United States Department of the Interior

U. S. GEOLOGICAL SURVEY
Box 25046 M.S. 911
Denver Federal Center
Denver, Colorado 80225

IN REPLY REFER TO:

February 8, 2005

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555

Gentlemen:

The U.S. Geological Survey is herein responding to your request for additional information (TAC No. MC5120) dated December 7, 2004. This concerns the USGS amendment request to its research reactor facility license (No. R-113, Docket 50-274) to allow the use of aluminum-clad TRIGA fuel in the core.

Correspondence concerning this response should be directed to Tim DeBey, Reactor Supervisor.

Sincerely,

Warren Day

Warren Day
Reactor Administrator

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 2/8/05 Warren Day

Copy to: Alexander Adams, NRC

A020

REQUEST FOR ADDITIONAL INFORMATION UNITED STATES GEOLOGICAL SURVEY
DOCKET NO. 50-264

1. The regulations in 10 CFR 50.30(b) require requests for license amendments to be made under oath or affirmation. Please resubmit your entire application (including original cover letter) with a new cover letter under oath or affirmation.

   Response: The resubmission under oath or affirmation was done prior to this response, on 12/03/04.

2. Technical specification (TS) D.1. contains a limit on calculated steady state power level of any fuel element of 22 kW. The purpose of this limit was to ensure that the departure from nucleate boiling ratio at the hottest point in the reactor fuel would not fall below two, an acceptable safe condition. Your proposed changes to the TS apply this limit only to stainless steel fuel elements. Why is this limit not applicable to all fuel, including aluminum-clad fuel placed in the core? If this limit is not applicable to the aluminum-clad fuel, what power limit per fuel element is applicable for the aluminum clad fuel? Please revise your proposed TS appropriately.

   Response: The proposed restriction on the aluminum fuel placement to the F and G rings prevents those elements from ever approaching the 22 kW limit for the stainless steel fuel, hence the application of this limit to the aluminum fuel is unnecessary. For the USGS core, a fuel element in the F-ring would produce 56% (or 12.3 kW maximum) of the power of a B-ring element and a fuel element in the G-ring would produce 47% (or 10.3 kW maximum) of the power of a B-ring element. (Also see request #9 and its response below for our proposed TS revision.)

3. Proposed TS D.3. discusses limiting measured fuel temperature in an aluminum-clad element to 530°C. This 530°C temperature limit for aluminum-clad fuel is the safety limit based on preventing the phase change in low hydride TRIGA fuel. The safety limit temperature is normally not used as the measured temperature limit because uncertainty in making the measurement results in needing to use a lower value. It appears that you do not possess the ability to measure the temperature in aluminum clad fuel elements. Even if you did measure the fuel temperature, fuel temperature is not an input into the reactor safety system that would result in a scram.

   The reactor safety system protects the integrity of the fuel cladding by initiating a scram if Safety Channel 1 or 2 exceeds 110 percent of full power. There is also a TS requirement that the reactor not be operated in a manner which would cause the measured fuel temperature to exceed 800 °C. Exceeding this limit does not result in any automatic action. Explain how the high power scram set point (1.1 MW) will prevent aluminum clad fuel elements from exceeding a temperature of 530°C in the F and G rings. Does limiting the temperature of the thermocouple fuel element in the B or C ring to 800°C provide additional protection of the aluminum clad fuel safety limit? Please revise your proposed TS appropriately.

Response: The proposed restriction on the aluminum fuel placement to the F and G rings prevents those elements from ever approaching 530°C. This is because of the significant reduction in power produced per element in the F and G rings. The highest power production fuel element (hottest element) is calculated to be in the B-ring, near the water hole of the central thimble irradiation position. Data from instrumented fuel elements in the GSTR show that when the reactor is operated at full power (1 MW), the B-ring fuel elements are operating at a centerline temperature of 344°C, the F ring is at 202°C and the G ring is at 172°C. Conservatively assuming that the heat transferred from the fuel elements is by free convection (i.e., ignoring radiation heat transfer), then the rate of heat transferred (i.e., power transferred) to the tank water is directly proportional to the difference in temperature (ΔT) between the fuel element's temperature and the cooling water's temperature. ($Q = hA\ (\Delta T)$ where Q is the heat transfer rate, h is the heat transfer coefficient, A is the surface area, and ΔT is the temperature difference. In this case, h and A are constants, giving $Q \propto \Delta T$.) Using this proportionality and the maximum expected cooling water temperature of 50°C, the following table can be generated.

Description of GSTR operation.	Cooling medium temp °C	B-ring peak temp °C	B-ring ΔT (°C)	F-ring peak temp C	F-ring ΔT (°C)	G-ring peak temp C	G-ring ΔT (°C)
Normal 1 MW ops	21	344	323	202	181	172	151
B-ring at 800C	50	800	750	470	420	401	351

These data show that the ratio of power generated in a F-ring element to the power generated in a B-ring element is 0.56. Likewise, the ration of power generated in a G-ring element to the power generated in a B-ring element is 0.47. Thus, limiting the B-ring fuel elements to 800°C would limit the aluminum-clad elements to 470°C in the F-ring and 401°C in the G-ring. These values are far above the fuel temperatures for 110% power operation. The calculated F and G-ring temperatures are also well below the 530°C safety limit for the aluminum-clad fuel. We propose to further restrict the operation of the GSTR to not cause the calculated aluminum-clad fuel temperature to exceed a value of 500°C (see below).

Proposed technical specification change:

Section D Reactor Core

Current wording:

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.

Proposed wording:

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 in a stainless steel clad element or the calculated fuel temperature to exceed 500°C in an aluminum clad element.*

4. Your application presents data on fuel temperature measurements in reactors. For measurements from the USGS reactor, where in the core was the thermocouple fuel element located?

   Response: For the GSTR fuel temperature data presented in the initial request and this response, instrumented fuel elements were located in the B, C, F, and G-rings.

5. Your application states that the maximum fuel temperature of stainless steel clad elements in the G ring of the reactor is approximately 180 °C. Please explain how this value was determined. What would the value be in the F ring?

   Response: The G-ring fuel temperature of approximately 180°C was estimated from empirical measurements in the GSTR at full power operation. Recent measurements of the F-ring and G-ring fuel temperatures using an instrumented element gave values of 202°C and 172°C at full power, respectively, with the water temperature at 21°C.

6. Your application quotes data from GA showing that 3.08$ pulses did not produce fuel temperatures over 500 °C. Please provide a reference for this data.

   Response: The GA pulsing data is from page 30 of GA publication GA 2025, "Hazards Report for the 250 KW TRIGA Mk II Reactor", published August 1961 (attached). It states that there were over 1000 pulses with 2.25% δk/k (~$3.08) insertions with a maximum measured fuel temperature less than 500°C. These pulses were performed with aluminum-clad fuel elements. Page 9 of this same reference indicates that the 250 KW facility would have between 64 and 85 fuel elements. Given that the GSTR currently has 125 fuel elements in its core and this response proposes that a core limit of at least 100 fuel elements be instituted (see item 9) for all operations above 100 kW, it follows that the maximum fuel temperatures from $3.00 pulses in the GSTR will be significantly lower than those seen in the GA testing.

