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TFE VERIFICATION PROGRAM

SEMIANNUAL REPORT
FOR THE PERIOD ENDING APRIL 30, 1989

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Prepared under
Contract DE-AC03-86SF16298
for the San Francisco Operations Office
Department of Energy

DATE PUBLISHED: SEPTEMBER 1989

GENERAL ATOMICS

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1. INTRODUCTION

1.1 OBJECTIVE OF PROGRESS REPORT

The objective of the semiannual progress report is to summarize the technical results obtained during the latest reporting period. The information presented herein will include evaluated test data, design evaluations, the results of analyses and the significance of results.

1.2 TFE VERIFICATION PROGRAM GOAL

The program objective is to demonstrate the technology readiness of a TFE suitable for use as the basic element in a thermionic reactor with electric power output in the 0.5 to 5.0 MW(e) range, and a full-power life of 7 years.

1.3 TECHNICAL APPROACH

The TFE Verification Program builds directly on the technology and data base developed in the 1960s and early 1970s in an AEC/NASA program, and in the SP-100 program conducted in 1983, 1984 and 1985. In the SP-100 program, the attractive features of thermionic power conversion technology were recognized but concern was expressed over the lack of fast reactor irradiation data. The TFE Verification Program addresses this concern.

The general logic and strategy of the program to achieve its objectives is shown on Fig. 1-1. Five prior programs form the basis for the TFE Verification Program:

- 1) AEC/NASA program of the 1960s and early 1970s.
- 2) SP-100 concept development program.
- 3) SP-100 thermionic technology program.
- 4) Thermionic irradiations program in TRIGA in FY-86.
- 5) Thermionic Technology Program in 1986 and 1987.

These programs provide both the systems and technology expertise necessary to design and demonstrate a megawatt class TFE.

Figure 1-1

LOGIC TO DEMONSTRATE TECHNOLOGY READINESS OF MEGAWATT CLASS TFE

The approach to be followed is to design a TFE that will meet the reliability and lifetime requirements for the 2 MW(e) conceptual reactor design, and initiate component testing in a fast reactor. The demonstration of a 7-year component lifetime capability will be through the combined use of analytical models and accelerated, confirmatory tests. Iterative testing will be performed where the results of one test series will lead to evolutionary improvements in the next test specimens.

The TFE components will undergo screening and initial development testing in ex-reactor tests. Several design and materials options will be considered for each component. As screening tests permit, down selection will occur. It is necessary to rapidly make baseline design and materials selections to make optimum use of irradiation testing.

In parallel with ex-reactor testing, and fast reactor component testing, components will be integrated into a TFE and tested in the TRIGA. Realtime testing of partial length TFEs will be used to test support, alignment and interconnective TFE components, and to verify TFE performance in-reactor with integral cesium reservoirs. Realtime testing will also be used to verify the relation between TFE performance and fueled emitter swelling, to test the durability of intercell insulation, to check temperature distributions, and to verify the adequacy over time of the fission gas venting channels.

Predictions of TFE lifetime will rest primarily on the accelerated component testing results, as correlated and extended to realtime by the analytical models developed.

The fast reactor testing of fueled emitters will be calibrated by verifying the accuracy of emitter temperature predictions through complementary analysis and ex-reactor and in-reactor diagnostic tests. Instrumented sheath insulators will be tested in a fast reactor with an applied voltage.

A test of prototypic TFEs will be run in a fast reactor as a verification of the basic TFE design. This design may be upgraded based on the final component testing results.

The deliverables of the program are:

- 1) Conceptual design of a megawatt class power system including component specifications and a system description.
- 2) Thermionic components with verified performance.
- 3) TFE demonstration in a fast reactor.
- 4) Fabrication process specifications.
- 5) Verified performance models.

2. CONCEPT DESIGN TASK

2.1 OBJECTIVE

Task 2 provides the design guidance for the TFE Verification Program. The primary goals of this task are:

- 1) Establish the conceptual design of an in-core thermionic reactor for a 2 MW(e) space nuclear power system with a 7-year operating lifetime.
- 2) Demonstrate scalability of the above concept over the output power range of 500 KW(e) to 5 MW(e).
- 3) Define the TFE which is the basis for the 2 MW(e) reactor design. This TFE specification will then be the basis for the test program.

2.2 TECHNICAL APPROACH

The technical approach being taken in the concept design effort can be characterized as follows:

- 1) Perform system-level tradeoffs to determine initial TFE features and reactor scalability trends;
- 2) Refine these results and identify the 2 MW(e) reactor general arrangement with primary emphasis on characterizing the features and performance of the TFE;
- 3) Enter this information into the program data base via two separate design description documents:
  - o Two MW(e) reactor-converter system description.
  - o TFE component specification.

2.3 REACTIVITY AUDIT OF 2 MW(e) POINT DESIGN

Summary. During the last reporting period, Rasor Associates, Inc. performed an independent evaluation of the 2 MW(e) point design with respect to the following issues:

- 1) Available excess reactivity
- 2) Adequacy of the reactivity control system
- 3) Areas requiring attention when a more detailed design effort is undertaken.

It was found that:

- 1) Excess reactivity requirements could probably be met with the current core design or with minor modifications thereto.
- 2) The number and/or size of the control rods should be increased, with a minor impact on core size.
- 3) The current TFE design is valid.

Reactivity requirements. The required excess reactivity at beginning-of-life (BOL) was estimated to be:

Burnup:	.056 ΔK
Temperature defect:	.015
Bowing and swelling:	.005
Uncertainty:	<u>.020</u>
	.096 ΔK

The reactivity controlled by the drums and rods must cover this amount and assure a subcriticality of at least .01 ΔK with one control device in its most reactive state.

Analysis model. A two-dimensional neutron transport theory code in R-Z geometry was used with 14 neutron energy groups. Six radial and 12

axial fuel zones were assumed. The central rod was shown explicitly, but 6 surrounding rods were modeled as a rod ring about 70% of the way between the core center and core edge. Each rod displaces 21 TFEs; the 7 control rods occupy 5.6% of the core volume.

Radial fuel zoning was achieved by varying the central hole in the UO_2 pellets within the emitter.

The core length-to-diameter ratio was assumed to be the reference value of 0.78.

Reactivity results. Key calculational results of the audit are shown on Table 2-1. Conclusions are:

- 1) The BOL reactivity is a little less than desired but modest changes to the design should yield the required reactivity.
- 2) In a detailed design effort, the rods and drums would have to be divided into at least two independent systems. Also, an operational strategy would have to be devised that recognized the perturbation that partially inserted control elements have on TFE performance.
- 3) Overall rod worth appears nearly adequate. However, some additional reactivity could be required to assure adequate subcriticality with one rod stuck out. This could be achieved using one or both of the following schemes:
  - 1) Use several small rods rather than 7 large ones.
  - 2) Increase the core length-to-diameter ratio.
- 4) Changes to the TFE specification are not indicated by the audit.

Areas of conservatism in the RAI audit. The RAI audit appears to underestimate core reactivity and drum/rod worth in the following areas:

- 1) An optimized placement of the rods could increase their worth.

TABLE 2-1
REACTIVITY ESTIMATES: KEFF

<u>Condition</u>	<u>K_{eff} Zoned</u>
BOL, rods out, drums out	1.08
BOL, rods out, drums in	1.05
BOL, 7 rods in, drums out	.95
BOL, 7 rods in, drums in	.93
BOL, control rod in, 6 rods out, drums out	1.05
Control element worth:	
Central rod	.03 $\Delta K/K$
Seven rods	.13
Six rods	.10
Drums	.03
All rods, all drums	.15

- 2) A larger number of smaller rods would also be worth more, for the same rod volume.
- 3) Adjustments in the core length/diameter could increase both rod and drum worth.
- 4) The reactivity change with core depletion could be less than assumed. Reactivity decrements for temperature and bowing may also be less than assumed.
- 5) Control rod followers would increase rod worth.
- 6) A capability to withdraw rod poison farther than the core-reflector interface, as assumed in the audit, would increase rod worth.

Areas of optimism in the RAI audit. The RAI audit appears to over-estimate core reactivity and drum/rod worth in the following areas:

- 1) The leakage that results from the rod "ring" could be larger than from the 6 individual rods, resulting in an over-estimate of rod worth.
- 2) A TFE pitch-to-diameter ratio greater than the assumed 1.02 will lower core reactivity.
- 3) The use of niobium rather than tantalum in the intercell region is optimistic.

Areas for further study. The areas of conservatism and optimism should clearly be studied in future analyses when a more detailed core design is undertaken. In addition, the following areas should be examined:

- 1) The use of low enrichment uranium to flatten the power distribution and reduce the burnup reactivity decrement. Burnable poison should also be considered.
- 2) Critical assemblies should be analyzed to calibrate the nuclear design codes.
- 3) Cross section data should be developed in a rigorous manner for all nuclides of interest.
- 4) The use of "gray" rods should be explored to minimize the impact on thermionic performance of rods which are inserted.

3. CONVERTER PERFORMANCE TASK

3.1 OBJECTIVE

The objective of the converter performance task is to establish accurate converter performance models which have been correlated to observe test data. The data base will be developed from near prototypic converters using emitter and collector materials of interest over the full range of anticipated operating parameters. Part of the data base will include off-design and non-ideal operation of the converters. The resulting models will be used to determine the optimum converter configuration (materials, additives, spacing, etc.) for use in the prototypic TFE and reactor design.

A thermionic reactor is composed of a large array of thermionic cells, each of which has a unique input power, emitter temperature, and operating current density. In addition, individual cells will vary in performance over the system lifetime due to changing operating conditions caused by fuel burnup and variations in operating power requirement, or losses of some of the cells. Thus, the design and performance prediction of a thermionic reactor require an extensive data base on prototypical cell performance over a wide range of operating conditions.

The data and models will also be used in the startup of the TFEs in TRIGA and the FFTF and also in the startup of thermionic reactors. The observed current-voltage data during startup can be related to system temperatures through the ex-reactor correlations.

3.2 TASK DESCRIPTION

The test matrix was described in Ref. 3-1. The planned planar and cylindrical tests are shown on Tables 3-1 and 3-2, where the test rationale is also shown.

TABLE 3-1

EX-REACTOR PLANAR CONVERTER TEST PROGRAM

Emitter	Collector	Cs		Rationale for Test
		Reservoir		
PC-1	Duplex W	Nb	Graphite	Early performance map with graphite-cesium reservoir.
PC-2	Duplex W	Nb Tilted	Pool	A verification of the model which relates converter output to emitter distortion. Use PD-6 from the Thermionic Technology Program. (Ref. 3-2)
PC-3	High Strength Emitter	Nb	Pool	Establish data base on performance with high strength emitter
PC-1	Duplex W (MOD)	Nb	Graphite	UO ₂ in contact with the emitter. When compared to PC-1, differential data on the UO ₂ effect will be available

TABLE 3-2

EX-REACTOR CYLINDRICAL CONVERTER TEST PROGRAM

Emitter	Collector	Reservoir	Life		Rationale
			Test		
CC-1	Duplex W	Nb	Graphite	Yes	Performance map with reference electrodes and reservoir. Life test for long term stability with graphite-Cs reservoir.
CC-2 (CD-1)	Duplex W	MoO/Nb (Oxygenated)	Pool	Yes	Long term stability of oxygenated electrodes. Use CD-1 from Thermionic Technology Program. (Ref. 3-2)
CC-3	High strength emitter	Nb	Pool	No	Performance maps for a high strength emitter

3.3 CYLINDRICAL CONVERTER CC-1/CC-1X

The design and fabrication of CC-1X were described in Ref. 3-1. Preliminary experimental data were also presented there. A complete discussion of the data and related analysis is given below.

3.3.1 Experimental Methods

The CC-1X emitter was heated by electron bombardment and the collector was coupled through a gas gap to a water cooling block. The gas gap could be filled with helium at low pressure to vary the heat conduction. Collector temperatures above 970 K could only be reached, however, by eliminating the helium. Even then, at 7 A/cm^2 , the maximum achievable T_C was about 1025 K, slightly lower than the design set point.

Emitter temperatures at the four locations shown in Ref. 3-1 were measured by tantalum-sheathed W-5% Re/W-26% rhenium thermocouples. These were calibrated against a black-body hole where temperature was determined by optical pyrometry. The thermocouple readings were used to get an average emitter temperature, hereafter referred to as the emitter temperature.

The JV characteristics were obtained by a single 60 Hz AC voltage sweep, with the converter set at a given DC point. This DC point could be adjusted by varying the output voltage of a DC power supply, with a constant load resistance around 10 mohms. Figure 3-1(a) shows a schematic of the electrical circuitry.

CC-1X was tested in the flashlight current configuration which is appropriate for series connections of several converters. In Fig. 3-1(b) the configuration of the voltage probes and current leads is shown. For TFE applications the current is taken off the emitter lead at V_{EL} and enters the converter at the bottom at V_{CB} . The converter lead voltage (or net output voltage) is $V = V_{EL} - V_{CB}$. Two additional probes were attached to the collector top, near the insulator, at V_{CT} , and on the Ta sheath of the second emitter thermocouple, giving the emitter electrode voltage V_{EE} .

Figure 3-1(a) - Schematic electrical circuit for CC-1X testing

Figure 3-1(b) - Current configuration and voltage probes for CC-1X flashlight. Converter lead voltage $V = V_{EL} - V_{CB}$

3.3.2 Emitter Temperature Profiles

The four thermocouple wells (Ref. 3-1, Fig. 3-1) were designated #1 and #4 according to their depth relative to the emitter top. At an average emitter temperature around 1800 K, differences of the order of 100°C were observed between #1 and #2, 1.27 cm apart. To check their validity, alternate temperature measurements were made by using these holes as black-body holes whose temperature could be determined with an optical pyrometer. Thermocouple #4 (the deepest) was used as a common reference point. The results are shown in Fig. 3-2 for $J=7 \text{ A/cm}^2$, together with the calculated temperature profile of in-core capsule 1H1. The surprising fact is that although CC-1X has an emitter three times as thick as 1H1, the temperature profile is less uniform. However, pyrometer and thermocouple readings are in good agreement. A comparison of temperature profiles at $J=3 \text{ A/cm}^2$ also shows a good agreement between thermocouple and pyrometer. Thus, the high temperature gradient observed between #1 and #2 could be the result of a nonuniform heating resulting from an off-axis bombardment filament. However, since in mapping the performance of CC-1X, the average of the four thermocouple readings is used in defining the emitter temperature, an 80° error in the first thermocouple would affect T_E by 20 K. The level of confidence in T_E is further increased by the similar performance between CC-1X and PD-6 (Ref. 3-2), as explained later.

3.3.3 Cesium-Graphite Reservoir Temperature: Families of Performance Curves

Ignited JV characteristics were recorded for each of the 36 combinations of converter parameters - emitter temperature T_E , collector temperature T_C , cesium-graphite temperature T_R - chosen in the following table:

$$\begin{aligned}T_E &= 1600, 1700, 1800 \text{ K} \\T_C &= 800, 900, 1000 \text{ K} \\T_R &= 950, 975, 1000, 1025 \text{ K}.\end{aligned}$$

Figure 3-3 shows a T_R family for $T_E=1800 \text{ K}$ and $T_C=1000 \text{ K}$.

Figure 3-2 - Comparison of emitter temperature profiles observed for CC-1X, and calculated for 1H1

CC-1X IGNITED J-V CURVES

Figure 3-3 - Cesium-graphite reservoir temperature family for an average emitter temperature of 1800 K and a collector temperature of 1000 K

Unignited characteristics were also recorded for diagnostic purposes under the following conditions:

$$\begin{aligned}T_E &= 1650, 1700, 1750, 1800 \text{ K} \\T_C &\sim 920 \text{ K} \pm 30 \text{ K} \\T_R &= 950, 975, 1000, 1025, 1050 \text{ K}.\end{aligned}$$

These unignited characteristics were used to calibrate the emitter temperature correlation of TFE-1H1. Figure 3-4 shows typical unignited characteristics at a constant T_R when T_E is varied.

