NUREG-1282

# Safety Evaluation Report
## on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors

Docket No. 50-163

GA Technologies, Inc.

**U.S. Nuclear Regulatory Commission**

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ABSTRACT

The properties and performance of the TRIGA higher weight percent (w%), low-enriched uranium fuels are compared with those of the currently licensed 8.5-w% fuels. Neutron physics considerations, materials properties, irradiation performance, fission product release, pulse heating, and limiting design basis were evaluated.

The performance of uranium-zirconium hydride fuel is substantially independent of uranium content up to 45-w% uranium. The behavior of the proposed 20- and 30-w% uranium fuels is indistinguishable from that of the currently approved 8.5-w% uranium fuel. Both the 20-20 and 30-20 uranium-zirconium hydride fuels are acceptable for use in the GA Mark F TRIGA reactor, and these two types of fuel are generically acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them.

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## 1 INTRODUCTION

By letter dated May 20, 1986, as supplemented by letters dated September 5, 1986 and January 22, 1987, GA Technologies, Inc. (GA) (licensee) requested an amendment to Operating License No. R-67 for its Training Research Isotopes GA (TRIGA) Mark F nonpower reactor. The amendment would change the Technical Specifications so that they would (1) define two types of TRIGA low-enriched uranium (LEU) fuel as "standard" fuel so they could be added to the list of approved standard fuels for the Mark F reactor and (2) permit continued tests under the "special" provisions for LEU fuels with uranium loadings that range from 30 weight percent (w%) to 45 w%. In addition, because the licensee is the developer and sole vendor for all TRIGA reactor fuels, it requested that the U.S. Nuclear Regulatory Commission (NRC) review and evaluate these two new standard fuels generically to qualify them for use in TRIGA reactors in general.

## 2 DISCUSSION

The two proposed LEU fuel types consist of the customary zirconium hydride matrix containing either 20 w% or 30 w% of uranium, with enrichment up to 20%. These have been designated 20-20 and 30-20 fuels, respectively. The licensee has been testing these two fuels in the GA TRIGA Mark F reactor under a license amendment issued in 1978. This amendment defined these two fuel types as "special fuels," and, accordingly, specific limitations on their use were placed in the Technical Specifications until additional tests and experiments were performed and evaluated. Since 1978, the licensee has completed various tests in the Mark F reactor and also has participated jointly in the Reduced Enrichment for Research and Test Reactor (RERTR) Program. Within this latter program, several fuel elements were irradiated in the Oak Ridge Research Reactor (ORR) to large burnups of uranium and were evaluated. Throughout the licensee's in-house program and the RERTR program, comparisons were made with TRIGA fuels that are currently approved by the NRC for use both at GA and other licensed TRIGA reactors. The tests and evaluations have been completed, and the licensee has submitted reports of the results in support of the application of May 1986.

The staff has reviewed and evaluated three higher weight percent LEU fuels (20 w%, 30 w%, and 45 w%) in accordance with the licensee's request and with respect to the rod-for-rod substitution of the new standard LEU fuels for any previously reviewed and approved standard high-enriched uranium or low-enriched uranium TRIGA fuels.

## 3 EVALUATION

This section provides a brief description of TRIGA fuels, followed by evaluations of neutron physics considerations, materials properties, irradiation performance, fission product release, pulse heating, and limiting design basis.

### 3.1 Fuel System Description

The uranium-zirconium hydride fuel used in TRIGA reactors is fabricated by hydriding an alloy that is a solid solution of uranium in zirconium. The zirconium is selectively hydrided, and the uranium remains as small metallic inclusions in the zirconium hydride matrix. The size of the uranium particles increases from 1 to 5 μm with increasing uranium content from 8.5 to 45 w%. Some important parameters for TRIGA fuels are provided in Table 1.

