FAST
TEST FACILITY
PROJECT REPORT**POTENTIAL FFTF ALLOY FUELS**

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DECEMBER 1967

**AEC RESEARCH &
DEVELOPMENT REPORT**

PROJECT NO.	REPORT NO.	DATE
C.R. Harn	32-108-367	FEB 17 1968

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

operated by

BATTELLE MEMORIAL INSTITUTE

for the

UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

PRINTED BY/ FOR THE U. S. ATOMIC ENERGY COMMISSION

3 3679 00060 6501

BNWL-514

UC-25, Metals, Ceramics,
and Materials

POTENTIAL FFTF ALLOY FUELS

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December 1967

NOTICE: PRELIMINARY REPORT

This report contains information of a preliminary nature prepared in the course of work under Atomic Energy Commission Contract AT(45-1)-1830. This information is subject to correction or modification upon the collection and evaluation of additional data.

REPRODUCED FROM THE
ORIGINAL REPORT

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

Printed in the United States of America

Available from

Clearinghouse for Federal Scientific and Technical Information

National Bureau of Standards, U.S. Department of Commerce

Springfield, Virginia 22151

Price: Printed Copy \$3.00; Microfiche \$0.65

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ABSTRACT

An ad hoc working group comprising knowledgeable fuel experts in the United States was convened at Pacific Northwest Laboratory (Battelle-Northwest) to assist PNL in evaluating the potential of plutonium- and/or uranium-enriched alloy driver fuels for the Fast Test Reactor, an integral part of the Fast Flux Test Facility. The members assembled at PNL during July 5-7, 1966, to delineate detailed research and development programs that would be required to assure an adequate alloy fuel element 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 alloy fuel combinations and operating conditions. Analyses of the various alloy-fueled FTR's demonstrated that sodium-bonded alloy fuel, in particular U-Pu-Zr, offers the possibility of an excellent driver fuel from several aspects including cycle cost, flux, performance, and fabrication requirements.

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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 in evaluating the potential of plutonium- and/or uranium-enriched alloy driver fuels⁽¹⁾ for the Fast Test Reactor (FTR), an integral part of the Fast Flux Test Facility.⁽²⁻⁷⁾ This report documents the work performed and includes further computer data generated after adjournment.

The participants assembled at PNL to contribute their experience in fuel evaluation; in fuel element design, fabrication, and testing; in research and development cost estimating and scheduling; and toward delineating detailed research and development programs that would be needed to assure an adequate alloy 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 July 5-7, 1966. Those participating were:

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Also, D. R. deHalas (BNW), W. E. Roake (BNW), R. J. Lobsinger (BNW), and R. J. Myjak (AEC-RLOO) were present for limited times.

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The group spent one day on the determination of the proper input data on economics for the subsequent analog computer runs. They also determined technical restraints such as pin manufacturing cost, reprocessing and conversion costs, maximum allowable fuel centerline temperature, maximum allowable fuel surface temperature, fuel bulk density, overpower factor, and burnup for each fuel case. One subsequent day was required to discuss and outline the detailed research and development program which would be needed to assure that adequate FTR fuel elements of the alloy type would be available by June 1, 1971. Following this, one day was devoted to running specific fuel cases on the PNL computer to determine core operating and economic parameters. An updated version of the MIDAS computer code application,⁽³⁾ developed by PNL for both the design analog and the economic evaluation and fuel cycle studies, was employed. 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, fast flux, etc.).

SUMMARY

The ad hoc working group recommended that four major fuel alloys with two types of cladding be evaluated as potential candidates for an alloy-fueled FTR. The alloys and claddings are listed in Table I.

The U-Pu-Zr alloy was selected as one of the fuels because of the plutonium enrichment and associated low critical mass, and because the sodium-bonded alloy is compatible with stainless steel cladding. The U-Zr alloy is of interest because of the high melting point, the highest among the four alloys proposed. The extensive fabrication and irradiation experience (especially in a fast flux) with U-Mo alloy led to its choice. The "adjusted" uranium was chosen because of the large amount of existing technology.

TABLE I. *Proposed Fuel Cases (all fuels sodium bonded to cladding)*

<u>Case No.</u>	<u>Fuel</u>	<u>Cladding</u>
1a	U-15 wt% Pu-15 wt% Zr	Stainless Steel
1b	U-15 wt% Pu-15 wt% Zr	Vanadium Alloy
2a	U-15 wt% Zr	Stainless Steel
2b	U-15 wt% Zr	Vanadium Alloy
3a	U-10 wt% Mo	Stainless Steel
3b	U-10 wt% Mo	Vanadium Alloy
4a	"adjusted" U ^(a)	Stainless Steel
4b	"adjusted" U ^(a)	Vanadium Alloy

(a) Additives to the uranium include: ~800 ppm aluminum (range: 300 to 1000 ppm), ~400 ppm iron (range: 400 to 1000 ppm), ~500 ppm carbon, and ~100 ppm silicon.

From a current technology standpoint, the fuels should have the relative ratings shown in Table II.

TABLE II. *Relative Ratings of Alloy Fuels*^(a)

<u>Alloy</u>	<u>If Pu is Required</u> ^(b)	<u>If Pu is not required</u> ^(b)
U-Pu-Zr	1	2
U-Zr	4	4
U-Mo	2	1
"adjusted" U ^(c)	3	3

(a) Ratings: 1 is high, 4 is low.

(b) For fast flux and/or physics reasons.

(c) Additives to the uranium include: ~800 ppm aluminum (range: 300 to 1000 ppm), ~400 ppm iron (range: 400 to 1000 ppm), ~500 ppm carbon, and ~100 ppm silicon.

The research and development program proposed for plutonium- and/or uranium-enriched alloy fuels generally covers the period FY-1967 to FY-1972 (first core loading date).

For each of the major alloy fuel types, estimates were made of the manpower (engineers and scientists) required for each task in the proposed research and development program. The total manpower estimates for each fuel type and for the combination are given in Table III.

TABLE III. Summary of Fuel Development Manpower Requirements

<u>Alloy Fuel</u>	<u>Engineers and Scientists, man-years</u>
U-Pu-Zr	95.9
U-Zr	75.3
U-Mo	67.1
"adjusted" U	67.1
Combination ^(a)	109.2 - 122.7

(a) *This is the combination of alloy fuel cases (i.e., carry all four alloys at first, then at a given time, select and carry on with two alloys, and later make a final selection and complete the work with one alloy) as shown on the R&D Schedule (See Discussion, p. 27).*

Based on the assumption that the proposed R&D programs will be supported and followed, it was estimated that a first core fuel loading with any of the four alloys could be commercially fabricated by June 1, 1971, and could achieve a burnup of 45,000 MWd/tonne (average) with the following conditions:

- Stainless steel cladding and a sodium outlet coolant temperature of 800 °F, or
- Vanadium alloy cladding and a sodium coolant outlet temperature of 1200 °F.

Transient tests need to be conducted with all four major alloys. The maximum credible accidents need to be studied and the results of the studies are needed by 1970. There is a need to investigate the problem of sodium expulsion (e.g., what is the effect if a closure fails on the lower end of a fuel pin). It was pointed out that vanadium alloy cladding offers some safety advantages (e.g., higher strength and melting point than stainless steel) in the event of a transient.

With a suitable R&D program extending beyond June 1971, the four major fuels could be commercially fabricated into fuel elements which achieve the following burnups (Table IV) if vanadium alloy cladding were employed.

TABLE IV. Estimated Future Burnup Capability of Alloy Fuels

<u>Case No.</u>	<u>Fuel</u>	<u>Sodium Outlet Temp., °F</u>	<u>Burnup, MWd/tonne</u>
1	U-Pu-Zr	1200	56,000 (average), 100,000 (maximum)
2	U-Zr	1200	56,000 (average), 100,000 (maximum)
3	U-Mo	1200	50,000 (average), (a) (maximum)
4	"adjusted" U	1200	56,000 (average), 100,000 (maximum)

(a) Value not stated.

For the FFTF fuel core and parametric studies, the fuel centerline temperatures were limited to the following values (Table V).

TABLE V. Fuel Centerline Temperature Limits^(a)

<u>Case No.</u>	<u>Fuel Alloy</u>	<u>Cladding</u>	<u>T_C, Centerline Temp., °F</u>
1a	U-Pu-Zr	Stainless Steel	~1290 or ~1470
1b	U-Pu-Zr	Vanadium Alloy	~1470 or ~1740
2a	U-Zr	Stainless Steel	~1290 or ~1470
2b	U-Zr	Vanadium Alloy	~1470 or ~1740
3a	U-Mo	Stainless Steel	~1245 or ~1380
3b	U-Mo	Vanadium Alloy	~1245 or ~1380
4a	"adjusted" U	Stainless Steel	~1200 or ~1380 or ~1470
4b	"adjusted" U	Vanadium Alloy	~1470 or ~1740

(a) For alloy fuels clad with stainless steel, the group imposed another limit--the fuel surface temperature must be <1300 °F (there is an alloy-stainless steel reaction at ~1350 °F).

The economic parameters for the mixed-oxide-fueled FFTF were reviewed and compared with the parameters for alloy fuels. The consensus was that the mixed oxide costs used by PNL were quite representative of a typical fuels process using mixed oxide. In specific areas the alloy fuel costs were modified to reflect the fuels' differences. The changes required to convert the computer input data for mixed oxide fuel* over to input data for alloy fuel included:

- Unit reprocessing costs - value raised 5%
- Conversion costs:
 - (a) For U-Pu-Zr, U-Zr, and U-Mo alloys - value raised 10%
 - (b) For "adjusted" uranium - value not changed
- Pin manufacturing costs (a low and a high cost used for each pin type):
 - (a) Stainless-Steel-Clad Fuel - value lowered 59% or 42%
 - (b) Vanadium-Alloy-Clad Fuel - value lowered 42% or 4.7%

The costs for the plutonium-containing alloy fuels were reduced 20% to arrive at costs for the nonplutonium alloy fuels. The rest of the economic data generated for the mixed oxide were then used for the alloy fuel studies. These data will be discussed in detail throughout this report. The effect of the economic data changes is described below.

The analog computer results for the alloy fuel cases were compared with the data for a cermet-fueled FTR and a mixed-oxide-fueled FTR as shown below:

<u>FTR With Cermet Fuel</u>	
<u>Flux, 10^{16} nv</u>	<u>Reactor Power, MW</u>
1.0	300
1.35	400

* Stainless steel cladding used.

FTR With Mixed Oxide Fuel

<u>Flux, 10^{16} nv</u>	<u>Reactor Power, MW</u>
0.83	360
1.0	437

In the comparisons below, it should be noted that the fuel core volumes employed were:

All alloy fuel cases	600 liters
Cermet fuel cases	600 liters
Mixed oxide fuel cases	800 liters*

In addition, all FTR cores were to meet the same test space requirements (i.e., ability to handle simultaneously 10 to 13 experimental fuel spaces).

With U Pu-Zr alloy fuel one finds that for the same flux, the reactor power is lower than that for mixed oxide fuel but higher than that for cermet fuel. For an average burnup of 45,000 MWd/tonne, the predicted fuel cycle cost with U-Pu-Zr alloy fuel is less than that with mixed oxide or cermet fuels.

The computer results indicate that for the same flux the nonplutonium alloys (U-Zr, U-Mo, and "adjusted" uranium) require a higher reactor power than that for either the cermet or mixed oxide fuel. However, the predicted fuel cycle costs for the nonplutonium alloys are less than those for cermet or mixed oxide when all are evaluated at an average burnup of 45,000 MWd/tonne without regard to flux.

Comparing the results for the U-Pu-Zr alloy fuels with those for the nonplutonium alloy fuels, one notes that for the same flux the reactor power with the latter fuels is higher. The predicted fuel cycle costs with the latter fuels are about the same as or higher than those associated with the former fuels.

* With mixed oxide fuel and the resultant lower flux, one needs this core volume to meet the test space requirements.

The performance for the various sodium-bonded alloy fuels followed the general trend set by the U-Pu-Zr alloy fuel. However, the U-Pu-Zr alloy showed the best performance in satisfying FTR testing requirements.

The major result of this study is to establish analytically the attractive aspects of alloy (in particular U-Pu-Zr) driver fuel for FFTF. Some of these aspects are flux level, safety characteristics, and fuel cycle costs. Specific cases are discussed in the Analog Computer Studies Section.

Table VI highlights the most interesting data for each of the fuel cases considered. As indicated in the notes, the data in Table VI are for an average burnup of 45,000 MWd/tonne. The effect of lowering the average burnup to 25,000 MWd/tonne is to raise the predicted fuel cycle cost 14 to 15%.

DISCUSSION

R&D PROGRAM FOR PROPOSED POTENTIAL FFTF ALLOY FUELS

Assumptions

The following assumptions apply to this study:

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

Fuel Cases Considered

See Summary, Table I.

