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THE UNIVERSITY OF TEXAS AT AUSTIN

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ATTN: Document Control Desk,
U.S. Nuclear Regulatory Commission,
Washington, DC 20555-0001

Allan Jason Lising
Project Manager
Division of Policy and Rulemaking
Research and Test Reactors Licensing Branch

SUBJECT: Docket No. 50-602, Request for Renewal of Facility Operating License R-129

REF: UNIVERSITY OF TEXAS AT AUSTIN - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE
LICENSE RENEWAL REQUEST FOR THE NUCLEAR ENGINEERING TEACHING LABORATORY TRIGA
MARK II NUCLEAR RESEARCH REACTOR (TAC NO. ME7694)

Sir:

Attached is a partial response to Request for Additional Information referenced above, including:

- RAI 1
- RAI 2.1
- RAI 2.2
- RAI 2.3
- RAI 3.1
- RAI 3.2
- RAI 6
- RAI 16
- RAI 20.2
- RAI 20.3
- RAI 21.1
- RAI 21.2
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A 020
NRR

- RAI 23
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- RAI 26
- RAI 31
- RAI 34.2
- RAI 35.2

We respectfully request an additional 90 days to complete response to the remaining items. If there are any questions, please feel free to contact P. M. Whaley at 512 232 5373 or whaley@mail.utexas.edu.

Your attention in this matter is greatly appreciated,

P. M. Whaley

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

S. Biegalski

ATT: RAI RESPONSES

RAI 1

The guidance in NUREG-1537 Section 1.8, "Facility Modifications and History," requests that the licensee provide descriptions of any changes to the facility that have been made, including changes made under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59. The UT safety analysis report (UT SAR) Section 1.6 provides brief information concerning two changes without indicating how they were accomplished. Please provide a list of all changes to the facility accomplished under 10 CFR Section 50.59 since the issuance of your current license.

RESPONSE

The guidance document indicates, "The modifications should be discussed briefly in this section in chronological order, including the number and date of the license amendment. Changes performed under the provisions of Section 50.59 of Title 10 of the Code of Federal regulations (10 CFR 50.59) that affect the SAR descriptions should be provided. If applicable, technical specifications changes should be also be given."

Information in the SAR was affected by two the changes in the proposed SAR, these changes did not require a license amendment or a revision to technical specifications, and the SAR as submitted is revised to incorporate the changes. There is no explicit requirement in the guidance document for a detailed description of the approval process for the changes (or for the changes themselves). A comprehensive list of all facility changes is attached; there are no other changes that affect the SAR.

FACILITY MODIFICATIONS

1991	Replaced DAC reset transistor with functional equivalent
1991	Replaced DAC PS high/low voltage test circuit resistors for better range
1992	Modified CSC program for 2000 MW (pulse) gain
1992	Replaced CSC disk controller with functional equivalent
1992	Replaced NP 1000 HV PS with functional equivalent
1992	Machined extension rod spacers in Shim 1/2 drive extensions
1992	Replaced shim rod drive circuit 220 ohm resistor with 700 ohm
1992	Replaced 2 ohm shim 1 position potentiometer with 5 ohm rheostat
1993	Changed FT1/2 setpoints from 410 to 500, test signal to 550
1994	Changed RMS location at BP2 to support installation of irradiation facility
1994	Modified console monitor to extract control commands
1995	Replaced HVAC damper access hatches for equipment clearance
1996	Replaced DAC floppy drive controller with functional equivalent
1996	Replaced DAC hard drive with functional equivalent
1996	Replaced transient rod down limit switch with functional equivalent
1996	Replaced HVAC alarm horn with functional equivalent
1997	Upgraded shim rod drives from AC motors to stepper motors drives
1997	Upgraded regulating rod stepper motor with functional equivalent
1998	Replaced Argon cam cabinet fan with functional equivalent

