

**TRIGA REACTOR
CHARACTERISTICS**

Learning Objectives

1. You should understand the basics of the physics and mechanical design of the TRIGA fuel.
2. You should understand the unique operational characteristics of the TRIGA fuel and realize the differences between TRIGA fuels and other, more traditional reactor fuels

Objective of Training

This module describes the general design, characteristics and parameters of TRIGA reactors and fuel. Much of this information should be incorporated into any reactor operator training program and, in many cases, the facility Safety Analysis Report.

The expected audience for this training consists of designated, certified or licensed reactor operators, operator trainees and regulators with some experience in nuclear reactor facilities and operations. The material is specifically directed at TRIGA fuelled reactor facilities and should only be used as information by other facilities. The unique characteristics of the TRIGA fuel (such as pulsing capabilities) are not generally applicable to other fuels (i.e. MTR reactors).

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1. TRIGA REACTOR PARAMETERS AND CHARACTERISTICS

1.1 History of TRIGA reactors

In the early 1950ies, the U.S. company General Atomics (GA) located in San Diego, California, developed a small research reactor which

- would be inherently safe
- is operationally flexible and relatively inexpensive
- allowed a large variety of experiments
- used low enriched uranium.

At the First Geneva Conference on Peaceful Uses of Atomic Energy in 1955, the first TRIGA-type reactor (Training, Research, Isotope Production General Atomic) was presented to the public. During the following decades, approximately 60 TRIGA reactors were constructed all over the world. Three basic TRIGA models have been produced:

- Mark-I: underground pool without beam tubes
- Mark-II: above ground tank with several beam tubes
- Mark-III: above ground oval tank with movable core.

A short description of the three TRIGA reactor types and list of all worldwide TRIGA reactors that were put into operation (some of them already decommissioned) is given in Appendix 1.

1.2 Critical mass of the TRIGA reactor

Criticality of a typical TRIGA Mark II reactor is usually achieved with about 57 standard (low enriched, < 20%) TRIGA fuel elements (about 2 kg ²³⁵U). To allow higher power operation (>100 kW), more fuel elements and several graphite reflector elements are added in the outer rings of the core grid plate. This added fuel produces a core excess reactivity which may be limited to about 2 \$ depending on the specific license of such a reactor. A typical core loading diagram of a circular grid reactor is shown in Figure 1. More recent TRIGA core designs will use a hexagonal, rather than a circular, design for better thermal and hydraulic characteristics.

Uranium-235 is consumed during operation of any reactor, the amount or burnup-rate depends on the reactor power. For a typical 250 kW TRIGA reactor operating for about 200 days a year, 8 hours per day the U-235 consumption is approximately 20 grams per year. Standard reactor physics calculations show that for 1 MWd (or 24000kWh) of thermal power production 1,25 grams of U-235 are consumed for most reactors. Experience has shown that for a well used 250 kW TRIGA reactor, approximately one fuel element has to added every second year to compensate for the reactivity loss by fuel burnup across the core; therefore, for 40 years of operation, approximately 20 fuel elements have to be added. More fuel may be required to compensate for the reactivity caused by the insertion of large experiments.

Since the development of the original TRIGA type reactors in the mid-fifties, new types of TRIGA fuel elements have been developed. A significant change was replacing (in the design) the aluminium cladding material to stainless steel (SST). Therefore, many TRIGA reactors may be operated with a mixed core with different types of fuel elements in the core.

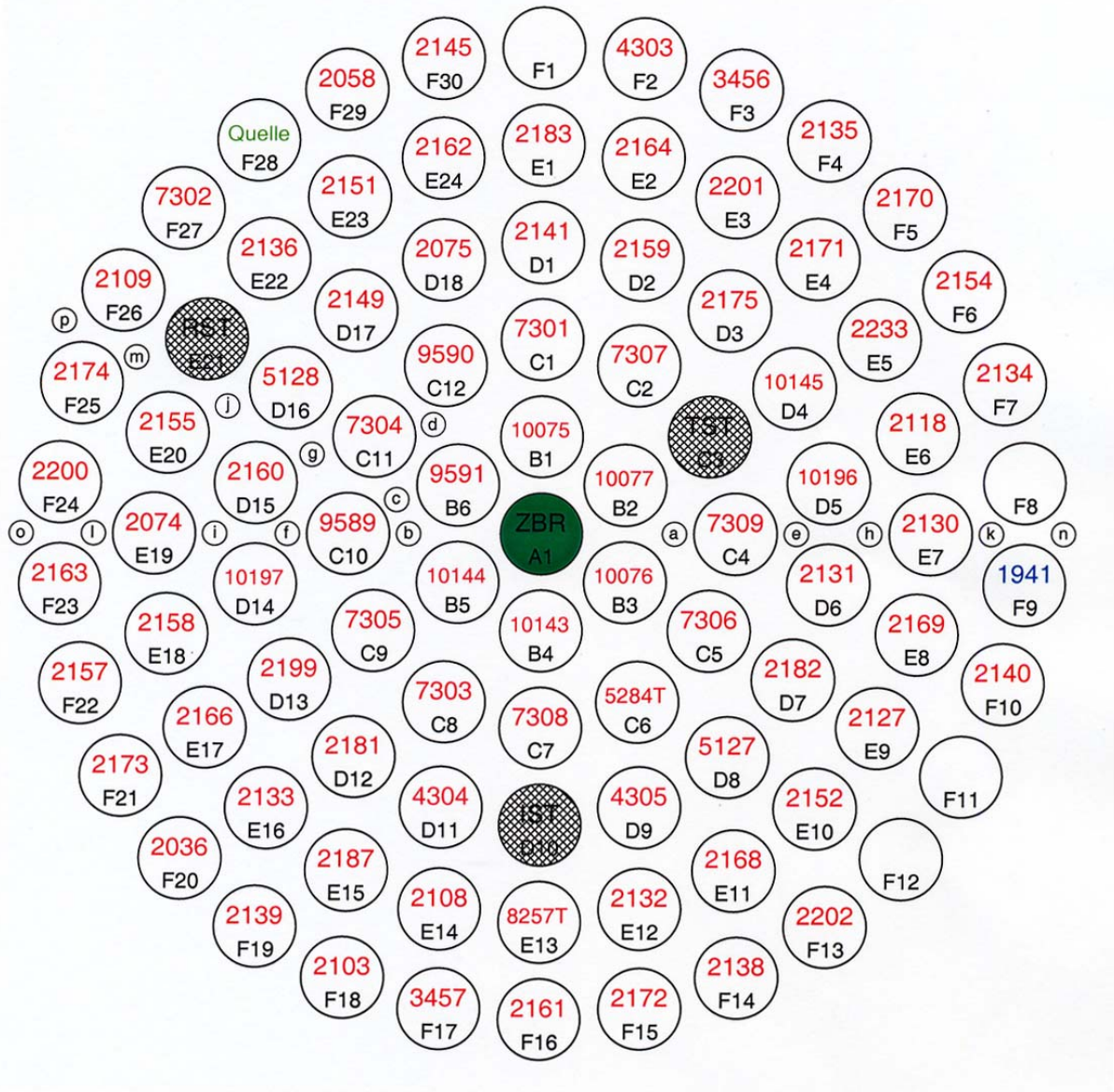


Figure 1: Standard core of a TRIGA

Among the SST elements there exist several different types of fuel, those most often seen in reactors today are produced with <20% enrichment (Standard SST) and those with approximately 70% enrichment (called FLIP = Fuel Lifetime Improvement Program). Details about these different fuel types are given in Table 1. Relatively recently, General Atomics has designed and certified new low enriched fuel with high uranium loading (as high as 30 wt-%) to replace the FLIP fuel. Gradually, FLIP research reactors will convert to low enriched uranium (LEU) fuel due to worldwide nuclear proliferation concerns. This new LEU fuel has higher uranium loading to give a long fuel lifetime without high enrichment. Additionally, General Atomics has produced a TRIGA fuel with a smaller diameter for higher (>3MW) operations.

The neutron reflector materials around the core and the absence or presence of graphite reflector elements in empty fuel positions also affect the reactor critical mass rather strongly. Replacing the graphite with a water reflector on a TRIGA Mark II raises the critical mass (more fuel needed) by about 25%. Adding a ring of graphite reflector elements in the G-ring (outer ring of the grid plate) around a water-reflected core cut will reduce the amount of fuel needed for criticality by about 10%.

