprehensive correlation between carbide fuel variables (e.g., linear power, burnup, fuel composition, fuel characterization, pin design) and fast flux irradiation performance in high temperature sodium.
- There is a need to study the effect of oxygen on the irradiation behavior of the fuels. Some work is currently underway on this problem. The effect of oxygen on the irradiation behavior of the fuel is unknown. At high temperature, the oxygen may react with carbon to form CO and add to the swelling problem.
- There is a need to study the effect of burnup and fission products on fuel-cladding compatibility.
- Irradiation tests (such as a thermal flux type) are needed with sodium-bonded fuel to study the effects (e.g., higher centerline temperatures, increased fuel swelling) of loss of bond (i.e., the problem of sodium expulsion if a weld fails on the lower end of a sodium-bonded fuel pin); however, this is a relatively long term problem and is probably negated by the effects of the expulsion of fission gases themselves, which can create an immediate problem.
- Same as for mixed carbide.
- Same as for mixed carbide, except that the oxygen content in the case of UC is low and is not a problem.
- Same as for mixed carbide.
- Same as for mixed carbide.

Tests are needed to evaluate the more serious effect of loss of gas (e.g., if the closure fails on the lower end of the pin) from a gas- or sodium-bonded fuel pin (i.e., what is the effect of high pressure gas being blown into the assembly and consequently voiding the assembly of sodium).

- Transient studies, in particular with irradiated fuels, need to be completed for accident analyses by 1969-1970. Transient tests with mixed carbide fuel pins are lacking and both helium- and sodium-bonded fuels need to be studied. Testing fuel pins to failure is necessary. Transient testing of failed fuel is necessary.
- There has been an insignificant number of irradiation tests performed at present to specify adequately a carbide fuel for FTR, a reactor which demands a very high degree of safety and reliability. Fast flux space availability will be a serious limitation for the testing program; however, the group indicated that perhaps a total of 90 pins need to be irradiated in a fast flux. The group took
- Same as for mixed carbide; however, a lower level of effort is required with UC fuel.
- See mixed carbide entry (UC pins are included in the total).

into account the fuel pins currently under test* (~ 30 to 40 in EBR-II) and felt that the additional 90 pins would be required to aid in solving the major problem with ceramic fuel elements. The major problem concerns the method of coping with fuel swelling** at high burnups and with high linear powers without undergoing severe cladding damage.

* Regarding future tests, Strasser indicated that by 1969, at least 55 (U,Pu)C helium-bonded pins will have been irradiated in fast reactors.

** To accommodate fuel swelling, variations of both helium-bonded and sodium-bonded fuel are proposed but certain problems will require solutions:

<u>Bond</u>	<u>Fuel Configuration</u>	<u>Problems to be Solved</u>
Helium	(a) High density, solid pellets with large helium annulus between fuel and cladding	(1) For configurations (a), (b), (c), and (d), one needs to solve the problem of how to provide sufficient void without excessive fuel temperatures.
	(b) High density, cored pellet	(2) For configurations (c) and (d), one needs to solve the problems of slumping and in-reactor sintering.
	(c) Low density, solid pellet	
	(d) Low density, packed powder	
Sodium	(e) High density, solid pellets with large sodium annulus	(3) For configurations (e) and (f), the following compatibility problems need to be solved:
	(f) Low density, solid pellets with large sodium annulus	(i) fuel-sodium-cladding, and (ii) fission product-sodium-cladding.
		(4) For configurations (e) and (f), a solution is required to the problem of loss of

sodium bond, which can occur in several ways:

- (i) by a small cladding leak,*
 - (ii) by bubble formation in the sodium bond, or*
 - (iii) by other means (e.g., a reactor transient).*
- (5) *For configurations (e) and (f), a solution is required to the problem of how to provide a fuel with the desired low swelling rate.*

UC-PuC Fuel Cases

- The program should be authorized by October, 1966, if this proposed task is to meet the required schedule (Figure 1).

g. Task - First Core Fabrication Development

- This work could start as early as January, 1967. See Figure 1 and note that the timing for Task g. (First Core Fabrication Development) is closely related to the end of Task a. (Fuel Process Development) and the start of Task h. (First Core Fabrication).
- This work can be completed in about 1-1/2 yr if helium-bonded fuel is employed.

UC Fuel Cases

- Same as for mixed carbide.
-
- Same as for mixed carbide.
-
- Same as for mixed carbide.

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FIGURE 1. Schedule for Development of Carbide Fuel for FFTF

- With sodium-bonded fuel, about 2 to 2-1/2 yr would be required to complete this work.
- The sodium-bonded UC fuel process is generally known except for the effects of going to smaller diameter pins.
- Methods for reliable reproducible production of carbides have been developed and excellent control of composition, structure, and dimensions have been demonstrated in the U.S. and U.K. on a laboratory scale of 1/2 kg batches.⁽⁹⁾ Development work is needed to scale the processes up to batch sizes on the order of several kilograms.
- Vibrationally compacted fuel pins have been made, but the large-scale economical preparation of feed material still needs to be developed.⁽⁹⁾

h. Task - First Core Fabrication

UC-PuC Fuel Cases

- This work could start as early as the middle of CY-1968 or could be delayed to the end of CY-1968 in order to take advantage of additional data from all tasks, but especially from Tasks a., (e.g., sodium-bonded

UC Fuel Cases

- Same as for mixed carbide, except that it could be delayed to the end of CY-1969.

UC-PuC and packed powder or low-density UC-PuC pellets), f. and g. (e.g., sodium-bonded, UC-PuC).

- This work could be completed in about two years.
- This work could be completed in about 1-1/2 yr.
- Synthesis and fabrication methods (see fourth item under UC-PuC in Task g., above) can and need to be adapted for mass production; they are currently being scaled up from the laboratory to the pilot plant.⁽⁹⁾
- Early utilization of large quantities of carbide fuels will develop the industrial fuel supply potential and reduce fuel fabrication costs by increasing fuel throughput.⁽⁹⁾
- Same as for mixed carbide.

i. Task - Fuel Element Inspection and Installation

UC-PuC Fuel Cases

- No specific comments were made by the group except that inspection will be simpler with helium-bonded fuel elements than with sodium-bonded fuel elements.

UC Fuel Cases

- Same as for mixed carbide.

j. Task - Advanced Development

UC-PuC Fuel Cases

- The development of vented UC-PuC fuel elements is lacking and

UC Fuel Cases

- Same as for mixed carbide.

- needs to be initiated. Vented fuels are of interest for several reasons: (1) one can avoid a sudden release of fission gas (see eighth item under Task f., above); (2) the stresses in the cladding can be reduced; and (3) one may be able to increase the operational range of the fuel element.
- There is a need to study high-strength cladding alloys (see second item under Task b., above).
 - Sodium-bonded UC-PuC fuel elements may or may not fall into this category - agreement was not reached among the participants on this point.
- Same as for mixed carbide.
 - Same as for mixed carbide.

R&D Program - Schedule

The schedule for the proposed R&D program for UC-PuC and UC fuel development is shown in Figure 1. It should be noted that the First Core Fabrication Task could be completed in 2-1/2 yr, starting either early or late in FY-1969 with UC-PuC fuels, or in 1-1/2 yr, starting early in FY-1969 or in mid FY-1970 with UC fuels.

