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Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors

Prepared by S. C. Hawley, R. L. Kathren

Pacific Northwest Laboratory
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ABSTRACT

Credible accidents were developed and analyzed for TRIGA and TRIGA-fueled reactors. The only potential for offsite exposure appears to be from a fuel-handling accident that, based on highly conservative assumptions, would result in dose equivalents of ≤ 1 mrem to the total body from noble gases and ≤ 1.2 rem to the thyroid from radioiodines. Credible accidents from excess reactivity insertions, metal-water reactions, lost, misplaced, or inadvertent experiments, core rearrangements, and changes in fuel morphology and ZrH_x composition are also evaluated, and suggestions for further study provided.

SUMMARY AND CONCLUSIONS

Credible accidents for TRIGA and TRIGA-fueled reactors were evaluated in the light of contemporary knowledge and the long operating history of this class of reactors. Seven categories of accidents were analyzed:

- excess reactivity insertions
- metal-water reactions
- lost, misplaced, or inadvertent experiment
- mechanical rearrangement of the core
- loss-of-coolant accident
- changes in fuel morphology and ZrH_x composition
- fuel handling.

The available licensed excess reactivity at TRIGA reactors is insufficient to produce a pulse or transient with sufficient energy to heat the fuel to the point at which failure could occur. With the exception of a core containing the $ZrH_{1.7}$ fuel, an additional 1% $\Delta k/k$ would be required as a minimum for the fuel to reach its limiting temperature. Hence, cladding failure or fuel melting can not occur. In the case of the reactor using $ZrH_{1.7}$ fuel, however, limiting temperatures could be approached.

The necessary conditions for a metal-water reaction would have to be produced by some large external force, and any credible metal-water reaction would be of much smaller magnitude and hence of secondary impact. Under certain circumstances, a lost, misplaced, or inadvertent experiment could cause gas pressures or noxious fumes from radiolytic or thermal decomposition, but an offsite hazard from these or any other effects of a credible accident from this category is remote.

Mechanical rearrangement of the core might result from extraordinary natural phenomena such as an earthquake or similar large forces. Even if this resulted in a criticality excursion in combination with a major loss-of-coolant, the activity released would not be sufficient to present an offsite hazard.

The long operating history of TRIGA type reactors raises questions regarding the effects of numerous pulses on the fuel. Swelling has been reported in FLIP fuels subjected to pulsing and 291 MWd of steady-state operation, and cracking of the U-ZrH_x may occur along with gas pressure buildup from dehydriding. Given the proper circumstances, this could lead to cladding failure and fission product release. The radiological consequences of such an accident would be small and produce no offsite hazard. Although the experimental data available to date suggest that the likelihood of this type of accident is small, additional experimental work may be needed.

The only credible accident that might produce significant offsite doses involves a fuel element that sustains major damage during a fuel-handling accident. The fuel element is assumed to contain 4% of the core inventory and the accident to occur 48 h postshutdown following operation of the reactor for 365 MWd. The calculated dose equivalents to the maximum exposed individual would be ≤ 1 mrem to the total body from noble gases and ≤ 1.2 rem to the thyroid from radioiodines. These calculations are based on highly conservative assumptions and therefore represent an upper limit on exposure.

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INTRODUCTION

The TRIGA (Training, Research, Isotope production, General Atomic) class of research reactors is one of the most prevalent in the world. As of 1981, 57 units were known to be operational world-wide; 25 are currently licensed in the United States (Table 1). The continuing popularity of these small multipurpose reactors stems to a large degree from their demonstrated versatility, reliability, and inherent safety features, coupled with their low capital and operating costs.

The basic TRIGA design was based on four principles: safety, simplicity, utility, and cost (Koutz et al. 1958). Of these, safety was given the greatest emphasis. Thus, when the first TRIGA reactor commenced operation in 1958, it was equipped not only with the usual safety systems based on scrams, but it also used zirconium hydride (ZrH_x) moderated fuel to produce a fail-safe, inherent, negative temperature coefficient. Indeed, the TRIGA reactor has become synonymous with ZrH_x moderated fuel, and reactors not of the General Atomic design that use TRIGA fuel are considered TRIGA (or more precisely TRIGA-fueled) reactors.

The other design principles have an effect, albeit smaller, on safety. Cost was to be kept low but not so low as to compromise the other three principles. Utility refers to the practical aspects and versatility of the reactor, which was designed as a multipurpose tool with emphasis on producing of radionuclides of sufficient strength to be useful for research, medical and industrial application. Finally, the reactor was to be kept simple so that it could be operated easily by persons with limited special skills and thus be used more widely. Moreover, this simplicity would provide for nearly maintenance-free operation over long periods of time.

Within months after completion of the first TRIGA reactor, the new design and fuel had been tested to 100 kW (Merton et al. 1958). Subsequently, because of the large, prompt, negative coefficient, a pulsing capability was designed into the TRIGA by rapid removal of a high-worth control rod. About half of the TRIGA or TRIGA-fueled reactors in the United States are licensed to perform routine pulsing, with peak power levels to 2 GW achievable during the pulse.

TABLE 1. TRIGA and TRIGA-Fueled Reactors Licensed in the United States as of September 1981

Licensee	Type[(a)]	Authorized Power, kW_{th}
AFRRI	TRIGA MK F	1000
Aerotest Operations, Inc.	TRIGA MK I	250
Columbia University	TRIGA	250
Cornell University	TRIGA	100
Dow Chemical Company	TRIGA MK I	100
General Atomic	TRIGA MK I	250
General Atomic	TRIGA MK F	1500
Kansas State University	TRIGA MK II	250
Michigan State University	TRIGA MK I	250
Northrop Corporation	TRIGA MK F	1000
Oregon State University	TRIGA FLIP	1000
Pennsylvania State University	TRIGA MK III	1000
Reed College	TRIGA MK I	250
Texas A and M	TRIGA FLIP	1000
U.S. Geological Survey	TRIGA MK I	1000
University of Arizona	TRIGA MK I	100
University of California/Berkeley	TRIGA MK III	1000
University of California/Irvine	TRIGA MK I	250
University of Illinois	ADVANCED TRIGA	1500
University of Maryland	TRIGA	250
University of Texas	TRIGA MK I	250
University of Utah	TRIGA	100
University of Wisconsin	TRIGA FLIP	1000
Veterans Administration	TRIGA	18
Washington State University	TRIGA FLIP	1000

(a) The two basic models, Mark I and Mark II, are distinguished by whether they are located above or below ground. The Mark III model has a movable core and pulsing capability.

Safe shutdown is ensured by the prompt negative temperature coefficient of the ZrH_x fuel. Although fuel with $ZrH_{1.1}$ has been safely pulsed to moderate power levels (General Atomic 1959; Reed College 1967), TRIGA reactors currently licensed for pulsing use only $ZrH_{1.6-1.7}$.

The inherent safety features of the TRIGA reactors depend largely upon the unique nuclear and physical properties of ZrH_x, which are briefly described elsewhere in this report. In addition, U-ZrH_x has excellent fission-product retention capability as verified by experiments and other empirical observations (General Atomic 1959; Foushee and Peters 1971). The relatively low steady-state power levels ($\leq$1.5 MW_{th}) and typical low-capacity factors and modes of operation create a relatively small cumulative averaged fission-product inventory, thus mitigating the consequences of a cladding failure. Even continuous operation at the maximum licensed steady-state power level (1.5 MW) will not provide enough decay heat to produce fuel melting after a loss-of-coolant accident (Jiacoletti et al. 1979).

Despite the inherent safety of the TRIGA reactors, there are several reasons why a new analysis of credible accidents is desirable at this time. Over the years, reactor safety and licensing concepts have changed, and new data have become available. Thus, the analysis of credible accidents can be re-examined in a modern light to take advantage of these factors plus the more than two decades of operating experience that has been gained. Also, some TRIGAs are now ready for license renewals, and a fresh, independent, credible accident analysis is in keeping with the long-standing practices in the nuclear safety field. Thus, the U.S. Nuclear Regulatory Commission requested that Pacific Northwest Laboratory(a) perform an independent and fresh overview analysis of credible accidents for TRIGA and TRIGA-fueled reactors that would include the information developed during the more than 20-year operating history of the reactors.

(a) Operated for the U.S. Department of Energy by Battelle Memorial Institute.

BASIC DESIGN AND OPERATING CHARACTERISTICS

A principal characteristic of TRIGA and TRIGA-fueled reactors is the fuel-moderator element, with the zirconium hydride moderator homogeneously mixed with the enriched uranium fuel. Although several types of fuel-moderator elements could be made by combining different hydride compositions, fuel enrichments, amounts of fuel, and cladding materials, only three basic types of TRIGA fuel elements are commonly used. The two types that use 20% enriched fuel are distinguished by the cladding material and hydride composition. Aluminum clad elements contain $ZrH_{1.1}$ but, although still widely used, are no longer manufactured. The second type has stainless steel cladding, which has replaced aluminum for all fuel elements, and $ZrH_{1.6-1.7}$. The third basic type, known as FLIP fuel has stainless steel cladding, $ZrH_{1.6-1.7}$ and 70% enriched uranium.

The general design of the TRIGA fuel elements is constant irrespective of type (Figure 1). Slight differences in the end-plug design, dimensions, or the inclusion of burnable poisons have no practical significance with respect to accident evaluations. The minimum critical assembly, or core, requires approximately 55 standard fuel-moderator elements, containing 20% enriched uranium or about 2 kg of ^{235}U. The required number of elements and core size would be reduced with Fuel Lifetime Improvement Program (FLIP) fuel.

The basic design and layout of a General Atomic-designed TRIGA reactor is shown in Figure 2. The core is a series of concentric circles ("rings") of vertical fuel elements held in place by circular grid plates at top and bottom. Appropriate fuel element positions are filled by control rods and experiment facilities. The core is usually located near the bottom of a large pool with a capacity of 8,000 to 60,000 gallons of water.

