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Office of the President

May 24, 1994

Alexander Adams, Jr.
Project Manager
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Docket No. 50-606

Subject: Response to Request for Additional Information

Dear Mr. Adams:

Arkansas Tech University's response to your request for additional information dated January 28, 1991 is enclosed. There are four attachments. Responses to items 5.a (fuel elements), 5.i, 5.l(2), 7.c, 8.b, 10.c, 15.e(3), 16.g(1), and 18.a..p (Chapter 10) will be provided at a later date.

Arkansas Tech University's response has been delayed by complications associated with construction of the Center for Energy Studies where the reactor will be located, the NRC inspection fee for research reactors, and issues relating to maintenance and operation of the reactor. It is our judgment at this time that a sufficient number of these items have been favorably resolved to warrant continuation of the licensing process.

Sincerely,

Robert C. Brown
President

enclosure

A120 1/1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

ARKANSAS TECH UNIVERSITY

1. Introduction page 1.1; **[Note: Page number format is 1-1 instead of 1.1]**

   a. Please use the standard notation of (%Δk/k), and (____S) throughout your documentation.

   The changes will be made in the revised SAR and the Tech Specs.

2. Section 1-2 page 1.1;

   a. You mention "core irradiation tubes." but have confined later discussion in the SAR to one central irradiation tube. Please clarify, or provide analyses for more than the central tube.

   There is only one core irradiation tube which is the core central irradiation tube. The sentence will be changed to read as follows.

Reactor experimental facilities will include a rotary specimen rack, a pneumatic transfer system, and a central core irradiation tube.

   b. Table 1-1, page 1.2 and 1.3; there are some entries that need to be addressed:

      (1) The notation of %Δk/k and dollar. (See comment No. 1.)

      Changes will be made.

      (2) The ratio of hydrogen to zirconium in the fuel/moderator material.

      Zr/H will be corrected to H/Zr.

      (3) The numerical magnitude of the reactor temperature coefficient of reactivity (justify value).

      0.11 %Δk/k will be corrected to 0.011 %Δk/k

      (4) The absence of a void coefficient of reactivity.

The void coefficient of reactivity is given as 0.1 %Δk/k/(1% void) in General Specifications and Description of TRIGA Mark I Reactor, GA 2070, and is given as 0.14 %Δk/k/(1% void) in Safety Analysis for Michigan State University TRIGA Reactor, June 1967. This information will be included in Table 1-1.

(5) Last part of Table needs the standard notation of cm(in) and m(ft).

Changes will be made.

c. Page 1-3; your reliance on 3.5 %Δk/k insertion, and peak powers of 8400 MW do not address possible differences of the GA reactor from the proposed ATU reactor. Please make quantitative comparisons of number of fuel elements, power distributions throughout the cores, average and maximum energy densities in the fuel, power densities in the fuel rods and average and maximum fuel temperatures.

Paragraphs 2 and 3 on page 1-3 will be replaced with the following paragraph.

The prompt shutdown mechanism has been demonstrated extensively in many thousands of transient tests performed on two prototype TRIGA reactors at the GA Technologies laboratory in San Diego, California, as well as other pulsing TRIGA reactors in operation. Because the reactor fuel is similar, the previously cited experience and tests apply to this TRIGA system. As a result, it has been possible to use accepted safety analysis techniques applied to other TRIGA facilities to update evaluations with regard to the characteristics of this facility [1-8].

3. Chapter 2;

a. Please provide distances from the reactor to the nearest major highways and rail lines.

Interstate Highway 40 passes about one-half mile north of the reactor site on ATU campus. Arkansas Highway 7 passes through ATU campus and adjacent (500 ft) to the reactor site. The Union-Pacific Railroad passes one mile south of the reactor site on ATU campus.

b. Do any major airways pass over the ATU campus? If yes, please address the density of air traffic and possible affect on the reactor.

The nearest airport is Russellville Municipal Airport, located approximately 3½ miles southeast of the ATU campus, which can accommodate small jets (4,450 ft runaway). There are no major airports with a control tower within 50 miles of the ATU campus. The only air traffic passing over the ATU campus is due to private planes and occasional commercial aircraft at high altitudes. The Russellville airport has no regularly scheduled passenger service.

4. Section 2.2: Please provide information on the distances and directions from the reactor to the nearest occupied building, such as a dormitory, and to the nearest permanent residence in the unrestricted area. In the later analyses for maximum potential radiation exposures to the public, include these locations for both routine operations and potential accidents, for both Argon-41 and fission products.

   The nearest occupied building is Jones Hall which is a dormitory. This building is situated approximately 120 ft southwest of the reactor site. The nearest permanent residence is located approximately ¼ mile north of the reactor building. The exposure potential is analyzed later.

5. Section 3.2:

   a. Itemize all components, including all fuel elements, of the ATU reactor that have been previously used. Give detailed history and conditions of their use, interim storage, refurbishment, etc. Provide explicit criteria used to determine acceptability for ATU, and reasons that ATU deems that integrity and operability for the requested period of the operating license is reasonably assured.

      Fuel Elements will be addressed separately. The following ATU reactor components have been previously used.

      1. Reflector Platform - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

2. Reactor Reflector Assembly - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage, the integrity of the weld will also be checked (see 5.i)

3. Neutron Source and Holder - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

4. Grid Plates - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

5. Rotary Specimen Rack, Guide Tubes, Drive and Indicator - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage, water tightness and operability will be checked.