Fuel element type	102	104	110 (FLIP)
Fuel moderator material	U-ZrH _{1.0}	U-ZrH _{1.65}	U-ZrH _{1.65}
Uranium content (wt%)	8.5	8.5	8.5
Enrichment (%)	20	20	70
Average ²³⁵ U content (g)	38	38	136
Burnable poison	SmO ₃ -disk	none	Erbium 1.6 wt%
Diameter of fuel meat	35.8 mm	36.3 mm	36.3 mm
Length of fuel meat	35.6 mm	38.1 mm	38.1 mm
Graphite reflector length	10.2 mm	8.73 mm	8.81 mm
Cladding material	Al-1100F	304 SS	304 SS
Cladding thickness	0.76 mm	0.51 mm	0.51 mm

Table 1: Principal fuel element design parameters

1.3 Calculated power per FLIP (=Fuel Lifetime Improvement Program) element

Power density is a significant parameter and reactor operators should be acquainted with the TRIGA values. Generally, operators should try and maintain the reactor flux as flat or constant as possible across the reactor by judiciously loading new elements in the outer ring and gradually moving them towards the center as the fuel is burned. Standard reactor physics calculations show that the neutron flux and local reactor power is peaked towards the center of the TRIGA core. The calculated results for a FLIP reactor are listed in Table 2. Power densities in the higher enriched fuels such as FLIP are slightly higher but the values are very close to a standard fuelled reactor of similar power level. Note that the highest power density occurs in the central B ring, as expected. The effect of the graphite elements in the outer ring is to slightly increase the power density in the outer fuelled rings and to slightly

reduce the power density in the inner fuelled rings, thus flattening the neutron flux across the core.

Average power per element (kW/element) at total core power for a 1000 kW TRIGA

B ring	C ring	D ring	E ring	F ring
18.85	14.68	12.63	9.99	9.22

Table 2: Calculated power per element (FLIP)

Table 3 lists the maximum, minimum, and the average power per element, along with ratios of the maximum-to-minimum power and the maximum-to-average power per element. Again, these values are calculated for a FLIP core but may be used as approximations for other TRIGA reactors. Each facility's Safety Analysis Report (SAR) should have values calculated specifically for the facility.

Maximum power per element P_{\max} (kW/element)	Minimum power per element P_{\min} (kW/element)	Average power per element P_{ave} (kW/element)	$\frac{P_{\max}}{P_{\min}}$	$\frac{P_{\max}}{P_{\text{ave}}}$
15.85	9.22	11.24	1.72	1.41

Table 3: Calculated maximum, minimum and average power per element (FLIP)

1.4 Loss of Coolant Accident (LOCA)

If the reactor pool is accidentally drained of water, the fission product decay heat will be removed primarily by natural convection of air. If the decay heat production is sufficiently low or if there is a long enough interval between reactor shutdown and coolant loss, the convective cooling by air will be enough to maintain the fuel at a temperature which will not damage the fuel elements. This accident was analysed with the following results:

- a) It has been demonstrated that standard fuel suspended in air can tolerate temperatures as high as 900 °C without damage to the clad. For FLIP fuel, this value is 938 °C.
- b) This temperature will not be exceeded under the conditions of coolant loss if the maximum thermal power in an element is equal to or less than 21 kW for standard fuel and 24 kW for FLIP fuel even if the reactor is operated for an infinite time prior to the accident.
- c) If reactor operations are limited to 70 MWh per week, power levels up to 24 kW/element for standard fuel and 28 kW/element for FLIP fuel will not cause element damage in the event of loss of coolant.

Calculations by General Atomics yielded similar results:

After prolonged operation at 2000 kW (about 32 kW per element), an instantaneous loss of water would produce a maximum fuel temperature of about 490 °C, well below the temperature necessary to cause any failure of the cladding (1150C). Higher power reactors with extended power histories must consider if an Emergency Cooling System is required. Typical low power TRIGA reactors are licensed to operate with a maximum steady state power level of 250 kW. The maximum power level expected for a fuel element is 6.5 kW, which is about 2.6% of the total thermal power production. Further the reactor is operated routinely only 40 hours a week, which equals 10 MWh. Therefore, with the maximum expected fuel element power level of less than 7 kW per element and a conservative prediction of a limiting element power of 24 kW, there appears to be a substantial margin of safety against cladding failure and subsequent release of fission products in the event of a loss of cooling accident.

1.5 Design Basis Accident or Maximum Hypothetical Accident

The design basis accident for a typical TRIGA reactor is usually defined as the loss of integrity of the fuel cladding for one fuel element and the simultaneous loss of pool water resulting in fission product release to the environment or building. The phrase “Design Basis Accident” is still used frequently but it is gradually being replaced with “Maximum Hypothetical Accident (MHA). For typical TRIGA reactors, the MHA is ANY accident at that facility that would release fission products from a fuel element or from the failure of a fuelled experiment. The actual mechanism may be unique to a particular facility and should be discussed in detail within the Safety Analysis Report. Generally, it will a “worst case scenario” that may or may not exist at the facility. For example, a Ring B element (highest power density), is removed following infinite reactor operations (maximum buildup of fission products) and is dropped out of the pool while moving the fuel through the air for maintenance or dropped in the pool immediately before a LOCA. The facility must be able to show that the public and environment is still protected if this accident were to occur.

1.6 TRIGA Reactor Moderation

The TRIGA fuel is a Uranium and Zirconium Hydride alloy. The intimate contact between the uranium and the hydrogen of the fuel results in a self-moderated reactor fuel. The prototype TRIGA Mark I reactor was designed through control of the hydrogen-to-uranium ratio, to be slightly under-moderated. Thus, a decrease of the water density of the coolant surrounding the fuel element would produce a negative reactivity response. Because of the fundamentally different neutron thermalisation characteristics of hydrogen bound in a solid hydride as to hydrogen bound in water, the optimum (or minimum) critical mass and the amount of over- or under-moderation are complex functions of the hydride, water, and uranium ratios.

It should be noted that an over-moderated reactor would have an initial positive temperature coefficient. If the water temperature increases then the water density decreases and the neutrons would be more optimally moderated up to a point. If the temperature continued to increase past that point then the moderating properties would decrease which would then result in a negative temperature coefficient.

1.7 Moderating properties of zirconium hydride

Experiments performed by GA personnel at the Brookhaven National Laboratory prior to the construction of the prototype TRIGA reactor showed that zirconium hydride has very unusual moderating properties for thermal neutrons. These experimental results can be explained by assuming that the hydrogen-atom lattice vibrations can be described by an Einstein model with a characteristic energy, $h\nu = 0.140$ eV. This description is consistent with the theory that the hydrogen atom occupies a lattice site at the centre of a regular tetrahedron of zirconium atoms.

The results of these experiments showed that:

1. ZrH_n alone is quite ineffective in slowing down neutrons below 0.14 eV.
2. For neutrons of energy above 0.14 eV, the moderating ability of ZrH_n is at least as good as that of free hydrogen.
3. Cold neutrons can gain energy in passing through ZrH_n by gaining amounts of energy which are integral multiples of $h\nu$ (0.14 eV). The higher the hydride temperature, the more readily this process will occur.
4. These observations are independent of the ratio over wide ranges in the ratio of hydrogen atoms to zirconium atoms in the hydride.

The results of the experiments show that hydrogen atoms bound in a ZrH lattice (Fig. 2) act like harmonic oscillators, or Einstein oscillators. In the so-called Einstein model, each atom is thought of as being bound isotropically to a fixed centre about which it can oscillate harmonically. Such an oscillator has possible energy states $[n + (3/2)]h\nu$, h being Planck's constant (6.62×10^{-34} J·s), ν the oscillator frequency, and n an integer. In a scattering event with such an individual oscillator, a neutron can hence gain or lose an integral multiple, $h\nu$, of energy.

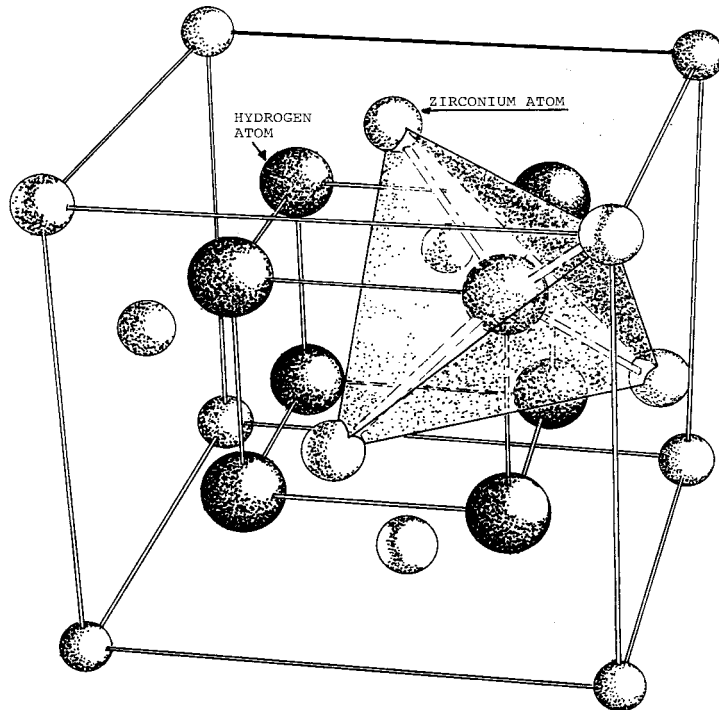


Figure 2: A unit cell of ZrH_2 . The tetrahedral structure of four zirconium atoms (light coloured) surrounding a hydrogen atom (black) is emphasized for one of the hydrogen atoms.

