~~CONFIDENTIAL~~ UNCLASSIFIED

Classification cancelled (or ~~changed to~~)

by authority of DOC

by ... T C, ... AUG 3 1 1973

SPECIAL REREVIEW FINAL DETERMINATION

Class.	Reviewers	Class.	Date
U	KAW	U	01/29/92
	RCD	U	06/29/92

TIM-870
STATUS OF MATERIALS DEVELOPMENT IN SUPPORT OF PWAR-20

AEC RESEARCH AND DEVELOPMENT REPORT

CONTRACT
AT(30-1)-2789

Issued
December 1, 1964

~~RICTED DATA~~

~~ontains restricted data as defined in the Act of 1954. Its transmittal or the disclosure of its contents in any manner to an unauthorized person is prohibited.~~

PRATT & WHITNEY AIRCRAFT

DIVISION OF UNITED AIRCRAFT CORPORATION

CANEL

MIDDLETOWN • CONNECTICUT

UNCLASSIFIED

~~CONFIDENTIAL~~

ISTRIBUTION OF [redacted]

DISCLAIMER

UNCLASSIFIED

DISTRIBUTION

	Copy No.
Mr. W. H. Pennington	1 - 3
Col. E. M. Douthett, HQ, AEC	4 - 7
Mr. T. M. Simmons, HQ, AEC	8 - 9
Mr. H. Potter, NY, Patent Office	10
Mr. F. Haines, MCR Project Office	11
Oak Ridge Technical Information Dept.	12-14
CANEL	15-60

TABLE OF CONTENTS

UNCLASSIFIED

TABLE OF FIGURES

UNCLASSIFIED

STATUS OF MATERIALS DEVELOPMENT IN SUPPORT OF PWAR-20

UNCLASSIFIED

Written By: M. A. DeCrescente
K. J. Kelly
P. V. Treciokas

Approved By: ____________________
H. Banach

____
L. M. Raring

CONFIDENTIAL RESTRICTED DATA	1		11/19/64
CLASSIFICATION	GROUP NO.	AUTHORIZED CLASSIFIER	DATE

INTRODUCTION

The present SNAP-50/SPUR powerplant development program concentrates on a system designed for unattended operation in space for 10,000 hours which produces a net electrical output of 300 Kwe. The reactor and shield for this powerplant has been designated as the PWAR-20. The purpose of this report is to summarize the status of materials development work conducted to date in support of the PWAR-20.

The preliminary design of the PWAR-20, including a summary of the major problem areas, is described in PWAC-445. In general, the design is based on conservative assumptions whenever this was consistent with the powerplant objectives rather than reliance on development breakthroughs which remain to be demonstrated. The preliminary design of the PWAR-20 provides a co-ordinated basis for continuing the development effort aimed at improvement and demonstration of materials capability for the PWAR-20.

UNCLASSIFIED

UNCLASSIFIED

A. Fuels

The PWAR-20 reactor fuel design requirements are:

Fuel Pin Diametral Swelling	≤ 2%
Fission Gas Release	≤20%
Fuel	UC (UN)

The present experimental status of uranium carbide and uranium nitride are summarized below:

Fuel	Demonstrated*	Future Potential
High Density UC	1.0 a/o U burnup with ≤ 20% FG release; ≤ 2% clad swelling attainable for 2200F clad, 2600F fuel center	1.5 a/o U burnup with ≤ 20% FG release; ≤ 2% clad swelling attainable if fuel center temp. reduced to 2300F.
High Density UN	1.4 a/o U burnup with ≤ 20% FG release; ≤ 2% clad swelling attainable for 2200F clad, 2600F fuel center	2.0 a/o U burnup with ≤ 20% FG release; ≤ 2% clad swelling attainable if fuel center temp. reduced to 2300F.

*Based on short-time tests

1. Uranium Carbide (UC)

Fourteen instrumented capsules, each containing three, ten percent-enriched, Cb-1 Zr-clad, sintered UC specimens immersed in lithium have been irradiated and examined. The specimen characteristics and range of test conditions are:

Cb-1 Zr Cladding Dimensions	0.296" OD; 25-mil wall
Barriers	2-mil Ta, 1-mil W (foil), none
Bond	1-mil cold He gas, ~10 psia
Fuel	92 to 97% theoretical density, 4.8 to 5.4%C
Clad Surface Temperature, F	1950 to 2225
Fuel Centerline Temperature, F	2350 to 2875
Test Time, Hrs.	670 to 3850
Power Density, kw/cc	1.1 to 1.9
Burnup, a/o U	0.3 to 1.6

The detailed results are summarized in Fig. 1. Fig. 2 shows the variation of swelling (as percent diametral clad increase) with fission burnup (as a/o U). The plot shows a general increase in swelling with burnup. The two percent swelling design limit has been exceeded at 1.4 a/o burnup under the test conditions. Some indications of the effect of lower centerline temperatures are apparent by comparing the tabulated irradiation fuel densities with density data obtained elsewhere.

Fig. 3 is a plot of fuel density decrease versus fission burnup, a/o U, showing CANEL high fuel centerline temperature data and the preliminary lower centerline temperature data furnished by the United Nuclear Corporation. The comparison shows that at a given burnup, significant improvements in fuel performance can be expected by lowering centerline temperatures below 2500F. It is expected that operating fuel centerline temperatures in the order of 2300F will allow fission burnup of ~1.5 a/o without exceeding two percent swelling. Testing to verify this performance is in progress.

