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## Transient Behavior of TRIGA, a Zirconium-Hydride, Water-Moderated Reactor

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This paper describes the transient behavior of TRIGA, a light-water-cooled reactor using fuel-moderator elements composed of uranium and zirconium hydride. The large, prompt negative temperature coefficient—an inherent characteristic of these fuel-moderator elements—limits reactor power transients primarily by means of fuel-element temperature rise rather than by void formation in the core. Step reactivity insertions of up to 1.6% resulted in peak powers of 250 Mw with no detectable boiling of the core water or expulsion of water from the core.

### REACTOR DESCRIPTION

The core of the TRIGA reactor (1) is at the bottom of an aluminum tank 6 ft in diameter and 21 ft deep, and is surrounded by a 1-ft-thick graphite reflector (see Fig. 1). The tank is filled with light water, which serves as moderator, radiation shield, and coolant. Top and bottom grid plates, containing 91 spaces, position the fuel-moderator elements in the core. Spaces not loaded with fuel are filled with either control-rod guide tubes or graphite dummy elements.

The fuel-moderator material is an alloy of uranium 20% enriched in  $U^{235}$  and zirconium hydrided to approximately one hydrogen atom for each zirconium atom (2). The fuel material in each element is in the form of a right circular cylinder  $\sim 1.5$  in. in diameter and 14 in. long (see Fig. 2). Four-inch graphite slugs at each end of the cylinder act as top and bottom reflectors. Between the graphite and the fuel-moderator material are inserted aluminum disks containing a burnable poison. The elements are clad with 0.030 in. of aluminum.

The reactor is controlled with four boron carbide rods: a central shim rod, two safety rods, and one pneumatically operated transient rod. The transient-rod withdrawal time is less than 0.1 sec. This time interval is negligible as far as reactor transient performance is concerned, because the rod is fully out of the core and the reactor attains its asymptotic period before a significant amount of power is generated.

### PRELIMINARY EXPERIMENTS

The initial experiments with the TRIGA reactor established that if fuel and water are heated by the same amount, there is only a slight change in reactivity (varying from  $+0.13$  cents/ $^{\circ}C$  at  $10^{\circ}C$  to  $-0.1$  cents/ $^{\circ}C$  at  $60^{\circ}C$ ). If the reactor is operated at a significant power level, on the other hand, thus raising the temperature of the fuel elements above that of the core water, there is a large decrease in reactivity (approximately  $-1.5$  cents/ $^{\circ}C$  average fuel-temperature rise). The average core water temperature does not change significantly compared with that of the fuel elements during high-power operation, because of the low flow impedance through the reactor core and the constant temperature of the inlet water.

As the uranium and the hydrogen in the fuel-moderator elements are intimately mixed, there is no time delay between fission and local heating of the hydrogen in the element; therefore, the observed large negative temperature coefficient, which is attributed to the bound hydrogen in zirconium hydride, is a prompt effect.

Because the major reactivity effects that have been observed in this reactor are prompt, they may be evaluated through a series of quasi-equilibrium, or static, experiments. The results obtained from these experiments may then be used to predict the transient behavior of the reactor. In these quasi-equilibrium experiments, simultaneous measurements of the average reactor fuel temperature and

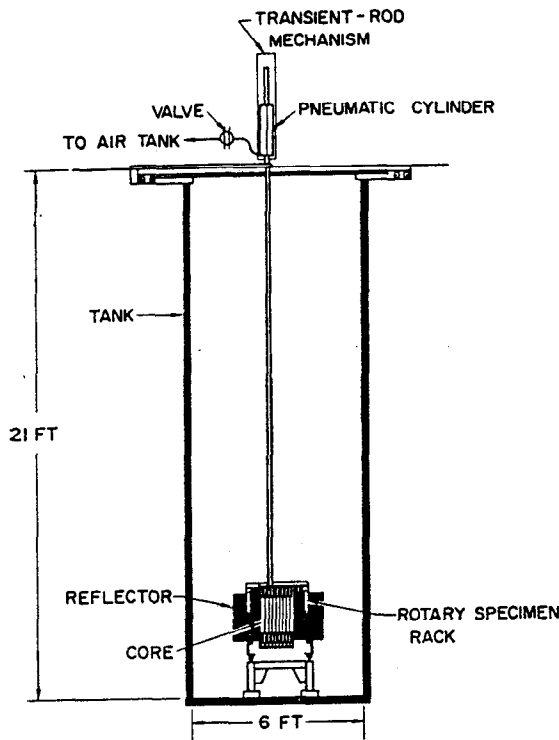


FIG. 1. Elevation view of TRIGA reactor.