R&D Program - Manpower Needs

The group estimated, for each of the four major fuel R&D categories (first category - Case 1; second category - Cases 4, 6, and 7; third category - Case 5; and fourth category - Cases 2 and 3), the manpower (engineers and scientists) required for each task shown above. The estimates are shown in Table VI.

TABLE VI. Manpower Needs for Proposed FFTF Carbide Fuels

Fuel R & D Category No.		1	2	3	4
Fuel Case No.		1	4 6 7	5	2 3
Fuel:	UC-PuC		X X	X	X X
	UC	X	X		
Stoichiometry:	Hypo.	X			X
	Stoichiometric	X			X X
	Slightly Hyper.				X
	Hyper.		X X X	X	
Bond:	Sodium	X			X X
	Helium		X X X	X	
Form:	Cast Slugs	X			X X
	Sintered Pellets	X	X		X X
	Low Density Pellets		X	X	
	Packed Powder		X	X	
	Annular Pellets		X		
TASK		ENGINEERS & SCIENTISTS, MAN-YEARS			
MANPOWER NEEDS (a)					
a.	Fuel Process Development	1.5	2.0	3.0	6.0
b.	Fuel Element Hardware Design and Evaluation	9.0	9.0	9.0	9.0
c.	Sodium Testing of Fuel Elements and Components	1.5	1.5	1.5	1.5
d.	Quality Certification Procedures	15.0- 22.5	22.5	22.5	22.5
e.	Nondestructive Test Development	1.6	1.5	1.5	1.6
f.	Irradiation Testing	30.0	37.5 ^(b)	37.5	45.0
g.	First Core Fabrication Development	6.0	9.0	9.0	12.0
h.	First Core Fabrication	15.0	19.5	19.5	21.0
i.	Fuel Element Inspection and Installation	3.1	3.0	3.0	3.1
j.	Advanced Development	<u>10.5</u>	<u>10.5</u>	<u>10.5</u>	<u>10.5</u>
TOTALS		93.2-100.7	116.0 ^(c)	117.0	132.2 ^(c)

(a) The absolute numbers for man-years are best estimates at this time but comparison of the carbide program with the present FFTF fuels program (See Appendix Table B-I) should be reasonably good. Consensus indicated that the manpower level per year would be about constant or it might show a linear increase (total) of as much as 10% in the period FY 1967 through 1971. Some of the manpower estimates may seem high but certain tasks span the entire period from FY-1967 through the installation of the first FTR core.

(b) There will be a considerable amount of data input from the current LM-FBR Program for Cases 4-7 and this will tend to reduce the manpower needs for this task.

(c) Total for any one fuel case in that category.

Facilities Required

Facilities Required for Proposed R&D Programs

Both AEC and commercial facilities for uranium carbide fuels are available now; however, the mixed carbide facilities are inadequate at present.

Facilities Required for Pilot Plant and Production Operations

Both AEC and commercial facilities will be available in 6 to 18 months.

Site Recommendations

Site Recommendations for Proposed R&D Program

Both AEC and commercial sites are recommended for the work on the proposed R&D Programs. The group declined to recommend specific sites.

Site Recommendations for Production

Commercial sites should be employed for the proposed production work, but the group declined to specify particular sites.

Continued Fueling of FFTF

The group expressed the opinion that one could expect a continuing decrease in core fabrication costs for core loadings prepared subsequent to the first core. The cost curves will probably follow the conventional learning curve.

ANALOG COMPUTER STUDIES ON PROPOSED CARBIDE FUELS FOR FFTF

Summary of the Carbide Fuel Analog Computer Studies

The mixed carbide analyses showed better results than the mixed oxide analyses in several categories. Specifically, at the same operating conditions, the carbide core required a lower fissile fraction due to its higher metal density and

permitted a larger fuel pin size. The values for flux level at constant power density are about 8% greater for the mixed oxide than they are for the mixed carbide due to the higher fertile metal density in the carbide (see Figure 2). The carbide, with the higher k_{eff} , can operate at a greater power density with the same size fuel elements. This allows approximately 20 to 40% higher flux level with the mixed carbide, in comparison with the mixed oxide, using the same pin size and a slightly lower fissile fraction. Conversely, at the same flux level, the mixed carbide has a considerably larger pin diameter than the mixed oxide, e.g., at a flux of 1×10^{16} nv the sodium-bonded mixed carbide pin clad OD is about 0.3 in. versus 0.2 in. OD for the mixed oxide pin with a flux of 0.75×10^{16} nv.

Both the sodium-bonded and helium-bonded mixed carbides showed cycle costs which are lower than the mixed oxide cycle costs at the same burnup and flux. This is due principally to the larger pin size allowable for the mixed carbide and thus the lower fabrication cost per kg of fuel. A typical example of cycle cost is: at the same flux of 0.83×10^{16} nv and at a burnup of 45,000 MWd/tonne, the oxide fuel cycle cost is $\$3.4 \times 10^6$ /yr. The cost for the mixed carbide would be less than that for mixed oxide for burnup levels greater than about 30,000 MWd/tonne. At the same conditions of burnup and flux, the mixed carbide cost is about 30% below that of the mixed oxide for sodium-bonded cases. The cycle costs are similar for helium-bonded cases. It should be pointed out, however, that the lower cycle costs are due in some degree to the lower core volume - 600 liters for the mixed carbide versus 800 liters for the mixed oxide - which lowers throughput. On a comparative basis, a "reasonably" conservative mixed-carbide-fueled FTR core could be designed to achieve a flux of about 1×10^{16} nv. For the same conditions, a "reasonably" conservative mixed oxide core could be designed which will produce a flux level of around 0.8×10^{16} nv.

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FIGURE 2. Centerpoint Flux Versus Core Volume
for Various FFTF Fuels*

From the standpoint of pin size and flux level, the uranium-carbide-fueled FTR's were on a par with or better than the mixed oxide (see Cases 1 and 6). However, the cycle costs for the uranium carbide were higher due to the lower neutron generation rate per MW involved in ^{235}U fission. The throughput is also higher for the ^{235}U cases because of their higher fissile inventory requirements (see Figure 3). The heat rejection rate for a ^{235}U -fueled FTR is about 30% higher than a mixed-oxide-fueled FTR at the same flux level and core volume. The larger inventory for the ^{235}U cases, which also results in a fuel fissile fraction that is greater than the ^{239}Pu cases, raises serious questions about the ^{235}U cases from a safety standpoint because of their low Doppler coefficient. See comments under R&D Program, Safety Problems.

An important operating aspect, which may be concluded from the analog runs, is that mixed carbide sodium-bonded fuel has a potential for high fluxes in the FTR. For instance, a flux of about $1.5 \times 10^{16} \text{ nv}$ can be achieved at a cycle cost of around $\$5 \times 10^6/\text{yr}$ (see Case 3, Run 5). This case uses a pin size of around 1/4 in. and a fissile fraction of about 28% -- this performance is virtually impossible with mixed oxide fuel. For this condition the mixed oxide fuel pin would be around 0.14 in. OD, and the fissile fraction would be close to 40%.

As the fuel program for FFTF progresses into the 1970's, it would be expected that at that time, carbide thermal performance data should be sufficiently established to demonstrate fluxes of $2 \times 10^{16} \text{ nv}$ with reasonable pin size and fuel fractions which could be used in the FTR. This could be achieved within the FFTF complex with a core volume of 400 to 500 liters to stay within the projected heat rejection requirements.