1999	Added braiding to fuel tool to protect the cable sleeve
2000	Operation with flooded reflector approved
2000	Controlled venting tool developed and used
2000	Replaced DAC with functional equivalent
2000	Replace CSC computer hard drive with functional equivalent
2001	Developed upgraded for the digital console hardware and software
2001	Replaced argon cam alarm relay rated at 6 a with 5 a relay
2001	Completed hardware/software replacement computer & UNIX with QNX
2002	Replaced Foxboro secondary flow transmitter with functional equivalent
2002	Performed GA software & hardware upgrade
2002	Replaced transient rod "down" optoisolator with functional equivalent
2002	Installed QNX upgrade
2002	Replaced NM 1000 IC (MB80 module) with functional equivalent
2004	Developed reflector modification for aluminum canisters and new placement for 7 element and both 3 element facilities
2004	Replaced pool cleanup filter piping with glued joint to threaded fitting
2004	Replaced pool RTD with a temporary, shirt lead RTD
2005	Replaced Tectronix display with monitor and terminal emulator
2005	Replaced DAC hard drive with functional equivalent
2006	Installed/restored temporary RACE accelerator lockout using Shim 2/4 magnet power
2009	Replaced heat exchanger magnahelic dp with double bellows pressure switch
2010	Replaced instrument air dryer with functional equivalent
2010	Replaced HVAC variable frequency controller circuit board with functional equivalent
2010	Replaced instrument air compressor with functional equivalent
2010	Replaced control console pushbutton switch lamps with LED devices
2011	Initiated VSE/security enhancements
2011	Replaced signal condition monitor with functional equivalent
2011	Replaced console LCD displays with functional equivalents
2011	Replaced ground detection action pack module with functional equivalent
2012	Installed shrink wrap on IFE thermocouple leads

RAI 2.1

UT SAR Section 4.2.1 discusses TRIGA fuel in general terms and refers to high enrichment fuel, fuel structures, and fuel stoichiometries but does not clearly state what fuel is to be used in UT TRIGA. Please identify the fuel that will be used under the renewed license. Clarify what fuel element types are allowed in the design features of the technical specifications (TS), and identify the geometries that are applicable.

RESPONSE

NUREG-1282, Safety Evaluation Report ON High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors "defines two types of TRIGA low-enriched uranium (LEU) fuels as "standard fuel…" including an 8.5 w% and a 12 w% fuel. The UT TRIGA reactor uses the standard, 8.5

w% fuel. As indicated in Chapter 4, the fuel is inserted in grid plates to establish a hexagonal core with a triangular pitch of 1.714 in.

Information will be provided in Chapter 4 as currently specified in Technical Specifications, Design Features, 5.1.3 indicating:

> "The high-hydride fuel element shall contain uranium-zirconium hydride, clad in 0.020 in. of 304 stainless steel. It shall contain nominally 8.5 weight percent uranium which has a maximum nominal enrichment of 20%. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom."

RAI 2.2

UT SAR Section 4.2.1 provides Figure 4.1 which relates TRIGA fuel phases to fuel temperature. A previously NRC-accepted diagram is from the Simnad Report (E-117-833, May, 20, 1986) which is the underlying basis for the TRIGA fuel SER (NUREG-1282). The Simnad diagram is significantly different from the one presented in the UT SAR. Please confirm that the UT SAR statements and conclusions are either based upon NUREG-1282 or provide a basis, methodology, and analysis for any differences.