Neutron savings or reflection occurs largely from interactions with the graphite end plugs of the elements and the water surrounding the core. Additional and necessary reflection is obtained in General Atomic TRIGA designs from the large graphite annulus surrounding the core and housing the rotary irradiation facility or "lazy susan". Further reflection is obtained from the common practice of filling an outer ring with graphite dummy elements.

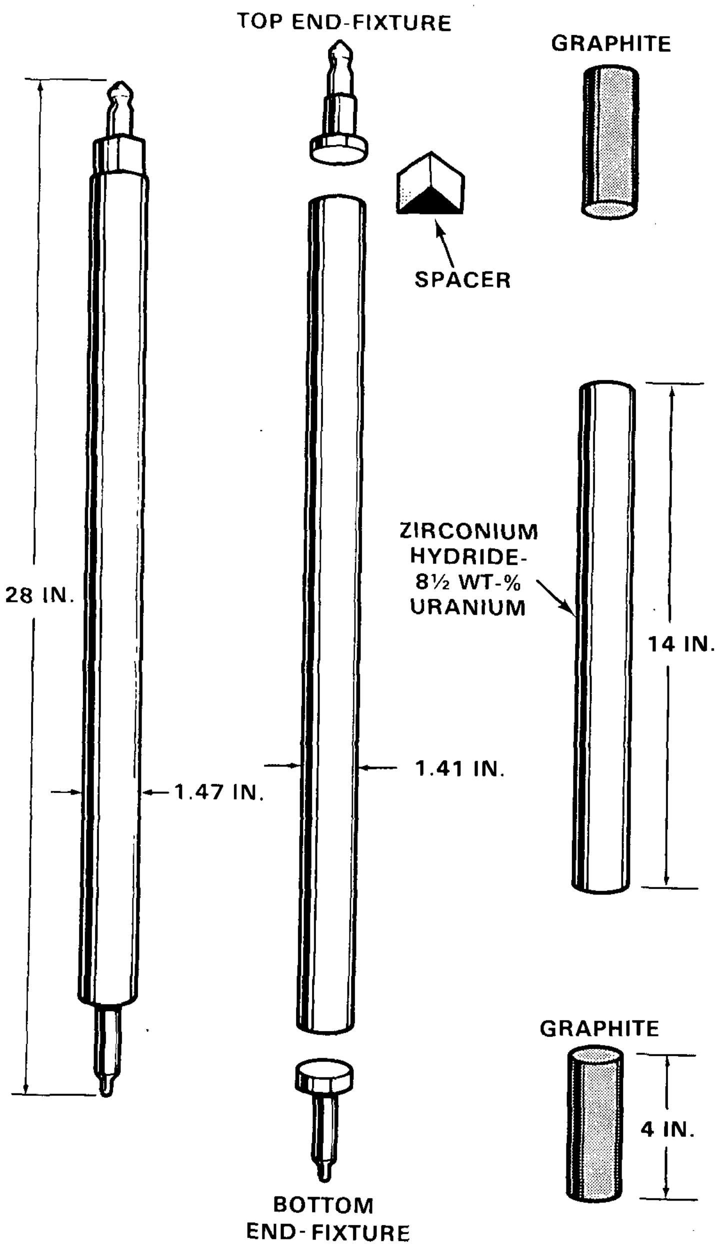

FIGURE 1. Basic TRIGA Fuel Element (Adapted from Reed College 1967)

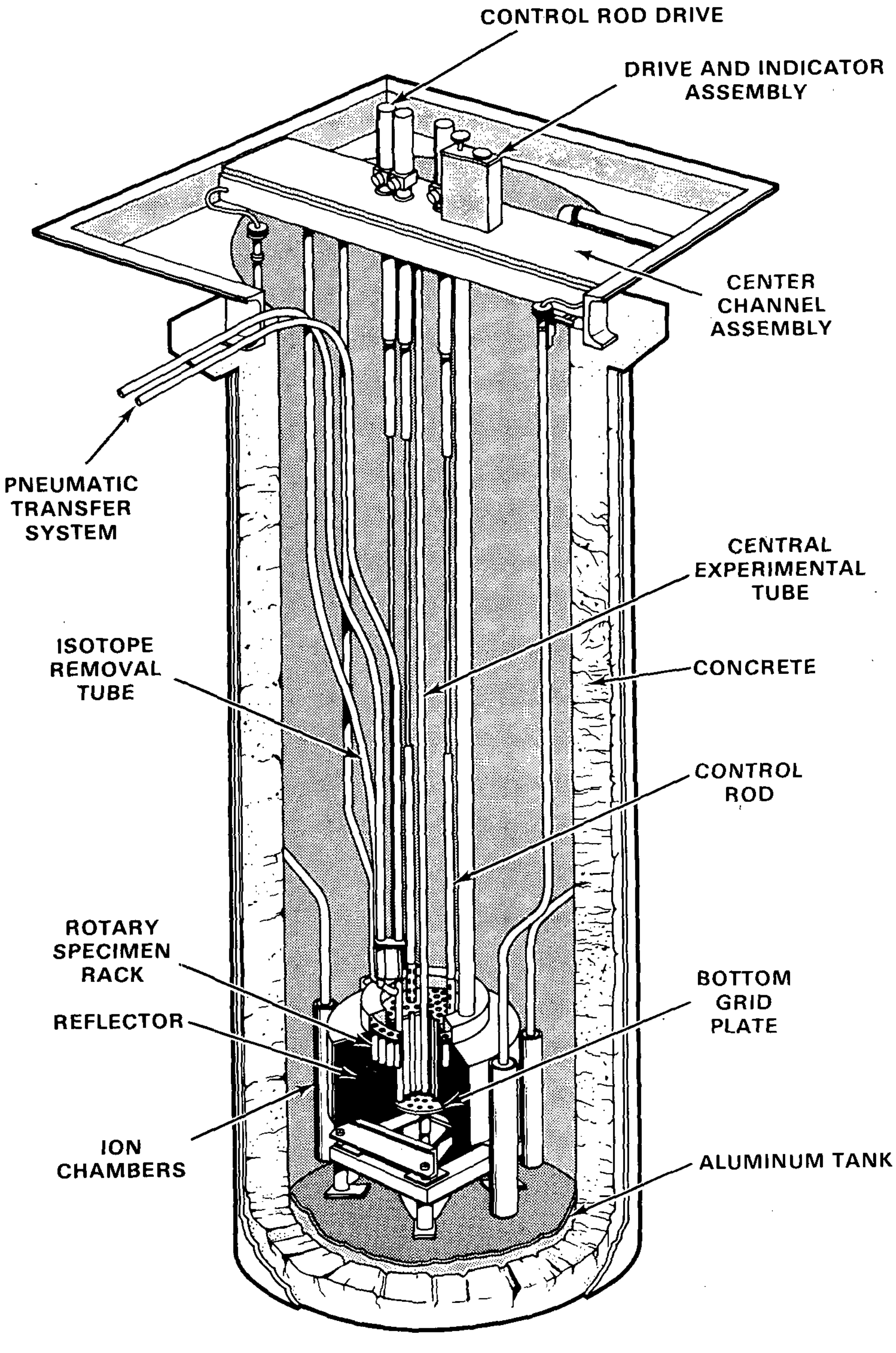

FIGURE 2. Cutaway View of basic Mark I TRIGA Reactor (Adapted from Reed College 1967)

In addition to General Atomic TRIGA configurations, several other pool reactors have been converted to TRIGA fuel using specially designed bundles of four elements. Typically, these reactors have been converted from MTR plate elements and have retained their original (usually rectangular) core configuration. Details of core layout and fuel element design can be obtained from several sources, including the safety analysis reports for specific TRIGA and TRIGA-fueled reactors (e.g., General Atomic 1959, Reed College 1967, Washington State University 1976).

Since the construction of the TRIGA fuel elements and their arrangements or lattice spacings have limited variability, several parameters among the TRIGA and TRIGA-fueled reactors have similar values. The (negative) fuel temperature coefficient is one of these. Approximately half of the coefficient is attributable to the zirconium hydride, with the remainder divided between the uranium in the elements (Doppler effect) and the general neutron leakage from the core. The total coefficient is typically on the order of 1×10^{-4} $\Delta k/k/^\circ C$. The values for the three components of the fuel temperature coefficient will vary depending on the type of fuel element, or more specifically, according to the degree of enrichment, the hydride composition, and the cladding material. Since the principal moderator, zirconium hydride, and the fuel are uniformly mixed, the total fuel temperature coefficient is also prompt.

The typical void coefficient for the interstitial water in the core is about -0.2% $\Delta k/k/1\%$ void, which makes all the core water worth about 20% $\Delta k/k$. The interstitial water provides necessary moderation in addition to the zirconium hydride and has a small, delayed temperature coefficient (bath coefficient), which is also negative.

Median values for the effective fraction of delayed neutrons (β_{eff}) and the prompt neutron lifetime are about 0.0072 and 60 μs respectively, with variations attributable primarily to differences in reflection of the core and fuel enrichment.

DEVELOPMENT OF ACCIDENTS

Although TRIGA reactors and fuel have been used safely for more than two decades, certain potential accident scenarios have not been fully evaluated, or need to be re-evaluated in the light of new knowledge and experience developed over the years. These can be conveniently combined into the following seven general areas:

- excess reactivity insertions
- metal-water reactions
- lost, misplaced, or inadvertent experiments
- mechanical rearrangement of the core
- loss-of-coolant accident
- changes in fuel morphology and ZrH_x composition
- fuel handling.

Specific accident scenarios were not evaluated. Rather the intent was to evaluate a more general category of potential credible accidents in light of available information and data obtained from the pertinent body of knowledge that has developed over the twenty-year operating history of the TRIGA and TRIGA-fueled reactors. No experimental work was performed in this study, although re-evaluation was made of experimental data and operating history reported by others. In general the basic logic was to consider worst-case situations and to ignore accidents with lesser consequences unless these have potential for major damage to physical facilities or offsite health effects.

