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DTIC XX

ATOMICS INTERNATIONAL A Division of North American Aviation, Inc		NAA-SR TDR NO 7872	APPROVALS <i>HA Moreau</i> <i>M. Silberberg</i>
TECHNICAL DATA RECORD		PAGE 1 OF 23	
AUTHOR M. Silberberg & P. Spiegler		DEPT & GROUP NO 713-31	DATE 10/17/62
TITLE Planning for Uranium Carbide Transient Heating Experiments in TREAT.		GO NO 7519	RECOMMENDED FOR OUTSIDE DISTRIBUTION ☐ YES ☒ NO <i>HA Moreau</i> SIGNATURE
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PROGRAM		TWR	
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		ABSTRACT The initial experimental program to study the transient heating of uranium carbide fuel in TREAT is outlined. Preliminary analyses are presented for the behavior of an HNPF prototype UC fuel slug during power bursts in the TREAT reactor. The analysis and evaluation conducted, center around the proposed utilization of the existing GE capsule, and indicate that this equipment will be suitable for the initial uranium carbide experiments with minor modification. A survey is presented and found not to be significantly different from the GE-UO₂ experiments. A planning schedule for the program is shown, and in addition, recommendations for additional studies which should be conducted are presented.	
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II. OBJECTIVES OF INITIAL EXPERIMENTAL PROGRAM

The initial experimental program for the project entitled "Fuel Element Transient Heating" (SCR) covers that portion of the effort which as originally proposed, is to be accomplished during FY 1963. The planning of this initial program is predicated on the utilization of existing hardware technology to accelerate in-pile transient heating experiments with prototype HNPF uranium carbide fuel. The need for early information on the performance of carbide fuel during power transients is related to its use in the Hallam second core loading. Nuclear transient heating experiments for the program will be conducted in the TREAT reactor at NRTS in Idaho. An in-pile capsule design and technique developed by GE-APED (San Jose) for TREAT experiments with UO₂-fast ceramic reactor fuel represents existing hardware which could be utilized for the initial experimental program. In subsequent sections of this report, the suitability of the GE capsule design is evaluated for our experiments by means of a preliminary analysis of carbide fuel behavior during TREAT power bursts.

Two individual capsule insertions in TREAT are planned for the initial program. In the first insertion, a prototype HNPF uranium carbide fuel slug (S.S. clad, sodium bonded slug - 6" length) will be subjected to several power bursts such that the maximum fuel temperature is distinctly below the melting point of the fuel. This test is expected to yield the following information:

1. Cladding deformation or failure before fuel melting as a result of high thermal stress.
2. Transient temperature profiles which can be compared with those predicted analytically.
3. Data on transient heat transfer such as importance of radiative heat transfer.
4. Changes in fuel structure associated with non-melting power transients.

In the second capsule insertion, the fuel specimen will be subjected to a power burst of sufficient energy release to melt a substantial portion of the fuel during the transient. Information which may be ob-

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tained from this test includes:

1. Gross structural damage and changes associated with fuel melting.
2. Nature of fuel distribution associated with fuel melting and subsequent resolidification.
3. Transient heat transfer data characterized by fuel melting and cladding failure, including an estimate of the heat of fusion of uranium carbide.
4. Transient pressure developed in sodium bond prior to cladding failure.
5. Inertial pressures, if any, generated by the disintegrating fuel rod.
6. Fission product release from UC fuel to sodium coolant (this will be evaluated by the "Fission Product Retention by Reactor Coolants" Project as part of their project objectives).

Post-irradiation examination of the capsules utilized in the above tests will be performed in the Component Development Hot Cell at Santa Susana.

Information obtained from the initial experimental program will be evaluated for significance with respect to the hazards summary for the Hallam second core loading with UC fuel, and as such, constitutes a major objective. Additionally, the results obtained from the initial effort will provide guidance for future project planning as well as advanced design of experimental apparatus.

It is recognized that transient heating of uranium carbide fuel having substantial irradiation history may yield significant results associated with accumulated fission gases in the fuel. A transient experiment which utilized a fuel specimen previously irradiated in one of the test reactors (e.g. MTR, G ETR), will be included in the second phase of the experimental program sometime during FY 1964.