6. Central Thimble - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

7. Control Rods (3): Drives, rods, Guide tubes. Drives - MSU, Drives were completely refurbished by GA, Stored at ATU, GA has responsibility for installation. Rods, will be new. Guide tubes - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

8. Detectors(3): Detectors, New. Mounting Ring, Guide tubes - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

9. Underwater Light Assembly - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

10. Fuel Element Storage Racks, Fuel Element Handling Tool - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage.

11. Control Console - GA, used at GA reactor facility, refurbished by GA, stored at ATU, will be installed by GA.

12. Primary System : Pump, Heat Exchanger, Demineralizer, Instrumentation - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected for any damage, pump will be refurbished and operability will be checked, demineralizer will be charged with resin.

15. Compressed Air System - MSU, used at the MSU reactor facility, Stored at ATU, will be visually inspected and operability will be checked.

b. In Section 3.2.1.1, first paragraph, second sentence, should it be: "...enriched to less than 20% U-235."?

Will be corrected.

c. In Section 3.2.1.1, second paragraph, the wording is confusing; please clarify. Also, Fig. 3-2 refers to a SS tube not a SS can, please use consistent wording and notation.

The paragraph will be corrected as given below.

Each element is sealed in a 0.020 in. (0.0508 cm) stainless steel tube (cladding) and all closures are made by heliarc welding. Two sections of graphite are inserted in the stainless steel tube, one above and one below the fuel, to serve as top and bottom reflectors for the core. A molybdenum disc separates the lower graphite section from the fuel.

d. Because GA has marketed both gapped and non-gapped stainless steel clad fuel, and because this affects heat transfer from both the fuel meat and the graphite end-pieces, please tell us which you will have, and address the effects on fuel temperatures and reactor performance in the appropriate sections of the SAR.

All calculations are done for non-gapped fuel elements. The Berkeley fuel that we expect to use is standard non-gapped fuel element.

e. In Section 3.2.1.2; Give the same information as in 5 a. for the instrumented fuel elements, and for your neutron detectors. How many in-

strumented fuel elements will be present at the ATUTR, and where will they be located in the core?

Instrumented fuel elements: There will be two instrumented fuel elements: one in the B ring and the other in the C ring. A typical core configuration is shown in Figure 3-6. These will be new non-gapped elements from GA.

Neutron detectors: There are three neutron detectors; one fission and two ionization chambers. All detectors will be new.

f. Section 3.2.1.3, Graphite dummy elements; If you intend to operate with such elements in any fuel location other than the outer ring, please provide an analysis of thermal and power density effects on nearby fuel elements, in both steady state and pulsed operation.

We expect to operate the reactor with the typical core configuration shown in Figure 3-6. Graphite dummy elements will be used in only the outer, F, ring. The graphite elements are not expected to be placed at any other location.

g. Section 3.2.2.1; Were the grid plates designed by GA, Michigan State, or Arkansas Tech? What design review did they receive? Please provide a reference or discussion.

The grid plates to be used for the ATUTR are standard GA design grid plates which were previously used in the Michigan State facility.

h. Section 3.2.3; What is the neutron source strength, in neutrons per second?

The strength of the Am-Be neutron source is 1.88 Ci. The neutron yield is 70 neutrons/10^6 primary alphas (Knoll, Radiation Detection and Measurement). Therefore, the neutron source strength is 4.9×10^6 n/s .

i. Section 3.2.4; If the graphite reflector assembly is not new, what precautions have been taken to assure that the water tight integrity is still valid and the graphite is dry? Discuss in further detail sealing and capping the unused beam port. What would be the effect if the seal fails?

Will be addressed later.

j. Section 3.2.5: Please reference the design review of the tank. What was the design criterion? What is the basis of the aluminum wall thickness being 1/4 in.?

The hydrostatic pressure due to the maximum water height of 25 ft is 10.8 psi. The maximum principle stress on the tank wall of radius 5 ft and thickness 0.25 in. due to the above hydrostatic pressure is 2600 psi and the maximum shear stress is 1300 psi. The principle stress is less than the yield strength for aluminum (35000 psi for 6061-T6, Beer and Johnson, Mechanics of Materials, 2nd ed.). This provides a factor of safety of approximately 13. The bottom of the tank will be made of 0.5 in. aluminum. Other structural reinforcements will also be present.

k. Are there any penetrations in the reactor tank? If so, please discuss the impact on reactor safety if failure of integrity were to occur.

No, there are no penetrations in the reactor tank.

l. (1) Tank coating to prevent corrosion has lost integrity in some non-power reactors. Please discuss details of the design, what tests have been performed, and what assurances you have that the life-time of the proposed coating will extend at least as long as the proposed operating license.

The aluminum tank will be surrounded by 2 ft of reinforced concrete. This concrete will, in turn, be enclosed in a 3/8" thick steel liner surrounded by another foot of reinforced concrete. The outer barriers of concrete and steel will prevent any water present in the surrounding ground from reaching the aluminum/concrete interface and thus prevent corrosion of the tank liner throughout the life of the facility. In addition to the barriers, the aluminum tank will have an epoxy coating.

(2) Discuss neutron fluxes beyond the tank surfaces and possible activation of soils and ground water.

Will be addressed later.

m. Section 3.2.5, third paragraph, second sentence: Is the isotope production facility actually the rotary specimen rack? If so, please use consistent terminology or cross reference.

Yes. The isotope production facility is the rotary specimen rack. Sentence will be corrected.

n. Please commit that you will develop written procedures for the use or movement of the "shielded isotope cask" above the reactor and its control rod mechanisms. Discuss the plans.

The next sentence will be added to the third paragraph in section 3.2.5. "When the isotope cask is moved over the reactor tank, written procedures pre-approved by the Reactor Safety and Utilization Committee will be followed." The paragraph will read as follows:

The center channel assembly is located at the top of the reactor tank directly over the reactor core. It provides support for the rotary specimen rack (isotope production facility) drive and indicator assembly, the control rod drives and the tank covers. The assembly consists of two 8 in. (20.3 cm) structural steel channels covered with steel plates 16 in. (40.7 cm) wide and 5/8 in (1.6 cm) thick. This assembly is 12 ft long and is designed to support a shielded isotope cask weighing 3.5 tons (3175 kg) placed over the specimen removal tube. When the isotope cask is moved over the reactor tank, written procedures pre-approved by the Reactor Safety and Utilization Committee will be followed.

o. Provide an analysis of the effects on the reactor of using evacuated vertical tubes extending to the top or sides of the reflector. Include consequences related to inadvertent flooding.