RESPONSE

NUREG-1282, *Safety Evaluation Report on High Uranium Content, Low Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors*, has the expressed purpose of authorizing the use of higher weight per cent fuels, above the 8.5 w% fuel previously approved in a license 1978 amendment. In comparing the new w% fuel to the "standard" fuel, NUREG 1282 references a General Atomics report (Simnad, M. T., *"The U-ZrH$_x$ Alloy: Its Properties an Use in TRGA Fuel,"* E-117-833, GA Technologies, Inc, San Diego, CA, February 1980). Although the report is on its face the underlying basis for approval of uranium content exceeding the "standard" 8.5 w%, the Simnad diagram is extracted below and will be placed in the revised, proposed SAR.
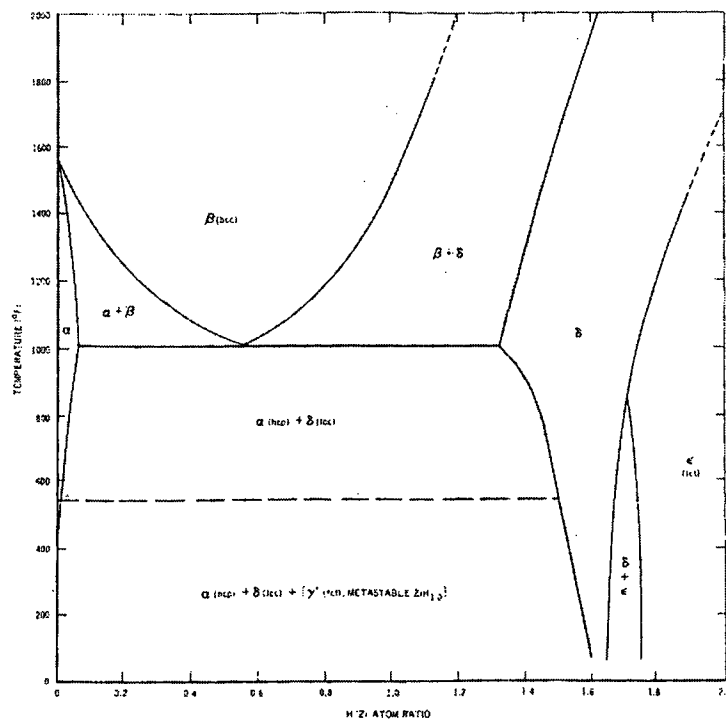
Fig., Zirconium Hydride Phase Diagram[1]

RAI 2.3

UT SAR Section 4.2.1 provides a statement on page 4-7 regarding the explosive potential of hydrogen resulting from reactions of zirconium with water. Please provide a discussion that confirms this potential does not exist for the UT TRIGA reactor under normal operations or accident conditions.

RESPONSE

NURGE/CR-2387 (PNL-4028), Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors discusses potential ZR-water reactions in TRIGA fuel:

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

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[1] Simnad, M. T., "The U-ZrHx Alloy: Its Properties an Use in TRGA Fuel," E-117-833, GA Technologies, Inc, San Diego, CA, February 1980, Fig. 2-4

explosion could then disperse the molten metal and thus provide the necessary physical conditions for the chemical reaction. However, this would take place so rapidly that, even if the water is vaporized by the molten metal, it will not be in contact with the metal sufficiently long to produce a chemical reaction of any significance. Indeed, the destruction of the BORAX-I reactor is thought to have been a result of a physical explosion rather than a chemical explosion (Epstein 1960).

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

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

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

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

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

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

The fuel elements recovered in this test showed a high degree of destruction but still retained large portions of their original form, although they were Hastelloy-N clad, not aluminum. Thus, it is reasonable to assume that a similar situation will occur with TRIGA fuels and that melting or a Zr-H$_2$O reaction will not occur.

Therefore, as long as the fuel remains below eth safety limit there is no potential for Zr-water reaction.

RAI 3.1

UT SAR Section 4.2.2 provides these descriptions; however, the absorber material is described as boron carbide. Typically in TRIGA control rods this material is boron carbide powder having a significantly different effective density that is based upon the particle size and packing fraction. Please confirm the type of absorber material used in the UT TRIGA control rods.

RESPONSE

ORNL/TM-2005/39 Version 6.1 (Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design), June 2011, Sect. M8 9Standards Composition Library) Table M8.2.4. Compounds indicates B4C (Boron Carbide, natural isotopic distribution) indicates theoretical density of 2.52 g cm^{-3}. General Atomics drawing TOS250D225 (Control Rod Fuel Follower) identifies drawing TOS250B226-1 as the sub assembly diagram for the poison section of the fuel follower control rod.

Drawing T135252D191 (Transient Rod Assembly) identifies drawing TOS250B226-2 as the sub assembly diagram for the poison section of the transient rod.

