January 30, 2006

Mr. Warren Day, Reactor Administrator
United States Department of the Interior
Geological Survey
Box 25046, MS 974
Denver Federal Center
Denver, CO 80225-0046

SUBJECT: UNITED STATES GEOLOGICAL SURVEY — AMENDMENT RE: USE OF ALUMINUM-CLAD FUEL (TAC NO. MC5120)

Dear Mr. Day:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 11 to Facility License No. R-113 for the United States Geological Survey TRIGA Reactor. The amendment consists of changes to the technical specifications (TSs) in response to your application of November 16, 2004, as supplemented on December 3, 2004, and February 8, April 11, and August 25, 2005.

The amendment allows the use of aluminum-clad fuel in the reactor.

A copy of the safety evaluation supporting Amendment No. 11 is also enclosed.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-274

Enclosures: 1. Amendment No. 11
2. Safety Evaluation

cc w/enclosures: Please see next page

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Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

DEPARTMENT OF THE INTERIOR

UNITED STATES GEOLOGICAL SURVEY

DOCKET NO. 50-274

AMENDMENT TO FACILITY LICENSE

Amendment No. 11
License No. R-113

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that

   A. The application for an amendment to Facility License No. R-113 filed by the Department of the Interior, U.S. Geological Survey (the licensee) on November 16, 2004, as supplemented on December 3, 2004, and February 8, April 11, and August 25, 2005, conforms to the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission as stated in Chapter I of Title 10 of the *Code of Federal Regulations* (10 CFR);

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance that (i) the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) such activities will be conducted in compliance with the regulations of the Commission;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. This amendment is issued in accordance with the regulations of the Commission as stated in 10 CFR Part 51, and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105 and publication of a notice for this amendment is not required by 10 CFR 2.106.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 3.B of Facility License No. R-113 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 11, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brian E. Thomas, Branch Chief
Research and Test Reactors Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Enclosure: Appendix A, Technical Specifications Changes

Date of Issuance: January 30, 2006

ENCLOSURE TO LICENSE AMENDMENT NO. 11

FACILITY LICENSE NO. R-113

DOCKET NO. 50-274

Replace the following pages of Appendix A, "Technical Specifications," with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
4	4
5	5
5a	5a

2. The pool water shall be sampled for conductivity at least weekly. Conductivity averaged over a month shall not exceed 5 micromhos per cm^2. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

3. The control console shall have an audible and visual water level alarm that will actuate when the reactor tank water level is between 12 and 24 inches below the top lip of the tank. This water level alarm shall be functionally tested monthly, not to exceed 45 days between tests. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

4. The pool water shall be sampled for pH at quarterly intervals, not to exceed 4 months. The pH level shall be within the range of 4.5 to 7.5 for continued operation. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

D. Reactor Core

1. The core shall be an assembly of TRIGA aluminum or stainless steel clad fuel-moderator elements, nominally 8.0 to 12 wt% uranium, arranged in a close-packed array except for (1) replacement of single individual elements with incore irradiation facilities or control rods; (2) two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be water or a combination of graphite and water. The reactor shall not be operated in any manner that would cause any stainless-steel clad fuel element to produce a calculated steady state power level in excess of 22 kW. Aluminum clad fuel-moderator elements will only be allowed in the F and G rings of the core assembly.

2. The excess reactivity above cold critical, without xenon, shall not exceed 4.9% delta k/k with experiments in place.

3. Fuel temperatures near the core midplane in either the B or C ring of elements shall be continuously recorded during the pulse mode of operation using a standard thermocouple fuel element. The thermocouple element shall be of 12 wt% uranium loading if any 12 wt% loaded elements exist in the core. The reactor shall not be operated in a manner which would cause the measured fuel temperature to exceed 735°C in a stainless steel clad element in the B ring or 652°C in a stainless steel clad element in the C ring.

4. Power levels during pulse mode operation that exceed 2500 megawatts shall be cause for the reactor to the shut down pending an

investigation by the reactor supervisor to determine the reason for the pulse magnitude. His evaluation and conclusions as to the reason for the pulse magnitude shall be submitted to the Reactor Operations Committee for review. Pulse mode operation will not be resumed until approved by the Committee.

5. If the reactor is operated in the pulse mode during intervals of less than six months, the reactor shall be pulsed semiannually with a reactivity insertion of at least 1.5% delta k/k to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value. If the reactor is not pulsed during intervals of six months, then for the first pulse after the time of the last comparative pulse, the reactor shall be pulsed with a reactivity insertion of at least 1.5% delta k/k to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.

6. Each standard fuel element shall be checked for transverse bend and longitudinal elongation after the first 100 pulses of any magnitude and after every 500 pulses or every 60 months, whichever comes first.