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FIGURE 3. Fissile Inventory Versus Core Volume
for Various FTFF Fuels*

Limitations

Some of the general limitations on the various fuel cases are as follows:

- The maximum interface temperature is limited to $<1800^{\circ}\text{F}$ since the stainless steel-carbide fuel eutectic melts at about 1000°C (1832°F).
- With hypostoichiometric UC fuel, the maximum temperature limit is $<2000^{\circ}\text{F}$ because of the melting point (1132°C or 2070°F) of uranium.
- With sodium-bonded, hypostoichiometric UC-PuC fuel, a recommended upper temperature limit is 1800°F because of the melting point of the metal phase.
- With helium-bonded carbide fuel some experience exists to 1800°C ($\sim 3300^{\circ}\text{F}$), but the allowable maximum centerline temperature is unknown. For the fuel cases, a maximum beginning-of-life temperature of 2000°C ($\sim 3600^{\circ}\text{F}$) was assumed (operating temperatures are expected to be much lower).
- With sodium-bonded stoichiometric or stabilized fuel, the probable maximum centerline temperature which should be allowed is 1400 to 1500°C (2552 to 2732°F).
- With hypostoichiometric UC, interaction of the free uranium with the stainless steel cladding has been observed at cladding temperatures starting as low as 760 to 870°C (1400 to 1598°F).
- Maximum fuel temperature for the helium-bonded cases was calculated in this report by using a gap conductance of 1000 Btu/hr-ft^2 over the entire lifetime of the fuel. It should be pointed out that more refined calculations will reduce the maximum fuel temperature.
- Heat flux through the cladding was not established as a limit and is noted as calculated. There may be heat flux limitations established by cladding thermal stresses, burnout potential, and mass transfer. There is

considerable experience up to 1×10^6 Btu/hr-ft², limited experience up to 1.5 to 2.0×10^6 Btu/hr-ft², and no experience above this. The carbide cases studied all involved heat fluxes above 1.0×10^6 Btu/hr-ft².

Summary of Fuel Cases, Performance, and Economic Data

The following Tables VII through XIII list the important core parameters for the carbide cases studied. A complete data list for the 31 cases examined is given in Appendix A. Listed with the table for each individual case are the various values and constraints such as centerline temperature, $\int kdt$, cladding thickness, smear density, gap coefficient, etc. It will be noted in several of the analog runs, in Tables VII-XIII and in Appendix A, that some of the constraint values, in particular centerline temperature, are never reached. This is due, primarily, to the high thermal capabilities of the carbide fuel, in that high flux levels (high power density) and satisfactorily large diameter pin sizes with low fissile fractions can be achieved at a margin well below the maximum centerline temperature specified.

The values listed below are used in each of the cases shown in Tables VII-XIII and the tables in Appendix A.

<u>Parameter Symbol</u>	<u>Parameter</u>	<u>Assigned Parameter Value</u>
V	Core Volume, liters	600
H/D	Height-to-Diameter ratio	0.9
F _T	Overpower factor	1.1 (unless otherwise noted)
ΔT	Coolant temperature rise, °F	300
T ₁	Inlet sodium temperature, °F	500
T ₂	Outlet sodium temperature, °F	800
ΔP	Core pressure drop, psi	60
N _S	Total number of subassemblies	61
N _T	Number of fueled tests	10
N _R	Number of control positions	6
P _F	Plant efficiency factor (operating availability)	0.75

TABLE VII. *Summary of Data for Case 1 Fuel
(sodium-bonded, hypostoichiometric
to stoichiometric UC sintered pellets
or cast slugs with SS cladding).*

- Maximum allowable fuel centerline temperature (T_{CL}) to be < 2000 °F
- Maximum fuel surface temperature (T_C) to be < 1300 °F
- Cladding surface temperature (on coolant side) to be < 1000 °F^(a)
- Stainless steel cladding thickness should be about 0.015 in.
- Fuel bulk (smear) density (ρ_s) should be about 80% and be based on the following equation:^(b) $d_i = 0.9 d_o - 0.027$
(in inches)
- $\int k_f dt = 0.128 (T_{CL} - T_C)$ in W/cm.

Run	1	2	3	4	5
T_{CL} , °F	2,000	1,900	1,900	2,150	2,163
Pin OD, in.	0.248	0.207	0.207	0.289	0.363
Reactor Power, MW	440	513	513	398	296
Pins/yr	11,320	19,910	12,270	7,250	3,310
Centerpoint Flux, 10^{16} nv	0.86	1.00	1.00	0.78	0.58
Average Burnup, MWd/tonne	45,000	45,000	73,180	45,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	3.98	4.99	3.89	3.36	2.40

(a) Note: Since this was the limit used by PNL on other FTR fuel studies, the group recommended using it on all the fuel cases.

(b) See Table XIV on nomenclature.

TABLE VIII. *Summary of Data for Case 2 Fuel
(sodium-bonded, hypostoichiometric
to stoichiometric UC-PuC sintered
pellets or cast slugs with SS
cladding)*

- Maximum allowable fuel centerline temperature (T_{CL}) to be < 1800 °F
- Maximum fuel surface temperature (T_C) to be < 1200 °F
- Cladding surface temperature (on coolant side) to be < 1000 °F (see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Fuel bulk (smear) density - same as Case 1
- $\int k_f dt = 0.107 (T_{CL} - T_C)$ in W/cm.

Run	1	2	3	4
T_{CL} , °F	1,800	1,800	1,800	1,800
Pin OD, in.	0.246	0.228	0.258	0.258
Reactor Power, MW	397	397	341	341
Pins/yr	10,450	9,400	6,060	13,400
Centerpoint Flux, 10^{16} nv	1.00	1.00	0.86	0.86
Average Burnup, MWd/tonne	45,000	60,000	60,000	26,830
Fuel Cycle Cost, \$ 10^6 /yr	3.18	2.82	2.23	3.64

TABLE IX. *Summary of Data for Case 3 Fuel
(sodium-bonded, stoichiometric or
slightly hyperstoichiometric, UC-PuC
sintered pellets or cast slugs with
SS cladding)*

- Maximum allowable fuel centerline temperature (T_{CL}) to be ≤ 2700 °F or < 3600 °F^(a)
- Maximum fuel surface temperature (T_C) to be < 1400 °F
- Cladding surface temperature (on coolant side) to be < 1000 °F (see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Fuel bulk (smear) density - same as Case 1
- $\int k_f dt = 0.107 (T_{CL} - T_C)$ in W/cm.