General Atomics drawing TOS250B226 indicates in the "next assembly" section drawings TOS250D225 and TOS252D191, with the specifications of items 1 and 2 of 1.300 in. diameter/15 in. length and 1.187 in. diameter/15 in. length, respectively. Note 1 states:

> MATERIAL TO BE BORON CARBIDE (B4C) HOT PRESSED. MINIMUM DENSITY SHALL NOT BE LESS THAN 2.48 GRMAS OF B4C PER CUBIC CENTIMETER OF COMPACT AND SHALL CONTIAN 78±1.0 WEIGHT PER CENT BORON. THE BORON SHALL CONTIAN THE NATURALLY OCCURRING ISOTOPE FRACTION OF ^{10}B, I.E., 18.5±0.2% THE TOTAL BORON TO TOTAL ATOM RATIO IN FINISHED PIECES SHALL BE 4.00±0.15.

Therefore the poison section of the in control rods is no less than 98% of theoretical density for B4C.

RAI 3.2

UT SAR Section 4.2.2 states that the control rod geometries and constituents are provided for both stainless steel and aluminum cladding. The design features in the UT SAR TS, Section 5, "Design Features," do not clearly indicate the types of control rods allowed for use in UT TRIGA. Please clarify what control rod types are allowed in the design features section of the technical specifications, and identify the geometries applicable to each of the four control rods in UT TRIGA.

RESPONSE

Chapter 4.2 will be revised to indicate:
1) The UT TRIGA currently has three fuel follower control rods (including one regulating rod and two shim rods) and one transient rod.

2) All control rods are capable of being scrammed. The fuel follower control rods are known as "standard" control rods, with electromagnets coupling the control rod drive to the control rod (through extension rods); scrams are accomplished in the standard rods using an interruption of power to the electromagnet. The transient rod extensions are coupled to the drive with a pneumatic piston; scrams are accomplished by venting air from the piston.

3) Fuel follower control rods currently installed in the UT reactor are fabricated in accordance with GA drawing TOS250D225 and revisions. The poison sections of the control rods are manufactured in accordance with GA drawing TOS250B226 (sub-parts 1 and 3) and revisions, with poison sections consisting of 15 in. B$_4$C cylinders, hot pressed to minimum density of 2.48 g cm^{-3}. The poison sections are 1.300 in diameter (fuel follower control rod) and 1.187 in. diameter (transient rod).

4) Standard fuel element follower control rod cladding is stainless steel tubes with welded end fittings approximately 45 in. (114 cm) long by 1.35 in. (3.429 cm) in diameter (1991 UT SAR). Control rod component dimensions are provided on GA drawing TOS250D225 (Control Rod – Fuel Follower). The cladding outer diameter is specified by the 1991 UT SAR as 1.35 in.

   The upper most 6.5 in. (16.51 cm) is an air void, separated from 15 in. (38.1 cm) of B_4C absorber by a 0.5 in. (1.27 cm) plug. The plug secured by a magneform weld and a small (0.12 in, 0.305 cm) air gap. Below the poison section is another 0.5 in. plug and magneform weld followed by a 0.25 in (0.635 cm) air gap, and fuel. The fuel rests on a double thickness 1 in. (2.54 cm) plug and magneform weld, followed by a 6.5 (16.51) in. air void. The bottom air void has an aluminum insert with wall thickness 0.35 in. (0.089 cm).

5) Transient control rod cladding is a 43 ¾ in aluminum tube with welded end fittings. The poison section of the transient rod is 15 in (38.1 cm). A double (1 in., 2.54 cm) plug with a magneform weld secures the poison section at the top and bottom. The air follower above the poison section is in an assembly 5.94 in., which includes the upper end fitting and the upper magneform weld. The air follower under the plug is 20.88 in. (53.02 cm) long, terminating in bottom fitting.

   The transient rod is surrounded by an aluminum guide tube. The guide tube is perforated by ½ in. holes at 90° rotations on 1 in. centers at the top and bottom of the core barrel

6) Control are in placed in gird plate positions fabricated such that the lower grid plate penetration is the same diameter as the upper grid plate penetration.

7) RAI 6: The guidance in NUREG-1537 Section 4.4, "Biological Shield," requests that the licensee provide a description of the biological shield and how it assures acceptable control of personnel exposure. UT SAR Section 4.4 includes a statement, "One goal of the design is a radiological exposure constraint of 1mrem/hour for accessible areas of the pool and shield system." Please describe methods used to demonstrate acceptable radiological exposure.