7. Will transverse bend and longitudinal elongation measurements be made on the aluminum clad fuel received from VA before the fuel is placed into service? If so, what were the results of the measurements? If not, how will you ensure that the

fuel meets the proposed TS requirements?

Response: Transverse bend and longitudinal elongation measurements were made on the aluminum-clad fuel received from the VA in August of 2003. All of the elements passed the examinations at the current criteria for stainless steel elements (i.e., 1/16" transverse bend and 1/10" elongation), which are equal or more restrictive than the proposed criteria for aluminum-clad elements.

8. You have proposed measuring transverse bend and longitudinal elongation every 50 months. Because this is a new fuel type please consider a requirement to measure approximately 20 percent of the aluminum-clad fuel annually. In addition, if any of the aluminum-clad fuel exceeded the measurement limits, all aluminum-clad fuel in the core would be checked. This would allow potential generic problems to be detected early, create a pool of data on aluminum clad fuel performance and result in a 50-month inspection schedule after the fourth year. The inspection schedule based on pulses would remain unchanged.

Response: The proposed measurement schedule (in agreement with existing technical specifications) is a 60-month schedule, not 50 months. We herein propose that, during the first 5 years of aluminum-clad fuel, to perform annual fuel transverse bend and longitudinal elongation measurements 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 would revert back to the pre-existing 60-month schedule.

<u>Proposed technical specification change:</u>

Section D <u>Reactor Core</u>

<u>Current wording:</u>

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.

__*Proposed wording:*__

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.

9. The accident analysis in the SAR for the USGS reactor discusses loss of reactor pool water. The analysis concludes that the maximum temperature reached in the core is 780 °C. While within the safety limit for stainless steel clad fuel, this temperature is greater than the safety limit for aluminum clad fuel. However, this analysis is for the hottest fuel element in the core. Please provide an analysis of the maximum temperature reached by an aluminum clad fuel element in the F and G ring following a loss of reactor pool water.

Response: The analysis in the SAR for the GSTR is excessively conservative. A less conservative, applicable analysis for a loss of pool water accident is given in the General Atomics publication, GA 9064 "Safety Analysis for Torrey Pines TRIGA Mk III", January 1970, (section 8.3 attached) as quoted below:

> *<u>Section 8.3 Loss of Reactor Pool Water</u>*
> *8.3.1 <u>Heat Removal</u>: "These calculations indicate that if the water in the pool is lost several minutes after a long period of operation at 2000 kW, the maximum temperature of the fuel, and consequently the stainless steel cladding, would be less than 520°C ... It was assumed that at the time the water was lost, the fuel temperature was 25°C. It was also assumed that the reactor had been operating for an infinite time at 2000 kW with 100 elements in the core."*
> *"An experiment was performed at Gulf General Atomic in which fuel element clad and coolant air temperatures were measured in a simulated loss of coolant situation. In this experiment seven electrically-heated dummy TRIGA fuel*

elements were placed in a grid plate. The removal of heat from these dummy elements was accomplished through the natural convection of air up between the fuel elements and out through the top grid plate. Correlation between measured temperatures in this experiment and calculations using the model described above was very good."

(Note: Data collection at the GSTR shows that fuel temperature equilibration with pool water occurs within approximately 5 minutes after a reactor shutdown.)

Unlike the Mk III TRIGA design, the GSTR has an "in-ground" tank that is surrounded by several feet of reinforced concrete and then earth. There are no penetrations into or through the concrete and the existing ground water is approximately the same level as the bottom of the tank. The heat exchanger and primary purification system components are all located higher than the reactor tank, eliminating any siphoning potential. These features make loss of the reactor pool water below the fuel level an incredible accident. Despite that, we will conservatively assume that the tank water can leak out at a maximum flow rate equal to the flow rate of the primary cooling pump (350 gpm). The primary coolant pump is actually not capable of emptying the tank for two reasons: 1) the suction pipe of the pump only reaches down about 3 feet below the top of the tank and 2) the pump cannot maintain sufficient net positive suction head to pull water up from over 20 ft below its location.

It will be assumed in the GSTR analysis that a reactor shutdown will occur when the reactor tank water level is no lower than 24" below the top lip of the tank. This will be enforced by installing an audible and visual alarm on the control console that will actuate at a level no lower than 24" below the top lip of the tank. The GSTR operating procedure will be revised to require that reactor operators scram the reactor upon receipt of this water level alarm. Functional testing of the alarm will also be required on a nominal monthly frequency.

a. There is a distance of approximately 22 ft 6 in from the top of the fuel in the GSTR core to the top of the reactor tank. The normal water level ranges from 6" to 15" below the top of the tank. If a reactor shutdown occurs when the water is 24" below the top of the reactor tank, that would leave a column of water that is 20 ft 6 in high and 7 ft 6 in diameter above the reactor fuel. This represents 6770 gallons of available cooling water. If we conservatively assume a leakage rate equal to the 350 gpm flow rate of the primary cooling pump, it would take over 19 minutes before the reactor core would lose its water cooling. This is about 4 times longer than assumed in the referenced GA analysis. ANS standard 5.1-1994, <u>Decay Heat Power in Light Water Reactors</u>, tabulates decay heat power for thermal fission of ^{235}U in Table 5. Assuming 210 MeV is released per thermal fission of U-235, linear interpolation of the data show that the decay heat power for 5 minutes and 19 minutes after shutdown are:

1. *5 minutes after shutdown ~ 2.39% of original power*
2. *19 minutes after shutdown ~1.76% of original power*

For the GA analysis, the initial decay heat would be ~ 2.39% of 2 MW, or ~47.8 kW. For the GSTR case, the initial decay heat would be ~ 1.76% of 1 MW, or ~17.6 kW. This shows that the GSTR core is a factor of 2.72 lower in decay heat power production when air cooling is initiated, relative to the GA Mk III core. Assuming a 100 element core and relative power factors in the F and G rings of 0.56 and 0.47, respectively, we calculate power produced per fuel element in the F-ring of 98.6 watts and the G-ring of 82.7 watts *at the time when air cooling begins. For the current condition of 125 fuel elements in the GSTR core, the power produced per fuel element would be 78.8 watts in the F-ring and 66.2 watts in the G-ring at the time when air cooling begins.*

b. The 0.03" thick aluminum cladding of the subject GSTR fuel has a thermal conductivity of ~ 170 W/mK while the 0.02" thick stainless steel cladding of the GA analysis has a thermal conductivity of ~ 16 W/mK. This gives improved heat transfer through the aluminum cladding because the net effect is a factor of 7 less resistance to heat flow through the aluminum cladding, relative to the stainless steel cladding. This conservatism in the GSTR case is noted here but is not accounted for in the calculations.

c. The aluminum fuel elements will not be located in the B-ring, but will be located in the F and G rings, reducing their decay heat production by factors of 0.56 and 0.47, respectively, below the hot element that was analyzed in the reference GA analysis.