Run	1	2	3	4	5	6
T_{CL} , °F	2,700	2,310	2,310	2,700	2,700	2,315
Pin OD, in.	0.324	0.288	0.288	0.254	0.233	0.286
Reactor Power, MW	398	397	397	538	593	400
Pins/yr	5,600	7,360	10,530	13,220	17,890	7,540
Centerpoint Flux, 10^{16} nv	1.00	1.00	1.00	1.35	1.49	1.00
Average Burnup, MWd/tonne	45,000	45,000	31,540	45,000	45,000	45,000
Fuel Cycle Cost, \$ 10^6 /yr	2.69	2.84	3.54	4.5	5.33	3.0

(a) *Some participants requested that this higher fuel temperature be used but due to other design considerations this value was never reached in the analog runs.*

TABLE X. *Summary of Data for Case 4 Fuel
(helium-bonded, hyperstoichiometric,
UC-PuC, sintered pellets with SS
cladding)*

- Maximum allowable fuel centerline temperature (T_{CL}) to be < 3600 °F
- Cladding surface temperature (on coolant side) to be < 1000 °F (see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Gap coefficient = 1000 Btu/hr-ft^2
- Fuel bulk (smear) density to be calculated from this equation,
 $d_i = d_o - 0.037$ (in inches), which assumes 0.015 in. thick
cladding and a 0.007 in. diametral gap
- $\int k_f dt = 0.107 (T_{\text{CL}} - T_{\text{C}}) \text{ W/cm.}$

Run	1	2	3	4
T_{CL} , °F	3,593	4,032	3,600	3,644
Pin OD, in.	0.296	0.238	0.210	0.177
Reactor Power, MW	298	398	398	398
Pins/yr	4,430	9,790	13,240	12,700
Centerpoint Flux, 10^{16} nv	0.75	1.00	1.00	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	45,000
Fuel Cycle Cost, \$ 10^6 /yr	1.89	2.91	3.21	3.13

TABLE XI. *Summary of Data for Case 5 Fuel
(helium-bonded, hyperstoichio-
metric, UC-PuC, packed powder or
low density pellets with SS
cladding)*

- Maximum allowable fuel centerline temperature (T_{CL}) to be <3600 °F
- Cladding surface temperature (on coolant side) to be <1000 °F
(see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Gap coefficient = 2,000 Btu/hr-ft²
- Thermal conductivity of fuel is 8.55 Btu/hr-ft-°F
- Fuel bulk (smear) density = 80%
- $\int k_f dt = 0.082 (T_{CL} - T_C)$ in W/cm

Run	1	2	3	4
T_{CL} , °F	3370	3213	2815	2523
Pin OD, in.	0.253	0.246	0.215	0.221
Reactor Power, MW	398	391	394	344
Pins/yr	9,870	10,290	14,260	11,720
Centerpoint Flux, 10^{16} nv	1.00	0.98	1.00	0.86
Average Burnup, MWd/tonne	45,000	45,000	45,000	45,000
Fuel Cycle Cost, \$ 10^6 /yr	3.13	3.12	3.38	2.93

TABLE XII. *Summary of Data for Case 6 Fuel (helium-bonded, hyperstoichiometric UC, packed powder or low density pellets with SS cladding)*

- Maximum allowable fuel centerline temperature (T_{CL}) to be <3600 °F
- Cladding surface temperature (on coolant side) to be <1000 °F (see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Gap coefficient = 1,000 Btu/hr-ft²
- Thermal conductivity of fuel to be 70% of the value for 100%-dense fuel
- Fuel bulk (smear) density = 80%
- $k_f dt = 0.0995 (T_{CL} - T_C)$ W/cm

Run	1	2	3
T_{CL} , °F	2926	3797	4736
Pin OD, in.	0.227	0.213	0.202
Reactor Power, MW	306	408	512
Pins/yr	9,650	14,980	21,190
Centerpoint Flux, 10 ¹⁶ nv	0.60	0.80	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	2.73	3.72	5.00

TABLE XIII. *Summary of Data for Case 7 Fuel
(helium-bonded, hyperstoichio-
metric, UC-PuC, sintered annular
pellets with SS cladding)*

- Maximum allowable fuel centerline temperature (T_{CL} to be <3600 °F)
- Cladding surface temperature (on coolant side) to be <1000 °F (see Case 1 note)
- Stainless steel cladding thickness should be about 0.015 in.
- Gap coefficient = $1,000$ Btu/hr-ft²
- Fuel bulk (smear) density to be about 80% (used same equation as in Case 1)^(a)
- $\int k_f dt = 0.160 (T_{CL} - T_C) \text{ W/cm}^{(a)}$

Run	1	2	3	4	5
T_{CL} , °F	3686	3771	3361	3200	5000
Pin OD, in.	0.247	0.246	0.210	0.246	0.210
Reactor Power, MW	389	398	398	342	537
Pins/yr	10,280	10,620	13,590	9,060	19,200
Centerpoint Flux, 10^{16} nv	0.98	1.00	1.00	0.86	1.35
Average Burnup, MWd/tonne	45,000	45,000	45,000	45,000	45,000
Fuel Cycle Cost, \$ 10^6 /yr	3.02	3.12	3.30	2.71	5.65

(a) *Annular volume reduction taken on perimeter for convenience. Increased thermal rate is accounted for in the equation for $\int k_f dt$.*

General Technical and Economic Input Data for the Parametric Studies

Changes Required to Convert $\text{UO}_2\text{-PuO}_2$ Economic Input Data to UC-PuC or UC Input Data

The previously used $\text{UO}_2\text{-PuO}_2$ input data and terminology may be found in Appendix B. The following changes were made to modify the $\text{UO}_2\text{-PuO}_2$ economic data for use in the UC and UC-PuC Fuel Cases.

Parameter Symbol	Parameter Title	$\text{UO}_2\text{-PuO}_2$ Value	UC-PuC Values ^(a)	Remarks
<u>UC-PuC Fuel Cases</u>				
C_1	Unit Conversion	\$500/kg ^{239}Pu	\$575/kg ^{239}Pu processed	Value Raised 15%
C_4	Unit Metal Reprocessing Cost	\$24,000/day	\$27,600/day	Value Raised 15%
C_p	Estimated Pin Fabrication Cost, Before Correction Factors are Applied	\$85.95/pin	\$98.90/pin	Value Raised 15%
<u>UC Fuel Cases</u>				
A_4	Reprocessing Plant Rate	20 kg/day	35 kg/day	Value Raised because of higher U criticality limits
C_1	Unit Conversion	\$500/kg ^{239}Pu processed	\$575/kg ^{235}U processed	Value Raised 15%
C_4	Unit Metal Reprocessing Cost	\$24,000/day	\$27,600/day	Value Raised 15%
C_p	Estimated Pin Fabrication Cost, before Correction Factors are Applied	\$85.95/pin	\$82.10/pin	Value reduced 20% because no plutonium present but raised 15% because it is carbide fuel.
C_{Pu}	Value of fissile material ^(b)	\$10,000	\$12,000	Value increased because of difference in price between enriched uranium and plutonium.

(a) All other $\text{UO}_2\text{-PuO}_2$ economic parameter values unchanged. A. Strasser indicated L_0 (Loss from Fabrication Plant, kg ^{239}Pu lost/kg ^{239}Pu processed) could be as high as 2% but the group used 1% as was done on the oxide cases. Strasser also indicated that conversion of nitrate to oxide could cost \$700/kg.

(b) The values are based on indicated AEC charge rates and were used by PNL in all studies (i.e., cermet, mixed oxide, and alloy). One offsite participant, C. W. Wheelock, felt that costing plutonium at \$10/gram was not realistic or reasonable for any case before the late 1970's and use of that value produces a distorted conclusion on the value of fueling with enriched uranium for the first few years of operation.