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

TABLE VI. Summary of the Most Interesting Cases for Each Alloy Fuel Type^(a-h)

Alloy Fuel, wt%	U-15 Pu-15 Zr	U-15 Pu-15 Zr	U-15 Zr	U-15 Zr	U-10 Mo	U-10 Mo	"adj"U ^(b)	"adj"U ^(b)
A. Cladding(C)	SS	V alloy	SS	V alloy	SS	V alloy	SS	V alloy
B. Assumed Fabrication Cost, \$/pin	35	50	28	40	28	40	28	40
C. Linear Power, kW/ft	22.1	30.8	21.6	22.5	21.6	17.9	28.4	40.6
D. Fuel Centerline Temp., °F(°C)	1470(800)	1740(950)	1470(800)	1740(950)	1380(750)	1245(670)	1470(800)	1740(950)
E. Cladding Surface Heat Flux, 10 ⁶ Btu/hr-ft ²	1.1	1.4	1.3	1.4	1.3	1.3	1.9	2.4
F. Fuel Pin Cladding OD, in.	0.257	0.280	0.213	0.212	0.214	0.187	0.199	0.222
G. Reactor Power, MW	310	352	389	403	386	400	467(d)	466(d)
H. Throughput, pins/yr	6280	5990	10,250	11,100	10,350	14,070	12,900	9,420
I. Centerpoint Flux, 10 ¹⁶ nv	0.89	1.0	0.84	0.87	0.83	0.86	1.0(e)	1.0(f)
J. Predicted Fuel Cycle Cost,(g) \$10 ⁶ /yr	1.67	1.93	2.74	2.99	2.78	3.06	3.41	3.54
K. Identification, Case No. (Run No.)	1a(1)	1b(2)	2a(1)	2b(4)	3a(1)	3b(5)	4a(7)	4b(7)

(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). Assume average burnup = 45,000 MWD/tonne. 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; 15.9; 4850(2680); 1.0; 0.200; 360; 19,500; 0.83; and 3.4; and for mixed oxide case; 16.6; 4850(2680); 1.4; 0.160; 437; 36,000; 1.0; and 4.65; respectively. Homogeneous cores used except in cermet case (4 zones).

(b) Additives to the "adjusted" uranium include: ~800 ppm aluminum (range: 300 to 1000 ppm), ~400 ppm iron (range: 400 to 1000 ppm), ~500 ppm carbon, and ~100 ppm silicon.

(c) The stainless steel or vanadium alloy cladding was sodium bonded to the fuel in all cases.

(d) Core power limit exceeded (see a. above).

(e) The test space (handles 15 elements) plus high flux are of interest.

(f) The test space (handles 20 elements) plus high flux are of interest.

(g) Lowering average burnup to 25,000 MWD/tonne increases predicted fuel cycle cost 14 to 15%.

(h) 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	1.0	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 alloy fuel cases.

- It was believed that the core designs (all with up-flowing coolant and vertical, skewed-tube array) allowed adequate space for fuel pin plenums above the fuel to accommodate any specified burnup within reasonable limits.

The U-Pu-Zr alloy was selected as one of the fuels because of the plutonium enrichment and associated low critical mass, and because the sodium-bonded alloy is compatible with stainless steel cladding. The U-Zr alloy is of interest because of the high melting point, the highest among the four alloys proposed. The extensive fabrication and irradiation experience (especially in a fast flux) with U-Mo alloy led to its choice. The "adjusted" uranium was chosen because of the large amount of existing technology.

Current State of Technology

By consensus, the fuels were given the following relative ratings (1 is high, 4 is low) from a current technology standpoint:

<u>Alloy</u>	<u>If Pu Is Required*</u>	<u>If Pu Is Not Required</u>
U-Pu-Zr	1	2
U-Zr	4	4
U-Mo	2	1
"adjusted" U	3	3

Alloys, especially those employed in Fermi (U-Mo alloy) and EBR-II,** can be considered to be proven fuels. Thousands of U-Mo alloy pins have been or are in DFR[†] and Fermi. There are 18 U-Pu-Zr alloy pins in EBR-II at present. Approximately 30,000 sodium-bonded EBR-II pins have been fabricated to date. Numerous fuel pins have been prepared for Hallam Reactor and for the blanket fuel loadings for Atomic Power Development Associates (APDA).

By consensus, the demonstrated burnup capability of alloy fuels at present was determined to be as follows:

* For physics and/or fast flux reasons.

** Experimental Breeder Reactor-II.

† Dounreay Fast Reactor.

TABLE VII. Demonstrated Burnup Capability of Alloy Fuels

Alloy	Demonstrated Burnup		Fuel Temperatures, °F	
	at. %	MWd/tonne	Surface	Center
U-Pu-Zr		70,000	(a)	(a)
U-Zr		0	(a)	(a)
U-Mo		25,000	(b)	1060-1250 (570-675 °C) max.
"adjusted" U	2.7 ⁽⁸⁾		1022 (550 °C)	1292 (700 °C)
	4.75 ⁽⁸⁾		914 (490 °C)	1166 (630 °C)

(a) At temperatures of interest to FFTF

(b) At DFR conditions (note-cladding temperature is 710 to 1000 °F or 380 to 540 °C max.)

The group consensus was that thorium-base alloys should not be overlooked since Th-U alloys, as well as Th-Pu alloys, show much better resistance to swelling than uranium-base alloys.

Four items regarding the design of alloy fuel pins were suggested by consensus. First, one might consider the use of voidage in the fuel plus fuel restraint (use strong cladding). Secondly, one might allow fission gas release by using a plenum and providing 30 to 35% voidage in the fuel. The third item was a recommendation to use the same design concept for fuel pins in Cases 1, 2, and 4. The fourth suggestion was that the pins should not be designed for zero stress in the cladding.

Pyro-reprocessing for application to potential FFTF alloy fuels was not considered at this time, since lower plutonium costs have diminished the incentive for this method in the last ten years. It was suggested that pyro-reprocessing be left out of the analog computer calculations now but that the process should be included in the R&D tasks as an item to be pursued on a long-term basis.

Anticipated Performance of Fuel Fabricated by June 1971

The participants agreed, in all four alloy cases, that a burnup of 45,000 MWd/tonne (average) could be expected with sodium-bonded fuel fabricated by June 1, 1971, and with these conditions:

- Stainless steel cladding and a sodium outlet coolant temperature of 800 °F, or
- Vanadium alloy cladding and a sodium coolant outlet temperature of 1200 °F.

These estimates are based on the assumption that the proposed R&D program is supported and followed.

Probability of Success

With the proposed R&D program, the group believed that a first core loading of any of the four major fuel cases could be completed by June 1, 1971.

Stretch Capability

It was the opinion of the group that, with a suitable R&D program beyond June 1971, the four major fuels could achieve the following burnups if vanadium alloy cladding were employed (Table VIII).

TABLE VIII. *Estimated Future Burnup Capability of Alloy Fuels*

<u>Case No.</u>	<u>Fuel</u>	<u>Sodium Outlet Temp., °F</u>	<u>Burnup, MWd/tonne</u>
1	U-Pu-Zr	1200	56,000 (average), 100,000 (maximum)
2	U-Zr	1200	56,000 (average), 100,000 (maximum)
3	U-Mo	1200	50,000 (average), (a) (maximum)
4	"adjusted" U	1200	56,000 (average), 100,000 (maximum)

(a) *Value not stated.*

Safety Problems

Transient tests need to be conducted with all four major alloy cases. The maximum credible accidents need to be studied and the results of the studies are needed by 1970. There is a need to investigate the problem of sodium expulsion (e.g., what is the effect of a failure of a closure on the lower end of a fuel pin).

Vanadium alloy cladding offers some safety advantages (e.g., higher strength and melting point than stainless steel) in the event of a transient. By consensus, an overpower factor* of 1.0 was employed in the fuel element performance and safety calculations. The melting point basis used for the overpower factor (1.25) for mixed oxide fuel is not valid for alloy fuel.

R&D Program - Tasks

The list of major task headings is shown below:

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

- (a) Fuel Process Development
- (b) Fuel Element Hardware Design and Evaluation
- (c) Sodium Testing of Fuel Elements and Components
- (d) Quality Certification Procedures
- (e) Nondestructive Test Development
- (f) Irradiation Testing
- (g) First Core Fabrication Development
- (h) First Core Fabrication
- (i) Fuel Element Inspection and Installation
- (j) Advanced Development

The task headings and descriptions (see following section) are essentially the same as those employed by PNL in the Fuels Development Schedule 189a.⁽⁹⁾

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

R&D Program-Task Descriptions, General

Before outlining the Fuels Program, the participants reviewed all the FFTF 189 submittals.⁽¹⁰⁾ Thus, 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 alloy fuel for FFTF. The review of the fuels-related FFTF programs indicated that these programs would be adequate for the alloy as well as for the other fuels considered for FFTF. The task descriptions are as follows:

(a) Fuel Process Development

The overall 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 containment of the fuel in a cladding that is compatible with the fuel and sodium bond and coolant. The technology for fast reactors requires the investigation of several candidate alloy fuel systems (e.g., U-Pu-Zr, U-Zr, U-Mo, "adjusted" uranium). Development of the driver fuel system will include three phases:

1. A process evolution period for selecting the leading process of a given system
2. A fabrication demonstration period in which pilot-scale operation will be initiated for prototypic fuel fabrication
3. A fabrication period for manufacturing the first core loading.

The scope of this task includes the first phase listed above.

(b) Fuel Element Hardware Design and Evaluation

The overall 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, turbulence, 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 overall 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 following up to see 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 nondestructive 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 fuels 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 overall objective of this task is the demonstration of a feasible process for the fabrication of driver fuel elements for the FFTF.

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 overall 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- 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 best candidate 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 preinstallation 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. Techniques will be established for the inspection (if required) 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 an advanced alloy driver fuel, to fabricate one-third to one-half of a core loading of the advanced driver 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 alloy 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

(a) Task - Fuel Process Development

The participants stated the following specific R&D needs for these alloys:

<u>U-Pu-Zr</u>	<u>U-Zr</u>	<u>U-Mo</u>	<u>U ("adjusted")</u>
<ul style="list-style-type: none"> • There is a need to study various homogenizing heat treatments to produce a ductile alloy and to minimize undesirable phase transformations. The ductility is needed at elevated temperatures to allow mechanical fabrication and to permit fuel deformation during irradiation without cracking. • Little work is needed on sodium bonding techniques since they are well established for this alloy. • If coextruded fuel is used and fuel-cladding compatibility is a problem, one should investigate the use of a thin diffusion-barrier layer. The layer could be coextruded on the alloy fuel or on the cladding. • There is a need to study alloy melting and synthesis. The problem with zirconium is that it floats during melting of the alloy-needs to be solved. 	<ul style="list-style-type: none"> • Extensive process development work is needed since fabrication techniques have yet to be developed for this alloy. Extrusion needs to be studied as a fabrication process. • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> • Essentially no process development work is needed since technology exists now for this alloy. However, there is a need to determine the feasibility of extrusion-swaging as a fabrication process. • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. • Process development work on injection casting, a desirable fabrication technique, needs to be done. Injection casting of this fuel alloy has not been demonstrated to date. 	<ul style="list-style-type: none"> • Same as U-Mo entry. • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. • Same as U-Mo entry.

<u>U-Pu-Zr</u>	<u>U-Zr</u>	<u>U-Mo</u>	<u>U ("adjusted")</u>
	<ul style="list-style-type: none"> Physical and mechanical properties need to be determined. There are no property data available for this alloy. 		
(b) <u>Task - Fuel Element Hardware Design and Evaluation</u>			
<ul style="list-style-type: none"> The R&D needs for this task with this fuel will be essentially the same as those pointed out in the mixed oxide fuel case. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry.
(c) <u>Task - Sodium Testing of Fuel Elements and Components</u>			
<ul style="list-style-type: none"> The R&D needs for this task with this fuel will be essentially the same as those pointed out in the mixed oxide fuel case. In comparison to the mixed oxide or carbide fuel cases, less effort will be required on fuel washout tests and studies with this alloy. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry.
(d) <u>Task - Quality Certification Procedures</u>			
<ul style="list-style-type: none"> Extensive work needs to be done in this task to assure that the FFTF will have safe, economic, and reliable fuel cores. The R&D needs are essentially the same as those indicated for the mixed oxide fuel case. Much work has been done in the past on uranium-enriched alloy fuels and some on plutonium-enriched alloy fuels. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> Same as U-Pu-Zr entry.