RESULTS OF URANIUM CARBIDE IRRADIATION TESTS

10% ENRICHED, 0.296 INCH OD X 0.025 INCH WALL Cb-1 Zr ALLOY CLADDING

Capsule	Specimen	Barrier	Compos., % C	Pretest Density, % T.D.	Avg. Temp, F Cladding	Avg. Temp, F Fuel Ctr	Time, hr	Burnup, a/o U	Power Density, Kw/cc	Gas Release, % Xe	Swelling, % + Spec. Dia.	Swelling, % - Fuel Density
	348			94.1	2050	2475		1.15	1.17	2.8	1.15	5.7
PW26-150	349	2 mil Ta	5.3	94.1	2100	2600	2560	1.31	1.33	3.3	1.25	5.2
	350			94.2	2050	2575		1.39	1.41	3.6	1.35	6.7
	364			95.9	---	---		0.66	1.68	5.3	0.9	---
PW26-151	356	2 mil Ta	5.4	95.2	2175	2825	1020	0.68	1.75	7.2	1.55	---
	358			95.2	2125	2825		0.72	1.84	6.8	1.3	---
	360			95.3	2050	---		Specimen ruptured			---	---
PW26-152	353	2 mil Ta	5.4	96.1	2150	---	2860	Specimen ruptured			---	---
	365			95.0	2025	---		Specimen ruptured			---	---
	354			95.4	---	---		0.34	1.39	1.4	0.44	2.5
PW26-153	368	2 mil Ta	5.4	94.5	2200	2700	670	0.32	1.31	2.7	0.64	4.1
	357			95.5	---	---		0.31	1.30	0.8	0.30	1.7
	345			95.2	1975	---		Specimen ruptured			5.4	---
PW26-154	346	2 mil Ta	5.4	95.5	2175	---	3090	Specimen ruptured			8.9	---
	347			95.4	2050	2475		1.4 Cs	1.2	---	4.2	19.1
	323			95.8	2150	2725		1.45	1.58	9.3	2.7	10.1
PW26-160	327	2 mil Ta	5.4	95.6	2200	2750	2370	1.38	1.50	24.8	3.4	13.3
	329			96.0	2175	2750		1.46	1.58	12.6	2.7	11.2
	514			97.3	2100	---		Specimen ruptured			---	---
PW26-170	516	None	5.1	96.8	2175	---	2210	Specimen ruptured			---	---
	517			97.2	2150	---		Specimen ruptured			---	---
	500			97.1	1975	2700		0.59	1.91	0.1	0.5	2.0
PW26-171	506	None	5.1	97.2	2125	2800	890	0.56	1.79	0.2	0.84	2.4
	510			97.4	1850	2450		0.51	1.64	0.1	Nil	1.8
	417			96.4	2075	---		Specimen ruptured			---	---
PW26-173*	419	None	5.4	96.8	2225	2875	2280	1.45	1.76	13.1	1.55	14.1
	424			96.2	2000	2750		1.44	1.74	12.9	2.02	9.1
	418			96.8	---	---		1.63	1.73	4.4	0.66	7.4
PW26-174*	420	None	5.4	97.0	2200	2825	2600	1.58	1.66	75.9	7.28	21.6
	426			95.7	2175	2875		1.83	1.93	12.4	2.18	16.5
	454			91.8	2025	2575		1.69	1.49	67.7	0.5	13.6
PW26-181	455	1 mil W	4.5	91.5	2175	---	3120	Specimen ruptured			14.4	---
	456			92.2	2075	2575		1.61	1.42	77.2	12.1	---
	405			92.7	1950	2575		0.66	1.70	58.2	0.4	---
PW26-190	410	1 mil W	4.8	92.4	2050	2725	1060	0.70	1.83	73.4	0.3	---
	409			92.8	1975	2600		0.64	1.68	59.3	0.1	---
	412			93.7	2000	2450		0.86	1.20	37.2	Nil	---
PW26-191	402	1 mil W	4.8	92.3	2150	2675	1570	1.01	1.40	37.5	0.64	---
	411			93.5	2100	2575		0.91	1.27	29.0	0.64	---
	403			93.1	---	---		1.78	1.29	65.9	0.2	11.7
PW26-192	408	1 mil W	4.8	92.4	2200	---	3850	Specimen ruptured			---	---
	407			95.8	1975	---		Specimen leaked			9.6	14 6

*0.289" OD x 0.025' wall

UNCLASSIFIED

FIG 1

TIM - 870

SWELLING VERSUS BURNUP FOR URANIUM CARBIDE

(1850-2225F CLAD, 2450-2875F FUEL CENTER)