The following theoretical model was therefore adopted. The hydrogen atoms behave as if isotropically bound in an approximately harmonic potential well and have identical frequencies, ν , such that $h\nu = 0.140$ eV, as was observed. The observed frequency spectrum is somewhat broadened by the thermal motions of the zirconium atoms to which the hydrogen atoms are bound. Successive levels may not be spaced quite evenly because of non-harmonic contributions to the potential.

In such a model, a neutron is caused to slow down by losing energy in multiples of $h\nu$, as long as its energy is above $h\nu$. Below 0.14 eV the neutron can still lose energy by the inefficient process of exciting acoustic Debye-type modes in which the hydrogen atoms move in phase with the zirconium atoms, which in turn move in phase with one another. These modes therefore, correspond to the motion of a group of atoms whose mass is much greater than that of hydrogen, and indeed even greater than the mass of zirconium. Because of the large effective mass, these modes are very inefficient for thermalising neutrons, but for neutron energies below 0.14 eV they provide the only mechanism for neutrons slowing down. (In a TRIGA core, the water provides for neutron thermalisation below 0.14 eV.) In addition, in the ZrH it is possible for a neutron to gain one or more energy units of ~ 0.14 eV in one or several scatterings, from excited Einstein oscillators. Since the number of excited oscillators present in a ZrH lattice increases with temperature, this process of neutron speeding-up is strongly temperature-dependent and plays an important role in the behaviour of ZrH moderated reactors.

With the assumption of an Einstein model for the slowing down of neutrons in zirconium hydride, the basic physical processes, which occur when the fuel-moderator elements are

heated can be described. A rise in temperature of the hydride increases the probability that a thermal neutron (~ 0.025 eV) in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. Since it has a longer mean free path for collision, the speeded-up neutron has an increased chance of escaping from the fuel-moderator element before being captured. However, the water channels between the fuel elements are very effective in re-thermalising any speeded-up neutrons which escape from the fuel; thus, the probability that a neutron will return to the fuel element before being captured elsewhere is not dependent on the temperature of the hydride but is in fact a function of the water.

1.8 Prompt negative temperature coefficient of reactivity

TRIGA reactors may be licensed to operate with:

1. Standard Al-clad TRIGA fuel (20% enriched);
 2. FLIP TRIGA fuel (70% enriched);
 3. Standard SST-clad TRIGA fuel (20% enriched)
- (Note: The 20% enriched fuel may have a fuel loading of 8.5, 11, 20 or 30 wt-%)

The basic parameter which allows TRIGA reactors to operate safely during either steady-state or transient conditions is the prompt negative temperature coefficient associated with the TRIGA fuel and core design. This temperature coefficient allows great freedom in steady-state and transient operations.

A major factor in the prompt negative temperature coefficient for the TRIGA cores is the core spectrum hardening that occurs as the fuel temperature increases. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (i.e., the spectrum is hardened). This spectrum hardening is used differently to produce the negative temperature coefficient in standard and FLIP fuel.

1.9 Temperature coefficients of standard 20% Enriched Fuel

As stated earlier, a large effort has gone into designing the TRIGA reactor in such a way that an increase in the temperature of the fuel element will result in a relatively large decrease in reactivity. This large prompt negative fuel temperature coefficient for 20% fuel is rather constant (see Fig. 3) and results from the following three effects:

1.9.1 Cell and Non-homogeneous effects

The 20% standard fuel element has an average neutron mean free path of ~ 3 cm, which is comparable to the average diameter of the element. Now the spectrum hardening effect discussed earlier, increases the mean free path for the neutrons considerably, and therefore, the probability of a neutron escaping the fuel element before capture is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly re-thermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel within a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water (keep in mind the TRIGA core is $\sim 1/3$ water). As a result, there is a temperature-dependent disadvantage factor for the core unit cell in which the ratio of absorptions in the fuel to total

cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance giving a loss of reactivity. Cell effect is an important contributor to the prompt negative coefficient. It should be noted that it is entirely dependent on the heterogeneous appearance of the core to thermal neutrons.

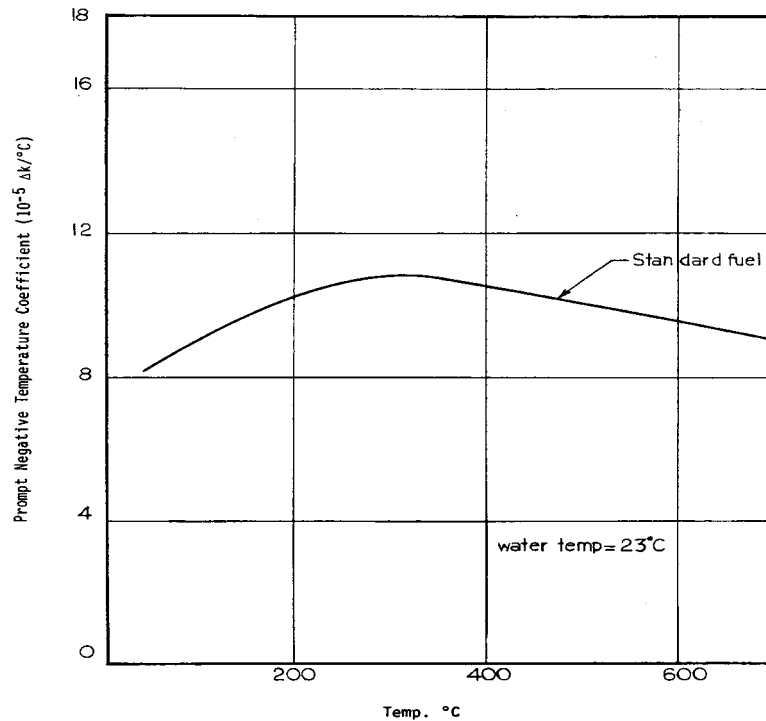


Figure 3: Typical prompt negative temperature coefficient vs. temperature.

1.9.2 Doppler broadening effects

The uranium in the fuel elements is approximately 20% ²³⁵U and 80% ²³⁸U. The capture resonances in ²³⁸U are Doppler-broadened by an increase in the fuel temperature which in turn causes a decrease in the resonance escape probability (p).

1.9.3 Core leakage effects

The core leakage contribution derives basically from the same mechanism which produces the cell effect. The core can be envisaged as a large super-cell with the reflector acting as the moderator. When the core heats up, leakage is increased and relatively more captures occur outside the fuel.

Table 4 shows the components of the temperature coefficient for both an aluminium clad (ZrH_{1.0}) and stainless steel clad (ZrH_{1.7}) core. Note that the cell effect is the dominant contributor and the stainless steel clad cell has a substantially larger value than the aluminium clad cell. The stainless steel clad adds a poison region between the fuel element and cooling water, which acts as a barrier to neutrons returning to the fuel after having gained energy and escaped.

	$(\times 10^{-5} \Delta k / ^\circ\text{C})$	
	Aluminium clad cell, graphite reflected core	Stainless steel clad cell, water reflected core
Cell	4.7(50%)	8.2(65%)
Doppler	2.1(20%)	1.8(15%)
Leakage	2.9(30%)	2.1(20%)
total	9.7	12.1

Table 4: Components of prompt negative temperature coefficient of TRIGA reactors with 20% enriched fuel

(The advantage of using $\text{ZrH}_{1.7}$ hydride lies in the fact that the fuel matrix remains in a single phase well above 1000°C . With $\text{ZrH}_{1.0}$, a fuel matrix phase change and an associated volume change occurs at approximately 530°C .)

Because of the neutron spectrum hardening phenomenon, the temperature coefficient components (in the standard fuel) are increasing functions with temperature up to about 400°C and then they begin to decrease (refer to Fig. 4). The Doppler coefficient decreases slightly with temperature over the entire range of interest.

The long delay times for transferring heat to the core cooling water coupled with the large prompt negative temperature coefficient of the TRIGA fuel-moderator element produce a reactor that is much more nearly linear than exponential in its response to reactivity changes.