<u>U-Pu-Zr</u>	<u>U-Zr</u>	<u>U-Mo</u>	<u>U ("adjusted")</u>
(e) <u>Task - Nondestructive Test Development</u>			
<ul style="list-style-type: none"> • There is a need to improve existing techniques or to establish new techniques for nondestructively assessing fuel pin and element quality. The major areas of concern are: sodium bond integrity, sodium level measurement, integrity of closures, and pin quality (e.g., size, weight, density) • There is a need to study methods for detecting a nonbond area in the sodium bond layer. It is important to be able to measure the voids experimentally and establish the allowable void size. Knowing this void size can result in a lower reject rate during pin fabrication. A current infrared technique for sodium level determination offers a good possibility for a sodium bond test method. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Same as U-Pu-Zr entry.
(f) <u>Task - Irradiation Testing</u>			
<ul style="list-style-type: none"> • See note below: 	<ul style="list-style-type: none"> • See U-Pu-Zr entry. 	<ul style="list-style-type: none"> • See U-Pu-Zr entry. 	<ul style="list-style-type: none"> • See U-Pu-Zr entry.

Note: *There is a need to demonstrate that the irradiation behavior of alloy fuel pins and elements can meet exacting standards and thus assure a safe, economic, and reliable FFTF alloy core loading. To do this, it was estimated that about 100 pins (U-Pu-Zr, U-Zr, U-Mo and "adjusted" U alloy), total, need to be tested in a fast flux. One*

U-Pu-Zr

U-Zr

U-Mo

U ("adjusted")

could probably put two subassemblies (38 encapsulated pins/subassembly) per year into EBR-II. If the pins are the nonencapsulated type, then 61 pins per subassembly can be used. The total number of pins per year (76 or 122) includes all four major alloy types. For tests in Fermi, 1.5 to 3 core positions might be available. This means that 72 unencapsulated pins (maximum) can be tested. It may be necessary to use encapsulated pins for U-Pu-Zr, U-Zr, and "adjusted" uranium alloy (Cases 1, 2, and 4) tests. The U-Mo alloy (Case 3) pins can probably be of the unencapsulated type.

The effort on U-Pu-Zr alloy (Case 1) should equal the combined efforts on U-Zr, U-Mo, and "adjusted" uranium alloys (Cases 2, 3, and 4).

For adequate quality and performance assurance, there is a need to conduct tests over a range of fission rates, burnups, and temperatures to study the characteristics of the fuel pins. Some of these characteristics are swelling, sodium-fuel-cladding compatibility, anisotropic growth, nucleation of fission gas bubbles, restraint by strong cladding, straightness, and void deployment.

The performance of the fuels needs to be demonstrated under overpower conditions to verify that the fuels meet the safety and reactor criteria.

There is a need to conduct transient tests. The maximum credible accidents need to be evaluated and the results are needed by 1970. The sodium expulsion problem (e.g., what is the effect of a failure of a closure on the lower end of a fuel pin) needs to be studied.

Thermal flux tests should be initiated at an early date (start in FY-1967, if possible) and used to screen the potential alloy fuel choices, to study various weld closures, and to facilitate fast-thermal flux reactor test data comparisons.

There is a need to study and test various methods for holding the fuel down in the sodium-bonded pins. Some work on this problem (an important one since small geometry changes in the reactor core can cause marked changes in reactor operation) is currently in progress.

Studies need to be conducted to establish the effect of burnup on fuel physical and mechanical properties, such as melting point, thermal expansion, etc. The tests should cover the range of 100 to 200% of anticipated goal values.

<u>U-Pu-Zr</u>	<u>U-Zr</u>	<u>U-Mo</u>	<u>U ("adjusted")</u>
(g) <u>Task - First Core Fabrication Development</u>			
<ul style="list-style-type: none"> • To meet the schedule goals of this task, there is a need to select the two most promising fuels (probably one plutonium-enriched and one uranium-enriched type) from the group of four studied in the Fuel Process Development Task (see Task a above). The selection needs to be made as soon as this task gets underway. • Because of the plutonium enrichment and the associated handling and facilities problems with plutonium, the U-Pu-Zr alloy fuel (Case 4) will require more (~6 mo) development time than any of the other three alloy fuel cases. To meet schedular goals in this task and related tasks (e.g., Irradiation Testing), the work should be initiated about the end of FY-1967. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Since facilities are currently available, this task could start in the middle of FY-1967. However, it probably would not be started until the end of FY-1968 because one would like to have as much irradiation test data available as soon as possible before choosing the two fuels. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Same as U-Zr entry. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry. • Same as U-Zr entry.
(h) <u>Task - First Core Fabrication</u>			
<ul style="list-style-type: none"> • It was estimated that about 2 yr would be required to complete this task with this alloy fuel, meeting applicable goals and producing a safe, economic, and reliable first core. These goals include those specified in the R&D schedule for this task, 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry, except the task could be completed in 1 1/2 yr with this alloy fuel. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry, except the task could be completed in 1 1/2 yr with this alloy fuel. 	<ul style="list-style-type: none"> • Same as U-Pu-Zr entry, except the task could be completed in 1 1/2 yr with this fuel.

U-Pu-Zr	U-Zr	U-Mo	U ("adjusted")
the reactor operational goals (e.g., loading date for first core), and the quality assurance goals (see Quality Certification and Inspection and Installation Tasks).			
(i) <u>Task - Fuel Element Inspection and Installation</u>			
• The R&D needs for this task with this fuel will be essentially the same as those pointed out in the mixed oxide fuel case.	• Same as U-Pu-Zr entry.	• Same as U-Pu-Zr entry.	• Same as U-Pu-Zr entry.
(j) <u>Task - Advanced Development</u>			
• See note below.	• Same as U-Pu-Zr entry.	• Same as U-Pu-Zr entry.	• Same as U-Pu-Zr entry.

Note: *Coextrusion studies need to be conducted with stainless steel-clad alloy fuels. These metallurgically bonded fuels may be attractive because of safety, economic, reliability, or performance characteristics. Alternate fuel geometries (e.g., annular) and fabrication techniques (e.g., adding diffusion barriers) need to be studied since they may offer a means by which fuel pins with superior operational characteristics (e.g., less swelling and less diffusion) can be produced.*

Fuel compartmentalization techniques may need to be evaluated. Reasons why segmented fuel cores might be considered include: inherent limitations in a given fabrication method (e.g., length/diameter ratio limits), improved operational performance of the fuel pin, and less severe problems during a transient.

The possibility of a built-in reference point on sodium-bonded fuel pins should be studied. The reference point would allow one to detect more accurately sodium level changes (e.g., by neutron radiography) and thus monitor fuel volume changes nondestructively.

R&D Program - Schedule

The schedule for the proposed R&D program for U-Pu-Zr, U-Zr, U-Mo, and "adjusted" uranium alloy fuel development is shown in Figure 1. Notice that all four alloys are carried at first, then at a given time two are selected, and later a final selection is made and the work completed with one alloy.

FIGURE 1. Schedule for Proposed R&D Program for U-Pu-Zr, U-Zr, U-Mo, and "Adjusted" Uranium Alloy Fuel Development for FFTF.

It should be noted that the First Core Fabrication Task could be completed in 2 yr with the U-Pu-Zr alloy, starting at the end of FY-1969. With the other alloy fuels, the task could be completed in 1 1/2 yr, starting in the middle of FY-1970.

R&D Program - Manpower Needs

The group estimated, for each of the four major alloy cases, the manpower (engineers and scientists) required for each task shown above. The estimates are shown in Table IX.

Facilities Required

(a) Facilities Required for Proposed R&D Programs

Both AEC and commercial facilities are available now for research and development work on alloy fuels.

(b) Facilities Required for Pilot Plant and Production Operations

For nonplutonium alloy fuels, both AEC and commercial facilities are available now. For plutonium-containing alloy fuels, both AEC and commercial facilities will be available in about 1 yr. The group indicated that a casting facility may be needed for U-Pu-Zr alloy (Case 1) studies.

Site Recommendations

(a) Site Recommendations for Proposed R&D Program

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

(b) Site Recommendations for Production

It was recommended that commercial sites be employed for the proposed production work, but the group declined to specify particular sites.

TABLE IX. Manpower Needs for Proposed Potential FFTF Alloy Fuels

Alloy Fuel Data:	Fuel Case No.	1	2	3	4	
	Fuel Alloy	U-Pu-Zr	U-Zr	U-Mo	"Adjusted" U	Combination
	Core-Clad Bond	Sodium				of cases as
	Cladding	Stainless steel or vanadium alloy				indicated on
						R&D schedule
						(Discussion,
						p. 27)

ENGINEERS AND SCIENTISTS, MAN-YEARS**

R&D Task:					
(a) Fuel Process Development	1.1	1.5	0.8	0.8	4.2
(b) Fuel Element Hardware Design and Evaluation	13.5	13.5	13.5	13.5	13.5
(c) Sodium Testing of Fuel Elements and Components	2.3	2.3	2.3	2.3	2.3
(d) Quality Certification Procedures	26.3	26.3	18.8	18.8	18.8-26.3
(e) Nondestructive Test Development	1.5	1.5	1.5	1.5	1.5
(f) Irradiation Testing*	22.5	7.5	7.5	7.5	45.0
(g) First Core Fabrication Development	3.2	1.2	1.2	1.2	2.4-4.4
(h) First Core Fabrication	6.0	2.0	2.0	2.0	2.0-6.0
(i) Fuel Element Inspection and Installation	4.5	4.5	4.5	4.5	4.5
(j) Advanced Development	15.0	15.0	15.0	15.0	15.0
TOTALS	95.9	75.3	67.1	67.1	109.2-122.7 [†]

* Estimates do not include the fabrication of irradiation test elements, capsules, etc.

** The absolute numbers of man-years are best estimates at this time, but comparisons, on a relative basis, of the alloy cases with the present FFTF fuels program (See Appendix Table B-1) should be reasonably good ones. The 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 FY-1971. One would probably see a decrease in manpower needs in the period after the irradiation tests have been started but before the start of first core fabrication.

† McHugh feels that these numbers may be low.

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 shape of the conventional learning curve.

ANALOG COMPUTER STUDIES ON PROPOSED ALLOY FUELS FOR FFTF

Summary of the Alloy Fuel Analog Computer Studies

The analog computer results for the alloy fuel cases were compared with the data for a cermet-fueled FTR and a mixed-oxide-fueled FTR as shown below:

FTR With Cermet Fuel

<u>Flux, 10^{16} nv</u>	<u>Reactor Power, MW</u>
1.0	300
1.35	400

FTR With Mixed Oxide Fuel

<u>Flux, 10^{16} nv</u>	<u>Reactor Power, MW</u>
0.83	360
1.0	437

All FTR cores were to meet the same test space requirements (i.e., ability to handle simultaneously 10 to 13 experimental fuel spaces), and in the comparisons below, it should be noted that the fuel core volumes employed were:

<u>Fuel Case</u>	<u>Core Volume, liters</u>
All alloys	600
Cermet	600
Mixed oxide	800*

* With mixed oxide fuel and the resultant lower flux, one needs this core volume to meet the test space requirements.

With U-Pu-Zr alloy fuel one finds that for the same flux the reactor power is lower than that for mixed oxide fuel but higher than that for cermet fuel. For an average burnup of 45,000 MWd/tonne, the predicted fuel cycle cost with U-Pu-Zr alloy fuel is less than that with mixed oxide or cermet fuels.

The computer results indicate that for the same flux the nonplutonium alloys (U-Zr, U-Mo, and "adjusted" uranium) require a higher reactor power than that for either the cermet or mixed oxide fuel. However, the predicted fuel cycle costs for the nonplutonium alloys are less than those for cermet or mixed oxide when all are evaluated at an average burnup of 45,000 MWd/tonne.

Comparing the results for the U-Pu-Zr alloy fuels with those for the nonplutonium alloy fuels, one notes that for the same flux the reactor power with the latter fuels is higher. The predicted fuel cycle costs for the latter fuels are about the same as or are higher than those associated with the former fuels.

The performance for the various sodium-bonded alloy fuels followed the general trend of U-Pu-Zr alloy fuel showing the best performance in satisfying FTR testing requirements, followed by the nonplutonium alloys.

The major result of this study was to establish analytically the attractive aspects of using alloy (in particular U-Pu-Zr) driver fuel for FFTF. Some of these aspects are flux level, safety characteristics, and fuel cycle costs.

Limitations

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

- An alloy surface temperature limit of 1300 °F (maximum) was set since an interface reaction between the stainless steel cladding and the alloy fuels starts at temperatures ≥ 1350 °F.

- If the mixed mean sodium coolant outlet temperature is limited to ≤ 800 °F, the mechanical properties of stainless steel cladding are such that alloy fuels with this cladding can attain an average burnup of 45,000 MWd/tonne.
- If the mixed sodium coolant outlet temperature is limited to ≤ 1200 °F, the mechanical properties of vanadium alloy cladding are such that alloy fuels with this cladding can attain an average burnup of 45,000 MWd/tonne.
- With the U-Pu-Zr alloy (solidus temperature is ~ 2120 °F or 1160 °C), the plutonium and zirconium concentrations should be held constant at 15 wt% each. If the plutonium content is increased above 15 wt%, the effect is to significantly lower the melting point of the alloy. It was recommended that ^{235}U enrichment be added to the alloy to supplement the plutonium enrichment to meet the reactor fissile material requirements.