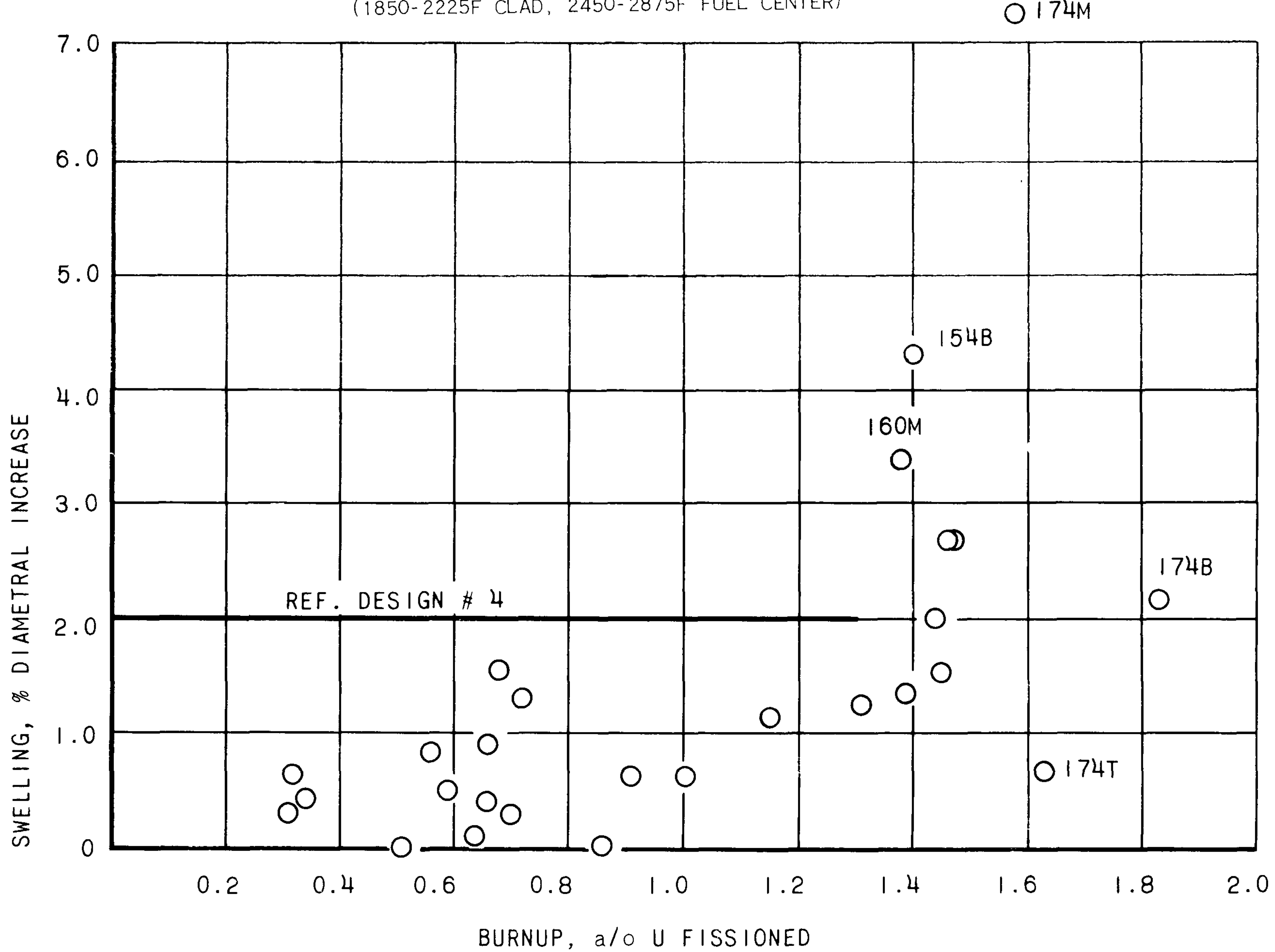


FIG 2

FUEL SWELLING VS BURNUP FOR UC & UN

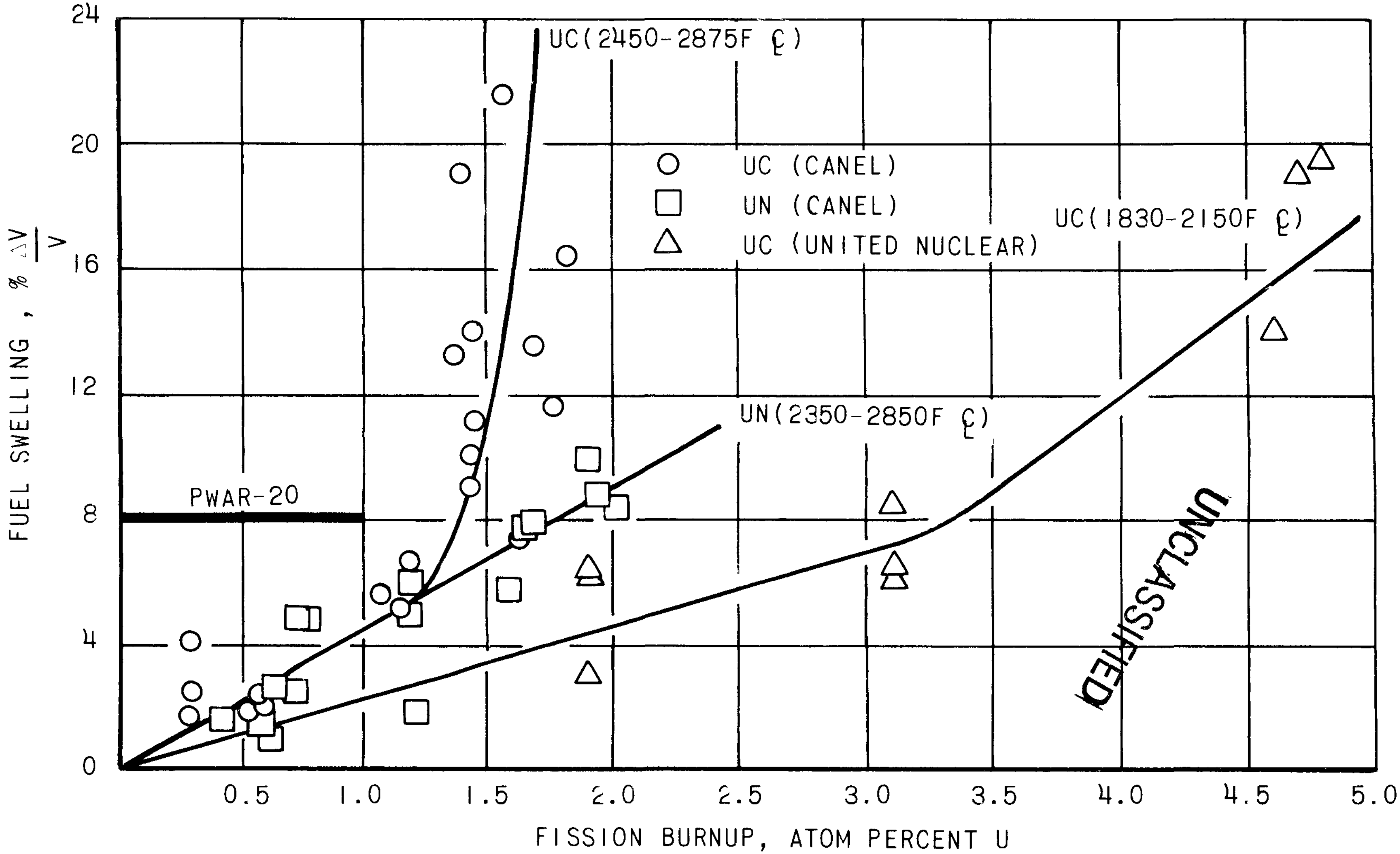


FIG 3

UNCLASSIFIED

The variation of fission gas release with burnup is illustrated in Fig. 4. The 20 percent release design value is exceeded at 1.4 a/o for UC fabricated to densities greater than 94 percent of theoretical under the conditions tested. Again, since it can be shown that fission gas release is related to the swelling, significantly lower releases are anticipated by lowering fuel centerline temperatures.

2. Uranium Nitride (UN)

Eight lithium-filled UN capsules have been irradiated and examined. The specimen characteristics and test conditions are:

Cb-1 Zr Cladding Dimensions	0.296, 0.309, 0.312" OD; 25 and 35-mil wall
Barriers	None and 1-mil W (foil)
Bond	1-mil cold He gas ~10 psia
Fuel	94 to 96% theoretical density
Clad Surface Fuel Temp, F	1875 to 2225
Fuel Centerline Temp, F	2350 to 2875
Test time, hrs	850 to 3000
Power Density, kw/cc	1 to 2.2
Burnup, a/o U	0.4 to 2.3

The detailed results of the UN capsule tests are summarized in Fig. 5. Fig. 6 shows the swelling results plotted against burnup. The largest swelling experienced with UN, two percent occurred at a burnup of 1.7 a/o U. These data would support the application of UN in a 2-Mw reactor where maximum full life burnup is less than 1.7% uranium burnup, maximum clad surface temperature is 2200F, and maximum fuel centerline temperature is below 2500F.

Fig. 7 would indicate that the fission gas release would be 25 percent for a design burnup of 1.5 a/o with fuel centerline temperatures above 2750F. However, test results (Fig. 5) for centerline temperatures below 2500F showed significantly lower gas releases.

An additional consideration which must be taken into account in the case of UN is the (n, α) and (n, p) reactions which occur with nitrogen in a fast flux. Preliminary calculations reveal that the ratio of helium atoms produced to gaseous fission product atoms is 0.15. The nature of the swelling mechanism would suggest, therefore, that in a fast flux, the swelling for a given burnup would be up to 15 percent higher than observed in thermal flux experiments. Hydrogen is produced at a rate of 0.03 atoms/fission. In a 1.5 a/o maximum burnup case, ~0.2 grams of hydrogen would be produced and, if distributed through the fuel clad pins in the PWAR-20 core, would represent an increase in hydrogen concentration of about 3 ppm.

