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PWAC-431
STATUS OF THE SNAP-50 REACTOR FUELS PARAMETRIC STUDIES-PART I

AEC RESEARCH AND DEVELOPMENT REPORT

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by H.F.C. TIC, date SEP 6 - 1973

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ADVANCED CONCEPTS FOR FUTURE
APPLICATION-REACTOR EXPERIMENTS

Reviewers	Class.	Date
KAW	U	2/9/82

SPECIAL RE REVIEW
FINAL
DETERMINATION
Class: U

PWAC-431
STATUS OF THE SNAP-50 REACTOR FUELS PARAMETRIC STUDIES-PART I

AEC RESEARCH AND DEVELOPMENT REPORT

CONTRACT
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Issued
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I. INTRODUCTION

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E. R. Dytko

I. INTRODUCTION

A. Relation of Reactor to Over-all Space Powerplant Goals

The present SNAP-50/SPUR powerplant development program concentrates on a system which produces a net electrical output of 300 Kwe, using a reactor thermal power of approximately 2 Mw. The development goals for this powerplant are an unshielded powerplant specific weight of 20 lb/ Kwe or less and unattended operation in space for 10,000 hours.

The SNAP-50/SPUR powerplant is expected to be used primarily for exploration of our solar system as the power supply for spacecraft using electric propulsion. Additional applications, such as providing electrical power for large earth orbiting systems and for lunar bases, are also projected.

It appears that the basic reactor concept should be capable of increased power output to about 8 Mw, which is equivalent to a net electrical output of 1 Mwe. Since the minimum reactor size is limited by criticality, and reactor size increases moderately with increased power, a reduction of powerplant specific weight is anticipated at higher power levels due to reactor considerations alone. Furthermore, in addition to those reductions in powerplant specific weight obtainable through improved component design, additional specific weight reductions can be obtained through increased operating temperatures.

B. Scope of Reactor Parametric Studies

The primary purpose of the current series of reactor parametric studies presented in this report is to investigate the effects of reactor fuels, fuel performance level and cladding materials on SNAP-50/SPUR reactor designs. Other variables affecting reactor design are not considered, except as needed to implement the study of effect of fuels. The scope of the investigation is limited to comparison of reactor characteristics and does not include over-all powerplant optimization. In addition to providing guidance to reactor design problems and the reactor fuel and cladding development program, the results of this study are intended to provide data for over-all powerplant optimization studies which will consider reactor effects on nuclear shielding and powerplant envelope and weight. The present base point for reactor performance requirements and all optimization studies is the current SNAP-50/SPUR powerplant concept (Ref. 1).

The reactor parametric studies are divided into two periods associated with what was at the time considered to be realistic fuel performance assumptions. Prior to October, 1963, only uranium carbide (UC) fuel (uranium nitride (UN) is interchangeable from a reactor design viewpoint) with Cb-1 Zr alloy cladding was extensively considered. Although UC/UN has the potential of providing the best reactor power to weight relationship, because uranium density is high and has good thermal conductivity, evaluation of UC/UN irradiation test data showed these fuels to be burnup-limited. Consequently, since October 1963, other fuels and cladding alloys have been considered more extensively in an effort to establish the best balance between fuel development problems and reactor performance for the lifetime and power requirements of space powerplant systems.

Since a number of fuels and cladding materials, the characteristics and limitations of which have not been explored experimentally, are potential candidates for high temperature liquid metal space reactors, it has been necessary to consider a range of operating conditions in order to guide immediate and long range design and development work. The effect on the reactor of each fuel and cladding combination of interest is examined considering performance levels which 1) it is expected can be achieved in the near future (1 to 3 years) and which are within present capability to fabricate and test, and 2) might be reached in the future (beyond 3 years) after extensive development.

The reactor parametric design study is based on both the physics and engineering principles involved in reactor design and explores the variations in reactor characteristics as influenced by operating conditions and material characteristics. The ability of such studies to accurately predict the final reactor weight and size is only as good as the assumptions and models used as input. An engineered reactor design, based on a set of specifications derived from parametric studies might differ considerably in weight and dimensions from parametric study predictions because definition of the complete control drive, for example, is not available for the parametric study. Furthermore, continued material or physics studies might show that assumptions used for quantities such as thermal conductivity, burnup limit, and neutron cross-sections require refinement. However, the parametric studies do result in a consistent design comparison of major variables which influence the design of the reactor, thereby providing the information to balance design and development work in the direction of best possibility of success in meeting the over-all reactor performance goals.

II. SUMMARY AND RECOMMENDATIONS

A. Summary

The fuels and cladding reactor parametric study results and conclusions are summarized below:

1. The parametric reactor weights for the fuels studied are summarized for comparison in Fig. 1 for both Present Capability and Advanced Capability at 2 Mw and 8 Mw design power levels. The fuels which result in minimum reactor weights are:

   2 Mw UC/UN (95 percent dense), UC/UN - 1W.

   8 Mw UC/UN-1W

2. Tantalum alloy and tungsten-rhenium fuel clads do not provide a significant weight advantage, as compared to the present capability of PWC-11 alloy (Cb-1 Zr-0.1 C) clad for the leading candidate fuels having low design fission gas releases. This effect is attributable to the minimum cladding thickness requirement of 0.015 inch used in the study, the near minimum PWC-11 clad thickness for the low gas release uses, and the increased density and the negative reactivity effects associated with the tantalum and tungsten clads. Carbide-fueled reactors with tungsten-rhenium clad weigh less than those with tantalum alloy, due to the lack of a diffusion barrier between the fuel and tantalum cladding.

3. The fuel maximum temperature over the limited range of the study (200F to 400F differences) does not have a significant effect on reactor weight for fuel pin diameters >0.250 inch, but does appreciably affect the number of fuel pins in the reactor.

B. Recommendations

The following recommendations for the 2 Mw SNAP-50/SPUR reactor immediate fuel element development and reactor design program are made, based upon the results of the parametric study considering reactor weight, core flow conditions, core fabrication and assembly, i.e., minimum number of fuel elements, fuel fabrication and test capabilities and available fuel performance data (Ref. 2).

1. Fuels

   The following fuels irradiation testing is required for immediate development:

   a. UC/UN (95 percent dense) to verify required 2 Mw performance

   b. UC/UN-1 W and 90 UC/UN-10 m/o ZrC at 2 Mw requirements to verify performance as a backup fuel.

2. Cladding

   a. PWC-11 alloy (Cb-1 Zr-0.1C) be used for the PWAR-20.

   b. Tungsten-rhenium clad to be developed as part of the long range program to provide a higher strength alloy if experimental UC/UN fission gas releases are higher than design values.

3. Reactor Design

   a. 2 Mw PWAR-20 for SNAP-50/SPUR powerplant

      Fuel - UC/UN (95 percent dense), maximum fuel burnup 1.5 percent (total uranium)

      Clad - PWC-11 alloy (Cb-1 Zr-0.1C)

PARAMETRIC REACTOR WEIGHT COMPARISON FOR VARIOUS FUELS AND CLADS

2 MW REACTORS

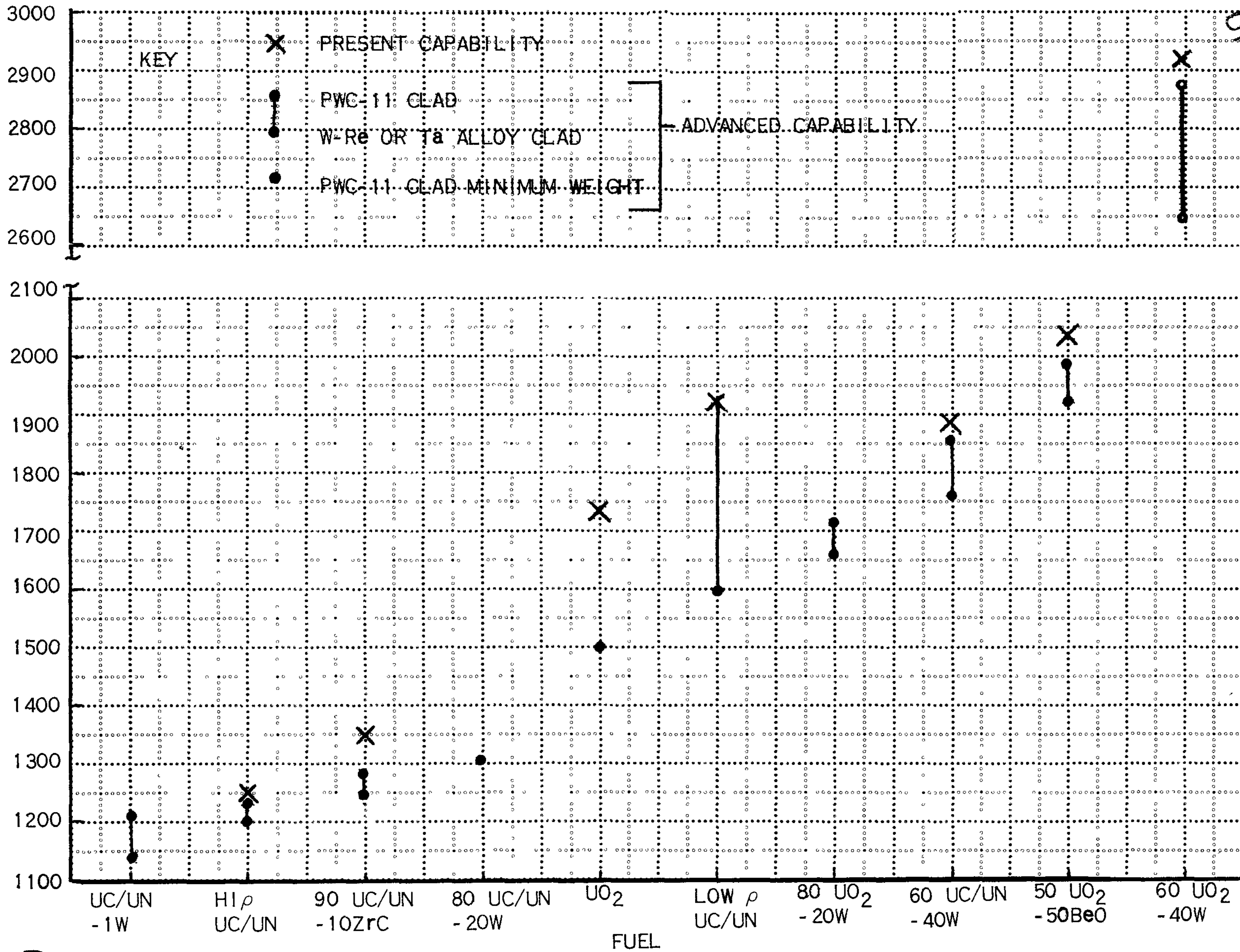

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FIG 1

PARAMETRIC REACTOR WEIGHT COMPARISON FOR VARIOUS FUELS AND CLADS

(CONTINUED)