Analytical Input for FFTF Fuel Cycle and Core Parameter
Analog Studies with Mixed Carbide and UC Fuels

The values used in the mixed carbide and UC core analyses are shown in Table XIV, and typical output data are shown in Figures 4 to 11. These are typical designs for the UC and UC-PuC systems. The graphical data show the variation of pin size, fissile throughput, fissile fraction, etc., with test volume and core performance.

The cores are not necessarily optimum FFTF cores using carbide fuels, but were used as reference points for economic evaluation and as cores which satisfy the FTR Testing requirements. Considerations in both the R&D programs and evaluation of the fuel cycle components were based initially on these designs. This allowed a common basis on which to start discussions.

Considerable information was requested from the participants. Manufacturing parameters were supplied describing the cost of fabrication in \$/pin, along with cost penalties for changes in the reference subassembly design, pin diameter, length, fissile content, pins-per-year, and other manufacturing variables. The required extent of these variations may be inferred from the accompanying charts. Cost penalty data were presented as multipliers to be applied to the basic reference manufacturing parameter or in graphical form. Penalties were of the form: $\text{cost} = a (\text{base cost}) \times (\text{size, number, kg, etc.})^b$. Although the group had the option to change the penalty forms, they chose to use the same forms, multipliers, and exponents that were used in the mixed oxide analysis. In addition, reprocessing costs were estimated, based on the reference throughput, and penalties for changes in throughput were included. Assumptions involved in generating the cost, such as plant write-off, fuel handling costs, and losses, were included.

TABLE XIV. *Parameters for Mixed Carbide and UC Core Analyses^(a)—Nomenclature and Reference*

		<u>Mixed Carbide</u>	<u>Uranium Carbide</u>
A ₄	Reprocessing rate, kg/day	20	35
BOND	(b)	Na	Na
BU Avg.	Average allowable burnup, MWd/tonne (Pu + U) (b)	45,000	45,000
C _{BL}	Cost of burnup, plus total ²³⁹ Pu or ²³⁵ U	(b)	(b)
C _{CONV}	Cost of converting nitrate to carbide, \$/yr	(b)	(b)
C _{FAB}	Pin manufacturing cost, \$/yr	(b)	(b)
C _{FC}	Total fuel cycle cost, \$/yr	(b)	(b)
C _h	Subassembly hardware cost, \$/subassembly	1620	1620
C _P	Average heat capacity of Na, Btu/lb-°F	0.311	0.311
C _{P'}	Pin fabrication cost, \$/pin (base value, not given design)	98.90	82.10
C _{Pu}	Value of ²³⁹ Pu or ²³⁵ U, \$/kg ^(c)	10,000	12,000
C _{RP}	Metal recovery costs, \$/yr	(d)	(d)
C _{SR}	Unit scrap recovery cost, \$/value ²³⁵ U or ²³⁹ Pu	0.1	0.1
C _{S1}	Pre-reactor shipping costs, \$/yr	(d)	(d)
C _{S2}	Spent fuel reactor shipping costs, \$/yr	(d)	(d)
C _{S4}	Unit spent fuel shipping cost, \$/kg ²³⁵ U or ²³⁹ Pu Processed	133	133
C' ₁	Unit conversion cost, \$/kg ²³⁵ U or ²³⁹ Pu Processed	575	575

TABLE XIV. (contd)

		<u>Mixed Carbide</u>	<u>Uranium Carbide</u>
C_4	Unit metal reprocessing cost, \$/day	27,600	27,600
D	Core diameter, ft. ^(b)	3.14	3.14
D_E	Fabrication efficiency factor	1.0	1.0
d_i	Pin fuel diameter, in. (nonirradiated), ^(b)	0.185	0.185
d_o	Pin outside diameter, in. ^(b)	0.24	0.24
d_1	Lattice dimension, in.	4.6	4.6
F_A	Axial power peaking factor	1.30	1.30
$F_R F_E$	Radial peaking factor, Engineer- ing peaking factor, ^(b)	1.33	1.33
F_T	Overpower factor (1.1 used throughout group's study)	1.1	1.1
f_{PuC} or f_{UC}	Volume fraction of ^{239}PuC or ^{235}UC	25.0	36.2
f_2	Pin fabrication correction factor for diameter	(e)	(e)
f_3	Pin fabrication correction factor for throughput	(e)	(e)
f_4	Subassembly hardware correction factor for length	(e)	(e)
f_5	Subassembly hardware correction factor for throughput	(e)	(e)
H	Core height, ft. ^(b)	2.83	2.83
K_C	Cladding average thermal con- ductivity, Btu/hr-ft-°F	11.9	11.9
$\int k_f dt$	$\left[\frac{2300^\circ F(b)}{750^\circ F} \right]$ [Btu/hr-ft]	14,700	14,700

TABLE XIV. (contd)

		<u>Mixed Carbide</u>	<u>Uranium Carbide</u>
L ₁	Loss from conversion plant, kg fissile lost/kg fissile processed	0.01	0.01
L _{1S}	Scrap from conversion plant, kg fissile lost/kg fissile processed	0.01	0.01
L ₂	Loss from fabrication plant, kg fissile lost/kg fissile processed	0.01	0.01
L _{2S}	Scrap from fabrication plant, kg fissile scrap/kg fissile processed	0.054	0.054
L ₃	Loss from scrap recovery plant, kg fissile lost/kg fissile processed	0.01	0.01
L ₄	Loss from reprocessing plant, kg fissile lost/kg fissile processed	0.01	0.01
M _C	Critical mass, kg ²³⁹ Pu or ²³⁵ U(b)	375	550
N _{DF}	Number of driver subassemblies ^(b)	44	44
N _P	Number of fuel pins/subassembly ^(b)	166	166
N _{PR}	Number of power producing subassemblies ^(f)	46	46
N _R	Total number of pins/sub- assembly ^(b)	169	169
N _S	Total number of subassemblies	61	61
N _T	Number of fueled tests	10	10
P _F	Plant efficiency factor (operating availability)	0.75	0.75

TABLE XIV. (contd)

		Mixed Carbide	Uranium Carbide
P_T	Core power, MW ^(b)	426	426
ΔP	Core pressure drop, psi	60	60
P_S	Smear density, vol% ^(b)	78	78
T_E	Maximum allowable ceramic fuel temperature, °F ^(b,g)	2300	2300
ΔT	Coolant temperature rise, °F	300	300
T_1	Inlet sodium temperature, °F	500	500
T_2	Outlet sodium temperature, °F	800	800
TP	Fuel throughput, kg fissile/hr ^(b)	620	905
t_T	Reprocessing plant turnaround time, days/batch	8	8
V	Core volume, liters	600	600
PPY	Pins per year throughput, pins/yr	12,000	12,000
ϕ_o	Centerpoint flux, $\frac{10^{16}n}{cm^2 \cdot sec}$	1.08	0.84
$\frac{Di}{t}$	Clad ID-to-clad thickness ratio	14.5	14.5

(a) These data are used in an updated version of an earlier MIDAS computer code application.⁽³⁾

(b) Other values used for individual Cases 1-7 (see Appendix A).

(c) Based on indicated AEC charge rates.

(d) Included in cycle cost analysis but not specifically listed.

(e) Mixed oxide equation form and coefficients used.

(f) $N_{DF} + 0.2 N_T = N_{PR}$

(g) Note - T_L and T_c used in $k_f dt$ equations, T_c = fuel surface temperature in °F.