3. Fuel Cladding Compatibility

The compatibility of uranium carbide and uranium nitride with fuel cladding has been evaluated over the following range of test conditions:

Fuel	$UC_{1.0}$, $UC_{1.1}$, UN
Cladding	Cb-1 Zr Alloy
Barriers	None
	Ta - 2 mils
	W-1, 5 mils

FISSION GAS RELEASE VERSUS BURNUP FOR URANIUM CARBIDE

(1850-2225F CLAD, 2450-2875F FUEL CENTER)

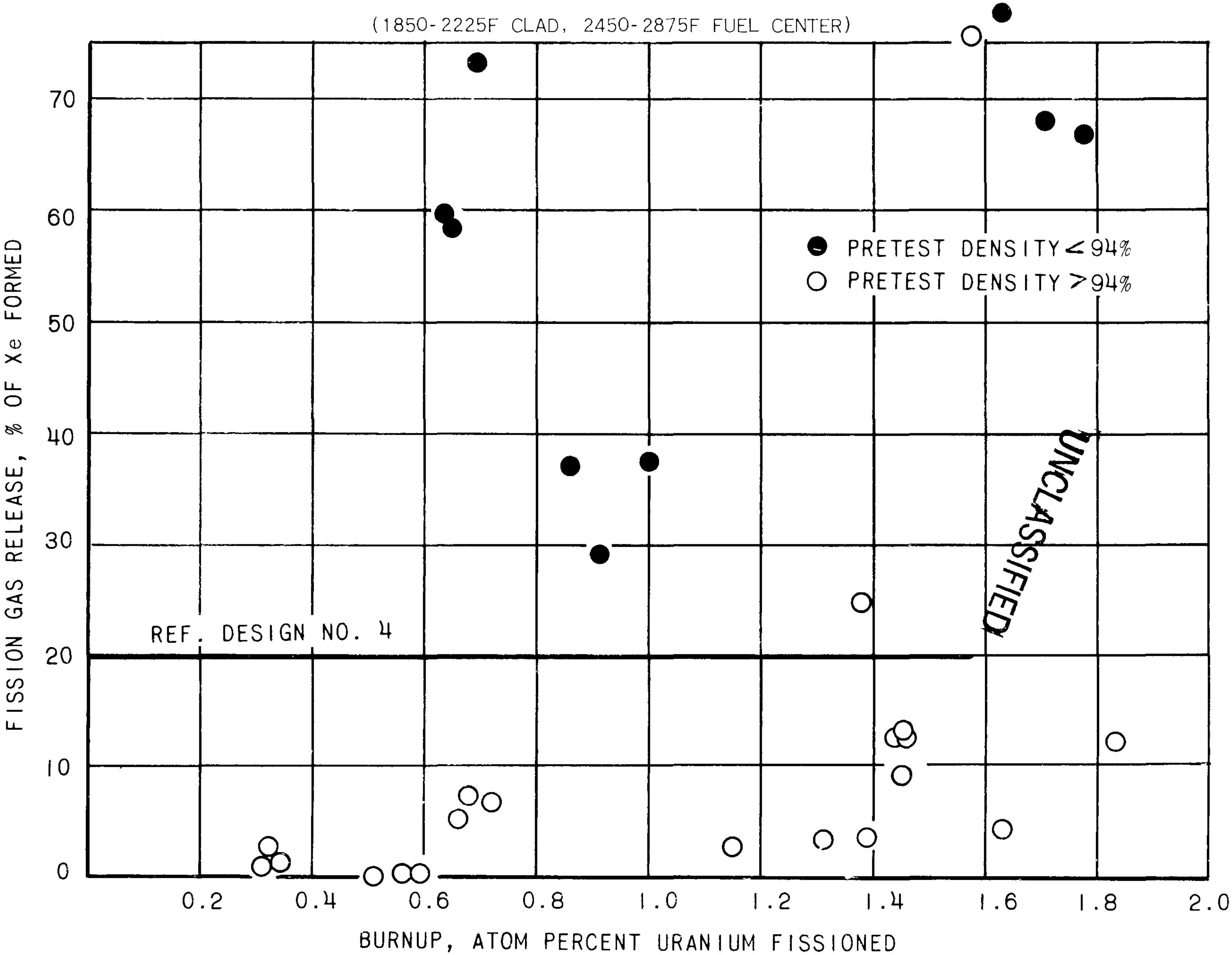


FIG 4

TIM - 810

RESULTS OF URANIUM NITRIDE IRRADIATION TESTS

35 MIL WALL Cb-1 Zr ALLOY CLADDING

UNCLASSIFIED

Capsule	Specimen	Cladding OD, in.	Barrier	Enrichment, %	Pretest Density % T.D.	Avg. Temp, F Cladding	Avg. Temp, F Fuel Ctr	Time, hr	Burnup, a/o U	Power Density, Kw/cc	Gas Release, % Xe	Swelling + Spec. Dia.	Swelling - Fuel Density
	42				90.4	1525	1675		0.06	0.44	0.08	Nil	---
PW26-200	43	0.314	None	93	91.0	1725	1850	390	0.06	0.41	0.09	Nil	---
	44				90.6	1550	1700		0.05	0.37	---	Nil	---
	39				90.3	1775	2075		0.54	0.93	---	Nil	---
PW26-201	40	0.314	None	93	90.8	1975	2300	1590	0.57	0.97	3.5	Nil	---
	41				90.3	1900	2300		0.72	1.23	5.1	Nil	---
	108				96.1	2025	2375		0.42	1.11	3.0	Nil	---
PW26-210	109	0.309	None	32	96.1	2125	2475	1090	0.42	1.12	3.3	Nil	1.6
	110				96.2	2125	2475		0.40	1.05	2.5	Nil	---
	118				94.1	1875	2350		1.20	1.54	4.8	0.47	5.0
PW26-220	119	0.309	None	12.3	94.3	2150	2625	1690	1.20	1.53	11.7	0.78	6.0
	121				94.0	1875	2400		1.21	1.55	4.2	0.37	1.9
	123				96.4	2075	2650		0.58	1.82	0.53	Nil	1.5
PW26-230	124	0.312	1 mil W	11.4	96.0	2225	2825	850	0.63	1.95	0.88	Nil	2.6
	125				95.7	2025	2625		0.61	1.90	0.68	Nil	0.9
	126				96.0	1975	2625		1.94	2.03	10.5	0.77	8.9
PW26-231	127	0.312	1 mil W	11.4	96.1	2225	2850	2750	1.91	2.00	30.8	1.54	10.0
	128				96.2	2050	2725		2.02	2.12	11.7	0.93	8.4
	132				95.0	2075	2600		1.59	1.59	17.5	1.2	5.8
PW26-240*	133	0.296	1 mil W	12.6	94.9	2200	2750	3000	1.69	1.73	33.6	2.0	8.0
	134				94.4	2125	2600		1.66	1.55	21.0	1.0	7.8
	135				94.9	2125	2775		0.78	2.06	5.8	0.20	4.9
PW26-241*	136	0.296	1 mil W	12.6	94.7	2225	2850	1100	0.74	2.00	3.9	0.37	2.5
	137				94.3	2100	2700		0.72	1.92	4.0	Nil	4.9

*25 mil wall cladding

FIG 5

SWELLING VERSUS BURNUP FOR URANIUM NITRIDE

(1875-2225F CLAD, 2350-2850F FUEL CENTER)