Table 1 Parameters for TRIGA fuels

| Type of fuel* | Weight percent Uranium | Weight percent Erbium | Uranium-235 (g/element) | Uranium enrichment (%) | $\alpha \times 10^5$ ($\Delta k/k^\circ C$) | Core lifetime (MWd) | Uranium volume percent |
|---|---|---|---|---|---|---|---|
| Original | 8.5 | 0.0 | 39 | 20 | 9.5 | 100 | 2.6 |
| FLIP | 8.5 | 1.6 | 137 | 70 | 10.5 | 3500 | 2.6 |
| LEU | 20 | 0.5 | 99 | 20 | 10.5 | 1200 | 6.8 |
| LEU | 30 | 0.9 | 162 | 20 | 8 | 3000 | 11.2 |
| LEU | 45 | 1.8 | 282 | 20 | 5 | 4000 | 19.5 |

*FLIP = Fuel Life Improvement Program conducted at GA;
LEU = low-enriched uranium.

The use of erbium burnable poison in conjunction with the higher U-235 loadings permits longer core lifetimes than would be obtainable with the original TRIGA fuel. It also permits maintaining a large prompt negative temperature coefficient of reactivity, $\alpha$, that is changed little from that of the original fuel through at least the 30-w% LEU fuel. As shown in Table 1, the volume percent (v%) of uranium increases with the increasing uranium loading but remains a small value, increasing from 2.6 v% in the original fuel to 11.2 v% for the 30-w% LEU fuel, and to 19.5 v% for the 45-w% fuel.

### 3.2 Physics

The primary intent of the GA Technologies Reactor Physics Qualification Program was to show that neutronically the 20-20 and 30-20 TRIGA LEU fuels behave essentially the same as the currently approved TRIGA Fuel Life Improvement Program (FLIP) fuel. To accomplish this, GA demonstrated the following:

- The power peaking factors in the LEU and FLIP fuels are very comparable. Any variations are due mainly to differences in the contained U-235 (not the total uranium loading).

- The prompt negative temperature coefficient, the reactivity worth, and the core lifetime of the TRIGA LEU and FLIP fuels are comparable primarily because of the adjustment of the erbium poison concentration. Also, the reactor kinetics parameters most important to power/burst behavior, prompt neutron lifetime, and effective delayed neutron fraction were similar.

### 3.3 Materials Properties

In this section, the materials properties of TRIGA fuels with higher uranium contents are reviewed relative to those of the currently licensed 8.5-w% TRIGA fuels.

### 3.3.1 Thermal Conductivity

Measurements were made of the thermal conductivity of 8.5-, 30-, and 45-w% uranium-zirconium hydride fuels. The data from these measurements, in conjunction with density and specific heat data, were used to determine the thermal conductivity of these materials. The thermal conductivity was found to be independent of uranium content within this range.

### 3.3.2 Heat Capacity

The specific heat of uranium-zirconium hydride was calculated as a function of uranium content using known specific heats for uranium and zirconium hydride and a linear interpolation. This method is a straightforward and acceptable approach, and the resulting values for heat capacity have been adequately factored into the analyses of kinetic behavior of the higher loaded LEU fuels.

### 3.3.3 Thermal Expansion

The coefficient of thermal expansion was measured for 45-w% uranium fuel and compared with that for 8- to 12-w% fuel. For a maximum power density TRIGA fuel element, the maximum radial expansion would be about 0.6% for 45-w% fuel as compared with 0.5% for 8.5-w% fuel, which is not a significant change.

### 3.3.4 Hydrogen Dissociation Pressure

The monitoring of hydrogen pressure during hydriding in the fabrication of high uranium content fuels showed that the equilibrium hydrogen dissociation pressure of the fuel depends only on the hydrogen/zirconium (H/Zr) ratio and the fuel temperature. It is independent of the uranium content.