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FIGURE 4. Centerpoint Flux Versus Number of Fueled Tests
(for 600-Liter Core)

Neg. PNL0672338-18

FIGURE 5. Centerpoint Flux Versus Number of Fueled Tests
(for 800-Liter Core)

Neg. PNL0672338-9

FIGURE 6. Throughput Versus Number of Fueled Tests
(for 600-Liter Core)

Neg. PNL0672338-8

FIGURE 7. Throughput Versus Number of Fueled Tests
(for 800-Liter Core)

Neg. PNL0672338-14

FIGURE 8. Number of Fueled Tests Versus Pins Per Year, $10^3/\text{yr}$ and Fuel Pin OD (Clad) (for 600-Liter Core)

Neg. PNL0672338-15

FIGURE 9. Number of Fueled Tests Versus Pins Per Year, $10^3/\text{yr}$ and Fuel Pin OD (Clad) (for 800-Liter Core)

Neg. PNL0672338-22

FIGURE 10. PuC Volume Fraction Versus Several Parameters

Neg. PNL0672338-23

FIGURE 11. UC Volume Fraction Versus Several Parameters

The carbide cores were mocked up on the computer, and rapid changes could be made in economic, design, and other parameters to assess the effect on cost and core performance as variables were changed. It should be noted that the assumed average burn-up in each supplied case is 45,000 MWd/tonne of heavy metal. The core designs themselves do not change with burnup; only the fissile throughput, pins-per-year, and cost change with burn-up. To accommodate any specified burnup within reasonable limits, the core designs must allow adequate space for pin plenums above the fuel. Since the cost for increased plenum height is not great, it is expected that the cost attributable to this item may be neglected.

FUTURE FFTF CARBIDE FUEL STUDIES

Analytical studies are being continued to investigate the use of carbide fuels for FTR under the Full-Performance Fuel Design Analysis Task which is a part of the FFTF Fuels Development Program. This task examines various fuels as potential candidates for FTR full-performance fuel. These follow-up analyses will reflect design improvements both in the reactor and the final FFTF core design. These design improvements include using a radially two-zoned core which, in the case of the mixed oxide fuel, increases the allowable centerpoint flux (due to the reduction of the radial peaking factor) about 15%, while maintaining the same maximum heat rate in the fuel element. Increasing the core coolant temperature drop from 300 to 400 °F would also provide design improvement by permitting larger fuel pins with lower fissile fractions due to the resultant lower pressure drop for the same power. The carbide-fueled FTR will be examined to explore the effects of increased core outlet temperature. Trade-off studies will be made between flux, test locations, cycle cost, control positions, pressure drop, etc.

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APPENDIX A

COMPLETE DESIGN ANALOG AND ECONOMIC DATA
FOR THE FFTF CARBIDE STUDIES

TABLE A-I. Analog Computer Output Data for Case 1 Fuel

Case No. 1	Run 1	Run 2	Run 3	Run 4	Run 5
Fuel Type: UC, hypo. to stoich., sintered pellets or cast slugs, SS clad, Na bond					
Smear Density, ρ_s					
Bond & (h gap), Material & Btu/hr-ft ² -°F					
	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000
$T_{\text{C}} \text{ max, } ^\circ\text{F}$	2000	1900	1900	2150	2163
$T_{\text{C}} \text{ max, } ^\circ\text{F}$	1164	1147	1147	1198	1187
Overpower Factor	1.1	1.1	1.1	1.1	1.1
Fuel OD, in.	0.196	0.159	0.159	0.232	0.298
Clad OD, in.	0.248	0.207	0.207	0.289	0.363
Core Volume ^(a) , liters	600	600	600	600	600
Core Height, ft.	} or H/D Ratio 0.9	0.9	0.9	0.9	0.9
Core Diameter, ft.					
Fuel Volume Fraction (fissile)	0.34	0.40	0.40	0.30	0.26
$\int_{\text{surface}}^L kdt$ (operating), W/cm	107	96.4	96.4	122	126
Linear Power (max), kW/ft	41.0	37.1	37.1	46.8	48.4
MC (critical mass), kg	540	540	540	540	540
Pins/Subassembly	161	208	208	127	93
Fueled Tests, no.	10	10	10	10	10
Reactor Power, MW	440	513	513	398	296
Fissile Throughput, kg/yr	868	1180	726	703	444
Pins/yr	11,320	19,910	12,270	7,250	3,310
ϕ_0 (center point flux), 10^{16} nv	0.86	1.00	1.00	0.78	0.58
Burnup, MWd/tonne (average)	45,000	45,000	73,180	45,000	45,000
Fuel Cycle Cost, $\$10^6/\text{yr}$	3.98	4.99	3.89	3.36	2.40
CRP (reprocessing) ^(b) , $\$10^6/\text{yr}$	0.87	1.12	0.77	0.74	0.53
C_{Fab} , $\$10^6/\text{yr}$	0.74	1.16	0.76	0.60	0.47
C_{Burnup} , $\$10^6/\text{yr}$	1.60	1.89	1.73	1.40	0.99

(a) Assumes 61 subassemblies

(b) Assumes rate = 35 kg/day

TABLE A-II. Analog Computer Output Data for Case 2 Fuel

Case No. 1	Run 1	Run 2	Run 3	Run 4
Fuel Type: UC-PuC, hypo. to stoich., sintered pellets or cast slugs, SS clad, Na bond				
Smear Density, ρ_s	0.81	0.81	0.81	0.81
Bond δ (h gap), Material δ Btu/hr-ft ² -°F	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000
T_{CL} max, °F	1800	1800	1800	1800
T_C max, °F	1050	1065	1056	1056
Overpower Factor	1.0	1.1	1.1	1.1
Fuel OD, in.	0.194	0.178	0.205	0.205
Clad OD, in.	0.246	0.228	0.258	0.258
Core Volume ^(a) , liters	600	600	600	600
Core Height, ft. } or H/D Ratio	0.9	0.9	0.9	0.9
Core Diameter, ft. }				
Fuel Volume Fraction (fissile)	0.232	0.239	0.214	0.214
$\int_{\text{surface}}^L kdt$ (operating), W/cm	80.6	79.0	80.0	80.0
Linear Power (max), kW/ft	31.0	30.3	30.7	30.7
MC (critical mass), kg	375	375	375	375
Pins/Subassembly	167	192	163	163
Fueled Tests, no.	10	10	10	10
Reactor Power, MW	397	397	341	341
Fissile Throughput, kg/yr	534	414	316	708
Pins/yr	10,450	9,400	6,060	13,400
ϕ_0 (center point flux), 10^{16} nv	1.00	1.00	0.86	0.86
Burnup, MWd/tonne (average)	45,000	60,000	60,000	26,830
Fuel Cycle Cost, $\$10^6$ /yr	3.18	2.82	2.23	3.64
CRP (reprocessing) ^(b) , $\$10^6$ /yr	0.96	0.79	0.66	1.20
C_{Fab} , $\$10^6$ /yr	0.86	0.79	0.62	1.01
C_{Burnup} , $\$10^6$ /yr	0.87	0.86	0.65	0.79