8 MW REACTORS

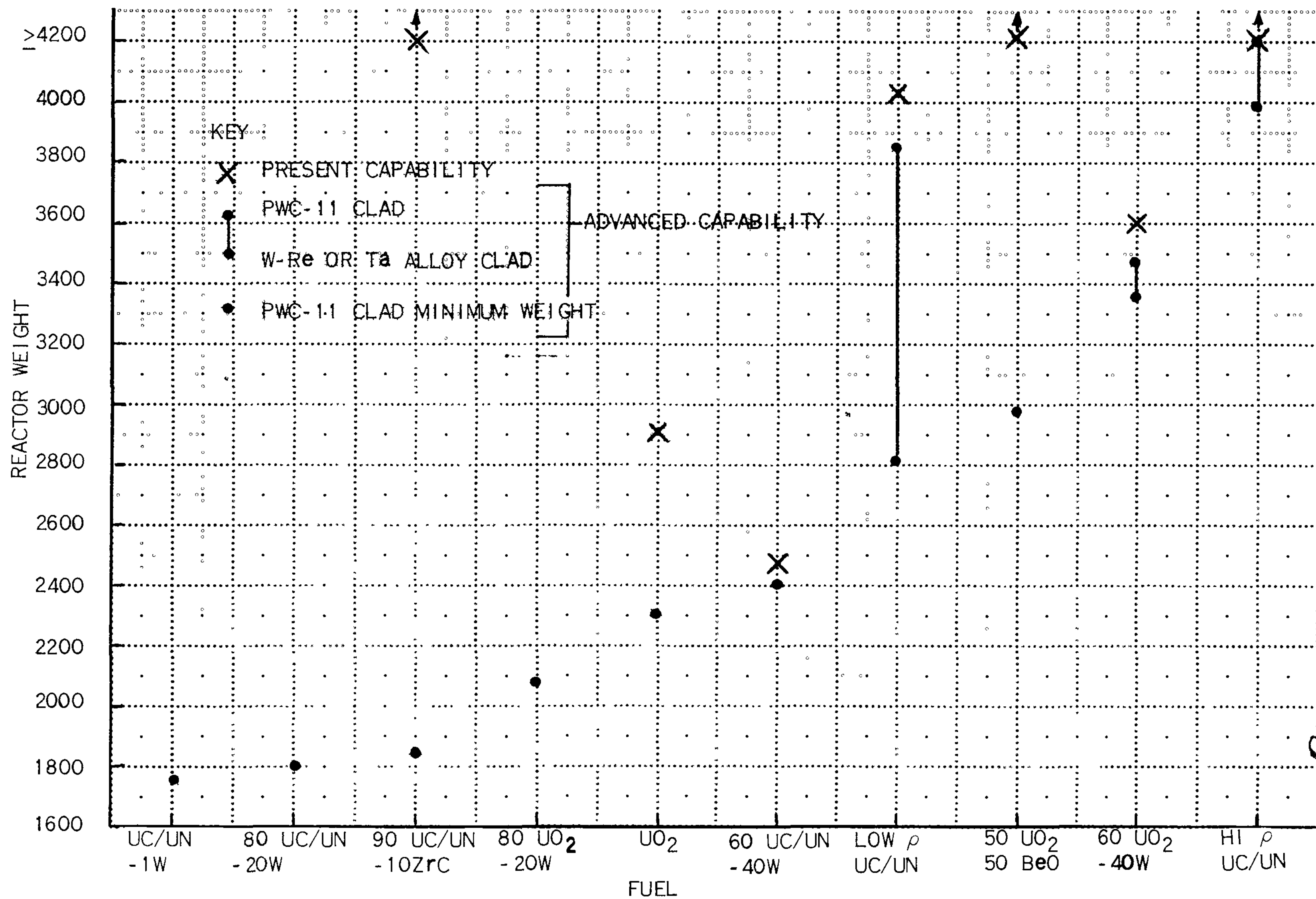

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FIG 1

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General Characteristics:

Parametric Cases 00,000, Fig. 6
Maximum Fuel Temperature, W/HCF $\leq$2300F
Effective Core Diameter - 10 inches (approx.)
Parametric Weight - 1300 pounds
Estimated Actual Weight - 1500 pounds

b. 2 Mw PWAR-20 Back-up Designs

(1) Fuel - 90 UC/UN-10 m/o ZrC or UC/UN-1 W

Clad - PWC-11 alloy

General Characteristics:

Parametric Case 11, Fig. 6
Maximum Fuel Temperature, W/HCF $\leq$2300F
Effective Core Diameter, 10.5 inches (approx.)
Parametric Weight, 1350 pounds
Estimated Actual Weight, 1600 pounds

(2) Fuel - UC/UN (95 percent dense), maximum fuel burnup 1.0 a/o (total uranium)

Clad - PWC-11 alloy

General Characteristics:

Parametric Case 6, Fig. 6
Maximum fuel temperature, W/HCF $\leq$2300F
Effective Core Diameter - 11.5 inches (approx.)
Parametric Weight - 1650 pounds
Estimated Actual Weight - 1900 pounds

4. Parametric Studies

The following additional reactor parametric studies are recommended:

a. 2 Mw reactors with plutonium and U^{235} fuels to determine their potential for this reactor application.

b. Uranium dicarbide (UC_2) and UC_2-coated particle fuels at both 2 and 8 Mw to determine their relationship with other uranium carbide type fuels considered in the present study.

c. Extend 2 Mw UC/UN base fuel studies to determine effects of increased fission gas release, coolant pressure drop, fuel-to-clad thermal resistance and fuel limits of 0.350 inch and 0.250 inch on parametric reactors.

d. Specific investigation of UN fuel at 2 Mw power level to determine parametric design.

III. DESCRIPTION OF PARAMETRIC METHOD

A. General Procedure

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1. Model

The basic tool used for parametric design studies is the Weight Optimization Parametric computer code (Ref. 3) which calculates reactor weights and dimensions from given physics data, desired operating conditions, and material properties, and selects the minimum weight reactor satisfying all input operating limitations.

The reactor model assumed for these calculations (Fig. 2) consists of a hexagonal core made up of pin-type fuel elements, arranged in a triangular pitch array with spacing between pins provided by spiral wires. Fuel pins contain the fuel matrix, end reflectors on each end of the core, and a void region for containment of fission gases. The fuel assemblies are supported by a core support plate. The reactor pressure vessel is a cylindrical shell with 2:1 ellipsoidal headers on each end. Core coolant inlet and outlet pipes are both located in the bottom header, with inlet coolant reaching the upper header through an annulus between the core and pressure vessel. Reactor control is provided by a movable side reflector outside the reactor pressure vessel.

2. Physics Input

The optimization code requires the fuel volume fraction, the total maximum-to-average power ratio, the radial-to-average power ratio, and the side reflector thickness over a range of the independent variables which are the effective core diameter, the core length-to-diameter ratio, the end reflector thickness, and the core coolant volume fraction.

The number of physics calculations required is minimized by utilizing the Composite Design technique (Ref. 3) to pre-select the cases to be run within the given range of the independent variables. Each set of calculated values is fitted to a second-order equation in terms of the independent variables by the method of least squares. The coefficients of these equations are then used by the optimization code.

3. Optimization Procedure

For each combination of the independent variables (core diameter, core length-to-diameter ratio, end reflector thickness, and core coolant volume fraction) considered during the optimization procedure, the corresponding reactor specifications are calculated as follows:

a. The second-order physics equations are solved for the core fuel volume fraction, the maximum-to-average power ratios, and the side reflector thickness.

b. The fuel pin diameter, matrix diameter, cladding thickness, and spacing are determined from the desired maximum fuel temperature and the core volume fractions on a unit triangular cell basis.

c. The length of the fuel pin void region required for the containment of helium and fission gases is determined as a function of power level, reactor lifetime, and pin dimensions.

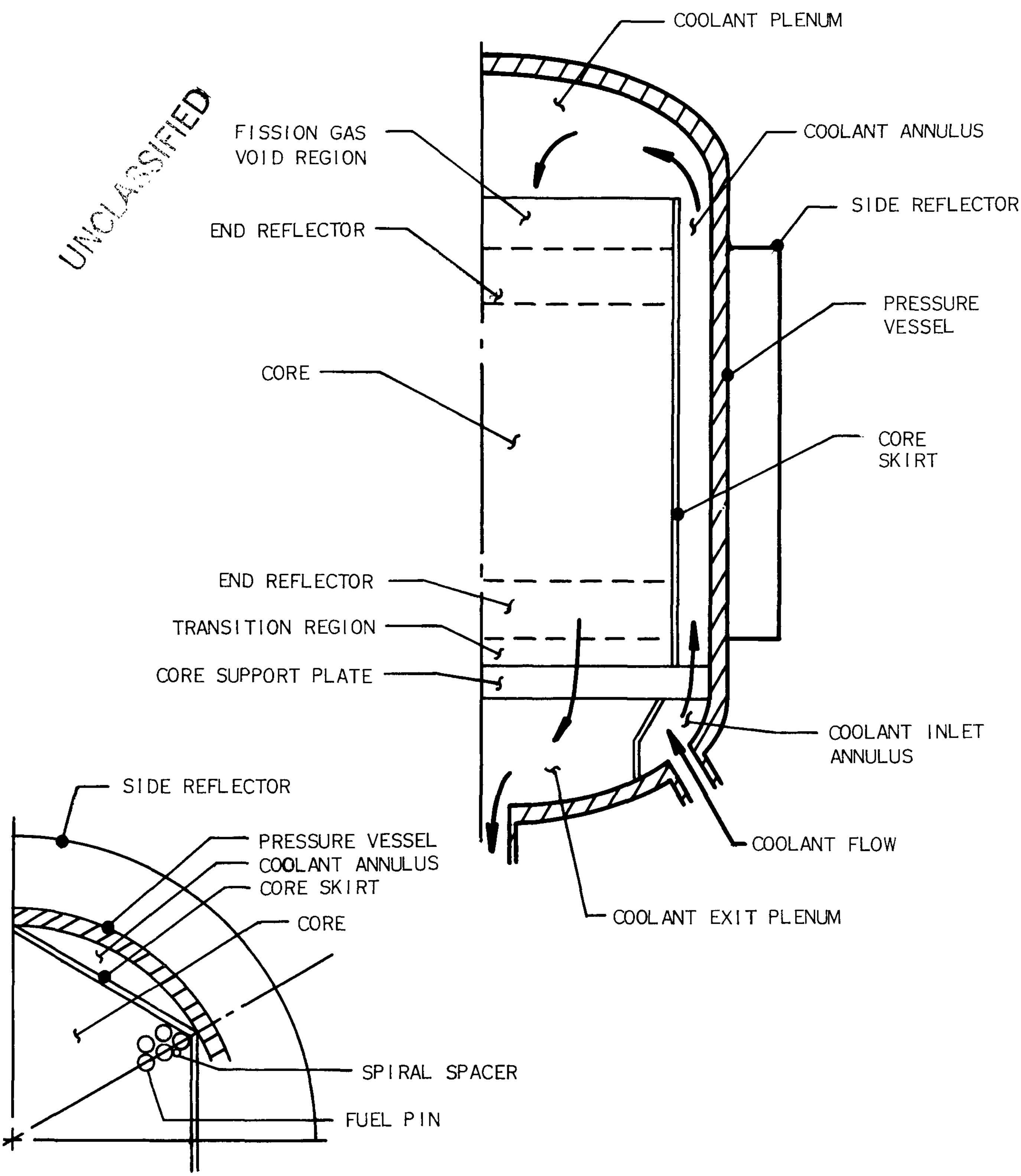
d. The total reactor pressure drop is the sum of the frictional pressure loss in the core and coolant annuli and the expansion and contraction losses through the upper and lower headers and from the inlet and outlet pipes.

e. The pressure vessel thickness is determined from the internal pressure level and stress-rupture strength of material including a safety factor.

f. The core plate thickness is taken as the maximum of two calculations based on 1) design launch loads and the low temperature yield strength and 2) operating pressure forces and

FIG 2

REACTOR PARAMETRIC STUDY MODEL



the high temperature stress-rupture strength.

g. Using these dimensions, the code calculates the total weight of the reactor, the over-all reactor dimensions, and such quantities as fuel burnup and power density.

These calculations are done throughout the optimization procedure where they are used in comparing reactor weights and in determining relationships to limitations imposed as input conditions.