### 3.3.5 Quench Response

Water-quench tests were performed on 45-w% uranium fuel heated to 800 to 1200°C temperatures to simulate cladding rupture and water ingress into TRIGA reactor fuel rods during operation. Minor cracking and small increases in density occurred in some samples. Hydrogen loss was accompanied by surface oxidation in all samples. These results are similar to those of earlier tests on 8.5-w% uranium fuel and indicate no difference in the response of higher uranium content fuel to water-quenching at high temperature.

### 3.3.6 Thermal Cycling

Thermal cycling tests were performed on 45-w% uranium fuel over the temperature range of 500 to 725°C, which includes the orthorhombic-to-tetragonal phase transformation at 653°C. Specimens were cycled 100 times out of pile and then 32 times in a neutron flux of $4 \times 10^{12}$ n/cm$^2$·s. There were no significant changes in dimensions in the out-of-pile tests, and a small decrease in weight was measured. The in-pile cycling test showed a small decrease in both length and diameter, which may be related to a loss of hydrogen. The dimensional stability of the high uranium content fuel is understandable considering the fine dispersion of the uranium in the zirconium hydride matrix. The dispersion of uranium in particles less than 5 μm in diameter evidently precludes anisotropic growth during cycling through the phase transformation because of accommodation by the matrix, which makes up 80% of the fuel volume in the case of 45-w% uranium fuel.

#### 3.3.7 Fuel/Cladding Compatibility

Uranium and zirconium form eutectics with iron, nickel, and chromium, the principal constituents of the four alloys (304 or 304 L stainless steel, Incoloy 800, and Hastelloy-x) that are licensed for use for fuel rod cladding according to the Technical Specifications. The uranium eutectics have lower melting temperatures than those of zirconium, which is tied up as a hydride in any case. The melting points of the eutectics with uranium are: iron, 725°C; nickel, 740°C; and chromium, 859°C. As the uranium content of the fuel is increased, the potential for the formation of low-melting eutectics is enhanced. Localized fuel melting has been observed in 45-w% uranium fuel in contact with Inconel 600 thermocouple sheathing at temperatures above 1050°C. The extent of potential eutectic melting due to fuel/cladding interaction should be less in the 20- and 30-w% uranium fuels than in 45-w% uranium fuel, but more than in the original 8.5-w% uranium fuel. In all cases, the extent of eutectic melting would be limited by the relatively small volume fraction of uranium in the fuels--11.2 v% or less for the fuels under review. The temperature at which eutectic fuel melting has been observed (1050°C) is 100C° above the lowest temperature at which cladding failure by hydrogen overpressure is predicted under conditions in which the cladding is at approximately the fuel temperature. Therefore, eutectic fuel/cladding melting does not constitute a more severe limit for fuel rod integrity than does hydrogen overpressure. It does, however, have the potential to produce fuel melting at temperatures about 80C° lower than the uranium melting point. This mechanism could lead to somewhat higher releases of fission products from the fuel rod in the temperature range 1050 to 1130°C under some accident conditions (such as loss of coolant) or during film boiling; however, these temperatures are above the safety limit of 950°C, which applies if the fuel rods are not immersed in water.

### 3.4 Irradiation Performance

The irradiation performance of 20-, 30-, and 45-w% uranium fuels was evaluated by irradiation testing to high burnups (>50% of the U-235) in the ORR and subsequent postirradiation examination. The burnups in the ORR exceed the design core lifetimes for the higher uranium content LEU fuels.

#### 3.4.1 Fuel Swelling

Fuel swelling was determined by measuring rod diametrical growth following the ORR irradiation and was compared with predictions based on a correlation developed for 8- to 10-w% uranium-zirconium hydride fuels during the Systems for Nuclear Auxiliary Power (SNAP) reactor program in the 1960s. Diametrical growth was measured on nine 45-, one 30-, and one 20-w% uranium fuel rods. The maximum swelling predicted was 0.025 in. (0.010 cm) (about a 4.6% increase in diameter). The measurements were generally within this value considering the ovality encountered and the accuracy of the measurements. Tests have shown that the existing correlation for the currently approved TRIGA fuels accounts for fuel swelling as a function of burnup in the high uranium content fuels. Rod ovality is related to the power shape within the rod, which depends on the water gap surrounding the rod.