INADVERTENT REACTIVITY INSERTIONS

Perhaps the most obvious generic credible accident is the stepwise or instantaneous inadvertent insertion of positive reactivity. The power transient thus produced, if sufficiently large, could result in fuel overheating and a possible breach of cladding integrity.

Fuel temperature effects may vary greatly according to the type of moderator. For $ZrH_{1.1}$, a phase change great enough to cause cladding failure occurs at about 535°C (Simnad 1980). $ZrH_{1.6-1.7}$, however, does not undergo a phase change even at temperatures above 2000°C. The limiting temperature for the higher hydride fuel elements is about 1100°C and is based on pressure buildup from the evolution of hydrogen. This has been clearly demonstrated by the SNAPTRAN tests (Buttrey et al. 1965; Cordes 1965,1966; Hasenkamp 1966), which established that large excursions in U-$ZrH_{1.85}$-fueled reactors would not produce fuel melting but rather an explosive disassembly of the fuel from gas pressure produced by dehydriding. Thus, an examination of the inadvertent transient accident should be made with consideration given to the different critical temperatures and effects in the low- and high-hydride fuel. However, since thousands of pulses have been safely performed with both low- and high-hydride fuel-moderator compositions, the conclusion that these reactors would not endanger the public welfare as a result of an inadvertent transient seems justified.

METAL-WATER REACTION

A second major accident category that needs examination is the exothermic metal-water reaction, which theoretically could unleash sufficient energy to produce explosive type forces or add to fuel heating. Considerable data have been gained from accidents and destructive tests since the TRIGA reactors were designed and initially evaluated from a safety standpoint. These data indicate that an explosive Al-H_2O reaction can occur under certain conditions (Baker and Liimatakinen 1973; Thompson and Beckerley 1964). Also, the Zr-H_2O reaction should be considered as well (Baker and Just 1962) since all the TRIGA fuel elements contain Zr as ZrH_x and some have Zr metal rods located in the center of the fuel meat.

LOST, MISPLACED, OR INADVERTENT EXPERIMENT

TRIGA reactors are widely used as research tools as well as for radionuclide production. Thus, the possibility of a misplaced or inadvertent experiment is real and, accordingly, should be examined. Improperly placed or designed experiments could produce sudden changes in reactivity or could directly affect cladding integrity through mechanical or chemical action. Similarly, chemical changes associated with irradiation might produce explosive mixtures within the various irradiation facilities. Also, the dropping of various objects into the open pool might be included in this category.

MECHANICAL REARRANGEMENT OF THE CORE

A major disruption or mechanical deformation of the core could be caused by severe natural forces such as earthquakes or tornadoes. The result might yield changes in reactivity or other core parameters and in addition might also damage or break the fuel elements, resulting in release of fission products and fuel to the environment.

LOSS-OF-COOLANT ACCIDENT

Natural forces can cause a loss of coolant through a rupture of the piping, or a beam port, or even of the reactor vessel. Early studies (General Atomic 1959) concluded that no fission product release would occur from a loss-of-coolant accident. For their analysis of the loss of pool water accident, General Atomic assumed that an instantaneous loss of all the water terminated prolonged operation at 1.5 MW. Since water is necessary for the normal lattice to maintain criticality, the reactor would shutdown before a significant portion of the fuel surface could be exposed (General Atomic 1959). The calculated temperature rise was about 110°C, producing a maximum fuel temperature of about 460°C. At this temperature, radiative loss of the core heat would be sufficient to ensure cladding integrity.

A loss-of-coolant accident was also analyzed for the Reed College TRIGA reactor, a typical Mark I model fueled with aluminum clad elements using $ZrH_{1.1}$ moderator (Reed College 1967). The postulated loss-of-coolant accident showed that the maximum fuel temperature would be less than 150°C after the infinite operation at 250 kW was terminated by the instantaneous loss of water. At this temperature the equilibrium pressure from fission gases, entrapped air and dissociated hydrogen was reported to produce a stress of only 660 psi which is well below the yield stress of >5000 psi for the aluminum cladding at 150°C.

In the safety analysis for the Annular Core Pulse Reactor (ACPR) a loss-of-coolant accident was assumed to occur five minutes after a pulse, which produced a 1000°C fuel temperature. Furthermore, a long period of steady-state operation at 300 kW was assumed to precede the pulse. The maximum fuel and cladding temperature after the coolant loss was calculated to be 390°C. The equilibrium pressure for gases inside the fuel element was calculated to be 36 psi, which produces a cladding stress of about 1300 psi. Since the ACPR used stainless steel clad fuel with a yield stress of 35,000 psi at 400°C, it was concluded that a loss-of-coolant accident would not lead to a fission product release (Hasenkamp 1966).

Jiacoletti (1979) recently re-examined the loss-of-coolant accident for research reactors and concluded that because of the low power level of these reactors, there would be no significant safety hazard. In an experiment with a 2-MW-pool (non-TRIGA) reactor, Knezevich et al. (1965) concluded that core meltdown following loss of coolant would not pose a problem, and that the only hazard would be from radiation exposure directly over the core. Largely because of the small quantity of decay heat available, further reduced by decay time since coolant loss time is not instantaneous, a meltdown from a loss-of-coolant accident is not credible. Therefore, this type of accident will not be considered further in this report.

CHANGES IN FUEL MORPHOLOGY AND ZrH_x COMPOSITION

The accumulated pulses on some fuel elements may number in the thousands, principally for the reactors that have been licensed for routine pulsing for a number of years. During pulsed operation the fuel is subjected to the greatest irradiation and thermal gradients. Since ZrH_x is brittle, fracturing may be expected to occur over the course of several hundred pulses, producing an increase in the surface area of the fuel meat.

The zirconium hydride in the fuel meat also disassociates to a degree, evolving a small amount of hydrogen gas that ultimately rehydrides. Although fracturing is minimal if cladding integrity is maintained, extensive cracking will occur if the evolved hydrogen can escape (Simnad 1980). This could lead to further evolution of hydrogen exacerbating the situation. Although the dehydriding reaction is endothermic, sufficient heat might be available during a pulse to sustain the process. However it is worthwhile to consider if the fuel meat might break into pieces as a result of numerous pulses, particularly for reactors that routinely pulse. The concern from a credible accident standpoint is whether increased hydrogen evolution would increase the cladding stress to a point where cladding integrity is lost.

The effects of sustained thermal gradients, produced during steady-state operation, have in one known case caused a change in the phase composition of the zirconium hydride (Richards and Randall 1981). The lower hydride phase that was created produced a distortion in several fuel elements, primarily as a result of the routine pulsing operations.

TRIGA and TRIGA-fueled reactors that operate at steady-state power levels of $\geq$1 MW may produce temperature gradients sufficient to create lower hydride phases in some of the fuel elements. However, such a condition would be latent unless the reactor were pulsed, either routinely or as a result of an inadvertent transient. The concern from a credible accident standpoint is if the maximum credible pulse could cause enough distortion in one or more elements to rupture the cladding. Such an event would also depend on the amount of lower hydride phase in the elements. Hence, the general questions of the possible effects of pulsing on fuel morphology and the *in situ* changes in the zirconium hydride composition will be considered.

FUEL-HANDLING ACCIDENTS

Since at some time all reactors using TRIGA fuels will refuel or remove fuel from the core for other reasons, it is credible that an irradiated fuel element may be damaged while outside the reactor, with concomitant release of radioactivity to the environment. Given the age, number, and type of usage of TRIGA reactors, this class of accident should be re-examined.

INADVERTENT TRANSIENT

Following a large, sudden reactivity insertion, the crucial safety parameter is the fuel-moderator temperature, which is related to the H:Zr ratio. The low-hydride moderator ($ZrH_{1.1}$) undergoes a phase change at approximately 535°C, which can cause cladding failure (Simnad 1980). Phase changes do not occur in the higher hydrides (i.e. those with H:Zr ratios greater than 1.5) at temperatures below 2000°C. The difference in hydrogen content between $ZrH_{1.6}$ and $ZrH_{1.7}$ is sufficient to create a significant difference in the limiting temperatures, which are 1150°C and 1000°C, respectively, for these two hydrides (Simnad 1980; WSU 1976). The disassociation of the zirconium hydride, which produces hydrogen gas and increases the internal pressure of the fuel element, is a function of the hydride composition and fuel-moderator temperature.

For the higher hydride fuel elements, the onset of cladding failure is determined by the yield strength of the cladding material at a given temperature. The pressure inside the fuel element is a function of both temperature, and additional gas that will be produced from dehydriding as the temperature rises. Since all the higher-hydride fuels are clad with stainless steel, only the hydride composition and its dynamics need be considered in determining the limiting temperature.

Although the current upper-limit hydride composition is $ZrH_{1.65}$, with $ZrH_{1.6}$ as the design value (Simnad 1980), there are some fuel elements with the higher hydride composition of $ZrH_{1.7}$ still in use (WSU 1976). The limiting values given above for hydrides with the H:Zr ratio greater than 1.5 are valid if the cladding temperature remains below 500°C. However, given equal cladding and fuel-moderator temperatures greater than 500°C, the limiting temperatures for transient or short-term operation are reduced to 940°C for $ZrH_{1.6}$ and 900°C for $ZrH_{1.7}$. Because long-term steady-state operation produces some fuel growth and deformation above 750°C, this value is usually taken as the limiting temperature for this type of operation (Simnad 1980). Since the only credible (indeed conceivable) way to initiate a pulse requires some water in the core, the fuel temperatures can reasonably be expected to be somewhat higher than the cladding temperatures. Hence, the lower limiting temperatures are generally used for the inadvertent transient analysis of reactors using the higher-hydride fuels.