Summary of Fuel Cases, Performance, and Economic Data

The following Tables X through XVII list the important core parameters for the alloy cases studied. A complete data list for the 56 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, fuel bulk density, gap coefficient, etc.

The values listed below are used consistently for the cases shown in the tables and 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	1.0
ΔT	Coolant temperature rise, °F	300

Parameter Symbol	Parameter	Assigned Parameter Value
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 sub-assemblies	61
N_T	Number of fueled tests	10
N_R	Number of control positions	6
P_F	Plant efficiency factor (operating availability)	0.75

It is important to note that the analog computer data for the U-Pu-Zr alloy fuel cases shown in Tables X, XI, A-I, and A-II are based on the assumption that all the fissile material is plutonium. The fissile fuel content is greater than the 15 wt% Pu in the alloy, so, as stated above, some ^{235}U enrichment will be required. Because the analog computer is not presently capable of coping with the variable dual enrichment scheme, the tabular data herein for this alloy does not reflect this substitution of ^{235}U for some of the ^{239}Pu . This substitution won't affect the design but will affect a number of parameters; in particular it will increase the critical mass* and reactor power (only slight effect on this) to yield the same flux in a given case. The substitution will also raise the fuel throughput and the predicted fuel cycle cost in a given case, but it is difficult to make a general statement on the magnitude of the change, except to say that it is estimated to be small.

* For example, in an analog computer case which calls for a fissile fraction of 20 vol% PuO_2 and a critical mass of 375 kg of ^{239}Pu , the required substitution to hold the plutonium content to 15 wt% in the fuel would result in the use of a critical mass of ~244 kg of ^{239}Pu plus ~170 kg of ^{235}U .

TABLE X. Case 1a Fuel (sodium-bonded, U-Pu-Zr alloy fuel in stainless steel cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 800 °C (~1470 °F) or 700 °C (~1290 °F)
- Maximum allowable fuel surface temperature to be 1300 °F
- Cladding surface temperature (on coolant side) to be <1000 °F^(a)
- Fuel bulk (smear) density ($\bar{\rho}_s$) to be 75 or 80% of theoretical
- Fuel composition to be U-15 wt% Pu-15 wt% Zr (the group recommended holding the plutonium and zirconium percentages constant and changing the ^{235}U content to obtain the proper fissile content. If more plutonium is added, the melting point of the fuel alloy decreases.)
- Burnup to be 45,000 MWd/tonne (average)^(a) except for two runs at 25,000 MWd/tonne (average), and
- Some of the tabular entries below are incorrect, since the results do not take into account the appropriate substitution of ^{235}U for some of the ^{239}Pu , and need further refinement.^(b)

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	35	35	35	35	35	50	35	35	35
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1470	1470	1470	1470	1290	1470	1470	1470	1470
Pins/Subassembly	169	195	217	217	217	217	217	217	217
Pin OD, in.	0.257	0.232	0.213	0.213	0.225	0.213	0.211	0.211	0.211
Fissile Fuel Fraction, % ^(b)	18.2	19.4	20.5	20.5	18.5	20.5	19.7	22.6	19.6
Reactor Power, MW	310	353	390	390	304	390	408	353	407
Pins/yr	6280	8710	11,360	20,460	7950	11,370	11,390	11,390	20,460
No. of Test Positions	10	10	10	10	10	10	10	17.7	10
Flux, 10^{16} nv	0.89	1.00	1.11	1.11	0.87	1.11	1.16	1.00	1.16
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	45,000	25,000
Fuel Cycle Cost, \$10/yr	1.67	1.99	2.29	3.36	1.70	2.41	2.29	2.17	3.35

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

(b) Important Note: See Summary of Fuel Cores, Performance, and Economic Data, last paragraph, p. 33.

TABLE XI. Case 1b Fuel (sodium-bonded, U-Pu-Zr alloy fuel in vanadium alloy cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 950 °C (~1740 °F) or 800 °C (~1470 °F)
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density (δ_s) to be 75 or 80% of theoretical
- Fuel composition to be U-15 wt% Pu-15 wt% Zr (see notes in Case 1a, above)
- Burnup to be 45,000 MWd/tonne (average) except for two runs at 25,000 MWd/tonne (average), and
- Some of the tabular entries below are incorrect, since the results do not take into account the appropriate substitution of ^{235}U for some of the ^{239}Pu , and need further refinement.^(a)

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	50	50	50	50	50	90	50	50	50
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1470	1740	1740	1740	1470	1740	1740	1740	1740
Pins/Subassembly	169	139	217	169	169	169	169	169	169
Pin OD, in.	0.241	0.280	0.197	0.241	0.256	0.241	0.239	0.239	0.239
Fissile Fuel Fraction, % ^(a)	21.3	19.1	24.9	21.3	18.8	21.3	20.1	20.1	25.1
Reactor Power, MW	423	352	531	423	317	423	443	443	355
Pins/yr	9730	5990	18,480	17,690	6460	9730	9950	17,890	11,990
No. of Test Positions	10	10	10	10	10	10	10	10	22
Flux, 10^{16} nv	1.20	1.00	1.50	1.20	0.90	1.20	1.27	1.27	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	2.49	1.93	3.56	3.68	1.76	2.76	2.59	3.80	3.46

(a) Important note: See Summary of Fuel Cores, Performance, and Economic Data, last paragraph, p. 33.

TABLE XII. Case 2a Fuel (sodium-bonded, U-Zr alloy fuel in stainless steel cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 800 °C (~1470 °F) or 700 °C (~1290 °F)
- Maximum allowable fuel surface temperature to be 1300 °F
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density ($\bar{\rho}_s$) to be 75 or 80% of theoretical
- Fuel composition to be U-15 wt%Zr, and
- Burnup to be 45,000 MWd/tonne (average); however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost \$/Pin	28	28	28	28	28	40	28	28	28
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1470	1470	1470	1470	1290	1470	1470	1470	1470
Pins/Subassembly	217	267	271	271	271	271	271	271	271
Pin OD, in.	0.213	0.182	0.179	0.179	0.191	0.179	0.177	0.177	0.177
Fissile Fuel Fraction, %	24.6	27.7	28.0	28.0	24.6	28.0	26.9	26.9	28.4
Reactor Power, MW	389	467	473	473	369	474	494	494	468
Pins/yr	10,250	17,100	18,850	32,290	12,250	17,880	17,930	32,260	16,250
No. of Test Positions	10	10	10	10	10	10	10	10	13
Flux, 10^{16} nv	0.84	1.00	1.02	1.02	0.80	1.04	1.06	1.07	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	2.74	3.49	3.57	5.01	2.69	3.68	3.63	5.07	3.47

TABLE XIII. Case 2b Fuel (sodium-bonded, U-Zr alloy fuel in vanadium alloy cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 950 °C (~1740 °F) or 800 °C (~1470 °F)
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density ($\bar{\rho}_g$) to be 75 or 80% of theoretical
- Fuel composition to be U-15 wt% Zr, and
- Burnup to be 45,000 MWd/tonne (average); however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8
Fabrication cost, \$/Pin	40	40	40	72	40	40	40	40
Smear Density	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1740	1740	1740	1740	1470	1740	1740	1740
Pins/Subassembly	169	217	217	217	217	217	217	217
Pin OD, in.	0.240	0.197	0.196	0.196	0.212	0.194	0.194	0.194
Fissile Fuel Fraction, %	25.3	29.6	29.7	29.7	25.4	28.6	28.6	34.4
Reactor Power, MW	426	534	534	537	403	560	561	468
Pins/yr	9140	17,120	30,800	17,270	11,100	17,330	31,180	12,760
No. of Test Positions	10	10	10	10	10	10	10	20
Flux, 10^{16} nv	0.92	1.15	1.15	1.16	0.87	1.21	1.21	1.00
Average Burnup, MWd/tonne	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, $\$10^6$ /yr	3.08	4.14	5.86	4.47	2.99	4.25	5.98	3.68

TABLE XIV. Case 3a Fuel (sodium-bonded, U-Mo alloy fuel in stainless steel cladding)

- Maximum allowable fuel centerline temperature (T_{c}) to be 750 °C (~1380 °F) or 675 °C (~1245 °F)
- Maximum allowable fuel surface temperature to be 1300 °F
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density ($\bar{\rho}_s$) to be 75 or 80% of theoretical
- Fuel composition to be U-10 wt% Mo (solidus temperature is 2120 °F or 1160 °C), and
- Burnup to be 45,000 MWd/tonne (average); however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	28	28	28	28	40	28	28	28	28
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_{c} , °F	1380	1380	1380	1380	1380	1245	1380	1380	1380
Pins/Subassembly	217	268	271	271	271	271	271	271	271
Pin OD, in.	0.214	0.186	0.180	0.180	0.180	0.191	0.178	0.178	0.178
Fissile Fuel Fraction, %	24.8	28.1	28.2	28.3	28.2	25.2	27.1	27.1	28.2
Reactor Power, MW	386	465	469	469	468	383	489	488	466
Pins/yr	10,350	17,500	17,800	32,200	18,050	13,070	17,690	32,100	16,410
No. of Test Positions	10	10	10	10	10	10	10	10	10
Flux, 10^{16} nv	0.83	1.00	1.01	1.01	1.01	0.83	1.05	1.05	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	2.78	3.52	3.59	5.04	3.71	2.83	3.65	5.10	3.51

TABLE XV. Case 3b Fuel (sodium-bonded, U-Mo alloy fuel in vanadium alloy cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 750 °C (~1380 °F) or 675 °C (~1245 °F)
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density (ρ_s) to be 75 or 80% of theoretical
- Fuel composition to be U-10 wt% Mo (solidus temperature is 2120 °F or 1160 °C); however, McHugh expressed interest in evaluating two other compositions (U-7 wt% Mo and U-9 wt% Mo) (a) also, and
- Burnup to be 45,000 MWd/tonne (average) however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	40	40	40	40	72	40	40	40	40
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1380	1380	1380	1380	1380	1245	1380	1380	1380
Pins/Subassembly	217	255	271	271	271	271	271	271	271
Pin OD, in.	0.211	0.187	0.177	0.177	0.177	0.187	0.175	0.175	0.175
Fissile Fuel Fraction, %	25.6	27.9	29.2	29.1	29.2	26.0	28.0	28.1	31.0
Reactor Power, MW	405	466	492	492	492	400	510	512	466
Pins/yr	11,210	16,540	19,340	34,880	19,350	14,070	19,340	34,770	16,440
No. of Test Positions	10	10	10	10	10	10	10	10	15
Flux, 10^{16} nv	0.87	1.00	1.06	1.06	1.06	0.86	1.10	1.10	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	3.01	3.61	3.90	5.50	4.22	3.06	3.95	5.54	3.67

(a) No analog computer runs were made with these DFR fuel alloy compositions.

TABLE XVI. Case 4a Fuel (sodium-bonded, "adjusted" uranium alloy fuel in stainless steel cladding)

- Maximum allowable fuel centerline temperature (T_{CL}) to be 800 °C (~1470 °F) or 750 °C (~1380 °F) or 650 °C (~1200 °F)
- Maximum allowable fuel surface temperature to be 1300 °F
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density ($\bar{\rho}_g$) to be 75 or 80% of theoretical
- Fuel composition to be uranium with ~2000 ppm of additives, and
- Burnup to be 45,000 MWd/tonne (average); however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	28	28	28	28	40	28	28	28	28
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_{CL} , °F	1470	1470	1470	1470	1470	1340	1470	1470	1470
Pins/Subassembly	217	204	271	271	271	271	217	217	217
Pin OD, in.	0.201	0.212	0.166	0.166	0.166	0.176	0.199	0.199	0.199
Fissile Fuel Fraction, %	28.3	27.1	33.2	33.2	33.2	29.4	27.1	27.0	29.6
Reactor Power, MW	494	466	594	595	594	500	510	510	467
Pins/yr	15,140	12,770	26,660	47,990	26,730	19,880	14,980	26,910	12,900
No. of Test Positions	10	10	10	10	10	10	10	10	15
Flux, 10^{16} nv	1.06	1.00	1.28	1.28	1.28	1.08	1.10	1.10	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	3.64	3.35	4.77	6.82	4.91	3.81	3.65	5.10	3.41

TABLE XVII. Case 4b Fuel (sodium-bonded, "adjusted" uranium alloy fuel in vanadium alloy cladding)

- Maximum allowable fuel centerline temperature (T_E) to be 950 °C (~1740 °F) or 800 °C (~1470 °F)
- Cladding surface temperature (on coolant side) to be <1000 °F (see note in Case 1a, above)
- Fuel bulk (smear) density ($\bar{\rho}_s$) to be 75 or 80% of theoretical
- Fuel composition to be uranium with ~2000 ppm of additives, and
- Burnup to be 45,000 MWd/tonne (average); however, investigate the effect of lowering burnup to 25,000 MWd/tonne (average).