### 3.4.2 Axial Growth

An increase in rod length indicates a fuel/cladding interaction. Fuel axial growth in the absence of interaction with the cladding is accommodated by the plenum. Six 45-, one 30-, and one 20-w% uranium fuel rods were measured to determine the increase in rod length, and the values ranged from 0.034 in. (0.014 cm) to 3.061 in. (0.024 cm) or up to 0.2% of the rod length. Axial rod growth in this range is acceptable and represents minimal cladding ratcheting by fuel/cladding mechanical interaction over the 100 startup and shutdown cycles of the ORR irradiations. An important reason for the lack of ratcheting is the good match between the thermal expansion properties of the uranium-zirconium hydride fuel and the Incoloy 800 cladding.

### 3.4.3 Rod Bowing

Measurements of bowing on five rods, as a result of ORR irradiation, showed a maximum of 0.063 in. (0.025 cm) for one rod and less than 0.025 in. (0.010 cm) for three of the rods. These are small values and are consistent with the relative ease with which the individual fuel rods were removed from and inserted into the fuel cluster many times during the irradiations. The rods were rotated 180° around their vertical axes about four times during the irradiation as an approximate method of balancing bending forces due to asymmetries in fuel burnup. Rod rotation is the accepted practice in TRIGA reactors, and the bowing measured at high burnup for the high uranium content rods is consistent with experience with the currently approved TRIGA fuels.

### 3.4.4 Hydrogen Migration

During sustained irradiation, hydrogen tends to migrate from the hot radial center of the fuel to a cooler annulus near the pellet periphery. Hydrogen/zirconium (H/Zr) ratios can vary by ±10 to 15% of their initial values. The increased H/Zr ratio near the outer radius of the fuel, coupled with high peak fuel temperatures that occur at the outer radius during a pulse, can cause excessive hydrogen pressures in the fuel matrix, which can weaken and deform the fuel matrix and cause excessive swelling and fuel element deformation. Such fuel element deformation has been observed with FLIP fuel irradiated in the Texas A&M University Nuclear Science Center TRIGA reactor. This experience suggests that pulse sizes or maximum fuel temperatures should be limited in higher burnup cores to account for the effects of hydrogen redistribution. This effect, however, is independent of uranium content in the TRIGA fuel, and the evidence suggests that an equilibrium hydrogen distribution is established within a moderate time scale.

## 3.5 Fission Product Release

The fission product source term for accident analysis in TRIGA reactors consists of those radionuclides that can be released from the fuel/cladding gap on loss of cladding integrity. Thus, it is the gap inventory of fission gases resulting from normal operation that is of interest.

A correlation of fractional release of fission gas as a function of irradiation temperature has been developed for 8.5-w% uranium fuels through measurements of release-to-birth ratios of fission products by heating fuel specimens in a furnace during irradiation within a TRIGA reactor. Fission product release tests have been run on trace-irradiated high uranium content fuels, and the fractional releases up to 400°C show somewhat higher releases than were found in earlier studies. On the other hand, similar release fractions for low and high uranium content fuels have been measured after irradiation at 800°C, and relatively smaller release fractions in high uranium content fuels have been measured at 1100°C. At 1100°C, the lower release fractions appear to be associated with larger uranium particle sizes (5 μm) in the high uranium content fuel (45 w%) compared with 1-μm uranium particles common in 8.5-w% uranium fuel.