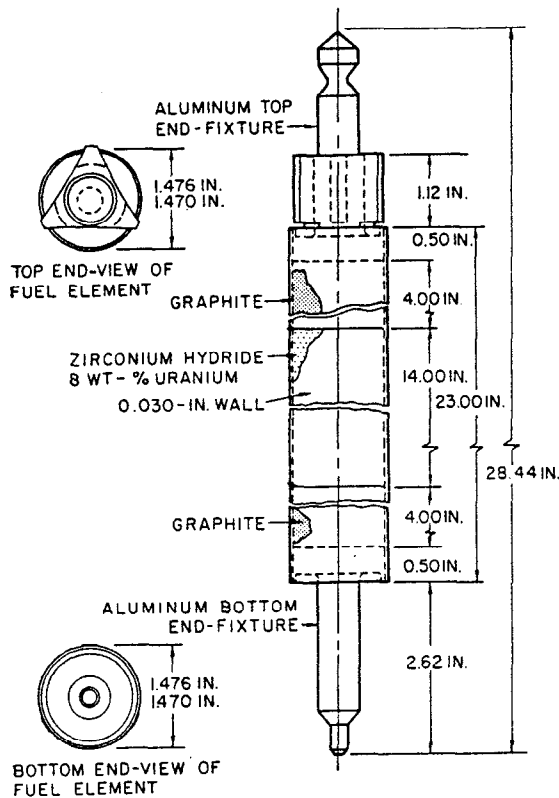


FIG. 2. TRIGA fuel-moderator element.

of reactivity loss are made as functions of reactor power level. These measurements provide a parametric relationship between reactivity loss and average fuel-temperature rise above average core-water temperature. The slope of the resulting curve is the reactor temperature coefficient.

### INSTRUMENTATION

To obtain a good experimental determination of average fuel temperature, eight of the fuel elements in the reactor core were each equipped with four internal thermocouples. These measured the central, top, bottom, and surface temperatures of the fuel. Core inlet and outlet water temperatures were also measured.

Neutron-sensitive ionization chambers supplied the input signal for the fast electronic amplifiers that provided power-level information.

A high-speed, 36-channel galvanometer recorder was used to measure transient fuel temperature, water temperature, transient-rod withdrawal time, and reactor power level. The core was photographed during each transient with a high-speed motion picture camera.

### QUASI-EQUILIBRIUM EXPERIMENTS

The reactor was loaded with the desired excess reactivity (with a maximum of approximately 4%), and the control rods and transient rods were calibrated. The reactor power level was then raised in small increments. At each new level, temperatures and changes in control-rod position were recorded. These data are shown in Figs. 3 and 4. Figure 5 shows reactivity loss as a function of average fuel-temperature rise with a small correction for changes in average core-water temperature. The slope of this

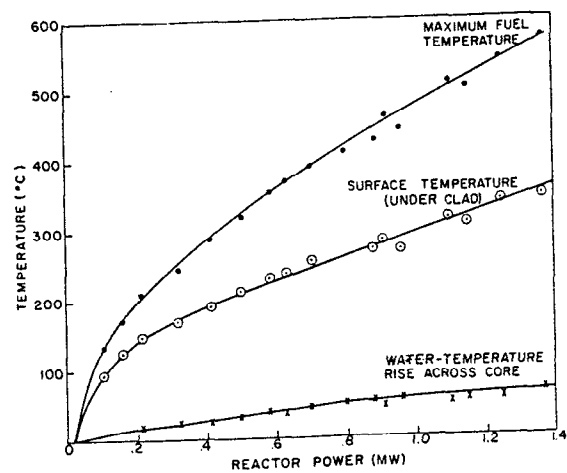


FIG. 3. Typical component temperatures in high-power quasi-equilibrium experiments.