Run Number	1	2	3	4	5	6	7	8	9
Fabrication Cost, \$/Pin	40	40	40	40	72	40	40	40	40
Smear Density	0.75	0.75	0.75	0.75	0.75	0.75	0.80	0.80	0.80
T_E , °F	1740	1740	1740	1740	1740	1470	1740	1740	1740
Pins/Subassembly	169	141	217	217	217	217	169	169	169
Pin OD, in.	0.226	0.261	0.181	0.181	0.181	0.199	0.223	0.223	0.222
Fissile Fuel Fraction, %	28.7	25.7	34.7	34.6	34.6	28.9	27.5	27.5	33.9
Reactor Power, MW	549	467	680	680	679	515	571	572	466
Pins/yr	13,280	8,390	25,400	45,650	25,500	16,120	13,250	23,770	9,420
No. of Test Positions	10	10	10	10	10	10	10	10	21
Flux, 10^{16} nv	1.18	1.00	1.46	1.46	1.46	1.11	1.23	1.23	1.00
Average Burnup, MWd/tonne	45,000	45,000	45,000	25,000	45,000	45,000	45,000	25,000	45,000
Fuel Cycle Cost, \$10 ⁶ /yr	4.08	3.31	5.58	8.02	5.97	3.93	4.14	5.77	3.54

General Technical and Economic Input Data for the Parametric Studies

Changes Required to Modify UO_2 - PuO_2 Input Data to Alloy Input Data

The changes made to modify UO_2 - PuO_2 economic data for use in the alloy fuels cases are shown in Table XVIII. The nomenclature and reference values, used in the UO_2 - PuO_2 cases, are shown in Table XIX.

Analytical Input for FFTF Fuel Cycle and Core Parameter Analog Studies With Alloy Fuels

The values used in the alloy core analyses are shown in Table XIX and typical output data are shown in Figures 2-17. These are typical designs for the U-Pu-Zr and U-Zr alloy 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 FTR cores using alloy 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 the 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.

TABLE XVIII. Changes Required to Modify UO_2 - PuO_2 Input Data^(a) to Alloy Input Data

Parameter Symbol	Parameter Title	UO_2 - PuO_2 Values	Alloy Values ^(b)	Remarks
(A) <u>Plutonium-Enriched Fuel Cases:</u>				
C'_1	Unit Conversion Cost	\$500/kg ^{239}Pu Processed	\$550/kg ^{239}Pu Processed	Conversion Cost Raised 10%
C_4	Unit Metal Reprocessing Cost	\$24,000/day	\$25,200/day	Reprocessing Cost Raised 5%
C_P^1	Estimated Pin Fabrication Cost Before Correction Factors are Applied	\$85.95/pin	(a) \$35/pin and \$50/pin for Stainless Steel Clad Pins (b) \$50/pin and \$90/pin for Vanadium Alloy Clad Pins	There was an Interest in Using a High and Low Value for Pins with each Cladding Material
C_{conv}	Cost of Converting Nitrate to Oxide (to Metal)	\$237,000/yr	\$260,700/yr	To Convert the Oxide to Metal, an Additional 10% was Required
F_T	Overpower Factor	1.25	1.0	The Overpower Factor is not Applicable in the Alloy Cases
L_{1s}	Scrap from Conversion Plant, (kg of ^{239}Pu lost/kg of ^{239}Pu processed)	0.01	0.02	
L_{2s}	Scrap from Fabrication Plant, (kg of ^{239}Pu lost/kg of ^{239}Pu processed)	0.054	0.02	
(B) <u>Uranium-Enriched Fuel Cases:</u>				
C'_1	Unit Conversion	\$500/kg ^{239}Pu	(a) for Cases 1-3 use \$550/kg ^{235}U processed (b) for Case 4 use \$500/kg ^{235}U processed	Conversion Cost Raised 10% for All Cases Except Case 4
C_4	Unit Metal Reprocessing Cost	\$24,000/day	\$25,200/day	Reprocessing Cost Raised 5%

TABLE XVIII. (Contd)

Parameter Symbol	Parameter Title	UO ₂ -PuO ₂ Values	Alloy Values ^(b)	Remarks
C ¹ _P	Estimated Pin Fabrication Cost before Correction Factors are Applied	\$85.95/pin	(a) \$28/pin and \$40/pin for Stainless Steel Clad Pins, and (b) \$40/pin and \$72/pin for Vanadium Alloy Clad Pins	There was an Interest in Using a High and a Low Value for Pins with Each Cladding Material. The Costs for Nonplutonium-Containing Pins were Believed to be ~20% Lower than the Costs for Plutonium-Containing Pins.
C _{Pu}	Value of ²³⁹ Pu	\$10,000/kg	\$12,000/kg	²³⁵ U Costs 20% More.
C _{conv}	Cost of Converting Nitrate to Oxide (to Metal)	\$237,000/yr	\$260,700/yr	To Convert Oxide to Metal, an Additional 10% was Required.
F _T	Overpower Factor	1.25	1.0	The Overpower Factor is not Applicable in the Alloy Cases.
L _{1s}	Scrap from Conversion Plant, (kg of ²³⁹ Pu lost/kg of ²³⁹ Pu Processed)	0.01	0.02	²³⁵ U used in place of ²³⁹ Pu.
L _{2s}	Scrap from Fabrication Plant (kg of ²³⁹ Pu lost/kg of ²³⁹ Pu Processed)	0.054	0.02	²³⁵ U used in place of ²³⁹ Pu.

(a) The UO₂-PuO₂ fuel pins employed stainless steel cladding and a helium bond.

(b) Note: All other UO₂-PuO₂ economic parameter values unchanged.

TABLE XIX. Values for U-Pu-Zr and U-Zr Alloy Cores Nomenclature and Reference

		U-Pu-Zr	U-Zr
A ₄	Reprocessing rate, kg/day	20	35
BOND	(a)	Na	Na
BU, Avg.	Average allowable burnup, Mwd/tonne (Pu + U) (a)	45,000	45,000
C _{BL}	Cost of burnup, plus total ²³⁹ Pu or ²³⁵ U	(a)	(a)
C _{CONV}	Cost of converting nitrate to metal, \$/yr (b)	(a)	(a)
C _{FAB}	Pin manufacturing cost, \$/yr	(a)	(a)
C _{FC}	Total fuel cycle cost, \$/yr	(a)	(a)
C _h	Subassembly hardware cost, \$/subassembly	1620 ^(c)	1620 ^(c)
C _p	Average heat capacity of Na, Btu/lb-°F	0.311	0.311
C _P ¹	Pin fabrication cost, \$/pin (base value, not given design)	(a)	(a)
C _{Pu}	Value of ²³⁹ Pu or ²³⁵ U, \$/kg	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}	Prereactor 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 ¹	Unit conversion cost, \$/kg ²³⁵ U or ²³⁹ Pu processed	(a)	(a)
C ₄	Unit metal reprocessing cost, \$/day	(a)	(a)
D	Core diameter, feet (a)	3.14	3.14
D _E	Fabrication efficiency factor	1.0	1.0
d _i	Pin fuel diameter, in. (nonirradiated) (a)	0.174	0.199
d _o	Pin outside diameter, in. (a)	0.245	0.28
d ₁	Lattice dimension, in.	4.6	4.6
F _A	Axial power peaking factor	1.30	1.30

TABLE XIX. (Contd)

		U-Pu-Zr	U-Zr
F_E	Engineering peaking factor	1.05	1.05
F_R	Radial peaking factor	1.13	1.13
F_T	Overpower factor (not applicable)	1.0	1.0
f_{Pu} or f_U	Volume fraction of Pu and/or ^{235}U in fuel, percent ^(a)	19.2	25.0
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 throughput	(e)	(e)
f_5	Subassembly hardware correction factor for length	(e)	(e)
H	Core height, ft ^(a)	2.83	2.83
K_C	Cladding average thermal conductivity, ^(a) Btu/hr-ft-°F	11.9	11.9
$\int k_f dt$	$\int_{800}^{T_C} k_f dt$, Btu/hr-ft	13,000	18,000
L_1	Loss from conversion plant, kg fissile lost/kg fissile processed	0.01	0.01
L_{1S}	Scrap from conversion plant, kg fissile scrap/kg fissile processed	(a)	(a)
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	(a)	(a)
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 and ^{235}U ^(a)	360	535
N_{DF}	Number of driver subassemblies ^(a)	44	44
N_P	Number of fuel pins/subassembly ^(a)	166	124
N_{PR}	Number of power producing subassemblies ^(g)	46	46

TABLE XIX. (Contd)

		U-Pu-Zr	U-Zr
N_R	Total number of pins/subassembly ^(a)	169	127
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
P_T	Core power, MW ^(a)	350	350
ΔP	Core pressure drop, psi	60	60
$\bar{\rho}_s$	Smear density, vol% ^(a)	75	75
T_E	Maximum allowable fuel temperature, °F ^(a,f)	2012	2156
ΔT	Coolant temperature rise, °F	300	300
T_1	Inlet sodium temperature, °F	500	500
T_2	Outlet sodium temperature	800	800
TP	Fuel throughput, kg fissile/yr	530	730
t_T	Reprocessing plant turn around time, days/batch	8	8
V	Core volume, liters	600	600
PPY	Pins per year throughput, pins/yr ^(a)	10,400	8,500
ϕ_o	Centerpoint flux, $\frac{10^{16}n}{cm^2\text{-sec}}$ (a)	1.01×10^{16}	0.78×10^{16}
$\frac{D_i}{t}$	Clad ID-to-clad thickness ratio	14.5/1	14.5/1

(a) Other values used for individual Cases 1a-4b (See Appendix A).

(b) If pyro-reprocessing employed, show costs plus cost for "hot" core handling. (Note - the group did not consider this process since commercial vendors probably would not be set up to use it).

(c) McHugh felt that this value is ~25% lower than it should be.

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

(e) Mixed oxide equation form and coefficients used.

(f) Note - T_E and T_C used in $\int k_f dt$ equations, T_C = fuel surface temperature in °F.

(g) $N_{DF} + 0.2 N_T$

FIGURE 2. Graph of Centerpoint Flux Versus Number of Fueled Tests for a 600-Liter, 61-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 3. Graph of Centerpoint Flux Versus Number of Fueled Tests for an 800-Liter, 91-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 4. Graph of Centerpoint Flux Versus Plutonium Content for a 600-Liter, 61-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 5. Graph of Centerpoint Flux Versus Plutonium Content for an 800-Liter, 91-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 6. Graph of Centerpoint Flux Versus Number of Fueled Tests for a 600-Liter, 61-Tube Reactor with a U-Zr Alloy Fuel Core.

FIGURE 7. Graph of Centerpoint Flux Versus Number of Fueled Tests for an 800-Liter, 91-Tube Reactor with a U-Zr Alloy Fuel Core.

FIGURE 8. Graph of Centerpoint Flux Versus Uranium Content for a 600-Liter, 61-Tube Reactor with a U-Zr Fuel Core.

FIGURE 9. Graph of Fuel Throughput (kg/yr) Versus Uranium Content in Fuel for a 600-Liter, 61-Tube Reactor with a U-Zr Alloy Fuel Core.

FIGURE 10. Graph of Centerpoint Flux Versus Uranium Content in Fuel for an 800-Liter, 91-Tube Reactor with a U-Zr Alloy Fuel Core.

FIGURE 11. Graph of Fuel Throughput (kg/yr) Versus Uranium Content in the Fuel for an 800-Liter, 91-Tube Reactor with a U-Zr Alloy Fuel Core.

FIGURE 12. Graph of Centerpoint Flux Versus Number of Fueled Tests with a 600-Liter, 61-Tube Reactor with U-Pu-Zr Alloy Fuel, Showing the Effect of Core Height-to-Diameter Ratio and Plutonium Content.

FIGURE 13. Graph of Centerpoint Flux Versus Number of Fueled Tests with an 800-Liter, 61-Tube Reactor with U-Pu-Zr Alloy Fuel, Showing the Effect of Core Height-to-Diameter Ratio and Plutonium Content.

FIGURE 14. Graph of Centerpoint Flux Versus Number of Fueled Tests with an 800-Liter, 91-Tube Reactor with U-Pu-Zr Alloy Fuel, Showing the Effect of Core Height-to-Diameter Ratio and Plutonium Content.

FIGURE 15. Graph of Core Coolant Outlet Temperature, Core Coolant Temperature Drop, and Plutonium Fraction in the Fuel Versus Centerpoint Flux for a 600-Liter, 61-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 16. Graph of Core Coolant Outlet Temperature, Core Coolant Temperature Drop, and Plutonium Fraction in the Fuel Versus Centerpoint Flux for an 800-Liter, 91-Tube Reactor with a U-Pu-Zr Alloy Fuel Core.

FIGURE 17. Graph of Plutonium Fraction in the Fuel Versus Centerpoint Flux for an 800-Liter, 91-Tube Reactor with a U-Pu-Zr Alloy Fuel Core, Showing the Effect of the Core Height-to-Diameter Ratio.