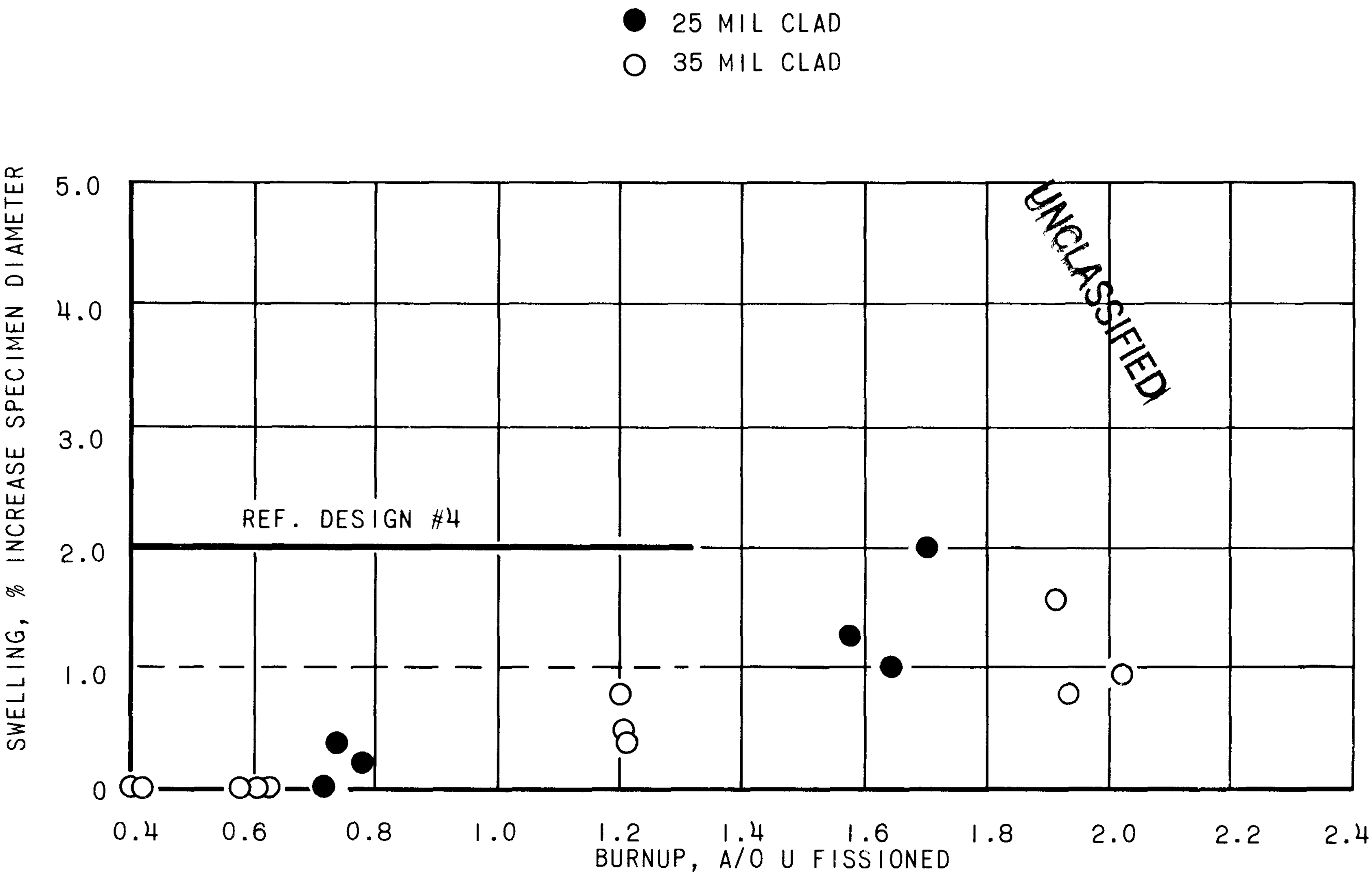
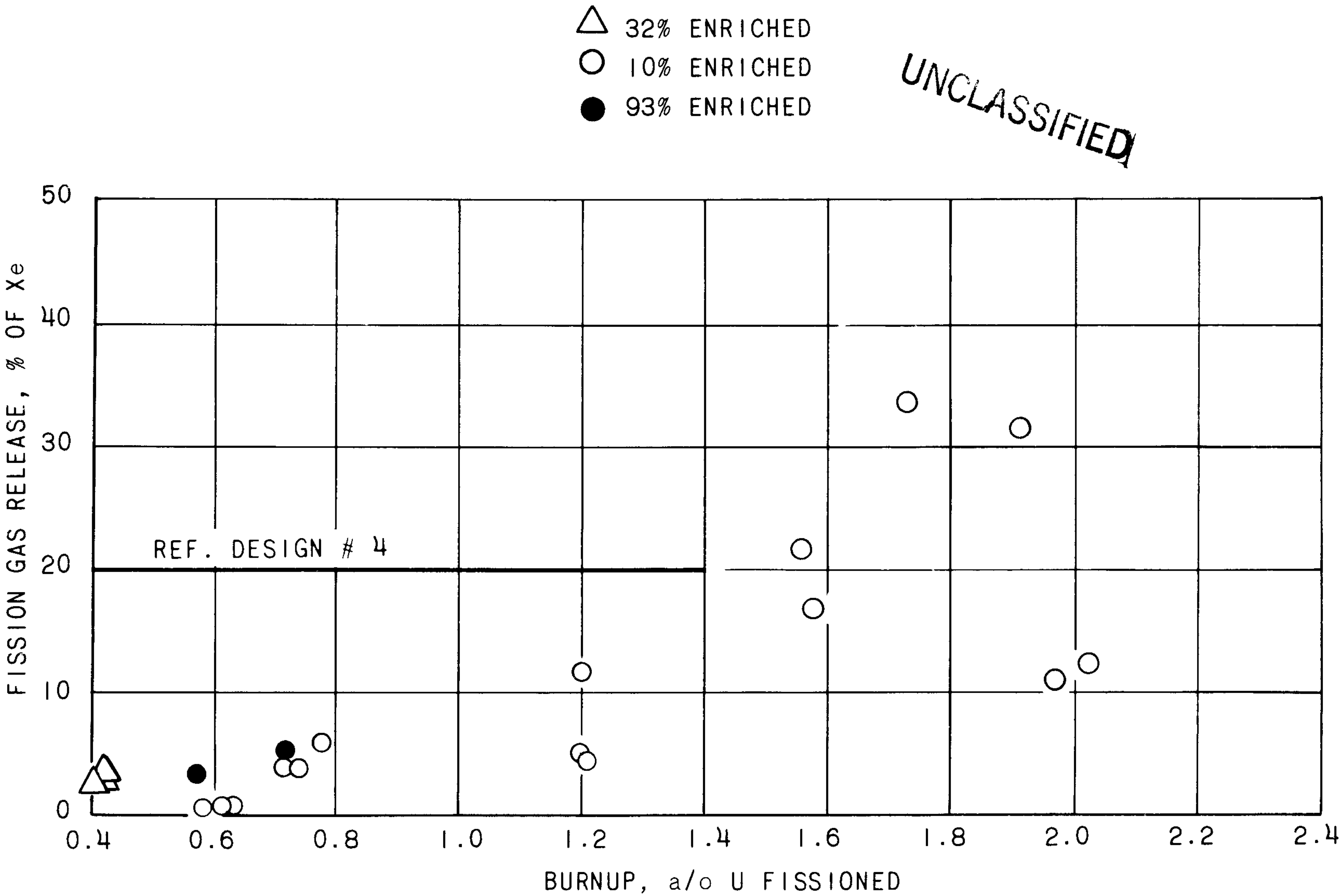


FIG 6

TIM - 870

FISSION GAS RELEASE VERSUS BURNUP FOR URANIUM NITRIDE

(1875-2225F CLAD, 2350-2850F FUEL CENTER)



UNCLASSIFIED

FIG 7

Temperature, F	2200
Max. Test Time, hours	3000 to 9000

Compatibility test results for hermetically sealed fuel pin specimens heated in a lithium environment are summarized in Fig. 8. At these test conditions, the interaction between UC and the Cb-1 Zr alloy cladding is significantly decreased by the use of a tantalum barrier and is essentially eliminated by the use of a tungsten barrier.