The envelope of a T_R family like the one in Fig. 3-4 determines, for a given set of emitter and collector temperatures, the optimum output voltage and T_R at each current density. It is useful to compare such optimized characteristics for CC-1X with the corresponding data obtained at the same temperatures and spacing on converters having similar electrode materials, such as PD-6 and CD-2 (Ref. 3-2). Since the only data available for PD-6 and CD-2 were obtained in the "cell-current" configuration (i.e. taking $V_{EL}-V_{CT}$), which was justified because both converters had very thick collectors, the comparison envelopes for CC-1X were also obtained in the cell configuration. Envelopes at $T_E=1800$ K and $T_C=1000$ K are shown for CC-1X and CD-2 in Fig. 3-5, and for CC-1X and PD-6 at $T_E=1800$ and 1600 K in Figs. 3-6 and 3-7. The agreement between the optimized characteristics of the three converters is quite good. In particular the closeness of the envelopes of CC-1X and PD-6 at two different emitter temperatures suggests that the average emitter temperature used to define T_E in CC-1X is as good a parameter as the temperature of a black-body hole drilled in the nearly isothermal emitter of PD-6.

3.3.4 Cesium Vapor Pressure Above the Graphite Reservoir

The performance mapping has been referenced to the cesiated graphite reservoir temperature T_R . It is worthwhile, however, to evaluate the cesium vapor pressure P_{Cs} in equilibrium with the graphite reservoir to compare with that predicted for the loading procedure followed during processing. Two methods, based on the comparison of JV characteristics, were used to estimate P_{Cs} as a function of T_R .

CC-1X (UNIGNITED)

Figure 3-4 - Emitter temperature family at constant collector and cesium-graphite temperatures

CC-1X & CD-2 ENVELOPE

Figure 3-5 - Comparison of cesium family envelopes for CC-1X and CD-2 for the temperatures indicated

CC-1X & PD-6 ENVELOPE

Figure 3-6 - Comparison of cesium family envelopes for CC-1X and PD-6 for the temperatures indicated

CONVERTER CC1X & PD6

Figure 3-7 - Comparison of cesium family envelopes for CC-1X and PD-6 for the temperatures indicated

CC-1X was tested at Thermo Electron Technologies Corporation (TTC) with a liquid cesium reservoir at T_{Cs} before the cesiation of the graphite. Thus, ignited JV characteristics obtained at TTC with the liquid and at RAI with graphite can be compared for the same emitter and collector temperatures to infer the desired correlation $P_{Cs}(T_R)$, from the known $P_{Cs}(T_{Cs})$ relationship. An example of such a comparison is shown in Figs. 3-8 and 3-9 where the current density and voltage at the knee of the characteristics are emphasized. In the other method, ignition current densities (i.e., the discontinuity in the characteristics in Fig. 3-4) are compared to calculations of the unignited saturation current densities. The results of these various evaluations of $P_{Cs}(T_R)$ are shown in Fig. 3-10 together with previous correlations obtained on POCO graphite for two cesium loadings, in grams of Cs per gram of graphite. The unignited method appears to correspond to a loading of .4 g/g, and even the ignited method gives a reasonable value (within a factor of 2 on P_{Cs}) around the graphite reservoir optimum point at 1025 K. The cesium loading inferred from these estimates appears adequate in the absence of cesium consumption.

3.3.5 Emitter Input Power Requirements and Converter Efficiency

The input power P_{in} was measured as a function of the converter current, at the two emitter temperatures of 1700 and 1800 K and a collector temperature of 1000 K.

The variation of the heat flux Q_{in} , at a given T_E , versus the current density J is shown in Fig. 3-11. Simple electron cooling theory predicts a linear correlation, which is indeed observed. Also shown in Fig. 3-11 are the corresponding data obtained in CD-2 (Ref. 3-2). The various slopes are quite similar for the two converters, but the heat flux at zero current, which is the sum of radiation and conduction losses from the emitter, is about 20% less for CC-1X. Since the radiation losses should be the same for both converters, the difference can be attributed to the shorter emitter sleeve of CD-2, which was not optimized.

Finally, at the bottom of Fig. 3-11, the measured lead efficiencies for CC-1X are plotted as a function of J . At 1800 K the maximum efficiency was

Figure 3-8 - Cesium family at given parameters: liquid reservoir

CC1X IGNITED J-V CURVE

Figure 3-9 - Cesium family at given parameters: graphite reservoir

Figure 3-10 - Cesium pressure above the cesium-graphite reservoir of CC-1X, evaluated from various methods

Figure 3-11 - Emitter input heat flux and lead efficiency of CC-1X as a function of the current density, at two emitter temperatures. Emitter heat flux data for CD-2 are also displayed for comparison.

obtained at 6 A/cm^2 , reaching 9.8% and decreasing slightly to 9.5% at 7 A/cm^2 . These efficiency measurements are the first ones dealing with a prototypic TFE converter at the design emitter temperature of 1800 K and operated in the flashlight current configuration, directly suitable for series connections.

3.3.6 Output Power Density: Comparison with CD-2

From the T_R family envelopes, the optimum output power density Q_{out} can be determined as a function of T_E and T_C at a given current density J . A comparison of Q_{out} with that of CD-2 is warranted, since CD-2 and CC-1X have the same electrode materials and interelectrode gap.

In Fig. 3-12, Q_{out} is plotted vs. T_E at constant collector temperatures for 7 A/cm^2 . In Fig. 3-13, at constant T_E , Q_{out} is plotted vs. the collector temperature. Although some differences between the two converters are noticeable at $T_E=1600 \text{ K}$, overall the performances are quite similar, in particular at the emitter design set point of 1800 K. The differences at lower T_E could be caused by different emitter temperature profiles, since the conduction heat loss along the emitter sleeve of CD-2 is larger.

It is worth mentioning that although the maximum efficiency for CC-1X was obtained around 6 A/cm^2 , more output power was available at 10 A/cm^2 .

3.4 PLANAR CONVERTER PC-2

The performance mapping of planar converter PC-2 provides data which can be used to validate a computer performance model of a fueled cylindrical converter with a distorted emitter. Measurements of emitter shapes after in-core operation in reactor life tests indicate fuel swelling induced emitter distortion reduces the interelectrode spacing. In general, the emitter acquires a barrel shape and the interelectrode spacing is reduced in the middle portion of emitter. This distortion can be approximated by dividing the emitter into several sections, determining the performance change for each section as the gap changes, and summing the results.

Figure 3-12 - Dependence of output power density on emitter temperature at given collector temperatures for CC-1X and CD-2 at $7 A/cm^2$

Figure 3-13 - Dependence of output power density on collector temperature, at given emitter temperature, for CC-1X and CD-2 at $7 A/cm^2$

A cost effective way to validate a model of the distorted emitter, in lieu of a very costly test of "bulging emitter" cylindrical converter, is to measure the performance of a planar converter with a tilted collector to simulate the distortion. JV (current-voltage) curves are measured for various spacings between the tilted collector and the emitter. Comparison with JV curves where the electrodes are parallel will give data on the change in performance when one electrode is not perfectly parallel with the second. The JV curve for a distorted cylindrical converter could then be constructed by the weighted superposition of the JV curves for a finite number of longitudinal sectors measured on a parallel planar converter at the prescribed emitter temperature (specified by the axial temperature profile) and spacing for each sector. The base data for comparing computed and measured tilted converter performance has been obtained on PC-2 and is reported in Ref. 3-1.

Figure 3-14 shows a cross-section of this converter. The collector usually stays parallel to the emitter for studying the influence of the spacing on the converter performance. The minimum spacing achievable in the parallel electrode configuration is actually determined by a shoulder built in the emitter flange structure, about $1.5 \pm .5$ mils. Tilting of the collector is achieved by adjusting the bellows.

3.4.1 Analytical Approach

To evaluate the JV characteristic of a converter with a varying spacing, one can decompose the whole converter into an ensemble of n elementary converters with a set of fixed spacings and assume that each component contributes a current $I_i = A_i J_i$, where A_i is the area and J_i is the current density at the particular spacing d_i . In this process, it is implicitly assumed that there is no interaction between the various adjacent converters and, therefore, no localization of the ignited plasma.

Figure 3-14 - Cross-sectional view of converter PC-2

This approach was tested on the tilted converter as follows:

- 1) JV curves for various spacings were recorded at $T_E=1800$ K, $T_C=1000$ K, and various cesium reservoir temperatures T_R on the parallel converter. Figure 3-15 is an example of such a spacing family.
- 2) The tilted converter was divided in five sections of equal width, as indicated in Fig. 3-16.
- 3) The average current density was calculated as
$$J = \sum_{i=1}^n A_i J_i$$
- 4) The computed J was compared to the current density at the average distance, $J(d)$ and to the experimental J_t obtained with a tilt $d_{\max} - d_{\min} = 10$ mils and $d = 11.5$ mils.

3.4.2 Results and Conclusions

Spacing families such as shown in Fig. 3-15 were originally obtained for integer values of d in mils, e.g., 7, 9, 11, etc... . A reevaluation of d resulted in values like 7.5, 9.5, etc. The comparison of J_t , J and $J(d)$ is shown in Figs. 3-17 and 3-18 as JV curves labeled 1, 2 and 3 respectively at two cesium temperatures 577 and 605 K; the agreement is excellent. Some uncertainty in the interpretation of the data reported in Ref. 3-1 has been resolved by this more careful analysis.

The conclusion is, therefore, that a tilted planar converter, which simulates the varying spacing induced by emitter distortion, behaves as a parallel planar converter with the average spacing. Furthermore, it has been shown that it is justified to compute the JV characteristics of a converter with a distorted emitter by decomposing this converter into an array of smaller converters with fixed spacings in parallel.

PC-2 PARALLEL CONVERTER DATA

Figure 3-15 - Spacing family for PC-2 in a parallel electrode configuration

Figure 3-16 - Geometry of parallel (top) and tilted (bottom) planar converter

PC-2 TILTED CONVERTER TEST

Figure 3-17 - Comparison of measured and calculated characteristics for PC-2 in a tilted electrode configuration, at the conditions indicated

PC-2 TILTED CONVERTER TEST

Figure 3-18 - Comparison of measured and calculated characteristics for PC-2 in a tilted electrode configuration, at the conditions indicated

3.5 PLANAR CONVERTER PC-1

Converter PC-1 is the baseline planar converter for the TFE Verification Program. It uses reference electrodes with a cesium-graphite reservoir.

Component fabrication for PC-1 is essentially complete. The emitter design includes a cavity for installing a UO_2 pellet, after the initial testing without a pellet is completed. The sides and back of the emitter are covered with a pressure bonded layer of $\text{Mo}(50\%)\text{Re}$ to aid in the subsequent sealing of the UO_2 loaded cavity. Difficulty was experienced in fabricating an emitter with a work function of about 4.9 eV. Process improvements were made and the emitter completed. Converter fabrication will be completed shortly.

3.6 THERMIONIC COMPUTER MODELING

An improved unignited mode planar converter model, MASTERPC, was developed during this period. As discussed in Section 8, discrepancies were found between measured IV curves from 1H1 and calculations from the UNIG model. MASTERPC eliminates most of the approximations used in UNIG and adds new physical phenomena which would be difficult to incorporate into UNIG.

MASTERPC uses pre-existing routines for solving a set of nonlinear differential equations and their associated boundary conditions. These equations describe the state of the interelectrode plasma. Routines for calculating the cesiated electrode work functions were taken over from UNIG with slight modifications to the collector work function calculation. Calculated IV curves were extensively compared with experimental data from PD-6 during development.

It was found that for emitter temperatures above 1700 K the formation of ions in the interelectrode space increases the apparent ignition current. This effect had been ignored in UNIG and would be difficult to

introduce into that model. The effect of ion formation is shown by the comparison in Fig 3-19. With the effect included, MASTERPC fits experimental data very accurately, as shown in Figs. 3-20 and 3-21. With the ionization included in MASTERPC, it is able to predict the ignition current and voltage for cases in which the ignition occurs before current saturation.

Examination of selected TFE-1H1 data indicates that the emitter temperature is about 40 K lower than previously indicated by UNIG. This temperature is consistent with estimates of TFE-1H1 performance at its operating point of 140 amperes.

References

- 3-1 TFE Verification Program Semiannual Report for the Period Ending September 30, 1988, GA-A19412.
- 3-2 Hatch, G. L., Thermionic Technology Program at Rasor Associates, Inc., DOE Contract DE-AC03-86SF15954, Final Report Number 533-003-B-053188, April, 1988.

Figure 3-19 - Effect of volume ionization on MASTERPC calculation of unignited mode I-V curve

Figure 3-20 - Comparison of MASTERPC calculation and experimental PD-6 I-V curve

Figure 3-21 - Comparison of MASTERPC calculation and experimental 1H1 I-V curve

4. INSULATOR SEAL TASK

4.1 OBJECTIVE

The overall objective of the insulator seal task is to develop and validate the performance of an insulator seal for use in the thermionic fuel element reference design. In particular, the objectives are:

1. Produce designs for the insulator seal.
2. Develop required fabrication processes for the insulator seals designed and document the process specifications.
3. Fabricate insulator seals for ex-reactor and in-reactor testing.
4. Verify the performance characteristics and lifetimes associated with insulator seals by means of ex-reactor and in-reactor testing. Perform postirradiation examination of theseal specimens and use the test results to improve the seal design.
5. Develop an analytical model of the performance and lifetime of the insulator seal and validate the model with test data.

The insulator seal must provide electrical isolation between adjacent collectors while maintaining a leak-tight seal separating the cesium vapor of the interelectrode space from the fission products. The current design requirements are listed below:

1. Insulator temperature	Maximum 1150° K
2. Nominal fast fluence, E>0.1 Mev	2.3×10^{22} n/cm ²
3. Maximum applied voltage	0.63 volts
4. Operating environment	1 torr Cs, fission gases
5. Leak tightness	$\leq 1 \times 10^{-8}$ standard cc/s
6. Electrical resistance	>10 ohms (1150° K)
7. Lifetime	≥ 7 years.

4.2 TASK DESCRIPTION

The insulator seal task consists of five subtasks:

1. Seal design. Seal designs consistent with the TFE requirements are to be developed. Designs have included the taper seal, the butt seal and trilayer seal.
2. Fabrication development. Select appropriate insulator materials. Develop appropriate bonding techniques for each insulator material selected. Fabricate sufficient insulator seal specimens to support the ex-reactor and in-reactor test effort. Prepare and issue process specifications. Insulator materials considered are Al_2O_3 , Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$.
3. Ex-reactor testing. Perform ex-reactor tests to evaluate the effects of thermal cycling, cesium compatibility, applied voltage and material interdiffusion on insulator seal performance and lifetime.
4. In-reactor testing. Perform in-reactor tests and the related postirradiation examinations to determine seal mechanical stability, electrical resistance and hermeticity after exposure to fast neutron fluences.
5. Modeling. Develop and validate analytical models to predict seal lifetime and performance.

4.3 RESULTS-TO-DATE

The seal design has been down selected to include only the taper and trilayer design. While the design of the trilayer seal is obviously different from that of the sheath insulator, the trilayer technology is identical for the two insulators. Trilayer seal fabrication development will be included in the sheath insulator task.

The insulator materials currently being developed as seals are single crystal (SC) and polycrystalline (PC) Al_2O_3 , polycrystalline (PC) Y_2O_3 and SC and PC $\text{Y}_3\text{Al}_5\text{O}_{12}$ (YAG). Initial PIE results of single crystal Al_2O_3 taper seal test samples in UCA-1 indicate good neutron damage resistance; additional testing is planned. Polycrystalline Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$ exhibit considerably less swelling than polycrystalline Al_2O_3 , and are more likely to remain hermetic, and will therefore be developed as a backup to single crystal Al_2O_3 .

4.3.1 Fabrication Development

Polycrystalline Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$ (YAG) slugs have been fabricated by TTC. The slugs are made from commercial powders. The sample preparation is the same for both powder types. The powder is mixed with water and several drops of Darvan C, a defloculant, and ball milled overnight. The slurry is then filter pressed to remove the water and allowed to dry overnight. The dry compact is cold pressed at 40 kpsi and fired at 1673°K for one hour. The Y_2O_3 is HIP'ed at 1883°K at 10 kpsi for three hours. The YAG is sintered to 97% theoretical density by heating it for two hours at 1973°K . The finished slugs are then machined to the desired shape.

A polycrystalline YAG sample in the taper configuration was successfully metalized and brazed to niobium. Metalization was done at 2073°K using a W-2% Y_2O_3 metalizing layer and the brazing was done at 2103°K using a V-Nb alloy. To eliminate erosion of the metalized layer by the braze alloy, a CVD coating of W was needed. The sample has gone through a final machining step and has successfully remained leak tight. The sample is being prepared for peel testing and for metallographical examination of the braze-ceramic interface region.

Development of metalization and brazing of polycrystalline Y_2O_3 to niobium is in the initial stages of development.