III. EXPERIMENTAL APPARATUS

A. Description of G.E. - TREAT In-Pile Capsule and Anticipated Changes for HNPf UC Fuel Slug

A drawing of the G.E. capsule is shown in Figure 1. The cap-

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capsule was designed to contain a prototype fuel rod for the Experimental Fast Ceramic Reactor. This fuel element consists of a Type 304 S.S., annealed, cylindrical tube which is 14 inches long, has an outer diameter of 1 inch, and a tube wall thickness of 32 mils. The tube contains UO_2 fuel pellets (3.9% enriched) which are separated by a 14-mil diameter gap from the tube wall. The end of the tube is sealed with a welded plug. Major design specifications for the capsule outside of being able to accommodate the fuel element are:

- 1) It must fit a dummy TREAT fuel element in order to be inserted in the core.
- 2) It must contain a heat sink capable of absorbing all the heat generated in the fuel during a transient without rising to temperatures above those expected in TREAT.
- 3) It must be sufficiently insulated from its surroundings to permit its use as a calorimeter.
- 4) It must withstand internal pressures generated in the course of planned experiments or in a possible accident situation.
- 5) The capsule must provide containment for fission products as well as sodium or other materials which might cause chemical reaction in the TREAT core.

As can be seen from Figure 1, these specifications were met by building two capsules, one located inside of the other. The outer capsule is just a dummy TREAT fuel element while the inner capsule meets the other specifications.

The G.E. capsule is equipped with temperature instrumentation consisting of a total of 11 thermocouples located at several external points, internally, at various radial positions, and in the case of the beryllium heat sink at five longitudinal positions on the outer surface. All thermocouples are chromel-alumel with the exception of the central fuel thermocouple which was originally Pt/Pt-10% Rh, but, in subsequent tests a rhenium-tungsten thermocouple (good to about 5000°F) has been used. The central fuel thermocouple is installed through the upper end plug of the specimen and is used to correlate reactor power with fuel tempera-

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ture in a series of preliminary low energy release transients. A heater element which is wrapped around the exterior of the heat sink and thermally bonded to it with sodium is provided for preheating the contents of the inner capsule to about 800°F. A 5/16" layer of Thermoflex insulation is placed between the inner capsule and the outer capsule shell to minimize heat losses during the approach to equilibrium temperature following the transient, in order to maintain the capability of the heat sink as a calorimeter.

The prototype HNPF fuel which is of interest for the initial UC experiments is a 0.892 inch diameter UC slug, 6 inches in length. The slug would be clad using 10-mil wall, 0.95 inch O.D., 304 S.S. tubing, with a 0.020-inch annulus for the sodium bond. The maximum diameter of the UC fuel specimen assembly is thus compatible with the corresponding dimension of the G.E. specimen. The overall length (6") of the UC element which is contemplated, is considerably shorter than the G.E. specimen. This difference in length can be accommodated by appropriate modification of the upper positioning tube which holds the fuel element in place. In this case, an additional volume of about 100 cc. of sodium coolant would be added to the capsule. In this adaption the capsule design is essentially unchanged. Another possible method for adapting the capsule to the shorter fuel length would be to shorten the entire inner capsule. This modification, however, greatly reduces the heat capacity of the inner capsule, with the result that its ability to act as a heat sink becomes questionable.

In their early experiments, G.E. did not include a pressure transducer as part of the capsule instrumentation, however, a transducer has been incorporated in recent experiments.⁽¹²⁾ The transducer being employed is apparently capable of operation at 1200°F. As will be shown later in the report, it is important that a suitable pressure transducer be included in the initial experiments. The transducer should be located so as to monitor the pressure within the clad fuel slug. Since G.E. monitors pressure in a similar location, the method of installation should not present a problem.

In the G.E. experiments, the space above the sodium coolant in the

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inner capsule is pressurized initially with 10 atm. of helium as a cover gas. In our experiments we may choose to operate with a lower initial pressure, however, for the present analysis, we will assume this pressure will be the same.