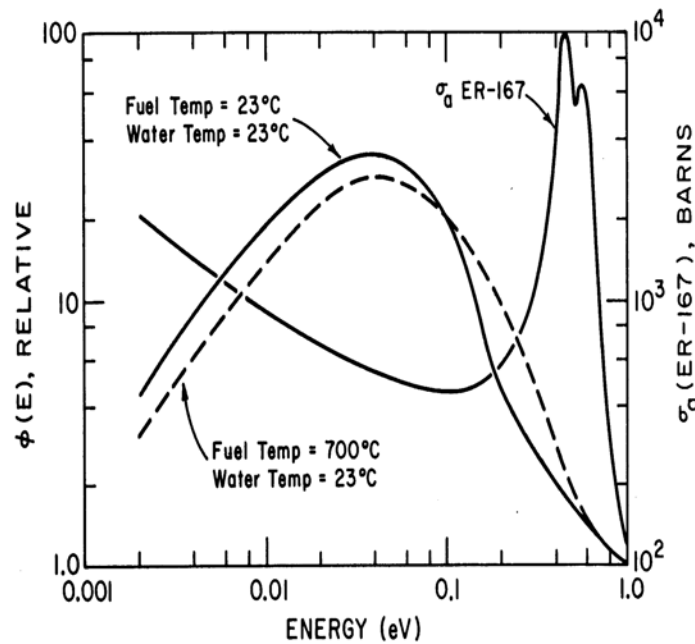


Figure 4: Thermal neutron spectra vs. fuel temperature relative to σ_a vs. energy for Erbium-167.

1.10 Temperature coefficients of 70% enriched fuel

For a TRIGA-FLIP fuel element, the ^{235}U loading is about 3.5 times that of a standard TRIGA element and this causes the neutron mean free path in the FLIP element to be much shorter (~ 3 mm). For this reason, the escape probability for neutrons in the fuel is not greatly enhanced as the fuel-moderator material is heated. In the TRIGA-FLIP fuel the temperature hardened spectrum is used to decrease reactivity through interaction with a low energy resonance material. Thus erbium with its double resonance at ~ 0.5 eV is used in the TRIGA-FLIP fuel both as a burnable poison and as a material to enhance the prompt negative temperature coefficient. The neutron spectrum shift, pushes more of the thermal neutrons into the ^{167}Er resonance as the fuel temperature increases, as illustrated in Fig. 4, where the cold and hot core spectra are plotted along with the energy-dependent absorption cross section for ^{167}Er . As with a standard TRIGA core, the temperature coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator and thus fuel and solid moderator temperatures rise simultaneously producing the temperature-dependent spectrum shift.

For a TRIGA-FLIP core the effect of cell structure on the temperature coefficient is small. Almost the entire coefficient comes from temperature dependent changes in f and η (thermal utilization factor and reproduction factor) within the core and $\sim 80\%$ of this effect is independent of the cell structure. The calculated beginning of life prompt negative temperature coefficient is shown in Figure 5.

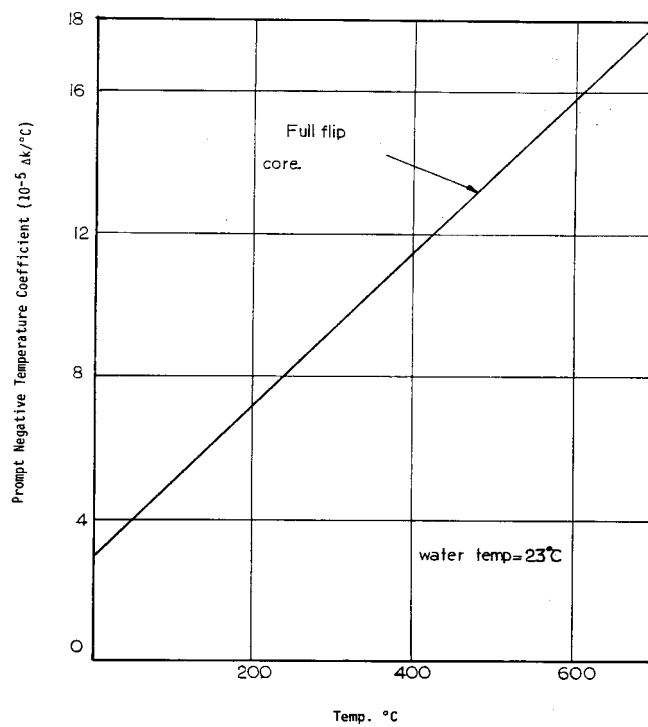


Figure 5: Prompt negative temperature coefficient vs. temperature for a typical TRIGA core.

It is highly temperature dependent and increases rapidly as a function of fuel temperature because of the steadily increasing number of thermal neutrons being pushed into the ^{167}Er resonance. The temperature-dependent character of the temperature coefficient of a TRIGA-FLIP core is advantageous in that a minimum reactivity loss is incurred in reaching normal operating temperatures, but any sizeable increase in the average core temperature results in a

suitably increased prompt negative temperature coefficient to act as a shut-down mechanism. (From an operational point of view, an average number to use for a prompt negative temperature coefficient for a typical TRIGA core is $\sim 1 \text{ } \rho/^{\circ}\text{C}$ [$7 \times 10^{-5} \text{ } \delta k/k \text{ per } ^{\circ}\text{C}$]).

Burn-up calculations made by GA indicate that after $\sim 3000 \text{ MW days}$ of operation the ^{235}U concentration averaged over the core will be $\sim 67\%$ and the ^{167}Er concentration will be $\sim 33\%$ of the beginning-of-life values. The temperature coefficient for such a core, including fission products will be less temperature dependent than the beginning-of-life coefficient because of the sizeable loss of ^{167}Er and the resulting increased transparency of the $\sim 0.5 \text{ eV}$ resonance region to such neutrons.

As a matter of interest, burn-up calculations also indicate that the FLIP cores will initially lose some reactivity for a short time and then will gain $\sim 1.5\%$ in reactivity during the first 1500 MW days of operation (see Fig. 6). This reactivity increase during the first half of core life is due to a mismatch between erbium and uranium burn-up. The reactivity gain due to erbium loss is somewhat greater than the reactivity loss due to ^{235}U burn-up.

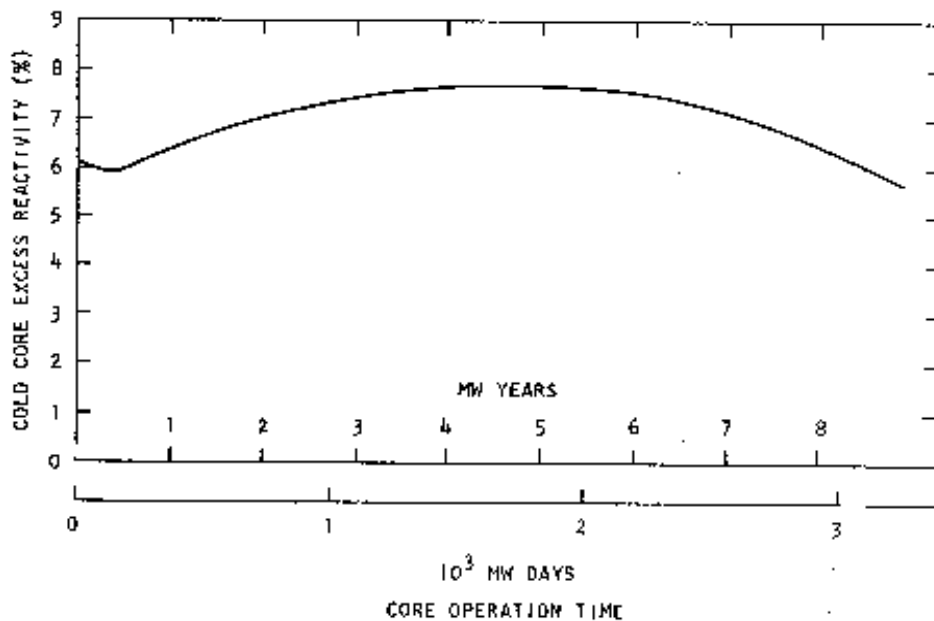


Figure 6: Cold core excess reactivity vs. core operation time.

The addition of Erbium to the fuel-moderator matrix in no way compromises the chemical, mechanical or nuclear characteristics of the fuel in comparison with its extensively demonstrated characteristics as standard TRIGA fuel. The major changes and effects resulting from the addition of the erbium are:

- a) Allowance of higher uranium enrichment to produce a much longer-burning core than standard TRIGA fuel.
- b) A shortened prompt neutron lifetime compared with a standard core ($\sim 19 \text{ } \mu\text{s}$ vs. $39 \text{ } \mu\text{s}$). This results from the heavier core loading of uranium and the erbium, but has no direct

effect on steady-state operation. The reduced lifetime does give shorter periods and higher peak powers during a transient when compared with a core with standard fuel (assuming equal reactivity insertions and an equivalent temperature coefficient). However, the reduced lifetime has no influence on the energy release or on fuel temperatures produced during a pulse. Since the operational limits of the system are based on fuel temperature, and since this change in neutron lifetime does not affect fuel temperatures, there is no resulting compromise in the safety of the system.