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. Also, reprocessing costs were estimated, based on the reference throughput, and penalties for changes in throughput were included.

The four alloy cores (U-Pu-Zr, U-Zr, U-Mo, and "adjusted" uranium) 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 burnup 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 burnup. 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 ALLOY FUEL STUDIES

Analytical studies are being continued in investigation of the use of alloy fuels for FTR under the Full-Performance Fuel Design Analysis Task which is 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

* *Note: Since this was the limit used by PNL on other FFTF fuel studies, the group recommended using it on all the alloy fuel cases.*

(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 provide design improvement by permitting larger fuel pins and, therefore, lower fissile fractions as a result of the accompanying lower pressure drop for the same power. The alloy-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.

ACKNOWLEDGEMENTS

The authors wish to thank R. D. Benham for his significant contribution in developing the MIDAS computer code application for both the design analog and economic evaluation-fuel cycle routine. The work performed by J. C. Draper and W. E. McClung was invaluable in preparing the input data and operating the analog computer "under fire" during the working period.

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APPENDIX A

COMPLETE DESIGN ANALOG AND ECONOMIC DATA FOR THE
FFTF ALLOY STUDIES

TABLE A-I. Case 1a (Fuel Type: U-Pu-Zr Alloy, Sodium-Bonded to Stainless Steel Cladding)

CASE NUMBER 1a	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U-Pu-Zr	Same as 1	Same as 1
Composition, wt%	15 Pu, 15 Zr	Same as 1	Same as 1
Cladding	SS	Same as 1	Same as 1
Fab. Cost, C' _p ; \$/Pin	35	Same as 1	Same as 1
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T _c (max); °F	1470	1470	1470
Fuel Surface Temp, ^(a) T _c (max); °F	913	919	924
Overpower Factor, F _T	1.0	1.0	1.0
Clad ID, in.	0.221	0.200	0.184
Clad OD, d _o ; in.	0.257	0.232	0.213
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	18.2	19.4	20.5
∫kdt (operating), W/cm	57.5	56.9	56.4
Linear Power (max), Operating; kW/ft	22.1	21.9	21.7
Critical Mass, M _C ; kg	390	390	390
Pins/Subassembly, N _R	169	195	217
Fueled Tests, N _T ; no.	10	100	10
Reactor Power, P _T ; MW	310	353	390
Fissile Throughput, kg/yr	330	400	467
Pin Throughput, PPY; Pins/yr	6280	8710	11,360
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	0.885	1.00	1.11
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C _{FC} ; (\$/yr) x 10 ⁶	1.67	1.99	2.29
Recovery Costs, C _{RP} ; (\$/yr) x 10 ⁶	0.62	0.707	0.79
Pin Manufacturing Costs, C _{Fab} ; (\$/yr) x 10 ⁶	0.26	0.32	0.38
Cost of Burnup, C _{BL} ; (\$/yr) x 10 ⁶	0.50	0.62	0.73

TABLE A-I. (contd)

CASE NUMBER 1a	RUN 4	RUN 5	RUN 6
Fuel Pin Data: Fuel Alloy	U-Pu-Zr	Same as 4	Same as 4
Composition, wt%	15 Pu, 15 Zr	Same as 4	Same as 4
Cladding	SS	Same as 4	Same as 4
Fab. Cost, C'_p ; \$/Pin	35	Same as 4	Same as 4
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_L (max); °F	1470	1290	1470
Fuel Surface Temp, ^(a) T_c (max); °F	923	866	925
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.184	0.194	0.184
Clad OD, d_o ; in.	0.213	0.225	0.213
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	20.5	18.5	20.5
$\int kdt$ (operating), W/cm	56.4	43.6	56.3
Linear Power (max), Operating; kW/ft	21.7	16.9	21.6
Critical Mass, M_C ; kg	390	390	390
Pins/Subassembly, N_R	217	217	217
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	390	304	390
Fissile Throughput, kg/yr	839	327	468
Pin Throughput, PPY; Pins/yr	20,460	7950	11,370
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.11	0.868	1.11
Burnup, BU avg; MWd/tonne (avg)	25,000	45,000	45,000
Fuel Cycle Cost; C_{FC} : (\$/yr) x 10 ⁶	3.36	1.70	2.41
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	1.05	0.62	0.79
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.54	0.31	0.49
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	0.85	0.50	0.73

TABLE A-I. (contd)

CASE NUMBER	1a	RUN 7	RUN 8	RUN 9
Fuel Pin Data:	Fuel Alloy	U-Pu-Zr	Same	Same
	Composition, wt%	15 Pu, 15 Zr	as 7 Same as 7	as 7 Same as 7
	Cladding	SS	Same as 7	Same as 7
	Fab. Cost, C' _P ; \$/Pin	35	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.		80	80	80
Bond, Material		Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T _{GL} (max); °F		1470	1470	1470
Fuel Surface Temp, ^(a) T _C (max); °F		936	936	935
Overpower Factor, F _T		1.0	1.0	1.0
Clad ID, in.		0.182	0.183	0.182
Clad OD, d _O ; in.		0.211	0.211	0.211
Core Volume, V; liters		600	600	600
Core Height/Core Diam, H/D; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		19.7	22.6	19.6
∫kdt (operating), W/cm		58.9	58.9	58.9
Linear Power (max), Operating; kW/ft		22.6	22.6	22.7
Critical Mass, M _C ; kg		390	390	390
Pins/Subassembly, N _R		217	217	217
Fueled Tests, N _T ; no.		10	17.7	10
Reactor Power, P _T ; MW		408	353	407
Fissile Throughput, kg/yr		467	445	839
Pin Throughput, PPY; Pins/yr		11,390	8,960	20,460
Center Point Flux, ϕ_O ; nv x 10 ¹⁶		1.16	1.00	1.16
Burnup, BU avg; MWd/tonne (avg)		45,000	45,000	25,000
Fuel Cycle Cost, C _{FC} : (\$/yr) x 10 ⁶		1.16	1.0	1.16
Recovery Costs, C _{RP} ; (\$/yr) x 10 ⁶		0.79	0.77	1.26
Pin Manufacturing Costs, C _{Fab} ; (\$/yr) x 10 ⁶		0.37	0.32	0.54
Cost of Burnup, C _{BL} ; (\$/yr) x 10 ⁶		0.73	0.71	0.84

(a) At overpower, not steady state (operating), condition.

TABLE A-II. *Case 1b (Fuel Type: U-Pu-Zr Alloy, Sodium-Bonded to Vanadium Alloy Cladding)*

CASE NUMBER 1b	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U-Pu-Zr	Same as 1	Same as 1
Composition, wt%	15 Pu, 15 Zr	Same as 1	Same as 1
Cladding	V alloy	Same as 1	Same as 1
Fab. Cost, C'_p ; \$/Pin	50	Same as 1	Same as 1
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_L (max); °F	1740	1740	1740
Fuel Surface Temp, ^(a) T_c (max); °F	973	963	989
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.209	0.242	0.170
Clad OD, d_o ; in.	0.241	0.280	0.197
Core Volume, V ; liters	600	600	600
Core Height/Core Diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	21.3	19.1	24.9
$\int kdt$ (operating), W/cm	79.0	80.6	77.8
Linear Power (max), Operating; kW/ft	30.5	30.8	29.8
Critical Mass, M_C ; kg	390	390	390
Pins/Subassembly, N_R	169	139	217
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	423	352	531
Fissile Throughput, kg/yr	516	388	762
Pin Throughput, PPY; Pins/yr	9730	5990	18,480
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.20	1.00	1.50
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	2.49	1.93	3.56
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.86	0.69	1.16
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.44	0.35	0.63
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	0.75	0.55	1.12

TABLE A-II. (contd)

CASE NUMBER 1b	RUN 4	RUN 5	RUN 6
Fuel Pin Data: Fuel Alloy	U-Pu-Zr	Same as 4	Same as 4
Composition, wt%	15 Pu, 15 Zr	Same as 4	Same as 4
Cladding	V alloy	Same as 4	Same as 4
Fab. Cost, C'_p ; \$/Pin	50	Same as 4	90
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F	1740	1470	1740
Fuel Surface Temp, ^(a) T_C (max); °F	973	893	973
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.209	0.220	0.209
Clad OD, d_o ; in.	0.241	0.256	0.241
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	21.3	18.8	21.3
$\int kdt$ (operating), W/cm	79.0	59.5	79.0
Linear Power (max), Operating; kW/ft	30.5	22.8	30.5
Critical Mass, M_C ; kg	390	390	390
Pins/Subassembly, N_R	169	169	169
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	423	317	423
Fissile Throughput, kg/yr	935	342	516
Pin Throughput, PPY; Pins/yr	17,690	6460	9730
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.20	0.90	1.20
Burnup, BU avg; MWd/tonne (avg)	25,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	3.68	1.76	2.76
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	1.38	0.64	0.86
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.63	0.34	0.71
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	0.88	0.48	0.75

TABLE A-II. (contd)

CASE NUMBER 1b	RUN 7	RUN 8	RUN 9
Fuel Pin Data: Fuel Alloy	U-Pu-Zr	Same as 7	Same as 7
Composition, wt%	15 Pu, 15 Zr	Same as 7	Same as 7
Cladding	V alloy	Same as 7	Same as 7
Fab. Cost, C'_p ; \$/Pin	50	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	80	80	80
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_L (max); °F	1740	1740	1740
Fuel Surface Temp, ^(a) T_C (max); °F	983	983	981
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.206	0.206	0.206
Clad OD, d_o ; in.	0.239	0.239	0.239
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	20.1	20.1	25.1
$\int kdt$ (operating), W/cm	84.0	84.0	84.0
Linear Power (max), Operating; kW/ft	32.1	32.1	32.1
Critical Mass, M_C ; kg	390	390	390
Pins/Subassembly, N_R	169	169	169
Fueled Tests, N_T ; no.	10	10	21.6
Reactor Power, P_T ; MW	443	443	355
Fissile Throughput, kg/yr	520	936	858
Pin Throughput, PPY; Pins/yr	9950	17,890	11,990
Center Point Flux, ϕ_o ; nv $\times 10^{16}$	1.27	1.27	1.00
Burnup, BU avg; MWd/tonne (avg)	45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) $\times 10^6$	2.59	3.80	3.46
Recovery Costs, C_{RP} ; (\$/yr) $\times 10^6$	0.85	1.37	1.28
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) $\times 10^6$	0.46	0.66	0.51
Cost of Burnup, C_{BL} ; (\$/yr) $\times 10^6$	0.82	0.97	0.91

(a) At overpower, not steady state (operating), condition.

TABLE A-III. Case 2a (Fuel Type: U-Zr Alloy, Sodium-Bonded to Stainless Steel Cladding)

CASE NUMBER 2a	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U-Zr	Same as 1	Same as 1
Composition, wt%	15 Zr	Same as 1	Same as 1
Cladding	SS	Same as 1	Same as 1
Fab. Cost, C'_p ; \$/Pin	28	Same as 1	Same as 1
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F	1470	1470	1470
Fuel Surface Temp, ^(a) T_C (max); °F	923	934	936
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.184	0.156	0.154
Clad OD, d_o ; in.	0.213	0.182	0.179
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	24.6	27.7	28.2
$\int kdt$ (operating), W/cm	56.6	55.6	55.2
Linear Power (max), Operating; kW/ft	21.6	21.2	21.2
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	217	267	271
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	389	467	473
Fissile Throughput, kg/yr	545	752	775
Pin Throughput, PPY; Pins/yr	10,250	17,100	18,850
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	0.84	1.00	1.02
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) x 10 ⁶	2.74	3.49	3.57
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.64	0.79	0.81
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.30	0.41	0.43
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.33	1.63	1.67

TABLE A-III. (contd)

CASE NUMBER 2a	RUN 4	RUN 5	RUN 6
Fuel Pin Data: Fuel Alloy	U-Zr	Same as 4	Same as 4
Composition, wt%	15 Zr	Same as 4	Same as 4
Cladding	SS	Same as 4	Same as 4
Fab. Cost, C'_P ; \$/Pin	28	Same as 4	40
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F	1470	1290	1470
Fuel Surface Temp, ^(a) T_C (max); °F	935	873	934
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.154	0.165	0.154
Clad OD, d_O ; in.	0.179	0.191	0.179
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	28.0	24.6	28.0
$\int kdt$ (operating), W/cm	55.0	43.0	55.3
Linear Power (max), Operating; kW/ft	21.2	16.5	21.2
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	271	271	271
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	473	369	474
Fissile Throughput, kg/yr	1395	530	772
Pin Throughput, PPY; Pins/yr	32,290	12,250	17,880
Center Point Flux, ϕ_O ; nv x 10 ¹⁶	1.02	0.795	1.04
Burnup, BU avg; MWd/tonne (avg)	25,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	5.01	2.69	3.68
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	1.30	0.62	0.81
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.62	0.34	0.54
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.89	1.27	1.66

TABLE A-III. (contd)

CASE NUMBER 2a	RUN 7	RUN 8	RUN 9
Fuel Pin Data: Fuel Alloy	U-Zr	Same as 7	Same as 7
Composition, wt%	15 Zr	Same as 7	Same as 7
Cladding	SS	Same as 7	Same as 7
Fab. Cost, C'_p ; \$/Pin	28	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	80	80	80
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F	1470	1470	1470
Fuel Surface Temp, ^(a) T_C (max); °F	948	947	948
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.152	0.152	0.153
Clad OD, d_o ; in.	0.177	0.177	0.177
Core Volume, V ; liters	600	600	600
Core Height/Core diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	26.9	26.9	28.4
$\int kdt$ (operating), W/cm	57.7	57.6	57.4
Linear Power (max), Operating; kW/ft	22.1	22.1	22.1
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	271	271	271
Fueled Tests, N_T ; no.	10	10	13.1
Reactor Power, P_T ; MW	494	494	468
Fissile Throughput, kg/yr	774	1392	759
Pin Throughput, PPY; Pins/yr	17,930	32,260	16,250
Center Point Flux, ϕ_o ; nv $\times 10^{16}$	1.06	1.07	1.00
Burnup, BU avg; MWd/tonne (avg)	45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) $\times 10^6$	3.63	5.07	3.47
Recovery Costs, C_{RP} ; (\$/yr) $\times 10^6$	0.81	1.30	0.80
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) $\times 10^6$	0.43	0.61	0.40
Cost of Burnup, C_{BL} ; (\$/yr) $\times 10^6$	1.72	1.94	1.61

(a) At overpower, not steady state (operating), condition.