B. Experimental UC Fuel Specimen Data

The pertinent information on the experimental UC fuel specimens which is required for the preliminary analyses in this report has been compiled^(11,13) and computed. This information is summarized below:

Fuel material: uranium monocarbide.
Uranium content: 95.35 wt. percent.
Carbon content: 4.65 wt. percent.
Fuel enrichment: 4.9 and 3.7 percent.
Fuel density: 11.4 gram/cc.
Fuel slug diameter: 0.892 inches.
Fuel slug length: 6 inches.
Fuel slug volume: 3.75 in³.
Fuel slug mass: 700 grams (1.54 lbs.).
Cladding I.D: 0.932 inches.
Cladding thickness: 0.010 inches.
Cladding volume: 0.176 in³.
Cladding mass: 22 grams (0.048 lbs.).
Sodium bond thickness: 0.020 inches.
Sodium bond volume: 0.33 in³.

	4.9% enr.*	3.7% enr.*
Mass of U-235 in slug, g.	32.7	24.7
Mass of U-238 in slug, g.	634	642
U-235 atoms/cc	13.6×10^{20}	10.3×10^{20}
Fission cross-section, cm ⁻¹	0.689	0.520

*Enrichments available for HNPF 2nd Core Loading.

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IV. TRANSIENT ENERGY RELEASE IN A UC FUEL SLUG IN TREAT

The following relationship suggested by J. Hume⁽¹⁾ was used to estimate the energy release in the fuel slug.

$$\begin{aligned} \text{Energy release rate on fuel slug per cc} &= \left[\frac{\sum_f \text{fuel slug}}{\sum_f \text{TREAT}} \right] \times \left[\text{slug disadvantage factor} \right] \times \\ &\quad \left[\text{horizontal peak flux to average flux} \right] \times \left[\text{average energy release in TREAT per cm}^3 \right] \end{aligned} \quad (1)$$

The relation assumes that the fuel slug is located at the center of the TREAT core. The following parameters were used:

$$\begin{aligned} \sum_f (\text{fuel slug } 4.9\%) &= 0.689 \text{ cm}^{-1} \\ \sum_f (\text{fuel slug } 3.7\%) &= 0.520 \text{ cm}^{-1} \end{aligned}$$

These values were calculated assuming a UC density of 11.4 gm/cm^3 and a slug disadvantage factor of 0.35. This is the disadvantage factor for a fuel element in the HNPF reactor⁽²⁾ and is believed to be adequate for these calculations. Further, using values obtained from Reference 1 for

$$\sum_f (\text{TREAT}) = 3.82 \times 10^{-2} \text{ cm}^{-1}$$

Horizontal peak flux to average flux ratio. = 1.28; Volume of TREAT = $2.83 \times 10^6 \text{ cm}^3$. The following results are obtained:

$$E_s (4.9\% \text{ enrichment}) = 2.86 \times 10^{-5} E_{\text{TREAT}} \quad (2)$$

$$E_s (3.7\% \text{ enrichment}) = 2.21 \times 10^{-5} E_{\text{TREAT}} \quad (3)$$

where

E_{TREAT} = Energy release in TREAT, Mw-sec

E_s = Energy release in slug, $\frac{\text{Mw-sec}}{\text{cm}^3}$

Since the TREAT reactor is capable of withstanding a transient with an energy release of 2500 Mw-sec without core damage⁽⁷⁾, this transient (although accidental with respect to present TREAT operating limits) will be used to compute the maximum energy release in the fuel slug.

$$E_{s\text{max}} (4.9\% \text{ enrichment}) = 7.12 \times 10^{-2} \frac{\text{Mw-sec}}{\text{cm}^3} = 68 \frac{\text{BTU}}{\text{cm}^3}$$

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$$E_{\text{max}} (3.7\% \text{ enrichment}) = 5.53 \times 10^{-2} \frac{\text{Mw-sec}}{\text{cm}^3} = 52.5 \frac{\text{BTU}}{\text{cm}^3}$$

The slug volume is 61.5 cm^3 , thus the maximum energy released in the slug is 4200 BTU for the 4.9% enrichment and 3200 BTU for the 3.7% enrichment.