- c) A change in shape of the prompt negative temperature coefficient as a function of temperature. This is the most significant change in a core with FLIP fuel vs. standard fuel, and the desire to have the greatest possible confidence in this parameter was the reason for extensive calculations and the conducting of an in-core test program. Results from these studies give a design prompt negative temperature coefficient which becomes increasingly more negative as the core temperature increases (compared with the more nearly constant value for standard fuel), and has an average value over the operational core temperature range (20°C - 700 °C) which is essentially the same as the coefficient for a standard core. For large temperature changes, the temperature coefficient is on the same order as the standard fuel. The nearly linear response of core reactivity due to a temperature change requires the operator to provide more control rod compensation in reaching full power when compared to a standard core but this prompt temperature rapidly inserts a negative reactivity proportional to the temperature change during any rapid power transient (see Fig. 7 for a graphic example of this inherent characteristic).

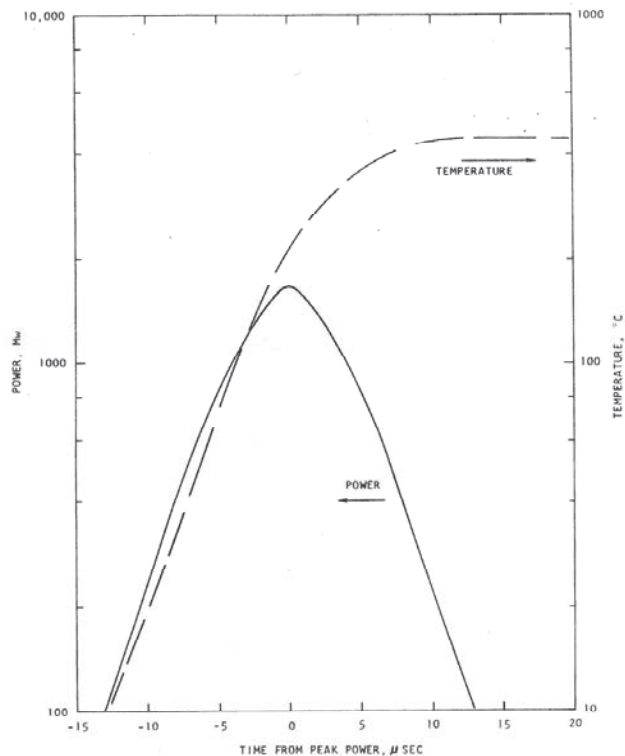


Figure 7: Calculated peak power and fuel temperature rise vs. time during power excursion.

1.11 Pool temperature coefficient of reactivity

The pool temperature coefficient (or sometimes called bath coefficient) describes the change in reactivity associated with a change in pool temperature and is defined as the change in reactivity per unit change in pool temperature. Whenever the reactivity and the temperature change in the same direction, the pool coefficient is positive. If the reactivity and the temperature change in opposite directions, the pool coefficient is negative. The pool coefficient has been measured experimentally in TRIGA reactors and has been found to be small but positive in the less than room temperature region, becoming negative at higher temperatures. The measured pool coefficient in a TRIGA averages $\sim 0.05 \text{ } \beta / ^\circ\text{C}$ ($1/2 \text{ } \beta$ for each $10 \text{ } ^\circ\text{C}$ change). Thus, very little reactivity change is a direct result of *pool temperature*. The temperature of the pool water does not change significantly during a prompt transient. The prompt transient behaviour of the reactor is determined primarily by changes in fuel temperature.

1.12 Void coefficient of reactivity

If water in the core is replaced with a very low density material, i.e., steam, gases, etc., a negative reactivity change will occur that is called the void coefficient of reactivity. The void coefficient of reactivity is defined as the change in reactivity per percent void of core water. The void coefficient is a function of position in the core, water fraction, and water temperature. This quantity is useful in studying the effects on reactivity produced by the formation of voids within the core, as could occur, for example, with the production of steam and/or radiolytic gases such as hydrogen or oxygen along the surface of a fuel element.

1.13 Fuel element reactivity worths

Perturbations of reactivity resulting from physical changes in the core and reflector are of importance to the safety of the reactor. The reactivity changes shown in Table 5 have been measured by removing one fuel element at a time. This gives a reactivity value relative to the water-filled fuel location in an otherwise completely loaded core.

Core position	Reactivity value in [β]
B	130
C	90
D	55
E	42
F	37
F (graphite element)	8

Table 5: Fuel element reactivity worth in cents measured at a TRIGA Mark II reactor.

1.14 Power coefficient of reactivity

The power coefficient describes the change in reactor power associated with a change in reactivity and is defined as the amount of reactivity needed to change the reactor power by one kilowatt.

TRIGA reactors have a strong negative temperature coefficient, and a certain amount of reactivity is needed to overcome this effect during reactor power increases. The variable temperature coefficient associated with the FLIP fuel precludes a constant power coefficient. However, an overall average power coefficient of $\sim 25 \text{ } \phi/\text{kW}$ can be used for a TRIGA with reasonable results. See Figure 8 for a typical TRIGA reactivity loss versus power curve.

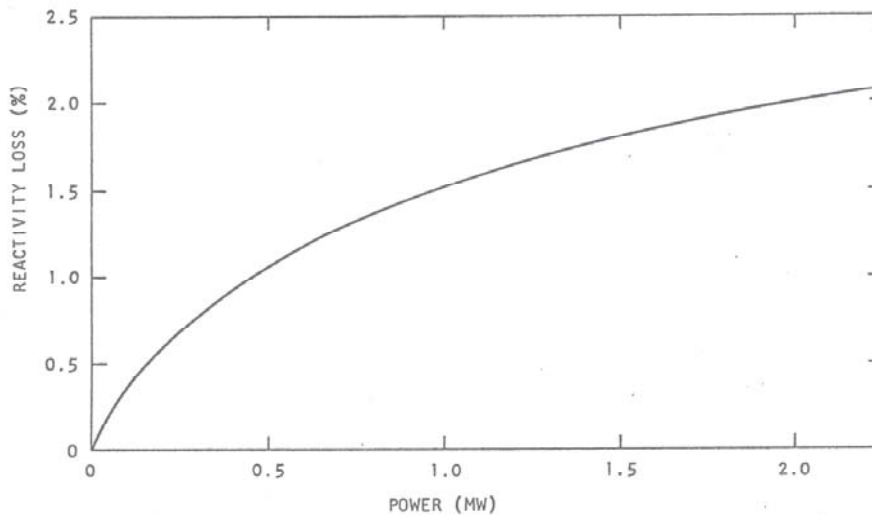


Figure 8: Typical reactivity loss versus power.

If an operator wants to know how much total reactivity would have to be added to the just critical core in order to reach 1000 kW, he could use the power coefficient (α_p):

$$\begin{aligned} \alpha_p &= \sim 0.25 \text{ } \phi/\text{kW}; & P &= 1000 \text{ kW} \\ (P)(\alpha_p) &= \text{reactivity } (\phi) \\ (1000 \text{ kW})(0.25 \text{ } \phi/\text{kW}) &= 250 \phi = \$ 2.50. \end{aligned}$$

This reactivity can be expressed in $\% \delta k/k$:

$$(\$)(\beta) = \rho; (\$ 2.50)(0.007) = 0.0175 \delta k/k = 1.75\% \delta k/k.$$

1.15 Reactor nuclear parameters

The parametric values listed in Table 6 are estimated for TRIGA reactors from calculations correlated with measurements made on other reactors of the same generic type. Thus, although each reactor is slightly different due to its core configuration, it is felt that the values given here are reasonable extrapolations. No unqualified estimate of the accuracy of the information presented herein has been made but, in general, it is felt that it should be good within about 5%.

Cold clean critical core	57 elements
	2.055 kg ²³⁵ U
Operational loading after 40 years of operation	90 elements
	3.52 kg ²³⁵ U
Prompt neutron lifetime	43×10 ⁻⁶ s
Effective delayed neutron fraction	β _{eff} = 0.0070
Prompt negative temperature coefficient	-1.0×10 ⁻⁴ δk/k °C

Table 6: Nuclear parameters of a typical TRIGA Mark II reactor

1.16 TRIGA Pulsing Performance

For transient (pulsing) operation the reactor is first brought to criticality at a low power level (usually 10 to 300 watts). This is accomplished by withdrawing only the standard control rods and leaving the transient rod in the core. Some facilities bring the reactor critical with all controls rods and then lower the transient rod to produce a subcritical pulse. Reactivity insertion is calculated by evaluating the differential transient rod worth curves and selecting the transient rod height. The pulse is then produced by rapidly firing the transient rod out of the core pneumatically within a fraction of a second. The reactor power will increase sharply to a value that will result in a fuel temperature increase which will compensate for the excess reactivity inserted. At this point the peak reactor power will occur. The fuel temperature will continue to rise resulting in an increasing loss of reactivity, and the reactor power will decrease to a comparatively low level. This final power level depends on the reactivity insertion and the heat transfer characteristics of the fuel element. The power level after the transient is referred to as the after-pulse tail and is produced from the power decay of the original pulse and the delayed neutron affects. After initiation of the transient pulse, the reactor is automatically scrammed after a few seconds to clip the power tail for operational convenience. A typical TRIGA reactor pulse is shown in Figure 9 for a reactivity insertion of \$ 1.994. A pulse for a FLIP reactor will be narrower with a higher peak power for the same reactivity insertion.