4.3.2 Ex-reactor Testing

Electrical testing was performed on as-received polycrystalline Y_2O_3

from TTC. The specimen consisted of a solid cylinder of Y_2O_3 , 0.728 inches in length and 0.72 inches in diameter, which has a niobium sleeve pressure bonded to the outer surface. A circumferential groove was machined in the niobium sleeve 0.040 inches wide to simulate a taper seal insulator. The sample resistance after five hours at $1090^{\circ}K$ was 5.4×10^4 ohms with an applied voltage of 0.6 volts. After a one week accelerated test at $1250^{\circ}K$ and 10V, the resistance dropped to 1×10^4 ohms from an initial value of 1.28×10^4 ohms. The resistance value was considered favorable and testing was stopped after one week. Further testing of polycrystalline Y_2O_3 will be done on samples with the taper seal design.

The ex-reactor test stand is currently being redesigned to permit exposure of the test sample to cesium vapor during electrical testing.

4.3.3 In-Reactor Testing

UCA-1. Irradiation of UCA-1 specimens in cycle 9 of FFTF was completed on October 10, 1987. The temperature of the five MOTA canisters was maintained at about $1100^{\circ}K$ during irradiation. The insulators were irradiated for approximately 341.1 effective full power days (EFPD) and received estimated peak fast fluences of 6.5 to 7.2×10^{22} n/cm² for level 4 canister (4B) and 2.5 to 4.1×10^{22} n/cm² for level 5 canisters (5:A,B,C,D). The variation of fluence is due to axial flux gradients over the 5-inch length of the MOTA canisters. The test matrix is shown in Table 4-1.

Table 4-2 shows the results of the initial visual inspection of UCA-1 seal insulators. The Al_2O_3 taper specimens appeared excellent with no detectable cracking. Some cracking was observed in the butt and trilayer seals. Sample B-4 had cracked completely. It is not known whether the cracking occurred during removal of the shorting wire or during irradiation. No further development of the butt seal is planned at this time.

Measurements of the electrical resistance at high temperatures for the seal insulators have been completed and are also summarized in Table 4-2. Measurements were taken during the rise to $1173^{\circ}K$ and during the decrease

TABLE 4-1
SEAL INSULATOR UCA-1 TEST MATRIX

ID Number	Material	Design	Canister	Irradiation Temperature	Fluence 10^{22} n/cm ²
L-1	Al ₂ O ₃ (SC)	Taper ³	5D	1127°K	3.9
L-2	Al ₂ O ₃ (SC)	Taper ³	5C	1110°K	3.1
L-3	Al ₂ O ₃ (SC)	Taper ³	4B	1160°K	7.2
B-1	Al ₂ O ₃ (SC)	Butt ¹	5A	1094°K	3.0
B-2	Al ₂ O ₃ (SC)	Butt ¹	5A	1114°K	4.1
B-4	Al ₂ O ₃ (SC)	Butt ¹	4B	1170°K	7.1
T-1	Y ₂ O ₃ (PC)	Trilayer ²	5B	1113°K	3.0
T-2	Y ₂ O ₃ (PC)	Trilayer ²	5D	1113°K	3.0
T-3	Y ₂ O ₃ (PC)	Trilayer ²	4B	1170°K	7.2

¹ Gas pressure bonded.

² HIP, 7 graded layers.

³ Metalized and brazed.

TABLE 4-2
PIE OF UCA-1 SEAL INSULATORS

ID Number	Material	Seal	Condition	Resistance (ohms)	
				300°K	1173°K
L-1	Al ₂ O ₃	Taper	No cracks	$>10^{10}$	3.7×10^7
L-2	Al ₂ O ₃	Taper	No cracks	$>10^{10}$	5.2×10^7
L-3	Al ₂ O ₃	Taper	No cracks	$>10^{10}$	9.9×10^7
B-1	Al ₂ O ₃	Butt	2 cracks//to axis	2.9×10^5	1.6×10^5
B-2	Al ₂ O ₃	Butt	3 cracks//to axis	$>10^{10}$	2.1×10^6
B-4	Al ₂ O ₃	Butt	Cracked		
T-1	Y ₂ O ₃	Trilayer	Circumferential	$>10^{10}$	182
T-2	Y ₂ O ₃	Trilayer	Hairline cracks	6.1×10^4	130
T-3	Y ₂ O ₃	Trilayer	Circumferential	9.0×10^{10}	54

back to room temperature. The measurements were done under a positive pressure of high purity He (99.999%) and with an applied potential of one volt. The system and procedure were modified by WHC to reduce the observed discrepancies in resistance measurements during heatup and during cooldown which was attributed to contamination of the sample ends by the measuring system. Before electrical testing, the specimen and resistance probe were baked out for 1.5 hours at 1223°K in a vacuum of better than 1.1×10^{-4} torr.

The single crystal samples had resistances of 10^5 ohms or greater at 1173°K. No correlation was observed between the resistance and irradiation fluence at 1173°K. The Y_2O_3 trilayers showed resistances of 200 ohms or less at 1173°K, well below requirements.

Examination of the UCA-1 insulator seals is continuing with helium leak testing and metallography. From these tests the following will be determined:

- 1) Extent of the ceramic cracking.
- 2) Distortion of the ends.
- 3) Bonding of the ceramic to the niobium.
- 4) Changes in microstructure of ceramic and niobium from the high neutron fluence.

These will help in determining why the Y_2O_3 trilayer seals have such a low resistance at high temperatures.

UCA-2. Irradiation testing of UCA-2 specimens in the FFTF was completed in cycle 10C on January 8, 1989. The temperature in the three MOTA canisters was maintained at 1073°K during irradiation. The insulators were irradiated for approximately 335 EFPD and received estimated peak fast fluences between 3.6 and 4.6×10^{22} n/cm² for level 1 canisters (1:A,B) and between 5.7 and 6.5×10^{22} n/cm² for level 4 canister (4C).

The MOTA is in the process of being disassembled and the canisters removed. The test matrix for the seal insulators in UCA-2 is shown in Table 4-3.

4.4 STATUS SUMMARY

A summary of the insulator seal development effort is shown on Table 4-4. The developmental priority at the moment is as follows:

Reference: Al_2O_3 single crystal taper.

Backups: Al_2O_3 polycrystalline taper
 Y_2O_3 polycrystalline taper
 $\text{Y}_3\text{Al}_5\text{O}_{12}$ (YAG) polycrystalline taper.

The polycrystalline Al_2O_3 taper seal has experienced over 30,000 real time test hours in the TRIGA reactor, and there is evidence that its lifetime is at least several years. Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$ are also being carried as backups since further testing may show that single crystal Al_2O_3 does not have the desired lifetime.

TABLE 4-3
UCA-2 SEAL INSULATOR TEST MATRIX

ID Number	Material	Design	Canister	Irradiation Temperature	Fluence 10^{22} n/cm^2
T-1	Y_2O_3 (PC)	Trilayer ¹	1A	1073°K	4.3
T-2	Y_2O_3 (PC)	Trilayer ¹	1A	1073°K	4.3
L-1	Al_2O_3 (SC)	Taper ²	1B	1073°K	4.3
Y-1	YAG(SC)	Taper ²	4C	1073°K	6.7

¹ HIP, 7 graded layers.

² Metalized and brazed.

TABLE 4-4

INSULATOR SEAL TECHNOLOGY STATUS

Material	Design	Priority	Electrical Properties	Neutron Stability	Fabricability	Test Matrix (No of Specimens)				
						UCA-1	UCA-2	TFE 1H1/1H2	Fueled Emitter	Ex-Reactor
Y ₂ O ₃	Taper P C	High	TBD	Good	TBD	3	2	1		(Task 5)
Y ₂ O ₃	Trilayer	High	Poor*	Good	Good					
YAG	Taper P C	High	TBD	Good	Good		1			Task 5
YAG	Taper S C	Low	TBD	Good	Marginal					
YAG	Trilayer	Medium	Good**	Good	Difficult***					
Al ₂ O ₃	Taper P C	Low	Good	Acceptable	Good	3	2	9		1
Al ₂ O ₃	Taper S C	High	Good	Acceptable	Good		1			
Al ₂ O ₃	Butt S C	Low	TBD	Acceptable	Marginal	3				Task 5
Al ₂ O ₃	Trilayer	Medium	Good**	Acceptable	Difficult***					

P C Polycrystalline

S C Single Crystal

* Due to impurities in starting material, Task 5 results

** Task 5 results

*** Not crack free

5. SHEATH INSULATOR TASK

5.1 OBJECTIVE AND TECHNICAL APPROACH

The overall objective of the sheath insulator task plan is to develop and validate the performance of a sheath insulator for use in the thermionic fuel element reference design. In particular, the objectives are:

1. Produce designs for the sheath insulator.
2. Develop required fabrication processes for the sheath insulators designed and document the process specifications.
3. Fabricate sheath insulators for ex-reactor and in-reactor testing (uninstrumented and instrumented testing).
4. Verify the performance characteristics and lifetimes associated with sheath insulators by means of ex-reactor and in-reactor testing.
5. Develop an analytical model of the performance and lifetime of the sheath insulator and validate the model with test data.

The sheath insulator must provide electrical isolation between the collector of each converter on the outer sheath tube which is in contact with the reactor liquid metal coolant. The sheath insulator must also provide good thermal conductance between the collector and the sheath tube. The current design requirements are listed below:

1. Insulator temperature	Maximum 1150°K
2. Nominal fast fluence, E>0.1 MeV	2.3×10^{22} n/cm ²
3. Maximum applied voltage	7.5 volts
4. Operating environment	1 torr Cs, fission gases
5. Electrical resistance	>1000 ohms (1150°K)
6. Thermal conductivity	>0.03 W/cm-°K)
7. Lifetime	≥7 years

The technical concerns are the unbending of the sheath insulator structure due to fast neutron induced damage, and electrolysis which could lead to low electrical resistance, electrical breakdown or poor bonding. The objective of the sheath insulator task is to resolve these concerns and to develop and validate the performance of a sheath insulator for use in a prototypical TFE.

5.2 TASK DESCRIPTION

The sheath insulator task consists of five subtasks:

1. Sheath design. Sheath designs consistent with the TFE requirements are to be developed. The primary design is the trilayer, consisting of an insulating layer between two niobium layers.
2. Fabrication development. Select appropriate insulator materials. Develop appropriate fabrication techniques for each insulator material. Fabricate sufficient sheath insulator specimens to support the ex-reactor and in-reactor test efforts. Prepare and issue process specifications. Insulator materials considered are Al_2O_3 , Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$. Fabrication techniques include plasma spraying (graded), and slip casting (cermet).
3. Ex-reactor testing. Perform ex-reactor tests to evaluate the effects of thermal cycling, applied voltage and material inter-diffusion on sheath insulator performance and lifetime.
4. In-reactor testing. Perform uninstrumented and instrumented in-reactor tests and the related postirradiation examinations to determine sheath mechanical stability, electrical resistance and hermeticity after exposure to fast neutron fluences and to both neutrons and an applied voltage.
5. Modeling. Develop and validate analytical models to predict sheath lifetime and performance.

5.3 RESULTS-TO-DATE

Two manufacturing processes and three insulator materials are currently being evaluated. The manufacturing processes are the plasma sprayed graded trilayer being investigated by GA and the slip cast (cermet) trilayer being investigated by TTC. The materials of choice are Al_2O_3 , Y_2O_3 and $\text{Y}_3\text{Al}_5\text{O}_{12}$ (YAG). The development will include fabricating samples for both ex-reactor and in-reactor testing and for H series TFEs.

5.3.1 Plasma Sprayed Graded Trilayer Fabrication Development

Postbonding metallographic examination of Al_2O_3 and YAG trilayers has frequently shown the presence of cracks. The cracks have been attributed to the thermal expansion mismatch between the niobium and ceramic. Elastic-creep analyses of typical Al_2O_3 trilayers have been performed to determine how to minimize or eliminate the cracking in the ceramic during the HIP process. The results indicate that by minimizing the thickness of the niobium mandrel and by improving the pressure/temperature cycle during the HIP process, tensile stresses in the ceramic can be reduced to levels where cracking should be minimized. A new design for a thin inner mandrel and outer sleeve were generated and drawings released for fabrication. Three thin mandrel designs with Y_2O_3 , YAG/ Al_2O_3 blend and Al_2O_3 were plasma sprayed and pressure bonded using the standard HIP pressure/temperature cycle and compared to standard mandrel design. Metallographical examination of the different sheath insulators showed a marked improvement with the thin mandrel design. There was a reduction in the number of cracks in the Al_2O_3 from approximately an average of thirty down to only two with the thin mandrel design.

The Y_2O_3 showed no cracks in both the standard and thin mandrel design. With the YAG/ Al_2O_3 blend, the improvement was not as dramatic as with Al_2O_3 : from 15 cracks with the standard mandrel to 8 cracks with the thin mandrel design.

Using recent thermal expansion data on the different materials, the elasto-plastic analysis is being reviewed and upgraded. A new HIP cycle

will be used in the next bonding run.

A combination of further refinements in the mandrel design, improved starting powders and an optimized HIP cycle should minimize cracks in the ceramic resulting from fabrication.

The matrix for the next batch of graded sheath insulators to be fabricated is shown in Table 5-1. The fabrication variables are ceramic material, starting powders, and HIP cycle history. A thin mandrel/outer sheath tube will be used throughout and the insulator will consist of only one graded layer of 50% niobium-50% ceramic. Using only one graded layer gives the same results as when using the standard three layer design (75%, 50%, 25%-Nb). An added advantage of the one layer design is that the thickness of the 100% ceramic layer is increased.

TABLE 5-1
SHEATH INSULATOR TEST MATRIX

Material	Sol-Gel		HIP Cycle	
	NH ₃ OH	N ₂ H ₄ CO	Standard	Modified
Al ₂ O ₃	2	2	2	2
Y ₂ O ₃	2	2	2	2
Y ₃ Al ₅ O ₁₂	2	2	2	2

Carbon contamination has been considered to be responsible for the low electrical resistance at high temperature of some of the Y₂O₃ sheath insulators during ex-reactor testing and possibly the UCA-1 Y₂O₃ test samples. The carbon is present in the as-received starting powders used for plasma spraying. A series of tests were performed to resolve the carbon contamination in the Y₂O₃ powder. The tests consisted of baking the starting powder in flowing oxygen at different temperatures for different times. The results are summarized in Table 5-2. As a result of this work powders used for plasma spraying would be calcined at 1300°K in flowing oxygen for 18 hours. This step should reduce the carbon content of the powder to less than 100 ppms.

TABLE 5-2

CARBON CONTENT OF MOLYCORP Y_2O_3

Sample Number	Temperature ($^{\circ}K$)	Time (Hours)	Carbon Content (ppm)
1	-	-	1696
4	-	-	1468
7	-	-	1341
2	1100	42	477
3	1100	66.5	266
5	1300	17.5	83
6	1300	63	66
7	1500	15.3	75
8	1500	41	34

It was also decided to use starting materials derived from Sol-gel processes in use at GA. Ceramic powders derived in this way have a high purity, a smaller and more uniform particle size and a denser micro-structure.

5.3.2 Slip Cast Cermet Trilayer Fabrication Development

The cermet sheath insulator development effort is continuing at TTC. Results from the Thermionic Technology Program (Ref. 5-1), showed that metal addition increases the strength of the ceramic insulator. Addition of metal particles to the ceramic has the disadvantage of exposing the ceramic to high electrical fields which can result in faster degradation or shorting of the ceramic. However analysis shows that if the metal addition is kept at or below 10% the degradation risk can be minimized.

Al_2O_3 and Y_2O_3 specimens with niobium powder or niobium spheres have been manufactured. These samples are in the process of being mechanically tested to determine strength as a function of metal shape. The strength will be determined from a ball-on-the-ring biaxial technique. Preliminary

strength testing was done on pure YAG and Y_2O_3 . The average strength was 32 kpsi and 22 kpsi, respectively. These results are in the range of values reported for these materials.

Alumina and Y_2O_3 cermet with 10% niobium powder have been manufactured. The Y_2O_3 samples have been electrically tested for about three weeks. The resistivity of these samples is between 10^6 to 10^7 ohm-cm at operating conditions.

The use of ceramic-ceramic composites for added strength as an alternate sheath insulator design is being investigated at TTC as part of the Thermionic Technology Program. Samples made by additions of Cr_2O_3 or Si_3N_4 to Al_2O_3 and $Y_3Al_5O_{12}$ (YAG) have been manufactured and are in the process of being tested both mechanically and electrically. The results of that effort are monitored in the TFE Verification Program to assess the desirability of technology transfer and assure cost effective efforts in both programs.