Although UN compatibility tests have reached only 3000 hours to date, results indicate that the extent of reaction between UN and cladding is appreciably less than that between UC and cladding in tests of similar duration. Tungsten proved to be compatible with UN and prevented interaction between UN and the cladding for the conditions tested. Testing to verify the performance of tungsten as a barrier between UC or UN and the fuel cladding is being continued for times beyond 10, 000 hours.

B. Structural Alloys

1. Fuel Element Cladding

Fuel clad design targets are as follows:

a. Rupture strength 10, 000 hours, 2200F - 1500 psi, minimum

b. Diametric growth at limit of secondary creep - 2 percent minimum

c. Diametric growth at long-time rupture - 8 percent minimum

d. Lithium compatibility - no deleterious effects in 10, 000 hours, 2200F

e. Productibility - High quality, uniform, fine bore tubing

Cb-1 Zr alloy with an extrapolated rupture strength of 700 psi at 2200F for 10, 000 hours falls short of the strength goal for fuel cladding. Of the columbium base alloys considered to offer improved strength, the Cb-1 Zr-0.06 C alloy was chosen to receive the principal effort, based on promising strength, fabricability, and compatibility data obtained from the Advanced Alloy Development and the LCRE programs. Preliminary data for biaxial stress rupture tests on tubing of typical fuel element cladding dimensions (0.250-inch OD x 0.025-inch wall and 0.312-inch OD x 0.013-inch wall) indicate an extrapolated rupture strength range of 1400 to 2000 psi for the 2200F, 10, 000-hour condition (Fig. 9). Although greater strength was obtainable by solution treatment followed by cold work, further effort in this direction was stopped because the fabrication process is not considered suitable for barrier-lined tubing. Of the total 34 rupture tests completed on solution treated and annealed tubing, seven have reached more than 4000 hours in 2200F lithium and four have reached more than 1000 hours in 2400F lithium. The goal of an eight percent minimum rupture ductility for the alloy appears attainable based on limited data for specimens solution heated at 2800F or 2700F plus a 2200F anneal.

Limited testing of D-43 alloy (Cb-10 W-1 Zr-0.1C) and T-111 alloy (Ta-8 W-2 Hf-0.01C) indicates that the strength and ductility goals are attainable with material in the as-received condition plus 2200F anneal.

UNCLASSIFIED

FUEL-CLADDING COMPATIBILITY AT 2200F[1] (SOUND PINS)

FUEL	BARRIER	MAX. TEST TIME, HR	EFFECT: FUEL	EFFECT: BARRIER	EFFECT: Cb-1 Zr CLAD
$UC_{1.1}$	NONE	9000	C LOSS	---	CARBURIZED
$UC_{1.1}$	Ta	9000	C LOSS	CARBURIZED	SL. CARBURIZED
$UC_{1.0}$	W	6000	NONE	NONE	NONE
UN	NONE	3000	NONE	---	NONE[2]
UN	W	3000	NONE	NONE	NONE

[1] Li ENCAPSULATED

[2] FREE U FORMED IN DIFFUSION CCUPLE TESTS

FIG 8

[redacted]

FIG 9

RUPTURE STRENGTH AND DUCTILITY OF FUEL ELEMENT CLADDING

UNCLASSIFIED

	Estimated Minimum 2200F/10,000 Hr. Rupture Strength psi	Range in Rupture Ductility, %	Number of Tests Completed
A. Cb-1 Zr-650 ppm C			
1. 0.304" OD/0.015" wall			
a. as-received + 2200F ann.	1200	11-25	9
b. sol. H.T. 2900F + swage + 2200F ann.	2100	3-9	52
c. swage + sol. H.T. 2900F + 2200F ann.	1500	3-9	16
2. 0.250" OD/0.025" wall			
a. as-received + 2200F ann.	1000	37-59	12
b. sol. H.T. 2875F + 2200F ann.	1400-2000	9-15	6
c. sol. H.T. 2875F + 2400F ann.	1400-2000	5-10	4
d. sol. H.T. 2875F + 2600F ann.	1400-2000	4-9	4
e. sol. H.T. 2800F + 2200F ann.	1400-2000	11-14	2
f. sol. H.T. 2700F + 2200F ann.	1400-2000	24-27	2
B. D-43 (Cb-10 W-1 Zr-.1 C)			
1. 0.304" OD/0.015" wall			
a. as-received + 2200F ann.	1700	26-28	4
b. sol. H.T. 2900F + swage	2300	6-10	20
C. T-111 (Ta-8-W-2 Hf-0.01 C)			
1. 0.312" OD/0.025" wall			
a. as received + 2200F ann.	4800	30-86	3

[redacted]

The objectives of the current mechanical properties evaluation program are to optimize heat treatment for strength and ductility and to determine rupture strength and creep behavior of Cb-1 Zr-0.06 C alloy at design conditions. Biaxial stress rupture and creep tests of fuel cladding with and without barriers are in test or are planned for conditions in the range of 1900 to 2600F and 1000 to 10,000 hours. These tests are expected to provide a more reliable basis for extrapolation of data by September, 1965, although the completion of the entire program is not expected before June, 1966.

For the compatibility of Cb-1 Zr-0.06 C with lithium to be acceptable, the alloy composition must be stable to the extent that strength is not reduced and lithium must be contained by the cladding without leakage at reactor clad maximum temperatures for 10,000 hours. In the ANP and LCRE programs, forced circulation loops of Cb-1 Zr alloy with carbon in the range of 200 to 600 ppm had been tested and showed no containment or transport effects related to the presence of carbon. Capsule tests indicated that alloys with 0.1 to 0.3 percent carbon were not significantly altered by lithium exposure. Although these preliminary results appear favorable, it is necessary to obtain more definitive test data concerning compatibility of Cb-1 Zr-0.06 C alloy with lithium at design temperature and flow conditions. The evaluation is being performed by capsule tests, stress rupture tests of tube specimens pre-soaked 1000 hours in 2200F lithium, and metallurgical and chemical analysis of forced circulation loops constructed with material combinations consistent with the reference design.

Preliminary results of lithium compatibility tests indicate that Cb-1 Zr-0.06 C alloy in the solution heated and 2200F annealed condition is capable of successfully containing lithium without penetration attack even when the alloy is contaminated with 1000 ppm oxygen. These results are consistent with the behavior of Cb-1 Zr alloy and suggest that carbon addition to the alloy does not adversely affect lithium containment. With regard to carbon stability in the alloy, initial results tabulated in Fig. 10. are inconclusive because of test difficulties and insufficient data. Plans for additional testing are summarized in 2-b following (Columbium Alloy section).