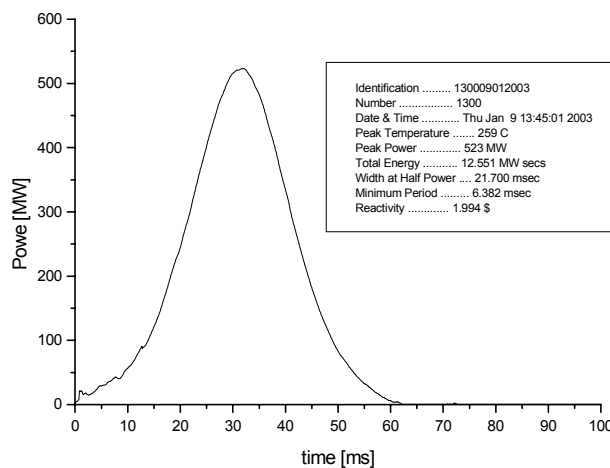


Figure 9: Typical TRIGA reactor power pulse.

1.17 The Fuchs-Nordheim model and pulsing characteristics

The performance of a TRIGA reactor pulsing beyond prompt critical can be quite well described by a space-independent kinetics model in which heat transfer and delayed neutrons are neglected. In this model, usually called the Fuchs-Nordheim model, the reactor parameters that appear are:

- ℓ = prompt neutron lifetime (s),
- α = prompt negative temperature coefficient (1/k dk/dt),
- θ = fuel temperature,
- h = ratio of peak to average fuel temperature in core,
- C = total heat capacity of metal in core.
- T = reactor period in seconds

If a sudden insertion of reactivity beyond prompt critical (δk_p) is made, the reactor power will initially rise exponentially as $\exp(t/T)$ where the period T is given by $\ell / \delta k_p$. This period will change when appreciable temperature is generated by the pulse. Assuming a constant heat capacity and a shutdown mechanism that is prompt and invariant with temperature, the reactor temperature will rise by $\delta k_p / \alpha$, until the reactivity inserted beyond prompt critical is just compensated by the fuel temperature rise. The power will then fall to very low values with a temperature overshoot by about a factor of two beyond the temperature at the peak power value.

The total energy release in the pulse will be:

$$E = \frac{2}{bT} = \frac{2C \delta k_p}{\alpha} .$$

The peak power will be:

$$P_{\max} = \frac{1}{2bT^2} = \frac{(C \delta k_p)^2}{2\alpha \ell} ,$$

and the width at half-maximum power of the pulse will be $\sim 3.5 T$. The shutdown coefficient, b , is given by

$$b = \frac{\alpha}{\ell C} .$$

The average fuel temperature is:

$$\bar{\theta} = \frac{E}{C} = \frac{2\delta k_p}{\alpha} ,$$

and the maximum fuel temperature is:

$$\hat{\theta} = h\bar{\theta} = \frac{hE}{C} .$$

It can be seen that the peak power varies as the square, and the energy release varies linearly with the prompt reactivity inserted (δk_p). The heat capacity of the TRIGA fuel element increases with temperature, and a Fuchs-Nordheim analysis with variable heat capacity has been derived by R. Scalettar* and is used for current TRIGA kinetics analysis. (For a \$ 3.00 pulse in the GA Advanced TRIGA Prototype Reactor, a difference of ~20% in peak power and energy release is noted between a constant versus a variable heat capacity analysis, with the higher values resulting from the constant heat capacity model.) The simple Fuchs-Nordheim model relationships are helpful in understanding the pulsing characteristics of the TRIGA reactors since the influence of each parameter on the pulse behaviour of the system is apparent. Figure 10 illustrates a power pulse from the GA TRIGA Mark I reactor using standard TRIGA fuel (aluminium clad) and compares it with a Fuchs-Nordheim model computation.

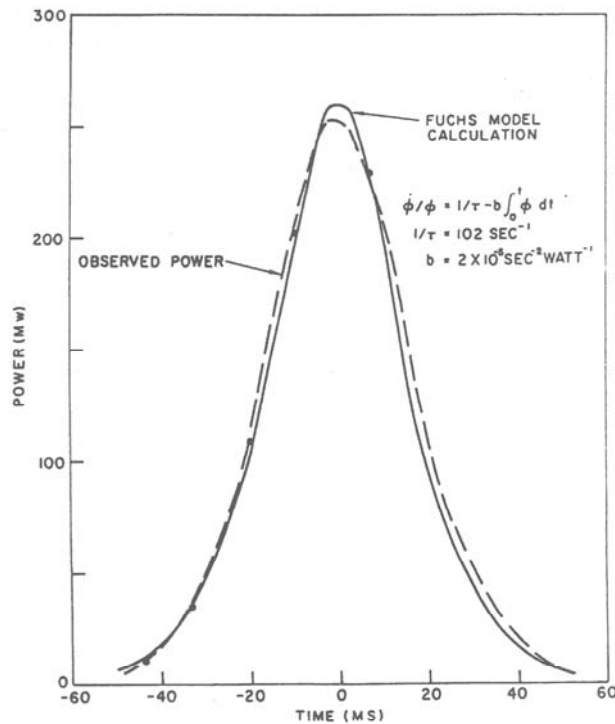


Figure 10: Power level as a function of time for GA TRIGA Mark I transient.

*R. Scalettar: "The Fuchs-Nordheim Model with Variable Heat Capacity". General Atomic Report GA-3416, General Atomic Division of General Dynamics Corporation, 13. 8. 1962.

1.18 How to estimate the peak power of a pulse

It is difficult to calculate peak power for any given pulse using the methods and formulas described so far. This is especially true when using the 70% enriched FLIP fuel. The reason is that there are too many variables involved. Some of these are:

1. Variable heat capacity
2. Variable negative temperature coefficient
3. Questionable reactivity insertions
4. Core history and xenon concentration
5. Transient rod air pressure at the instant of the pulse (affects rod acceleration).

Computer program would be required to correlate precise predictions for peak power vs. reactivity insertion for various pulses. Routine operational estimates of the pulse size can be made by using a ratio with known quantities. The expression most commonly used is:

$$\frac{p_1}{(\delta k_{p_1})^2} = \frac{p_2}{(\delta k_{p_2})^2}$$

p_1 = last known pulse size (MW)

p_2 = projected pulse size (MW)

δk_{p_1} = prompt reactivity (\$-1) for the last known pulse
(Remember this quantity is squared in the expression.)

δk_{p_2} = prompt reactivity selected for the projected pulse.

Example

Question: What size pulse will a \$ 2.10 reactivity insertion produce?

Answer: Look in the log book for the last pulse parameters;

p_1 = 2100 MW

δk_1 = \$ 2.25 (Remember that prompt reactivity is reactivity greater than \$ 1.00.)

δk_{p_1} = \$ 1.25

given:

δk_2 = \$ 2.10

δk_{p_2} = \$ 1.10

$$\frac{p_1}{(\delta k_{p_1})^2} = \frac{p_2}{(\delta k_{p_2})^2}$$

$$\frac{2100}{(1.25)^2} = \frac{p_2}{(1.10)^2}$$

$$\frac{(2100)(1.21)}{1.56} = p_2$$

$$1629 = P_2$$

1629 MW peak power.

Energy releases can be estimated in the same way by using the expression:

$$\frac{E_1}{\delta k_{p1}} = \frac{E_2}{\delta k_{p2}},$$

E = energy released (MWs).

Notice in the energy expression that the prompt reactivity is not squared.

1.19 Reactivity accident

It has been shown in various TRIGA analyses and confirmed by an adiabatic model that accidental pulsing from a high power level, e.g. 1 MW, would actually result in lower peak fuel temperatures as compared to regular pulses initiated by the same reactivity insertion from a low power level. This is due to the larger prompt negative temperature coefficient at the higher fuel temperatures, and also due to the fact that the feedback effects are significant from the moment the transient rod starts to withdraw. Thus, a much smaller pulse is experienced and the adiabatic fuel temperature peak is less than in a comparable pulse from low power, in spite of the higher initial fuel temperatures.