The following procedure is used by the Weight Optimization Parametric code to determine which cases to consider within the input range and finally to select the minimum weight reactor satisfying the input conditions:

a. The fuel element cladding thickness is calculated over the range of the variables, and the calculated values are fitted to a second-order equation in terms of the independent physics variables by the method of least squares.

b. Values are selected for the core length-to-diameter ratio, the core diameter, and the end reflector thickness. The equation found above in (a) is solved for the core coolant volume fraction corresponding to the input value of the minimum allowable cladding thickness.

c. Five cases of varying coolant volume fraction from the bottom of the range to the volume fraction found in (b) are run to find the coolant pressure drop as a function of the coolant volume fraction.

d. The case with the coolant volume fraction corresponding to the input value of the desired pressure drop is run. If none of the input limits such as burnup, heat flux, or maximum side reflector thickness are exceeded, the specifications for the case are stored. Otherwise the case is not considered further in the optimization.

e. Steps (b) through (d) are performed over the complete range of end reflector thickness. A curve of reactor weight versus end reflector thickness is thus obtained for the selected core length-to-diameter ratio and core diameter. The specifications for the minimum weight reactor along this curve are stored.

f. Steps (b) through (e) are performed over the complete range of core diameter. A curve of reactor weight versus core diameter is thus obtained for the selected core length-to-diameter ratio. The specifications for the minimum weight reactor along this curve are stored.

g. Steps (b) through (f) are repeated for the complete range of core length-to-diameter ratios to obtain a curve of reactor weight versus core length-to-diameter ratio. The minimum weight reactor along this curve is the weight-optimized reactor which satisfies the input conditions.

During the Phase II and III studies, this full optimization procedure was used to examine the reactors under consideration.

For the Phase II studies, the major assumptions affecting the reactor weight were 1) the use of maximum fuel temperatures of 3200F without hot channel factors, 2) a columbium alloy clad 10,000-hour rupture strength of 4000 psi, and 3) thin side reflectors (about 1.5 inches) giving a shutdown reactivity of 0.97 with reflector removed.

During Phase III studies, lower values of fuel temperature and clad strength were examined and the Phase III reactor (Ref. Design 2) temperature of 2500F with hot channel factors and a columbium alloy clad strength of 1500 psi. Consideration of reactor handling and safety problems resulted in the side reflector thickness being increased from the approximate 1.5 inches in the Phase II studies to 4.0 inches.

The 4.0 inches was estimated to be the minimum thickness required to insure that with the reflector removed the reactor would be subcritical when reflected by an infinite thickness of water. This resulted in lowering the parametric reactor base reactivity from 0.97 to about 0.93. The core to reflector distance was increased to reflect engineering studies which had been done at that time. The Ref. 2 reactor was based on a 19-can core configuration, rather than the 7-can configuration used in Phase II.

B. Simplified Procedure

1. Optimization Assumptions

Because of the limited time available for completion of the fuel survey study, it was necessary to reduce the number of physics calculations normally required for input to the weight optimization code. This was accomplished by reducing the number of independent physics variables from four to two, the core diameter and the core coolant volume fraction. Experience with the parametric studies showed that for reasonable input conditions, the weight-optimized reactors usually have core length-to-diameter ratios and end reflector thicknesses of approximately 1.1 and 3 inches, respectively. These values were chosen as constant values for this study. In addition to the reduction in the number of independent variables, the number of calculations for each combination of these variables was reduced to one, the determination of the core fuel volume fraction required for criticality assuming a constant 4-inch side reflector in its most reactive position.

2. Physics Assumptions

The physics model used on the Phase IV studies was based on the following assumptions:

a. Constant end reflector thickness - 3 inches (each end)

b. Constant core length-to-diameter ratio - 1.1

c. Constant side reflector thickness - 4 inches

d. Maximum neutron multiplication factor - 1.05

e. Core-to-side reflector gap thickness T, and volume fractions -

$$T = 0.07\, D_c + 0.788 \text{ inch}$$

$$\text{Cb Volume Fraction, } VF_{cb} = D_c + 3.0/D_c + 11.3$$

$$\text{Void Volume Fraction, } VF_{void} = 0.3/T$$

$$\text{Li Volume Fraction, } VF_{li} = 1 - VF_{cb} - VF_{void},$$

where D_c is the core diameter.

The equation for the thickness and composition of the core-side reflector gap was empirically derived from preliminary design studies in this region.

The physics calculations were carried out with the use of the two-dimensional neutron transport theory multi-group code TDC. The neutron cross-sections used in these calculations, with the exception of the tungsten cross-sections used for the tungsten cermet fuels, were taken from LAMS-2543, although the absorption cross-sections for columbium and tantalum were altered to obtain better agreement between calculated worth and measured material coefficients. Tungsten cross-sections were prepared from basic resonance data and comparison with critical experiment material coefficient results.

The physics calculations for the tantalum alloy clad cases assumed a constant six percent volume fraction of columbium for the core structure other than fuel element cladding. The physics data produced for these tantalum cases was assumed to hold also for the tungsten-rhenium alloy clad cases.

Because all calculations were run with the side reflector in the maximum reactivity position, the startup maximum-to-average power ratios had to be obtained by scaling the calculated end-of-life values to agree with previous calculations of startup values.

3. Engineering Assumptions

The engineering model for the fuels study is fixed in the optimization code (Ref. Section III A-1). However, the reactor weight printed out by the optimization code was scaled to reflect the following characteristics which are in closer agreement with results of design and engineering accomplished to date and the configuration shown in Fig. 3.

a. Seven-can, rounded-edge core

b. Pressure vessel sized considering hoop stress only

c. Approximately 250 pounds for local reactor support structure, bearings, and control drives including drive shaft, gears, etc.

A 0.005-inch tungsten barrier was included between the fuel matrix and the fuel element cladding in all cases where columbium or tantalum alloy cladding material was used with the following fuels:

a. UC/UN

b. 90 UC/UN-10 m/o ZrC

c. UC/UN-1 W

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Since the same physics data was used for both tantalum and tungsten-rhenium cladding cases with these fuels, the only difference between the cases was the exclusion of the fuel-cladding barrier in the tungsten-rhenium clad fuel elements.

In every case in the study, a factor of 90 percent was applied to the cladding stress rate-to-rupture to account for the possibility of small variations in tubing thickness. Based on fabrication considerations, the fuel pin cladding thickness was limited to a minimum of 0.015 inch.

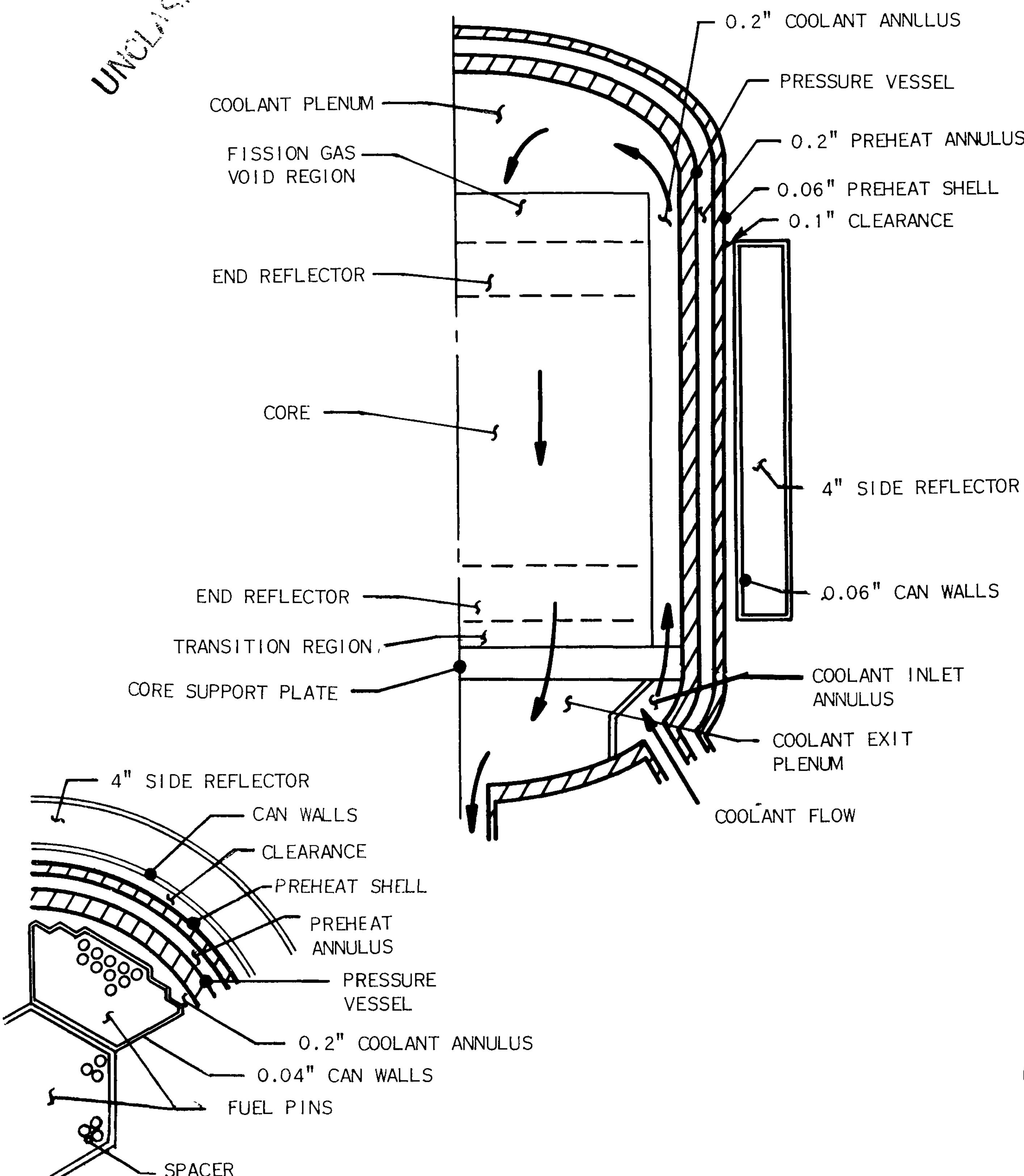
In all cases, it was assumed that there was a constant fuel-to-matrix contact resistance equivalent to 0.0005 inch of helium. This assumed contact resistance is the same as that used to calculate CANEL irradiation capsule maximum fuel temperatures.

Reactor design criteria presently specify that the fuel pin swelling or creep due to hoop stress produced in the clad by matrix swelling and fission gas containment be limited to one percent in 10,000 hours. The secondary creep strain rate is a function of stress and temperature and, at the conditions presently anticipated to exist in the reactor core, insufficient test data is available to allow satisfactory prediction of cladding behavior. For this reason, although results of a preliminary analysis show that a fuel pin swelling of nearly five percent is required to raise cladding surface temperatures to the 2200F design value, the maximum allowable strain is limited to one percent, which will not be exceeded if more than one-half of the stress-rupture strength is used as a design basis. The foregoing is based on the established creep properties of Cb-1 Zr alloy. The stress rupture criteria are used to design the parametric reactor fuel pins for gaseous fission product containment. Fuel swelling effects are considered by applying a fuel burnup limit to those fuels (UC/UN, UO_2-BeO) for which matrix swelling is indicated to be a function of fuel burnup at SNAP-50/SPUR reactor conditions based on irradiation testing to date. For other fuels, it is assumed that fuel pin diametral swelling will be less than one percent.

During the course of this study, it was found that many burnup-limited cases were larger and heavier than required, due to the assumption of a constant high fuel enrichment. Consequently, the weight optimization code was revised to allow lowering of the fuel enrichment if this resulted in a weight saving.

FIG 3

REACTOR WEIGHT DESIGN MODEL



IV. SUMMARY OF RESULTS OF REACTOR DESIGN AND PARAMETRIC STUDIES, 1961 TO OCTOBER 1963

Three distinct steps in UC/UN reactor studies for space applications are apparent, spanning the time from start of space powerplant investigations at CANEL in 1961 to October 1963, when the study was expanded to consider other fuels. The basic ground rules for these reactor studies were:

Reactor coolant	Lithium
Reactor structural material	Cb-1 Zr alloy
Reactor coolant outlet temperature	2000F
Reactor lifetime	10,000 hours
Uranium carbide U^{235} enrichment	84 percent

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A summary of this work is shown in Fig. 4, which shows the history of predicted SNAP-50/SPUR reactor weights as a function of fuel and cladding performance assumptions.