2. Columbium Alloy

The reactor structure is basically a thin-walled pressure vessel adapted internally to support the core and externally to accept the required liquid metal piping. The essential forms, therefore, are die forgings (heads), extrusions (cylinder), and rolled or forged flats (core support). These components are welded together after machining to form a hermetically sealed pressure tank in which flowing lithium is heated by the fueled core.

The material chosen for the reactor structure was a higher carbon version of the Cb-1 Zr alloy which was evaluated under the ANP and LCRE programs. The nominal carbon content of 150 ppm of the Cb-1 Zr alloy was increased to a nominal 800 ppm to meet future strength requirements of the alloy for advanced reactor development beyond PWAR-20. However, the strength and design criteria used for PWAR-20 were based conservatively on the lower carbon Cb-1 Zr alloy data.

No serious problems were anticipated in procuring mill products or in fabricating and assembling reactor components of the higher carbon alloy. Studies to date have shown that this increase in carbon content has not significantly affected workability and weldability. On the basis of limited test experience, the carbon increase provides a threefold increase in 10,000-hour rupture strength at 2000F compared to the original Cb-1 Zr alloy.

a. Mechanical Properties

In regard to strength criteria for the design of PWAR-20, the strength behavior of the Cb-1 Zr (lower carbon) alloy has been extensively investigated at CANEL over the past seven years.

LITHIUM COMPATIBILITY EVALUATION OF CARBON-MODIFIED Cb-1 Zr ALLOYS IN ALKALI METALS

Lithium Tests	Total Tested	Temp. Range, F	Time, hrs	Results
Oxygenated specimens (1150 ppm 0_2) in static capsules	2	2000	50	No attack when in annealed condition.
Tube specimens exposed in Cb-1 Zr forced circulation loop.	3	2000	500	No detectable change in chemistry or strength.
	3	2200	1000	Inconclusive, uncertain thermal history.
	3	2200	1000	Slight C and Zr depletion, no significant change of rupture strength.
Forced Circulation Loop Cb-1 Zr-0.06 C Hot Leg Cb-1 Zr Remainder	1	2200/1900	2175 (leaked)	Inconclusive, contaminated by external gas environment.

UNCLASSIFIED

FIG 10

TIM - 870

Laboratory strength studies in lithium on the Cb-1 Zr alloy to 2400F and for times to 10,000 hours are essentially completed. Over 290 uniaxial and biaxial creep-rupture tests were made on approximately 30 heats of material. The test data are tabulated in CNLM-4444, PWAC-631, PWAC-632, PWAC-633, PWAC-634, and PWAC-641. Minimum 10,000-hour rupture strength values derived from tests on various fabricated forms of material are listed below:

Cb-1 Zr Alloy, Annealed 2200F

	Minimum 10,000-Hour Rupture Strength		
Temp. F	Forgings and Extrusions (49 Tests)	Bar, Plate, Rod, Pipe (47 Tests)	Sheet and Tubing (83 Tests)
1600	11,000 psi	8200 psi	7800 psi
1800	5800	4400	3400
2000	3000	2300	1400
2200	1500	1300	700

In addition to the laboratory tests, engineering tests were completed on Cb-1 Zr alloy ten-inch and 14-inch diameter pressure vessels. The vessels,which contained lithium, were pressurized with helium and tested in static inert gas. The ten-inch diameter vessel which completed 4370 hours at 2000F under an effective stress of 2135 psi and 5630 hours at 1900F under an effective stress of 2475 psi exhibited a creep growth of three percent. The 14-inch diameter vessel which recently completed 10,000 hours at 2000F under an effective stress of 1400 psi is currently undergoing disassembly for post-test evaluation. Test details are presented in PWAC-398, Part II, and in TIM-789.

Fabrication of two Cb-1 Zr and three Cb-1 Zr-0.1 C ten-inch pressure vessels has begun. Uniaxial creep-rupture tests in lithium were started on improved columbium base alloys with the principal effort on the Cb-1 Zr-0.1 C alloy. These tests are intended to provide preliminary design data for use in the pressure vessel program. Test data are tabulated in PWAC-641 and PWAC-642.

In FY-1965 and FY-1966, uniaxial creep-rupture tests will be continued on the Cb-1 Zr-0.1 C alloy in lithium and in a vacuum of 10^{-8} torr. Also, preliminary strength curves will be established for Cb-10 W-1 Zr-0.1 C, Cb-18 W-8 Hf, Cb-27 Ta-10 W-1 Zr, and Ta-9.5 W-2.5 Hf-0.01 C alloys. In addition, engineering tests of two Cb-1 Zr and three Cb-1 Zr-0.1 C ten-inch diameter pressure vessels will be completed or in progress. All five tests will be conducted at 2000F for 3000 hours with the outer surfaces of the specimen subjected to a vacuum of 10^{-8} torr. The vessels will be filled with lithium and pressurized to produce a stress that will cause one percent or five percent diametral creep growth in the cylinder section. Details of the program are presented in TIM-843.

b. Coolant Compatibility

Extensive evaluation of Cb-1 Zr alloy compatibility with lithium in loop and pressure vessel tests (PWAC-343, PWAC-355, PWAC-381, PWAC-633, PWAC-634) has shown that the columbium-zirconium system is capable of containing 2000F lithium for 10,000 hours when oxygen contamination of the alloy is minimized. Investigations have established that zirconium in 2200F annealed alloy provided corrosion resistance to lithium unless the oxygen contamination exceeded an oxygen to zirconium atom ratio of approximately two (PWAC-343 and PWAC-1015).

Although the general containment behavior of carbon-modified columbium-zirconium alloys was expected to be similar to that of Cb-1 Zr alloy, a study was necessary to evaluate the effect of added carbon (from nominal 150 ppm C in Cb-1 Zr alloy to nominal 850 ppm C in PWC-11 alloy) on oxygen tolerance. This study was begun to determine the extent to which carbon interactions with zirconium altered the annealing behavior and the allowable O:Zr atom ratio. In addition, it was recognized that verification of the chemical stability of carbon-modified alloys under thermal and composition gradient conditions was required. Tests are being conducted using 1) laboratory capsules, 2) a forced circulation loop constructed of Cb-1 Zr alloy with facilities for exposure of specimens in 2000F-2200F flowing lithium, and 3) forced circulation corrosion loops with heater regions of carbon modified Cb-1 Zr alloy.

Results to date on a limited number of tests, summarized in Fig. 10 and detailed in PWAC-643 and PWAC-1015, suggest the following:

1) The oxygen threshold for lithium corrosion of annealed carbon-modified Cb-1 Zr alloy is above 1150 ppm; thus, carbon addition does not cause a catastrophic reduction in the oxygen tolerance determined for Cb-1 Zr alloy.

2) Surface depletion of carbon and zirconium did not result in significant change in rupture strength.