8) Attached is an excerpt from the Nuclear Engineering Teaching Laboratory Startup Program describing the methods used to perform acceptance surveys of the biological shield during initial commissioning of the facility.

(1) The energy response of the instrument shall be in compliance with the currently applicable standard, American National Standard Radiation Protection Instrumentation Test and Calibration, ANSI N323-1978[12], over the gamma-ray energy range of 0.2 - 3.0 MeV.

(2) Instruments shall be available for measurements within the accuracies stated in the above item (1), over the range of exposure rates from 0.05 to 25000 mR/h.

c. Neutron Instrumentation. The features that neutron instruments shall possess include:

(1) The instrumentation accuracy for determining dose equivalent rates shall be in compliance with current applicable standards ANSI-N323-1978(12), at thermal neutron energies and over the neutron energy range from 0.2 to 7 MeV. The inherent overresponse of the dose equivalent rate in the range from thermal to 0.2 MeV should not exceed a factor of four.

(2) Instruments shall be available for measurement within accuracies stated in the preceding item (1), over the range of dose equivalent rates from 0.05 to 5000 mrem/h (0.5 to 50,000 µSv/h).

(3) The detectors shall be capable of gamma-ray discrimination consistent with the gamma-ray background in the areas to be suveyed. If sufficient neutron spectral data are available, flux measuring instruments may be used and dose equivalent rates calculated. Dose equivalent rate estimates can be

computed by using flux to dose conversion factors developed
from American National Standard Neutron and Gamma-Ray
Flux-to-Dose-Rate Factors, ANSI/ANS-6.1.1-1977(13).

3. Radiation Base Points

   a. Radiation background measurements shall be made to provide values
   against which measurements made during the startup and operating
   phases can be compared. The background measurements should be
   made with the same type of survey instrument that is used in the
   shield surveys. These measurements will often be most
   conveniently made prior to facility initial startup.

   b. Horizontal Shield Sections. Scans should be taken at the one
   meter level above the floor and should be performed in a manner
   such that the scanning paths are not more than one meter apart.
   If, during a scan, measured radiation levels are above design
   levels corrected to correspond to source intensities at the time
   of the measurements, a scan shall be made to determine the
   location of the possible shielding deficiency. A new RBP shall
   be designed for this location until the deficiency is corrected.

   c. Vertical Shield Sections. Measurements shall be made with the
   detector as close to the shield section as is practical.
   Scanning of the areas should be performed in a manner such that
   the scanning paths on the shield surface are not more than one
   meter apart. If, during a scan, high dose equivalent rates are
   measured, a thorough survey shall be made to determine the
   location of the possible shielding deficiency. A new RBP shall be
   designated for this location until the deficiency is corrected.

WALL   **WALL**
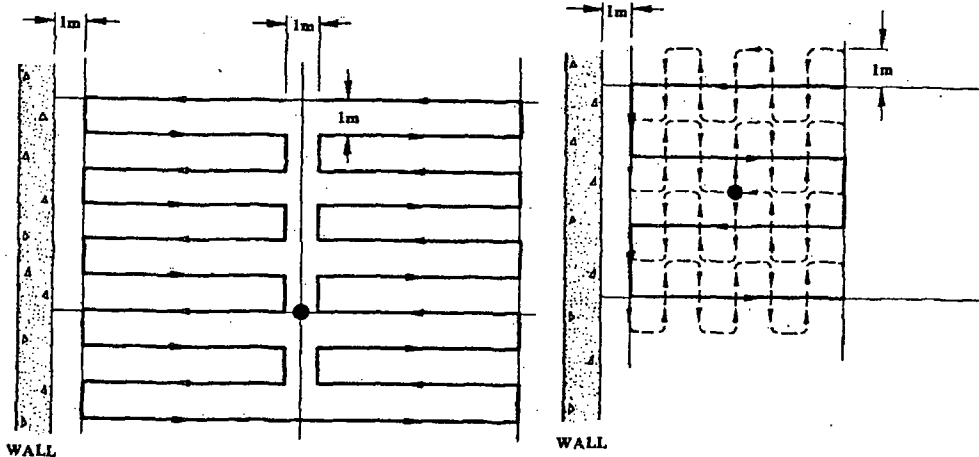
● = Radiation Base Point (At 1m Level)
——— Surveyor's Path & Instrument Scan (At 1m Level)