A. Phase I, Preliminary Conceptual Studies, November 1961 - July 1962

The first space reactor data (Part A) was scaled from the UC/UN-fueled PWAR-18 reactor which was typical of the Lithium-Cooled Reactor Experiment (LCRE) reactor, then being designed at CANEL, and which featured liquid metal-cooled reflectors with neutron absorber sections contained in a rotating control drum configuration. The UC/UN fuel and clad performance design assumptions were identical to those used in the LCRE reactor design, and were considered realistic at that time. The initial scaling included the effects of changing the coolant temperature drop from 400F to 100F, reduction in design power from 10 Mw to 8 Mw, and revising control drive weights.

During this phase (Part B), the reactor concept was revised to use radiatively-cooled reflectors. Fuel and cladding design assumptions remained unchanged during this concept revision. The weight data available from engineering work accomplished on the LCRE reactor type was scaled and adjusted to obtain the 1300-pound reactor weight shown for the initial radiatively-cooled reactor concept. During this time, powerplant considerations specified an 8-Mw reactor weight target of 1150 pounds.

Also during this phase of space reactor design, parametric studies as described in this report were not conducted. The proposed reactors were scaled from work accomplished on the LCRE type reactor.

B. Phase II, Parametric Design and Application Studies, December 1962 - October 1963

The first parametric studies were accomplished in this period, based on the reactor concept evolved in Phase I. The first series of studies was conducted to provide data for powerplant optimization studies and included reactor power levels of 2, 5 and 8 Mw, reactor coolant temperature rise variations of 50F to 150F and reactor coolant pressure drops of 10 to approximately 120 psi. The studies showed the reactor weight difference between the 2-Mw (250 to 300-Kwe space powerplant) and 8-Mw (1 Mwe) reactor designs to be approximately 200 pounds. Because this predicted weight difference was small, a single 8-Mw reactor design having the capability of being used in 250-Kwe to 1000-Kwe powerplants was selected as the most practical reactor approach for the SNAP-50/SPUR power systems.

These reactor parametric study results were utilized in a powerplant optimization study (Ref. 4) which showed 8-Mw, 100F coolant temperature rise, and 30 psi coolant pressure drop to be optimum for a 1-Mwe powerplant. The weight of the reactor meeting the above requirements was determined to be 890 pounds, including meteoroid shielding. The reactor was selected for the SNAP-50/SPUR First Flight Powerplant Preliminary Design Specifications (Ref. 5) and designated Reference Design No. 1.

HISTORY AND COMPARISON OF UC REACTOR WEIGHT ESTIMATES FOR SNAP-50 APPLICATIONS

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LIFE 10,000 HOURS
COOLANT LITHIUM
COOLANT OUTLET TEMPERATURE - 2000F
UC FUEL - 84% ENRICHED

		Reactor Data as Published									Unpublished Data and Comments	Distribution of Total Weight, lb				
		Power, Mw	Burnup, %	Bare Core K	Matrix Temp, F	No. Pins & Pin Diameter, inches	Clad Thickness Cb-1 Zr, inches	Core Diameter, Fuel Length, Void Length, inches	Reflector Thickness, inches	Total Weight, lb	Comments	Side Reflector	End Reflector & Core	Liquid Metal	Pressure Vessel	Structure & Drives
Phase I - Preliminary Conceptual Studies											Fission Gas Release 7.5% Clad Strength 3400 psi at 2200F					
	A. BeO Reflectors, L.M. Cooled, Rotating Poison (B_4C) Drum Control															
11-7-61	CNLM-3886, Preliminary Design Specifications for the PWAR-18, UC Reactor	10	7.87(1)		2700, max	217(3) per can 0.295	0.025	10	6	*1932 (2340)	PWAR-16 reactor- reflector configuration. Coolant ΔT 400F. Bare core K estimate 0.85. *Weight underestimated - correct figures shown in parentheses	*400 (810)	410	32	230	860
5-22-62	CNLM-4083, Preliminary SNAP-50 Reactor Weight Parametric Study	8							6	1632	This reactor scaled from PWAR-18 coolant ΔT 100F. Optimized control drives.					
	B. Radiatively Cooled External Reflector															
5-3-62	CNLM-4068, SNAP-50 Reference Requirements	8								1200	1200 lbs stated to be a guessed weight					
5-24-62	MPR-62-4-1, Monthly Progress Report, April 1962										Parametric studies for Phase II were initiated April 16, 1962					
7-17-62	CNLM-4168, SNAP-50 Reference Requirements	6.5								1304	Major reasons for weight change from 5-22-62 are power reduced 7.76 Mw to 6.5 Mw, outlet pressure increased 21 psia to 51 psia, reactor ΔP increased 30 psi to 40 psi.					
7-18-62	CNLM-4167, SNAP-50 Basic System Requirements	8								1150	Intended to guide component designers indicating attainable target with vigorous design and development					
Phase II - Parametric Design and Application Studies											Maximum matrix temp, 3400F Fission gas release, 10% Clad strength, 4000 psi					
12-27-62	CNLM-4357, SNAP-50 System Optimization Studies	2 5 8								510-627 660-769 718-1020	Without meteoroid shield (120 lb)					
1-15-63	CNLM-4385, SNAP-50 Reactor Development Concept	Optim.2 8 8 at 2	1.77(1) 5.21(1) 1.30	0.97 0.97 0.97	3200 3200 2262	0.746 0.405 0.405	0.0304 0.015 0.015	9 1.89 10 7.3 10 7.3	1.5 1.7 1.7	500 710 710	Without meteoroid shield (120 lb)	80 120 120				
2-4-63	CNLM-4372, Parametric Results for UC Fueled SNAP-50 Type Reactors	8		0.97	3200	0.405	0.015	10 7.3	1.5	830	Including meteoroid shield (120 lb)	122	407	15	82	204
6-10-63	PWAC-631, Quarterly	8	4.9(1)		3200	434 0.432	0.025	10.43 10.43 3.00	1.6	860	Including meteoroid shield weight estimate revised from 2-4-63					

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FIG 4

HISTORY AND COMPARISON OF UC REACTOR WEIGHT ESTIMATES FOR SNAP-50 APPLICATIONS

CONTINUED

Date	Report	Power, Mw	Burnup, %	Bare Core K	Matrix Temp, F	No. Pins & Pin Diameter, inches	Clad Thickness Cb-1 Zr, inches	Core Diameter, Fuel Length, Void Length, inches	Reflector Thickness, inches	Total Weight, lb	Comments	Side Reflector	End Reflector & Core	Liquid Metal	Pressure Vessel	Structure & Drives
		Reactor Data as Published									Unpublished Data and Comments	Distribution of Total Weight, lb				
6-21-63	CNLM-5118, SNAP-50/SPUR First Flight Powerplant Preliminary Design Specifications	8								890	Including meteoroid shield. Revised weight estimate from 2-4-63.					
7-26-63	PWAC-406, Study of Adapting SNAP-50 Space Powerplant to a Lunar Base	8								652	Spherical core shape, no meteoroid shield. Design not considered to be practical.					
8-27-63	MPR-63-7-1, Monthly Progress Report, July 1963	8								1100-2700	Based on parametric studies 2300F-3200F. Fuel temperature 1200-6000 psi Clad strength, 3-50% Fission gas release					
10-4-63	CNLM-5275, Presentation of Adapting SNAP-50 Powerplant to a Lunar Base	8								890	Revised weight estimate from 2-4-63, with meteoroid shield					
Phase III - Parametric and Preliminary Engineering Design Studies											Maximum matrix temp, 2500F Fission gas release, 10% Clad strength, 1500 psi at 2200F Core diameter, 12 inches Reflector thickness, 4 inches Burnup, 4 3%(2)					
10-18 63	MPR 63-9-1, Monthly Progress Report, September 1963	8		0 93		1603 0.254	0.022	12.0 13.2 9		2000*	Major Reasons for Reactor Weight Change, Phase II to III	450	850	10	295	220
11-25-63	PWAC-633, Quarterly Progress Report, July 1, 1963 to September 30, 1963	8		0.93		1603 0.254	0.0225	12 0 13.2 9	4 0	1996*	Clad strength, 4000 to 1500 psi (47 % weight change) Bare core K, 0.97 to 0 93 (17% weight change) Upgrading of nuc. data (17% weight change) Matrix temp, 3200F to 2300F (18 % weight change) *Includes 175 lbs of SNAP-50 powerplant wiring and piping not included in other wt. estimates.					
Phase IV Present											Maximum matrix temp, 2300F Fission gas release, 10% Clad strength, 1500 psi Maximum fuel burnup, 1.5%					
12-9-63	CNLM-5340, SNAP-50/SPUR Reference Design No. 3	2				1600		12 0 13 2		1000	Flight test reactor wt goal					
	MPR-63-11-1, Monthly Progress Report, November 1963	8	4.3(2)		2500 max					2000	Same as 10-18-63					
		2	1.5(2)		2300 max					1400-1600	Estimated wt. of SNAP-50 initial flight reactor designed for 2 Mw (Ref. Design No. 3).					

(1) Avg total burnup, % U^{235} mass
(2) Max. fission burnup, total uranium mass
(3) 7 cans - 1519 pins.

Burnup Conversion

Avg. Total Burnup, % U^{235} Mass	Max. Fission Burnup, % Total Uranium Mass.
7 87	6.10
1.77	2.04
5.21	6.10
1.30	1.521
4.90	5.89

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FIG 4

The fuel operating temperature in these studies during the early part of Phase II (3200 w/o HCF) was selected on the same basis as in UO_2-BeO reactor design (LCRE) - 75 percent of the fuel melting temperature. The clad strength assumption (4000 psi) was also known to be optimistic for presently available materials. It was recognized that an extensive development program was required to substantiate these fuel and clad performance assumptions, with a fair possibility that these conditions could not be successfully demonstrated.

Therefore, a new series of parametric studies was initiated in the latter part of Phase II to determine the effect of fuel performance assumptions on 8-Mw UC/UN-fueled reactors. A range of fuel and cladding conditions was investigated, the lower limits of which now appear to be within present fuel and clad capability. In addition, the nuclear mockup was for this study changed to a 19-fuel can core configuration and nuclear cross-sections were modified. Parametric reactor weights ranging between 1100 and 2700 pounds were indicated by this study.

C. Phase III, Parametric and Preliminary Engineering Design Studies, October 1963

A review of fuel and cladding performance potential indicated that the performance assumptions associated with 8-Mw reactor weighing less than 1000 pounds (Reference Design No. 1) were too optimistic. Accordingly, fuel and cladding design assumptions were reduced:

Maximum fuel temperature from 3400F to 2500F W/HCF

Fuel cladding strength from 4000 to 1500 psi and a compatibility barrier incorporation

These reduced fuel assumptions and the incorporation of a four-inch thick side reflector for handling safety considerations (core base reactivity reduced from 0.97 to 0.93) with the 19-can core reactor parametric model resulted in an 8-Mw reactor weight of 2000 pounds.

The major factors, and their relative effect in increasing the reactor weight from approximately 890 pounds in Phase II to 2000 pounds in Phase III, as determined by changing the variables one at a time using the Phase II reactor mockup are:

Factor and Change	% of Weight Change
Fuel cladding strength, 4000 psi to 1500 psi and incorporation of compatability barrier	47
Fuel maximum temperature w/o HCF 3200F to 2300F	18
Nuclear mockup, 7 to 19 can, and more accurate material cross-sections	18
Reactor K_{min} reduced, 0.97 to 0.93	17

V. RESULTS OF PARAMETRIC STUDIES, OCTOBER 1963 TO PRESENT

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A. Fuels and Clad Assumptions

Late in the Phase II effort, it was determined that the fuel design assumptions used in these studies could not be substantiated (Ref. 2) and further reduction in parametric study design assumptions was required. Using conservative fuel design assumptions, the weight of optimized 8-Mw reactors was clearly in the order of two times the weight of 2-Mw reactors. Accordingly, the parametric studies were expanded to consider other fuel and clad possibilities at 2-Mw and 8-Mw power levels. The range of fuel and clad assumptions is shown in Fig. 5. The performance levels cover values which 1) appear with reasonable confidence to be demonstrable and are considered within present capability of fabrication and test, denoted as Present Capability, and 2) advanced performance, denoted Advanced Capability which might be demonstrated in the future if sufficient promise in powerplant performance improvement warrants such an approach.