TABLE A-IV. *Case 2b (Fuel Type: U-Zr Alloy, Sodium-Bonded to Vanadium Alloy Cladding)*

CASE NUMBER 2b		RUN 1	RUN 2	RUN 3
Fuel Pin Data:	Fuel Alloy	U-Zr	Same	Same
	Composition, wt%	15 Zr	as 1	as 1
	Cladding	V alloy	as 1	as 1
	Fab. Cost, C'_p ; \$/Pin	40	Same	Same
			as 1	as 1
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		75	75	75
Bond, Material		Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F		1740	1740	1740
Fuel Surface Temp, ^(a) T_C (max); °F		972	990	989
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.207	0.169	0.170
Clad OD, d_o ; in.		0.240	0.197	0.196
Core Volume, V; liters		600	600	600
Core Height/Core Diam, H/D; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		25.3	29.6	29.7
$\int kdt$ (operating), W/cm		79.1	77.3	77.0
Linear Power (max), Operating; kW/ft		30.5	29.8	29.7
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		169	217	217
Fueled Tests, N_T ; no.		10	10	10
Reactor Power, P_T ; MW		426	534	534
Fissile Throughput, kg/yr		630	923	1663
Pin Throughput, PPY; Pins/yr		9140	17,130	30,800
Center Point Flux, ϕ_o ; nv x 10 ¹⁶		0.92	1.15	1.15
Burnup, BU avg; MWd/tonne (avg)		45,000	45,000	25,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		3.08	4.14	5.86
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		0.70	0.93	1.51
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.36	0.52	0.76
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		1.47	1.89	2.15

TABLE A-IV. (contd)

CASE NUMBER	2b	RUN 4	RUN 5	RUN 6
Fuel Pin Data:	Fuel Alloy	U-Zr	Same	Same
	Composition, wt%	15 Zr	as 4	as 4
	Cladding	V alloy	Same	Same
	Fab. Cost, C'_p ; \$/Pin	72	as 4	as 4
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		40	75	80
Bond, Material		Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F		1740	1470	1740
Fuel Surface Temp, ^(a) T_C (max); °F		985	901	1000
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.169	0.182	0.167
Clad OD, d_o ; in.		0.196	0.212	0.194
Core Volume, V ; liters		600	600	600
Core Height/Core Diam, H/D ; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		29.7	25.4	28.6
$\int kdt$ (operating), W/cm		77.8	58.6	81.3
Linear Power (max), Operating; kW/ft		29.9	22.5	31.3
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		217	217	217
Fueled Tests, N_T ; no.		10	10	10
Reactor Power, P_T ; MW		537	403	560
Fissile Throughput, kg/yr		932	600	937
Pin Throughput, PPY; Pins/yr		17,270	11,100	17,330
Center Point Flux, ϕ_o ; nv x 10 ¹⁶		1.16	0.87	1.21
Burnup, BU avg; MWd/tonne (avg)		45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		4.47	2.99	4.25
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		0.94	0.67	0.94
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.83	0.40	0.52
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		1.90	1.39	1.97

TABLE A-IV. (contd)

CASE NUMBER	2b	RUN 7	RUN 8	RUN
Fuel Pin Data:	Fuel Alloy	U-Zr	Same as 7	
	Composition, wt%	15 Zr	Same as 7	
	Cladding	V alloy	Same as 7	
	Fab. Cost, C'_P ; \$/Pin	40	Same as 7	
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.		80	80	
Bond, Material		Na	Na	
Gap Coeff., h; Btu/hr-ft ²		41,000	41,000	
Fuel Centerline Temp, ^(a) T_G (max); °F		1740	1740	
Fuel Surface Temp, ^(a) T_C (max); °F		999	1000	
Overpower Factor, F_T		1.0	1.0	
Clad ID, in.		0.167	0.167	
Clad OD, d_O ; in.		0.194	0.194	
Core Volume, V; liters		600	600	
Core Height/Core Diam, H/D; ratio		0.9	0.9	
Fuel Volume Fraction (fissile), vol%		28.6	34.4	
$\int kdt$ (operating), W/cm		81.3	81.3	
Linear Power (max), Operating; kW/ft		31.2	31.3	
Critical Mass, M_C ; kg		514	514	
Pins/Subassembly, N_R		217	217	
Fueled Tests, N_T ; no.		10	19.6	
Reactor Power, P_T ; MW		561	468	
Fissile Throughput, kg/yr		1681	881	
Pin Throughput, PPY; Pins/yr		31,180	12,760	
Center Point Flux, ϕ_O ; nv x 10 ¹⁶		1.21	1.00	
Burnup, BU avg; MWd/tonne (avg)		25,000	45,000	
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		5.98	3.68	
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		1.52	0.89	
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.77	0.43	
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		2.24	1.59	

(a) At overpower, not steady state (operating), condition.

TABLE A-V. Case 3a (Fuel Type: U-Mo Alloy, Sodium-Bonded to Stainless Steel Cladding)

CASE NUMBER 3a	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U-Mo	Same as 1	Same as 1
Composition, wt%	10 Mo	Same as 1	Same as 1
Cladding	SS	Same as 1	Same as 1
Fab. Cost, C'_p ; \$/Pin	28	Same as 1	Same as 1
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F	1380	1380	1380
Fuel Surface Temp, ^(a) T_C (max); °F	923	933	935
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.185	0.157	0.155
Clad OD, d_o ; in.	0.214	0.186	0.180
Core Volume, V ; liters	600	600	600
Core Height/Core Diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	24.8	28.1	28.2
$\int kdt$ (operating), W/cm	56.4	54.9	54.9
Linear Power (max), Operating; kW/ft	21.6	21.2	21.0
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	217	268	271
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	386	465	469
Fissile Throughput, kg/yr	562	765	775
Pin Throughput, PPY; Pins/yr	10,350	17,500	17,880
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	0.833	1.00	1.01
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	2.78	3.52	3.59
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.65	0.80	0.82
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.30	0.42	0.53
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.34	1.63	1.67

TABLE A-V. (contd)

CASE NUMBER	3a	RUN 4	RUN 5	RUN 6
Fuel Pin Data:	Fuel Alloy	U-Mo	Same as 4	Same as 4
	Composition, wt%	10 Mo	Same as 4	Same as 4
	Cladding	SS	Same as 4	Same as 4
	Fab. Cost, C'_p ; \$/Pin	28	40	28
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		75	75	75
Bond, Material		Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F		1380	1380	1245
Fuel Surface Temp, ^(a) T_C (max); °F		936	934	883
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.155	0.155	0.165
Clad OD, d_o ; in.		0.180	0.180	0.191
Core Volume, V; liters		600	600	600
Core Height/Core Diam, H/D; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		28.3	28.4	25.2
$\int kdt$ (operating), W/cm		54.9	54.7	44.6
Linear Power (max), Operating; kW/ft		21.0	21.0	17.1
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		271	271	271
Fueled Tests, N_T ; no.		10	10	10
Reactor Power, P_T ; MW		469	468	383
Fissile Throughput, kg/yr		1397	778	565
Pin Throughput, PPY; Pins/yr		32,200	18,050	13,070
Center Point Flux, ϕ_o ; nv x 10 ¹⁶		1.01	1.01	0.83
Burnup, BU avg; MWd/tonne (avg)		25,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		5.04	3.71	2.83
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		1.31	0.82	0.65
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.63	0.54	0.35
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		1.89	1.66	1.33

TABLE A-V. (contd)

CASE NUMBER 3a	RUN 7	RUN 8	RUN 9
Fuel Pin Data: Fuel Alloy	U-Mo	Same as 7	Same as 7
Composition, wt%	10 Mo	Same as 7	Same as 7
Cladding	SS	Same as 7	Same as 7
Fab. Cost, C'_P ; \$/Pin	28	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.	80	80	80
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F	1380	1380	1380
Fuel Surface Temp, ^(a) T_C (max); °F	947	947	946
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.153	0.154	0.154
Clad OD, d_o ; in.	0.178	0.178	0.178
Core Volume, V ; liters	600	600	600
Core height/Core Diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	27.1	27.1	28.2
$\int kdt$ (operating), W/cm	56.9	56.9	57.0
Linear Power (max), Operating; kW/ft	21.8	21.8	21.8
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	271	271	271
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	489	488	466
Fissile Throughput, kg/yr	770	1387	754
Pin Throughput, PPY; Pins/yr	17,690	32,100	16,410
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.05	1.05	1.00
Burnup, BU avg; MWd/tonne (avg)	45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	3.65	5.10	3.51
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.81	1.31	0.81
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.43	0.62	0.51
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.72	1.94	1.62

(a) At overpower, not steady state (operating), condition.

TABLE A-VI. *Case 3b (Fuel Type: U-Mo Alloy, Sodium-Bonded to Vanadium Alloy Cladding)*

CASE NUMBER	3b	RUN 1	RUN 2	RUN 3
Fuel Pin Data:	Fuel Alloy	U-Mo	Same as 1	Same as 1
	Composition, wt%	10-Mo	Same as 1	Same as 1
	Cladding	V alloy	Same as 1	Same as 1
	Fab. Cost, C'_p ; \$/Pin	40	40	40
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		75	75	75
Bond, Material		Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_G (max); °F		1380	1380	1380
Fuel Surface Temp, ^(a) T_C (max); °F		904	913	917
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.182	0.161	0.152
Clad OD, d_o ; in.		0.211	0.187	0.177
Core Volume, V ; liters		600	600	600
Core Height/Core Diam, H/D ; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		25.6	27.9	29.2
$\int kdt$ (operating), W/cm		58.9	57.7	57.3
Linear Power (max), Operating; kW/ft		22.6	22.2	22.0
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		217	255	271
Fueled Tests, N_T ; no.		10	10	10
Reactor Power, P_T ; MW		405	466	492
Fissile Throughput, kg/yr		605	761	840
Pin Throughput, PPY; Pins/yr		11,210	16,540	19,340
Center Point Flux, ϕ_o ; nv $\times 10^{16}$		0.87	1.00	1.06
Burnup, BU avg; MWd/tonne (avg)		45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) $\times 10^6$		3.01	3.61	3.90
Recovery Costs, C_{RP} ; (\$/yr) $\times 10^6$		0.68	0.80	0.86
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) $\times 10^6$		0.40	0.51	0.56
Cost of Burnup, C_{BL} ; (\$/yr) $\times 10^6$		1.40	1.64	1.74

TABLE A-VI. (contd)