● = Radiation Base Point (At 1m Level)
— — —Instrument Scan (At 1m Level)
——— Surveyor's Path

FLOOR
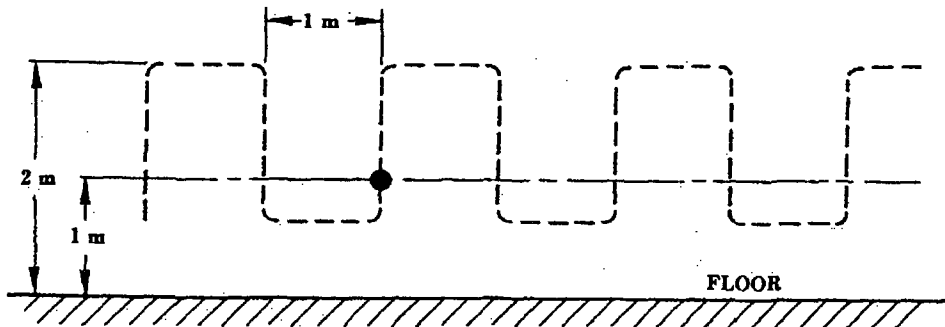
● = Radiation Base Point (At Vertical Shield Surface)
— — —Instrument Scan Path

     d.   Penetrations. Scanning measurements shall be made around each

penetration or cluster of penetrations. If, during a scanning

survey, greater than design level dose equivalent rates are

measured, a thorough survey shall be made to determine the

location of the possible shielding deficiency. The RBP shall

remain until the measured dose equivalent rates are reduced to

within the design limit.

RAI 6: The guidance in NUREG-1537 Section 4.4, "Biological Shield," requests that the licensee provide a description of the biological shield and how it assures acceptable control of personnel exposure. UT SAR Section 4.4 includes a statement, "One goal of the design is a radiological exposure constraint of 1mrem/hour for accessible areas of the pool and shield system." Please describe methods used to demonstrate acceptable radiological exposure.

Attached is an excerpt from the Nuclear Engineering Teaching Laboratory Startup Program describing the methods used to perform acceptance surveys of the biological shield during initial commissioning of the facility.

Radiation Survey of Reactor Shield System

1. Instruments

The instruments used to perform the testing of facility shielding
should be calibrated using the methods described in Reference (12).
Some specifications for the survey instruments of the types used in
the testing of facility shielding are given below:

a. General Characteristics. Gamma-ray and neutron survey
instruments used in performing the low-power shield tests shall
be capable of detecting exposure rates and dose equivalent rates,
respectively, as low as 0.05 mR/h and 0.05 mrem/h (0.5 μ Sv/h).
More than one instrument may be necessary to cover the expected
range of measurements. Considerations in the selection of
instrumentation include:

(1) Instruments shall retain their calibration and response
characteristics for the duration of each testing period.

(2) Instruments shall be capable of measurements of the dose
rates in a mixed field of gamma-rays and neutrons with
capability of gamma-ray discrimination for neutron
instruments.

(3) Instruments shall meet the specified performance
requirements under the radiation, temperature, and humidity
conditions of measurement.

(4) Directional dependence of the instrument shall be known.

b. Gamma-ray Instrumentation. The features that gamma-ray
instruments shall possess include:

RAI 16.1

The pool dimensions of a "tall tank formed by the union of two half-cylinders with a radius of 6½ ft. (1.9812 m) with 6½ feet separating the half-cylinders," appears to be inconsistent with the stated tank nominal water volume of 40.57 cubic meters. Please confirm and revise accordingly.