In FY-1965 and FY-1966, the oxygen tolerance limit for carbon modified Cb-1 Zr alloys will be determined over a range of carbon levels and conditions for annealing will be optimized. The effect of temperature and concentration gradients on carbon migration will be investigated by at least two thermal convection loop tests and by at least seven forced circulation corrosion loop tests as summarized below.

Lithium Compatibility Loop Program

UNCLASSIFIED

Type	Loop Material: Hot Leg	Loop Material: Remainder	Number Planned	Temp. F Max/Min	Scheduled Time, hrs
Thermal Convection	Cb-1 Zr-0.06 C	Cb-1 Zr-0.1 C	1	2000/1500	1000
			1	2000/1500	3000
Forced	Cb-1 Zr-0.6 C	Cb-1 Zr	1	2200/1900	3000
	Cb-1 Zr-0.06 C	Cb-1 Zr-0.1 C	2	2200/1900	3000
			1	2200/1900	10,000
	Cb-1 Zr-0.1 C	Cb-1 Zr-0.1 C	2	2200/1900	3000
			1	2200/1900	10,000

3. Titanium Alloy

A titanium alloy (Ti-8 Al-1 Mo-1 V) was selected for reactor and shield support members having preheat, launch and operational temperatures $\leq$ 1000F based on strength-weight considerations. The titanium alloy chosen is currently in pilot lot evaluation of various fabricated shapes by the aerospace industry for advanced structural requirements.

Long time creep-rupture strengths extrapolated from present data, which are limited to temperatures to 800F, show promise for this alloy for our application. A realistic creep-rupture program at current design temperatures (1000F max.) and allowable creep deformation (one percent in 1000 hours, max.) will be conducted to confirm present extrapolated data. The following strength properties are considered typical for this alloy.

0.2% Yield Strength at 1000F	=	60,000 psi
1% Creep/1000 hr./1000F	=	~ 5000 psi
1% Creep/10,000 hr./1000F	=	~ 2500 psi

UNCLASSIFIED

4. Coatings

The columbium alloy reflector support structure and other sections in the proximity of the reactor will require an oxidation protective coating to permit prelaunch heating of the SNAP-50/SPUR powerplant at 1000F in impure noble gas or air. The coating is to afford 1000-hour protection from oxidation and is not to adversely affect strength of structural members when subsequently exposed to design temperatures in space environment.

a. Oxidation protection of Cb-1 Zr alloy for 1000 hours in 1200F air has been obtained on laboratory specimens with 95.5 percent reliability using a tin-aluminum slurry coating PWK-35 (PWAC-1008).

b. Tests are in progress to evaluate oxidation protection of engineering-size specimens and to evaluate the effect of coating on creep strength of Cb-1 Zr alloy.

During FY-1965 and FY-1966, the stability of candidate high-emissivity coatings will be studied at design environmental conditions. Evaluation of the oxidation protection reliability of tin-aluminum coatings on engineering-size specimens will be completed. Procedures for coating full-size components will be verified.

C. Reflector and Shield

1. Reflector

Sintered BeO was chosen for prime consideration as the neutron reflector material, based on an assessment of nuclear properties, high temperature capability, and advanced state of development. A critical literature survey was conducted to collect and evaluate available data concerning: 1) physical and mechanical properties, 2) compatibility, 3) fabrication technology, and 4) irradiation effects. A summary of the status is as follows:

a. Physical and mechanical properties selected for use in parametric studies and in reflector design are shown in PWA-DIM-201.

b. High temperature compatibility of BeO with columbium-zirconium alloy container has been demonstrated through out-of-pile couple tests at 2000F, 2400F and 2800F for 200, 500, and 1000 hours (PWAC-634). Irradiation tests of LCRE-type BeO end reflectors clad in Cb-1 Zr alloy for 1000 hours at 2200F showed no adverse effect on compatibility (Ref. TIM-830, Supp. 1, 2).

c. The fabrication technology for sintered BeO in various sizes and configurations has been established and parts are manufactured in production quantities (Ref. GEMP-106A).

d. Irradiation stability tests of BeO indicate that the material is capable of retaining integrity for the SNAP-50/SPUR design conditions of 1×10^{21} nvt (>1 Mev) at temperatures above 2000F (GEMP-270A, ORNL 3523). Irradiation effects which result in anisotropic growth and cracking of BeO appear to be dependent upon temperature, time, and flux. Volume expansion of less than one percent has been observed for irradiation conditions of 1850F and 0.4×10^{21} nvt (> 1 Mev) at 1×10^{14} to 3×10^{14} n v (>1 Mev), (GEMP-270A). These data suggest that swelling of BeO at SNAP-50/SPUR reactions would not be a problem, although verification is needed at specific design conditions of temperature, time and flux.

The BeO reflector is designed to be radiatively cooled; therefore, stable emissivity characteristics are required.

a. Emissivity of unirradiated BeO has been measured, but verification of irradiation stability has not yet been established.

b. Improved BeO cooling through the development of a stable high temperature coating with better emissivity is desirable in order to reduce the maximum internal temperature of the reflector. Current efforts in this direction are minimal, but are planned to expand in FY-1966.

During FY-1966, an irradiation program will be initiated if information is insufficient to verify design extrapolation concerning BeO stability for 10,000-hour, 2300F to 2500F service.

2. Shield

Lithium hydride was selected for prime consideration as the neutron shield, since it offers the lightest shield capable of operating at temperatures up to 1000F. A critical literature survey was conducted covering: 1) mechanical, physical, and thermodynamic properties, 2) reactivity and compatibility, 3) manufacture and control, and 4) effects of irradiation. Physical and mechanical properties data selected for use in parametric studies and in shield design are shown in MDI-301. Irradiation stability did not appear to be a serious problem (CNLM-5757). High quality lithium hydride, outgassed and enclosed in evacuated or inert gas-filled stainless steel capsules, experienced negligible swelling, powdering, or gas evolution when irradiated at 10^{16} to 10^{19} nvt for 100 to 180 hours at 200 to 1000F. These test conditions approximate the temperatures and total doses expected for the PWAR-20 reactor application, but the test durations were no more than one-fiftieth of the required 10,000-hour life. In view of the importance of time in a diffusion-controlled mechanism for radiation damage, additional irradiation experiments are needed at specific design conditions of neutron flux in order to verify design assumptions (TIM-814).

Fabrication methods for casting honeycomb-reinforced lithium hydride have been established, and tests indicate that required density, hydrogen content and resistance to thermal cracking can be attained (APEX 915, NASA-SR-9400). This technology would be applied to the fabrication of the PWAR-20 reactor shield. A survey of potential fabrication facilities indicated that Atomics International expected to enlarge their casting equipment to sizes which would accommodate PWAR-20 shield requirements.

To improve confidence in the reactor design, additional information is required in the areas listed below. Work would be initiated by CANEL (TIM-814) only if not available from other Government-sponsored programs.

a. Hydrogen migration under imposed thermal gradients.

b. Hydrogen permeation through selected container materials.

c. Lithium hydride compatibility of improved strength oxidation resistant alloys.

d. Irradiation and vacuum stability of high-emissivity coating for the shield container.