There are several accident analyses of inadvertent transients for TRIGA and TRIGA-fueled reactors in the literature. The majority of these analyses are based on information originally developed by General Atomic, the designer and manufacturer of TRIGA reactors, usually in conjunction with safety analysis reports or similar licensing documents. A review of some of the analyses performed to date provides a convenient starting point for the inadvertent transient analysis in this study.

An inadvertent transient created by the addition of the entire licensed excess reactivity of 1.46% $\Delta k/k$ (\$2.00) was considered in the Torrey Pines Safety Analysis Report (General Atomic 1959). This reactor used aluminum clad fuel with a H:Zr ratio of approximately one and was licensed for pulsing with up to 1.46% $\Delta k/k$ (\$2.00) of reactivity (General Atomic 1959). The analysis of an inadvertent transient was based on actual experiments, and the results showed that the maximum fuel temperature remained below 400°C. This is well below the 535°C limiting temperature for cladding failure for the $ZrH_{1.1}$ used in the elements (Simnad 1980). The fuel elements suffered no observable deleterious effects from repeated pulses with a \$2.00 insertion, which leads to the conclusion that there would be no hazard effects from either routine pulsing or an accidental insertion of \$2.00.

The postulated reactivity accident included in the Reed College Safety Analysis Report (Reed College 1967) assumed essentially instantaneous insertion of all the available licensed excess reactivity, i.e., \$3.00 (2.25% $\Delta k/k$). The analysis was concise and, although prepared for a reactor without pulsing capability, noted that numerous pulses had been safely performed with the General Atomics TRIGA with both identical fuels and reactivity insertions. Although the fuel had been extensively distorted by the pulses, there was no breach of integrity or fission product release in nonpulsing fuel subjected to a stepwise reactivity insertion of \$3.00 or less.

The Washington State University (WSU) reactor is licensed for 1 MW_{th} steady-state operation and pulsing with reactivity insertions of $2.50 or less. The core is a mixture of standard, 20% enriched, stainless steel clad TRIGA fuel and the 70% enriched FLIP fuel. FLIP fuel differs from the standard fuel by its greater enrichment of 70% and the addition of about 1.5 wt% erbium to the fuel-moderator meat. The FLIP fuel also has a slightly smaller H:Zr ratio (1.6:1) than the H:Zr ratio in the standard fuel (1.7:1). The WSU reactor was not designed by General Atomic and is not of the basic TRIGA design. It was converted from MTR fuel elements by using special bundles of TRIGA fuel elements that allowed the original core shape and structure to be retained.

The inadvertent transient accident was evaluated in the Safety Analysis Report (Washington State University 1976). Two ways were postulated for the inadvertent transient to occur: 1) accidental fuel insertion, or 2) inadvertent ejection of the transient rod when at its maximum worth. The worth of the centrally located fuel bundle was reported as $3.75, which is the same value given for the total worth of the transient rod. For both cases the accidents were postulated to occur while the reactor was operating at full steady-state power. Stepwise addition of $3.75 of reactivity while at full power was calculated to produce a maximum fuel temperature of 1142°C that was stated in the Safety Analysis Report to be below the 1150°C safety limit for FLIP fuel, and hence within acceptable limits.

Although a great variety of inadvertent transient accidents can be postulated, the most serious credible accident would result from an instantaneous insertion of all available excess reactivity in the reactor. To produce a more serious accident additional reactivity from an outside source would have to be inserted simultaneously with the reactivity from the reactor, or in a quantity greater than is available from the reactor. Such an accidental event is simply not credible.

West et al. (1967) used a modified Nordheim-Fuchs point kinetics model to calculate pulse characteristics for the Advanced TRIGA Prototype Reactor (ATPR), which used $ZrH_{1.7}$, and compared the results with measured values for the same characteristics, i.e., peak power, energy release and fuel temperature. As shown in Table 2, the values calculated by West and his co-workers are all higher than the measured values, and thus the calculation appears to be conservative.

A simpler equation (Equation 1) that assumes a constant heat capacity can also be used. This equation yields more conservative results in that it predicts a greater mean temperature rise for a given reactivity insertion as shown:

$$\Delta k_p = \frac{E\alpha}{2C} = \frac{\Delta T\alpha}{2} \tag{1}$$

where Δk_p = $\Delta k/k\ (1-\beta_{eff}) - \beta_{eff}$

E = total energy release, MW-s

C = heat capacity of core, MW-s/°C

ΔT = temperature rise, °C

α = prompt negative temperature coefficient, $-\ \Delta k/k/°C$.

Values for Δk_p, C, and α can be readily obtained from the literature. The prompt negative temperature coefficient, α, varies somewhat, but a typical value of 1×10^{-4} $\Delta k/k/°C$ will be used. The core heat capacity, C, is a function of the core size or number of fuel elements and can be calculated from the heat capacity for a single element which is about 820 watt-s/°C (West et al.

TABLE 2. Energy Release and Fuel Temperatures from Transients

Reactivity Inserted $\Delta k/k$	Energy Release MW-s		Fuel Temperature[(a)] °C		Peak Fuel Temperature[(a)] °C
	Measured	Calculated	Measured	Calculated	Calculated
3.50	54	57	750	790 ± 20	1050
3.15	45	—	675	---	---
2.80	39	40	585	615 ± 15	830
2.10	24	25	405	430 ± 10	590

(a) From the position of the thermocouple, this is the approximate mean temperature in hottest elements.

NOTE: Data from West et al. 1967.

1967). In this study, these calculations were made for cores containing 60 and 100 elements. Values of Δkp can be calculated from β_{eff} which depends on the core type, with the aluminum clad low-hydride core having a value of 0.0073 and the stainless steel higher-hydride core having a value of 0.0070 (West et al. 1967). Using the limiting temperatures specified earlier and assuming a typical initial or low-power fuel temperature of 30°C, temperature rises and required reactivity insertions for various H:Zr ratios are calculated and are shown in Table 3. The typical maximum licensed-core excess for each H:Zr ratio and the subsequent temperature rise if this amount of reactivity were instantaneously inserted into the reactor is also shown.

As shown in the Table 3, the calculated temperatures produced from stepwise insertion of typical maximum licensed reactivities are well below the limiting temperatures. More reactivity than is available would be required for the fuel temperature to reach the limiting temperature.

If the reactor were operating at full power before an inadvertent transient, the initial fuel temperature would be higher than the 30°C used above. Thus, the value of ΔT would be smaller and a smaller amount of reactivity would be needed to raise the fuel to the limiting temperature. However, a part of the available reactivity would have been used in reaching the higher power

TABLE 3. Reactivity-Temperature Relationships

	Parameter			
H:Zr Ratio	Limiting Temperature, °C	$\Delta k/k$, %(a)	Typical Licensed $\Delta k/k$, %	Calculated Mean Temperature, °C(b)
1.1	535	3.28	2.25	330
1.7	1000	5.62	3.50 to 5.60	579 to 996
1.6	1150	6.37		

(a) $\Delta k/k$ required to reach limiting temperatures.

(b) Calculated from maximum licensed reactivity in operating TRIGAs (may contain mixed H:Zr ratios).

level, thus offsetting this apparent reduced amount. Indeed, the calculated reactivities required for full-power operation (and to heat the fuel to a maximum "normal" operational temperature) and to provide sufficient energy to raise the fuel temperature to the limiting value would be about 3.93% $\Delta k/k$ for the aluminum clad hydride cores and 7.24% to 7.99% $\Delta k/k$ for the stainless steel clad higher-hydride cores, assuming 100 and 200°C nominal full-power fuel temperatures for the two cases.

These conservative calculations show that the available excess reactivity falls short of the amount required to raise the fuel cladding to the limiting temperatures. Hence, even in the event of an uncontrolled stepwise insertion of reactivity, there would be no cladding failure or fuel melting. Only if sufficient reactivity was available from abnormal fuel loadings or external augmentation of reactivity (already considered to be no-credible) would the potential exist for cladding failure. Such an event is considered incredible. Even so the safety limits are themselves conservative and the fuel of all three H:Zr ratios has been pulsed to temperatures reaching and even exceeding the limiting values without loss of cladding integrity (Simnad 1980).

The above analysis may be conservative in other ways as well in that it assumes that transients are limited only by the amount of reactivity inserted stepwise. As has been shown by Young et al. (1964), the large, prompt negative coefficient of the ZrH_x moderated fuel will terminate the power rise and the total energy release will be a maximum of ~20 MW-s, well below that necessary to raise the fuel to the critical temperature.

Since the only credible way a core can be pulsed requires some water in the lattice, any fission product release from one or more failed fuel elements will be reduced by the water. This was clearly shown in the destructive tests with the SNAPTRAN reactors. These reactors were loaded with fuel-moderator elements containing fully enriched uranium alloyed with zirconium hydride ($ZrH_{1.85}$) (Buttrey et al. 1965; Cordes 1966; Hasenkamp 1966). The complete SNAPTRAN fuel-moderator element was approximately the same size and shape as the fuel meat of TRIGA elements. These elements, clad with Hastelloy-N, were assembled into a right cylindrical compact lattice. In the SNAPTRAN 3 test the core was submerged under approximately 1 m of water in a tank 3.1-m high with a

4.25-m diameter. The other SNAPTRAN destructive test was conducted with a dry core, using beryllium reflector pieces instead of water. The cores were subjected to large, rapid reactivity insertions ($3.80 for the wet core, $5.10 for the dry core) that produced an energy release of approximately 50 MWs in each case.

The SNAPTRAN tests showed that fuel-element disassembly occurred before any fuel melting could take place, and the water environment significantly reduced the amount of fission products released to the atmosphere (Buttrey et al. 1965; Cordes 1966). In the submerged-core test the fuel-moderator disintegration temperature for the U-ZrH$_{1.85}$ was determined to be 1054°C. Reported values for the melting points of unclad and clad SNAP fuel are 1899°C and approximately 1400°C, respectively (Cegelski 1965b).