RESPONSE

A sketch of the pool surface area:
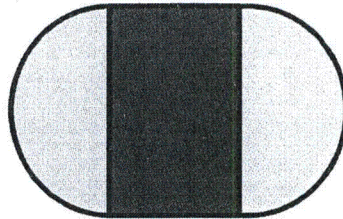
Decomposed to:
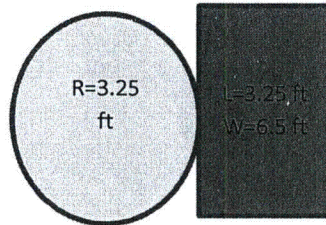
POOL DIMENSIONAL ANALYSIS

Dimension	ft	m	AREA (m^2)
R	3.25	0.9906	3.08
L	6.5	1.9812	1.963
W	3.25	0.9906	
TOTAL			5.045

VOLUME ANALYSIS

Level	Elevation (m)	Volume Between Elevations
Normal	8.179	41.266
Minimum (TS)	6.5	32.795
Core Top	0.51	2.573
Core Bottom	0.2519	1.271
TOTAL		77.905

With a reflector height of approximately 0.54 m, a reflector radius of approximately 0.6 m, and the hexagonal core metal to water ratio of approximately 1:3, the volume occupied by the reflector and core

is on the order of 0.5 m and can be neglected in evaluating pool water volume. Therefore during normal operations pool volume is 41.3 m^3, with approximately 37.4 m^3 over the core elevation.

RAI 20.2

Describe any measuring systems used to confirm that acceptable reactivity levels are maintained in storage locations, how those systems are controlled by procedures or UT TRIGA TS, and how they are calibrated.

RESPONSE

A criticality alarm system is installed to support material under a non-reactor special nuclear materials license. No other controls or instrumentation exist specifically to support storage locations.

RAI 20.3

UT SAR Section 9.4.5 states that fuel elements are required to be stored in a configuration with k_{eff} less than 0.8. This is inconsistent with the statement in UT SAR Section 5.2.3 of the proposed UT TRIGA TS, which states, "The k_{eff} of all fuel elements or fueled devices in storage is less than 0.9." Please explain this discrepancy.

RESPONSE

Both values should be 0.9 as per revised ANSI standard; the UT TRIGA TS was incorrectly incorporated directly from existing Technical Specifications without updating to the current standard.

RAI 21.1

UT SAR Section 9.4.3 states that a top loading fuel transfer cask is used and that there is "no potential for failure or mishandling as exists in a bottom loading cask." Please describe how the use of such a cask eliminates the potential for such accidents.

RESPONSE

The wording will be revised to indicate that potential for a dropped fuel element is restricted to insertion and removal of the element from the cask, with distance for the drop limited within the pool to clearance of the element from the core as it is removed to a maximum distance between the element and the bottom of the pool.

RAI 22.1

In the process of confirming the dose calculations in UT SAR Section 11.1.1.1.1 the NRC staff finds that the beam tube volume cited should be in units of cm^3. Please confirm.

RESPONSE

The beam tube volume and RSR volume unit of cm^{-3} is a typographical error and should be cm^3.

RAI 22.3

UT SAR Section 11.1.1.1.1 does not describe the whole body dose to facility staff. Please provide a discussion of facility worker doses, and whether these doses are ALARA.

RESPONSE

Dosimetry results for NETL staff over the past several years shows that typical annual occupational doses are less than 100 mrem which is well below the NETL ALARA goal of 1 rem per year.

RAI 23

The guidance in NUREG-1537 Section 11.1.1.2, "Liquid Radioactive Sources," requests that the licensee describe liquid radioactive effluents. UT SAR Section 11.1.1.2 describes the major liquid radioactive source term as being the primary coolant, but does not describe if there are any liquid effluents, their point of discharge, and whether the effluents are within the limits of 10 CFR Part 20. Please describe the liquid effluents providing this detail.

RESPONSE

During normal operations, there is no liquid effluent and thus no points of discharge or comparison to limits to be discussed.