There are no immediate plans to use evacuated vertical tubes. A detailed safety analysis will be performed and the approval of the Reactor Safety and Utilization Committee will be obtained before such an arrangement is used in the reactor. The following sentences are removed from the SAR.

Evacuated vertical tubes inserted into the reactor and located on the top, or the side, of the reflector may be used to obtain a collimated beam of radiation. Special shielding may be required whenever this is done.

p. (1) Pneumatic transfer system (PTS); Please discuss who will be in control of this system and its samples How is use of the PTS controlled with the potential for reactivity changes if the receiver/sender unit is outside of the reactor room? What organizational group is responsible if the receiver/sender unit is in a location not explicitly covered in the reactor operating license?

(2) Discuss radiological impacts related to PTS use and operation.

The following will be added as the second paragraph to section 3.2.6.2.

Use of the pneumatic transfer system will be under control of the reactor operating staff. A permit switch will be located in the control room to allow or deny use of the pneumatic system. The PTS receiver/sender unit will be located in one of the lab rooms in the reactor building and users will undergo proper training as to use, radiological impact and reactivity effects of samples. During operation, use of the PTS will result in production of Ar-41 gas from the air used in the blower. For this reason, the blower will be vented into the reactor room so that its exhaust will be monitored upon release. If use of the PTS results in excessive Ar-41 production, as measured by the monitors in the reactor building and the building exhaust, then the recirculation method noted in Figure 3.13 will be used. All users of the system will receive training in use and radiation protection. An area radiation monitor will be located near the PTS receiver/sender unit in the lab with both local and remote (to the control room) alarms.

q. Are the control rods new? If not, has there been any "burn-up" of the B-10? If used, discuss the implications to reactor operation and safety.

All the control rods will be new ones obtained from GA, with no burnup for B-10.

r. Section 3.2.7.2, Rod Drive Assemblies; Do the "limit switches" perform any function other than causing lights to indicate positions? Please discuss.

The limit switches, in addition to switching lights that indicate positions, actuate circuits that stop the drive motor at the top and bottom of travel. In the event of a scram the rod DOWN limit switch actuates the circuit to drive the magnet down, unless the UP push button is depressed. The last sentence of the second paragraph will be replaced wit h the following sentence.

Limit switches mounted on the drive assembly actuate circuits to stop the drive motor at the top and bottom of travel, drive the magnet down in the event of a scram, and indicate the following at the console:

s. Figure 3-16; It is suggested this figure should be labeled Rack and Pinion Control Rod Drive.

Will be changed.

t. Administrative controls to limit the transient rod reactivity addition might not be sufficient. A mechanical stop on the transient rod may be more appropriate. Please discuss.

The last sentence in paragraph 2 on page 3.29 will be changed as given below.

The rod may be withdrawn from zero to a maximum of 15 inches from the core; however, a mechanical stop is used during pulsing operation to restrict the travel so that the maximum permissible step insertion of reactivity (1.4 %Δk/k or 2.0 S) will not be exceeded.

u. Section 3.2.9.2, Storage Racks: How many 10 position racks are present in the tank? If all fuel elements must be removed from the core for some reason, how and where will they be moved and stored? Discuss reactivity and shielding conditions of all of the fuel storage facilities, for both irradiated and unirradiated fuel rods.

There are eight racks each capable of holding 10 fuel elements in the tank. Two 0.75 in diameter aluminum rods hold the rack securely in position. These rods are secured to the tank flange. Two racks are attached to the aluminum connecting rods by locating one rack at a different vertical level and offsetting the horizontal position slightly. The design of these racks are standard and similar to the ones used at other facilities. The number and positioning of the fuel in the racks is such that the configuration will remain subcritical. A minimum of 8 ft of water above the racks will be maintained to provide shielding.

Fuel storage facilities are discussed in section 6.2.2 of the SAR. Storage pipes outside the pool are pits in the reactor floor. These pits are fabricated of 10 in diameter stainless steel pipes. They are 12 ft deep, and located 3 ft from the adjacent pipe. Nineteen elements may be stored in each pipe, and water may be added to provide radiation shielding. An element spacing rack will provide an array for the fuel equivalent to the inner most 3 rings of the reactor core (including the central A ring). Locked cover plates on the pipes provide access control. The cover plates may include some shielding. This configuration of 19 fuel elements will remain subcritical (see 15 b). Analysis at the SNRS (McClellan Air Force Base) facility shows that the expected radiation level at floor level from 19 irradiated fuel elements in the pit, after 1000 kW operation and cooling for 24 hours, will be less than 1 mr/hr with the pit flooded or with a shield

plug in place. Without shielding, the radiation level is estimated to be 15 rem/hr.

6. Section 3.3;

   a. A NRC SER is not considered to be an acceptable substitute for a case-specific technical analysis by an applicant for a license.

      The references 3 and 4 will be removed from this chapter and references to them will also be changed.

   b. Pages 3-33; With all of the "operating experience with TRIGA reactors" that you have noted, in addition to the University of Texas analyses, give a specific reference to experimentally verified operation of a 70 element reactor. Cite power levels, peak to average power densities, maximum operating thermal power level, peak fuel temperature, and burn-out ratio. Please provide the reference for the 1250 C phase transition for ZrH.

      Will be addressed later.

   c. What is the power density (watts/gm) corresponding to the 180 ^{o}C and the 265 ^{o}C fuel temperatures?