(a) Assumes 61 subassemblies

(b) Assumes rate = 20 kg/day

TABLE A-III. Analog Computer Output Data for Case 3 Fuel

Case No. 3	Run 1	Run 2	Run 3	Run 4	Run 5	Run 6
Fuel Type: UC-PuC, stoich. or slightly hyper., sintered pellets or cast slugs, SS clad, Na bond						
Smear Density, ρ_s	0.81	0.81	0.81	0.81	0.81	0.81
Bond ξ (h gap), Material & Btu/hr-ft ² - °F	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000	Na; 41,000
T_L max, °F	2700	2310	2310	2700	2700	2315
T_C max, °F	1299	1200	1200	1326	1337	1198
Overpower Factor	1.1	1.1	1.1	1.1	1.1	1.1
Fuel OD, in.	0.266	0.231	0.231	0.201	0.182	0.230
Clad OD, in.	0.324	0.288	0.288	0.254	0.233	0.286
Core, Volume ^(a) , liters	600	600	600	600	600	600
Core Height, ft. } Core Diameter, ft. } or H/D Ratio	0.9	0.9	0.9	0.9	0.9	0.9
Fuel Volume Fraction (fissile)	0.210	0.215	0.215	0.260	0.280	0.220
$\int_{\text{Surface}}^L k_{\text{eff}}$ (operating), W/cm	150	119	119	148	146	120
Linear Power (max), kW/ft	57.6	45.8	45.8	56.5	56.0	46.0
MC (critical mass), kg	375	375	375	375	375	375
Pins/Subassembly	101	127	127	140	156	127
Fueled Tests, no.	10	10	10	10	10	10
Reactor Power, MW	398	397	397	538	593	400
Fissile Throughput, kg/yr	476	496	710	812	976	505
Pins/yr	5600	7360	10,530	13,220	17,890	7,540
ϕ_0 (center point flux), 10^{16} nv	1.00	1.00	1.00	1.35	1.49	1.00
Burnup, MWd/tonne (average)	45,000	45,000	45,000	45,000	45,000	45,000
Fuel Cycle Cost, $\$10^6$ /yr	2.69	2.84	3.54	4.50	5.33	3.00
CRP (reprocessing) ^(b) , $\$10^6$ /yr	0.87	0.91	1.20	1.33	1.56	0.92
C_{Fab} , $\$10^6$ /yr	0.72	0.65	0.80	1.00	1.17	0.76
C_{Burnup} , $\$10^6$ /yr	0.77	0.81	0.89	1.41	1.70	0.85

(a) Assumes 61 subassemblies

(b) Assumes rate = 20 kg/day

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TABLE A-IV. Analog Computer Output Data for Case 4 Fuel

Case No. 4	Run 1	Run 2	Run 3	Run 4
Fuel Type: UC-PuC, hyper., sintered pellets, SS Clad, He bond				
Smear Density, ρ_s	0.89	0.89	0.89	0.89
Bond & (h gap), Material & Btu/hr-ft ² -°F	He; 1000	He; 1000	He; 1000	He; 1000
$T_{\text{C}}^{\text{max}}$, °F	3593	4032	3600	3644
T_{C} max, °F	2797	3230	2967	2990
Overpower Factor	1.1	1.1	1.1	1.1
Fuel OD, in.	0.258	0.200	0.173	0.177
Clad OD, in.	0.296	0.238	0.210	0.214
Core Volume ^(a) , liters	600	600	600	600
Core Height, ft. } or H/D Ratio	0.9	0.9	0.9	0.9
Core Diameter, ft.				
Fuel Volume Fraction (fissile)	0.170	0.205	0.221	0.217
$\int_{\text{surface}}^{\text{L}} kdt$ (operating), W/cm	85.0	86.3	68.7	70.1
Linear Power (max), kW/ft	32.8	33.6	26.0	26.6
MC (critical mass), kg	375	375	375	375
Pins/Subassembly	132	177	223	217
Fueled Tests, no.	10	10	10	10
Reactor Power, MW	298	398	398	398
Fissile Throughput, kg/yr	288	476	512	502
Pins/yr	4,430	9,790	13,240	12,700
ϕ_0 (center point flux), 10^{16} nv	0.75	1.00	1.00	1.00
Burnup, MWd/tonne (average)	45,000	45,000	45,000	45,000
Fuel Cycle Cost, $\$10^6/\text{yr}$	1.89	2.91	3.21	3.13
CRP (reprocessing) ^(b) , $\$10^6/\text{yr}$	0.61	0.87	0.92	0.91
C_{Fab} , $\$10^6/\text{yr}$	0.60	0.83	0.966	0.95
C_{Burnup} , $\$10^6/\text{yr}$	0.40	0.76	0.82	0.80

(a) Assumes 61 subassemblies

(b) Assumes rate = 20 kg/day

TABLE A-V. Analog Computer Output Data for Case 5 Fuel

Case No. 5	Run 1	Run 2	Run 3	Run 4
Fuel Type: UC-PuC, hyper., packed powder or low density pellets, SS clad, He bond				
Smear Density, ρ_s	0.80	0.80	0.80	0.80
Bond δ (h gap), Material δ Btu/hr- ft ² -°F	He;2000	He; 2000	He; 2000	He; 2000
T_G max, °F	3370	3213	2815	2523
T_C max, °F	2210	2130	1972	1790
Overpower Factor	1.1	1.1	1.1	1.1
Fuel OD, in.	0.200	0.195	0.166	0.172
Clad OD, in.	0.253	0.246	0.215	0.221
Core Volume ^(a) , liters	600	600	600	600
Core Height, ft. } or H/D Ratio	0.9	0.9	0.9	0.9
Core Diameter, ft. }				
Fuel Volume Fraction (fissile)	0.227	0.229	0.246	0.230
$\int_{\text{surface}}^L kdt$ (operating), W/cm	95.3	88.6	70.0	60.4
Linear Power (max), kW/ft	36.4	34.0	26.6	23.1
MC (critical mass), kg	375	375	375	375
Pins/Subassembly	160	169	217	217
Fueled Tests, no.	10	10	10	10
Reactor Power, MW	398	391	394	344
Fissile Throughput, kg/yr	527	518	563	460
Pins/yr	9,870	10,290	14,260	11,720
ϕ_0 (center point flux), 10^{16} nv	1.00	0.98	1.00	0.86
Burnup, MWd/tonne (average)	45,000	45,000	45,000	45,000
Fuel Cycle Cost, $\$10^6$ /yr	3.13	3.12	3.38	2.93
CRP (reprocessing) ^(b) , $\$10^6$ /yr	0.95	0.94	1.00	0.86
C_{Fab} , $\$10^6$ /yr	0.84	0.86	0.89	0.90
C_{Burnup} , $\$10^6$ /yr	0.85	0.84	0.91	0.75