1.20 TRIGA Safety Compared to Other Research Reactors

1.20.1 Summary

The most pertinent information regarding safety of research reactors comes from specific test data developed in experiments at the National Reactor Testing Station (now the Idaho National Engineering Laboratory) SPERT facilities in Idaho, and at the GA TRIGA facilities in San Diego, California. An analysis of these data clearly indicates that the TRIGA reactor has the greatest inherent safety of any available megawatt level research reactor. The safety of the TRIGA is due to the large prompt negative temperature coefficient of reactivity, which is an intrinsic characteristic of the standard U-Zr-H fuel. The TRIGA temperature coefficient acts independently of any external controls to assure safe and reliable self-shutdown in the event of an accidental reactivity insertion.

1.20.2 Review of Some Reactor Safety and Failure Tests

It was shown in the SPERT tests with aluminium-uranium alloy plate type fuel with light water cooling and moderation, that a reactivity insertion of about \$ 2.40 results in fuel plate damage and that a \$ 3.60 reactivity insertion results in a violent destruction of the core. Tests on plate-type aluminium-uranium alloy fuel using D₂O moderator and coolant show that fuel melting will occur with about \$ 3.50 insertions. In one SPERT test, using a core loading of low-enrichment stainless clad UO₂ fuel, a reactivity insertion of about \$ 2.70 resulted in a rupture-failure of two fuel elements with an additional 140 to 150 elements bowed by a 3.4 bar to 4.1 bar pressure surge.

The above reactivity insertions resulted in damage or failure with a single pulse, indicating that considerably smaller insertions may cause failure for repeated routine pulses. By contrast, TRIGA pulse experiments have shown that the TRIGA standard fuel can safely withstand

numerous \$ 4.00 reactivity insertion pulses with complete safety. Further, the TRIGA can handle much larger pulses. In this regard, it is significant that the GA TRIGA Prototype Reactor facility was licensed for pulses with up to \$ 5.00 reactivity insertions. The facility is located in a conventional building adjacent to the GA Research Laboratories near San Diego, California. The GA TRIGA reactors in San Diego have been shutdown at this time and are in the process of decommissioning.

It has been demonstrated that TRIGA reactors can operate with the dual capability for up to 5 MW steady-state and 9000 MW routine pulses. Steady operation up to 1.5 MW is conducted with natural convection core cooling, and \$ 4.00 pulses to 5000 MW are run routinely at rates up to 12 pulses per hour.

APPENDIX 1

TRIGA Mark I Reactor

General Data

Below ground; fixed core; graphite reflector.
100 kW to 2,000 kW steady state power level.
Up to 6,400,000 kW pulsing power level.
 8.0×10^{13} n/cm² s maximum thermal flux (<0.21 eV) at 2000 kW.
 9.6×10^{13} n/cm² s maximum fast flux (10 keV) at 2000 kW.
UZrH_{1,6} fuel elements using uranium enriched to 20%.

The below-ground TRIGA Mark I is extremely simple in physical construction. It is a graphite-reflected core capable of operating up to 2,000 kW steady state and pulsing routinely and reproducibly with reactivity insertions up to 3.2% $\delta k/k$. The reactor core rests at the bottom of an aluminium tank. Surrounding earth and demineralised water provide the required radial and vertical shielding. No special containment building is necessary and installation in existing buildings is often feasible. The Mark I TRIGA can be installed in a circular pool or in a large, oblong pool for greater experimental access to the reactor core. Core cooling is achieved through natural convection, eliminating the need for an expensive and restrictive forced cooling system.

TRIGA Mark II Reactor

General Data

Above ground; fixed core; graphite reflector; 4 beam tubes; thermal column.
250 kW to 2,000 kW steady state power level with natural convection cooling (3,000 kW with forced cooling).
Up to 6,400,000 kW pulsing power level.
 8.0×10^{13} n/cm² s maximum thermal flux (<0.21 eV) at 2,000 kW.
 9.6×10^{13} n/cm² s maximum fast flux (>10 keV) at 2,000 kW.
UZrH_{1,6} fuel elements using uranium enriched to 20%.

The TRIGA Mark II, which provides experimental capabilities greater than the TRIGA Mark I, is an above-ground fixed-core research reactor. Its core, identical to that of the TRIGA Mark I, is located in a pool surrounded by a concrete shield structure which is above the reactor room floor. The pool water provides natural convection cooling. The Mark II features include:

- Thermal Column - a graphite thermal column, 1.2 m by 1.2 m by 1.65 m, extending from the reflector through the concrete structure, provides a source of well-thermalised neutrons suitable for physical research or biological irradiation. A movable high-density concrete door with a removable 20-cm concrete plug shields the outer face of the column.

- Beam Ports - four 15-cm-diameter horizontal beam ports, extending through the concrete shield to the face of the reflector, permit the extraction of core radiations, or the insertion of specimens for irradiation. Two of the beam tubes extend radially to the reflector, a third penetrates the reflector to the core's edge, and the fourth is tangent to the core.

TRIGA Mark III Reactor

General Data

Above ground; movable core; beam tubes; thermal columns; exposure room; removable core-irradiation facilities.

1,000 kW to 2,000 kW steady state power level with natural convection cooling (3,000 kW with forced cooling).

Up to 6,400,000 kW pulsing power level.

6.6×10^{13} n/cm² s maximum thermal flux (<0.21 eV) at 2,000 kW.

6.2×10^{13} n/cm² s maximum fast flux (>10 keV) at 2,000 kW.

UZrH_{1,6} fuel elements using uranium enriched to 20%.

The TRIGA Mark III, the most versatile of the standard TRIGA series, is available in either above- or below-ground configurations. Its water-reflected movable core greatly increases the reactor's flexibility. The core can be moved to one end of the pool for experiments in an adjacent dry, walk-in exposure room, or to the opposite end for experiments involving the thermal column and beam ports. The ability to move the radioactive core away from the experimental facilities greatly eases set-up of experiments.

The reactor tank is approximately 7.5 m long by 7.5 m deep, with a maximum width of 3 m at the center. Because it is natural convection cooled up to 2000 kW, the reactor can be operated anywhere in the pool. Reactor facilities include:

- Two Thermal Columns with internal void - a graphite thermal column, 1.2 m by 1.2 m by 3 m, extends from the periphery of the reactor core through the concrete shield structure. A 0.9 m by 0.9 m by 1.05 m Hohlraum space is provided in this horizontal thermal column with a vertical thermal column directly above. Four ports through the concrete shielding allow access to the two thermal columns.
- Beam Ports - four 15-cm-diameter horizontal beam ports penetrate the concrete shield and the reactor pool water to the edge of the core shroud, and two 20-cm-diameter through beam ports intersect in the thermal column adjacent to the core.
- Walk-in Exposure Room - This room is 3 m wide by 3.6 m long by 2.9 m high and easily accommodates very large experiments. Access to the room is provided by several 15-cm-diameter conduits and a motor-driven concrete door.

INTERNATIONAL

Country	Location	TRIGA Model	Power Steady State kW(t)	Pulsing MW(t)	Status	Initial Criticality
Austria	Federal Ministry of Education, <i>Wien</i>	Mark II	250	250	Operating	1962
Bangladesh	Atomic Energy Research Establishment, <i>Dhaka</i>	Mark II	3,000	3,900	Operating	1986
Brazil	University of Minas Gerais, <i>Belo Horizonte</i>	Mark I	100		Operating	1960
Colombia	Institute of Nuclear Science & Alternative Energy, <i>Bogota</i>	Conversion	100		Operating	1997
England	Imperial Chemical Industries, <i>Billingham</i>	Mark I	250		Decommissioned	1971
Finland	The State Institute for Technical Research, <i>Helsinki</i>	Mark II	250	250	Operating	1962
Germany	Univ. of Frankfurt, <i>Frankfurt</i> Medical College Hannover, <i>Hannover</i> German Cancer Research Center, <i>Heidelberg</i> Johannes Gutenberg Univ., <i>Mainz</i> Association for Radiation Research, <i>Munich</i>	Conversion Mark I Mark I Mark II Mark III	1,000 250 250 100 1,000	250 2,000	Decommissioned Shutdown Shutdown Operating Decommissioned	1977 1973 1966 1965 1972
Indonesia	National Atomic Energy Agency, <i>Bandung</i> National Atomic Energy Agency, <i>Yogyakarta</i>	Mark II Mark II	2000 250		Operating Operating	1997 1979
Iran	Nuclear Research Center, <i>Tehran</i>	Conversion	5,000		Suspended	
Italy	University of Pavia, <i>Pavia</i> ENEA Cassaccia Research Center, <i>Rome</i>	Mark II Mark II	250 1,000	250	Operating Operating	1965 1960
Japan	Japan Atomic Energy Research Institute, <i>Tokai-mura</i> Musashi Institute of Technology, <i>Tokyo</i> Rikkyo University, <i>Yokosuka</i>	ACPR Mark II Mark II	300 100 100	22,000	Operating Operating Operating	1975 1963 1961
Korea	Korea Advanced Energy Research Institute, <i>Seoul</i> Korea Advanced Energy Research Institute, <i>Seoul</i>	Mark II Mark III	250 2,000	2,000	Shutdown Shutdown	1962 1972
Malaysia	Malaysian Inst. for Nuclear Technology, <i>Kuala Lumpur</i>	Mark II	1,000	1,200	Operating	1982
Mexico	National Institute for Nuclear Research, <i>Mexico City</i>	Mark III	1,000	2,000	Operating	1968
Morocco	National Institute for Nuclear Science & Energy, <i>Rabat</i>	Mark II	2,000		Under Construction	2005
Philippines	Philippine Atomic Energy Commission, <i>Quezon City</i>	Conversion	3,000	1,000	Operating	1988
Romania	Institute for Nuclear Research, <i>Pitesti</i> Institute for Nuclear Research, <i>Pitesti</i>	ACPR MPR 16	500 14,000	22,000	Operating Operating	1979 1979
Slovenia	Jozef Stefan Nuclear Institute, <i>Ljubljana</i>	Mark II	250		Operating	1966
Taiwan	National Tsing Hua University, <i>Taipei</i>	Conversion	1,000		Operating	1977
Thailand	Office of Atoms for Peace, <i>Bangkok</i> Ongkharak Nuclear Research Center, <i>Bangkok</i>	Conversion MPR 10	1,000 10,000		Operating Under Construction	1977 2005
Turkey	Technical University of Istanbul, <i>Istanbul</i>	Mark II	250	250	Operating	1979
Viet Nam	Institute of Nuclear Research, <i>Dalat</i>	Mark II	250		Decommissioned	1963
Zaire	Nuclear Science Commission, <i>Kinshasa</i> Nuclear Science Commission, <i>Kinshasa</i>	Mark II Mark I	1,000 50	1,600	Operating Shutdown	1972 1959