In the test conducted in the dry core, approximately 20% of the total fission-product inventory, consisting of 75% of the noble gases, 70% of the iodines, 45% tellurium and 5% solids, was released. In contrast less than 1 of the total inventory--consisting solely of 4% of the noble gas activity--was released to the atmosphere from the test with the submerged core.

METAL-WATER REACTIONS

Although metal-water reactions have occurred in some reactor accidents or destructive tests (e.g., NRX, Borax-I, SPERT-I, SL-1, TMI), the evidence from these events and laboratory experiments shows that a dispersed liquid metal is required for a violent chemical reaction to occur (Baker and Liimatakinen 1973; Miller, Sola and McCardell 1964; Rogovin and Frampton 1979; Thompson and Beckerley 1964). The conditions for a solid metal-water reaction are not readily conceivable in a reactor system (Epstein 1960). The reaction of powdered or finely divided solid metal with water has been verified in the laboratory, but any event in a reactor capable of reducing the metal components to this state would probably create enough damage so that further destruction by the relatively small amount of energy released by a chemical reaction would be trivial in comparison. However, a physical explosion could occur if molten metal reacts with water in a manner that produces rapid vaporization of the water. This type of explosion could then disperse the molten metal and thus provide the necessary physical conditions for the chemical reaction. However, this would take place so rapidly that, even if the water is vaporized by the molten metal, it will not be in contact with the metal sufficiently long to produce a chemical reaction of any significance. Indeed, the destruction of the BORAX-I reactor is thought to have been a result of a physical explosion rather than a chemical explosion (Epstein 1960).

A metal-water reaction is only a secondary hazard, following a major primary destructive event that creates droplets of liquid metal. Production of molten metal appears incredible without an extraordinary initiating event such as a deliberate initiation of a large energy release in the core, as from an explosion or externally added reactivity. Nonetheless, the potential effects of metal-water reactions alone will be examined, assuming that any metal melted during an accident would react chemically and physically to completion producing water, which then may also contribute to the explosion through steam formation. Also considered is the chemical production of hydrogen that may also explode and produce further damage. An explosion resulting from a metal-water reaction is thought to have occurred in the NRX reactor accident (Epstein 1960; Hurst 1962; Thompson and Beckerley 1964).

The typical TRIGA reactor system includes several materials that potentially are reactive with water. The cladding, either aluminum or stainless steel, while more likely subjected to less heat than the fuel meat, has lower melting temperatures than either Zr or ZrH_x. Since the fuel meat is essentially ZrH_x, the uranium need not be considered for the purposes of this section. However, the question of whether the ZrH_x will react with water is addressed, along with whether dehydriding occurs, making chemically reactive Zr available to react with the fuel meat. Under credible accident conditions, the calculated heat production is far below that required for temperatures in the core to reach the melting point of Zr (1823°C). Thus, even if massive dehydriding occurred to produce metallic Zr from the ZrH_x, the principal hazards would be from release of hydrogen gas and fission products, since molten Zr metal would not be produced.

In numerous experiments with heated samples of ZrH_x, no potential for a ZrH_x reaction with water was found (Simnad 1980). Quench tests at temperatures as high as 1200°C have shown only minor cracking and density increases for some unclad fuel samples. Above approximately 1050°C some localized melting was observed but was determined to be a result of thermocouple contact with the cladding which formed eutectics. Even if the temperature of the stainless steel cladding approaches that of the fuel for several minutes, the limiting temperature would still be determined by the evolved hydrogen pressure (critical pressure occurring at 950° to 1000°C) and not by potential alloy formation with the cladding (Simnad 1980).

Production of metallic zirconium from dehydriding should be maximum with the unclad samples used in these tests. Given the water environment also used in the testing, the conditions for a zirconium-water reaction would be very similar to those encountered in an accident situation. Thus the likelihood of Zr production and subsequent potential for a Zr-water reaction could be evaluated from these tests. The actual amount of reaction was negligible and no evidence of a zirconium-water reaction was reported (Simnad 1980). Whether massive or essentially complete dehydriding could occur under credible accident conditions and produce reactive Zr is beyond the scope of this study. However, substantial dehydriding does occur at much lower temperatures than the melting point of Zr, and the analysis of the SNAPTRAN tests produced no evidence of a metal-water reaction (Buttrey et al. 1965; Cegelski 1965a; 1965b; Simnad 1980).

If high temperatures are generated in a fuel element, the element will disassemble before the melting point of zirconium or the stainless steel or aluminum cladding can be reached. Thus, the resulting pieces of hot fuel/moderator, even if dehydrided, will not react explosively with the cooler water.

Low-hydride aluminum clad fuel changes phase about 120°C below the melting point of aluminum, causing distortion of the fuel elements, altering the core geometry, and lowering k_{eff}. This in turn reduces the amount of available excess reactivity and hence the energy release and core heating. It is not clear whether the heat associated with this energy release would be adequate to raise the aluminum cladding to its melting point. If a core of aluminum clad elements could somehow be pulsed to 75 MW-s for these types of reactors, the cladding should rupture, but actual dispersion of molten cladding seems unlikely. However, as has been noted above, the prompt negative temperature coefficient limits the pulse to about 20 MW-s (Young et al. 1964). Quantification of cladding behavior is beyond the scope of this study and might require experimental effort. However, if the heat transfer capabilities of the ZrH fuel meat are sufficiently poor, the element would disassemble from the hydrogen gas pressure before the cladding could melt. In any case, the gas pressure from hydrogen evolved from the dehydriding process will ultimately disassemble the element. At this point, heat production should essentially cease, based on observation in the SNAPTRAN-3 test (Buttrey et al. 1965; Cordes 1966).

The fuel elements recovered in this test showed a high degree of destruction but still retained large portions of their original form, although they were Hastelloy-N clad, not aluminum. Thus, it is reasonable to assume that a similar situation will occur with TRIGA fuels and that melting or a Zr-H_2O reaction will not occur.

LOST, MISPLACED, OR INADVERTENT EXPERIMENT

In some circumstances, materials or objects may be unintentionally inserted into the core, or left in the core longer than planned. Such situations devolve into three basic categories, with rather fine differences among them insofar as accident potential is concerned. For the sake of clarity, the three categories have been rigorously defined as lost, misplaced, or inadvertent experiments, and these terms are used precisely as defined throughout this report.

A lost experiment is defined as a sample or device inserted into the core and whose intended length of irradiation is significantly exceeded. A sample or device inserted in the incorrect experiment facility or elsewhere in the core would be a misplaced experiment. Accidental or inadvertent experiments result from objects dropped onto or near the core or into the reactor tank or from an accidental or premature startup or with an object or experiment in the core.

There are two credible potential hazards from a lost or misplaced experiment: 1) production of hazardous products from excessive irradiation in amounts greater than planned, and 2) gas overpressurization which could yield an explosive release of activity from the experiment, possibly damaging the core or releasing radioactivity to the environment. A misplaced or inadvertent experiment might also affect reactivity or produce damage or activation products.

The normal experimental facilities of the TRIGA reactors designed by General Atomic include a rotary irradiation rack (commonly called a lazy susan), a central thimble or central irradiation facility, rabbit tubes, and fluence-rate (i.e., flux) monitor positions. In addition, there are one or more beam tubes, in either a vertical or a horizontal position, that may have been added later by the user. If the potential for fuel element damage is taken to be directly proportional to the number and proximity of elements to an experimental access point, then the central thimble, the fluence-rate monitor holes, and the rabbit tubes have the greatest potential for causing damage.

The central thimble or other in-core single element positions are usually water-filled aluminum tubes penetrating both upper and lower grid plates. The central thimble, as its name implies, is a vertical tube in the geometric center of the core and, thus, provides access to the highest levels of neutron and photon fluence rate. The fluence monitoring positions are small holes in the grid plates at interelement locations. These positions have the severest size restriction but are also physically closest to the elements. Flux wires, encased in special holders, are inserted directly into these holes and are used to determine core characteristics and for calibration purposes.

The rabbit system includes an air-filled metal tube, usually aluminium, located near the edge of the core in a region of high-thermal neutron-fluence rate. Although the terminus itself is normally surrounded by water, sample heating would be somewhat greater because of the higher insulating quality of the air inside the terminus. Rabbit irradiations are generally of short duration, with compressed air or vacuum being used to move the rabbit through the system.

Irradiation-induced heating is assumed to play a role in sample failure. However, the induced heating in any credible experiment in any of these facilities would not be great enough to cause melting of the cladding. Even if a flux monitoring experiment were improperly installed so that it touched a fuel element and created a hot spot, the small area of contact would probably not be sufficient to cause localized melting of the fuel or its cladding, or damage or melting of the experiment.

The administrative review system for planned experiments usually requires an evaluation or other assurance that the sample or device, upon failure, will not damage the reactor or produce an increase in reactivity. This procedure does not necessarily account for the behavior of lost, misplaced or inadvertent experiments except in a superficial manner. Experiments inadvertently left in the core would likely be subjected to a prolonged and possibly more intense irradiation and heating than intended. Prolonged irradiation would increase radiolytic decomposition, including a buildup of gaseous products, a process that might be further accelerated by heating. Thus, excessive or prolonged irradiation could create a hazard by the production of explosive gases from chemical decomposition or changes or simply by pressure buildup from gases produced within the sample, causing the encapsulation to fail. Heat generation alone should not be a problem since the heat transfer is likely to remain adequate throughout the irradiation period, particularly in a water-filled irradiation facility. Continued intense irradiation or irradiation in the rabbit tube for long periods could result in weakening, degradation, or even failure of the encapsulation.

An inadvertent experiment, such as an unnoticed object dropped into the pool, might also release its contents into the reactor, possibly causing some adverse chemical reaction or even physical damage to fuel elements. If the cladding is not damaged by impact but the object remains near or in contact with a fuel element, then the potential hazards are similar to those of a lost or misplaced experiment.