As noted in Fig. 5, the fuel cladding materials studied were columbium alloys with assumed 10,000-hour rupture strengths ranging from 750 to 2000 psi, a tantalum alloy and a tungsten-rhenium alloy, each with an assumed 10,000-hour rupture strength of 4000 psi. The fuel candidates examined were high and low density UC/UN (UC and UN fuel have approximately the same uranium density and thermal conductivity, so that in parametric studies they are examined as one fuel with reasonable accuracy), UO_2, UO_2-BeO with 50 volume percent BeO, UC/UN-Zrc with 10 m/o ZrC, UC/UN-1 W, UC/UN-W cermets with 20 and 40 volume percent tungsten, and UO_2-W cermets with 20 and 40 volume percent tungsten.

The low density UC/UN fuels assuming 100 percent gas release were considered as one alternative to improve UC/UN dimensional-stability, temperature, and burnup limitations by eliminating fission gas pressure buildup in the matrix. The UC/UN-W and UC/UN-ZrC fuels appear to offer an alternative approach to this problem by increasing the matrix creep strength. This is indicated by out-of-pile creep tests of UC/UN-1 W (Ref. 6) which show this material has a significantly higher creep strength than UC/UN. Also hot hardness tests at CANEL indicate the UC/UN-10 m/o ZrC hardness is greater than for UC/UN (Ref. 7). Each fuel was examined with each appropriate cladding material.

The UC/UN fuel was investigated for three densities. The high density fuel was assumed to have relatively low fission gas release ranging from one to ten percent, and the lower density fuels to have 100 percent fission gas release. The other fuel candidates were generally assumed to have ten percent fission gas release, except for UO_2-BeO. The UO_2-BeO was assumed to have a 25 percent fission gas release which is estimated to be equivalent to a 100 percent fission gas release plus a 20 percent helium release based on helium to gaseous fission product generation rates determined for the LCRE reactor. (Ref. 8).

Fuel maximum centerline temperatures were varied over a range which seemed consistent with the assumed values of gas release and fuel element swelling as indicated by presently available data. In general, this restricted temperatures to 2300F for present capabilities and up to 2700F to 2900F for advanced fuel capability. Where present data seemed to indicate required limitations on burnup in order to maintain limited swelling or gas release, these limits were imposed on the fuels.

Fuel thermal conductivities for the UC/UN, UO_2-BeO, and UO_2 fuels used in the studies are based on experimental measurements. Those for the cermet and UC/UN-ZrC fuels were determined analytically by the volumetric ratio of constituents, or for the latter from electrical resistivity measurements. The fuel conductivities (Btu/hr-ft-F) used for each fuel are:

UC/UN, 95% theoretical density	11.0
UC/UN, 87% theoretical density	10.1
UC/UN, 80% theoretical density	9.26

SUMMARY OF FUEL PERFORMANCE VARIATIONS STUDIED AT 2-MW AND 8-MW DESIGN POWER LEVELS

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	Fuels Studied for Present and Advanced Capability							Fuels Studied for Advanced Capability Only		
	UC	UN	90 UC-ZrC	60 UC-W	UO_2	60 UO_2-W	UO_2-BeO	UC-1W	80 UC-W	80 UO_2-W
1. Fuel Density	0.80 0.87	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95
2. Maximum Fuel Temperature, F										
Present Capability	2500	2300	2300	2300	2300	2300	2300	--	--	--
Advanced Capability	2700 2900	2500 2700	2500	2500	2700	2700	2300 2500 2700	2500	2500	2700
3. % Gas Release										
Present Capability	100	1, 10	10	10	10	10	25	--	--	--
Advanced Capability	100	10	10	10	10	10	25	10	10	10
4. % Burnup (Design Limit)										
Present Capability		1 1.5 2.0					3.0			
Advanced Capability		1.5					3.0 6.0			
5. Fuel Claddings										
Present Capability	PWC-11 (Cb-1 Zr-0.1 C) and Cb alloys									
Advanced Capability	PWC-11, T-222, and W-Re									

90 UC/UN-10 m/o ZrC	10.4
UC/UN-1 W	11.0
80 UC/UN-20 W	15.6
60 UC/UN-40 W	23.2
UO_2	1.4
50 UO2-50 BeO	6.7
80 UO_2-20 W	8.3
60 UO_2-40 W	16.8

B. 2-Mw Reactors

Results of the 2-Mw reactor fuel parametric studies are tabulated in Figs. 6 and 7.

C. 8-Mw Reactors

Results of the 8-Mw reactor fuel parametric studies are tabulated in Figs. 8 and 9.

D. Effect of Fuel and Clad Variables on Reactor Weight

1. Fuel Versus Reactor Weight

In Table 1 is presented the approximate range of reactor weights for each fuel material as extracted from the study (neglecting very low-strength columbium alloy cladding strength cases). Two-Mw reactor weights for operating conditions in the category of present fuel capability fall in the range of 1200 to 3000 pounds, while the present capability 8-Mw reactor weights are generally heavier than the minimum 2-Mw reactor weights by a factor of two or more.

Table 1
Summary of Reactor Weights From Parametric Studies

	2-Mw		8-Mw	
	Present Capabilities	Advanced Capability	Present Capabilities	Advanced Capability
UC/UN 0.95 density	1200[a]-1600[b]	1200[a]	>4200[a]	>4200[a] (1900)
UC/UN 0.87, 0.80 density	1600[c]-1900	1500-1900	3200[c]-4000	2700-3800
UO_2	1700-2000	1500-1800	2900	2300-3000
50 UO_2-50 BeO	1800[c]-2000	1800-2200	2700[c]-3400[d]	3000-3400[d]
90 UC/UN-10 m/o ZrC	1400	1300	4000[a]	1900-2100
60 UC/UN-40 W	1900	1800	2500	2400-2550
60 UO_2-40 W	3000	2600-3100	3600	3400-3500
80 UC/UN-20 W		1300		1800-1900
80 UO_2-20 W		1700		2300-2400
UC/UN-1 W		1200		1800-1900

(a) Fuel burnup limit 1.5% total uranium
(b) Fuel burnup limit 1.0% total uranium
(c) PWC-11 clad design temperature 2100F
(d) Fuel burnup limit 3.0% total uranium

FIG 6

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SNAP-50 2-MW REACTORS PRESENT CAPABILITY

COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	0	00[4]	000[4]	1	2	3
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10	10	10
Design Fuel Temp, F WHCF	2100	2150	2300	2300	2300	2300
Clad Material	PWC-11	PWC-11	PWC-11	PWC-11	PWC-11	Cb Alloy
Design Clad Temp, F	2200	2200	2200	2200	2100	2200
Design Clad Rupture Stress, psi	1500	1500	1500	1500	2070	750
Fission Burnup, %[1]	1.28	1.5[2]	1.5[2]	1.5[2]	1.5[2]	1.36
Clad Thickness, in.	0.025	0.016	0.020	0.019	0.017	0.038
Maximum Power Density Kw/cc, w/o HCF	0.36	0.42	0.42	0.42	0.42	0.38
Pin Diameter, in.	0.250	0.250	0.350	0.357	0.354	0.406
Pin Spacing, in.	0.018	0.014	0.016	0.029	0.029	0.029
Number of Pins, approximate	1653	1212	599	576	575	563
Effective Core Diameter, in.	12.00	10.15	9.85	10.15	10.05	11.30
Core Length, in.	13.20	11.16	10.83	11.17	11.06	12.43
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	1.9	5.2	7.2	5.7	3.4	12.0
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	31.0	30.9	32.4	31.5	28.9	39.8
Reactor OD, in.	22.0	19.9	19.6	19.9	19.8	21.2
Core Coolant Reynolds Number, average	4300	5900	8600	8500	8600	7700
Weight, lb	1639	1276	1240	1255	1205	1647

(1) Atoms of U^{235} fissioned per original uranium atom, maximum.
(2) Case reached design burnup limit. Enrichment lowered.
(3) Case run at a reactor pressure drop less than 5 psi.
(4) Coolant pressure drop 10 psi.

FIG 6

SNAP-50 2-MW REACTORS PRESENT CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

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Case No.	4	5	6	7	8
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.95	0.95	0.95	0.87	0.87
Gas Release (Helium plus Fission), %	1	10	10	100	100
Design Fuel Temp, F WHCF	2300	2300	2300	2500	2500
Clad Material	PWC-11	PWC-11	PWC-11	PWC-11	PWC-11
Design Clad Temp, F	2200	2200	2200	2100	2200
Design Clad Rupture Stress, psi	1500	1500	1500	2070	1500
Fission Burnup, %[1]	1.5[2]	1.55	1.0[2]	1.40	1.33
Clad Thickness, in.	0.015	0.016	0.020	0.030	0.043
Maximum Power Density Kw/cc, w/o HCF	0.42	0.44	0.28	0.36	0.34
Pin Diameter, in.	0.351	0.344	0.441	0.486	0.520
Pin Spacing, in.	0.028	0.032	0.025	0.041	0.039
Number of Pins, approximate	575	582	512	360	356
Effective Core Diameter, in.	9.95	9.95	11.50	10.90	11.50
Core Length, in.	10.95	10.95	12.65	11.99	12.65
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	0.9	8.2	6.7	23.4	26.6
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	26.3	33.6	34.9	50.4	54.7
Reactor OD, in.	19.7	19.7	21.5	20.8	21.4
Core Coolant Reynolds Number, average	8700	8600	8000	10,000	9500
Weight, lb	1158	1224	1652	1630	1893

FIG 6

SNAP-50 2-MW REACTORS PRESENT CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	9	10	11	12	13
Fuel	UC/UN	UC/UN	90 UC 10 ZrC	60 UC 40 W	UO_2
Fuel Density, %	0.80	0.80	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	100	100	10	10	10
Design Fuel Temp, F WHCF	2500	2500	2300	2300	2300
Clad Material	PWC-11	PWC-11	PWC-11	PWC-11	PWC-11
Design Clad Temp, F	2100	2200	2200	2200	2200
Design Clad Rupture Stress, psi	2070	1500	1500	1500	1500
Fission Burnup, %(1)	1.31	1.28	1.5(2)	1.32	1.12
Clad Thickness, in.	0.031	0.039	0.018	0.027	0.016
Maximum Power Density Kw/cc, w/o HCF	0.31	0.30	0.38	0.22	0.24
Pin Diameter, in.	0.507	0.527	0.368	0.655	0.202
Pin Spacing, in.	0.036	0.037	0.030	0.016	0.015
Number of Pins, approximate	362	357	582	275	2747
Effective Core Diameter, in.	11.25	11.60	10.50	12.10	12.60
Core Length, in.	12.38	12.76	11.55	13.31	13.86
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	21.6	28.6	6.4	7.1	2.1
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	49.3	56.8	33.3	36.4	32.4
Reactor OD, in.	21.2	21.6	20.3	22.1	22.7
Core Coolant Reynolds Number, average	9700	9400	8200	10,300	3100
Weight, lb	1691	1921	1353	1882	1733

FIG 6

SNAP-50 2-MW REACTORS PRESENT CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	13-A	14	15	16	17[3]
Fuel	UO_2	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	60 UO_2 40 W
Fuel Density, %	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium Plus Fission), %	10	25	25	25	10
Design Fuel Temp, F WHCF	2300	2300	2300	2300	2300
Clad Material	PWC-11	PWC-11	PWC-11	Cb Alloy	PWC-11
Design Clad Temp, F	2200	2200	2100	2200	2200
Design Clad Rupture Stress, psi	1500	1500	2070	750	1500
Fission Burnup, %[1]	0.80	1.41	1.53	0.94	0.80
Clad Thickness, in.	0.015	0.023	0.016	0.067	0.038
Maximum Power Density Kw/cc, w/o HCF	0.17	0.15	0.16	0.10	0.10
Pin Diameter, in.	0.237	0.497	0.465	0.691	0.912
Pin Spacing, in.	0.017	0.018	0.020	0.007	0.020
Number of Pins, approximate	1800	596	617	502	229
Effective Core Diameter, in.	11.90	13.70	13.15	17.00	15.30
Core Length, in.	20.50	15.07	14.47	18.70	16.83
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	2.9	11.1	9.3	11.0	5.8
Side Ref. TK., in.	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	39.1	43.0	40.0	48.7	40.9
Reactor OD, in.	21.9	24.0	23.3	27.8	25.8
Core Coolant Reynolds Number, average	4100	6200	6300	5400	8900
Weight, lb	2000	2031	1809	3381	2596