   During the first 5 years of aluminum-clad fuel usage, annual fuel transverse bend and longitudinal elongation measurements will be made on 20% of the aluminum-clad fuel elements that have been in the core at any time during that year. The measurement schedule will be controlled such that different fuel elements are measured each year for this initial 5-year period. After this initial 5 years of aluminum-clad fuel usage, if no generic problems have been detected, the inspection schedule will revert back to the standard fuel 60-month schedule.

   The limit of transverse bend shall be 1/16-inch over the total length of the clad portion of the element (excluding end fittings). The limit on longitudinal elongation shall be 1/10 inch for stainless steel clad elements and ½-inch for aluminum clad elements. The reactor shall not be operated in the pulse mode with elements installed which have been found to exceed these limits.

Any element which exhibits a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation of routine operation. Fuel elements that have been removed from service do not need to be checked for transverse bend or longitudinal elongation.

7. Observance of the license and technical specification limits for the GSTR will limit the thermal power produced by any single fuel element to less than 22 kW if the reactor has at least 100 fuel elements in the core. Therefore the reactor must have at least 100 fuel elements in the core if it is to be operated above 100 kW. Operations with less than 100 fuel elements in the core will be restricted to a maximum thermal power of 100 kW.

E. Control and Safety Systems

1. The standard control rods shall have scram capability and the poison section shall contain borated graphite, or boron and its compounds in solid form as a poison in an aluminum or stainless steel clad.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO

FACILITY LICENSE NO. R-113

DEPARTMENT OF THE INTERIOR

UNITED STATES GEOLOGICAL SURVEY

DOCKET NO. 50-274

1.0 INTRODUCTION

By letter dated November 16, 2004, as supplemented on December 3, 2004, and February 8, April 11, and August 25, 2005, the U.S. Geological Survey (USGS or the licensee) submitted a request for amendment of the Technical Specifications (TSs), Appendix A of Facility License No. R-113 for the USGS TRIGA Reactor (GSTR). The requested changes to the TSs would allow the use of aluminum-clad fuel elements in the reactor.

2.0 BACKGROUND

The GSTR is a Mark I TRIGA reactor licensed to operate at steady-state thermal power levels up to 1 megawatt and in the pulse mode with reactivity insertions up to 2.1% $\Delta k/k$. The reactor is on the grounds of the Denver Federal Center near Denver, Colorado, and is used to perform nuclear research in the basic earth sciences in support of the USGS. The reactor is currently authorized to use low-enriched stainless-steel-clad uranium-zirconium hydride TRIGA fuel containing 8.5 to 12 weight percent (wt%) uranium. The licensee has acquired 56 fuel elements from the Alan J. Blotcky Nuclear Reactor, which was operated by the Veterans Administration (VA) in Omaha, Nebraska. The VA reactor has been permanently shut down and plans to decommission. The fuel elements acquired from the VA are low-enriched aluminum-clad uranium-zirconium hydride TRIGA fuel containing 8.0 wt% uranium. These fuel elements have low fuel burnups and significant remaining life.

3.0 EVALUATION

The regulations in 10 CFR 50.36 require nuclear reactors to have TSs. The TSs contain limitations on the types of fuel elements the licensee may possess and limitations on the use of fuel elements in the reactor. The staff has determined that the changes proposed by the licensee continue to meet the requirements of 10 CFR 50.36.

The licensee has used a significant amount of the uranium in the existing core so that the remaining excess reactivity in the core is not sufficient under all allowed operational conditions to overcome poisons, such as xenon, that build up in the core during operation. The licensee therefore needs to replace some of its high-burnup fuel with fresher fuel to increase the excess reactivity available to operate the reactor. The licensee acquired some low-burnup fuel from the VA when the VA permanently shut down the Alan J. Blotcky Nuclear Reactor.

General Atomics (GA), the manufacturer of TRIGA reactors, has produced various types of fuel elements for TRIGA reactors over the years. One of the first forms of TRIGA fuel developed by GA was a low-enriched, aluminum-clad, uranium-zirconium hydride (low-hydride type) fuel. Later, GA developed a low-enriched, stainless-steel-clad, uranium-zirconium hydride (high-hydride type) fuel. USGS currently uses stainless-steel-clad fuel with two different weight percents of uranium, 8.5 wt% and 12 wt%. The higher the weight percent of uranium in the fuel, the more uranium there is in each fuel element and the longer the fuel can be used in the reactor (Amendment No. 8 dated March 16, 1998, approved the use of 12 wt% fuel in the reactor).