      The power density is approximately 1.3 W/g. This value is for a core with 90 fuel elements. A typical 70 element core will have a power density of approximately 1.7 W/g. The average and peak temperatures should be corrected to about 200 C and 294 C respectively, for steady state operation. Corrections will be made.

7. Section 3.4;

a. Give the basis for the "neutron lifetime" for your reactor being 41 micro-seconds.

GA-2070 provides a prompt neutron lifetime of 65 μs, the value given in GA-2599 is about 45 μs, and is given as 41 μs in the Texas SAR. The typical value may vary by about 10 % between graphite and water reflected cores.

b. Page 3-34, paragraph 3, sentence 3; Please be more careful of your use of the term "shutdown margin." See the definition in you Technical Specifications.

The sentence will be changed as given below.

With the core maximum excess reactivity of 2.25 %Δk/k, the shutdown margin, the minimum reactivity available to shut down the reactor with the most reactive rod stuck out, is about 0.44 %Δk/k (0.63 $).

c. Page 3-34, last sentence, and Table 3-5; Please give your basis for this table.

Values given in this table will be revised later.

d. Page 3-35, sentence preceding Table 3-6, and the table; This sentence implies that ad hoc estimation of reactivity effects can serve to replace a measurement. Please discuss your basis for that.

This paragraph will be changed as given below.

The estimated reactivity effects associated with the introduction of some of the experiments in the reactor core are given in Table 3-6. These numbers should only be used as a guide. The effects of materials not given here must be investigated before insertion into the reactor core.

8. Section 3.5;

   a. What, specifically, are the relationships between Safety Settings and Safety Limits (Safety Limits are not mentioned)? What is the technical basis for stating that the Safety Settings are "conservative?"

Section 3.5 will be revised as given below.

3.5 SAFETY SETTINGS IN RELATION TO SAFETY LIMITS

As mentioned previously, the main safety consideration in operation of a TRIGA reactor is the fuel temperature. As noted earlier in this chapter, the upper limit for fuel temperature for prevention of fuel cladding failure for standard TRIGA fuel can be taken as 920 C. Thus we take a conservative limit of 800 C as the safety limit for maximum fuel temperature in the core. The operating parameters that can be controlled to effect this limit in a TRIGA reactor are:

a. Maximum licensed steady-state power level
b. Fuel temperature measured by thermocouple
c. Maximum reactivity insertion during pulsing
d. Core inlet coolant water temperature

The safety settings as listed in Table 3-7 are such that in all operations, normal and abnormal, the safety limits indicated in the reactor design bases will not be exceeded.

Table 3.7 ATUR Triga Safety Settings

Parameter	Safety Setting	Function
Maximum steady-state power level	250 kW (t)	Reactor scram
Maximum measured fuel temperature	500 C	Reactor scram

Administrative controls are imposed to limit the excess reactivity, transient conditions and coolant water temperature to insure that the safety limits are not exceeded as follows:

a. Maximum core excess reactivity of 2.25 %Δk/k with a shutdown margin of at least 0.2 %Δk/k with the most reactive control rod fully withdrawn.
b. Maximum reactivity insertion for pulsing operations of 1.4 %Δk/k. (Also limited by mechanical stop in place during pulsing operations.)
c. Core inlet water temperature of 50 C.

These safety settings are conservative in the sense that adherence to them will result in fuel and cladding temperatures well below the safety limits indicated in the reactor design bases. Thus, the safety settings indicated here give a margin of additional safety above that absolutely required for safe operation of the reactor.

b. Please give quantitative analysis, showing relation of temperature at the thermocouple in the scram circuit to the peak fuel temperature in the core. Discuss procedures for ensuring that no fuel temperature reaches 500 C, for both pulse and steady state operation.

Will be addressed later.

c. Second paragraph of Section 3.5; Isn't there a scram on both temperature and power? If so, suggested wording might be: "... and if either 250 kW or 500 C is exceeded, the reactor will scram."

Please see revised section 3.5 given in 8.a.

d. Please summarize in Section 3.5 the quantitative margins between these set-limits and the values of the corresponding parameters when you consider the hazards to be "significant," and discuss the bases.

Please see revised section 3.5 given in 8.a.

9. Section 4.1;

a. Table 4.1; Are the set points scrams, interlocks, or some other action

Table 4-1 will be changed as follows. The set point on the wide range log power channel is an interlock. The set points on the percent power channel #1 and #2 are scrams.

Table 4-1 Operating Ranges and Set Points for Neutron Channels

Channel	Detector	Range	Set points
Wide range log power	Fission	< source level to 250 kW	2 cps Interlock
Multirange linear power	Fission (same as above)	source level to 250 kW	none
Percent Power #1	Ion	1% to 110%	100% (250 kW) Scram
Percent power #2	Ion	1 % to 110 %	100% (250 kW) Scram

b. Table 4.1 and Section 4.2.3

Power level set points should not exceed the licensed power level of the reactor. Either change the percent power set points to "100% or less" or increase the licensed power level to 275 kW(t). If power level is increased, please analyze the increased power level in the SAR.

The scram set points on the percent power channels will be changed to 100 % (250 kW). This will be done in Table 4-1, section 4.2.3, and section 4.4. Section 4.4 will be revised as given below.

4.4. REACTOR SAFETY SYSTEM

The reactor safety system prevents reactor conditions from deviating beyond safe limits and mitigates the consequences if the safe limits are exceeded. A reactor

protective action interrupts magnet current to the safety-shim and regulating rods and releases air pressure to the safety-transient rod resulting in the immediate insertion of all rods. All scram conditions are indicated automatically by the annunciators located in the control console. Appropriate checks, tests, and calibrations are provided to verify the operability and satisfactory performance of the scram functions. The following conditions will result in the immediate insertion of all rods. These represent the minimum safety channels required for operation.