Single crystal Al_2O_3 is also being considered, both at GA and TTC, for sheath insulator developmental work based on encouraging results from single crystal Al_2O_3 insulator seal work (Task 4). Sapphire tubes are commercially available and can be machined to desired thickness. Tubes have been ordered and some bonding tests performed on them with very encouraging results.

5.3.3 Ex-reactor Testing

Ex-reactor testing of the sheath insulator is done by Rasor Associates Inc. (RAI). The tests are divided into screening and life tests. The life tests include long term tests at nominal operating conditions and short term accelerated tests at higher temperatures. These tests allow for a better understanding of the degradation mechanisms.

Ovens A, B, and C were converted to isothermal condition, and the electrical tests are now performed in these test rigs under a vacuum rather than a helium atmosphere. Initially, thermal flux furnaces were used for

insulator testing. While this test strategy results in conditions similar to conditions experienced during reactor operation, the problems were significant. The temperature variations along the length of the heater was quite large, and the spread in measured temperatures were as large as 200°K. The uncertainties in sample temperature can be important when considering activated processes such as mass transport. Moreover, the thermal fluxes were small compared to design conditions.

A new test protocol is being developed to evaluate trilayer sheath insulators. The new procedures cover details such as sample handling, cleaning, mounting and testing. Five sheath insulator specimens are currently being tested in test rig A. The test conditions and resistances for the test specimens are shown in Table 5-3. The alumina specimens continue to show resistances of better than 10^7 ohms and have been stable for more than six months.

TABLE 5-3
EX-REACTOR TEST MATRIX: TEST RIG A

Sample Number	Material	Source	Temperature (°K)	Time (Hours)	Resistance* (ohms)
255	$Y_3Al_5O_{12}$	GA	1096	4608	1.1×10^4
250	Al_2O_3	GA	1094	4608	1.4×10^8
251	Al_2O_3	GA	1092	4608	1.3×10^8
252	Al_2O_3	GA	1089	4608	7.1×10^8
A14	Al_2O_3 -10% Nb	TTC	1086	4608	7.1×10^7

* At an applied potential of 10 V.

5.3.4 In-Reactor Testing

UCA-1. The irradiation conditions for the sheath insulators was essentially the same as that for the insulator seals as described in Section 4. The test matrix for the sheath insulators in UCA-1 is shown in Table 5-4. The initial visual inspection of the sheath samples is summarized on Table 5-5.

TABLE 5-4
SHEATH INSULATOR UCA-1 TEST MATRIX

Sample Number	Material	End Design	Canister	Temperature (°K)	Fluence 10^{22} n/cm^2
TTC-A1-1	Al_2O_3	Flat end cermet	5A	1090	3.1
TTC-A1-2	Al_2O_3	Flat end cermet	4B	1137	7.0
TTC-A1-3	Al_2O_3	Flat end cermet	5C	1111	3.5
TTC-YAG-1	$\text{Y}_3\text{Al}_5\text{O}_{12}$	Flat end cermet	5A	1096	3.4
TTC-YAG-2	$\text{Y}_3\text{Al}_5\text{O}_{12}$	Flat end cermet	4B	1130	6.6
TTC-YAG-4	$\text{Y}_3\text{Al}_5\text{O}_{12}$	Flat end cermet	5D	1106	3.4
Y-1	Y_2O_3	Flat end graded	5D	1104	3.1
Y-2	Y_2O_3	Flat end graded	5B	1112	3.7
Y-3	Y_2O_3	Flat end graded	4B	1133	6.8
AL-1	Al_2O_3	Op stress relief	5B	1109	3.5
AL-2	Al_2O_3	Op stress relief	5C	1108	3.3
AA-1	Al_2O_3	Alt stress relief	5D	1112	3.6
AA-2	Al_2O_3	Alt stress relief	5B	1104	3.2
ALH-1	Al_2O_3	Flat end graded	5B	1115	3.6
ALN-1*	Al_2O_3	Op stress relief	4B	1135	6.9
AAN-1*	Al_2O_3	Alt stress relief	5C	1114	3.8

* Nickel plated.

TABLE 5-5
INITIAL VISUAL INSPECTION OF THE UCA-1 SHEATH INSULATORS

Sample Number	Material	Design	Condition
TTC-AL-1	Al_2O_3	Cermet	No cracks
TTC-AL-2	Al_2O_3	Cermet	No cracks
TTC-AL-3	Al_2O_3	Cermet	No cracks
TTC-YAG-1	$Y_3Al_5O_{12}$	Cermet	No cracks
TTC-YAG-2	$Y_3Al_5O_{12}$	Cermet	No cracks
TTC-YAG-4	$Y_3Al_5O_{12}$	Cermet	No cracks
Y-1	Y_2O_3	Graded	Circumferential cracks
Y-2	Y_2O_3	Graded	Circumferential cracks
Y-3	Y_2O_3	Graded	Circumferential cracks
AL-1	Al_2O_3	Graded	Circumferential cracks
AL-2	Al_2O_3	Graded	Circumferential cracks
AA-1	Al_2O_3	Graded	Circumferential cracks
AA-2	Al_2O_3	Graded	Circumferential cracks
ALH-1	Al_2O_3	Graded	Circumferential cracks
ALN-1	Al_2O_3	Graded	Circumferential cracks
AAN-1	Al_2O_3	Graded	Circumferential cracks

Measurements of the electrical resistance at high temperatures for the sheath insulators has been completed and is summarized in Table 5-6. Measurements were taken during the rise to 1173°K and during the return to room temperature. The measurements were done under a positive pressure of high purity He, (99.999%) and with an applied potential of one volt. The system and procedures used were modified twice by Westinghouse Hanford Company (WHC), to reduce the initial observed discrepancies in resistance measurements during heatup and during cooldown. These anomalies appear to have been due to some type of contamination of the sample ends. The final guideline requires that before electrical testing, the specimen and resistance probe should be baked out in a vacuum for 1.5 hours at 1225°K and at a pressure lower than 1.1×10^{-4} torr. Specimen ALH-1, a half length sheath, was not tested because of damage during specimen recovery.

Only the alumina samples appeared to have acceptable resistance at high temperature.

Examination of the UCA-1 sheath insulators is continuing. Further examination includes helium leak testing and metallography. From these tests the following will be determined.

1. Extent of the ceramic cracking.
2. Distortion of the ends.
3. Bonding of the ceramic to the niobium.
4. Changes in microstructure of ceramic and niobium from the high neutron fluence.

UCA-2. Irradiation testing of UCA-2 specimens in the FFTF was completed in Cycle 10C on January 8, 1989. The temperature in the three MOTA canisters was maintained at about 1100°K during irradiation. The estimated peak fast fluence ranged up to about 7×10^{22} n/cm² for the level 4 canister.

TABLE 5-6
UCA-1 SHEATH INSULATOR ELECTRICAL RESISTANCE

Sample Number	Fluence 10^{22} n/cm^2	Irradiation Temperature	Resistance	
			300°K	1173°K
TTC-AL-1	3.1	1090	27	N.A.
TTC-AL-2	3.5	1111	1460	N.A.
TTC-AL-3	7.0	1137	50	N.A.
TTC-YAG-1	3.4	1096	8.1×10^9	217
TTC-YAG-2	6.6	1130	2900	N.A.
TTC-YAG-4	3.4	1160	1.2×10^6	200
Y-1	3.1	1104	N.A.	10
Y-2	3.7	1112	363	8
Y-3	6.8	1106	634	3
AL-1	3.5	1109	2.9×10^5	4180
AL-2	3.3	1108	3.9×10^5	1530
AA-1	3.6	1112	156	N.A.
AA-2	3.3	1108	1.6×10^8	7.1×10^5
ALH-1	3.6	1115	N.A.	N.A.
ALN-1	6.9	1135	3.6×10^4	4820
AAN-1	3.8	1114	4250	570

N.A. = Not available.

The MOTA is in the process of being disassembled and the canisters removed. The test matrix for the sheath insulators is shown in Table 5-7. In addition to sheath insulators, seal insulators and cesium reservoirs as in UCA-1, the UCA-2 canisters also include test specimens of the TFE end restraint (Fig. 5-1) and the intercell insulation (Fig. 5-2).

IFAC-SI. The IFAC-SI is a test program consisting of sheath insulator test specimens to be irradiated in a fast reactor environment which in addition to fast neutrons and temperatures will have a voltage applied across the insulator to simulate actual TFE conditions. The test, scheduled for irradiation in a modified MOTA test vehicle, will consist of four to six sheath insulator samples with guard rings in a full instrumented test train.

5.4 STATUS SUMMARY

A summary of the insulator seal development effort is shown in Table 5-8. The development priority at the moment is as follows:

Reference: Y_2O_3 , plasma sprayed graded or slip cast cermet.

Backups: Al_2O_3 , single crystal.

Al_2O_3 , plasma sprayed graded or slip cast cermet.

$Y_3Al_5O_{12}$ (YAG), plasma sprayed or slip cast cermet.

The designation of Y_2O_3 as the reference material assumes that the use of pure (carbon free) material will yield satisfactory electrical performance. The polycrystalline Al_2O_3 plasma sprayed graded trilayer has experienced over 30,000 realtime test hours in the TRIGA reactor, and there is good evidence that its lifetime is at least several years.

References

- 5-1 V. P. Cone and J. B. Dunlay, Thermionic Technology Final Technical Report for FY-86, Thermo Electron Technology Corporation, 1987.

TABLE 5-7
SHEATH INSULATOR UCA-2 TEST MATRIX

Material	Samples	Design	Canister	Temperature °K	Fluence 10^{22} n/cm^2
Y_2O_3	2	HIP end restraint	1A	1073	4.3
Y_2O_3	1	Intercell insulation	1A	1073	4.3
Al_2O_3	1	Intercell insulation	1A	1073	4.3
Y_2O_3	1	Cermet trilayer	1A	1073	4.3
Al_2O_3	1	Cermet trilayer	1A	1073	4.3
Y_2O_3	1	Graded trilayer	1A	1073	4.3
Y_2O_3	2	HP end restraint	1B	1073	4.3
Al_2O_3	1	Intercell insulation	1B	1073	4.3
$\text{Y}_3\text{Al}_5\text{O}_{12}$	1	Intercell insulation	1B	1073	4.3
Y_2O_3	1	Graded trilayer	1B	1073	4.3
Al_2O_3	1	Graded trilayer	1B	1073	4.3
$\text{Y}_3\text{Al}_5\text{O}_{12}$	1	Graded trilayer	1B	1073	4.3
Y_2O_3	1	Intercell insulation	4C	1073	6.7
$\text{Y}_3\text{Al}_5\text{O}_{12}$	1	Intercell insulation	4C	1073	6.7
Y_2O_3	1	Cermet trilayer	4C	1073	6.7
Al_2O_3	1	Cermet trilayer	4C	1073	6.7
Y_2O_3	1	Graded trilayer	4C	1073	6.7
Al_2O_3	1	Graded trilayer	4C	1073	6.7
$\text{Y}_3\text{Al}_5\text{O}_{12}$	1	Graded trilayer	4C	1073	6.7

Figure 5-1 - UCA-2 TFE end-restraint test specimen

Figure 5-2 - UCA-2 intercell insulation test specimen

TABLE 5-8

SHEATH INSULATOR TECHNOLOGY STATUS

Material	Design	Priority	Electrical Properties	Neutron Stability	Fabricability	Test Matrix (No. of Specimens)				
						UCA-1	UCA-2	TFE 1H1/1H2	Fueled Emitter	Ex- Reactor
Y ₂ O ₃ (P.C.)	Trilayer	High	Poor*	Good	Good	3	4	1		14
YAG (P.C.)	Trilayer	Medium	Good	Good	Difficult**	3	2			7
Al ₂ O ₃ (P.C.)	Trilayer	Medium	Good	Acceptable	Difficult**	10	4	1	9	13
Al ₂ O ₃ (S.C.)	Trilayer	Medium	Good***	Acceptable	TBD					

P.C. Polycrystalline.

S.C. Single Crystal

* Due to impurities in starting material,

** Not crack free.

*** Task 4 results

6. FUELED EMITTER TASK

6.1 OBJECTIVE AND TECHNICAL APPROACH

The function of the fueled emitter is to generate the heat necessary to drive the thermionic conversion process with the geometry, power densities and temperatures required by the reference TFE design, using materials compatible with both the thermionic process and reactor requirements. Specific requirements for the emitter are given in Table 6-1.

TABLE 6-1
FUELED EMITTER DESIGN REQUIREMENTS

<u>Emitter</u>	
Material	Duplex Tungsten
Outside diameter, cm	1.27
Thickness, cm	0.10
Emitter/collector gap, cm	0.025
Nominal temperature, K	1800
<u>Fuel</u>	
Outside diameter, cm	1.0
Inside diameter, cm	Variable
Enrichment, % U-235	93
Fuel length, cm	4.65
Nominal fuel burnup, atom percent	4.1
Peak fuel burnup, atom percent	5.3

The key technical concern is the emitter distortion as a result of fuel swelling over the 7-year lifetime. It is the objective of the fueled emitter task to develop and demonstrate by appropriate ex-reactor testing, in-reactor testing and analytical verification a fueled emitter capable of meeting the above design requirements.

The strategy to demonstrate the performance and lifetime of the fueled emitter is based on extensive in-reactor testing. While duplex tungsten will be the focus of most of the testing, alternate materials, particularly

the HfC strengthened materials, look very promising and will be considered. Additionally, alternate fuel forms will be evaluated which either weaken the fuel or change its swelling characteristics, thereby minimizing emitter distortion. These forms include insulated fuel, and wafered fuel.

During the current reporting period the following tasks were performed:

UFAC fast reactor irradiation program

- 1) The irradiation testing of the UFAC Batch-1 emitters continued and accumulated about 2% burnup on the accelerated and about 0.5% burnup on the prototype emitters.
- 2) PIE of the four fractured UFAC Batch-1 emitters continued with the dimensional and metallographic examinations.
- 3) Design and fabrication of the Batch-2 emitters were initiated.

TRIGA irradiations program

- 1) TFE-1H1, the initial TRIGA test in the TFE Verification Program, began irradiation on September 9, 1988 and has successfully achieved about 5500 hours of irradiation testing (mid-May 1989).
- 2) The irradiation testing of Capsule 2 continued. The three fueled emitters in the capsule accumulated about 5500 hours of testing.
- 3) Capsule 3 was removed from the TRIGA core after 23,871 hours of testing.
- 4) The initial phase of the PIE of fueled emitter Capsules 1 and 3 and the sheath insulator Capsule 4 was completed.

Advanced emitter materials

- 1) Four bars of tungsten hafnium carbide (WHfC) material were purchased from the Advanced Energy Systems of Westinghouse.

6.2 EX-REACTOR TESTING

Ex-reactor analysis has focused on the possibility that bladder and thermal ratcheting phenomena could be important in emitter distortion. Modeling studies were performed early to determine if there is the possibility for thermal ratcheting and/or fission gas induced bladder deformation of the emitter. The studies showed that these deformation mechanisms are not important to overall fuel emitter distortion. No further work is planned at this time.

A second ex-reactor activity is to investigate the creep strength characteristics of the duplex tungsten and other candidate emitter materials. Tungsten hafnium carbide (WHfC) has been selected as the emitter material which could offer a significant lifetime improvement over CVD tungsten, the current reference emitter material. The selection considered both a demonstrated mechanical strength advantage and an available fabrication technology.

Four ingots of tungsten - 0.35 hafnium - 0.025 carbon (WHfC) have been ordered from Westinghouse. These bars will be used to characterize the material, fabrication development, and as a candidate advanced emitter material for irradiation testing in UFAC Batch-2. The characterization will include mechanical strength/creep testing, chemical compatibility with the UO_2 fuel, and determining the electronic work function of the bare material. The fabrication development will consist of the machining and joining techniques required to fabricate thermionic emitters from the extruded bar.

The four ingots, 4 in. long by 2.5 in. diameter, have been successfully arc-cast. In this condition the material has very large

grains and is not very strong. A final extrusion to size for grain refinement and improved mechanical properties is required. Prior to this final extrusion, the ingots are coated with molybdenum, which acts as a lubricant. Difficulty has been experienced in making this final extrusion. While the exact cause is not known, it is believed to be caused by the molybdenum coating which was too thin, causing an interaction with the dies during the extrusion. Several dies were damaged during the extrusion when the material would "hang up" and eventually freeze in the dies. New dies have been ordered and the four ingots have been reground and encased in an adequate molybdenum coating. Delivery of the material has been delayed into mid-to-late May 1989.