V. PRELIMINARY THERMAL ANALYSIS OF UC EXPERIMENT

A. Energy Necessary to Melt the Slug

As a first approximation, assuming equilibrium conditions and the absence of temperature gradients, the heat necessary to melt the slug can be evaluated from

$$E = m [c (T_m - T_s) + L] \quad (4)$$

where

m = mass of fuel slug, 1.55 lbs.

c = specific heat of UC, 0.071 BTU/lb-°F

L = heat of fusion, ~ 70 BTU/lb

T_m = UC melting temperature, 4350°F

T_s = starting temperature, 800°F

From the computation with Equation (4), with the fuel slug preheated to 800°F, an energy release of 500 BTU is necessary to melt the slug. The transient energy release in TREAT necessary to achieve such energy releases in the slug is found from Equations (2) and (3) to be about 300 and 400 Mw-sec for the 4.9% and 3.7% enriched slugs. In these calculations, it is assumed that there is no heat transfer from the slug. Actually, the sodium bond in the slug will allow appreciable heat transfer from the slug by transient conduction during the experiment. Consequently, it is expected that transients with energy release greater than those computed above will actually be necessary for an experiment in which the fuel slug is melted. Numerical evaluations of the actual energy required to reach melting with transient heat conduction from the fuel slug can be made using a multi-region digital computer code for cylindrical geometry.

B. Equilibrium Temperature Estimates

The heat capacity of the inner capsule has been estimated at

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4.3 BTU/°F. This estimate is arrived at from the following data:

Material	Specific Heat, BTU/lb-°F	Weight lb	BTU/°F
Fuel	0.07	1.5	0.11
Steel	0.12	6.9	0.83
Sodium	0.30	0.7	0.21
Beryllium	0.68	4.6	<u>5.13</u>
			4.28 BTU/°F

The maximum possible energy release is 4200 BTU and 3200 BTU for the 4.9% and 3.7% enriched slug, respectively, as shown previously in Section IV. Such energy release will raise the equilibrium temperature of the inner capsule by 1000°F and 750°F. Thus, if the transient occurs at an initial capsule temperature of 800°F, equilibrium temperatures of 1800°F and 1550°F would be expected, assuming negligible heat loss. The actual experiments will call for transients in which the energy release in TREAT will be around 500 Mw-sec. For such transients the energy release in the 4.9% and 3.7% samples will be about 850 BTU and 650 BTU, respectively. Assuming an initial temperature of 800°F, the resulting equilibrium temperatures will be about 1000°F and 950°F, respectively.

C. Transient Temperature Distribution

The detailed radial temperature distribution through the UC fuel, sodium bond, cladding, sodium annulus, and beryllium heat sink was estimated for the most severe transient anticipated in the experiment. Based upon calculations in Section VB, the transient energy release in TREAT required to melt a 4.9% enriched UC fuel slug was 300 Mw-sec. Due to heat conduction from the fuel during the transient, it was assumed for the purpose of the following calculation that an energy release in TREAT of 500 Mw-sec will be required. The temperature distribution was estimated by means of an analog computer utilizing a preliminary model based upon the relationship

$$\rho_i c_i \frac{\partial T_i}{\partial t} = k_i \frac{1}{r} \frac{\partial}{\partial r} \left(r \frac{\partial T_i}{\partial r} \right) + M f(t) \quad (5)$$

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where

r is the radius

T_i is the temperature at some point in the i th region

t is the time

$Mf(t)$ is the power source term in the fuel as a function of time

ρ_i is the density of the i th region

C_i is the specific heat of the i th region

k_i is the thermal conductivity of the i th region

The configuration analyzed was based on the actual fuel and test capsule dimensions and the following material properties:

	Fuel	Clad	Sodium	Heat Sink
Density lb/in ³	0.42	0.28	0.03	0.07
Thermal Conductivity BTU/hr-ft-°F	13.3	12.5	36.0	64.0
Specific heat BTU/lb-°F	0.071	0.12	0.30	0.68

The transient temperature distribution which was obtained for the case above is shown in Figure 2. For these calculations, the heat generation rate was assumed uniform throughout the slug. Actually, the fuel disadvantage factor causes more heat generation at the periphery than at the center of the slug. In addition, the limitations of the preliminary computer model do not allow the inclusion of phase changes. Thus, the fuel temperatures shown in Figure 2 are considerably higher than expected in the actual case as the heat of fusion of UC has been neglected. This heat of fusion would be equivalent to a ΔT of about 1000°F, thus, the cladding and sodium temperatures shown in Figure 2 can be considered as pessimistic.