d. The net result is that the GA analysis, on its own, is sufficient to show that the peak fuel temperature from a complete loss of coolant accident would not exceed the 530 C safety limit for aluminum-clad fuel in any of the core rings. However, for the GSTR there are the added safety margins that result from a factor of 2.72 lower decay heat at the beginning of air cooling (see a. above) and factors of 1.79 and 2.13 lower decay heat, for the F and G rings, respectively, because the aluminum-clad elements are on the core periphery. The total reduction factors of 4.87 and 5.80, respectively for the F and G ring elements of the GSTR aluminum-clad fuel results in significantly lower peak fuel temperatures. The rate of heat transferred (i.e., power transferred) to the surrounding air is directly proportional to the difference in temperature (ΔT) between the fuel element's temperature and the air temperature. ($Q = hA\ (\Delta T)$ where Q is the heat transfer rate, h is the heat transfer coefficient, A is the surface area, and ΔT is the temperature difference. In this case, h and A are conservatively assumed to be the same for the aluminum-clad and stainless steel-clad fuels, giving $Q \propto \Delta T$.) Using this proportionality and a maximum expected air temperature of 100°C, the following table can be generated:

Description	Cooling medium temp °C	B-ring peak temp °C	B-ring ΔT (°C)	F-ring peak temp C	F-ring ΔT (°C)	G-ring peak temp C	G-ring ΔT (°C)
GA Mk III air cooling from 2 MW ops	100	520	420	---	---	---	---
GSTR air cooling from 1 MW ops	100	254	154	186	86	172	72

The resulting peak temperatures in the GSTR aluminum-clad fuel (F and G-rings) are well below the 530 °C safety limit.

Proposed technical specification changes:

Section D. Reactor Core

__*Current wording:*__

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 ± 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.

__*Proposed wording:*__

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.*

Section C. Reactor Pool and Bridge

__*Proposed additional specification wording:*__

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.*

10. The accident analysis in the SAR for the USGS reactor discusses a reactivity accident where 3.00$ of reactivity is added to the reactor operating at a steady state power level of 1.4 MW. The results show a peak fuel temperature of 804°C. While within the safety limit for stainless steel clad fuel, this temperature is greater than the safety limit for aluminum clad fuel. However, this analysis is for the hottest fuel element in the core. Please provide an analysis of the maximum temperature reached by an aluminum clad fuel element in the F and G ring following a reactivity addition at power.

    Response: The proposed restriction on the aluminum fuel placement to the F and G rings prevents those elements from ever approaching 530°C. This is because of the significant reduction in power produced per element in the F and G rings. Empirical data from instrumented fuel elements in the GSTR show that when the B-ring fuel elements are operating at a centerline temperature of 344°C, the F ring is at 202°C and the G ring is at 172°C. Conservatively assuming that the heat transferred from the fuel elements is by free convection (i.e., ignoring radiation heat transfer), then the rate of heat transferred (i.e., power transferred) to the tank water is directly proportional to the difference in temperature (ΔT) between the fuel element's temperature and the cooling water's *temperature. Using this proportionality and the maximum expected cooling water temperature of 50°C, the following table can be generated.*

Description of GSTR operation.	*Cooling medium temp °C*	*B-ring peak temp °C*	*B-ring ΔT (°C)*	*F-ring peak temp C*	*F-ring ΔT (°C)*	*G-ring peak temp C*	*G-ring ΔT (°C)*
Normal 1 MW ops	*21*	*344*	*323*	*202*	*181*	*172*	*151*
B-ring at 804C	*50*	*804*	*754*	*473*	*423*	*402*	*352*

These data show that limiting the B-ring fuel elements to 804°C would limit the aluminum-clad elements to 473°C in the F-ring and 402°C in the G-ring.

11. Controlling pH in addition to conductivity is important to prevent corrosion in systems with aluminum at elevated temperatures such as found on the surface of operating fuel elements. Information is available about corrosion protection of aluminum systems (see, for example, "Handbook of Power Plant Chemistry, "Hans-Gunter Heitmann, CRC Press, Boca Raton, 1993 or "Criteria for Corrosion Protection of Aluminum-Clad Spent Nuclear Fuel in Interim Wet Storage," James P. Howell, Westinghouse Savannah River Company, WSRC-MS-99-QO601). Controlling pH in the 5.5 to 7.5 range will minimize uniform corrosion. Please discuss the need to control pH in addition to conductivity and propose changes to TS C.2 as needed.

Response: Chemically, the protective film is a hydrated form of aluminum oxide. The corrosion resistance of aluminum depends upon this protective oxide film that is stable in aqueous media when the pH is between about 4.0 and 8.5. The oxide film is naturally self-renewing and accidental abrasion or other mechanical damage of the surface film is rapidly repaired. The conditions that promote corrosion of aluminum and its alloys, therefore, must be those that continuously abrade the film mechanically or promote conditions that locally degrade the protective oxide film and minimize the availability of oxygen to rebuild it.

The reference, "Criteria for Corrosion Protection of Aluminum-Clad Spent Nuclear Fuel in Interim Wet Storage," James P. Howell, Westinghouse Savannah River Company, WSRC-MS-99-QO601 is quoted below:

> *"Effect of pH*
> *Aluminum is passivated and protected by its oxide film in the pH range of about 4-8.5. The limits vary somewhat with temperature and the specific form of oxide present, and with the presence of substances that can form soluble complexes or insoluble salts with aluminum. The oxide coating is soluble at pH values below 4 and above 8.5. General corrosion in distilled water at 60 °C has been shown minimum at pH 4 rising slightly in the passive range and faster between pH 9 and 10. For pitting corrosion, which is the predominant mechanism for aluminum in water, the pitting potential in chloride solutions has been found to be relatively independent in the range of 4-9."*

More directly applicable data is given in DOE Handbook 1015/1-93, "Department of Energy Fundamentals Handbook". It states on page 17 under Module 2, Corrosion of Aluminum: "For those reactor plants in which aluminum is used for cladding and other structural components, pH is controlled in an acidic condition because of the corrosion properties of aluminum. Plant pH has a marked effect on the rate of chemical reaction between the coolant water and aluminum. In the area of the cladding, the corrosion reduces the thickness and forms an oxide film that is a thermal barrier. Extensive tests carried out in support of DOE test reactors have revealed that minimum aluminum corrosion results with a pH of 5.0 at normal operating temperatures. Additionally, studies have shown that the aluminum corrosion products also exhibit a minimum solubility at a pH near 5.5 at 25(C. The aluminum corrosion products tend to reduce the substrate (base) aluminum metal corrosion rates. Because it is desirable to maintain dissolved aluminum in the reactor coolant at the lowest practicable level, it is desirable to maintain the system pH level in the range of minimum oxide solubility. Figure 9 shows the effect of pH on aluminum oxide solubilities for various forms of oxide, and the effect of pH on corrosion rates. It should be noted that the values at which minimum corrosion and solubility are found shift to a lower pH as the temperature is increased. For example, at 300 C, the value for minimum aluminum corrosion is near pH 3.0. Therefore, the optimum pH for operation is determined by the operating temperature."

The DOE data are from operating test reactors, giving more direct applicability to the GSTR.

Based on the inconsistent reference data presented above, we would put more reliance on the data from DOE test reactors that show pH levels of 5.0 and lower will minimize aluminum metal corrosion rates, especially at high temperatures. Therefore we propose to set an acceptable pH range of 4.5 to 7.5 for the USGS reactor water. Limited historical data from the USGS facility show that the pH has been in the range of 5 to 7 over the life of the facility. We propose to monitor the primary water pH by performing quarterly pH tests, at the same time the primary water is being sampled for radioactivity levels.