The low-hydride ($U\text{-}ZrH_{1.0}$) aluminum-clad-fuel and the high-hydride ($U\text{-}ZrH_{1.6}$) stainless-steel-clad fuel have different failure mechanisms because the different ratios of hydrogen to zirconium atoms in the fuel places the two fuel types at different locations on the U-ZrH phase diagram. Fuel damage in the low-hydride fuel is caused by a phase change that the fuel undergoes at about 530°C. The fuel occupies more volume in the phase above 530°C than in the phase below 530°C. This change in volume causes the fuel meat to swell and press on the aluminum clad, causing it eventually to fail. During reactor operation, low-hydride fuel is kept below 530°C, which is the safety limit, to prevent this mode of clad failure.

The high-hydride fuel failure mechanism is not dependent on a change in phase of the U-ZrH with increasing temperature. Instead, as the fuel temperature increases, pressure builds up inside the fuel element from hydrogen produced by dehydriding of the fuel and other gases in the gap between the fuel meat and the cladding. With increasing temperature, the pressure from these gases inside the fuel element increases, and the stainless-steel-clad yield strength decreases until the cladding fails. Because the physical properties of the stainless steel cladding vary with temperature, the fuel failure temperature varies with clad temperature. For clad temperature at or below 500°C, the peak fuel temperature safety limit is 1150°C. For clad temperature above 500°C, the peak fuel temperature safety limit is 950°C.

In addition to the safety limits discussed above, there is also a high-hydride-fuel steady-state operational fuel temperature design limit of 750°C based on consideration of irradiation- and fission-product-induced fuel growth and deformation. The fuel growth is time and temperature-dependent. A maximum temperature of 750°C is used as the operational design basis temperature because the resulting average core fuel temperatures lead to insignificant calculated fuel growth from temperature-dependent irradiation effects. This is a steady-state operating limit. Because the time at high temperature during pulsing is short, this limit is not a concern in the pulse mode of operation.

The fuel acquired from the VA is aluminum-clad low-hydride fuel. The NRC has approved the use of mixed cores containing both aluminum and stainless-steel-clad fuel (see NUREG-1312, "Safety Evaluation Report Related to the Renewal of the Facility License for the Research Reactor at the Dow Chemical Company," and NUREG-1096, "Safety Evaluation Report Related to the Renewal of the Operating License for the TRIGA Training and Research Reactor at the University of Utah"). Also, a number of NRC-licensed TRIGA reactors have operated and continue to operate on all aluminum-clad fuel cores. Because the aluminum-clad fuel safety limit is governing, core limits must be chosen to protect the aluminum clad fuel from over-heating.

The licensee proposed changes to the TSs to allow the use of aluminum-clad low-hydride fuel in the reactor. The licensee provided a technical justification showing that controlling the temperature of the stainless-steel-clad fuel in the B and C rings of the reactor core provides reasonable assurance of protection of the safety limit for the aluminum clad fuel in the F and G rings of the reactor core.

TS D.1 concerning reactor core conditions currently reads as follows:

1. The core shall be an assembly of TRIGA stainless steel clad fuel-moderator elements, nominally 8.5 to 12 wt% uranium, arranged in a close-packed array except for (1) replacement of single individual elements with incore irradiation facilities or control rods; (2) two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be water or a combination of graphite and water. The reactor shall not be operated in any manner that would cause any fuel element to produce a calculated steady state power level in excess of 22 kW.

The licensee has proposed changing this TS to read as follows (with bold type showing proposed changes):

1. The core shall be an assembly of TRIGA **aluminum or** stainless steel clad fuel-moderator elements, nominally 8.**0** to 12 wt% uranium, arranged in a close-packed array except for (1) replacement of single individual elements with incore irradiation facilities or control rods; (2) two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be water or a combination of graphite and water. The reactor shall not be operated in any manner that would cause any **stainless-steel clad** fuel element to produce a calculated steady state power level in excess of 22 kW. **Aluminum clad fuel-moderator elements will only be allowed in the F and G rings of the core assembly.**

The licensee has also proposed changes to TS D.3 concerning fuel temperatures in the reactor core. The TS currently reads as follows:

3. Fuel temperatures near the core midplane in either the B or C ring of elements shall be continuously recorded during the pulse mode of operation using a standard thermocouple fuel element. The thermocouple element shall be of 12 wt% uranium loading if any 12 wt% loaded elements exist in the core. The reactor shall not be operated in a manner which would cause the measured fuel temperature to exceed 800°C.