Characterization of the base material, however, was actively pursued during the reporting period despite the above setbacks. Westinghouse provided GA several smaller bars from prior runs and from incomplete extrusions. Metallographic examination of these samples showed a fine grained structure as expected of the final material. A secondary phase consisting of iron and chromium was, however, dispersed throughout the matrix. This secondary phase was probably caused by interaction with the die during the extrusion is not representative of the final material.

Fabrication development of the WHfC was also pursued. Electrical discharge machining (EDM) will be used to form the emitter cup from the extruded bar. Sample pieces were successfully shaped by EDM. Honing the machined surface to remove surface microcracks has not been tried but does not appear to be a problem. Joining the WHfC to tantalum was also successfully done by using the same diffusion bonding cycle that is used to bond CVD tungsten to tantalum. It therefore appears that fabricating an emitter and joining it to the rest of the TFE is feasible using standard fabrication/machining techniques.

Finally, specimens of WHfC and UO_2 fuel were fabricated to run high temperature chemical compatibility tests. The specimens will be honed and tested in the the next reporting period.

6.3 UFAC IN-REACTOR TESTING

The UFAC testing will be performed in three separate batches consisting of about ten emitters each. Batch-1 is presently being irradiated in EBR-II. Figure 6-1 shows the current test schedule for the UFAC experiments. The first line of the schedule gives the EBR-II operating intervals (known as cycle numbers) and the duration of each operating period. The basic UFAC design consists of individually encapsulated fueled emitter test specimens which are contained within the EBR-II irradiation assemblies. This was described in Ref. 6-1. The test matrices for Batches-1 and -2 and the identification scheme for the individual emitters is shown on Table 6-2.

6.3.1 UFAC Batch-1

Status. The ten UFAC Batch-1 emitters began irradiation testing in May 1987 at the start of EBR-II run 144. Seven half sized emitters were installed into subassembly UFAC-1 and loaded into core location 5E2 for accelerated irradiation testing. In this location the rate of UO_2 fuel burnup is about a factor of four to five faster than in a 2 MWe reactor thus enabling accelerated burnup testing. Coincidentally, however, the rate of fluence accumulation is about a factor of two faster than in the 2 MWe reactor leading to accelerated material degradation. Three full sized prototypic emitters were installed in subassembly UFAC-2 and loaded into core location 7F1 for testing. In this location these emitters accumulated burnup at about the same rate as in the 2 MWe reactor fuel. Fluence accumulation, however, is about a factor of four greater than in the 2 MWe reactor. The projected irradiation history for Batch-1 is shown in Table 6-3.

Postirradiation examination. Three accelerated emitters from UFAC-1 and one prototype emitter from UFAC-2 are undergoing examination in the Hanford hot cells. These were removed from the experiments when it was observed they were fractured near the tantalum/tungsten transition joint. Emitter diameter profiles were measured at four angles, i.e., 0° , 45° , 90° , and 135° , with the results shown on Figs. 6-2, 6-3, 6-4, and 6-5. SU1-6 and SU1-10 showed similar profiles with the peak emitter deformation

Figure 6-1

EBR-II UFAC FUELED EMITTER TESTING SCHEDULE

TABLE 6-2
UFAC EMITTER IDENTIFICATION SCHEME

<u>ID Number</u>				<u>Emitter</u>	<u>Fuel</u>	
<u>Real</u>	<u>Accelerated</u>	<u>Diameter</u>	<u>Thickness</u>	<u>Temperature</u>	<u>Design</u>	<u>Comments</u>
<u>Time</u>						
<u>UFAC Batch-1</u>						
SU2-1		0.5	0.040	1800	15% void	Reference design
SU2-2		0.5	0.050	1800	15% void	Thick emitter
SU2-3		0.5	0.040	1800	Solid	Solid fuel
	SU1-4	0.25	0.020	1700	15% void	Ref des/low temp
	SU1-5	0.25	0.020	1800	15% void	Reference design
	SU1-6	0.25	0.020	1900	15% void	Ref des/high temp
	SU1-7	0.25	0.020	1800	Solid	Solid fuel
	SU1-8	0.25	0.020	1900	Solid	Solid fuel/high temp
	SU1-9	0.25	0.025	1800	15% void	Thick emitter
	SU1-10	0.25	0.025	1900	15% void	Thick emit/high temp
<u>UFAC Batch-2</u>						
SU2-11		0.5	0.040	1800	15% void	Batch-1 replica
SU2-12		0.5	0.040	1800	25% void	High fuel void
SU2-13		0.5	0.040	1800	15% void	Advanced material
SU2-14		0.5	0.040	1800	-	Insulated fuel
SU2-15		0.5	0.040	1800	-	Wafered fuel
	SU3-16	0.25	0.020	1800	15% void	Batch-1 replica
	SU3-17	0.25	0.020	1800	-	Insulated fuel
	SU3-18	0.25	0.020	1800	15% void	Advanced material
	SU3-19	0.25	0.020	1900	15% void	Adv mat/high temp
	SU3-20	0.25	0.020	1800	-	Wafered fuel

Figure 6-2 - Axial diameter profile for fueled emitter SU1-6

Figure 6-3 - Axial diameter profile for fueled emitter SU1-7

Figure 6-4 - Axial diameter profile for fueled emitter SU1-10

Table 6-5 - Axial diameter profile for fueled emitter SU2-3

TABLE 6-3
UFAC BATCH-1 BURNUP PROJECTION
 (ATOM-PERCENT)

EBR-II Run	UFAC-1: Accelerated Specimen							UFAC-2: Real Time Specimen		
	SU1-4	SU1-5	SU1-6	SU1-7	SU1-8	SU1-9	SU1-10	SU2-1	SU2-2	SU2-3
144A	.80	.80	.81	.67	.68	.90	.91	.22	.24	.17
145A	*	*	*	*	*	*	*	.36	.40	.28
146B	1.46	1.47	1.49	1.24	1.25	1.65	1.67	.55	.60	.42
147A	1.86	1.88	1.90	1.58	1.60	2.11	2.13	.66	.73	.51
149A	2.48	2.50	**	**	2.13	2.80	**	.84	.93	**
150B	3.07	3.10			2.64	3.46		1.02	1.12	
151A	3.41	3.44			2.93	3.84		1.12	1.24	
152B	4.11	4.15			3.51	4.59		1.32	1.46	

* UFAC-1 removed for interim examination.
 ** Capsules removed for terminal examination.

occurring at the top of the fuel column. Peak diameter increases were less in SU1-10 with a 25 mil thick emitter than in SU1-6 with a 20 mil thick emitter. The diameter profile in SU1-7 peaked at the lower end of the fuel column. SU1-7 contained solid pellet fuel while SU1-6 and SU1-10 contained fuel pellets with a fabricated central void. SU2-3 with solid pellet fuel showed a diameter profile similar to SU1-7.

Discrete measurements of SU1-6, SU1-10 and SU2-3 were taken to characterize the diameter profile at the top of the emitters in the vicinity of the fracture. In all cases, the diameter increase toward the top end of the emitter is consistent with an extrapolation of the continuous profilometer traces. The SU1-6 measurements appear to be offset by approximately .5 mil. Similar measurements of SU1-7 were not possible; the lower end of SU1-7 fractured during the handling to remove fracture surface replicas for examination.

Scanning electron microscopy (SEM) of the fractured emitter and transition pieces for both SU1-7 and SU2-3 was completed. A comparison of the replicas from the fractured emitter surface and the corresponding transition piece showed a one-to-one correlation, indicating no material was lost from either surface during the fracture or during handling and examination.

SEM of the replicas show two distinct structures (Fig. 6-6). One microstructure is the radially oriented grains typical of CVD tungsten. The second is a rather mottled structure, suggestive of the lapped interface on the tantalum. The replicas indicate approximately 50% intergranular fracture occurred in the accelerated burnup emitter SU1-7 and 90% intergranular fracture in the "real-time" emitter SU2-3.

The fractured emitters were sectioned to obtain samples for fuel burnup analysis, metallographic examination of the fracture zone (both W and Ta), and the fuel column. In addition to providing data to identify the cause of the unexpected fractures, destructive examinations of the fueled emitters will provide verification data on operating conditions. Fuel isotopic measurements will provide data on fuel burnup and heat rate, while fuel metallography provides fuel temperature estimates for extrapolation to emitter surfaces.

Diameter measurements of the tantalum transition pieces near the fracture were taken at 45° increments around the circumference (Table 6-4). Diameters averaged .2503 in. for the three accelerated burnup emitters and .5005 in. for the prototype emitter. Fabrication specifications for the Ta transition pieces were .2500 $\pm$ .0005 in. and .5000 $\pm$ .0005 in. respectively.

TABLE 6-4
DIAMETERS OF SU1-6, SU1-7, SU1-10, SU2-3 TANTALUM
TRANSITION PIECES AT FRACTURE

Average Emitter Diameter of Ta Transition Piece at Fracture (inches)						
	0°	45°	90°	135°	180°	
UFAC-1						
SU1-6	.2507	.2504	.2508	.2515	.2511	.2509
SU1-7 *	.2495	.2496	.2494	.2495	.2495	.2495
SU1-10	.2501	.2494	.2504	.2501	.2509	.2502
	.2510	.2507	.2505	.2506	.2510	.2507
UFAC-2						
SU2-3	.5003	.5003	.4999	.5018	.5001	.5006

* Two sets of measurements were taken.

TUNGSTEN →

**LAPPED INTERFACE
OR TANTALUM**

Figure 6-6 - Morphology of the SU1-7 transition piece fracture surface

Collector examinations were completed on these four emitters. Outer diameters were measured at 0° and 90° at equally-spaced axial locations for SU1-6, SU1-10, and SU2-3. Measurements showed essentially constant diameters over the axial length of the collectors and were averaged over the length of the samples. Measurements are summarized in Table 6-5. The two collectors from the accelerated burnup emitters, i.e., SU1-6 and SU1-10, had diameter increases of approximately .001 in., and the collector from the real-time burnup emitter, i.e., SU2-3, had a diameter increase of .002 in. The collector from capsule SU1-7 was sectioned axially, and the inner surface visually examined and photographed. The inner surface appeared normal, though slightly discolored.

TABLE 6-5
SUMMARY OF SU1-6, SU1-10, SU2-3 COLLECTOR DIAMETER MEASUREMENTS

<u>Capsule</u>	<u>Average Collector Outer Diameter,* inch</u>	
	<u>Preirradiation**</u>	<u>Postirradiation</u>
SU1-6	.7439 ± .0001	.7451 ± .0001
SU1-10	.7442 ± .0001	.7453 ± .0001
SU2-3	.7441 ± .0001	.7460 ± .0004

* Average ± one standard deviation.

** Measured at 0° and 90° prior to encapsulation in stainless steel capsules. (Fabrication dimensions .7445 ± .0005 inch.)

6.3.2 UFAC Batch-2

Description. The second batch of fueled emitters is scheduled for irradiation testing beginning August 1989 in EBR-II run 153. This batch will include five full-size prototype emitters and five half-size accelerated emitters. The five prototype emitters will be installed, along with the two Batch-1 prototype emitters, in subassembly UFAC-2B and loaded into core location 7F1. The five accelerated emitters will be installed into subassembly UFAC-3 and loaded into core location 5E2. Since these locations are the same as for Batch-1, both the burnup and fluence acceleration for Batch-2 will be similar to those cited previously for Batch-1.

The Batch-2 test matrix was shown in Table 6-2. The emphasis of this second series of UFAC emitters is on advanced emitter configurations with the objective of increasing emitter lifetime/performance. The matrix includes:

- 1) Batch-1 replicates - for performance comparison.
- 2) Insulated fuel - to investigate the effect of a depleted layer of UO_2 adjacent to the emitter. The depleted UO_2 will result in a higher average temperature for the fissioning fuel.
- 3) Wafered fuel - to investigate the effect of lower fuel temperature on emitter distortion.
- 4) Advanced emitter material - to investigate the irradiation performance and stability of an advanced, higher strength emitter material.

Figure 6-7 is a comparative illustration of the fueled emitter configurations for Batch 2. The various fuel forms can be seen in the figure. (Note that no distinction is made in the figure for emitter materials.)

In addition to the fuel form and emitter material variations in the Batch 2 emitters, the SU2-11 and SU2-12 test articles incorporate a solid fission product trap similar to the interim design added to 1H3 and future TFEs, as described in the last semiannual report (Ref. 6-X, Sec. 8.3.1, p. 134). The SU2-12 and -14 emitters feature a machined 0.020 in. thick region below the tungsten to tantalum bond intended as stress relief for the bond. This feature replicates the H-series TFE emitter "stem" configuration in the region below the bond. Both the fission product trap and emitter stress relief are shown in Fig. 6-7.

Insulated fuel design. The fuel for this concept consists of an enriched solid pellet contained within a depleted annular "rind". The depleted layer is 0.8 mm thick in the real-time test and 0.5 mm thick in the accelerated. The fission and gamma heat generated in the central

Figure 6-7 - UFAC Batch-2

pellet must be conducted through the depleted outer UO_2 prior to rejection from the emitter. The low thermal conductivity of the UO_2 will cause the central zone to operate at a higher temperature than the outer rind causing the active fuel to have a lower creep strength. The fission product induced swelling of the central zone will be reacted by the stronger, outer fuel rind instead of the tungsten emitter. Diametral distortion of the emitter should therefore be reduced.

A thermal model of the insulated fuel design has been developed for both real-time and accelerated specimen. The model considers heat generated from fission and from gamma heating with an emitter temperature maintained at 1800 K. Figure 6-8 presents a comparison of the calculated radial temperature profile for a real-time, prototype, insulated specimen to a standard specimen. The real-time specimen's maximum fuel temperature has been increased by about 260 K. Similarly, Fig. 6-9 compares the radial temperature profiles of standard versus insulated accelerated specimen. A maximum fuel temperature increase of about 317 K is achieved for the accelerated specimen.

Wafered fuel concept. The design for the wafered fuel concept is shown in Fig. 6-10. The fuel system in this concept is replaced by alternate wafers of fuel and tungsten. A 3-to-1 ratio of fuel-to-tungsten has been chosen to minimize both the temperature and the gradient in the fuel. Both axial and radial heat flow will occur in the fuel. Axially, the heat will flow to the tungsten wafers and then radially outward along the wafer for rejection out of the emitter. Once the fuel redeposits on the emitter surface, the radial heat flow will further lower the fuel temperature. The fuel will therefore operate isothermally only slightly above the emitter temperature of about 1800 K.

A two dimensional thermal model of the wafered fuel concept was developed for both the real time and accelerated specimens. Heat generation from fission and gamma heating were considered to obtain the 1800 K emitter temperatures. Results of this analysis show the wafered fuel to significantly reduce the maximum fuel operating temperature.

Figure 6-8 - Real time specimen fuel radial temperature profile:
standard vs. insulated fuel designs

Figure 6-9 - Accelerated specimen fuel radial temperature profile:
standard vs. insulated fuel designs

Figure 6-10 - Wafered fueled emitter concept

Figures 6-11 and 6-12 present a comparison of the wafered fuel concept to other fuel concepts for UFAC Batch-2 real-time and accelerated specimen.

In addition to reducing fuel temperature, the wafered fuel design causes larger axial than radial fuel temperature gradients. This may further assist in reducing radial fuel swelling and emitter distortion. Table 6-6 presents a comparison of some of the calculated operating conditions for real-time and accelerated UFAC Batch-2 standard, wafered, and insulated fuel concepts.

TABLE 6-6
UFAC BATCH-2 FUEL CONFIGURATIONS: COMPARISON OF OPERATING CONDITIONS

	Real Time			Accelerated		
	Standard	Wafer	Insulated	Standard	Wafer	Insulated
Emitter temperature, K	1800	1800	1800	1800	1800	1800
Cell power, W	613	625	569	320	310	333
Fuel power, W/cc	196	233	203	709	879	895
Fuel maximum temp, K	2119	1874	2379	2119	1876	2436
Radial temp gradient, K/in	1400	303	2067	2800	752	4183
Axial temp gradient, K/in		857			1814	

Several alternate concepts were investigated for the wafered design. These included the sectioned design shown in Fig. 6-13, and the "sleeved" design shown in Fig. 6-14. Serious concerns were raised in the analysis of both of these concepts concerning the stability of the void areas within the fuel region. Evaporation/condensation of the UO_2 fuel upon heatup and subsequent operation is likely to fill in the voids and cause a fuel/metal structure which could be stronger than the original UO_2 fuel. It was therefore decided that for this initial wafered irradiation, the proposed disc concept would be investigated. A possibility of including simple sleeves in some of the wafers will be investigated. Alternative designs will be also considered for UFAC Batch-3.