Proposed new technical specification:

C. Reactor Pool and Bridge

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.

In addition to the above technical items, the GSTR Technical Specifications were reformatted and repaginated to accommodate the proposed changes and give the document a consistent appearance. A copy of the GSTR Technical Specifications with the proposed changes is attached to this document.

APPENDIX A

TECHNICAL SPECIFICATIONS FOR THE

U.S. GEOLOGICAL SURVEY TRIGA REACTOR

DOCKET NO. 50-274

The dimensions, measurements, and other numerical values given in these specifications may differ from measured values owing to normal construction and manufacturing tolerances, or normal accuracy of instrumentation.

A. Definitions

1. Shutdown

The reactor, with fixed experiments in place, shall be considered to be shutdown (not in operation) whenever all of the following conditions have been met: a) the console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator (or stored in a locked storage area); b) sufficient control rods are inserted so as to assure the reactor is subcritical by a margin greater than 0.7% delta k/k cold, without xenon; c) no work is in progress involving fuel handling or refueling operations or maintenance of the control mechanisms.

2. Steady State Mode (SS)

Steady state mode shall mean operation of the reactor at power levels not to exceed 1 megawatt utilizing the scrams in Table I and the interlocks in Table II.

3. <u>Pulse Mode</u>

Pulse mode shall mean operation requiring the use of the scrams in Table I and the interlocks in Table II to assure that no more than one rod is pneumatically withdrawn to produce power pulses.

4. <u>Square Wave Mode (SW)</u>

Square wave mode shall mean operation of the reactor with the mode selector switch in the square-wave position requiring use of the scrams in Table I and the interlocks in Table II.

5. <u>Operable</u>

A system or component shall be considered operable when it is capable of performing its intended functions.

6. <u>Experiment</u>

Experiment shall mean: (a) any apparatus, device, or material installed in the core or experimental facilities (except for underwater lights, fuel element storage racks and the like) which is not a normal part of these facilities or (b) any operation to measure reactor parameters or characteristics.

7. <u>Experimental Facilities</u>

Experimental facilities shall mean the rotary specimen rack, vertical tubes, pneumatic transfer system, central thimble, and in-pool irradiation facilities.

8. Reactor Safety Systems

Reactor safety systems shall mean those systems, including their associated input circuits, which are designed to initiate a reactor scram.

9. Standard Thermocouple Fuel Element

A standard thermocouple fuel element shall contain thermocouples imbedded in the fuel halfway to the vertical centerline at the midplane of the fuel section and one inch above and below the midplane.

B. Reactor Building

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume in the reactor room shall be 3.1×10^8 cubic centimeters.
2. All air or other gas exhausted from the reactor room and from associated experimental facilities during reactor operation shall be released to the environment at a minimum of 21 feet above ground level.
3. The concentration of argon 41 in the reactor building stack effluent air shall be limited to a maximum of 4.8×10^{-6} uCi/ml averaged over a year.
4. The stack effluent air shall be analyzed quarterly to determine the isotopic composition of the radionuclides emitted. The limit of B.3 above shall apply only to argon 41; limits on concentrations for other radionuclides shall be as specified in 10 CFR Part 20.

C. Reactor Pool and Bridge

1. The reactor shall not be operated if the pool water level is less than 16 feet above the top grid plate. The bulk pool temperature shall be monitored while the reactor is in operation and the reactor shall be shut down if the temperature exceeds 60°C. The reactor core shall be cooled by natural convective water flow.

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. <u>Reactor Core</u>

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 800°C in a stainless steel clad element or the calculated fuel temperature to exceed 500°C in an aluminum clad element.

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.

2. The control rods shall be visually inspected at least once every two years. If indication of significant distortion or deterioration is found, the rod(s) will be replaced.

3. Only one pulsing control rod may be used in the core. The poison section of this rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The pulse rod shall be designed to release and fall upon initiation of a scram signal. The maximum reactivity worth of the rod fully inserted by the drive in relation to fully withdrawn shall be equal to or less than 2.9% delta k/k.

4. A pulse may be initiated only when the reactor is at power less than 1 kW. Pulsed reactivity insertion shall not exceed 2.1% delta k/k.

5. The minimum shutdown margin (with fixed experiments in place) provided by operable control rods (including the pulse rod) in the cold clean condition, with the most reactivity of the operable control rods fully withdrawn, shall be 0.4% delta k/k.

6. The maximum rate of reactivity insertion associated with movement of a standard rod shall be no greater than 0.2% delta k/k/sec.

7. The type and minimum number of safety systems which shall be operable for reactor operation are shown in Table I.

8. The type and minimum number of interlocks which shall be operable for reactor operation are shown in Table II.

9. The reactor instrumentation channels and safety systems for the intended modes of operation as listed in Table I shall be verified to be operable at least once each day the reactor is operated unless the operation extends continuously beyond one day, in which case the operability need only be verified prior to beginning the extended operation.

10. A licensed reactor operator shall be present during maintenance of the reactor control and safety systems.

11. Following maintenance or modification of the control or safety systems, the associated system shall be verified to be operable before the reactor is placed in operation.

12. The conditions listed below shall be verified at least once semi-annually, with the exception that if the reactor is operating continuously, the conditions shall be verified after the first shutdown that occurs more than six months after the previous tests. Those items marked with an * are not applicable if the reactor is completely defueled, but they must be verified upon startup if more than six months have passed after the previous tests.

a. *All reactor interlocks are operable.

b. *Control element drop times are less than one second (two seconds for pulse rod). If drop time is found to be greater than this, the rod shall not be considered operable.

c. *Power level safety circuits are operable. The circuits will be tested by the introduction of an electrical signal into the circuit at a point between the detector and the control system.

d. Ventilation system interlocks are operable.

e. *The safety channels indicate the actual power level as determined by a thermal power measurement.

13. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed. Semi-annually, at intervals not to exceed eight months, the transient (pulse) rod drive cylinder and the associated air supply system shall be inspected, cleaned and lubricated as necessary.

F. Radiation Monitoring

1. The radiation levels within the reactor laboratory shall be monitored by at least one area radiation monitor during reactor operation or when work is done on or around the reactor core or experimental facilities. The monitor shall have a readout and provide a signal which actuates an audible alarm. During short periods of repair to this monitor, reactor operations may continue while a portable gamma-sensitive ion chamber is utilized as a temporary substitute.
2. A continuous air monitor with readout and audible alarm shall be operable in the reactor room when the reactor is operating.
3. The alarm set points for the above radiation monitoring instrumentation shall be verified at least once a week. This instrumentation shall be calibrated at least once a year.

G. Fuel Storage

1. All fuel elements or fueled devices shall be rigidly supported during storage in a safe geometry (k_{eff} less than 0.8 under all conditions of moderation).
2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling such that the fuel element or fueled device temperature will not exceed design values.

H. Administrative Requirements

1. The facility shall be under the direct control of the Reactor Supervisor. He shall be responsible to the Reactor Administrator for safe operation and maintenance of the reactor and its associated equipment. He or his appointee shall review and approve all experiments and experimental procedures prior to their use in the reactor. He shall enforce rules for the protection of personnel against radiation.