CASE NUMBER	3b	RUN 4	RUN 5	RUN 6
Fuel Pin Data:	Fuel Alloy	U-Mo	Same	Same
	Composition	10 Mo	as 4	as 4
	Cladding	V alloy	Same	Same
	Fab. Cost, C'_p ; \$/Pin	40	as 4	as 4
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		75	72	40
Bond, Material		Na	75	75
Gap Coeff., h ; Btu/hr-ft ²		Na	Na	Na
Fuel Centerline Temp, ^(a) T_{CL} (max); °F		41,000	41,000	41,000
Fuel Surface Temp, ^(a) T_c (max); °F		1380	1380	1245
Overpower Factor, F_T		917	915	868
Clad ID, in.		1.0	1.0	1.0
Clad OD, d_o ; in.		0.152	0.153	0.162
Core Volume, V ; liters		0.177	0.177	0.187
Core Height/Core Diam, H/D ; ratio		600	600	600
Fuel Volume Fraction (fissile), vol%		0.9	0.9	0.9
$\int kdt$ (operating), W/cm		29.1	29.2	26.0
Linear Power (max), Operating; kW/ft		57.0	57.3	46.5
Critical Mass, M_C ; kg		22.1	21.9	17.9
Pins/Subassembly, N_R		514	514	514
Fueled Tests, N_T ; no.		271	271	271
Reactor Power, P_T ; MW		10	10	10
Fissile Throughput, kg/yr		492	492	400
Pin Throughput, PPY; Pins/yr		1510	837	606
Center Point Flux, ϕ_o ; nv x 10 ¹⁶		34,880	19,350	14,070
Burnup, BU avg; MWd/tonne (avg)		1.06	1.06	0.86
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		25,000	45,000	45,000
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		5.50	4.22	3.06
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		1.39	0.86	0.68
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		0.82	0.89	0.46
		1.98	1.75	1.39

TABLE A-VI. (contd)

CASE NUMBER	3b	RUN 7	RUN 8	RUN 9
Fuel Pin Data:	Fuel Alloy	U-Mo	Same as 7	Same as 7
	Composition, wt%	10 Mo	Same as 7	Same as 7
	Cladding	V alloy	Same as 7	Same as 7
	Fab. Cost, C'_P ; \$/Pin	40	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.		80	80	80
Bond, Material		Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F		1380	1380	1380
Fuel Surface Temp, ^(a) T_C (max); °F		929	929	930
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.151	0.151	0.151
Clad OD, d_o ; in.		0.175	0.175	0.175
Core Volume, V ; liters		600	600	600
Core Height/Core Diam, H/D ; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		28.0	28.1	31.0
$\int kdt$ (operating), W/cm		59.3	59.3	59.3
Linear Power (max), Operating; kW/ft		22.8	23.0	22.3
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		271	271	271
Fueled Tests, N_T ; no.		10	10	15
Reactor Power, P_T ; MW		510	512	466
Fissile Throughput, kg/yr		835	1501	813
Pin Throughput, PPY; Pins/yr		19,340	34,770	16,440
Center Point Flux, ϕ_o ; nv x 10 ¹⁶		1.10	1.10	1.00
Burnup, BU avg; MWd/tonne (avg)		45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) x 10 ⁶		3.95	5.54	3.67
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		0.85	1.40	0.84
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.57	0.82	0.51
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		1.79	2.03	1.61

(a) At overpower, not steady state (operating), condition.

TABLE A-VII. Case 4a (Fuel Type: "Adjusted" Uranium, Sodium-Bonded to Stainless Steel Cladding)

CASE NUMBER 4a	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U	Same as 1	Same as 1
Composition, wt%	"adjusted"	Same as 1	Same as 1
Cladding	SS	Same as 1	Same as 1
Fab. Cost, C'_p ; \$/Pin	28	28	28
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_C (max); °F	1470	1470	1470
Fuel Surface Temp, ^(a) T_C (max); °F	995	992	1011
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.173	0.183	0.143
Clad OD, d_o ; in.	0.201	0.212	0.166
Core Volume, V ; liters	600	600	600
Core Height/Core Diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	28.3	27.1	33.2
$\int kdt$ (operating), W/cm	71.5	72.0	68.9
Linear Power (max), Operating, kW/ft	27.5	27.6	26.5
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	217	204	271
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	494	466	594
Fissile Throughput, kg/yr	815	733	1149
Pin Throughput, PPY; Pins/yr	15,140	12,770	26,660
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.06	1.00	1.28
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) x 10 ⁶	3.64	3.35	4.77
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.84	0.78	1.11
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.37	0.34	0.55
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.74	1.63	2.15

TABLE A-VII. (contd)

CASE NUMBER	4a	RUN 4	RUN 5	RUN 6
Fuel Pin Data: Fuel Alloy		U	Same as 4	Same as 4
Composition, wt%		"adjust- ed" U	Same as 4	Same as 4
Cladding		SS	Same as 4	Same as 4
Fab. Cost, C'_p ; \$/Pin		28	40	28
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.		75	75	75
Bond, Material		Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F		1470	1470	1340
Fuel Surface Temp, ^(a) T_c (max); °F		1013	1012	955
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.143	0.143	0.152
Clad OD, d_o ; in.		0.166	0.166	0.176
Core Volume, V ; liters		600	600	600
Core Height/Core Diam, H/D ; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		33.2	33.2	29.4
$\int kdt$ (operating), W/cm		68.9	68.7	58.2
Linear Power (max), Operating; kW/ft		26.5	26.5	22.4
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		271	271	271
Fueled Tests, N_T ; no.		10	10	10
Reactor Power, P_T ; MW		595	594	500
Fissile Throughput, kg/yr		2074	1153	859
Pin Throughput, PPY; Pins/yr		47,990	26,730	19,880
Center Point Flux, ϕ_o ; nv $\times 10^{16}$		1.28	1.28	1.08
Burnup, BU avg; MWd/tonne (avg)		25,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) $\times 10^6$		6.82	4.91	3.81
Recovery Costs, C_{RP} ; (\$/yr) $\times 10^6$		1.83	1.11	0.88
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) $\times 10^6$		0.80	0.69	0.45
Cost of Burnup, C_{BL} ; (\$/yr) $\times 10^6$		2.47	2.15	1.77

TABLE A-VII. (contd)

CASE NUMBER 4a	RUN 7	RUN 8	RUN 9
Fuel Pin Data: Fuel Alloy	U	Same as 7	Same as 7
Composition, wt%	"adjusted" U	Same as 7	Same as 7
Cladding	SS	Same as 7	Same as 7
Fab. Cost, C'_p ; \$/Pin	28	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	80	80	80
Bond, Material	Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F	1470	1470	1470
Fuel Surface Temp, ^(a) T_C (max); °F	1010	1011	1012
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.172	0.172	0.171
Clad OD, d_o ; in.	0.199	0.199	0.199
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	27.1	27.0	29.6
$\int kdt$ (operating), W/cm	73.8	74.0	73.9
Linear Power (max), Operating; kW/ft	28.4	28.3	28.4
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	217	217	217
Fueled Tests, N_T ; no.	10	10	15
Reactor Power, P_T ; MW	510	510	467
Fissile Throughput, kg/yr	808	1449	784
Pin Throughput, PPY; Pins/yr	14,980	26,910	12,900
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.10	1.10	1.00
Burnup, BU avg; MWd/tonne (avg)	45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶	3.65	5.10	3.41
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.84	1.34	0.82
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.38	0.54	0.34
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.78	2.01	1.61

(a) At overpower, not steady state (operating), condition.

TABLE A-VIII. Case 4b (Fuel Type: "Adjusted" Uranium, Sodium-Bonded to Vanadium Alloy Cladding)

CASE NUMBER 4b	RUN 1	RUN 2	RUN 3
Fuel Pin Data: Fuel Alloy	U	Same as 1	Same as 1
Composition, wt%	"adjusted" U	Same as 1	Same as 1
Cladding	V alloy	Same as 1	Same as 1
Fab. Cost, C'_p ; \$/Pin	40	40	40
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h ; Btu/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{GL} (max); °F	1740	1740	1740
Fuel Surface Temp, ^(a) T_c (max); °F	1066	1054	1089
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.194	0.225	0.156
Clad OD, d_o ; in.	0.226	0.261	0.181
Core Volume, V ; liters	600	600	600
Core Height/Core Diam, H/D ; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	28.7	25.7	34.7
$\int kdt$ (operating), W/cm	101.5	103.5	98.3
Linear Power (max), Operating; kW/ft	39.0	39.7	37.7
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	169	141	217
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	549	467	680
Fissile Throughput, kg/yr	920	699	1373
Pin Throughput, PPY; Pins/yr	13,280	8,390	25,400
Center Point Flux, ϕ_o ; nv x 10 ¹⁶	1.18	1.00	1.46
Burnup, BU avg; MWd/tonne (avg)	45,000	45,000	45,000
Fuel Cycle Cost, C_{FC} ; (\$/yr) x 10 ⁶	4.08	3.31	5.58
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶	0.93	0.75	1.29
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶	0.45	0.36	0.67
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶	1.94	1.63	2.49

TABLE A-VIII. (contd)

CASE NUMBER 4b	RUN 4	RUN 5	RUN 6
Fuel Pin Data: Fuel Alloy	U	Same as 4	Same as 4
Composition, wt%	"adjust- ed" U	Same as 4	Same as 4
Cladding	V alloy	Same as 4	Same as 4
Fab. Cost, C'_p ; \$/Pin	40	72	40
Fuel Bulk (Smear) Density, $\bar{\rho}_s$; % (by volume) of theo.	75	75	75
Bond, Material	Na	Na	Na
Gap Coeff., h; But/hr-ft ²	41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F	1740	1740	1470
Fuel Surface Temp, ^(a) T_C (max); °F	1090	1089	978
Overpower Factor, F_T	1.0	1.0	1.0
Clad ID, in.	0.156	0.156	0.171
Clad OD, d_o ; in.	0.181	0.181	0.199
Core Volume, V; liters	600	600	600
Core Height/Core Diam, H/D; ratio	0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%	34.6	34.6	28.9
$\int kdt$ (operating), W/cm	98.3	98.6	74.3
Linear Power (max), Operating; kW/ft	37.8	37.8	28.6
Critical Mass, M_C ; kg	514	514	514
Pins/Subassembly, N_R	217	217	217
Fueled Tests, N_T ; no.	10	10	10
Reactor Power, P_T ; MW	680	679	515
Fissile Throughput, kg/yr	2468	1375	868
Pin Throughput, PPY; Pins/yr	45,650	25,500	16,120
Center Point Flux, ϕ_o ; nv $\times 10^{16}$	1.46	1.46	1.11
Burnup, BU avg; MWd/tonne (avg)	25,000	45,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) $\times 10^6$	8.02	5.97	3.93
Recovery Costs, C_{RP} ; (\$/yr) $\times 10^6$	2.13	1.28	0.89
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) $\times 10^6$	0.92	1.06	0.50
Cost of Burnup, C_{BL} ; (\$/yr) $\times 10^6$	2.86	2.49	1.82

TABLE A-VIII. (contd)

CASE NUMBER	4b	RUN 7	RUN 8	RUN 9
Fuel Pin Data:	Fuel Alloy	U	Same as 7	Same as 7
	Composition, wt%	"adjusted" U	Same as 7	Same as 7
	Cladding	V alloy	Same as 7	Same as 7
	Fab. Cost, C'_P ; \$/Pin	40	Same as 7	Same as 7
Fuel Bulk (Smear) Density, $\bar{\rho}_S$; % (by volume) of theo.		80	80	80
Bond, Material		Na	Na	Na
Gap Coeff., h; Btu/hr-ft ²		41,000	41,000	41,000
Fuel Centerline Temp, ^(a) T_{CL} (max); °F		1740	1740	1740
Fuel Surface Temp, ^(a) T_C (max); °F		1082	1083	1083
Overpower Factor, F_T		1.0	1.0	1.0
Clad ID, in.		0.192	0.192	0.192
Clad OD, d_O ; in.		0.223	0.223	0.222
Core Volume, V; liters		600	600	600
Core Height/Core Diam, H/D; ratio		0.9	0.9	0.9
Fuel Volume Fraction (fissile), vol%		27.5	27.5	33.9
$\int kdt$ (operating), W/cm		106.0	106.0	105.8
Linear Power (max), Operating; kW/ft		40.6	40.6	40.6
Critical Mass, M_C ; kg		514	514	514
Pins/Subassembly, N_R		169	169	169
Fueled Tests, N_T ; no.		10	10	20.7
Reactor Power, P_T ; MW		571	572	466
Fissile Throughput, kg/yr		918	1648	859
Pin Throughput, PPY; Pins/yr		13,250	23,770	9,420
Center Point Flux, ϕ_O ; nv x 10 ¹⁶		1.23	1.23	1.00
Burnup, BU avg; MWd/tonne (avg)		45,000	25,000	45,000
Fuel Cycle Cost, C_{FC} : (\$/yr) x 10 ⁶		4.14	5.77	3.54
Recovery Costs, C_{RP} ; (\$/yr) x 10 ⁶		0.94	1.50	0.88
Pin Manufacturing Costs, C_{Fab} ; (\$/yr) x 10 ⁶		0.45	0.65	0.36
Cost of Burnup, C_{BL} ; (\$/yr) x 10 ⁶		2.00	2.26	1.59

(a) At overpower, not steady state (operating), condition.

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,^{(9)*} 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)*

TASKS	ENGINEERS & SCIENTISTS	
	%	MAN-YEARS
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	3.0 ^(a)
j. Advanced Development	7.0	10.5 ^(b)
TOTAL	100.0	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 dollar level and manpower requirements have not changed significantly.

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