The licensee has proposed changing this TS to reads as follows (with bold type showing the proposed change):

3. Fuel temperatures near the core midplane in either the B or C ring of elements shall be continuously recorded during the pulse mode of operation using a standard thermocouple fuel element. The thermocouple element shall be of 12

> wt% uranium loading if any 12 wt% loaded elements exist in the core. The reactor shall not be operated in a manner which would cause the measured fuel temperature to exceed **735°C in a stainless steel clad element in the B ring or 652°C in a stainless steel clad element in the C ring.**

The licensee provided information on the physical attributes of the 8.5 and 12 wt% stainless-steel-clad fuel and the 8.0 wt% aluminum-clad fuel. The most significant difference from a safety standpoint is the different H/Zr atom ratios which result in the different fuel failure mechanisms and safety limits, as discussed above. The aluminum-clad fuel elements are slightly longer than the stainless-steel-clad fuel elements (28.37 inches [72.06 cm] for stainless steel verses 28.44 inches [72.23 cm] for aluminum) and larger in diameter (1.47 inch [3.73 cm] for stainless steel versus 1.48 inch [3.76 cm] for aluminum). The differences are not significant and the licensee discussed the fact that sufficient clearance exists in the holes in the grid plate (1.505 in [3.82 cm]) to accommodate the larger diameter of the aluminum clad fuel. As discussed below, fuel elements are periodically checked for transverse bending and longitudinal elongation. One purpose of the checks is to help ensure that fuel elements do not develop transverse bends to the point where they cannot be easily removed from the grid plate.

The uranium in the aluminum clad fuel has a lower weight percent than the stainless-steel-clad fuel. This difference in weight percent and the shorter fuel meat (15 inches [38.1 cm] for stainless steel versus 14 inch [35.6 cm] for aluminum) result in there being less uranium in the aluminum-clad fuel (36 grams for aluminum versus 39 grams for 8.5 wt% stainless steel and 55 grams for 12 wt% stainless steel). This means that everything else being equal, an aluminum-clad fuel element will generate less heat (power) than a stainless-steel-clad fuel element.

The ends of the fuel meat of the aluminum-clad fuel contain a neutron poison in the form of samarium wafers. The purpose of the poison is to maintain the reactivity worth of the fuel element at a constant value during initial operation. This type of fuel element has been safely used at other NRC-licensed TRIGA research reactors.

The nuclear characteristics of the two fuels also differ. The prompt neutron lifetime of the aluminum clad fuel is longer (60 μsec) than that of the stainless steel clad fuel (43 μsec). This is because the high-hydride stainless-steel-clad fuel contains more hydrogen (neutron moderator) than the aluminum-clad fuel and neutrons are thermalized more quickly. The prompt-negative temperature coefficient of the aluminum-clad fuel is smaller (-11×10^{-5} Δk/k per degree C) than that for the stainless-steel-clad fuel (-13×10^{-5} Δk/k per degree C). Finally, the effective delayed neutron fraction for the aluminum-clad fuel is slightly larger (0.0073) than for the stainless-steel-clad fuel (0.007).

Aluminum-clad fuel with these characteristics has been approved by NRC in both aluminum-clad and stainless-steel-clad mixed cores and all aluminum-clad cores (this fuel was previously used in the NRC-licensed research reactor at VA). The NRC staff finds that the basic characteristics of the aluminum clad fuel are acceptable for use in the GSTR.

The fuel is arranged on the grid plate in rings. The licensee proposed limiting the use of aluminum-clad fuel elements to the F and G rings of the core. These are the two outer fuel rings. Generally, the further fuel is from the center of the reactor core, the less power the fuel generates per fuel element and the lower the temperature of the fuel during operation. The temperature of the fuel during both steady-state and pulsing operation needs to be considered.

The licensee has fuel elements that contain thermocouples that can measure the temperature of the fuel during operation.

The actual peak fuel temperature in the fuel element may differ from the measured temperature. The actual temperature is determined by calculation. The NRC staff asked the licensee to discuss the relationship between the measured temperature in a thermocouple fuel element and the actual maximum temperature in the fuel element. The licensee discussed two sources for the difference between measured and actual temperature: the accuracy of the temperature-measuring instrumentation and the difference in locations of the thermocouple and the hot spot in the fuel element. The instrumentation accuracy is about ±5°C (for conservatism, calculations assume that the instrument is reading low).

The effect of thermocouple location differs in measuring steady-state and pulsing temperatures. The maximum temperature during steady-state operation is on the fuel element centerline near the thermocouple location. The calculated actual temperature is about 10°C more than the measured temperature. The effect of the instrumentation and thermocouple location during steady-state operation is that the actual peak fuel element temperature could be up to 15°C higher than the measured temperature.

During pulsing, the peak temperature in the fuel is near the fuel cladding. The difference between the measured fuel temperature and the actual peak temperature depends on the amount of reactivity added to the reactor during the pulse. The difference increases as the reactivity addition increases. At the licensed limit for the GSTR, 2.1% delta k/k, the difference is about 25% of the measured temperature. For example, a measured temperature of 400°C would represent an actual temperature of 500°C. To this would be added the effect of instrumentation accuracy of ±5°C. The licensee considered these fuel element temperature accuracies in discussing the proposed changes to the TSs.