Advanced material design. The advanced material emitter design is similar to the standard CVD tungsten emitter design (Fig. 6-15). Three advanced emitters are planned for Batch-2. A prototype and an accelerated

Figure 6-11 - Comparison of the temperature profile for a prototype wafered and standard emitter design

Figure 6-12 - Comparison of the emitter temperature profiles for an accelerated wafered and standard emitter design

Figure 6-13 - Wafered and sectioned fuel design was analyzed but is not considered for UFAC Batch-2 testing

Figure 6-14 - "Sleeved" wafered fueled emitter concept

Figure 6-15 - Advanced emitter
material concept

emitter operating at the 1800 K design temperature, and an accelerated emitter operating at 2000 K. This latter emitter would investigate the capability of the material and provide a data point for further high temperature testing in Batch-3. The performance based on available data shows that the material should perform exceptionally well up to 2400 K.

As explained in Section 6.2.1, fabrication of the base material is still pending and insertion of these advanced emitters into EBR-II in run 153 is uncertain at this point. Fabrication development and compatibility testing using available material is proceeding and all attempts will be made to complete fabrication on time. Insertion into run 154 in December 1989 is the fallback position.

6.3.3 TRIGA irradiations

Status summary. Irradiation testing of fueled emitters and partial length TFEs in the TRIGA Mark-F is continuing. Postirradiation testing of fueled emitter Capsules 1 and 3 and sheath insulator Capsule 4 was initiated.

TFE-1H1, the initial TFE fabricated under the TFE Verification Program, was installed in the TRIGA core and began irradiation testing on September 16, 1989.

Diode 2.1, the upper diode in Capsule 2, remains operational after having accumulated about 33,000 hours of irradiation.

Postirradiation examination. Destructive postirradiation examination of three capsules was initiated. The examination is being done in the GA hot cell and is expected to last about one month. The objective of this effort is to recover the test specimens from these three capsules. A one to two month effort is planned in FY-90 to perform further detailed dimensional and metallographic examination of the recovered emitters and sheath insulators. Visual examination and some gamma scanning will be done in the current effort.

The three capsules were stored at the TRIGA Mark-F poolside since their removal from irradiation testing. Prior to transfer to the hot cell, the upper portion of the capsules was sectioned and removed to ease the transfer (Figs. 6-16 thru 6-20). The transfer was done by inserting the core section of each capsule into a leaded cask and transferred using a 120 ton crane.

Capsule 4, which contained the sheath insulators, was the first to be destructively examined. The capsule was sectioned using a small air grinder and the sheath insulator specimens recovered (Fig. 6-21). Visual examination at a 4X magnification showed no insulator distortion or debonding. The thermocouples and attachments were carefully examined since several gave open signals and were believed to have failed during service. The visual examination showed that the thermocouples had slipped out of the welded niobium pockets into which they were installed and had not really failed (Fig 6-22). This method of attachment should be noted so that it is not repeated in later designs, such as IFAC-SI, which has similar instrumentation requirements.

Fueled emitter Capsule 1 was then examined. Detailed gamma scanning of the upper portion of the capsule was done to determine the activity of the fission product plenums. The only detected activity in the plenum was cesium 134 at about twice background level. It should be noted that this is not a fission product but an activation product from irradiation of cesium 133 from the cesium reservoirs. The plenums were pinched off and individually recovered and gamma scanned. Reduction of the data is still in progress but the primary constituent in the traps was krypton 85 with small traces of cesium 134. The only fission gas line scanned was for diode 2.2. The results, in count per minute, are shown in Table 6-7. Cesium 137 is the only species attributable to a fission product. The ends of the fission lines were visually examined and were found to be clean and free of blockages.

A visual examination of the recovered diodes showed that all three seal insulators were cracked. Prior neutron radiographs had clearly shown

Figure 6-16 - Sectioning capsules at TRIGA

Figure 6-17 - Axial section of capsule removed uncovering internal structures

Figure 6-18 - Cutting capsule instrumentation

Figure 6-19 - Capsule completely severed

Figure 6-20 - Upper section removed and ready for disposal

Figure 6-21 - Recovered sheath insulator specimen

Figure 6-22 - Thermocouple attachment on sheath insulator capsule. Bottom thermocouple properly installed. Top thermocouple slipped out of niobium pocket

TABLE 6-7
COUNT RATE FOR FISSION GAS LINES OF DIODE 2.2

Distance From Top of Emitter (in.)	Cs-134 (CPM)	Cs-137 (CPM)
0.0 - 3.0	5000	2600
3.0 - 6.0	1800	1200
6.0 - 9.0	1200	860
9.0 - 12.0	550	420
12.0 - 15.0	590	390
15.0 - 18.0	200	170
18.0 - higher	traces	traces

that the insulators for diodes 2.2 and 2.3 were cracked. The crack in 2.1 was similar and is believed to have been caused during handling and section of the capsule in the hot cell. All three emitters could only be partially removed from the collectors. The remaining portion of the broken seal insulator on the collector would cause the emitters to jam and prevent complete removal. The as-fabricated clearance between the emitter and the inner diameter of the seal insulator is 0.005 in.; thus the jamming gives some indication of emitter diametral distortion. Emitter 2.2 jammed only at the very bottom end (Fig. 6-23). Several attempts were made to completely remove it from the collector. As it finally did come out the stem of the emitter was broken (Fig. 6-24). The other emitters have not been recovered.

Sectioning and recovery of the diodes in Capsule 3 was then initiated. The same procedure was followed as for Capsule 1. No major activities were found in the fission product traps except for krypton and cesium 134, as in Capsule 1. Gamma scanning of the individual traps and recovery of the diodes is in process.

6.4 MODELING

The primary objective of the fueled emitter modeling task is to calibrate and validate existing analytical tools for the prediction of the

Figure 6-23 - Emitter 2.2 partially removed from collector housing

Figure 6-24 - Emitter 2.2 removed from collector housing

fueled emitter lifetime. The model will be used to evaluate and define fueled emitter specimens for in-core tests and to evaluate the lifetime of the fueled emitter in the 2 MW(e) thermionic reactor to assure that it can reliably achieve its 7-year lifetime goal.

Over the last 12 months, the fueled emitter modeling efforts concentrated at first on exploring the effect of reducing the predicted fuel surface tension at equilibrium. It was concluded that the operating histories for the TRIGA experiments should be reviewed before further recalibration work. This review was concluded late in 1988, and with availability of the reviewed operating histories, work to recalibrate or validate LIFE-4 was started.

The effects of 1) increasing the calibration constant controlling solid fission product swelling, and 2) increasing the coefficient of friction between fuel and emitter were explored, as well as the effect of surface tension. Further, fission gas release constants were changed to more closely predict the plenum reservoir pressures for the fueled emitters irradiated in TRIGA. The incubation time model for release of fission gas built into LIFE was changed. Increasing all of the above constants caused substantially more emitter strain, a change that was needed, because with less gas retention, the strain being measured was consistently under-predicted. While solid fission product swelling is fairly well known, increasing the prediction of such swelling can be viewed as a surrogate for increased gas pressure swelling. Reduction of the predicted nuclear fuel surface tension also permits more swelling without weakening the fuel. Adjustment of fuel creep strength has limited effect because of two competing effects that tend to cancel out each other. When the creep strength is reduced, more swelling is allowed, but the fuel supports less interface pressure. When the creep strength is increased, just the opposite occurs.

Fairly good success was achieved in predicting the deformation and fission gas pressure of six of the fueled emitters irradiated in TRIGA. These data have been reviewed and accepted. This calibration is considered

to be tentative until the effects of radial power depression can be checked and until the ability of the calibration to predict the "historical" data and the UFAC data currently being collected have been checked. The results are tabulated for the end points of the accepted data in Table 6-8.

As can be seen, deformations for Capsule 1 Diode 2, Capsule 2 Diode 2, and Capsule 3 Diode 1 are predicted quite well. The correlations between calculated and experimental deformations are within 0.6 mils. However, there is greater disagreement with the other three test diodes. Experimental deformations were less than calculated deformations by a range of 1.1 to 1.9 mils. Plenum pressures correlated quite well, except for Capsule 3 Diode 3, where the calculated pressure was almost a factor of two greater than the measured pressure. The measured pressure in Capsule 3 Diode 3 seems different from the others in this respect, the others showing experimental fission gas release approaching 100%.

Figures 6-25 and 6-26 compare calculated fission gas reservoir pressures and fueled emitter deformations to measured reservoir pressures and deformations as a function of time for Capsule 3 Diode 1. This demonstrates that the calculations correlated with the experimental data reasonably well.

References

- 6-1 TFE Verification Program, Semiannual Report for the Period Ending April 30, 1988, GA-A19269, June 1988.

TABLE 6-8
EXPERIMENTAL VERSUS CALCULATED DEFORMATIONS AND RELEASED FISSION
GAS PRESSURES

Test Diode	Calculation Deformation (mils)	Time* (Hours)	Measured Deformation (mils)	Time* (Hours)	Calculation Plenum P (mbar)	Measured Plenum P (mbar)
Capsule 1						
Diode 2	2.0	10,558	1.4	10,946	23.3	-
Capsule 2						
Diode 2	7.0	20,658	7.2	20,100	49.9	46.3
Diode 3	5.3	18,858	3.7	18,000	36.3	35.3
Capsule 3						
Diode 1	9.1	15,058	8.9	14,200	26.5	29.9
			10.7	16,200		
Diode 2	5.1	14,458	3.2	13,600	35.6	-
Diode 3	4.6	16,058	3.5	16,200	38.0	22.0

* Measured deformations were not, in general, taken at the same time as the calculation ceased. The times in the first column are times at cessation of the calculation, and also measured plenum pressures. Times in the second column are times at deformation measurement. Where possible, a second measurement bracketing the calculated deformation was provided.

Figure 6-25 - Comparison of calculated and experimental reservoir pressures for Capsule 3, Diode 1. The calculated reservoir pressure for a 100% release of fission gas is also shown.

Figure 6-26 - Comparison of calculated and experimental deformations for Capsule 3, Diode 1. The deformation of the fueled emitter measured at the midpoint is shown and is identified as experimental.

7. CESIUM RESERVOIR AND INTERCONNECTIVE TFE COMPONENTS

7.1 OBJECTIVE

The overall objective of the cesium reservoir and interconnective TFE components task is to develop and validate the performance of cesium reservoirs and interconnective TFE components, such as intercell insulation, fission product vents and fission product traps. In particular, the objectives are:

1. Design integral cesium reservoirs and interconnective TFE components.
2. Develop required fabrication processes for integral cesium reservoirs and interconnective TFE components and document the process specifications.
3. Fabricate integral cesium reservoirs and interconnective TFE components for in-reactor and ex-reactor tests.
4. Verify the performance characteristics and lifetimes associated with integral cesium reservoirs and interconnective TFE components by means of in-reactor and ex-reactor testing.
5. Develop an analytical model of the performance and lifetime of the Cs reservoir and interconnective TFE components.

7.2 TASK DESCRIPTION

The integral cesium reservoir and interconnective TFE tasks consists of five subtasks:

1. Cesium reservoir and interconnective TFE components design.
Design integral cesium reservoir test specimens consistent with the TFE design requirements. Design test specimens of intercell insulation and interconnective TFE components such as TFE end

restraint and fission product traps, ports and tubes consistent with TFE design requirements.

2. Fabrication development. Determine integral cesium reservoir materials with the capability for storing cesium either as intercalation compounds or as sorption reservoirs. Develop systems for containing the integral reservoir material in a TFE environment. Select insulator materials such as Al_2O_3 , Y_2O_3 and YAG and develop processes for fabricating TFE end restraints and intercell insulation components. Develop materials for use as fission product traps and develop components for fission product vent and port tubes.
3. Ex-reactor testing. Perform ex-reactor tests to determine the cesium pressure of integral reservoir materials as a function of cesium loading and temperature. Examine the ability of intercell insulation coatings to reduce parasitic discharges. Determine the mechanical stability of TFE end restraints.
4. In-reactor testing. Determine the effect of fast neutron fluence on the stability of the pressure versus temperature characteristics of integral cesium reservoirs at a fixed cesium loading. Also examine the mechanical stability of prototypical TFE end restraints and intercell insulation coatings. Determine fission product release characteristics from the fuel and evaluate the effectiveness of the fission product control components and assemblies.

7.3 PROGRESS DURING PRESENT REPORTING PERIOD

7.3.1 UCA-1/2 Postirradiation Examination

From the neutron radiographs it was observed that the UCA-2 HOPG specimen failed in a fashion very much like the failures in UCA-1: the niobium subcapsule was ruptured and the tantalum inner liner was greatly distended upon removal from FFTF.

The Nb subcapsule for the POCO specimen was undamaged and the Ta inner liner was intact. (A harp distortion in the Ta is visible in preirradiation x-ray radiographs and resulted from intercalation in the isotropic POCO graphite.) The graphite discs have shrunk as a result of irradiation. The postirradiation thickness of these discs is about 14% smaller than the preirradiation value and about 4% larger than the as-built, preintercalation value. The changes in the disc diameter are hard to quantify due to the decreasing absorption length as the edge of the disc is approached. However the same qualitative changes occurred in the diameter as did in the thickness.

The Cs vapor pressure over the reservoir material was determined using the Cs meter described in the last semiannual progress report (Ref. 7-1). The Cs meter was calibrated against a liquid Cs reservoir. Similarly, the capacitance manometer used at GA to characterize the unirradiated graphite was calibrated against a liquid Cs reservoir. An additional cross calibration was made using Stackpole PC-113 porous carbon loaded to ~1300 mg Cs/gC, with the P_{Cs} versus T_{carbon} being measured both in the lab at GA (capacitance manometer) and in a hot cell at WHC (Cs meter). The comparison is shown in Fig. 7-1. The agreement is better than ~20% between 2 and 4 torr, the region of interest in the TFE program.

No Cs was detected for the PC-113 specimen removed from UCA-1 over the temperature range from 100°C to 1100°C. Figure 7-2 shows the pressure vs. inverse temperature behavior of the UCA-2 POCO CZR-2 specimen before and after irradiation. At any given temperature, the postirradiation pressure is about 75% larger than the preirradiation pressure, between 2 and 4 torr. The calibration of Fig. 7-1 suggests that this pressure increase may actually be somewhat larger than indicated.

When the data of Fig. 7-2 are compared with theoretical P-T behavior for the Cs intercalation compounds (Ref. 7-2), it is found that the preirradiation P-T behavior lies intermediate between the isosteres for the equilibrium reactors:

Figure 7-1 - Cross calibration with Stackpole
PC-113

Figure 7-2 - UCA-2 S/N 199 POCO CZR-2

From the measured Cs loading of this sample (222 mg Cs/g the $C_{60} \longleftrightarrow C_{48}$ ($4^{th} \rightarrow 5^{th}$ stage) equilibrium should apply. After irradiation the data falls more nearly on the line corresponding to the $2^{nd} \rightarrow 3^{rd}$ stage equilibrium $2 C_{36}Cs + Cs \rightarrow 3 C_{24}Cs$. At the present time it is not clear if these correlations, and the observation that the graphite thickness returned very nearly to its preintercalation value are meaningful. Indeed the neutron fluence may act to convert the original C_{12n} Cs compound to a new compound with a different coordination number and a different Cs vapor pressure. Only the C_{12n} Cs compounds have been observed and studied to date.

While the reason for the pressure increase with irradiation is not understood at this time, a potential impact on TFE design is suggested. Previous scenarios involving a loss of Cs, e.g., a cracked insulator seal, have led to a decrease in Cs pressure in the interelectrode space. The UCA-2 results with POCO graphite represent an offsetting effect which may tend to stabilize the Cs pressure in the event of such an occurrence.

7.3.2 UFAC Batch-2 Reservoir Samples

The decision was made during the present reporting period to perform additional in-core irradiation tests of candidate reservoir materials. These in-core tests will be in EBR-II during cycles 153 and 154 coinciding with the second batch of UFAC fueled emitters. These tests will commence in August, 1989 and end in February, 1990. This will provide neutron damage data approximately 18 months ahead of UCA-3.