1. Power on one of the two percent power (safety) channel exceeds 100% of the full power (250 kW) during steady state operation.
2. Peak power or energy on one percent power channel exceeds 100% of peak power during pulsing operation.
3. High voltage power supply to the fission or ion chambers is le 90% of the normal operating voltage.
4. Fuel temperature measured by one of the two thermocouples is greater than 500 C during steady state or pulsing operation.
5. Loss of magnet current.
6. Loss of console power.
7. Manual scram.
8. External scram (not required for operation).
9. Minimum Period (not required for operation, available for use as desired).

The following conditions cause an alarm which is visually and aurally annunciated at the console. Manual scram may be initiated if necessary.

1. Bulk pool temperature above 50 C.
2. Pool level not within 1.0 ft of normal operating level.
3. Primary pump pressure less than 90% of normal operating pressure which initiates pump trip.
4. High radiation level (see section 10).

The sentence " The reactor may be operated at power levels not to exceed 275 kW during short periods for test or calibration. " will be removed from page 2-4 of Technical Specifications. Table on page 3-6 of the Technical specifications will be corrected as given below.

Safety Channel	Number Operable	Function	Effective Mode Steady-State	Pulse
1. Manual Scram Bar	1	Scram on operator demand	X	X
2. Fuel Temperature	2	Scram at 500 C	X	X
3. Percent Power Level	2	Scram at 100% of full power	X	
4. Percent Power Level (Peak Pulse Power)	1	Scram at 100% of peak power		X
5. High Voltage	1	Scram on loss of	X	X
6. Magnet Current	1	Scram on loss of	X	X

c. Control console, Please provide information on type, model number, year of initial operation and history of use, including any modifications, improvements, and refurbishments.

This a typical desk type console (T91634, drawing number EL0349) which was used by GA at their 250 kW reactor facility. After refurbishments by GA the console was shipped to ATU in 1991. Currently the console is stored at ATU. The console and control systems will be installed and tested by GA before releasing to ATU.

d. Page 4-2, first paragraph; If water temperatures are read through a selector switch, please explain how the "core inlet coolant water temperature below 50 C" is ensured at all times.

Bulk pool temperature probe is connected directly to the alarm circuit. Figures 4.2 and 4.3 will be modified to show this.

10. Section 4.2;

a. Discuss whether the reactor period signal from the Wide Range Log Power Channel is used to provide a reactor scram signal.

Reactor period is also derived from the log power channel and is used to provide a non-required reactor scram signal. This period scram circuit may be bypassed as set forth in SOP's

b. Section 4.2.3; It is stated that the two safety channels are completely independent. Do they operate from independent high voltage (ion-collecting voltage) supplies? Please discuss.

Yes, the high voltage to the two ion chambers providing signals to the two percent power channels comes from the left and right hand drawers. The scram signals come from these two independent channels. Therefore, the two safety channels are completely independent.

c. Section 4.2.5; Two temperature scram channels are discussed. Please discuss how both are correlated to the peak fuel temperature in the core. How are power density and fuel temperature distributions within individual fuel rods, including the instrumented elements, accounted for between pulse conditions and steady state conditions? By what criteria do you determine which of the two temperature channels is the Technical Specification LSSS?

Will be addressed later.

11. Section 4.3;

a. Are all of the control/safety rods scramable?

Yes, all control/safety rods are scramable.

b. Please discuss the use of the percent power channel in Transient Mode. How is linearity ensured, and how is the channel calibrated?

In the Transient mode only the percent power channel in the right hand drawer (#2) is used. It provides signals for the peak power (nv) and the integrated power (nvt) circuits. In addition a scram signal is generated at the Peak power or energy during pulsing. The % power linear amplifier (ELC 266-2120, GA) provides a 10 V output for detector currents between 10^{-6} A and 10^{-3} A with 1/2% accuracy. Test and calibration signals provided at the console ensure the operability of the channel. An external calibrated current source is used to substitute the detector current signal to perform an absolute current calibration.

12. Section 4.4:

a. Last paragraph; Reference is made to pump pressure less than 90%. What pump? Please describe.

The pump referred to is the one in the primary system (figure 5.1). The sentence will be changed as follows.

3. Primary pump pressure less than 90% of normal operating pressure which initiates pump trip.

b. Please provide the basis for the alarms listed in this section.

1. Bulk pool temperature above 50 C. The temperature rise of the coolant through the core is roughly 30 C at 250 kW. Assuming the core inlet temperature to be the same as the bulk pool temperature of 50 C the outlet temperature would be 80 C. This provides a margin of 33 C (saturation temperature of water at 23.4 psia is 113 C) to prevent film boiling. (Tech Spec. LCO 3.3.1.a)

2. Pool level not within 0.5 ft (to be changed to 1.0 ft) of normal operating level. This alarm signals any water leak into or out of the reactor tank. This is also an indirect indication of leak into or out of the primary system. (Tech Spec. LCO 3.3.1.d.). This requirement would also ensure Tech Spec. LCO 3.3.1.c is satisfied.

3. Primary pump pressure less than 90% of the operating pressure which initiates a pump trip. This would indicate a break in the primary loop and the pump will shut down to minimize effects of the leak and damage to the pump.

4. High Radiation level. The alarm, when the radiation level is above normal levels, would warn operating personnel to radiation hazards. Steps would be initiated to minimize personnel exposure and radioactivity release .

13. Chapter 4, references; It is not indicated in the text where the various references apply. Please address this comment.

These are general references that provided the information for this chapter.

14. Chapter 5:

a. Section 5; Does the statement about the cooling system being "above grade" mean that all parts of it are at an elevation higher than the surface of the water in the reactor tank? Please discuss.