2. A Reactor Operations Committee shall review and approve safety standards associated with the operation and use of the facility. Its jurisdiction shall include all nuclear operations in the facility. The Committee shall meet to monitor reactor operations at least semi-annually.

The Reactor Operations Committee shall be composed of at least four members, appointed by the Director, U.S. Geological Survey, and who shall be knowledgeable in field relating to nuclear safety. The Reactor Supervisor and a qualified health physicist shall be members of the Committee. The Committee shall be responsible for determining whether a proposed change, test, or experiment would constitute a change in technical specifications or an unreviewed safety question as defined in 10 CFR Part 50. The Committee shall establish written procedures concerning its activities, quorums, review of experiments and procedures, and other aspects as appropriate.

3. Written instructions shall be in effect and followed for:

a. Testing and calibration of reactor operating instrumentation and control systems, control rod drives, area radiation monitors and air particulate monitors.

b. Reactor startup, routine operation and reactor shutdown.

c. Emergency and abnormal conditions, including evacuation, reentry and recovery.

d. Fuel loading or unloading.

e. Control rod removal and replacement.

f. Maintenance operations which may affect reactor safety.

4. Any additions, modifications, or maintenance to the core and its associated support structure, the pool structure, and rod drive mechanisms, or the reactor safety system, shall be made and tested in accordance with the specifications to which the systems or components were originally designed and fabricated, or to specifications approved by the Reactor Operations Committee as suitable and not involving an unreviewed safety question. The reactor shall not be placed in operation until the affected system has been verified to be operable.
5. The reactor facility emergency plan, emergency procedures and physical security plan shall be audited by the Reactor Operations Committee biennially, with the interval not to exceed 30 months.

I. <u>Experiments</u>

1. Prior to performing any new reactor experiment, the proposed experiment shall be evaluated by a person or persons appointed by the Reactor Administrator to be responsible for reactor safety. He shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including, where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. He shall determine whether, in his judgement, the experiment by virtue of its nature or design does not constitute a significant threat to the integrity of the core or to the safety of personnel. Following a favorable evaluation and prior to conducting an experiment, he shall sign an authorization form containing the basis for the favorable evaluation.

2. A favorable evaluation of an experiment shall conclude that failure of the experiment will not lead to a direct failure of a fuel element or of other experiments.

3. No new experiment shall be performed until the proposed experimental procedures for that experiment or type of experiment have been reviewed and approved by the Operations Committee.

4. The following limitations on reactivity shall apply to all experiments:

   a. The reactivity worth of any individual in-core experiment shall not exceed $3.00.

   b. The total, absolute, reactivity worth of in-core experiments shall not exceed $5.00. This includes the potential reactivity which might result from experimental malfunction, experiment flooding or voiding, and removal or insertion of experiments.

   c. Experiments having reactivity worths greater than $1.00 shall be securely located or fastened to prevent inadvertent movement during reactor operation.

5. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, or liquid fissionable materials shall be doubly encapsulated.

6. Explosive materials such as (but not limited to) gun powder, dynamite, TNT, nitro-glycerine, or PETN in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities without out-of-core tests which shall indicate that with the containment provided no damage to the reactor or its components shall occur upon detonation of the explosive. Explosive materials in quantities less than 25 milligrams may be irradiated without out-of-core

tests provided that the pressure produced in the experiment container upon detonation of the explosive shall be shown to be less than the design pressure of the container.

7. Experiment materials, except fuel materials, which could off-gas, sublime, volatize or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor or (c) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limits of Appendix B of 10 CFR Part 20.

8. In evaluating experiments, the following assumptions shall be used:

   a. If the effluent from an experiment facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, the assumption shall be used that at least 10% of the aerosols produced can escape.

   b. For materials whose boiling point is above 130°F and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, the assumption shall be used that at least 10% of these vapors can escape.

9. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum strontium-90 inventory is no greater than 5 millicuries.

10. If a container fails and releases material which could damage the reactor fuel or structure by corrosion or other means, physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Operations Committee and determined to be satisfactory before operation of the reactor is resumed.

TABLE I

MINIMUM REACTOR SAFETY SYSTEMS

Originating Channel	Setpoint	Mode in which effective SS	Pulse	SW
1. Safety Channel 1	110% of full power	X		X
2. Safety Channel 2	110% of full power	X		X
3. Scram button	Manual push	X	X	X
4. Preset timer	Less than or equal to 15 seconds		X	
5. CSC watchdog timer	Loss of refresh signal	X	X	X
6. DAC watchdog timer	Loss of refresh signal	X	X	X

TABLE II

MINIMUM INTERLOCKS

Action Prevented	Mode in which effective SS	Pulse	SW
1. Control rod withdrawal with neutron level less than 10^{-7}% power on the digital power channel.	X		
2. Simultaneous manual withdrawal of two control rods, including the pulse rod.	X		
3. Simultaneous manual withdrawal of two control rods excluding the pulse rod.			X
4. Initiation of pulse above 1 kW.		X	
5. Application of air pressure to pulse rod drive mechanism unless cylinder is fully inserted.	X		
6. Withdrawal of any control rod except pulse rod.		X	

GENERAL ATOMIC
DIVISION OF
GENERAL DYNAMICS
JOHN JAY HOPKINS LABORATORY FOR PURE AND APPLIED SCIENCE
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GA-2025

HAZARDS REPORT
FOR THE
250 KW TRIGA MARK II REACTOR

August, 1961

1.2 TECHNICAL DATA AND NUCLEAR CHARACTERISTICS SUMMARY

Reactor core

Composition	Cylindrical lattice of fuel-moderator elements and control rods, 35% H_2O by volume
Active lattice dimensions	14 inches (35.6 cm) in diameter and 14 inches high
Number of fuel-moderator elements	~64*
Number of graphite dummy elements	~21*
Composition of fuel-moderator elements	Central slug of fuel-moderator material, 2 burnable poison wafers and two 4-inch (10.16 cm) graphite end reflectors.
Fuel-moderator element cladding	0.030 inch (0.76 mm) aluminum
Fuel-moderator material	Homogeneous alloy, 8 wt-% uranium, 91 wt-% zirconium and 1 wt-% hydrogen
U^{235} enrichment	20%
Zirconium-to-hydrogen atomic ratio	1.0

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*The total number of fuel-moderator and dummy elements is 85. The number of fuel-moderator elements, sufficient to provide an initial excess reactivity of 2.25% $\delta k/k$, is approximately 64. The balance will be graphite dummy elements.

U^{235} loading per element	36.76 grams (average)
Initial core loading (U^{235})	~2.3 kg (5.1 lbs)

Reflector

Material	Graphite
Cladding material	Aluminum
Radial thickness	12 in. (30.48 cm)
Top and bottom thickness	4 in. (10.16 cm)

Structures

Reactor structure	Ordinary concrete; 21.5 ft (6.55 meters) high, 22 ft 10 in (6.96 meters) wide, 28 ft 4 in (8.63 meters) long
Reactor tank	6.5 ft (2.0 meters) ID by 20.5 ft (6.25 meters) deep

Shielding

Radial	1.5 ft (45.7 cm) of water and a minimum of 8 ft 2 in (2.49 meters) of ordinary concrete or equivalent
Vertical	
Above core	~16 ft (4.9 meters) of demineralized H_2O
Below core	~2 ft (61.0 cm) of H_2O and a minimum of 3 ft (91.4 cm) of ordinary concrete

Experimental and Irradiation Facilities

Rotary specimen rack	40-position rack located in graphite reflector (each position can hold two

The prototype TRIGA reactor at General Atomic has been pulsed safely over 1000 times with 2.25% $\delta k/k$ insertions. The resulting power excursions attained a peak power of 1000 Mw, on a reactor period of 4.0 msec, with a total energy release during the burst of approximately 16 Mw-sec. The maximum measured fuel temperature for this pulse was less than 500°C.