Measurement of fission gas in the plenum of one 45-w% uranium rod irradiated to 64% U-235 burnup in the ORR indicated a release fraction of approximately $4 \times 10^{-4}$. This value is greater than that for trace-irradiated 45-w% uranium fuels, which was approximately $1 \times 10^{-4}$ at 400°C (roughly the average of fuel centerline peak and surface temperatures in the ORR irradiation), and greater than the fraction ($1 \times 10^{-5}$) for 5.5% U-235 burnup in 8.5-w% uranium fuel. However, the release fractions measured at 400°C are small even for the 45-w% fuel. The data on releases from high-temperature trace-irradiated high uranium content fuel indicate that the somewhat larger uranium particles are of some benefit. In general, the data suggest that fission product release from the high uranium content LEU fuel is comparable to that from the currently approved 8.5-w% uranium fuel. However, confirmatory data are not available on release fractions of fission gases from high-burnup fuel irradiated at temperatures above approximately 800°C. When such data are available from GA, the staff will review the results. In the meantime, no special limitations will be imposed on the operating conditions at GA because the data currently available provide reasonable assurance that potential risks of cladding failure are within the limits previously evaluated.

### 3.6 Pulse Heating

A 45-w% uranium LEU fuel rod that was instrumented for measuring temperature and pressure was subjected to a series of 30 power pulses in a TRIGA reactor to maximum temperatures in the range of 1050 to 1100°C. Only very modest (generally less than 2 psi) pressure pulses were measured in the rod as a result of the pulsing, in agreement with previous data on negligible hydrogen release during the pulsing of 8.5-w% uranium fuel to temperatures up to 1150°C. All surveillance examinations on rod deformation were satisfactory. Tests have shown that the pulse response of uranium-zirconium hydride TRIGA fuel is independent of the uranium content of the fuel and is dominated by the behavior of the zirconium hydride, along with the prompt temperature coefficient of reactivity.

As mentioned in Section 3.4.4, pulse sizes or maximum fuel temperatures should be limited in higher burnup cores to account for the effects of hydrogen redistribution. GA has proposed changes to Sections 5.5.1 and 9.2.2 of the Technical Specifications that address this potential problem by imposing limits on maximum operating temperatures in standard TRIGA fuels. In conjunction with the surveillance requirements in Section 5.5.1(c) of the Technical Specifications, the staff concludes that there is reasonable assurance that the effects of hydrogen migration will not lead to unreviewed fission product releases.

### 3.7 Limiting Design Basis

The basic limit for TRIGA fuel is dictated by the dissociation of hydrogen from the uranium-zirconium hydride fuel. A fuel temperature safety limit of 1150°C for pulsing precludes loss of cladding integrity when the cladding temperature is below 500°C. When the cladding temperature equals the fuel temperature, the fuel temperature limit is 950°C. Experience with fuel rod deformation following pulsing of TRIGA fuel rods that had experienced significant hydrogen migration has shown that peak pulse temperatures should be reduced for fuel in which equilibrium hydrogen redistribution had occurred.

Tests of uranium-zirconium hydride fuels have shown that the limiting design basis for the operation of TRIGA fuels is independent of uranium content in the fuel up to 45 w%.

## 4 ENVIRONMENTAL CONSIDERATIONS

The amendment involves changes in the installation or use of a facility component located within the restricted area as defined in Part 20 of Title 10 of the Code of Federal Regulations (10 CFR 20) and changes in inspection or surveillance requirements. The staff has determined that the amendment involves no significant hazards consideration, there is no significant change in the types or significant increase in the amounts of any effluents that may be released off site, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

The staff has concluded, on the basis of the considerations discussed above, that (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a safety margin, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

## 5 CONCLUSION

The staff concludes that the performance of uranium-zirconium hydride fuel is substantially independent of uranium content up to 45-w% uranium. The behavior of the proposed 20- and 30-w% uranium fuels is indistinguishable from that of the currently approved 8.5-w% uranium fuel. The categorization of fuels with uranium contents between 30 and 45 w% as "special fuels" provides continuing limitations on its use while additional tests are in progress.

On the basis of its evaluation, the staff concludes that both the 20-20 and 30-20 uranium-zirconium hydride fuels are acceptable for use in the GA Mark F TRIGA reactor, as proposed and analyzed. The staff further concludes that these two types of fuel are generically acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them.

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