The licensee presented data on measurements of fuel temperatures in the core. For 1 MW steady-state operation with a 125-element core a temperature of 365°C was measured in a 12 wt% stainless steel thermocouple element located in the C ring of the reactor. An additional measurement in the B ring of the reactor with a 8.5 wt% instrumented fuel element resulted in a temperature of 344°C. Fuel elements in the B ring usually produce the highest power in the core and thus have the highest temperature. An instrumented fuel element with 12 or 8.5 wt% fuel would produce more power than a similar element with 8 wt% uranium content and thus would have a higher temperature. The licensee measured a fuel temperature of 202°C in the F ring and 172°C in the G ring. These measurements were taken with a 8.5 wt% instrumented fuel element with a coolant temperature of 21°C. The licensee calculates that power produced in a fuel element in the F ring is 56% of the power in a B ring element and the power produced in a fuel element in the G ring is 47% of the power in a B ring element.

Proposed TS D.3 limits the measured temperature of a stainless-steel-clad fuel element to 735°C in the B ring and 652°C in the C ring, which corresponds to a calculated peak fuel of 750°C in the B ring of the reactor. The instrumented fuel element is restricted by TS D.3 to the B or C ring. The licensee performed calculations where the temperature of the fuel element in the B ring was set at 750°C with a coolant temperature of 60°C. The limiting case was a 125-element core. The resulting temperatures were 667°C in the C ring, 447°C in the F ring, and 383°C in the G ring. The calculated reactor power level needed for a B ring temperature of 750°C was about 2.1 MW, significantly above the high power scram limit of 1.1 MW. These calculations show that the proposed measured temperature limits of 735°C (calculated temperature of 750°C) in the B ring and 652°C (calculated temperature of 667°C) in the C ring result in temperatures in the F and G rings below the aluminum-clad fuel safety limit of 530°C.

The licensee also calculated fuel element temperatures for a number of allowable core conditions. The number of fuel elements in the core, the power level, and the coolant temperature were varied. Calculations were performed for a 125-element core, which is the current size of the licensee's core, and for a 100-element core, which is the smallest core allowed by the technical specifications for operation at a power level above 100 kW. The power levels used were 1 MW, the licensed power level, and 1.1 MW, the high power level scram setpoint limit. Coolant temperatures were 25°C, the normal operating temperature, and 60°C, the technical specification limit on coolant temperature. The limiting case for temperatures in the F ring was a 100-element core with a power level of 1.1 MW and a coolant temperature of 60°C. The maximum calculated fuel temperatures for this case were 702°C in the B ring, 652°C in the C ring, 284°C in the F ring, and 213°C G ring. The limiting case for temperatures in the G ring was a 125-element core with a power level of 1.1 MW and a coolant temperature of 60°C. The maximum calculated fuel temperatures for this case were 415°C in the B ring, 373°C in the C ring, 259°C in the F ring, and 226°C in the G ring. The temperatures in the F and G rings are below the aluminum-clad fuel safety limit of 530°C. These calculations provide additional assurance that the mixed core of aluminum-clad and stainless-steel-clad fuel elements can be operated safely during steady-state operation under the limitation proposed by the licensee.

The purpose of the reactor protection system is to protect the safety limits of the fuel by scramming the reactor before a condition develops that can lead to fuel damage. The reactor protection system in the GSTR has two power channels either of which will scram the reactor if the power level in the reactor exceeds 110% of licensed power (1.1MW). The discussion above shows that scramming the reactor at this power level will prevent temperatures in fuel elements in the F and G rings from exceeding the aluminum-clad fuel safety limit of 530 °C.

Based on the above discussed measurements and calculations of fuel temperatures in the GSTR, the limit on measured fuel temperatures of 735°C in the B ring and 652°C in the C ring of the reactor core, and the restriction on the use of aluminum clad fuel elements to the F and G rings of the reactor core, the staff concludes that aluminum-clad fuel elements in the GSTR will be operated in the steady-state mode below the safety limit temperature of 530°C. The use of a mixed core of aluminum-clad and stainless-steel-clad fuel elements during steady-state operation as proposed by the licensee is therefore acceptable.

The licensee also discussed the effect of a mixed aluminum-clad and stainless-clad core on pulsing. As discussed above, the nuclear characteristics of the low-hydride and high-hydride fuel differ. With the F and G rings containing all aluminum clad fuel, the core would be about one-half aluminum-clad fuel. A pulse in a mixed aluminum and stainless steel clad core would be broader (have a longer period) than in a stainless-steel-clad core due to the longer prompt neutron lifetime of the aluminum-clad fuel. The smaller prompt-negative coefficient of the aluminum-clad fuel would result in a mixed core having larger pulse energy and a higher fuel temperature than an all stainless-steel-clad core. The licensee stated that in an all aluminum-clad core, the maximum fuel temperature would increase by about 20%.