UNITED STATES

Country	Location	TRIGA Model	Power Steady State kW(t)	Pulsing MW(t)	Status	Initial Criticality
Arizona	University of Arizona, <i>Tucson</i>	Mark I	250	300	Operating	1958
Arkansas	Arkansas Technical Univ., <i>Russelville</i>	Mark I	250	2,000	Suspended	
California	General Atomic, <i>San Diego</i>	Mark I	250	1,000	Shutdown	1958
	General Atomic, <i>San Diego</i>	Mark F	1,500	6,400	Shutdown	1960
	General Atomic, <i>San Diego</i>	Mark III	2,000		Decommissioned	1966
	University of California, Davis, <i>Sacramento</i>	Mark II	2,300	1,200	Operating	1990
	Norair Division of Northrop Corp., <i>Hawthorne</i>	Mark F	1,000	1,600	Decommissioned	1963
	University of California, <i>Berkeley</i>	Mark III	1,000	1,200	Decommissioned	1966
Colorado	University of California, <i>Irvine</i>	Mark I	250	250	Operating	1969
	Aerotest Operation, <i>San Ramon</i>	Conversion	250		Operating	1965
Colorado	U.S. Geological Survey, <i>Denver</i>	Mark I	1,000	1,200	Operating	1969
Idaho	Argonne Nat'l. Lab. West (HFEF, INEL), <i>Idaho Falls</i>	Conversion	250		Operating	1977
Illinois	University of Illinois, <i>Urbana</i>	Mark II	1,500	6,500	Shutdown	1960
	University of Illinois, <i>Urbana</i>	LOPRA	10		Shutdown	1971
Kansas	Kansas State University, <i>Manhattan</i>	Mark II	250	250	Operating	1962
Maryland	Harry Diamond Labs. (U.S. Army), <i>Forest Glen</i>	Mark F	250	1,000	Decommissioned	1961
	AFRRI, <i>Bethesda</i>	Mark F	1,000	3,300	Operating	1962
	University of Maryland, <i>College Park</i>	Conversion	250		Operating	1974
Michigan	The Dow Chemical Company, <i>Midland</i>	Mark I	300		Operating	1967
	Michigan State University, <i>East Lansing</i>	Mark I	250		Decommissioned	1969
Nebraska	Veterans Administration Hospital, <i>Omaha</i>	Mark I	18		Shutdown	1959
New Mexico	Sandia Nat'l. Laboratories, <i>Albuquerque</i>	ACPR	600	12,000	Operating	1967
New York	Columbia University, <i>New York</i>	Mark II	250	250	Suspended	
	Cornell University, <i>Ithaca</i>	Mark II	500	250	Shutdown	1962
Oregon	Oregon State University, <i>Corvallis</i>	Mark II	1,000	3,200	Operating	1967
	Reed College, <i>Portland</i>	Mark I	250		Operating	1968
Pennsylvania	Pennsylvania State University, <i>University Park</i>	Mark III	1,000	2,000	Operating	1965
Puerto Rico	Puerto Rico Nuclear Center, <i>Mayaguez</i>	Conversion	2,000		Decommissioned	1972
Texas	Texas A&M University, <i>College Station</i>	Conversion	1,000	2,000	Operating	1968
	University of Texas, <i>Austin</i>	Mark I	250		Decommissioned	1963
	University of Texas, <i>Austin</i>	Mark II	1,100	1,600	Operating	1992
Utah	University of Utah, <i>Salt Lake City</i>	Mark I	250		Operating	1975
Washington	Westinghouse-Hanford-300 Area, <i>Richland</i>	Mark I	1,000		Shutdown	1977
	Washington State University, <i>Pullman</i>	Conversion	1,000	2,000	Operating	1967
Wisconsin	University of Wisconsin, <i>Madison</i>	Conversion	1,000	2,000	Operating	1967

APPENDIX 2

Technical data of a typical TRIGA Mark II reactor

1. REACTOR CORE

fuel-moderator material	8 wt% uranium 91 wt% zirconium 1 wt% hydrogen
uranium enrichment	20% uranium-235
fuel element dimensions	3.75 cm in diameter 72.24 cm in length
cladding	0.76 mm aluminium or 0.51 mm steel
active core volume	max. 49.5 cm diameter, 35.56 cm high
core loading	2.3 kg of uranium-235

2. REFLECTOR

material	graphite with aluminium cladding
radial thickness	30.5 cm
top and bottom thickness	10.2 cm

3. CONSTRUCTION

reactor mounting	heavy and standard concrete 6.55 m high 6.19 m wide 8.76 m long
reactor tank	1.98 m in diameter 6.40 m in depth

4. SHIELDING

radial:	30.5 cm of graphite; 45.7 cm of water and at least 206 cm of heavy concrete
vertical:	above the core 4.90 m of water and 10.2 cm graphite; underneath the core 61.0 cm water, 10.2 cm graphite and at least 91 cm standard concrete.

5. IRRADIATION DEVICES

- (1) four beam holes 15.2 cm in diameter
- (2) one central irradiation tube (middle of core)
- (3) five reflector irradiation tubes
- (4) one pneumatic transfer system (near core edge)
- (5) a thermal column with cross section 1.22x1.22 m and length 1.68 m

- (6) experimental tank with surface area 2.44x2.74 m and depth 3.66 m; connected to the reactor by means of a neutron radiography collimator 0.61x0.61 m in cross section and 1.22 m long.

6. CONTROL SYSTEM

- Two boron carbide control rods with electric motor and rack and pinion drive;
- One boron carbide pulse rod with compressed air drive (5 bars);
- Maximum reactivity insertion rate - time rate of change (excluding pulse operation): 0.04% $\delta k/k$ per second
- Total rod worth about 4.8% $\delta k/k$.

7. CHARACTERISTICS IN CONTINUOUS OPERATION

Thermal power output:	250 kW
Fuel element cooling:	natural convection of the tank water below 100 kW, pump circulation cooling above 100 kW
tank water cooling:	heat exchanger
thermal flux:	$1 \times 10^{13} \text{ cm}^{-2}\text{s}^{-1}$ in the central irradiation tube $1.7 \times 10^{12} \text{ cm}^{-2}\text{s}^{-1}$ in the irradiation tubes
prompt temperature coefficient:	$-1.2 \times 10^{-4} \delta k/k \text{ } ^\circ\text{C}$
mean prompt neutron lifetime:	$6.0 \times 10^{-5} \text{ s}$.

8. CHARACTERISTICS IN TRANSIENT OPERATION

peak power	250 MW
prompt pulse energy yield	10 MW s
prompt pulse lifetime	40 ms
total energy yield	16 MW s
minimal period	10 ms
maximum reactivity insertion	1.6% $\delta k/k = 2\beta$
maximum repetition frequency	12/h
number of fissions during a pulse	3×10^{17}
maximum fuel temperature:	
during the pulse	240 $^\circ\text{C}$
9 seconds after the pulse	360 $^\circ\text{C}$

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