FIG 7

SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	1	2	3	4	5	6
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.95	0.95	0.95	0.95	0.87	0.87
Gas Release (Helium plus Fission), %	10	10	10	10	100	100
Design Fuel Temp, F WHCF	2500	2700	2500	2500	2700	2900
Clad Material	PWC-11	PWC-11	T-222	W-Re	PWC-11	PWC-11
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	1500	4000	4000	1500	1500
Fission Burnup, %(1)	1.5(2)	1.5(2)	1.5(2)	1.5(2)	1.35	1.39
Clad Thickness, in.	0.022	0.029	0.016	0.017	0.046	0.049
Maximum Power Density Kw/cc, w/o HCF	0.42	0.42	0.42	0.42	0.35	0.36
Pin Diameter, in.	0.452	0.537	0.442	0.435	0.594	0.676
Pin Spacing, in.	0.035	0.036	0.032	0.033	0.045	0.051
Number of Pins, approximate	351	255	363	365	259	194
Effective Core Diameter, in.	9.95	9.95	9.85	9.75	11.30	11.00
Core Length, in.	10.95	10.95	10.84	10.73	12.43	12.10
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	7.7	6.0	2.0	1.8	30.4	34.2
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	33.1	31.4	27.1	26.9	58.3	61.2
Reactor OD, in.	19.7	19.7	19.6	19.4	21.2	20.9
Core Coolant Reynolds Number, average	11,100	13,000	11,000	11,100	11,500	13,500
Weight, lb	1237	1223	1216	1202	1880	1851

(1) Atoms of U^{235} fissioned per original uranium atom, maximum.
(2) Case reached design burnup limit. Enrichment lowered.
(3) Case run at a reactor pressure drop less than 5 psi.

FIG 7

SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

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Case No.	7	8	9	10	11	12
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.87	0.87	0.80	0.80	0.80	0.80
Gas Release (Helium plus Fission), %	100	100	100	100	100	100
Design Fuel Temp, F WHCF	2700	2700	2700	2900	2700	2700
Clad Material	T-222	W-Re	PWC-11	PWC-11	T-222	W-Re
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	4000	4000	1500	1500	4000	4000
Fission Burnup, %(1)	1.34	1.39	1.26	1.29	1.30	1.36
Clad Thickness, in.	0.021	0.022	0.049	0.053	0.022	0.022
Maximum Power Density Kw/cc, w/o HCF	0.35	0.36	0.30	0.31	0.31	0.32
Pin Diameter, in.	0.559	0.541	0.627	0.713	0.575	0.550
Pin Spacing, in.	0.044	0.044	0.037	0.042	0.037	0.040
Number of Pins, approximate	257	261	260	195	276	280
Effective Core Diameter, in.	10.50	10.30	11.65	11.45	11.05	10.75
Core Length, in.	11.55	11.33	12.82	12.60	12.16	11.83
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	17.5	16.6	26.5	28.8	14.9	15.8
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	43.7	42.5	54.9	56.7	42.0	42.5
Reactor OD, in.	20.3	20.1	21.6	21.4	20.9	20.6
Core Coolant Reynolds Number, average	12,300	12,500	11,000	13,000	11,300	11,500
Weight, lb	1563	1513	1919	1898	1662	1596

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SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	13	14	15	16	17	18
Fuel	UC-1 W	UC-1 W	UC-1 W	90 UC 10 ZrC	90 UC 10 ZrC	90 UC 10 Zrc
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10	10	10
Design Fuel Temp, F WHCF	2500	2500	2500	2500	2500	2500
Clad Material	PWC-11	T-222	W-Re	PWC-11	T-222	W-Re
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	4000	4000	1500	4000	4000
Fission Burnup, %[(1)]	1.57	1.51	1.58	1.61	1.51	1.59
Clad Thickness, in.	0.022	0.015	0.015	0.022	0.016	0.015
Maximum Power Density Kw/cc, w/o HCF	0.44	0.42	0.44	0.41	0.38	0.40
Pin Diameter, in.	0.443	0.443	0.422	0.450	0.456	0.433
Pin Spacing, in.	0.037	0.032	0.034	0.033	0.029	0.031
Number of Pins, approximate	354	362	368	378	386	394
Effective Core Diameter, in.	9.85	9.85	9.55	10.25	10.40	10.05
Core Length, in.	10.84	10.84	10.51	11.28	11.44	11.06
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	7.6	2.0	2.0	7.0	1.8	1.9
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	32.7	27.2	26.7	32.9	28.0	27.5
Reactor OD, in.	19.6	19.6	19.2	20.0	20.2	19.8
Core Coolant Reynolds Number, average	11,200	11,000	11,300	10,400	10,100	10,400
Weight, lb	1213	1219	1140	1282	1323	1251

FIG 7

SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

UNCLASSIFIED

Case No.	19	20	21	22	23	24
Fuel	60 UC 40 W	60 UC 40 W	80 UC 20 W	80 UC 20 W	UO_2	UO_2
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10	10	10
Design Fuel Temp, F WHCF	2500	2500	2500	2500	2700	2700
Clad Material	PWC-11	T-222	PWC-11	T-222	PWC-11	T-222
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	4000	1500	4000	1500	4000
Fission Burnup, %(1)	1.34	1.38	1.68	1.61	1.26	1.02
Clad Thickness, in.	0.036	0.016	0.023	0.017	0.016	0.016
Maximum Power Density Kw/cc, w/o HCF	0.23	0.23	0.38	0.36	0.26	0.21
Pin Diameter, in.	0.837	0.804	0.535	0.540	0.268	0.294
Pin Spacing, in.	0.016	0.015	0.028	0.022	0.020	0.017
Number of Pins, approximate	168	173	272	276	1342	1317
Effective Core Diameter, in.	12.00	11.70	10.10	10.20	11.60	12.40
Core Length, in.	13.20	12.87	11.11	11.22	12.76	13.64
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	6.8	2.9	8.5	1.7	3.4	0.8
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	35.9	31.5	34.1	27.6	32.0	30.7
Reactor OD, in.	22.0	21.7	19.8	20.0	21.6	22.5
Core Coolant Reynolds Number, average	13,300	13,500	12,500	12,200	4900	4600
Weight, lb	1854	1757	1308	1314	1500	1787

 [redacted]

FIG 7

UNCLASSIFIED

SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	25	26	27	28	29	30[3]
Fuel	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	60 UO_2 40 W
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	25	25	25	25	25	10
Design Fuel Temp, F WHCF	2500	2700	2300	2500	2700	2700
Clad Material	PWC-11	PWC-11	T-222	T-222	T-222	PWC-11
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	1500	4000	4000	4000	1500
Fission Burnup, %[1]	1.41	1.44	1.12	1.24	1.33	0.81
Clad Thickness, in.	0.032	0.034	0.015	0.016	0.015	0.053
Maximum Power Density Kw/cc, w/o HCF	0.15	0.15	0.12	0.13	0.14	0.10
Pin Diameter, in.	0.633	0.710	0.541	0.643	0.722	1.329
Pin Spacing, in.	0.017	0.020	0.012	0.013	0.015	0.021
Number of Pins, approximate	375	298	580	380	284	105
Effective Core Diameter, in.	13.70	13.55	14.50	13.90	13.50	15.00
Core Length, in.	15.07	14.91	15.95	15.29	14.85	16.50
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	9.3	15.9	3.0	4.0	5.6	7.8
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	41.0	47.4	36.2	36.2	37.0	42.4
Reactor OD, in.	24.0	23.8	24.9	24.2	23.7	25.5
Core Coolant Reynolds Number, average	7800	9300	5900	7600	9100	13,400
Weight, lb	2006	1985	2202	2026	1918	2886

[redacted]

FIG 7

SNAP-50 2-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 5 PSI

Case No.	31(3)	32	33
Fuel	60 UO_2 40 W	80 UO_2 20 W	80 UO_2 20 W
Fuel Density, %	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10
Design Fuel Temp, F WHCF	2700	2700	2700
Clad Material	T-222	PWC-11	T-222
Design Clad Temp, F	2200	2200	2200
Design Clad Rupture Stress, psi	4000	1500	4000
Fission Burnup, %(1)	0.83	1.21	1.17
Clad Thickness, in.	0.020	0.030	0.016
Maximum Power Density Kw/cc, w/o HCF	0.10	0.20	0.20
Pin Diameter, in.	1.272	0.681	0.669
Pin Spacing, in.	0.026	0.023	0.022
Number of Pins, approximate	105	244	247
Effective Core Diameter, in.	14.40	11.95	11.84
Core Length, in.	15.84	13.15	12.98
End Reflector Total Length, in.	6.0	6.0	6.0
Gas Void Length, in.	5.4	7.3	2.3
Side Refl. TK., in.	4.0	4.0	4.0
Pressure Vessel Length, in.	38.9	36.3	31.0
Reactor OD, in.	24.8	22.0	21.8
Core Coolant Reynolds Number, average	14,000	11,100	11,200
Weight, lb	2645	1716	1659

FIG 8

SNAP-50 8-MW REACTORS PRESENT CAPABILITY

COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

UNCLASSIFIED

Case No.	1	2	3	4	5
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	90 UC 10 ZrC
Fuel Density, %	0.95	0.87	0.80	0.80	0.95
Gas Release (Helium plus Fission), %	10	100	100	100	10
Design Fuel Temp, F WHCF	2300	2500	2500	2500	2300
Clad Material	PWC-11	PWC-11	PWC-11	PWC-11	PWC-11
Design Clad Temp, F	2200	2200	2100	2200	2200
Design Clad Rupture Stress, psi	1500	1500	2070	1500	1500
Fission Burnup, %[1]	1.5[2]	4.04	4.04	3.87	1.5[2]
Clad Thickness, in.		0.030	0.025	0.025	
Maximum Power Density Kw/cc, w/o HCF		1.04	0.96	0.92	
Pin Diameter, in.		0.295	0.290	0.297	
Pin Spacing, in.		0.041	0.035	0.042	
Number of Pins, approximate		1440	1480	1425	
Effective Core Diameter, in.	>17[3]	14.00	13.75	14.00	>17[3]
Core Length, in.		15.40	15.13	15.40	
End Reflector Total Length, in.		6.0	6.0	6.0	
Gas Void Length, in.		64.3	44.1	73.2	
Side Refl. TK., in.		4.0	4.0	4.0	
Pressure Vessel Length, in.		97.6	76.9	106.6	
Reactor OD, in.		24.6	24.3	24.6	
Core Coolant Reynolds Number, average		15,600	15,600	15,700	
Weight, lb		3935	3183	4032	

(1) Atoms of U^{235} fissioned per original uranium atom, maximum.
(2) Case reached design burnup limit. Enrichment lowered.
(3) Exceeded Parametric Study Limits

FIG 8

SNAP-50 8-MW REACTORS PRESENT CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