A misplaced or inadvertent experiment might also affect the reactivity of the reactor. A sudden change in reactivity would produce an inadvertent transient in an operating reactor. If the reactor can accommodate the transient, then there is no hazard and the event has no practical safety significance. If a large rapid change in reactivity results, the safety systems should effect a scram, shutting down the reactor and preventing further irradiation of the experiments. Significant reactivity changes not sufficiently great to activate automatic scram systems should be observed by the reactor operators who presumably would take appropriate action. However, even if the operators fail to act, the reactor would slowly rise in power to its equilibrium level regulated by the negative temperature coefficient. Thus, the misplaced or inadvertent experiment would not result in an excess reactivity (i.e., nuclear excursion) accident.

The potential hazards from an experiment decomposing and releasing its contents are mitigated by the physical configuration of the experimental facilities. The central thimble is normally water filled but the water can be displaced by a continuous flow of air or other gas, or by a solid object. In either case the flammable or explosive vapors generated would be largely diluted, and the water or air flow would remove the heat sufficiently fast to preclude explosion or ignition. If corrosive substances were released and the thimble were completely air or gas filled, these might escape into the reactor room but would not create a hazard outside of the immediate vicinity. An off-site radiological hazard would not be created.

A release may not occur, since the irradiation facilities are wet, and the pool water would likely dilute or act as a solvent for activation products produced by irradiation. Similarly, air flow in the rabbit system would remove noxious gaseous products of irradiation, discharging them via the stack. If special holders for fluence-rate monitors are not used, then it is possible the material could come into contact with a fuel element. However, this contact would occur under water and would benefit from the dilution and heat transfer capabilities of the pool water.

MECHANICAL REARRANGEMENT

An infinite variety of mechanical rearrangements of the core can occur, depending on the forces and energy available to move the fuel and supporting structures. However, despite the number of potential geometries that might be produced, mechanical rearrangement can be quite simply divided into two broad categories; core crushing with fuel element damage, and simple rearrangement of the lattice.

CORE CRUSHING AND FUEL ELEMENT DAMAGE

This category of potential accidents considers the effects of events with sufficient force to crush the core and related structures, perhaps dispersing the fuel elements or shifting or making major rearrangements in the lattice. The most severe effects would occur if there was little or no water surrounding the core, which would increase the severity of any fission product release (Cordes 1965). Moreover there would be no buoyancy forces from the water to partially offset gravity so the damage to the core would predictably be greater than if the pool were normally filled. However, a catastrophic event which caused the roof to collapse or other major structural damage might also reasonably be expected to rupture the tank or otherwise destroy the integrity of its water-holding capability.

The consequences of such forces acting on the core can be divided into two general cases. In the first, damage could occur to most if not all of the fuel elements, causing a breach in the cladding and breaking and dispersing the fuel. Thus, fission products would be released from the fuel. The amount would depend largely on the size and other characteristics of the rupture, as well as on the size of fuel particles created by the initiating event which in turn determines the amount of surface area exposed. If the reactor were at power or only recently shutdown, then water would be present and the fission product release would be mitigated even though short half-lived gaseous products were present (Cordes 1965). If the pool were empty, then the damage might be more extensive and the potential for fission product release greater. However, only the longer-lived fission products would be present and the total activity available for release would be much less. Only a small fraction of the total fission product inventory would be released, even in the event of a cladding failure of all elements.

The second general case involves a dry core. With a dry pool, the actual release might well be more severe than the case with water, described above. However, a dry core could not be critical with a normal configuration. The short-lived fission products and total activity would be much less than if the reactor had just been shutdown. Even with dry fuel elements (the worst case) the fission product release fraction is very small—usually taken as 1.5×10^{-5} (Reed College 1967). A rearrangement of the fuel elements could occur from external forces, which would create a supercritical lattice. This situation is similar to a criticality accident, and a corresponding hazards analysis can be made. However, since the nuclear excursion of a dry lattice is not considered credible, this type of analysis was not performed.

LATTICE REARRANGEMENT

Fuel element rearrangements occurring as a result of severe mechanical shock, turbulence or other disruptive forces might result in the creation of a critical or supercritical lattice. Assigning probabilities to the creation of specific arrangements and performing criticality calculations are beyond the scope of this present study and are, in fact, unnecessary since the problem lends itself to generic treatment.

Small changes in the lattice spacing may change the excess reactivity of the system. One safety analysis of a TRIGA-fueled reactor (Hasenkamp 1966) reported that the critical loading with optimum close packing is about half the nominal loading, although the dimensions of optimum spacing are not reported. However, operating experience gained in fuel element removal and insertion shows that small deviations from normal spacing, as is the case when a fuel element is not reseated properly, decrease the available excess reactivity of the system. This is reasonable since the system is designed to be undermoderated, and water is replaced with a less efficient moderator, in this case the fuel element. The overall net reactivity of the system would then be decreased.

Generally, the only ways to increase the reactivity of the system would be to exchange the water for a better moderator (such as D_2O) or to add fuel. If two or more elements were somehow brought closer together while the remainder retained normal spacing, the overall reactivity of the system would remain constant or only slightly changed but the power distribution might be altered, producing a small region that becomes hotter than normal. This situation would be serious only during a pulse where the hottest spot might then reach and remain at a high temperature long enough to cause a cladding failure either directly by melting or indirectly as a result of the fuel-moderator volume change for $ZrH_{1.1}$ or by excessive gas pressure generated by the dehydriding of $ZrH_{1.7}$. The normal heat transfer capabilities of the reactor system should eliminate any serious heat buildup in the cladding. Heat buildup in the fuel would decrease reactivity.

CHANGES IN FUEL MORPHOLOGY AND ZrH_x COMPOSITION

High-level pulsing reactors, defined as producing fluences of 10^{15} n/cm^2, can operate over a reasonable fuel lifetime without adverse safety consequences (Simnad 1980). Although "reasonable" is not further quantified, more than 25,000 pulses have been performed with standard TRIGA fuel elements (Simnad 1980), and at least one reactor has accumulated 700 pulses on some of its fuel elements over a period of 14 years. Therefore, it is not unreasonable to assume that some fuel elements still in service have accumulated even more pulses, perhaps as many as 1,000. The usual evaluation criteria for fuel elements are based on bowing, elongation or visual changes in the cladding. These parameters are used to estimate fuel serviceability and apparently give a reliable indication of thermal gradients to which the fuel element has been subjected during its operating history. In addition, the long-term effects of fuel burnup create swelling of the fuel moderator that presumably would also be indicated by changes in fuel element dimensions. Fuel elements that have been pulsed many times may have cracking or fracturing of the fuel-moderator meat, which would not be observed during the fuel inspections. ZrH_x is a brittle material and susceptible to thermal cracking (Simnad 1980). Even though thermal cycling tests on 45 wt% U-$ZrH_{1.7}$ samples have shown apparently no adverse effects (Simnad 1980), the brittleness of ZrH_x might ultimately cause the fuel-moderator meat to break into relatively small pieces.

The rates of heating and cooling of the fuel elements during a typical pulse are rapid (on the order of 10^3°C/s) and thus thermal explosion and contraction in the fuel moderator could conceivably produce fractures, which ultimately break the fuel moderator into pieces. At what point these effects might occur, if at all, has not been established, although cracking of the fuel moderator due to swelling of a central Zr rod has been observed (Richards and Randall 1981).

Other tests (Simnad 1980) have shown that pulsing will produce extensive cracking and fracturing of the fuel moderator meat if the hydrogen evolved from disassociation of ZrH_x is allowed to escape. However, the effects of pulsing on a fuel element with a cladding break or defect have not been determined experimentally. Experiments designed to measure the equilibrium hydrogen pressure produced during a pulse have recorded pressures that were not sufficient to cause rupture of the cladding (Coffer et al. 1966; Simnad 1980). The equilibrium hydrogen pressure is dependent on the diffusion coefficient, and although the measured diffusion coefficients of hydrogen in zirconium hydride suggest that equilibrium pressure can be obtained during a pulse, a pressure transient is still credible given the uncertainty about fuel moderator breakup over very long operation and the higher operating temperature of pulsed fuel elements (Coffer et al. 1966; Moss 1966; and Simnad 1980).

The reported disassociation pressures for ZrH_x are a function of temperature and ZrH_x composition (Simnad 1980). The effect of ZrH_x morphology is not reported, but a relationship must exist inasmuch as hydrogen accumulation in the gap is diffusion controlled. In the case of very small pieces or particulates, the hydrogen will, on the average, travel a shorter distance before reaching the surface and hence have a greater probability of escaping from the fuel and accumulating in the gap and interstitial spaces. Hydrogen resorption or hydriding would also proceed more rapidly due to the increased surface area available for reaction. However, the absolute pressure should be larger since the amount of hydrogen released would be greater and the gap volume would not be increased to any large extent by the cracks and fractures.

Therefore, two situations need to be considered: 1) if normal pulsing will produce chunks or break the fuel moderator after a given number of pulses, and 2) if a cladding break during a pulse will enhance or contribute to fuel-moderator cracking. Suppose a pressure transient was created during a pulse due to increased hydrogen diffusion and accumulation in gaps and cracks. This could lead to cladding failure and subsequent fission product release in the first case and, for the second, an increased fission product release. Since these accidents are postulated to occur only with elements that have acquired a long operating history of pulses, the fission product inventory should be relatively great. Because these situations require a pulse as the initiating or terminal event, the fission product inventory will include not only the long-lived fraction but also the short-lived fission gases. Finally, not only are the gap activities likely to be released but also activity from the fuel because of the change in fuel-moderator structure and surface area from the cracking and fracturing. Despite the increased likelihood and quantity of fission product release from this postulated mechanism, the consequences would be small as the water overlying the core would act to limit the activity escaping into the reactor room. Thus, the dose equivalents projected for persons offsite would be considerably lower--perhaps two or more orders of magnitude--than those from the fuel-handling accident postulated below.