The water cooling and purification systems are located on the south wall of the reactor room above the level of the reactor pool.

b. Section 5.1; You have N-16 produced by a (n,β) reaction instead of (n,p). Please correct.

Will be corrected.

c. Section 5.1, page 5-3; Is the 1 psi pressure differential independent of operation of the primary coolant system pump? Please discuss, including radiological implications in the event of a water leak between primary and secondary systems within the heat exchanger.

The 1 psi pressure differential applies only when the primary coolant pump is operating. The heat exchanger and cooling system is designed to produce a 1 psi differential between primary and secondary systems. The secondary system is a closed system and so any such leakage would be contained within the facility and that periodic checks (monthly) will be made of the secondary water to insure that there are no leaks.

d. Section 5.3; How is back flow from the pool to the city water system prevented under all possible water pressure conditions? What precautions are taken to assure no primary or secondary water enter the city or campus water supply?

The pool is located below grade. The primary and the secondary systems are isolated from the city water system. There are no pipes connected directly to the city water system from either the primary or the secondary systems. Make up water is added to the primary system by a flexible hose which is disconnected when it is not in use. Make up water for the chilled water system is taken from a small reservoir which is isolated from the city water system. This reservoir is replenished as needed manually. Check valves may be installed wherever a backflow is possible.

e. An inadvertent leaking of the pool water down to the siphon break would result in about 950 gallons lost. Assuming this water contains the maximum calculated radio-nuclide level resulting from prolonged operation at the maximum licensed power level:

(1) What precautions, if any, are taken to ensure this water is not released to the unrestricted area? Assess radiological consequences to restricted area personnel.

(2) If this water is allowed to enter the unrestricted area (sewer or storm drain or etc.), assess the potential dose consequences to personnel in the unrestricted area.

The potential for loss of reactor cooling water to an unrestricted area is minimal. The reactor pool is located at the lowest elevation in the reactor room and thus any leakage from the cooling/purification system will tend to drain back into the pool. There are no floor drains leading to an unrestricted area (ie. sewer or storm drain) in

the reactor room. Due to the high purity of the coolant water, the levels of radionuclides in the pool water, even after extended operations at full power, would be rather low. There is no conceivable pathway for water leaking from the reactor pool or cooling system to enter a drinking water system prior to massive dilution in the Arkansas river

f. How often is the radio-nuclide level ir the primary coolant system (PCS) checked and compared with 10 CFR Part 20 allowable concentration for release to the environment? What is the maximum allowed electrical conductivity level in the PCS?

Radionuclide concentrations in the primary coolant system are checked each day of reactor operation as a means of detecting water impurities and/or fuel element leakage. Pool water conductivity will also be measured each day of operation and the upper limit of 5 micromhos/cm should be sufficient to maintain low concentrations of radionuclides in the pool water.

g. How is the pool water level determined?

The pool level is determined using a float meter. The float generates indication for the water level in the pool for ± 1.0 m of the normal operating level. Alarms are also generated if the water level is 1.0 ft above or below the normal operating level.

15. Chapter 6;

a. Figure 6.1; Please show and define the "restricted area" as defined in 10 CFR Part 20, and the "reactor facility" to which the reactor operating license will apply.

Restricted area as defined in 10 CFR Part 20, (Any area access to which is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.), is the "Reactor Building " shown in Figure 6-1 of the SAR. The reactor facility to which the reactor operating licence will apply is also the same as the "Reactor Building".

b. Section 6.2.2, Fuel storage; Discuss K_{eff} for fuel storage pits containing 19 elements, both dry and water flooded. How does one transfer the fuel from the reactor; discuss procedures.

The five fuel storage pits are similar in dimensions and configuration to the storage pits at SNRS, McClellan Air Force Base. Analysis at the SNRS facility of the storage pits shows that the largest k_{eff} for pit containing 19 fuel elements is approximately 0.75 fully flooded and 0.45 when dry when all five pits are fully loaded. The spacing between storage pits is sufficient to stop any neutrons escaping from one pit from entering another and thus k_{eff} will be the same for each pit regardless of whether the other pits contain fuel or not.

Procedures will be developed for transfer of fuel from the reactor to these storage pits which will address safety both of the personnel performing the transfer and of the fuel. Movement of fuel elements to and from these pits will utilize the element handling tool. Only un-irradiated fuel will be stored in the pits in a dry state.

c. Section 6.2.3, Ventilation system; please discuss:

(1) Fail-safety features in case of loss of electricity.

(2) Normal configuration; where is the fresh air intake relative to exit from the exhaust stack?

(3) Automatic features causing dampers to close and isolation to be achieved. What functions are changed by monitor response?

(4) Are "sealed doors" and windows closed during operation?

(5) Discuss the actions, automatic and other wise, of the ventilation system in the event discussed in Section 7.3 of the SAR.

6) Must air from the Control Room and Room 3 enter the Reactor Room to be exhausted? What is the pressure difference between the Reactor Room and the rest of the building?

(7) What is the air flow path in the Control Room 3, and the Reactor Room? Is there any time when reactor room air is forced or circulate into any other room in the reactor building?

(8) Under what circumstances is air routed through the HEPA filter? Is switch over to the HEPA system automatic?

(9) Do all facility fans shutdown upon Reactor Room isolation?

Section 6.2.3 will be revised addressing these questions as given below.

6.2.3 Ventilation System

The ventilation system is designed to supply fresh, conditioned air to the control and reactor rooms; to remove normal release of radioactive gaseous effluents, to filter effluents through a high efficiency filter when necessary; and to isolate the reactor room in the event of abnormal levels of radioactive material.