UNCLASSIFIED

Case No.	6	7	8	9	10
Fuel	60 UC 40 W	UO_2	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO
Fuel Density, %	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	25	25	25
Design Fuel Temp, F WHCF	2300	2300	2300	2300	2300
Clad Material	PWC-11	PWC-11	PWC-11	PWC-11	PWC-11
Design Clad Temp, F	2200	2200	2200	2100	2100
Design Clad Rupture Stress, psi	1500	1500	1500	2070	2070
Fission Burnup, %(1)	4.35	3.10	4.48	3.0(2)	4.57
Clad Thickness, in.	0.016	0.015	0.016		0.016
Maximum Power Density Kw/cc, w/o HCF	0.73	0.65	0.47		0.48
Pin Diameter, in.	0.323	0.131	0.270		0.267
Pin Spacing, in.	0.020	0.012	0.020		0.018
Number of Pins, approximate	1292	9729	2411		2429
Effective Core Diameter, in.	13.50	15.80	15.65	> 17(3)	15.45
Core Length, in.	14.85	17.38	17.22		17.00
End Reflector Total Length, in.	6.0	6.0	6.0		6.0
Gas Void Length, in.	12.5	4.0	20.7		11.5
Side Refl. TK., in.	4.0	4.0	4.0		4.0
Pressure Vessel Length, in.	44.8	40.6	56.9		47.1
Reactor OD, in.	24.0	26.7	26.5		26.3
Core Coolant Reynolds Number, average	17,100	5,300	10,800		10,900
Weight, lb	2474	2902	3045		2714

FIG 8

SNAP-50 8-MW REACTORS PRESENT CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

Case No.	11	12	13
Fuel	50 UO_2 50 BeO	50 UO_2 50 BeO	60 UO_2 40 W
Fuel Density, %	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	25	25	10
Design Fuel Temp, F WHCF	2300	2300	2300
Clad Material	PWC-11	Cb Alloy	PWC-11
Design Clad Temp, F	2200	2200	2200
Design Clad Rupture Stress, psi	1500	750	1500
Fission Burnup, %[1]	3.0[2]	3.0[2]	2.86
Clad Thickness, in.			0.018
Maximum Power Density Kw/cc, w/o HCF			0.36
Pin Diameter, in.			0.440
Pin Spacing, in.			0.019
Number of Pins, approximate			1025
Effective Core Diameter, in.	> 17[3]	>17[3]	16.00
Core Length, in.			17.60
End Reflector Total Length, in.			6.0
Gas Void Length, in.			17.0
Side Refl. Tk., in.			4.0
Pressure Vessel Length, in.			53.6
Reactor OD, in.			26.9
Core Coolant Reynolds Number, average			16,100
Weight, lb			3602

FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

Case No.	1	2	3	4	5	6
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.95	0.95	0.95	0.95	0.87	0.87
Gas Release (Helium plus Fission), %	10	10	10	10	100	100
Design Fuel Temp, F WHCF	2500	2500	2500	2500	2700	2700
Clad Material	PWC-11	PWC-11	T-222	W-Re	PWC-11	PWC-11
Design Clad Temp, F	2200	2200	2200	2200	2100	2200
Design Clad Rupture Stress, psi	1500	1500	4000	4000	2070	1500
Fission Burnup, %(1)	4.88	1.5(2)	1.5(2)	1.5(2)	4.56	4.14
Clad Thickness, in.	0.026			0.016	0.026	0.037
Maximum Power Density Kw/cc, w/o HCF	1.37			0.42	1.18	1.07
Pin Diameter, in.	0.258			0.437	0.312	0.346
Pin Spacing, in.	0.026			0.013	0.045	0.044
Number of Pins, approximate	1581			1185	1104	1050
Effective Core Diameter, in.	12.40	> 17(3)	>17(3)	16.90	13.00	13.80
Core Length, in.	13.64			18.59	14.30	15.18
End Reflector Total Length, in.	6.0			6.0	6.0	6.0
Gas Void Length, in.	7.6			2.5	53.8	63.9
Side Refl. TK., in.	4.0			4.0	4.0	4.0
Pressure Vessel Length, in.	37.7			41.5	85.2	96.8
Reactor OD, in.	22.7			28.0	23.4	24.3
Core Coolant Reynolds Number, average	16,800			14,200	19,200	18,600
Weight, lb	1900			3985	3073	3833

(1) Atoms of U^{235} fissioned per original uranium atom, maximum.
(2) Case reached design burnup limit. Enrichment lowered.
(3) Exceeded Parametric Study Limits

FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

Case No.	7	8	9	10	11	12
Fuel	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN	UC/UN
Fuel Density, %	0.87	0.87	0.80	0.80	0.80	0.80
Gas Release (Helium plus Fission), %	100	100	100	100	100	100
Design Fuel Temp, F WHCF	2700	2700	2700	2700	2700	2700
Clad Material	T-222	W-Re	PWC-11	PWC-11	T-222	W-Re
Design Clad Temp, F	2200	2200	2100	2200	2200	2200
Design Clad Rupture Stress, psi	4000	4000	2070	1500	4000	4000
Fission Burnup, %(1)	3.92	4.20	4.17	3.90	3.89	4.14
Clad Thickness, in.	0.019	0.017	0.028	0.035	0.018	0.017
Maximum Power Density Kw/cc, w/o HCF	1.01	1.08	0.99	0.93	0.92	0.98
Pin Diameter, in.	0.324	0.300	0.333	0.356	0.328	0.307
Pin Spacing, in.	0.038	0.042	0.040	0.041	0.034	0.038
Number of Pins, approximate	1045	1071	1076	1040	1126	1153
Effective Core Diameter, in.	12.80	12.30	13.40	14.00	13.30	12.80
Core Length, in.	14.08	13.53	14.74	15.40	14.63	14.08
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	31.3	35.3	48.4	63.1	29.9	32.3
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	62.2	65.3	80.4	96.3	61.7	63.2
Reactor OD, in.	23.2	22.6	23.9	24.6	23.8	23.2
Core Coolant Reynolds Number, average	20,000	20,500	18,800	18,300	18,600	19,100
Weight, lb	2845	2709	3115	3856	2985	2818

 [redacted]

FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

Case No.	13	14	15	16	17	18
Fuel	UC/1W	UC/1W	UC/1W	90 UC 10 ZrC	90 UC 10 ZrC	90 UC 10 ZrC
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10	10	10
Design Fuel Temp, F WHCF	2500	2500	2500	2500	2500	2500
Clad Material	PWC-11	T-222	W-Re	PWC-11	T-222	W-Re
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	4000	4000	1500	4000	4000
Fission Burnup, %(1)	5.09	4.52	4.78	5.31	4.46	4.76
Clad Thickness, in.	0.016	0.015	0.016	0.016	0.016	0.016
Maximum Power Density Kw/cc, w/o HCF	1.42	1.26	1.33	1.35	1.13	1.21
Pin Diameter, in.	0.239	0.252	0.235	0.241	0.263	0.244
Pin Spacing, in.	0.031	0.025	0.027	0.028	0.023	0.024
Number of Pins, approximate	1547	1603	1641	1671	1653	1712
Effective Core Diameter, in.	11.70	12.20	11.70	12.10	12.75	12.20
Core Length, in.	12.87	13.42	12.87	13.31	14.13	13.42
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	12.5	3.3	3.4	11.6	3.0	3.1
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	41.6	33.3	32.5	41.4	33.9	33.0
Reactor OD, in.	21.9	22.5	21.9	22.4	23.1	22.5
Core Coolant Reynolds Number, average	18,000	17,000	17,500	16,800	16,000	16,400
Weight, lb	1768	1933	1796	1856	2094	1931

[redacted]

FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)

COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

UNCLASSIFIED

Case No.	19	20	21	22	23	24
Fuel	60 UC 40 W	60 UC 40 W	80 UC 20 W	80 UC 20 W	UO_2	UO_2
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10	10	10
Design Fuel Temp, F WHCF	2500	2500	2500	2500	2700	2700
Clad Material	PWC-11	T-222	PWC-11	T-222	PWC-11	T-222
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	4000	1500	4000	1500	4000
Fission Burnup, %(1)	4.52	4.13	5.06	4.51	3.77	2.66
Clad Thickness, in.	0.020	0.015	0.016	0.015	0.015	0.016
Maximum Power Density Kw/cc, w/o HCF	0.76	0.70	1.14	1.02	0.79	0.56
Pin Diameter, in.	0.415	0.435	0.288	0.307	0.164	0.191
Pin Spacing, in.	0.023	0.017	0.026	0.020	0.016	0.013
Number of Pins, approximate	762	747	1154	1140	4964	4615
Effective Core Diameter, in.	13.20	13.50	11.70	12.10	14.10	15.40
Core Length, in.	14.52	14.85	12.87	13.31	15.51	16.94
End Reflector Total Length, in.	6.0	6.0	6.0	6.0	6.0	6.0
Gas Void Length, in.	14.7	3.5	13.1	3.0	5.9	1.6
Side Refl. TK., in.	4.0	4.0	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	46.4	35.8	42.1	32.7	39.3	37.3
Reactor OD, in.	23.7	24.0	21.9	22.4	24.7	26.2
Core Coolant Reynolds Number, average	22,700	22,400	20,900	20,200	84,000	79,000
Weight, lb	2410	2457	1812	1913	2313	3022

FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

UNCLASSIFIED

Case No.	25	26	27	28	29	30
Fuel	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO	50 UO_2 50 BeO
Fuel Density, %	0.95	0.95	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	25	25	25	25	25	25
Design Fuel Temp, F WHCF	2700	2700	2700	2700	2300	2300
Clad Material	PWC-11	PWC-11	T-222	T-222	T-222	T-222
Design Clad Temp, F	2200	2200	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	1500	4000	4000	4000	4000
Fission Burnup, %[1]	4.73	3.0[2]	3.46	3.0[2]		3.0[2]
Clad Thickness, in.	0.023		0.015	0.017		
Maximum Power Density Kw/cc, w/o HCF	0.50		0.36	0.32		
Pin Diameter, in.	0.394		0.447	0.484		
Pin Spacing, in.	0.024		0.016	0.013		
Number of Pins, approximate	1100		1030	985		
Effective Core Diameter, in.	15.15	> 17[3]	16.20	17.00	> 17[3]	> 17[3]
Core Length, in.	16.67		17.82	18.70		
End Reflector Total Length, in.	6.0		6.0	6.0		
Gas Void Length, in.	22.9		7.4	6.5		
Side Refl. TK., in.	4.0		4.0	4.0		
Pressure Vessel Length, in.	58.0		44.4	45.1		
Reactor OD, in.	25.9		27.2	28.1		
Core Coolant Reynolds Number, average	16,400		15,900	15,600		
Weight, lb	2980		3067	3403		

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FIG 9

SNAP-50 8-MW REACTORS ADVANCED CAPABILITY

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(CONTINUED)
COOLANT OUTLET TEMPERATURE - 2000F
COOLANT TEMPERATURE RISE - 100F
SIDE REFLECTOR THICKNESS - 4 INCHES
COOLANT PRESSURE DROP - 30 PSI

Case No.	31	32	33	34
Fuel	60 UO_2 40 W	60 UO_2 40 W	80 UO_2 20 W	80 UO_2 20 W
Fuel Density, %	0.95	0.95	0.95	0.95
Gas Release (Helium plus Fission), %	10	10	10	10
Design Fuel Temp, F WHCF	2700	2700	2700	2700
Clad Material	PWC-11	T-222	PWC-11	T-222
Design Clad Temp, F	2200	2200	2200	2200
Design Clad Rupture Stress, psi	1500	4000	1500	4000
Fission Burnup, %(1)	2.93	2.82	4.00	3.44
Clad Thickness, in.	0.031	0.016	0.019	0.015
Maximum Power Density Kw/cc, w/o HCF	0.37	0.36	0.67	0.58
Pin Diameter, in.	0.666	0.661	0.361	0.384
Pin Spacing, in.	0.018	0.016	0.026	0.021
Number of Pins, approximate	451	444	1001	976
Effective Core Diameter, in.	15.80	15.50	13.40	13.80
Core Length, in.	17.38	17.05	14.74	15.18
End Reflector Total Length, in.	6.0	6.0	6.0	6.0
Gas Void Length, in.	14.0	5.5	13.5	3.5
Side Refl. TK., in.	4.0	4.0	4.0	4.0
Pressure Vessel Length, in.	50.3	41.2	45.4	36.1
Reactor OD, in.	26.7	26.4	23.9	24.4
Core Coolant Reynolds Number, average	24,600	25,400	19,500	19,200
Weight, lb	3470	3356	2296	2394

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The results of the study can be further summarized by ranking the fuels with respect to weight. The first and second candidates for 2-Mw and 8-Mw reactors limited by present capability considerations would be:

2-Mw			8-Mw
0.95 UC/UN	a.	(Cases 00 and 000 - Fig. 6)	60 UC/UN-40 W (Case 6 - Fig. 8)
90 UC/UN-10 m/o ZrC	b.	(Case 11 - Fig. 6)	UO_2, UO_2-BeO (Cases 7, 10 - Fig. 8)

The UO_2-fueled present capability reactor, a second choice candidate for the 8-Mw reactor based on weight considerations, requires approximately 9000 fuel pins, which is considered excessive from an assembly and fabrication viewpoint.