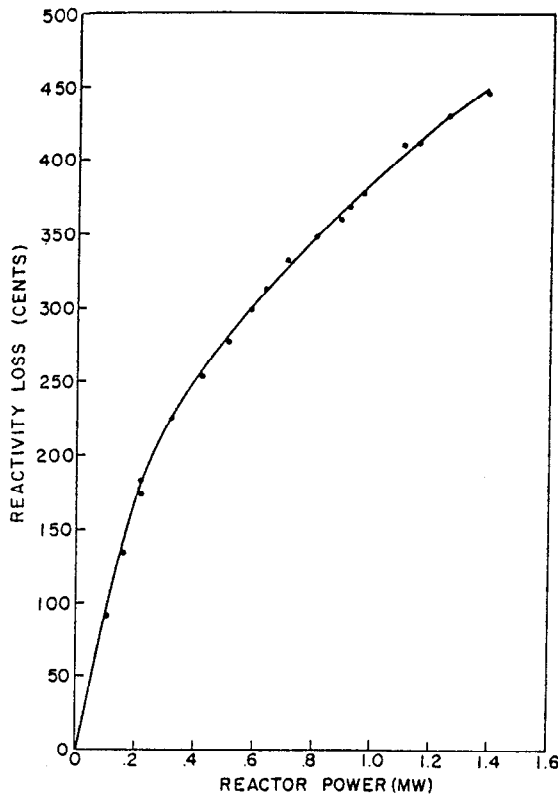


FIG. 4. Reactivity loss vs power level in high-power quasi-equilibrium experiments.

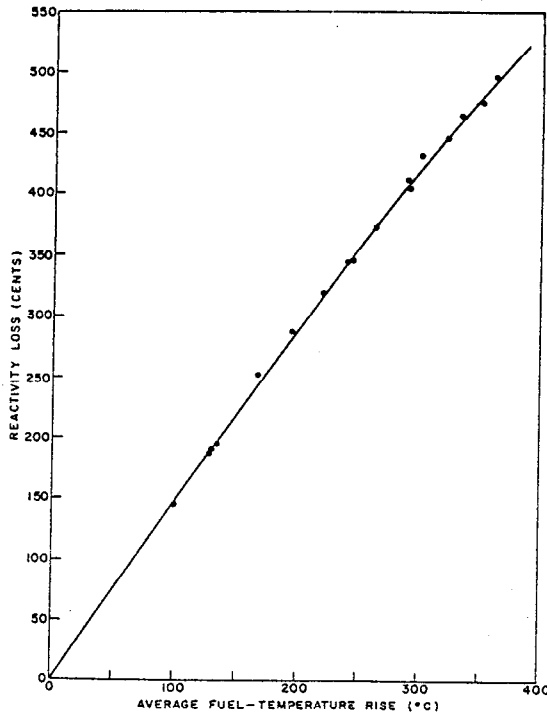


FIG. 5. Reactivity loss as a function of average fuel-temperature rise with constant average core water temperature.

curve yields an initial fuel-temperature coefficient of  $-1.5$  cents/ $^{\circ}\text{C}$ , which decreases slowly to approximately  $-1.0$  cent/ $^{\circ}\text{C}$  at  $400^{\circ}\text{C}$ .

These measured values of the reactor temperature coefficient were subsequently used in kinetics calculations to predict the behavior of the reactor following step reactivity insertions.

TRANSIENT EXPERIMENTS

For each transient, the reactor was brought to criticality and held at a constant low power. The transient rod was adjusted so that the system reactivity rose above delayed critical by a predetermined amount when the pneumatic rod was driven out of the core.

The transient rod was removed from the core in less than 0.1 sec. After it was fully removed, the power of the reactor increased on the asymptotic period for several periods before an appreciable amount of power was generated. Thereafter, the temperature of the core increased rapidly. The reactor power increased to a maximum and was then observed to decrease with a period approximately equal to the asymptotic period, as shown in Fig. 6. Following the prompt burst, the reactor power level decreased monotonically to the level measured in the quasi-equilibrium experiments.