Even with allowance for heat of fusion of UC, a TREAT transient as large as 500 Mw-sec may be somewhat higher than that required to melt fuel, but perhaps of the correct magnitude to melt fuel and cladding. The presence of the sodium bond allows for rapid heat dissipation from the fuel resulting in sodium bond temperatures that could result in generation of appreciable sodium vapor pressures within the clad fuel slug

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during the transient. The relationship between cladding strength and sodium bond pressure will be discussed further in Section VI. The calculations performed above also demonstrate the significant difference in heat transfer characteristics between the GE-UO₂ fuel rod and the UC rod in our experiment. The higher thermal conductivity of UC and the presence of a sodium thermal bond result in higher transient heat flow than in the case of the unbonded, UO₂ rod. Consequently, we will experience higher cladding temperatures than in the UO₂ case during equivalent transients.

VII. PRELIMINARY HAZARDS ANALYSIS

A. Internal Pressure Capability of Fuel Cladding

In considering the pressure capability of the fuel cladding, one can first examine the characteristics of the pressure environment during the transient. In Figure 2 we observed that the temperature of the sodium thermal bond reaches values during a 500 Mw-sec transient which are sufficiently high to cause boiling and associated generation of appreciable pressures from sodium vapor. This, of course, assumes bulk boiling in the bond. Using the temperature-time function of the sodium bond in Figure 2 we can evaluate the bond pressure history during the transient, and this is shown in Figure 3. In Figure 3 the constant pressure of 350 psia represents the helium cover gas pressure above the sodium coolant in the capsule. This gas pressure is developed from expansion of the initial capsule cover gas pressure of 10 atm at 70°F, when heated to 800°F during capsule preheating. Since this pressure acts externally on the fuel cladding, the net pressure available to stress the cladding is the difference between the internal sodium bond pressure and 350 psia. The net pressure pulse acting on the cladding is estimated from Figure 3₁ to have a maximum value of 430 psi and a duration of the order of 1 second. In Figure 2 we observed that the maximum average cladding temperature during a 500 Mw-sec transient approached the melting point of 304 ss. Under these conditions the cladding will undoubtedly fail. If we postulate that the maximum cladding temperature shown in Figure 2 is overly pessimistic, and assume a maximum cladding temperature of 1800°F, we can evaluate the bursting stress for this temperature. The hoop stress for thin wall tubing is

$$\sigma_h = \frac{Pd}{2t}$$

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where

P = internal pressure, psi
 t = wall thickness, 10 mil
 d = internal diameter, 0.932 in.

thus,

$$P = 0.0206 \sigma_h$$

Using the ultimate strength of 304 ss at 1800°F, 12,000 psi,⁽³⁾ the bursting pressure is

$$P_b = (0.0206)(12,000) = 247 \text{ psi}$$

This pressure is clearly within the range of the amplitude of the pressure pulse shown in Figure 3 even if we assume a correspondingly lower sodium bond temperature in Figure 2. One must finally consider how the duration of the pressure pulse (~ 1 second) compares with the natural time for plastic flow when evaluating dynamic stress. The natural time for plastic flow for the cladding at 1800°F is computed to be approximately 0.1 millisecond, thus, any portion of the pressure pulse can be considered to act statically with respect to the cladding wall.

It is evident from the foregoing discussion that the subject of transient heating effects as related to fuel cladding behavior is in itself an important area for investigation in the experimental program.

B. Internal Pressure Capability of the Inner Capsule

The calculations performed in Section VIA indicated that when a 4.9% enriched UC fuel slug is subjected to a TREAT transient with sufficient energy (~ 500 Mw-sec) to melt a substantial portion of the fuel, cladding failure would most likely also occur. With respect to experimental hazards, however, failure of the cladding does not represent a significant hazard, as fission products remain trapped within the sodium coolant which is contained by the primary enclosure, the inner capsule. One must then examine the internal pressure capability of this container.

The equilibrium temperature of the contents of the inner capsule following a 500 Mw-sec transient was computed in Section V-B to be 1000°F, for an initial capsule temperature of 800°F.