(a) Assumes 61 subassemblies

(b) Assumes rate = 20 kg/day

TABLE A-VI. Analog Computer Output Data for Case 6 Fuel

Case No. 6	Run 1	Run 2	Run 3
Fuel Type: UC, hyper., packed powder or low density pellets, SS Clad, He bond			
Smear Density, ρ_s	0.80	0.80	0.80
Bond ξ (h gap), Material ξ Btu/hr- ft ² -°F	He; 1000	He; 1000	He; 1000
T_{ξ} max, °F	2926	3797	4736
T_C max, °F	2374	3059	3820
Overpower Factor	1.1	1.1	1.1
Fuel OD, in.	0.177	0.164	0.154
Clad OD, in.	0.227	0.213	0.202
Core Volume ^(a) , liters	600	600	600
Core Height, ft. } Core Diameter, ft. } or H/D Ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile)	0.309	0.357	0.406
$\int_{\text{surface}}^L kdt$ (operating), W/cm	54.0	73.0	93.0
Linear Power (max), kW/ft	20.8	28.1	35.2
MC (critical mass), kg	540	540	540
Pins/Subassembly	217	217	217
Fueled Tests, no.	10	10	10
Reactor Power, MW	306	408	512
Fissile Throughput, kg/yr	546	848	1200
Pins/yr	9,650	14,980	21,190
ϕ_0 (center point flux), 10^{16} nv	0.60	0.80	1.00
Burnup, MWd/tonne (average)	45,000	45,000	45,000
Fuel Cycle Cost, $\$10^6$ /yr	2.73	3.72	5.00
CRP (reprocessing) ^(b) , $\$10^6$ /yr	0.62	0.86	1.14
C_{Fab} , $\$10^6$ /yr	0.66	0.86	1.05
C_{Burnup} , $\$10^6$ /yr	0.93	1.23	1.71

(a) Assumes 61 subassemblies

(b) Assumes rate = 35 kg/day

TABLE A-VII. Analog Computer Output Data for Case 7 Fuel

Case No. 7	Run 1	Run 2	Run 3	Run 4	Run 5
Fuel Type: UC-PuC, hyper., sintered annular pellets, SS clad He bond					
Smear Density, ρ_s	0.80	0.80	0.80	0.80	0.80
Bond δ (h gap), Material δ Btu/hr-ft ² -°F	He; 1000	He; 1000	He; 1000	He; 1000	He; 1000
T _C max, °F	3686	3771	3361	3200	5000
T _C max, °F	3219	3290	2980	2827	4400
Overpower Factor	1.1	1.1	1.1	1.1	1.1
Fuel OD, in.	0.195	0.193	0.160	0.194	0.180
Clad OD, in.	0.247	0.246	0.210	0.246	0.210
Core Volume ^(a) , liters	600	600	600	600	600
Core Height, ft. } Core Diameter, ft. } or H/D Ratio	0.9	0.9	0.9	0.9	0.9
Fuel Volume Fraction (fissile)	0.230	0.231	0.267	0.218	0.307
$\int_{\text{surface}} k_{\text{eff}}$ (operating), W/cm	74.0	74.7	60.0	57.9	100
Linear Power (max), kW/ft	28.3	28.5	24.0	23.2	39.9
MC (critical mass), kg	375	375	375	375	375
Pins/Subassembly	169	169	217	179	180
Fueled Tests, no.	10	10	10	10	10
Reactor Power, MW	389	398	398	342	537
Fissile Throughput, kg/yr	516	537	540	434	1040
Pins/yr	10,280	10,620	13,590	9,060	19,200
ϕ_0 (center point flux), 10^{16} nv	0.98	1.00	1.00	0.86	1.35
Burnup, MWd/tonne (average)	45,000	45,000	45,000	45,000	45,000
Fuel Cycle Cost, $\$10^6$ /yr	3.02	3.12	3.3	2.71	5.65
CRP (reprocessing) ^(b) , $\$10^6$ /yr	0.93	0.95	0.97	0.82	1.38
C _{Fab} , $\$10^6$ /yr	0.86	0.87	1.01	0.79	1.33
C _{Burnup} , $\$10^6$ /yr	0.72	0.79	0.80	0.71	1.59

(a) Assumes 61 subassemblies

(b) Assumes rate = 20 kg/day

APPENDIX B

FUELS DEVELOPMENT SCHEDULE 189a DATA INTERIM MIXED OXIDE FUEL

APPENDIX B

FUELS DEVELOPMENT SCHEDULE 189a DATA INTERIM MIXED OXIDE FUEL

The manpower needs for the reference mixed oxide or cermet fuel for FFTF, as indicated in the Fuels Development Schedule 189a, ^{(8)*} are shown in Table B-I.

TABLE B-I. *Manpower Needs for Proposed R&D Program for Mixed Oxide or Cermet FFTF Driver Fuel FY-1967 through FY-1971*

	<u>Tasks</u>	<u>Engineers and Scientists</u>	
		<u>%</u>	<u>Man-Years</u>
a.	Fuel Process Development	4.0	6.0
b.	Fuel Element Hardware Design and Evaluation	6.0	9.0
c.	Sodium Testing of Fuel Elements and Components	1.0	1.5
d.	Quality Certification Procedures	15.0	22.5
e.	Nondestructive Test Development	5.0	7.5
f.	Irradiation Testing	32.0	48.0
g.	First Core Fabrication Development	11.0	16.5
h.	First Core Fabrication	17.0	25.5
i.	Fuel Element Inspection and Installation	2.0 ^(a)	3.0 ^(a)
j.	Advanced Development	7.0 ^(b)	10.5 ^(b)
	Total	100.0 ^(c)	150.0 ^(c)

(a) *Includes 1.4 man-years occurring in FY-1972.*

(b) *Includes 5.5 man-years occurring in FY-1972.*

(c) *As stated to the group, it was assumed by PNL at that time that three fuels (mixed oxide, cermet, and a third one) would be carried in Task a. plus certain other tasks, then two fuels would be used in Task g. plus certain other tasks, and then one fuel would be used in Task h. plus certain other tasks.*

* *The FFTF Fuels 189a content is presently undergoing extensive revisions from that presented to the group. However, the total manpower (engineer and scientist) requirements have not changed significantly.*

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R. C. Free
E. E. Garrett
R. L. Gibby
S. M. Gill
J. W. Hagan
J. E. Hanson
R. A. Harris
H. Harty
R. A. Harvey
B. R. Hayward (2)
J. W. Helm
L. G. Henke
R. J. Hennig
G. M. Hesson
C. W. Higby
P. L. Hofmann
M. T. Jakub
B. M. Johnson
H. G. Johnson
R. L. Junkins
J. H. Kinginger
R. L. Knecht
D. J. Lagrou
D. D. Lanning
G. A. Last
H. D. Lenkersdorfer
W. R. Lewis
C. W. Lindenmeier
H. E. Little
W. W. Little, Jr.
R. J. Lobsinger
C. E. Love
M. K. Mahaffey
W. B. McDonald
M. H. Meuser
M. K. Millhollen
R. A. Moen
C. A. Munro
D. M. Nero
R. E. Nightingale
W. M. Noble
A. Padilla
M. G. Patrick
L. A. Pember
J. A. Perry
R. E. Peterson
H. G. Powers
F. G. Richards

Battelle-Northwest (contd)

W. E. Roake
J. D. Schaffer
F. H. Shadel
D. W. Shannon (2)
B. O. Shaver
D. E. Simpson
R. J. Squires
G. H. Strong
C. D. Swanson
J. P. Thomas
G. L. Tingey
J. C. Tobin
W. L. Townsend
K. G. Toyoda
M. Vogel
D. M. Walley
J. H. Westsik
L. A. Whinery
R. D. Widrig
M. Wiitala
N. G. Wittenbrock
M. R. Wood
F. W. Woodfield
H. H. Yoshikawa
BNW Central Files (3)
FFTF File (2)
Technical Information File