The licensee also measured fuel temperature during pulsing. The pulse reactivity addition limit is 2.1% Δk/k (3.00$). Maximum temperature during a 2.49$ pulse in a 125-element core was 320°C (400°C calculated temperature adjusting for temperature measurement error) as measured by a 12 wt% stainless steel thermocouple fuel element located in the B ring of the reactor. A 3.00$ pulse in a 78-element core of new 8.5 wt% fuel had a measured B ring fuel temperature of 411°C (514°C calculated temperature adjusting for temperature measurement error). Fuel temperatures in the F and G rings of the reactor will be significantly lower. The licensee calculated fuel temperatures for a 3.00$ pulse in a 100-element core with a coolant temperature of 60°C of 500°C in the B ring, 442°C in the C ring, 296°C in the F ring, and 252°C in the G ring. Assuming an all aluminum-clad core would increase the fuel temperatures to 355°C in the F ring and 303°C in the G ring. In addition, GA pulsed the prototype TRIGA reactor with an aluminum-clad core over 1000 times with 2.25% Δk/k (3.08$) reactivity additions with acceptable results.

Based on measured and calculated temperatures in TRIGA reactors with aluminum-clad fuel during pulsing and the restriction of aluminum-clad fuel elements to the F and G rings in the GSTR, the staff concludes that aluminum-clad fuel elements in the GSTR will be operated in the pulse mode below the safety limit temperature of 530°C.

Based on the discussion above, the staff concludes that the licensee has shown that aluminum-clad fuel can be safely used in the F and G rings of the reactor core during both steady-state and pulsing operation without exceeding the safety limit for aluminum clad fuel. Therefore, the NRC staff finds that the use of aluminum clad fuel in the reactor is acceptable.

The licensee has proposed changes to the TS D.7 requirements for the reactor core. TS D.7. currently reads as follows:

> 7. The power produced by each fuel element while operating at the rated full power shall be calculated if the reactor is to be operated at greater than 100 kW with less than 100 fuel elements in the core. Recalculations shall be performed:
>
>   a) at 6 $\pm$ 1 month intervals, or
>   b) whenever a core loading change occurs.
>
>   Power per element calculations are not required at any time that the core contains at least 100 fuel elements or if reactor power is limited to 100 kW. If the calculations show that any fuel element would produce more than 22 kW, the reactor shall not be operated with that core configuration.

The licensee proposes to change this TS to read as follows:

> 7. Observance of the license and technical specification limits for the GSTR will limit the thermal power produced by any single fuel element to less than 22 kW if the reactor has at least 100 fuel elements in the core. Therefore the reactor must have at least 100 fuel elements in the core if it is to be operated above 100 kW. Operations with less than 100 fuel elements in the core will be restricted to a maximum thermal power of 100 kW.

The licensee has proposed this change because it does not intend to operate the reactor at a power level above 100 kW with a reactor core containing fewer than 100 fuel elements. The proposed wording removes the option to operate the reactor over 100 kW with less than 100 fuel elements if calculations are performed to show that the power produced by any single fuel element is less than 22 kW. The 22 kW limit was established by Amendment No. 8, issued on March 16, 1998, which allowed the use of 12 wt% fuel. The purpose of the 22 kW limit was to ensure that nucleate boiling would not occur. Because the licensee’s proposed change maintains the restriction that the core must contain at least 100 elements to be operated above 100 kW, thus maintaining the power limit of 22 kW by any single fuel element, the proposed change is acceptable to the staff.

The licensee has proposed changes to TS D.6. concerning surveillance requirements for fuel elements. TS D.6. currently reads as follows:

> 6. Each standard fuel element shall be checked for transverse bend and longitudinal elongation after the first 100 pulses of any magnitude and after every 500 pulses or every 60 months, whichever comes first. The limit of transverse bend shall be 1/16-inch over the total length of the clad portion of the element (excluding end fittings). The limit on longitudinal elongation shall be 1/10 inch. The reactor shall not be operated in the pulse mode with elements installed which have been found to exceed these limits.
>
> Any element which exhibits a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation of routine operation.

The licensee proposes to change this TS to read as follows:

> 6. Each standard fuel element shall be checked for transverse bend and longitudinal elongation after the first 100 pulses of any magnitude and after every 500 pulses or every 60 months, whichever comes first.
>
> **During the first 5 years of aluminum-clad fuel usage, annual fuel transverse bend and longitudinal elongation measurements will be made on 20% of the aluminum-clad fuel elements that have been in the core at any time during that year. The measurement schedule will be controlled such that different fuel elements are measured each year for this initial 5-year period. After this initial 5 years of aluminum-clad fuel usage, if no generic problems have been detected, the inspection schedule will revert back to the standard fuel 60-month schedule.**
>
> The limit of transverse bend shall be 1/16-inch over the total length of the clad portion of the element (excluding end fittings). The limit on longitudinal elongation shall be 1/10 inch **for stainless steel clad elements and ½-inch for aluminum clad elements.** The reactor shall not be operated in the pulse mode with elements installed which have been found to exceed these limits.
>
> Any element which exhibits a clad break as indicated by a measurable release of fission products shall be located and removed from service before continuation

of routine operation. **Fuel elements that have been removed from service do not need to be checked for transverse bend or longitudinal elongation.**