On the basis of this experience on the operating TRIGA prototype, it is concluded that there is no hazard associated with the sudden accidental insertion of the total available excess reactivity (2.25% $\delta k/k$) in this reactor. Curves of the transient power level and the fuel temperature resulting from such an insertion of reactivity are shown in Fig. 13.

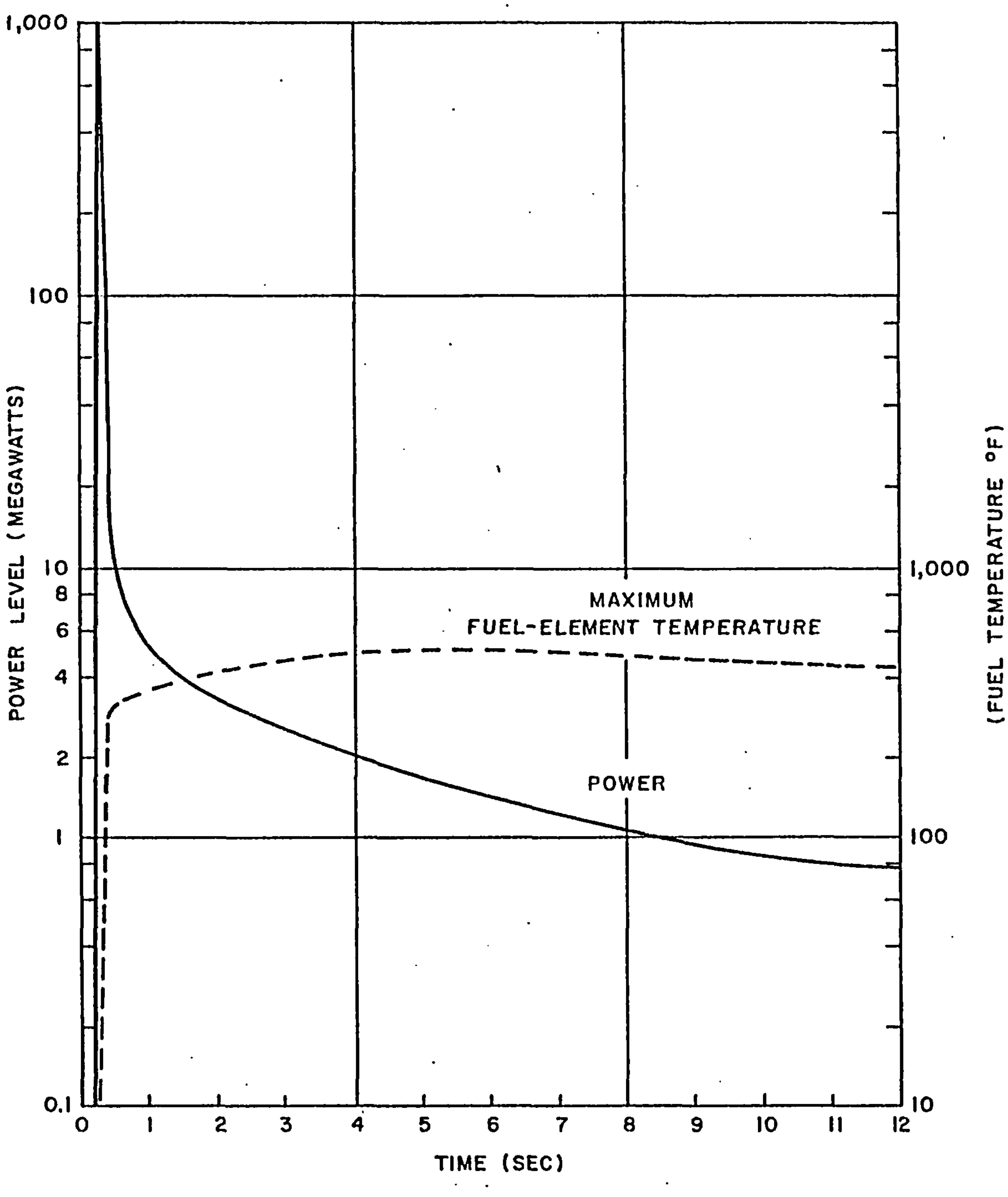

Figure 13 - Transient Power and Fuel Temperature as Functions of Time after 2.25% $\delta k/k$ (3.00 dollar) Reactivity Insertion

Gulf General Atomic
Incorporated
P.O. Box 608, San Diego, California 92112

GA-9064

SAFETY ANALYSIS REPORT

for the

TORREY PINES TRIGA MARK III REACTOR

Gulf General Atomic Project 1251.0000 January 5, 1970

It should be noted that kinetics calculation show that had the insertion occurred while the reactor was at full power the resulting maximum fuel temperatures would have been much lower.

8.3 LOSS OF REACTOR POOL WATER

8.3.1 Heat Removal

Although the total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine the maximum fuel temperature rise resulting from such a coolant loss. These calculations indicate that if the water in the pool is lost several minutes after a long period of operation at 2000 kW, the maximum temperature of the fuel, and consequently the stainless steel cladding, would be less than 520°C. At this temperature the equilibrium hydrogen pressure for $ZrH_{1.60}$ fuel material plus the pressure exerted by trapped air and fission product gases is less than 45 psi. This pressure produces a stress in the clad of about 1570 psi, whereas the yield stress for the stainless steel clad is 19,000 psi at 540°C. Therefore, the fission products will be retained in the fuel elements and the principal hazard from this accident is from the high radiation levels from the unshielded core.

It is reasonable to assume that the reactor is shut down for several minutes before the water is lost, as there would be ample indication that such a loss occurred and a scram could be initated (e.g. high radiation level.) However, calculations also show that if the loss occurred with the reactor at operating temperature the conclusions drawn above are not altered. The fuel and clad temperature would rise to about 590°C and the clad stress would be 1680 psi whereas the yield strength of the clad would be 17,700 psi.

Use was made of TAC, a two-dimensional, transient heat transport computer code, developed by Gulf General Atomic, for calculating the maximum temperature in the core after a water loss. It was assumed that at the time the water was lost, the fuel temperature was 25°C. It was also assumed that the reactor had been operating for an infinite time at 2000 kW with 100 elements in the core. The rate of energy release in the hottest element was determined from consideration of the energy deposition of fission product gammas and betas only. The energy release from delayed neutrons is relatively small (about 150,000 watt-sec total in the hottest element) and has an average decay constant of about 0.08 sec^{-1}.