To recapitulate, the fuel moderator may undergo changes in a long period of service leading to a greater likelihood of cladding failure and fission product release. Existing experimental data suggest that the possibility of such an accident is small, and the potential consequences would be relatively minor. However, the need for further empirical work, perhaps only in the form of observation at operating facilities, is indicated.

In September 1976, routine inspection of fuel elements at the Texas A and M reactor revealed swelling deformation in three fuel elements (Feltz, Randall, and Schumacher 1977; Richards and Randall 1979). The elements were those of the FLIP type exposed to the highest fluxes and power densities within the core. Swelling was maximal in the element exposed to the greatest flux--i.e., the element in the center of the core that had been pulsed more than 50 times and had a power history of 291 MWd of steady-state operation. Cladding integrity was maintained and no deformation was noted in the standard elements contained in the mixed-core loading.

The Texas A and M reactor is licensed for steady-state operation of 1000 kW and pulsed reactivity insertions of up to $2.70. The peak fuel temperature calculated by Texas A and M from an insertion of $2.70 was 883°C, well below the safety limit of 1150°C (Richards and Randall 1979). Subsequent tests and measurements revealed that the fuel had been subjected to a maximum temperature of 920°C, still well below the safety limit (Richards and Randall 1981).

The damaged fuel elements were examined metallographically by Argonne National Laboratory. On the basis of these and other tests, it was determined that phase change had occurred in the element. Operation at 1 MW produced a thermal gradient of more than 400°C across a 1-cm depth of fuel, with the temperature being greatest (~575°C) in the center of the fuel and least (~150°C) at the fuel surface. When subjected to thermal gradients, the hydrogen in ZrH_x migrates to the colder regions (Simnad, Foushee and West 1976), and the central portion of the fuel element becomes depleted in hydrogen, while the outer portions become hydrogen rich. Thus, the lower hydride in the center would change from the normal delta-epsilon mixture to an alpha-delta mixture, while the higher hydride near the cooler fuel surface would move further into the epsilon phase. This phenomenon produced the observed deformation.

A phase change is not expected to occur solely as a result of pulsing, since the temperature gradients must be present for periods of time much longer than those associated with pulsing for hydrogen migration to occur. (Simnad, Foushee and West 1976). Similarly, steady-state operation at power levels lower than 1 MW will not produce such large temperature gradients in the fuel moderator, and hence the problem will be less. The amount of phase change is apparently determined by the temperature gradients achieved during steady-state operations. The exact combinations of temperature gradients and pulse sizes that could cause fuel element damage have not been characterized, nor has a limiting temperature been established for the mixture of α-δ phase material that might result. Clearly, additional work is needed in this area.

Swelling of the fuel could lead to cladding rupture and release of fission product activity into the pool. The radiological consequences of such a release would in general be confined to the immediate vicinity of the reactor. Even assuming the relatively large release fraction of 10^{-4}, offsite, lifetime, whole body dose equivalents would not exceed 1 mrem, mostly from noble gases. Radioiodines and other fission products would be largely retained in the pool, and the dose equivalents to critical organs of offsite observers would be insignificant--i.e., less than the one millirem value *de minimis* guidance level adopted at DOE sites.

FUEL-HANDLING ACCIDENT

At some point in the lifetime of a TRIGA reactor, used fuel within the core may be moved to new positions or removed from the core. Individual elements or single bundles of up to four elements are moved while the reactor is shutdown. The most serious fuel-handling accidents involve spent or used fuel that has been removed from the core and dropped or otherwise damaged causing a release of fission products. The accident identified below is a "worst-case" situation because it assumes an extremely unfavorable meteorology and a large fission product inventory in the damaged fuel. Thus, the projected offsite dose equivalents represent an upper limit and should be down-scaled appropriately for less-hazardous accidents. The method used to calculate the radiological consequences is essentially identical to that developed in an earlier work on Argonaut reactors (Hawley, Kathren and Robkin 1981) and is discussed in detail there.

The accident described herein assumes that a TRIGA reactor was operated at 1 MW for 1 yr prior to shutdown, or 365 MWd. Such an operating history, although possible, is extraordinary in view of the known operating history of TRIGA reactors and introduces a conservatism of one to two orders of magnitude. For this example, assume that during fuel unloading a fuel element is hit by a shipping cask dropped from the overhead crane. The fuel element is assumed to have a large cladding rupture and severe physical damage to the fuel. The activity in the fuel element can be calculated from the power history by using the RIBD subroutine of the computer code ISOSHLD and assuming a flux of 1×10^{13} n/cm^2-s (Engel, Greenborg and Hendrickson 1966). To remain consistent with worst-case conditions, it is assumed that the TRIGA reactor had only 50 elements in its core, and this particular element was located near the center of the core and hence had greater than average burnup. In a typical TRIGA, maximum burnup for any element is about twice the average burnup, and since the damaged element is 1/50 or 2% of the total, it contained a maximum of 4% of the total activity in the core, or in other words, was assumed to have a power history of 14.6 MWd. The calculated noble gas and radioiodine activities in this element are shown in Table 4. Refractories have been ignored as these would

TABLE 4. Gaseous Fission Product Activity in the TRIGA Element Containing the Greatest Activity Following Operation at 365 MWd

Nuclide	$T_{1/2}$	Activity at Shutdown, Ci	Activity at 48 h Postshutdown, Ci
^{83m}Kr	1.9 h	124	---
^{85m}Kr	4.4 h	287	0.02
^{85}Kr	10.8 y	4.8	5
^{87}Kr	1.3 h	552	---
^{88}Kr	2.8 h	789	0.05
^{89}Kr	3.2 m	970	---
^{90}Kr	32 sec	1102	---
^{133m}Xe	2.3 d	39	29
^{133}Xe	5.3 d	2271	1972
^{135m}Xe	0.3 h	598	---
^{135}Xe	9.1 h	1025	57
^{131}I	8.1 d	1080	1012
^{132}I	2.3 h	1662	7×10^{-4}
^{133}I	20.3 h	1932	407
^{134}I	0.9 h	2544	---
^{135}I	6.7 h	2213	15.5

not contribute significantly to the exposure because of the relatively small fraction released (Foushee and Peters 1971).

As can be seen from the table, at 48 hours after shutdown, the major activities are from ^{133}Xe, ^{131}I and ^{133}I. ^{133}Xe is a noble gas that decays by the emission of a beta particle with a maximum energy of 0.346 MeV. Since this nuclide is chemically inert, it mainly poses an external exposure hazard from immersion in a gaseous cloud containing the ^{133}Xe. On the other hand, the iodine isotopes pose an internal hazard. Radioiodine, if inhaled or absorbed directly through the skin, is absorbed into the bloodstream and localizes in the thyroid, which thus receives the greatest dose equivalent.

All of the radioactivity would not be released from the element. Indeed, the fuel matrix acts to strongly retain the fission products. Foushee and

Peters (1971) have collected and summarized empirical data regarding release fractions. Even with unclad, heated, irradiated fuel, the fraction of gaseous activity released did not exceed 7 x 10^{-5} and was generally 2 to 3 times below this value. Gap activity fraction was approximately 1.5 x 10^{-5}. Negligible amounts of nonvolatiles would be released to the atmosphere. Based on these experimental results, an upper limit release fraction of 10^{-4} was assumed for gaseous activity. This highly conservative value is greater than any previously noted and should easily exceed the fraction reasonably expected to be released by the type of accident hypothesized. Release of nongaseous fission products is typically one or more orders of magnitude lower (Foushee and Peters 1971), and thus, exposure from the source is secondary to exposure from gaseous radionuclides.

Using the method developed for Argonaut reactors (Hawley, Kathren and Robkin 1981), dose equivalents at downwind locations outside of the facility can be calculated from Equation (2) for noble gases and Equation (3) for radioiodines:

$$H = 0.27 \sum E_i \, A_i f \, \chi/Q \tag{2}$$

in which

H = the whole body dose equivalent, in rem

$\Sigma(E_i)$ = the effective absorbed energy of the ith nuclide, in MeV

A_i = the activity of the ith nuclide in the fuel, in curies

f = the fraction of activity released from the fuel

χ/Q = the atmospheric dispersion factor in s/m^3.

$$H_{th} = 2.78 \times 10^{-4} \int_{t_1}^{t_2} A_i f \, \chi/Q \, V f_a D_i e^{-\lambda_i t} \, dt \tag{3}$$

in which

H_{th} = total, integrated, lifetime dose equivalent commitment to the thyroid, in rem

V = the breathing rate, in m^3/h

f_a = the fraction of inhaled activity reaching the thyroid

D_i = the dose conversion factor in rad/Ci (converted to dose equivalent by assuming a quality factor of one)

λ_i = the decay constant for the ith nuclide, in hours^{-1}

t = the time of plume passage, in hours

t_1 and t_2 = the start and end time of the release, in hours after shutdown, and

A_i, f, χ/Q are as defined for Equation (2).

Each of the above equations is based on a constant activity release over a period of one hour. A ground level release one hour in duration occurring 48 hours after shutdown was assumed, with no credit taken for partial containment, plateout, or other potential mitigating mechanisms, however realistic and probable. An extremely conservative value of 1×10^{-2} was assumed for χ/Q. Values of (E_i) for immersion doses from the noble gases were obtained from published data of the International Commission on Radiological Protection (ICRP 1960) and are 0.44, 0.24, 2.8, 0.19, and 0.62 MeV for ^{85m}Kr, ^{85}Kr, ^{87}Kr, ^{133}Xe and $^{135}Xe + ^{135m}Cs$, respectively. Using the values in Table 4 and the pessimistic release fraction of 10^{-4}, the whole-body immersion dose equivalents are calculated at less than 1 mrem, even if no credit is taken for decay, a veritable impossibility.