The original choice of materials for the EBR-II test is shown in the test matrix in Table 7-1. The choice of HOPG and POCO CZR-2 was based on the desirability of utilizing an intercalation compound for the TFE Cs reservoir. The use of one-half as much graphite in the EBR-II test as compared with UCA-1/2 is based on the possibility that neutron swelling could be a contributor to the failure of the HOPG specimens in UCA-1/2. The use of both Cs loaded and unloaded graphite specimens is to determine the influence of Cs on graphite dimensional changes in the neutron fluence.

TABLE 7-1

TEST MATRIX FOR IN-REACTOR EBR-II TEST OF INTEGRAL CESIUM RESERVOIRS

Material	Reservoir Type	<u>Pretest Load Conditions</u>	
		Cesium Pressure (torr)	Integral Reservoir Temperature (K)
Mo	Sorption	TBD	TBD
Al ₂ O ₃	Sorption	TBD	TBD
HOPG-ZYH	Intercalation	TBD	TBD
HOPG-ZYH	-	0	-
POCO CZR-2	Sorption/Intercalation	TBD	TBD
POCO CZR-2	-	0	-

The new materials, porous alumina spheres and low density sintered molybdenum, are included since these materials display large BET surface areas (Ref. 7-3) and should be relatively unaffected by neutron bombardment compared to carbon.

The Cs loading capability of the Al₂O₃ and the Mo samples is not known. If during initial characterization it is determined that the volume specific loading is not adequate for TFE use, these materials will be eliminated from the EBR-II test matrix. In addition, the very recent UCA-2 PIE results have suggested that the present EBR-II matrix (Table 7-1) is not the optimal choice. This matter is currently under review and may be altered.

7.3.3 Reservoir Materials Characterization: Stackpole PC-113

The Cs loading and pressure vs. temperature behavior of this material was measured in the following parameter space:

$$\begin{aligned}
 750^{\circ}\text{K} &\lesssim T_{\text{reservoir}} \lesssim 1100^{\circ}\text{K} \\
 500 \text{ mgCs/gC} &\lesssim \text{specific loading} \lesssim 1300 \text{ mgCs/gC} \\
 1 \text{ torr} &\lesssim \text{PCs} \lesssim 7 \text{ torr.}
 \end{aligned}$$

These data are shown in Figs. 7-3 and 7-4. The Cs specific loading of PC-113 is comparable with that of POCO CZR-2 reported earlier (Ref. 7-2) on a grams Cs per cm³ of reservoir basis. There is a major disadvantage to the PC-113 material however. Both the loading curves and the P-T curves are very steep. This manifests itself in extreme sensitivity of the Cs pressure to small losses of Cs from the reservoir material. Unlike POCO or HOPG, which form Cs intercalation compounds, the partial loss of Cs over a 7-year mission with a PC-113 reservoir would result in a considerable reduction in Cs pressure. For this reason it is not deemed to be a viable candidate for the TFE Cs reservoir use. This decision is reached independent of the negative neutron damage effects mentioned in Section 7.3.1 above.

7.3.4 Fission Product Trap Materials

Fission product issues. From U-235 fission a large number of daughter nuclides are formed and subsequently decay. The stable nuclides resulting from these decay sequences are given in Table 7-2. In addition, each fission event releases two oxygen atoms in the fuel lattice which can undergo subsequent chemical reactions with uranium or daughter nuclides. There is a very large number of possible chemical reactions that could occur in the 1800°K fuel matrix between the stable nuclides listed in Table 7-2 and also between these nuclides and oxygen. Some of the reaction products, as well as some of the unreacted elemental nuclides, will be volatile at 1800°K and will emanate from the fuel. It is these volatile species, and not simply the stable nuclides listed in Table 7-2, that are considered fission products in the present context.

There are two potential hazards presented by these volatile fission products: 1) if the gas cannot escape the fuel volume, pressure buildup and emitter distortion are long-term possibilities, and 2) if fission products should enter the Cs space via a cracked emitter or insulator seal, contamination of the emitter is possible. The potential problems associated with gas pressure buildup can be reduced by providing a large, evacuated volume to collect the gaseous fission products. Due to space constraints, this fission product volume must be some distance removed from the fuel space

Figure 7-3 - Cesium pressure vs. loading for four carbon temperatures.
The filled triangles are the loading points for the present sample (#202) and the UCA-1 samples (B,E).

Figure 7-4 - Pressure versus temperature at four specific loadings

TABLE 7-2
YIELD OF STABLE AND LONG-LIVED NUCLIDES FROM
THERMAL NEUTRON FISSION OF U-235*

Element	Atomic Weight of Nuclide	Percent Yield	Element	Atomic Weight of Nuclide	Percent Yield
Ge	72,73	0.000126	Sb	121,123	0.0163
Se	77,78,79	0.0853	Te	125,126,128,130	2.441
Br	81	0.14	I	127,129	1.03
Kr	83,84,85,86	3.857	Xe	131,132,134,136	21.83
Rb	85,87	3.50	Cs	133,135,137	19.15
Sr	88,90	9.34	Ba	138	5.74
Y	89	4.79	La	139	6.55
Zr	91,92,93,94,96	31.05	Ce	140,142	12.39
Mo	95,97,98,100	24.44	Pr	141	6.0
Tc	99	6.06	Nd	143,144,145,146,148,150	21.04
Ru	101,102,104	10.9	Sm	147,149,151,152,154	4.322
Rh	103	3.0	Eu	153	0.15
Pd	105,106,107	1.47	Gd	155,156,157	0.0538
Ag	109	0.030	Cd	158	0.002
Cd	111,112	0.029	Tb	159	0.00107
In	115	0.0104	Dy	161	0.00107
Sn	117	0.011			

* Reprinted from **Nucleonics**, Vol. 16, No. 4, p. 78 ff, April 1958.

and be connected to it by tubes and channels, which will be at a temperature in the vicinity of the collector temperature ($\sim 1000^{\circ}\text{K}$). It is possible that fission product condensation may occur in these tubes and even plug them, allowing the pressure in the fuel space to rise. This possibility could be alleviated by tailoring the local tube cross section according to the local temperature and anticipated flow conditions so that the local fission product pressure is less than its vapor pressure, thus preventing condensation.

A more commonly proposed method of eliminating the possibility of tube plugging, and which also is effective in reducing converter contamination in the event of a leak between the fission product space and the Cs space, is to locate a "filter" or fission product trap directly above the fuel. Before proceeding to a more detailed discussion of fission product

trapping, some observations made during the pre-1973 GA thermionics program are in order (Refs. 7-4, 7-5, 7-6).

Pre-1973 results. Postirradiation examinations (PIE) have been previously conducted on several converters and multi-converter TFEs using both UC and UO_2 fuel. The most extensive information is available for the carbide fueled, two cell TFE 2E3, the oxide fueled converter I4, and the oxide fueled, 2 cell TFE 2E1 (Refs. 7-4 and 7-6).

For the $\text{UC}_{1.03}$ fueled TFE 2E3, the fission product vent in the emitter transition piece of one cell was plugged. The fission product paths to the other cell formed by grooves in the collector were also plugged. There were also surface deposits on both the Al_2O_3 and the Nb surfaces in the vent tubes. PIE radiochemical analysis indicated that the plugs consisted of ~95% Sr and the remainder Cs. The surface deposits, on the other hand, had less than 50% Sr and contained Er and Ce as well as Cs. No fission product gases were found in the fission product volume outside the core area, indicating complete tube blockage.

Contrasting results were obtained with the $\text{UO}_{2.004}$ fueled TFE 2E1, which had the same geometry as 2E3 and operated in-core for over 3 times as long. In this case, no Sr was detected and 96% of the fission product gas was collected in the fission product volume. The vent tubes in one cell were filled with a porous material containing Cs and Te with a small amount of Ba. There were also some surface deposits of this composition.

The in-core operating history of the F series TFEs affords some indication of the severity of cell degradation when communication accidentally occurs between the Cs and fission product spaces. For TFE's 2F2, 6F2, 2F3 and 6F3, leaks developed between these two spaces in the form of either a cracked insulator seal or emitter. The first two listed above had carbide fuel and suffered a loss in performance when the leak occurred. The last two had oxide fuel and were unaffected by the leak (Ref. 7-6).

These pre-1973 results with the E and F series TFEs raises the possibility that the fission products escaping from an oxide fuel at ~1800 K are

somewhat benign in terms of plugging up the fission gas handling pathway and contaminating the emitter. Experience with the H series TFEs of the present TFE Verification Program will clarify the issue, however, the magnitude of the fission product management problem in TFEs is somewhat diminished by these earlier results.

Fission product trapping. This discussion will be limited to the selection of a suitable fission product trapping material. The mechanical design of a fission product trap will be given at a later time but certain features will inevitably be incorporated: the trap will fit into the emitter transition region above the fuel; its construction must also accommodate heat shields and a fuel holddown spring; the gas path through the trap must be such as to maximize collision of fission products with the trap material.

A suitable material for a trapping medium is one that removes volatile fission products other than rare gases, from the flow before they leave the emitter stem region. The trapping medium must react with, or promote reaction between, fission products at a temperature of about 1000°K leading to nonvolatile products. Even a qualitative model that attempts to describe the chemistry occurring in the trap will be complex. However the question of trap chemistry is preceded by an even more complex system, i.e., the chemistry leading to products in the fuel matrix that are volatile at 1800°K and, therefore, have the opportunity to appear as reactants at the fission product trap.

One approach to determining the volatile fission products that are present is to consider only the oxides of the metals shown in Table 7-2. The next step is to rank these metal oxides on the basis of their free energy of formation, ΔG_f° , with the largest negative values appearing first. The amount of oxygen available is assumed to be 2 atoms per fission and will, according to this approach, be consumed to exhaustion in the sequence of decreasing negative ΔG_f° . Using values of ΔG_f° at 1800°K from Refs. 7-7 and 7-8 and data on the dependence of ΔG_o^f for UO_{2+x} upon x (Ref. 7-9), it is found that the daughter nuclides are oxidized in the sequence

given in Table 7-3. The last column of this table gives the vapor pressure of the oxide or metal as appropriate. Thus, on the basis of this greatly oversimplified model, the fission products evolving from the fuel will be BaO and the metals Rb, Pd, Cs, Te. It is not clear if the rare earth oxides will be volatile at 1800°K. There will be no gaseous oxygen emanating from the fuel in this model.

The analysis underlying the results depicted in Table 7-3 are approximate. First, as mentioned above, the only reactions considered were those between the individual fission product nuclides and oxygen; reactions occurring between fission product nuclides themselves were not considered. As an example there are several possible reactions between the alkaline earths Sr and Ba and the halides Br and I that are products of U-235 fission (Table 7-2). As a specific case, consider the nuclide Sr, with a fission yield of 9.34%. Table 7-3 considers only the reaction whereas the

formation of two alkaline earth halides may also occur according to

and

From the values of ΔG° given above (Ref. 7-8) it is seen that the formation of Sr Br₂ is thermodynamically favored over either SrO or SrI₂. Table 7-3, for the model considering only oxides, indicates that SrO is only very slightly volatile. Its calculated vapor pressure at 1800° K is ~10⁻⁶ torr. The vapor pressure of Sr Br₂ at these same temperatures is calculated as 10 torr. Thus the bromine reaction channel provides a mechanism for removing Sr from the fuel that was not available in the simple model. Similar results are obtained with the barium halides.

TABLE 7-3
OXIDATION OF STABLE NUCLIDES IN FUEL BASED ON FREE ENERGIES OF
FORMATION FOR $M + O_2 \rightarrow MO_2$

Element	Fission Yield	Forms Oxide	Vapor Pressure of the Oxide or Element as Appropriate at 1800°K (Torr)
La	7	Yes	
Sr	9	Yes	2×10^{-6}
Ba	6	Yes	2×10^{-2}
Pr	6	Yes	
Nd	21	Yes	
Sm	4	Yes	
Y	5	Yes	
Zr	31	Yes	6×10^{-11}
Ce	12	Yes	
Mo	24	Yes	10^{-3} (MoO_2)
Tc	6	No	10^{-9}
Rb	4	No	> atmos
Ru	11	No	10^{-8}
Rh	3	No	10^{-5}
Pd	2	No	10^{-2}
Cs	19	No	> atmos
Te	2	No	> atmos

Another concern of the simple oxide model [and also the halide estimates of reactions 1) and 2)] results from ignoring the pressure of the gases in the reaction. For example the free energy of reaction for 1) above is

$$\Delta G = \Delta G^\circ - RT \ln P(Br_2) \dots \dots \quad (3)$$

with

$$\Delta G^\circ = \Delta G_f^\circ (SrBr_2) \dots \dots \dots \quad (4)$$

and $\Delta G < 0$ favors formation of the bromide product. If atmospheric

pressure Br_2 is assumed then $\Delta G = \Delta G^\circ = \Delta G_f^\circ (\text{SrBr}_2) = -507 \text{ kJ/mole}$, from the JANAF tables (Ref. 7-8). In reality the bromine pressure will be less than atmospheric (as will the oxygen pressure for the reactions considered in constructing Table 7-3) and $\Delta G > \Delta G^\circ$ will result. Since these pressures are not known or calculated, the logarithmic pressure term is ignored throughout this document. The application of equations (3) and (4) and the computation leading to Table 7-3 further assumes that the solids are homogeneous. The situation is more complicated if, for example, the solids are in solid solution in the fuel.

Materials for the fission product trap. As inferred from the previous section, there is uncertainty in the composition of the fission products emanating from the fuel. In addition, the choice of plasma-sprayed Al_2O_3 on Mo discs as the trapping medium in H series TFEs appears prudent as a first estimate. This arises from pre-1973 experience with the F series TFEs and also laboratory experiments conducted at that time (Ref. 7-10). Given these facts, the present discussion will be confined to the special class of reactions that might occur between Al_2O_3 (or MoO_2) at 1000°K and elemental fission products, their oxides and halides. The goal is to indicate those types of reactions that are thermodynamically favored at equilibrium and not make quantitative estimates of "trapping efficiency".

There are two general types of interactions to consider: "physical" trapping, leading to condensation, and chemical reaction.

Generally speaking, the fission product oxides have a very low vapor pressure at 1000°K ($<10^{-15}$ atm for BaO , for example). Condensation of the oxides in the trap is therefore to be anticipated. However, since the vapor pressure is also low at the fuel temperature of 1800°K ($<10^{-5}$ atm for BaO), these oxides are not likely to escape the fuel and be collected. The halides have a vapor pressure in the range of 10^{-8} atm at 1000°K and may also be expected to condense. Unlike the oxides, the halides will escape from the fuel readily as their vapor pressure is in the range of 10^{-2} atm at 1800°K . Thus appreciable condensation of bromides might be expected. Most of the elemental fission products that can escape the 1800°K fuel have too large a vapor pressure at 1000°K for condensation to occur.

For fission product oxides reacting with Al_2O_3 or MoO_2 there is the possibility of new phases being produced of the form

If there is excess oxygen available, the corresponding reactions with MoO_2 are:

In equilibrium, at 300°K , reactions (5) through (7) proceed to the right as shown (Ref. 7-11). Thermodynamic data for $\text{Eq. SrAl}_2\text{O}_4$ at 1000°K has not been found, so it is not possible at this time to state that these reactions could occur in a fission product trap at 1000°K , although experimental data from the pre-1973 GA program indicates that they do (Ref. 7-10).

This earlier experiment was conducted with a series of plasma-sprayed Al_2O_3 coated Mo baffles which were exposed to a flow of Sr atoms in vacuum. The baffle temperature was held at 1350°K for ~16 hours with Sr flow rate of $\sim 10^{13}$ atoms/sec entering the baffle chamber. Post-mortem studies indicated that about 50% of the Sr atoms were trapped on the Al_2O_3 . X-ray diffraction suggested the presence of the $\text{SrO} \cdot \text{Al}_2\text{O}_3$ phase. Metallography indicated that this phase extended 20-40 μ m into the Al_2O_3 .

To explain these earlier experimental results, reaction (5) is inappropriate since the impinging species in the experiment was Sr, not SrO. However, if excess oxygen were present, the reaction

could proceed (again using a room temperature value for ΔG ($\text{SrO} \cdot \text{Al}_2\text{O}_3$)). Note that analogous reactions also occur for the fission product oxides on MoO_2 . The formation of aluminates and molybdates according to reactions (5) through (8) constitutes an effective trapping mechanism only if the

vapor pressure of the new compound is low. Adequate data is not available, however, at 1350°K, $\text{Sr Al}_2\text{O}_3$ is apparently nonvolatile, as evidenced by the experiments mentioned above.