For the advanced technology or long range design and development reactors the candidates would be:

2-Mw			8-Mw
UC/UN (0.95); UC/UN-1 W	a.	(Cases 1, 2, 13 - Fig. 7)	UC/UN-1 W; 80 UC/UN-20 W (Cases 13, 21 - Fig. 9)
90 UC/UN-10 m/o ZrC; 80 UC/UN-20 W	b.	(Cases 16, 21 - Fig. 7)	90 UC/UN-10 m/o ZrC (Case 16, Fig. 9)

The fuels resulting in a minimum reactor weight are:

2 Mw	0.95 UC/UN, UC/UN-1 W
8 Mw	UC/UN-1 W, 80 UC/UN-20 W

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Comparing the above fuels with optimum UO_2 and UO_2-BeO reactors at the 2-Mw power level, the UO_2 reactors are approximately 300 to 800 pounds heavier and the UO_2-BeO 600 to 1000 pounds heavier than the minimum weight UC/UN reactor. At the 8-Mw power level, the UO_2 and UO_2-BeO reactor are respectively, $\gtrsim$500 pounds and $\geq$900 pounds heavier than the minimum weight UC/UN-1 W reactor.

The above fuel candidates, except for the UC/UN (95 percent dense), 90 UC/UN-10 m/o ZrC and UO_2-BeO cases indicated, are assumed not to be burnup-limited, i.e., fuel pin diametral growth is $\leq$1 percent at the design fuel temperatures and the fuel burnups indicated in Figs. 6 to 9. Experimental verification of these performance requirements must be accomplished to insure validity of these reactor designs.

The effect of a burnup limit on the potential of fuels is considerable, especially at the 8-Mw power level, as evidenced by comparison of specific cases in Figs. 6 to 9. Comparison of 8-Mw parametric designs for UC/UN (95 percent dense) (Fig. 9, Cases 1 and 2) and 90 UC/UN-10 m/o ZrC (Fig. 8, Case 5 and Fig. 9, Case 16) indicates that the weight of 1.5 percent burnup-limited designs are in excess of a factor of 2 greater than similar non-limited burnup cases. At the 2-Mw power level for UC/UN (95 percent dense), the 1.5 percent burnup limit results in a negligible weight penalty while a 1 percent burnup limit increases the reactor weight by approximately 35 percent as compared to the non-burnup-limited case (Fig. 6, Cases 1, 5, and 6).

All 2-Mw parametric cases were based on a 5 psi reactor pressure drop except for Cases 00 and 000 in Fig. 6 which have 10 psi pressure drops. The negligible weight difference between UC/UN-fueled Cases 000 and 1 (Fig. 6), similar except for coolant pressure drop, is felt to be primarily due to the 1.5 percent burnup limit constraint and is not considered indicative of weight trends for non-burnup limited design.

2. Cladding Material Versus Reactor Weight

Table 2 presents selected, approximate reactor weights for various fuel and cladding material combinations studied. The PWC-11 columbium alloy clad was assumed to have a 10,000-hour rupture strength of 1500 psi and the tantalum and tungsten-rhenium alloys assumed to have a strength of 4000 psi. It can be seen that there is not always an advantage from a weight standpoint in going to the advanced alloys and, in some cases, it is a distinct disadvantage.

In general, a change to the advanced alloys reduces reactor weight if the parametric design optimizes with a thick PWC-11 clad. Under this condition, the reduction in clad thickness, which can be achieved in going to the strong alloy, more than compensates for the negative reactivity effect caused by the increased neutron captures in these alloys and their increased density. At columbium cladding thicknesses of about 0.020 inch, the effects just compensate so that no weight change occurs. For columbium cladding thicknesses of about 0.015 inch, which is the lower limit of thickness allowed in the study, the reactivity effect predominates and higher weights result.

Table 2
Reactor Weights for Various Fuel-Clad Combinations
Advanced Capability

	2-Mw				8-Mw			
	Cb	Ta	W-Re	Cb to Ta Difference	Cb	Ta	W-Re	Cb to Ta Difference
UC/UN 0.95 density	[(1200)]	1200	1200	Same	>4200	>4200	4000	--
UC/UN 0.87 density	1900	1550	1500	-350	3800	2850	2700	-950
UO_2	1500	1800		+300	[(2300)]	3000		+700
50 UO_2-50 BeO	2000	1800		-200	[(3000)]	3400		+400
90 UC/UN-10 M/O ZrC	(1300)	1300	[1250]	Same	[1850]	2100	1900	+250
60 UC/UN-40 W	1850	1750		-100	[(2400)]	2450		+50
60 UO_2-40 W	3100	2600		-50	3500	3350		-150
80 UC/UN-20 W	[1300]	1300		Same	[1800]	1900		+100
80 UO_2-20 W	1700	1650		-50	2300	2400		+100
UC/UN-1 W	1200	1200	[1150]	Same	[1750]	1900	1800	+150

Optimum 2 and 8-Mw reactors, 1st choice ____

1st and 2nd choice if comparison is limited to present capability ( )

1st and 2nd choice if comparison is based on long range design and development [ ]

The clad cases of interest for the first and second choice fuels as determined in Section V-D-1, and the weight differences between PWC-11 and tantalum alloy clads are shown in Table 2. The data shows that for the fuels of interest, tantalum or tungsten-rhenium alloys result in little or no weight savings as compared to PWC-11 alloy.

The indicated weight difference between the tungsten-rhenium and tantalum alloy clads is due to the elimination of the 0.005 inch diffusion barrier required between the UC/UN fuel and the tantalum clad in the tungsten-rhenium clad cases.

[redacted]

3. Columbium Alloy Cladding Strength Versus Reactor Weight

The following trends are indicated from the limited columbium alloy cladding strength variation cases included in the study:

a. In general, improving the strength of the cladding material will reduce reactor weight appreciably until the clad thickness reaches its minimum thickness of 0.015 inch. The effect is greatest for high gas release systems which require the larger clad thicknesses.

b. For the optimum 2-Mw uranium carbide or nitride fuel with low gas release rates, an increase in cladding strength from 1500 to 2000 psi results in a weight reduction of approximately five percent. This relatively small difference is due to the common burnup limit. A decrease in clad strength from 1500 psi to 750 psi results in a weight increase of approximately 35 percent.

4. Maximum Fuel Centerline Temperatures Versus Reactor Weight

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Table 3 presents some selected results of reactor weights as a function of fuel maximum temperatures. For the range of temperatures and conditions examined in this study, the effect on reactor weight of changes in fuel temperature was, in general, relatively small (five percent or less for fuel temperature variations of 200F to 400F), except for UO_2 fuel. In general, the maximum fuel matrix temperature variations between 2300F to 2700F appear to have little effect on reactor weight for cores with fuel pins $\geq$0.250 inch in diameter. However, since centerline temperature affects primarily fuel pin diameter, the number of fuel pins can be cut approximately in half in going from 2300F to 2700F temperatures. UO_2-fueled reactor weights vary up to 15 percent over the range of 2300F to 2700F.

Table 3
Reactor Weights for Various Maximum Fuel Centerline Temperatures

	2-Mw				8-Mw		
Temperatures	2300	2500	2700	2900	2300	2500	2700
UC/UN 0.95 density	1250	1240	1220				
UC/UN 0.87 density		1890	1880	1850		3930	3830
UO_2	1730		1500		2900		2310
50 UO_2-50 BeO	2030	2000	2000		3040		2980
90 UC/UN-10 m/o ZrC	1350	1282					
60 UC/UN-40W	1882	1854			2474	2410	

The calculated maximum fuel centerline temperature depends strongly on the assumed fuel-cladding contact resistance, which was assumed constant for all fuels. If, because of failure to maintain intimate contact between the fuel matrix and the cladding, the contact resistance increased, the centerline temperature could be appreciably higher. This effect would be more pronounced for the cermet fuels of high fuel matrix conductivity because the temperature drop across the gap is a larger percentage of the total temperature drop across the pin. If the desired centerline temperature were held constant, increased contact resistance would result in increased reactor weight. The effect of a factor of five increase in the contact resistance is shown in Table 4 for fuels containing varying amounts of tungsten. The results indicate the gap resistance has little effect on parametric reactor weights at 2-Mw power levels. However, at 8 Mw, the reactor weight is increased appreciably with the increased gap resistance, but the weight trend between the fuels remains essentially unchanged.

Table 4

Fuel to Clad Gap Resistance	2-Mw Design Value	2-Mw 5X Design Value	8-Mw Design Value	8-Mw 5X Design Value
UC/UN-1 W	1213	1286	1768	2286
80 UC/UN-20 W	1308	1368	1812	2224
60 UC/UN-40 W	1854	1909	2410	2741

E. Effect of Core Flow Conditions on Reactor Weight

The reactor is to be designed for operation in a turbulent flow region to insure flow stability in all core coolant passages, thereby precluding the possibility of flow oscillations causing thermal cycling of fuel pin clad in local areas. A minimum local coolant Reynolds Number of 3000, based on design coolant flow, was selected as the lower design limit. This is considered adequate, since spiral wire spacers used on all fuel pins in the element (except the boundary pins) promote flow mixing throughout the element. Preliminary evaluation of several reactor cores in a seven-fuel can configuration indicates the minimum coolant Reynolds Number in the fuel element boundary passages is approximately 50 percent of the core average Reynolds Number. Thus, a core average coolant Reynolds Number of 6000 was considered the minimum acceptable value for evaluation of parametric cases.

One small pin UC/UN and several UO_2 2-Mw reactors (Cases 0, 13, 13a - Fig. 6 and cases 23, 24 - Fig. 7) do not meet this criteria and, therefore, are not considered desirable for the 2 Mw reactor. Two non-weight optimum UO_2-BeO cases also do not meet the criteria. The 8-Mw present capability UO_2 reactor (Case 7 - Fig. 8) also fails to meet the flow criteria. In general, the study indicates fuel pin diameters ≥ 0.250 inch for 10-inch diameter cores and ≥ 0.350 inch for 11 to 11.5-inch diameter cores are required to provide the 6000 core average Reynolds Number and turbulent flow.

F. Possible Differences Between Parametric Results and Final Design

The parametric studies are based on a consistent set of assumptions and, therefore, the results do represent a fair comparison. However, because of the simplifying physics and engineering assumptions it is necessary to make in the parametric studies, it is possible and indeed probable that the absolute value of reactor weights and sizes may be changed as a result of final design and analysis.

The engineered reactor weight can be influenced appreciably by the reactor mechanical design concepts and nuclear safety requirements which cannot be accurately estimated due to lack of detail design or definition at this time. The following are typical of such features: control method and drive, support structures including launch locks for which an estimated weight is used, and detailed core and pressure vessel design, i.e., removable fuel element, vessel closures, vessel head pipe reinforcements and possible nuclear safety devices (destruct mechanisms which are not considered in the parametric studies). The estimated control drive and support structure weight used in this study (Ref. Section III-A-3) accounts for approximately 20 percent of the weight of the optimum 2-Mw UC/UN reactor (Case 00,000 Fig. 6); thus the engineered reactor weight will be quite sensitive to the final control method and drive and support structure design. In addition, results of reactor engineering and nuclear analytical and experimental programs and fuels and material development programs may require revision of the parametric study assumptions and, consequently, the PWAR-20 reactor design.

Since the final PWAR-20 reactor design will be predicated on achieving an optimum between minimum weight, high reliability and the development program required to substantiate all phases of the design, further weight changes may be encountered.

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