For the thyroid dose calculations, the breathing rate, V, was taken as $1.2\ m^3/h$ (ICRP 1975), f_a as 0.3 (ICRP 1978), and D_i as 6.3×10^6, 2.3×10^5, 1.8×10^6, 1.1×10^5, and 5.4×10^5 rem per curie inhaled for $^{131-135}I$, respectively, assuming a quality factor of one (Kathren 1964). Decay constant values can be calculated from the values for half-life given in Table 4. The same release fraction and χ/Q were assumed as for the noble gases, and the exposure or cloud passage time was taken to be one hour, or equal to the release time. The calculated lifetime dose equivalent commitment to the thyroid was 1.2 rem, assuming zero time elapsed between shutdown and release of activity. If the release were to occur at 48 hours after shutdown, the calculated lifetime dose equivalent commitment to the thyroid would be on the order of 0.7 rem, largely from ^{131}I.

In addition to the previous method, 50-year dose equivalent commitments to the whole body, red marrow, lungs and thyroid were calculated using the computer code WRAITH (Scherpelz, Borst and Hoenes 1980). The 48-hour postshutdown values from Table 4, multiplied by the 10^{-4} release fraction, were used as the source term. A ground-level release occurring under stable meteorological conditions and light wind speed (i.e., Pasquill stability Class F and 1 m/s) was modeled for several downwind receptor points or ranges. Since the code assumes that the duration of the release is equal to the exposure time, no times or durations need to be specified. Although the code calculates E/Qs, these are numerically equal to X/Qs and can be compared to the previous results.

At a range of 100 m, where the E/Q was about 1.5×10^{-2} s/m^3, the 50-year dose equivalent commitments were as follows: 0.3 mrem whole body, 0.3 mrem red marrow, 1.8 mrem lungs, and 0.57 rem thyroid. These values agree reasonably well with the previous results.

In addition to gaseous radionuclides, other fission products will be present, with the radiostrontiums and radiocesiums being most significant from a hazards standpoint. Table 5 gives the quantity of the significant radiostrontiums and radiocesiums at zero and 48 hours after shutdown in the fuel element containing the greatest activity (i.e., 4% of the total inventory in the core) following operation at 365 MWd.

The air concentration of each nuclide, C_i, can be calculated from Equation (4)

$$C_i = 2.78 \times 10^{-4} \int_{t_1}^{t_2} A_i f \; X/Q \; e^{-\lambda_i t} \, dt \tag{4}$$

in which A_i, f, t, λ_i, t_1 and t_2 and X/Q are as defined for Equations (2) and (3). These values can then be multiplied by the breathing rate and

TABLE 5. Activity from Radiocesium and Radiostrontium in the TRIGA Element Containing the Maximum Activity Following Reactor Operation at 365 MWd

Nuclide	$T_{1/2}$	Activity at Shutdown, Ci	Activity at 48 h Postshutdown, Ci
^{89}Sr	52.7 d	1000	978
^{90}Sr	27.7 y	31	31
^{91}Sr	9.7 h	1292	2.3
^{92}Sr	2.7 h	1462	0.002
^{93}Sr	8.3 m	1658	---
^{134m}Cs	2.05 y	1.8	---
^{134}Cs	2.9 h	3.0	3.0
^{136}Cs	13.7 d	26	23
^{137}Cs	30.0 y	496	496
^{138}Cs	32.2 m	2060	---

exposure time to obtain the intake of each radionucide, and the dose can then be calculated according to the method put forth by Committee II of the International Commission on Radiological Protection (ICRP 1960).

The only uncertainty in Equation (4) is the proper value for the release fraction, f. Review of the literature, and especially the work of Foushee and Peters (1971), suggests that release fractions for nongaseous fission products from damaged irradiated TRIGA fuels have not been well characterized but are, in fact, very small, perhaps so small as to be considered negligible. To maintain the high degree of conservatism in the hazard estimate from the nonvolatile fission products, a release fraction of 1 x 10^{-6} was assumed. This is consistant with hazards analyses generally as well as experimental data which show that the release of solid fission products from damaged reactor fuels is two or more orders of magnitude smaller than the release of noble gases.

In Table 6 the air concentration of radioactivity from each nuclide is given along with the intake of each, calculated from Equation (4) using the conservative parameters discussed below. Also presented for the sake of comparison are the Maximum Permissible Body Burdens (MPBBs) put forth by the ICRP 60. Comparison of intakes with the corresponding MPBB reveals that the

TABLE 6. Air Concentrations, Radionuclide Intake and Doses to Maximum-Exposed Individual from Radiostrontiums and Radiocesiums Following Hypothetical Maximum Credible Fuel-Handling Accident

Nuclide	Air Concentration (Ci/m^3)	Intake (Ci)	MPBB[a] (μCi)
^{89}Sr	2.7×10^{-9}	3.3×10^{-9}	4×10^{-6}
^{90}Sr	8.6×10^{-11}	1.0×10^{-10}	2×10^{-6}
^{91}Sr	3.6×10^{-9}	4.3×10^{-9}	3×10^{-6}
^{92}Sr	4.1×10^{-9}	4.9×10^{-9}	2×10^{-6}
^{134m}Cs	5.0×10^{-12}	6.0×10^{-12}	1×10^{-4}
^{134}Cs	8.3×10^{-12}	1.0×10^{-11}	2×10^{-5}
^{136}Cs	7.2×10^{-11}	8.7×10^{-11}	3×10^{-5}
^{137}Cs	1.4×10^{-9}	1.7×10^{-9}	3×10^{-5}

(a) Maximum Permissible Body Burden (ICRP 1960).

former are, in general, a small fraction of a percent of the latter, resulting in all cases except ^{90}Sr in calculated lifetime dose equivalents of less than 1 mrem. ^{90}Sr, because of its long effective half-life (15.6 yr) and relatively high uptake (40%), could deliver a bone dose equivalent of several mrem to the bone over a period of a decade or more. However, the annual dose equivalent from the intake given in Table 6 is initially (and hence always) less than the one mrem/yr de minimis level of exposure put forth by Rodger et al. (1978) and proposed as ALARA for environmental protection (Kathren et al. 1980).

Since the uranium in standard TRIGA fuel elements is largely ^{238}U, the offsite hazard from plutonium produced by neutron capture was also evaluated. Plutonium production is a complex function of the isotopic composition of the fuel, the specific operating conditions of the reactor, and time. A conservative first order estimate of plutonium production can be made for a typical TRIGA by calculating the conversion ratio (CR), which is the number of fissionable nuclei produced (i.e., ^{239}Pu) per fissionable nucleus consumed (i.e., ^{235}U) by all processes, as shown in Equation (5) (Glasstone and Sessonke 1981):

$$CR = \frac{\sigma_c}{Y\sigma_t} + \epsilon P_1 \ (1-\rho)\eta \qquad (5)$$

in which

σ_c is the microscopic neutron capture cross section of ^{235}U
σ_t the total microscopic neutron cross section for ^{238}U
Y the ratio of ^{235}U to ^{238}U
ϵ the fast fission factor
P the fast nonleakage probability
ρ the resonance escape probability, and
η the number of neutrons per fission of ^{235}U.

Equation (4) is conservative in that it provides the initial CR, which is somewhat greater than CR at a later point in the operating time history of the reactor, i.e., after some fuel has been consumed, and fission and other activation products that may compete for neutrons have been produced. For a typical TRIGA fueled with 20% enriched uranium, ϵ is 1.0680, P is 0.6464, and ρ is 0.9356 (Reed College 1967). Values for σ_c, σ_t and η are readily obtained from the literature (cf., for example, Nero 1979) as 2.7b, 683b, and 2.06, respectively, for thermal neutrons. Using these values and solving Equation (5) gives a CR of 0.11 that corresponds to a yield of 40 g of ^{239}Pu for 365 MWd of operation, since 1 MWd consumes 1 g of ^{235}U. Thus, the maximum amount of ^{239}Pu in any one element would be 0.04 x 40 = 1.6 g or 0.1 Ci. Using Equation (4), this would correspond to an uptake of about half a picocurie, which in turn would deliver an insignificant dose equivalent.

The calculated dose equivalents are extremely conservative and thus represent an extreme upper limit. If such an accident occurred, exposure levels would more realistically be one to several orders of magnitude lower.

Hence, even under the worst of circumstances, the potential exposure to personnel outside the facility from any credible fuel-handling accident would be small and of little or no health significance. Whole body and thyroid life-time dose equivalents are well within those put forth by regulatory requirements or by international bodies concerned with radiation protection (ICRP 1977, 1978; NCRP 1971, 1975, 1976).

SUGGESTIONS FOR FUTURE WORK

Three areas concerning the analysis of credible accidents would benefit from further work. The first of these is to determine if prior pulse history has adverse affects on the integrity of fuel or cladding. As TRIGAs age and elements accumulate more pulses, the need for investigation in this area becomes more pressing.

The second area relates to development of realistic source terms and modeling of the radiological consequences of a fission product release. Small-scale experiments might be conducted with fuel elements, particularly those that have a history of a large accumulation of pulses. These would determine, among other things, the physical state of the fuel-moderator and the fission product release fraction from a broken or crushed fuel element. These data could be used along with computer models to predict the nature and extent of environmental contamination and personnel exposures.

Finally, additional effort might be directed towards quantification of cladding behavior under extreme conditions as might occur following a metal-water reaction or initiating event. This area of study might include heat transfer and similar factors that could mitigate as well as potentiate the effects of the reaction.

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Credible accidents were developed and analyzed for TRIGA and TRIGA-fueled reactors. The only potential for offsite exposure appears to be from a fuel-handling accident that, based on highly conservative assumptions, would result in dose equivalents of ≤ 1 mrem to the total body from noble gases and ≤ 1.2 rem to the thyroid from radioiodines. Credible accidents from excess reactivity insertions, metal-water reactions, lost, misplaced, or inadvertent experiments, core rearrangements, and changes in fuel morphology and ZrHx composition are also evaluated, and suggestions for further study provided.

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