There was no evidence of any system instability, boiling of water in the core, or disturbance of the

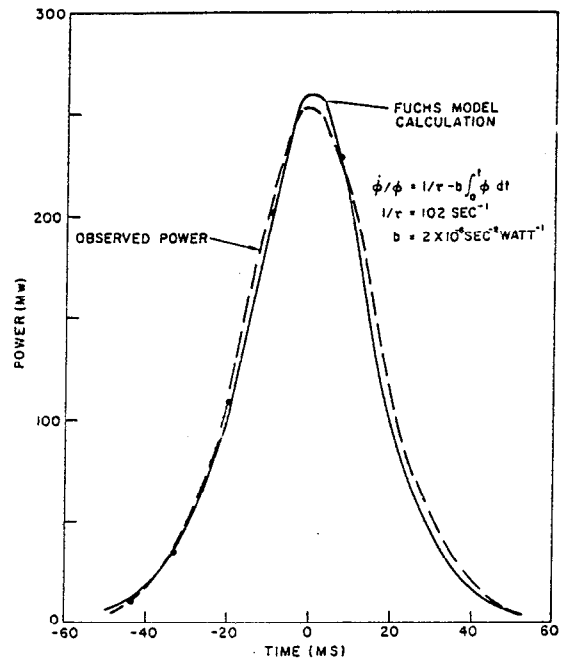


FIG. 6. Power level as a function of time for the 1.6%  $\delta k/k$  TRIGA transient.

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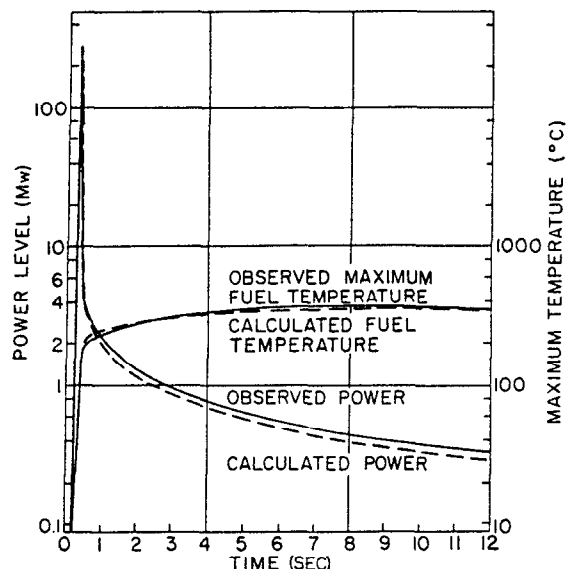


Fig. 7. Transient power and fuel temperature as functions of time after a 1.6% reactivity insertion

water level in the tank during these transients. The major portion of the transient energy release was stored as thermal energy in the fuel elements during the transient, and it subsequently diffused slowly into the cooling water. Reactor power and temperature as functions of time are shown in Fig. 7 for the 2.00-dollar ( $\sim 1.6\%$   $\delta k/k$ ) transient. This figure also shows predictions of power and fuel temperature based on the temperature-coefficient data obtained in the quasi-equilibrium experiments as used in the reactor model discussed in the following section.

A summary of the results obtained from the series of transient experiments performed to date is given in Table I. From these data,  $l/\beta = 0.01$ , with an estimated uncertainty of 0.001. Data obtained from the 2.00-dollar transient are given in Table II.

#### SOURCE OF THE PROMPT TEMPERATURE COEFFICIENT OF THE HYDRIDE

Experimental evidence indicates that the bound hydrogen in the uranium-zirconium-hydride fuel-moderator material acts like an Einstein oscillator with energy levels  $(N + 3/2)h\nu$ , where  $h$  is Planck's constant,  $N$  is an integer, and  $\nu$  is the oscillator frequency. The energy  $h\nu$  has been measured to be 0.13 eV (3). Therefore, while the moderating properties of this bound hydrogen are much like those of free hydrogen above 0.13 eV, they are greatly inhibited below this energy. Neutrons can be slowed down effectively below this energy only by the water surrounding the elements. However, the bound

TABLE I

SUMMARY OF RESULTS OF TRANSIENT EXPERIMENTS

Reactivity insertion ( $\delta$ )	Reactor period (sec)	Peak power (Mw)	Maximum fuel temperature ( $^{\circ}\text{C}$ )
0.74	1.74	0.2	122
0.91	0.47	0.58	153
1.06	0.15	1.6	177
1.16	0.069	5	193
1.30	0.034	20	222
1.69	0.013	100	285
2.00	0.0098	250	360