Estimates of the internal pressure capabilities of the inner capsule are also based on Hoop stresses as in the case of the fuel cladding, thus, yielding pressure, $P_y = \frac{2\sigma_y t}{d}$, σ_y = yield strength

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bursting pressure, $P_b = \frac{2\sigma_y t}{d}$, σ_u = ultimate strength
The minimum value of t/d for the inner capsule occurs just outside the heat sink where $t = 1/8"$ and $d = 3"$. Using values⁽³⁾ of yield and ultimate strength for 304 ss at 1000°F of 18,000 psi and 50,000 psi respectively, we obtain:

$$P_y = \frac{\sigma_y}{12} = \frac{18.0 \times 10^3}{12} = 1500 \text{ psi}$$
$$P_b = \frac{\sigma_u}{12} = \frac{50.0 \times 10^3}{12} = 4170 \text{ psi}$$

Thus, we observe that the inner capsule pressure capability based upon yield stress alone is at least a factor of four greater than the pressures associated with the capsule during a 500 Mw-sec transient.

C. Radiation Hazards

The following section closely follows that prepared by GE⁽¹⁰⁾ with appropriate changes to reflect differences in fuel. In a 500 Mw-sec transient an estimated 2.70×10^{16} fissions will result in the fuel sample. This will lead to gamma-ray dose rates listed below:

Activated Component	Time after Shutdown	Total Activation (curies)	Distance from Center of Capsule	Dose Rate r/hr
Fuel	2 days	.180	6 inches	1.9
Sodium	" "	.034	" "	0.35
Beryllium	---	---	---	---
Steel	" "	.013	" "	0.14

The indicated fuel activity is due to fission products and was calculated for a representative gamma ray energy of 2 Mev from curves by Moteff.⁽⁴⁾ Sodium activation is due to the formation of sodium 24. The steel activity is that of stainless steel 304 as given by the method of Bopp and Sisman.⁽⁵⁾ The dose rate at 6 inches includes the shielding effect of the capsule.

Fission products provide the primary gamma ray source since activation of other materials can be seen to be small by comparison. It is calculated that increasing the time following the transient from 2 days to 2 weeks decreases the dose rate by a factor of 6. Two hours after the transient the fission product dose rates are expected to be a factor of 6 greater than those at 2 days. In the event of a major accident,

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fission gases may be released from the capsule. Thus, it would be desirable to know the total activity of these gases. The following table lists the estimated activities of the krypton and xenon gases as well as those of the iodines, following a 500 Mw-sec transient. The activity values are for the total fission product yields with no radioactive decay or build-up (immediately following the transient). Fission product yields and half life values were taken from the data reported by Katcoff.⁽⁶⁾

Fission Product Gas Activation

FISSION PRODUCT GAS ACTIVATION (For 2.7×10^{16} fissions)

<u>Isotope</u>	<u>Half-life</u>	<u>Yield</u> <u>%</u>	<u>Immediate</u> <u>Activity - curies</u>
Kr-85	10.6 yr	.22	4.5×10^{-6}
Xe-133	5.27 d	5.0	7.0×10^{-2}
I-129	1.7×10^7 yr	0.6	7.5×10^{-11}
I-130	12.6 hr	3.8×10^{-4}	5.6×10^{-5}
I-131	8.05 d	2.3	2.3×10^{-2}
I-133	20.8 hr	5.2	4.7×10^{-1}
I-134	52.5 min	5.9	$1.3 \times 10^{+1}$
I-135	6.7 hr	4.6	1.3
I-136	86 sec	2.3	1.8×10^2

The activities of I-134, I-135, and I-136 will die out quickly with time. The Xe-135 activity will decrease more slowly while Xe-133, I-131 and I-133 may increase in activity for a short time after shutdown.

The probability of release of fission products or radioactive gases during normal operation should be quite low due to the triple containment of the fuel and the airtight seals at each point. In the case of failure of the fuel cladding during an experiment, double containment is provided by the inner capsule and the outer assembly container.