The licensee has proposed limits on transverse bending (bowing) and longitudinal elongation for the aluminum-clad fuel elements. The transverse bending limit is the same as for the stainless steel, 1/16 inch (0.159 cm) over the total length of the clad portion of the element. The longitudinal elongation limit proposed by the licensee is ½ -inch (1.27 cm). The proposed values are within the values suggested by the designer of the reactor, GA, and within the values found acceptable to the NRC staff in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors."

The licensee has proposed a schedule for performing bend and elongation measurements on the fuel. This schedule recognizes that the aluminum-clad fuel elements will be subjected to higher power levels than in the VA reactor. Twenty percent of the aluminum-clad fuel used in the reactor would be measured each year for the first 5 years of aluminum-clad fuel element usage. Different fuel elements would be measured each year such that over the 5-year interval all of the aluminum-clad fuel elements used in the reactor would be measured. This would allow potential generic problems to be detected early and would create a pool of data on aluminum-clad fuel performance. If no generic problems with the fuel are detected over the 5-year period, the licensee would return to its longstanding surveillance intervals.

Because the licensee's proposed values are within the values recommended by the reactor manufacturer and accepted by NRC staff, the NRC staff finds acceptable the licensee's proposed transverse bend and longitudinal elongation limits and surveillance intervals.

The licensee has added a statement to the TS that fuel elements that have been removed from service need not be checked for transverse bending or longitudinal elongation. The purpose of measuring transverse bending and longitudinal elongation is to prevent fuel elements with unacceptable transverse bending and longitudinal elongation from being used in the reactor. Having been removed from service, these elements will not be used in the reactor. The licensee's proposed TS addition is therefore acceptable to the NRC staff.

The licensee's current TSs contain limits on water chemistry to control corrosion of reactor components. The TSs currently limit the conductivity of the primary coolant. In a request for additional information, the NRC staff asked the licensee about the need to control primary coolant pH given the addition of aluminum-clad fuel to the reactor. Several references were discussed that indicated the importance of controlling pH to maintain a protective oxide film on aluminum surfaces. Based on research in the literature (DOE Handbook 1015/1-93, "Department of Energy Fundamentals Handbook," Module 2, Corrosion of Aluminum), the licensee proposed a new TS C.4 on control of primary coolant pH:

4. The pool water shall be sampled for pH at quarterly intervals, not to exceed 4 months. The pH level shall be within the range of 4.5 to 7.5 for continued operation. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

The NRC staff has reviewed the licensee's proposed TS. The staff has determined that the proposed pH limits will minimize corrosion of the aluminum-clad fuel, and will not result in undue corrosion of the stainless-steel-clad fuel that is also used in the reactor. Therefore, the proposed pH limits are acceptable to the NRC staff.

The licensee discussed the maximum hypothetical accident for the reactor, the failure in air of the fuel element with the highest power production that has been operating for a very long time. Because the aluminum-clad fuel elements have a lower weight percent of uranium than the stainless-steel-clad elements and are restricted to the F and G rings of the reactor, the amount of fission products available for release from an aluminum-clad fuel element will be less than from the fuel element assumed for the maximum hypothetical accident. The staff concludes that the use of aluminum-clad fuel elements in the reactor core will not change the results of the maximum hypothetical accident.

The staff reviewed the GSTR hazards summary report (safety analysis report) and noted that the maximum fuel temperatures given for the loss-of-pool-water accident (780°C) and the large reactivity addition accident (804°C) were greater than the safety limit for aluminum-clad fuel. The analyses used very conservative assumptions and the temperatures given were the maximum for the core and would not be experienced in the F and G rings of the reactor where the aluminum-clad fuel would be located. The licensee was asked to address these issues in requests for additional information from the NRC staff.

For the loss of coolant, the licensee referred to the "Safety Analysis Report for the Torry Pines TRIGA Mark III Reactor" (GA-9064). In that report, GA determined that if coolant is lost several minutes after a long period of operation at 2 MW, the maximum temperature the fuel would reach is 520°C, which is below the safety limit for aluminum-clad fuel. GA's methodology and the licensee's methodology in its hazards summary report are similar.