The ventilation system for the reactor building consists of two parts. The first part is standard HVAC equipment utilizing both fresh and re-circulated air for all areas of the building except for the control and reactor rooms. The second part of the system branches from the primary system to supply fresh air to the control and reactor rooms. A common cooling and heating plant conditions air for the entire building. Supply ducts extend from the utility/storage and furnace rooms (rooms 5A and 5 in figure 6-1) around the outer rooms of the building ending in the control room with a single duct extending into the reactor room. These ducts will be set to supply approximately 150 cfm of conditioned air to the control room and approximately 500 cfm to the reactor room. A separate duct system extending from the control room to the reactor room will allow the lower air pressure in the reactor room to draw air from the control room. Recirculation ducts take air from all areas of the building other than the control and reactor rooms. No recirculation ducting extends into the control or reactor room.

An exhaust fan is mounted at roughly the center of the south wall of the reactor room approximately 7 m from ground level. This fan exhaust approximately 850 cfm of air from the reactor room to the outside. If conditions warrant, such as high levels of radioactive particulates in the reactor room air, this air can be routed, by manual controls, through a HEPA filter located in the duct work leading to the

exhaust fan. Each of the ducts leading into (air supply) or out of (exhaust fan) the reactor room contain a remote control damper which will automatically close upon the following conditions:

1. Loss of electrical power to the ventilation system.
2. Loss of power to exhaust fan.
3. Alarm on exhaust particulate radiation monitor
4. High level alarm on exhaust Ar-41 monitor
5. Actuation of isolation system by reactor operator

The building ventilation system will be required to supply roughly 850 cfm of make-up air from outside the facility. The fresh air intake that supplies this air is located in the south wall of the utility/supply room (room 5A) and thus at the southwest corner of the reactor building at approximately ground level. Due to the large dilutions expected, even in the lee of the building, the separation between the reactor room exhaust and building fresh air intake should be sufficient to prevent any significant recirculation of exhausted reactor room air through the rest of the building.

The operation of the air supply and exhaust systems will result in a negative pressure differential between the reactor room and both the building exterior and other portions of the reactor building. This negative pressure differential will persist, although at a lower magnitude, in the event that interior doors, such as between the control room and reactor room, are left open. Normal operating procedure will call for both doors into the control room to remain closed except for personnel passage. The roll-up door at the west end of the reactor building is too large for the ventilation system to maintain negative pressure if it is fully open. Procedures will be developed to restrict the use of this door during reactor operation. The remote dampers mentioned previously will also be able to be controlled manually from a panel in the control room which will also include readouts from the monitoring system. In addition, a similar control panel will be installed in the Emergency Support Center described in the facility Emergency Plan.

Control of exhaust air from the pneumatic transfer system will be accomplished by ducting surrounding the PTS piping and the receiver/sender station in Lab 3. This ducting will be connected to a small blower located in, and exhausting to, the reactor room so that all air, both from operation and leakage, from the PTS system will also exhaust the building through the reactor room exhaust fan and monitors.

d. Section 6.2.4; Discuss the Ar-41 monitor in more detail. Where is it located, how is it calibrated, and what functions does it perform? Justify not listing it in the Technical Specifications.

The Ar-41 monitor is based on a NaI detector. Air from the roof stack is routed through a sampling chamber which houses the NaI detector. The readout will be displayed in the control room. The calibration is performed using standard calibrated Ar-41 source annually. The system will display Ar-41 radiation levels in the reactor room air as well as provide alarms when preset levels are exceeded.

The following will be included in the Technical specifications 3.3.3.

c. A continuous Ar-41 monitor (NaI) shall be operable with readout and audible alarm.

The reference to the Ar-41 monitor in this section will be removed. Since the exhaust fan will be shutdown on loss of electricity, there will be no need for an Ar-41 monitor. The first sentence in paragraph 2 will be corrected as given below.

Power from a battery system will supply the intrusion alarm and radiation monitors (area monitor, particulate monitor) under emergency conditions for about 15 hours.

e. Section 6.3.1; Radioactive Waste

(1) Describe your planned facilities to collect and store liquid radioactive waste pending release. Discuss which waste drains lead to installed storage facilities, and how inadvertent or uncontrolled release of radioactive liquid to the public sanitary sewer system is prevented.

There are no drains from the reactor room and the control room. Drains from the toilet, utility, furnace, and storage rooms (4, 5, 5A of Figure 6-1) are connected to the sewer system. Storage or handling of the radioactive material in these will be prevented through administrative controls. There are two sinks in room 2. A regular sink drains to the sewer system. Disposal of hot waste into this sink will be prevented through administrative controls. The second sink is for hot waste. A catch basin under this sink provides storage for the hot waste until it is poured manually into the waste storage tanks in room 3. In room 3 there is a sink for hot waste which drains into two 30 gallon polyethylene storage tanks. The drains on these tanks are

secured with locked ball valves. Water level in the storage tanks will be checked at least monthly or more frequently as needed. Samples will be drawn from this tank for analysis. The liquid waste will be disposed off into sewer drains only if approved by and under the supervision of health physics personnel. The shower in room 3 is a portable shower with a catch basin and is completely self-contained.

(2) Discuss plans and provided equipment to assess radioactive content of liquids before release. What are the criteria to be used in determining if a release will comply with applicable regulations?

Samples from the waste holding containers will be analyzed for radioactivity by health physics personnel. Samples will be analyzed for isotopic content by gamma ray spectroscopy using a solid state detector and will also be analyzed for gross beta activity. The waste will be disposed off according to criteria specified in 10 CFR part 20 or as appropriate.

(3) In accordance with the Nuclear Waste Policy Act of 1982, do you have a written agreement with DOE that they will accept all used fuel for reprocessing or other disposition?

Will be addressed separately.

f. Section 6.3.2, Counting Laboratory; Please describe and discuss the disposition of air-borne and liquid radioactive materials used in this Laboratory including potential effluents to the unrestricted environment.