From reaction (8), it is seen that the aluminate reaction for elemental Sr does not proceed if excess oxygen (gaseous or adsorbed on the alumina) is not available. The straight oxidation reaction

is very slightly exothermic and may proceed; for Ba, this reaction is endothermic. For a MoO_2 trap, reactions of the type (9) are thermodynamically favorable for both Sr and Ba. It should be recalled, however, that elemental fission product nuclides are unlikely evolvents from the fuel.

Fission product bromides and iodides that could form in the fuel as discussed earlier do not form compounds analogous to reactions (5) to (7), based on available data. Furthermore, direct oxidation reactions such as (9) are endothermic for these halides.

Summary. In summary, the trapping mechanisms for postulated fission products from UO_{2+x} fuel at 1800°K, flowing over plasma sprayed alumina on molybdenum disks maintained at 1000°K, are likely to be the following:

- 1) Elemental fission products:
  - a) Formation of aluminates and molybdates if excess oxygen available.
  - b) Slight possibility of conversion to oxide in some instances.
- 2) Fission product oxides:
  - a) Conversion to aluminates and molybdates.
- 3) Fission product bromides:
  - a) Condensation, but not confined to Al_2O_3 or MoO_2 ; on encapsulation of trap and surroundings also.

7.4 Cs RESERVOIR MATERIALS SUMMARY

Due to its excessive sensitivity of pressure to loading, PC-113 has been dropped from consideration as a reservoir material. Although neutron induced failure of three HOPG reservoirs has been observed in UCA-1/2, this material remains in consideration pending additional tests in EBR-II. DFP-1 was dropped due to its similarity to CZR-2. These changes are contained in the updated overall evaluation of candidate reservoir materials, Table 7-4, which shows that four materials remain: two graphite, one ceramic and one refractory metal.

References

- 7-1 Thermionic Fuel Element Semiannual Report for the Period Ending October 31, 1988, GA-A19412.
- 7-2 Thermionic Fuel Element Semiannual Report for the Period Ending September 30, 1987, GA-A19115.
- 7-3 Thermionic Fuel Element Semiannual Report for the Period Ending April 30, 1988, GA-A19269.
- 7-4 Gulf GA-A11073, July 1972 (Thirdly Summary Report).
- 7-5 Gulf GA-A12251, January 1973 (Thirdly Summary Report).
- 7-6 Gulf GA-A12608, June 1973 (Final Summary Report).
- 7-7 "The Thermochemical Properties of the Oxides, Flourides and Chlorides up to 2500°K", A. Glassner, Argonne National Laboratory Report ANL-5750, 1956.
- 7-8 JANAF Thermochemical Tables, J. Physical and Chemical Reference Data, Vol. 14, Supplement 1, Parts I and II, 1985.
- 7-9 T. B. Lindemer and T. M. Besmann, J. Nucl. Mat. 130, 473 (1985).
- 7-10 Gulf-GA-A12397, June 1973 (Thirdly Summary Report).
- 7-11 "The NBS Tables of Chemical Thermodynamic Properties", J. Physical and Chemical Reference Data, Vol. II, Supplement 2, 1982.

TABLE 7-4

OVERALL EVALUATION OF CANDIDATE RESERVOIR MATERIALS AFTER DOWN SELECTION
 OK=MEETS DESIGN CRITERIA; P=MEASUREMENT PLANNED; IP=MEASUREMENT IN PROGRESS

Material	Interim Rank, 4/88	Specific Cs Loading	Cs P-T Reproducibility /Stability	Dimensional Stability in Neutron Flux		Impurities				
		From BET Area	From P-L Test	BOL(Ex-reactor)	EOL (UCA PIE)	Inferred From Data Base	Meas-ured (UCA PIE)	Bulk Anal-ysis	Out-gas-sing Test	Chemical Compati-bility
<u>Graphite</u>										
HOPG-ZYH	(2)	OK	OK	OK	(Interim Fail) ¹	OK	(Interim Fail) ¹	OK	OK	OK
POCO CZR-2	1	OK	OK	OK	(Interim OK) ²	OK	(Interim OK) ²	OK	OK	OK
<u>Ceramic</u>										
Activated Alumina, AA-400(ALCAN)	3	OK	P	P	P	(OK)	P	OK	OK	OK
<u>Refractory</u>										
Sintered Mo (Rhenium Alloys, Inc.)	4	OK	P	P	P	(OK)	P	OK	OK	OK

¹ Failure in UCA-1/2, but retest planned in UFAC Batch-2; see Sec. 7.3.2.

² Conditional success in UCA-2, retest planned in UFAC Batch-2; see Sec. 7.3.1, 7.3.2.

8. THERMIONIC FUEL ELEMENT

8.1 OBJECTIVE

The overall goal of the thermionic fuel element task is to demonstrate that TFEs prototypic of a 2 MW(e) thermionic space nuclear power system of seven year life can be fabricated from well modeled components, and that their performance is as predicted when operated in a prototypic thermionic reactor environment. Derivative goals include:

- 1) Produce a TFE engineering design and specifications.
- 2) Develop required TFE assembly processes, process specifications, and demonstrate manufacturing capability.
- 3) Fabricate TFEs and test them to demonstrate the processes and the integrated performance of the components.
- 4) Develop and verify a TFE model that can predict TFE performance and lifetime.

Components with demonstrated performance and which are projected to meet lifetime requirements will be used in these TFEs. Most of the TFEs will undergo irradiation in the TRIGA reactor. The performance of one prototypic TFE will be demonstrated in the FFTF.

8.2 TASK DESCRIPTION

8.2.1 Testing Logic

All TFEs tested will be as close to the baseline TFE design in geometry and performance as practicable. Test and design requirements are shown Table 8-1.

Six TFE tests are planned for the TRIGA reactor before a demonstration in the fast-neutron spectrum FFTF. These are outlined in Table 8-2.

TABLE 8-1
COMPARISON OF TFE TEST REQUIREMENTS WITH SYSTEM
BASELINE DESIGN REQUIREMENTS

	Test Requirements (BOL)	Baseline Design Requirements (BOL)
Cell power density average (W/cm^2)	3.2	3.2
Current density average (A/cm^2)	7	7
Emitter temperature average (K)	1800	1800
Collector temperature average (K)	1070	1070
Sheath-collector voltage maximum (V)	± 7.5	± 7.5
TFE geometry	Baseline	Baseline
TFE materials	Baseline or variants for performance/ structural improvements	Section 2.2
Fuel enrichment (% U-235)	Variable enrichment	93; variable fuel volume fraction
Fast fluence: nominal ($E > 0.1$ Mev)	Real time	$2.7 \times 10^{22} / 7$ years
Fuel burnup: nominal (a/o)	4.1	4.1

TABLE 8-2
TFE IN-REACTOR TEST MATRIX

TFE ID	Fuel/Emitter Collector	Cs Reservoir	Purpose	Reactor
1H1	UO ₂ /W/Nb	Pool	Verification of reference cell design performance and fabrication processes. Sheath integration.	TRIGA
1H2	UO ₂ /W/Nb	Integral	Backup to 1H1. Verification of integral cesium reservoir design, performance.	TRIGA
1H3	UO ₂ /W/Nb	Pool	Study effect of fission products mixing with cesium.	TRIGA
3H1	UO ₂ /W/Nb	Integral	Backup to 3H1. Introduction of prototypic components outside cell.	TRIGA
6H1	UO ₂ /W/Nb	Integral	Verification of long-TFE fabricability. Consider evolutionary improvements. Verification of fission gas ventings. Increased voltages.	TRIGA
6H2	UO ₂ /W/Nb	Integral	Backup to 6H1.	TRIGA
6H3	UO ₂ /W/Nb	Integral	Verification of TFE performance in fast reactor environment.	FFTF

The logic behind this test series is a step-by-step development of the fabrication processes, culminating in the fast reactor prototype. The first step is the development of the processes for a single cell, and the integration of the single cell into a TFE sheath tube with appropriate end fittings. These end fittings involve the cesium reservoir, fission gas venting and the conduction of electrical current produced. This single-cell TFE is designated "1H", where the H refers to the latest generation of thermionic cell in a megawatt class TFE, and the "1" refers to the number of cells in the TFE.

The second step is the development of the intercell region of the TFE. The test vehicle will be the 3H-series TFE which will contain three of the H-series cells welded end-to-end in an electrical series circuit. The center cell in the series is isolated from TFE end-fittings and is thereby typical of cells within the interior of a thermionic reactor. The testing of the three cell TFE also allows the study of fission gas venting in a representative multicell environment at a minimum expense.

As the last step in the TRIGA testing, 6H-series TFEs will be built to demonstrate the fabricability of a long TFE where axial alignment is critical. Testing of these TFEs will provide additional demonstration of fission product venting and the capability to maintain unobstructed fission gas passages.

Success in the process development will be evidenced by the observed in-reactor performance and stability.

8.2.2 TRIGA Facility

The Mark F TRIGA reactor will be used for TFE testing. This includes a neutron radiography facility located within the reactor pool for periodic nondestructive diagnostic examinations of the TFE internals.

The TRIGA typically operates at about 1.4 MW(t). At this power level, the fast fluence is about 1×10^{21} nvt ($E > .1$ Mev) per year.

8.2.3 Design Description

A description of the TFE design was presented in the last semiannual report, Ref. 8-1. The component matrix defining the differences between the TFEs listed on Table 8-2 are shown on Table 8-3.*

TABLE 8-3
TFE TEST MATRIX

	<u>1H1</u>	<u>1H2</u>	<u>1H3</u>	<u>3H1</u>	<u>3H5</u>	<u>6H1</u>	<u>6H2</u>	<u>6H3</u>
Number of converters	1	1	1	3	3	6	6	6
Alignment scheme	Old	Old	Old	New	New	—————→		
Fission product trap in emitter	No	No	Interim	Interim	Optimize	—————→		
Mixed Cs and fission product space	No	No	Yes	No	No	—————→		
Converter trilayer	Y ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃	Y ₂ O ₃	—————→		
Converter seal	Litton	Litton	Litton	Litton	Litton	—————→		
Cesium reservoir	Liquid	Graphite	Liquid	Graphite	Graphite	—————→		
Emitter transition material	Ta —————→						Nb/Ta —→	
Fuel holddown device	Spring	—————→				Flight device	—————→	

8.3 TFE-1H1 STARTUP AND OPERATION

TFE-1H1 was charged into the TRIGA reactor in early September, 1988 and the rise-to-power occurred on September 15, 1988. As of March 1, 1989 about 4000 hours of operation on 1H1 had been accumulated. Performance has been excellent.

Secondary containment. An unusual incident occurred while making the final weld on the 1H1 secondary containment tube. Resistance checks just

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*This updates a similar matrix shown in Table 8-6 of Ref. 8-1.

before and after the weld was made showed a drop in the sheath insulator voltage probe to emitter bus and current probe to emitter bus resistances. It could not be positively determined whether the changes in resistances were caused by spurious arcing, the rotation of the capsule during welding, or some other reason. A modified welding procedure will be developed and tested prior to future operations.

The objective of the current probe and the voltage probe was to include in the 1H-TFE series the capability to apply a higher potential across the sheath insulator than is generated naturally by the TFE (~0.5 V). This test was considered a low priority test to be done towards the end of the irradiation period. After reviewing the available data, it was determined that the probable location of the change in resistance was within the primary containment. It was decided to proceed with the test and forego the capability for the high voltage sheath insulator test.

Rise-to-power. During the rise-to-power it became evident that the 1H1 collector was heating faster than the cesium reservoir. Consequently a significant amount of electrical power had to be supplied to the cesium reservoir heater to maintain the required cesium temperature of 600 K. At about 800 kW (TRIGA power) the collector temperature reached its design value of 1070 (without any external electric power to the collector heater), and the cesium temperature was about 590 with full power to the reservoir heater. A hold was called in the rise-to-power to change the secondary 50% helium/50% argon mixture to one higher in helium to cool the collector. As the gas change was initiated the collector temperature began to drop slowly. However, the cesium temperature began to drop rapidly and cesium was lost in the interelectrode gap. With the loss of electron cooling the emitter temperature rose to about 2100 K, and the collector temperature climbed up to about 1300 K. The secondary was completely evacuated which restored the cesium pressure. As the cesium reservoir temperature increased, ignition occurred again. Examination of the unignited and ignited mode I-V curves showed that some change had occurred in them, probably due to gas contamination.

A procedure was established that no "on-line" changes would be made to the secondary gas mixture. Instead the reactor would be brought to about 500 kW, at which point there would be no electrical cooling current to the emitter and a loss of cesium would not effect the cell.

A loss of site power shortly after the above incident scrambled the reactor.

The next day, the secondary containment was filled with a 50% helium/50% argon mixture, and the reactor brought slowly up to power until a collector temperature of about 1070 K was reached. At this point the reactor was secured and the test article monitored for stability. The operating conditions for TFE-1H1 were as follows:

Emitter temperature	1780 K
Collector temperature	1112 K
Cesium reservoir temperature	601 K
Current	137 amps
Converter lead voltage	0.45 v
Secondary gas mixture	50% He/50% Ar

The converter lead voltage corrected to the design collector temperature (1070 K) is 0.52 v.

Diagnostic software. The new HP-916 computer system dedicated to 1H1 diagnostics was installed at the TRIGA facility with the software package developed by Rasor Associates, Inc. During 1H1 startup and operation, the data acquisition system and the diagnostics software functioned as designed.

Since a thermocouple does not exist on the 1H1 emitter, the diagnostics software package enables fast correlation of 1H1 operation with out-of-core thermionics test data to provide an estimate of emitter temperature. An I-V sweep is taken of 1H1 and the data is transferred digitally from an oscilloscope to the diagnostics computer for display. An operator can then input all known operating parameters (i.e., collector

temperature, cesium reservoir temperature, etc.) as well as guesses for emitter temperature. The software package plots a characteristic thermionic curve for the ignited or unignited mode based on the estimated emitter temperature. This is repeated until the characteristic curve of the estimated temperature corresponds to the I-V sweep. Figure 8-1 shows a match of the 1H1 unignited mode with a characteristic unignited mode curve for a thermionic device operating at 1805 K. Periodic I-V sweeps indicate that the emitter operates very steadily near 1800 K.

Subsequent operation. During December, there were two interruptions to the irradiation of TFE-1H1. On December 17, 1988 the TRIGA reactor was inadvertently scrambled. A portable air monitor was tested using the TRIGA uninterruptible power supply (UPS). The monitor momentarily overloaded the UPS causing the scram. The reactor was returned to full power the same day without incident.

On December 22 the TRIGA reactor was shutdown for the holiday vacation. The shutdown followed a scheduled reduction in power down to 500 kW at 10 kW/min. Thermionic parameters were continuously monitored and emitter temperature was maintained near 1800 K. The reactor power reduction rate was then increased to 25 kW/min until final shutdown.

During this shutdown, the TRIGA Mark F core was reconfigured: several of the fuel and dummy elements were shifted to flatten the core power profile; dummy rods were inserted in the locations that will eventually be occupied by 1H2 and 1H3; and an incore power detector was located closer to 1H1 for better power calibration. Restart of the reactor occurred on January 5, 1989.

Output voltage of TFE-1H1 declined slightly in the last half of January but performance was relatively stable at the month's end. Estimated converter output power changed about 6%, from 3.3 W/cm^2 to 3.1 W/cm^2 . The emitter temperature was down 25 K from 1800 K, while other temperatures appear unchanged, and reactor power has been held essentially constant. A tentative conclusion was that the output voltage change was caused by a partial short within the TFE between the emitter and collector

Thermionic Data

Emitter:	1805.00 K
Collector:	1109.00 K
Cesium Reservoir:	501.00 K
Gap:	10.00 mils
Phi-efficiency:	4.78 volt
Phi-C:	1.95 volt

Figure 8-1 - Emitter temperature determination technique

sheath insulators. The converter current density has changed from 7.0 amps/cm² to 7.35 amps/cm².

On March 1, 1989 the line fuse blew on the power supply unit for the 1H1 cesium reservoir and stem heaters, resulting in a reactor operator alarm on low reservoir temperature and, immediately following, an automatic scram of the TRIGA test reactor on low TFE current. The reactor was brought back to full power that day without incident. Of major significance was the absence upon restart of the partial internal short previously detected in January. 1H1 operation returned to the normal output exhibited during the period from initial startup in mid-September to the onset of the apparent short in mid-January.

References

- 8-1 TFE Verification Program Semiannual Report for the Period Ending October 31, 1988, GA-A19412 (issued January, 1989).