The GSTR loss-of-pool-water analysis assumes that complete water loss in the core occurs immediately after the reactor has been shut down from infinite operation at full power. This assumption results in a very conservative level of decay heat generation in the fuel and an elevated initial fuel temperature. The maximum fuel temperature reached during the event depends on the decay heat produced in the fuel and the temperature of the fuel when the coolant is lost. Even though the operating power of the reactor in the GA analysis is twice that of the GSTR, the assumption that the reactor was shut down for several minutes before the loss of coolant occurred in the GA reactor resulted in a reduction in maximum fuel temperature from 780°C to 520°C.

The GSTR reactor is a Mark I TRIGA type with an in-ground pool. There are no piping or experimental facilities that can drain the primary coolant to the core level. The reactor is at the bottom of a replacement tank that sits in the original reactor tank. The tank is about 25 feet (7.6 meters) below grade. The original tank is surrounded by a concrete shield. The shield is surrounded by earth. The two tanks and the shield would have to fail to allow a leakage path for the primary coolant. The rate of coolant loss would be limited to the ability of the earth surrounding the reactor to absorb water. The license estimates there are 6770 gallons (25,600 liters) of coolant above the core. The NRC staff concludes that it would take a significant amount of time for the earth surrounding the reactor pool to absorb this water. The licensee assumed a leakage rate of 350 gpm (1325 lpm), the rating of the primary coolant pump. The primary cooling system is the designed so that the primary pump suction line only reaches 3 feet (1 meter) below the top of the tank. Given this leakage rate, it would still take about 19 minutes to empty the pool to the top of the reactor.

The amount of decay heat produced by the reactor core varies with the time since the reactor was shut down. At 5 minutes after shutdown, the decay heat level is about 2.39% of full power; at 19 minutes, the decay heat level is about 1.76% of full power. The GA analysis maximum fuel temperature of 530°C is based on a core decay heat level of about 48 kW. The GSTR decay heat level 19 minutes after shutdown is about 18 kW. Therefore, the maximum temperature in the GSTR core will be substantially less than 530°C. In addition, the aluminum-clad fuel elements in the F and G rings will be at a lower temperature than the maximum temperature fuel element due to their location in the core. The licensee calculates that the aluminum-clad fuel temperature will be less than 200°C.

The licensee proposes a new TS to help ensure that the reactor operator will be aware of a loss-of-coolant event and will take steps to shut down the reactor, reducing decay heat levels in the core. The licensee proposes to install of an audible and visual water level alarm that will alert the reactor operator if the reactor pool level is dropping. A surveillance requirement for monthly testing of the alarm is also proposed. The proposed TS C.3 reads as follows:

3. The control console shall have an audible and visual water level alarm that will actuate when the reactor tank water level is between 12 and 24 inches below the top lip of the tank. This water level alarm shall be functionally tested monthly, not to exceed 45 days between tests. This item is not applicable if the reactor is completely defueled and the pool level is below the water treatment system intake.

The NRC staff concludes that given the below-ground-level design of the GSTR, the assumption of a 19-minute time to empty the reactor pool is conservative and acceptable. The NRC staff also concludes that based on the analysis performed by the licensee and GA and the addition of a low-pool-level alarm, there is reasonable assurance that the maximum temperature of aluminum-clad fuel elements in the F and G rings will not exceed the safety limit temperature of 530°C. Therefore, the results of the analysis of the loss-of-coolant event are acceptable to the NRC staff.

The GSTR hazards analysis report discusses a reactivity addition event where 3.00$ of reactivity is added to the reactor operating at a steady-state power level of 1.4 MW. The peak fuel temperature for the event is 804°C, which is above the aluminum-clad fuel safety limit. In a request for additional information, the staff asked the licensee to address this issue.

The licensee responded that fuel temperatures in the F and G rings, the only core locations where the aluminum-clad fuel elements are allowed, are significantly lower than at the peak fuel temperature location in the core. The hazards analysis report concludes that the average temperature at the conclusion of the reactivity addition is 470°C, within the aluminum-clad fuel temperature safety limit of 530°C. Based on measured fuel temperatures in the core and the relationship between the B ring and F and G ring fuel temperatures, the licensee determined that the peak temperature in aluminum clad fuel elements in the F and G rings of the reactor core would be 473°C for the G ring and 402°C for the F ring.

Based on the information in the hazards analysis report and the licensee's analyses, the staff concludes that there is reasonable assurance that the fuel temperature in aluminum-clad fuel elements in the F and G rings of the reactor will not exceed the safety limit temperature during a reactivity addition event. Therefore, the results of the reactivity addition event are acceptable to the NRC staff.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in inspection and surveillance requirements. The staff has determined that this amendment involves no significant hazards consideration, no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and 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.

5.0 CONCLUSION

The staff has concluded, on the basis of the considerations discussed above, that (1) the amendment does not involve a significant hazards consideration because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, create the possibility of a new kind of accident or a different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety; (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.

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Date: January 30, 2006