The after-shutdown power density (in Btu/hr-ft^3) in the hottest fuel element is given by

$$\frac{q}{V} = 0.1\, p \frac{P}{V_f} \cos\left[0.78 \frac{\pi}{L}\left(x - \frac{L}{2}\right)\right]$$
$$\times \left\{\left[t + t_0 + 10\right]^{-0.2} - 0.87\left[t + t_0 + 2 \times 10^7\right]^{-0.2}\right\} \qquad (1)$$

where p = peak-to-average power density in the core = 2.0,

P = operating reactor power = 6.82×10^6 Btu/hr (2000 kw),

V_f = volume of the fuel in the core = 1.49 ft^3,

L = length of the fuel = 1.25 ft,

x = distance measured from the bottom of the fuel element, ft,

t = time after the core is exposed to the air, sec,

t_0 = time from shutdown to the time the core is exposed, sec.

Equation (1) is the Untermeyer-Weill formula that matches the work of Stehn and Clancy* to about 5×10^4 sec after shutdown. It is also conservatively assumed that all the energy produced by fission product decay in the element is deposited in the element.

While the decay gammas and betas are raising the fuel element temperature, the flow of air between the fuel elements will be removing heat and attempting to lower the fuel temperature. The air velocity through a central channel can be determined by setting the frictional pressure loss equal to the buoyancy . Entrance and exit losses will be negligible compared with the frictional losses.

(1) Mass Flow Rate of Air

The mass flow between fuel elements has been derived by Dee† but modified for laminar flow. The following is based on this analysis.

* Stehn, J. R., and E. F. Clancy, "Fission Product Radioactivity and Heat Generation," Paper 1071, Proceedings, Second United Nations International Conference on the Peaceful Uses of Atomic Energy, United Nations, Geneva, Switzerland, September 1958.

† Dee, J. B., "Steady State Natural Circulation in the 15 Mw(t) Pwr-SRGA," November 11, 1964.

The driving head is given by the integral over the effective channel length

$$\Delta P_b = \int \rho \, dh \, . \qquad (2)$$

This can be expressed as

$$\Delta p_b = \frac{h}{144} (\rho_0 - \rho_1) \qquad (3)$$

where h is the effective height between heating and cooling centers in ft, ρ is density in lb/ft^3, and the buoyancy force Δp_b is in $lb/in.^2$.

Evaluation of the effective height, h, is complicated by the difficulty in calculating the effect of cooling and mixing of the air after it has exited from the upper grid plate as well as predicting the friction losses as the rising air entrains the bulk air in the environment. This distance has been taken as 1.5 feet.

The friction force acting to retard the flow is the product of the wetted surface area, the velocity head and a friction coefficient. The friction force is manifested as a pressure drop, Δp_f, acting over the flow cross-sectional area, i.e.,

$$\Delta p_f = \frac{\rho}{2g} \frac{L}{144} \left(\frac{Pe}{A}\right) f \, v^2 \qquad (4)$$

where ρ is the average fluid density in lb/ft^3, L is the length of the channel in ft, Pe is the friction perimeter in ft, A is the flow area in ft^2, v is the velocity in ft/sec and f is a friction coefficient.

The flow velocity is related to the mass flow rate by

$$vA\rho = w$$

with w, the mass flow rate in lb/sec, and, for laminar flow, the friction coefficient f is given by

$$f = \frac{16}{Re} = \frac{16\mu A}{3600 \, wD}$$

where Re is the Reynolds number, D is the hydraulic diameter in feet, ($=4\,A/\pi\ D_{el}/2$), D_{el} is the fuel element diameter in ft, and μ is the dynamic viscosity in lb/ft-hr.

Combining the last three equations yields

$$\Delta p_f = \frac{1.5432 \times 10^{-5}}{g}\left(\frac{w\mu}{\rho}\right)\left(\frac{LPe}{A^2 D}\right)$$

$$= 0.10185\left(\frac{w\mu}{\rho}\right)\left[\frac{L}{d^4}\left(\frac{d}{d_{el}}\right)\right] = 0.10185\,F\left(\frac{w\mu}{\rho}\right) \tag{5}$$

where d is the hydraulic diameter in inches and d_{el} is the fuel element diameter in inches, and

$$F = \frac{L}{d^4}\left(\frac{d}{d_{el}}\right) .$$

The pressure loss associated with the acceleration of the fluid is given (in psi) by

$$\Delta p_v = \frac{\rho_1 v^2}{2g}$$

$$= \frac{w^2}{288\rho_1 A^2 g} \tag{6}$$

where A is the flow area in ft^2, and w in lb/sec, and ρ_1 is the exit air density in lb/ft^3.

Equating the buoyant force with the friction and acceleration losses yields

$$\frac{w^2}{288\rho_1 A^2 g} + 0.10185F\left(\frac{w\bar{\mu}}{\bar{\rho}}\right) = \frac{h}{144}(\rho_0 - \rho_1) \tag{7}$$

with the viscocity and density in the friction term averaged over the length of the channel assuming a linear temperature rise. The mass flow is then

$$w = \left\{\left[14.666\rho_1 A^2 g\, F\left(\frac{\bar{\mu}}{\bar{\rho}}\right)\right]^2 + \rho_1 A^2 gh\,(\rho_0 - \rho_1)\right\}^{1/2} - 14.666\rho_1 A^2 g\, F\left(\frac{\bar{\mu}}{\bar{\rho}}\right) \quad (8)$$

Over the range of temperatures of interest (up to 2000°F), the properties of air have been approximated by linear equations. Thus,

$$\rho = \frac{1}{2.5 \times 10^{-2}\, T} \ \text{lb/ft}^3$$

and

$$\mu = (0.01135 + 0.6017 \times 10^{-4} T)\ \text{lb/hr-ft}, \quad (9)$$

where T is the temperature in °Rankine.

(2) Heat Transfer Coefficient

For air being heated as it flows upward in vertical tubes, McAdams* suggests that the heat transfer is characteristic of a combination of forced and natural convection as the velocity near the wall is increased by the buoyant force owing to the difference in temperature. The heat transfer correlation used has the form

$$h = 1.75\, F_1 \frac{k_f}{D}\left[\frac{wc_{pb}}{k_b L} + 0.0722\, F_2 \left(\frac{D}{L}\, GrPr\right)^n\right]^{1/3}. \quad (10)$$

The factors F_1 and F_2 are functions of the ratio $Z = (T_1 - T_0)/\Delta\bar{T}_f$ where $\Delta\bar{T}_f$ is the average film temperature drop in the channel. For purposes of calculation the factor Z was assumed to be 1.0, and according to McAdams the factors F_1 and F_2 are 0.912 and 0.588 respectively. Also, the subscripts b and f refer to evaluation of the property at average bulk temperature and average film temperature respectively.

Again, over the temperature range of interest, one can write for the thermal conductivity and specific heat of the air

* McAdams, W. H., "Heat Transmission," 3rd edition, McGraw-Hill Book Co., 1954, p. 233.

$$k_a = (0.0009 + 0.26 \times 10^{-4}\ T)\ \mathrm{Btu/hr\text{-}ft\text{-}^\circ F}\ ,$$

and (11)

$$C_{pa} = 0.240\ \mathrm{Btu/lb\text{-}^\circ F}\ ,$$

with temperature T in $^\circ$R.