D. Maximum Credible Accident

The TREAT reactor will withstand an accidental 2500 Mw-sec transient without damage,⁽⁷⁾ therefore, attention will be given to such a transient. Beyond this point, it is considered that the effect of the sample would not add significantly to the core damage already brought

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about by elevated core temperatures which would result from a transient in excess of 2500 Mw-sec. It has been shown earlier that for such a transient, an energy release of 4200 BTU occurs in the 4.9% enriched fuel slug, raising the equilibrium temperature of the inner capsule to 1800°F. In these calculations, no heat losses are taken into account, consequently they are pessimistic. Now, above the sodium coolant, there will be a helium blanket of 10 atmospheres at room temperature. If we assume that this gas blanket is raised to an average temperature of 1500°F at equilibrium, an internal pressure of 540 psi would result. This value is below the yield pressure estimated for 1800°F and nearly a factor of 2 below the bursting pressure as shown below:

$$P_y = \frac{\sigma_y}{12} = \frac{8 \times 10^3}{12} = 750 \text{ psi}$$

$$P_b = \frac{\sigma_u}{12} = \frac{12 \times 10^3}{12} = 1000 \text{ psi}$$

Some consideration must be given to the fact that higher internal pressures may occur in the period between the power burst and equilibrium. At any time prior to equilibrium, however, the inner capsule wall will be at a lower temperature which will result in higher pressure capability.

A 2500 Mw-sec transient in TREAT would most likely be capable of vaporizing a major portion of the fuel sample. However, before the fuel reached the boiling point, it is reasonable to assume that molten fuel material would be in direct contact with sodium upon dispersal, thus resulting in rapid transfer of a considerable fraction of the heat generated in the fuel. As a result, we can assume that the average fuel temperature would not exceed the vaporization temperature, which for UC is reported to be about 7900°F at 1 atm.⁽⁸⁾ The sodium temperature in the inner capsule will in turn increase during this time to the point where the sodium vapor pressure is equal to the pressure of the expanding helium cover gas at the same temperature. When this temperature is achieved, bulk boiling of sodium will take place and additional heat transferred from the fuel will superheat the sodium vapor to an average temperature intermediate between the fuel vaporization temperature (7900°F) and the sodium boiling point. The temperature at which bulk boiling of sodium will occur is estimated to be 3000°F from consideration of ideal

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gas law expansion of helium and sodium vapor pressure data.⁽⁹⁾ Using the heat capacities previously estimated in Section V-B for the sodium and fuel, one obtains an average sodium vapor temperature of 4650°F. If the sodium vaporizes at 3000°F and 65 atm. we can predict the internal pressure of the sodium-helium vapor mixture at 4650°F:

$$P_2 = \frac{T_2}{T_1} (P_1) = \frac{5110^\circ R}{3460^\circ R} (65 \text{ atm.}) = 96 \text{ atm.} \\ \text{or } 1420 \text{ psia}$$

This maximum pressure is lower than the yield pressure of the inner capsule at 800°F (1800 psi) and almost a factor of 4 less than the bursting pressure (5200 psi). It should be noted that in our experiments it may be desirable to operate without the initial helium cover gas pressure of 10 atm. In the case of the absence of the cover gas, the transient sodium pressures experienced during the maximum credible accident would be even lower than the pressures estimated above. It is concluded, therefore, that the mechanical integrity of the capsule is sufficient to preclude violent rupture under maximum credible conditions.

Failure of the fuel specimen during an accident may result in the insertion of a small amount of reactivity under certain conditions. Dilution of the specimen fuel reduces the self-shielding of the fuel, thus increasing the reactivity. G.E.⁽⁷⁾ computed that for their sample, a uniform distribution of the specimen fuel throughout the inner capsule increases the TREAT reactivity by 1.4% $\Delta k/k$. In the matter of reactivity insertion, it may be assumed that test capsule failure results in the distribution of fuel throughout the central region of the TREAT core. G.E. estimated its specimen under such conditions results in an insertion of less than 2.8% $\Delta k/k$ reactivity above that of the core with a normal test specimen and capsule. G.E. calculates that if all its specimen fuel is distributed uniformly throughout the TREAT core, a reactivity insertion of 1.2% $\Delta k/k$ results. Since our sample has less U-235, we may use their results as an upper limit.