RAI 24

The guidance in NUREG-1537 Section 11.1.5, "Radiation Exposure Control and Dosimetry," provides guidance on radiation exposure control and dosimetry, indicating that exposure limits should be administratively established for all accessible locations of the facility, including exposure limits to visitors. The UT SAR does not provide a description of exposure limits to visitors, or the dosimetry

provided to determine compliance with those limits. Please provide a description of how UT meets NUREG-1537 Section 11.1.5.

RESPONSE

While ALARA has always been applied to visitor doses, an exposure limit was not explicitly stated in the NETL ALARA program. A revised version of the NETL ALARA program with a visitor ALARA goal of 50 mrem per year has been drafted and will be submitted to the NETL Reactor Oversight Committee for approval. As stated in UT SAR Section 11.1.5.5.2, direct reading dosimeters (pocket ion chambers or electronic dosimeters) are used by visitors when in restricted areas.

RAI 26

NUREG-1537 Section 11.2.3, "Release of Radioactive Waste," provides guidance on the release of radioactive waste. UT SAR Section 11.2.2.3 provides an estimate of the annual average generation of solid waste of 25 cubic feet, while Table 13.1 in UT SAR Section 11.1.1.3 states that annual solid waste volume is typically less than 2 cubic feet. Please discuss this discrepancy.

RESPONSE

As stated in UT SAR Section 11.2.2.3, approximately 25 cubic feet of solid radioactive waste is produced annually at NETL. However, most of this waste has a relatively short half-life and decays to background levels relatively quickly. The remaining solid radioactive waste which contains material with a relatively long half-life consists of approximately 2 cubic feet of material. Since the waste is generated roughly uniformly throughout the year, the shorter half-life material decays to background roughly uniformly throughout the year while the longer half-life material accumulates roughly uniformly throughout the year. Thus, the value of 2 cubic feet of solid radioactive waste was entered into Table 13.1 as an estimate of amount of solid radioactive waste on hand at any one time.

RAI 31

The guidance in NUREG-1537 Section 13.1.9, "Mishandling or Malfunction of Equipment," requests that the licensee provide an accident analysis assuming equipment mishandling or malfunction. The UT SAR does not identify any potential effects from movements of heavy loads using the 5-tonne crane. Please identify potential incidents, including loss of power or dropped loads, related to the operation of the crane and discuss the consequences.

RESPONSE

As addressed in response to RAI 21.2, The overhead crane has a dynamic brake that does not release if power is secured. Loads over the core while the reactor is operating are prohibited by procedures. The bridge over the core is directly over the core, and provides significant interference from impact by falling objects.

RAI 34.2

UT TRIGA TS 2.2.The REQUIRED ACTION, B.1 and B.2, which support condition B seem to be reversed and the completion times are both labeled B.2. Please discuss this error and/or revise accordingly.

RESPONSE:

The second action is changed, requiring evaluation that the safety limit was not exceeded

	B.1 ENSURE REACTOR SHUTDOWN condition	B.1 IMMEDIATE
B. An INSTRUMENTED FUEL ELEMENT in the B or C ring indicates greater than 550°C	AND	
	B.2 EVALUATE whether Safety Limit was exceeded	B.2 IMMEDIATE

RAI 35.2

Section 3.1 of the guidance describes that limits be placed on core excess reactivity and the corresponding regulatory interpretation includes provisions that account for experiment worth. Proposed UT TRIGA TS 3.1 "Core Reactivity," Specification A excludes consideration of experiments having positive reactivity.

RESPONSE

The statement that consideration of experiments with positive reactivity worth are excluded in determining excess reactivity is not correct.

EXCESS REACTIVITY: That amount of reactivity above the critical condition which would exist if all the control rods were moved to the maximum positive reactivity condition

TS 3.1:

The maximum available core reactivity (EXCESS REACTIVITY) with all control rods fully withdrawn does not exceed 4.9% Δkk ($7.00) when:
1. REFERENCE CORE CONDITIONS exists
2. No MOVEABLE EXPERIMENTS with net-negative reactivity worth are in place

There is no intent or wording that excludes positive worth of experiments in evaluating excess reactivity, i.e., positive experiment worth contributes to core excess reactivity, although no credit is given in reducing shutdown margin by the negative worth of experiments that might be moved during reactor operation.