For the purpose of this analysis, the fuel element itself was considered to be a 1.25-ft-long, 0.1225-ft-diameter cylinder of TRIGA fuel with specific heat and thermal conductivity given by

$$C_{pf} = (26.3 + 0.0245\ T)\ \mathrm{Btu/ft^3\text{-}^\circ F}\ ,$$

and (12)

$$k_f = (10.7 - 6.42 \times 10^{-4}\ T)\ \mathrm{Btu/hr\text{-}ft\text{-}^\circ F}\ ,$$

where T is the local temperature in $^\circ$R. The temperature drop in the clad was ignored because it is insignificant.

For the computer program, the fuel element was divided into five radial and five axial regions, and the temperature in each region was computed as a function of time after water loss.

In Figure 8.6 the maximum fuel temperature calculated is plotted as a function of time after the loss of coolant water. The reactor was assumed to have been shut down several minutes before the core became dry and therefore the fuel temperature at the time of the water loss was about 25°C. It is difficult to conceive a set of conditions such that the pool could be drained in less time than required for the core to cool down to the ambient temperature. Nevertheless, a separate calculation was made with the water lost at shutdown and with the initial fuel temperature distribution of the operating reactor. The results of this calculation showed that the consequences were not significantly different from the accident analyzed with peak fuel temperatures about 75°C higher than for the much more reasonable postulated conditions.

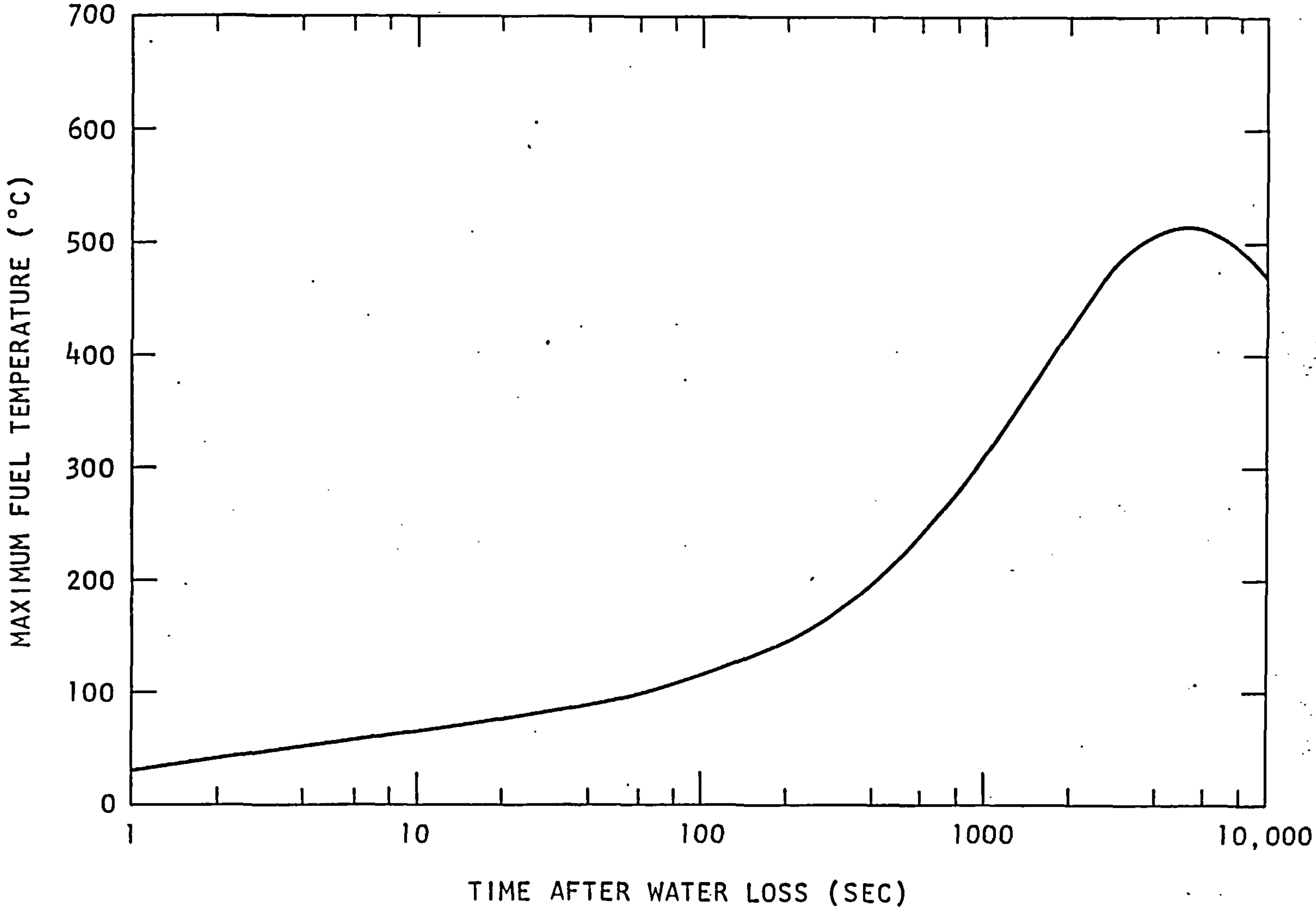

Fig. 8-6. Maximum fuel temperature vs time after water loss, 2 MW operation, 100 element core

An experiment was performed at Gulf General Atomic in which fuel element clad and coolant air temperatures were measured in a simulated loss of coolant situation. In this experiment* seven electrically-heated dummy TRIGA fuel elements were placed in a grid plate. The removal of heat from these dummy elements was accomplished through the natural convection of air up between the fuel elements and out through the top grid plate. Correlation between measured temperatures in this experiment and calculations using the model described above was very good.

8.3.2. Internal Pressure

To determine the pressure exerted on the cladding by released hydrogen, fission products, and air trapped in the fuel can, the conservative assumption will be made that the entire system is at the peak fuel temperature, i.e. 540°C.

The total number of fission product nuclei released in the gap between the fuel and clad was determined from Blomeke and Todd† and the results of an experiment.** The total quantity of Br, I, Kr, and Xe, released to the gap in the central fuel element after 4 years operation at 2000 kW will be

$$N_i = 4.6 \times 10^{-5} \times 4.00 \times 10^{22} = 1.8 \times 10^{18} \text{ atoms .} \tag{13}$$

The number of moles in the gap is

$$n_{fp} = \frac{1.8 \times 10^{18}}{6.02 \times 10^{23}} = 3.0 \times 10^{-6} \text{ moles .} \tag{14}$$

The partial pressure exerted by the fission product gases is

$$P_{fp} = n_{fp} \frac{RT}{V} , \tag{15}$$

* Shoptaugh, J. R., "Simulated Loss-of-Coolant Accident for TRIGA Reactors," GA-6596, August 18, 1965.

† Blomeke, J. O., and Mary F. Todd, "U-235 Fission Product Production as a Function of the Thermal Neutron Flux, Irradiation Time, and Decay Time," ORNL-2127, Oak Ridge National Laboratory, November 1958.

**Foushee, F. C., "Release of Rare Gas Fission Products from U-ZrH Material," GA-8597, March 29, 1968.