E. Nuclear Effects on Reactor Performance

No calculation of the effect of the sample and test capsule on

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the reactivity of the TREAT reactor have been performed for the present evaluation. However, G.E.⁽⁷⁾ performed such calculations for its experiment and arrived at a net decrease in reactivity of about 1.8% $\Delta K/K$. Since we intend to use the same set-up with perhaps somewhat less uranium, we may guess a similar decrease in reactivity. G.E. also calculated that the effect of the test assembly on the temperature coefficient of reactivity of TREAT is negligible. Further, G.E. found that the insertion of the capsule changes the control rod worth by +7% which is in agreement with TREAT operating experience for experiments of comparable reactivity worth.

VI. CONCLUSIONS AND RECOMMENDATIONS

The preliminary analysis and evaluation performed for an initial experimental program on transient heating of prototype HNPF uranium carbide fuel indicate that the in-pile capsule design and technique utilized by G.E. for UO_2 experiments in TREAT is suitable for carbide experiments. The use of a single, clad uranium carbide fuel slug of HNPF 2nd core specification is dimensionally compatible with the existing capsule design, only minor compensation being required for the shorter carbide slug length. A preliminary analysis of the hazards associated with the use of carbide fuel in the capsule indicates no significant difference from the G.E.- UO_2 experiments. Additional highlights of the study conducted can be summarized as follows:

1. A transient energy release of ~500 Mw-sec for the TREAT reactor is estimated to be necessary to melt a substantial portion of the UC fuel slug. This energy release corresponds to a TREAT power burst with an initial reactor period of approximately 75 milliseconds.
2. The presence of the sodium bond and relatively high thermal conductivity of uranium carbide results in significantly greater transient heat transfer from the fuel than in the case of the UO_2 fuel element tested by G.E.
3. A preliminary analog estimate of temperature profiles in the clad fuel specimen during fuel melting indicates that the cladding melting point may be achieved, and that cladding failure may occur prior to

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this as a result of vapor pressures generated in the sodium bond.

The results obtained in this study indicate several areas which are worthy of further examination and analysis in more detail. In particular, additional analyses should be performed with respect to the pressures generated in the sodium bond for various transient conditions approaching fuel melting, and the relationship of these pressures to the strength of the cladding in each case. The temperature profile computations required for such an analysis would be performed using existing transient conduction programs available for the digital computer. This study would attempt to predict whether or not cladding rupture can occur prior to fuel melting, and if so, under what circumstance. The results of the study would be useful to ASCR Reactor Safety and could at the same time be incorporated directly into the final analyses required for the UC experiments in TREAT.

It also appears worthwhile to consider conducting an experiment in advance of the TREAT study utilizing the KEWB reactor, in order to examine the ^{subject} of cladding failure due to sodium bond vapor pressure. A small mock-up of the cylindrical geometry involved could be assembled with a central, ^{poorly} conducting dummy rod having a few mils of fully enriched uranium electroplated on its outer surface to serve as the heat source. This rod would then be bonded with sodium and clad in the usual manner. A thin, fully enriched heat source of this type can develop sufficient energy release during short period KEWB transients to cause it to melt and to vaporize a major fraction. It is also expected that the temperature of a thin, fully enriched sample of uranium carbide can probably be raised to the melting point (4350°F).

VIII. PROGRAM PLANNING SCHEDULE

The tasks associated with the initial experimental program of UC experiments in TREAT and the corresponding planning schedule for these tasks is shown in the following chart:

SCHEDULE OF INITIAL EXPERIMENTAL PROGRAM

TASK	MONTH FROM INITIATION OF PROJECT									
	1	2	3	4	5	6	7	8	9	10
1. Detailed thermal and nuclear analysis of experiments.	→									
2. Hazards summary and test specifications for TREAT.	→									
3. Preparation of fuel specimens.	→									
4. Procurement of Components and Materials.	→									
5. Capsule fabrication.		→								
6. Capsule loading and out-of-pile tests.			→							
7. Capsule irradiations in TREAT.					→					
8. Hot cell examination.						→				
9. Analysis and evaluation of data.						→				
10. Preparation of reports.							→			

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Figure 1
G.E. Treat Capsule
Assembly

Figure 2

Fig. 3 - Capsule Pressure as a Function of Time During 500 Mw-sec Transient