TABLE II

TWO-DOLLAR (1.6%  $\delta k/k$ ) STEP-INSERTION PARAMETERS

Duration of prompt burst	40 ms
Energy release, prompt burst	10 Mw-sec
Fissions in prompt burst	$3 \times 10^{17}$
Total energy release	16 Mw-sec
Radiation dose just above pool	3 mrem
Maximum fuel-moderator-element temperature	360 $^{\circ}\text{C}$

hydrogen in the elements can speed up neutrons with energy less than 0.13 eV by integral multiples of 0.13 eV. As the fuel temperature rises, the occupation density of the excited states of the oscillators increases exponentially, and the neutron speeding-up probability therefore increases rapidly. During a transient, the hydride temperature and the fuel temperature rise simultaneously and the neutron spectrum is hardened promptly. This results in a decrease in the fission probability and an increase in the fraction of neutrons lost because of leakage from the core and parasitic capture in the water and in the control rods. The water temperature does not increase significantly over the short time interval of the prompt radiation burst.

Other factors contribute about 20% of the total prompt negative temperature coefficient. Doppler broadening of the  $\text{U}^{238}$  resonances increases parasitic neutron capture, and prompt expansion of the fuel elements displaces water in the center of the reactor, where the void coefficient is negative.

#### COMPARISON OF THEORY AND EXPERIMENT

Two methods are presently used to predict the transient behavior of the reactor. These are (1) an empirical description based on the Fuchs model, which neglects the effects of delayed neutrons and of heat transfer, and (2) a detailed space-independent thermal and neutronic model of the reactor.

Because of the assumptions made, the application of the Fuchs model is limited to transients in which

the reactivity insertion is large compared with 1.00 dollar. This model accurately predicts the shape and magnitude of the prompt burst, as shown in Fig. 6.

For step reactivity insertions, the Fuchs model takes the form (4)

$$\frac{\phi}{\phi} = \frac{1}{\tau} - b \int_0^t \phi(t) dt$$

where  $\phi$  is the power level in watts,  $1/\tau$  is the reactivity step in units of a reciprocal asymptotic reactor period ( $\text{sec}^{-1}$ ), and  $b$  is the energy shutdown coefficient ( $\text{watts}^{-1} \text{sec}^{-2}$ ).

The experimental value of  $b$  for the TRIGA reactor is  $2 \times 10^{-6} \text{ watts}^{-1} \text{sec}^{-2}$ ; this value was obtained from transient data beyond prompt critical. Using this formulation, one can make the following performance predictions ( $\tau$  in sec. throughout):

Maximum power:

$$P_{\max}(\text{watts}) = 2.5 \times 10^4 / \tau^2.$$

Energy release in prompt burst:

$$E_p(\text{watt-sec}) = 10^5 / \tau.$$

Prompt-burst temperature rise:

$$T_p(^{\circ}\text{C}) = 2/\tau.$$

(An additional approximately constant temperature rise of  $\sim 200^{\circ}\text{C}$  due to delayed-neutron contributions will subsequently occur.)

The more detailed model is used for digital computation of the behavior of the reactor. In this model, the space-independent kinetic equations are coupled to a space-independent thermal model of the reactor through the reactor temperature coefficient (5). The thermal model consists of an average fuel heat capacity, an average core-water heat capacity, and a fuel-water heat-transfer resistance. Heat-transfer from the core water to the bulk shielding water is represented by a second transfer resistance. These impedances are determined from the data obtained in the quasi-equilibrium experiments, as is the temperature coefficient required to complete this model. These data are

fitted with a constant-water-temperature coefficient, a Doppler coefficient, and a hydride coefficient based on the Einstein model (6). The results obtained agree with experiment and are illustrated in Fig. 7.

Further experiments of this type are being prepared. Bench-testing of the present fuel-moderator elements, including thermal cycling, indicates that transient temperatures of at least  $800^{\circ}\text{C}$  can be obtained repeatedly without significant damage to the elements (7). Preparations are now being completed to achieve transients to this temperature. On the basis of the negative-temperature-coefficient data obtained from Fig. 5, it is expected that a peak power of 2000 Mw can be reached with a central flux of about  $10^{17}$  neutrons/cm<sup>2</sup>-sec before a maximum temperature of  $800^{\circ}\text{C}$  is obtained.

#### ACKNOWLEDGMENTS

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