An exhaust duct surrounds the PTS receiver/sender (in room 3) and tubes and all gaseous products are thus vented to the reactor room. The exhaust stack monitor warns if contaminated effluents are released to the unrestricted environment. Air-borne and liquid radioactive samples will be in closed containers while in the counting laboratory. After counting the samples will be disposed of in accordance with approved disposal procedures. Samples will be handled in an air hood which will be vented into the reactor room. (See also answer to 15.e.1)

g. Section 6.4.1, Argon;

Section 6.4 of the SAR will be revised as shown in Attachment #1.

(1) Please give the projected annual dose to the most highly exposed individual in the unrestricted area from all sources of Ar-41. Describe the actual location and elevation of the exhaust stack and provide a quantitative estimate of the factor by which projected doses are over-estimated. Justify the arguments.

The exhaust stack is located in the middle of the reactor building at roof level and at an elevation of 7. m above ground level (please see 15.c). The newly added section 6.4.3 of the SAR (Attachment #1) addresses this question.

(2) Please give the projected annual dose at the site of the nearest permanent residence and the nearest temporary residence (dormitories, e.g.) in the unrestricted area. Show methods and details in going from reactor room concentration to annual doses. Provide an estimate of the factor by which the projected doses are over-estimated, and justify it.

Please see the newly added section 6.4.3 of the SAR (Attachment #1).

(3) Please state the assumptions that entered into your calculations of releases from experimental facilities. What are the experimental facility exhaust paths?

Please see the revised section 6.4.1.2 of the SAR (Attachment #1).

The exhaust path for the central thimble is through the top of the tube. For the rotary specimen rack the exhaust path is through the drive shaft tube and the specimen loading tube. For the pneumatic transfer system the exhaust path is through the receiver/sender section and the blower exhaust . For the pneumatic system the exhaust occurs mainly during its operation for small durations.

(4) At the bottom of page 6-13 it is mentioned that air from experimental facilities may be filtered... "to further reduce Argon-41 activities." Please discuss the plans, including methods and equipment.

This statement, "Air from experimental facilities may be filtered if necessary to further reduce the argon-41 concentration levels." is wrong and is removed from the SAR. Please see the revised section 6.4.1.2 of the SAR (Attachment #1).

(5) Please give the maximum projected annual dose to reactor operations personnel in the restricted area.

Please see section 6.4.3 of the SAR (Attachment #1).

(6) Please justify your factor of 25 reduction in the Ar-41 level at the bottom of page 6-13.

Please see section 6.4.3 of the SAR. (Attachment #1).

(7) Discuss the operating principles of the pneumatic transport system, including the formation and release of Ar-41 in neutron-irradiated air.

The pneumatic transfer system is discussed in section 3.2.6.2. Formation and release of argon-41 in neutron irradiated air from all facilities, including the pneumatic transfer system is discussed in section 6.4.1.2. Argon-41 produced in PTS is carried into room 3 when the system is operated. The air from the pneumatic transfer system is exhausted through the exhaust stack in the reactor room.

(8) It is understood that the University of Texas is revising the material of your reference number 1. Please update your submittal as necessary.

This revision is already incorporated into the ATU SAR.

h. Section 6.4.2, Nitrogen-16; Please discuss the expected dilution factor of nitrogen-16 by the diffuser system at the ATU facility. Provide the bases.

Please see revised section 6.4.2 with new values for the reactor room volume and average flux.

Without the diffuser the transit time from the core to the pool surface is 109.5 s. In this time the nitrogen-16 decays by a factor of 3.6×10^{-5}. The discharge from the diffuser is 9464 cm^3/s (150 gpm). This flow will spread the convective column of nitrogen-16 bearing water from the core into the cross sectional area (72966 cm^2) of the pool. The discharge from the core is about 2205 cm^3/s. If we assume that this core discharge (neglecting any other the effect of the diffuser discharge on the vertical flow rate) is spread into only 1% of the pool cross sectional area and then rises up through this cross section, the effective upward velocity would be approximately 3.02 cm/s (2205 $cm^3/s \div 729.7\ cm^2$). The transit time for the water to reach the surface of the pool is 212 s (640 ÷ 3.02 cm/s). In this time the

nitrogen-16 decays by a factor of 2.5×10^{-9}. This estimate shows that the diffuser reduces the nitrogen-16 concentration by about 4 orders of magnitude.

i. Section 6.4.2, page 6-15, paragraph 1; It is stated that water of conductivity = 2 micromhos affects nitrogen-16 chemical reactions in certain ways. Will your Technical Specifications require this conductivity during operations? If not, what is the effect on the assumptions in this section of the SAR of the projected conductivity?

If the water conductivity is approximately 2 μmhos most of the nitrogen-16 formed will combine with oxygen and hydrogen atoms of the water. The ones that combine in anion form, which makes up almost half of ions, will remain in water and will not be released. This factor will further reduce the nitrogen-16 concentration released into the reactor room. To obtain conservative estimates this factor is not taken into account in the calculations. The reduction in the concentration from the decay of the nitrogen-16 during transport to the surface is more significant. Since the dose rates resulting from the nitrogen-16 without taking into account the above conductivity effect is low enough, the Technical specifications does not require this conductivity during operations. However, low values of conductivity should be used if possible to meet ALARA requirements. If the conductivity effect is included the resulting dose rate from nitrogen-16 would be reduced by at least a factor of two.

j. Section 6.4.2, last paragraph; Please justify why a thin disk on the pool surface is an adequate representation of the nitrogen-16 distribution in the pool water, and give a more quantitative discussion of the last sentence, about transport times and "substantial" dose reductions.

It was assumed that the pool water containing the nitrogen-16 moved to the surface from the core region with a velocity of 5.85 cm/s and then spread into a thin disk. Assuming a radius of 125 cm for this disk the thickness is obtained as 0.97 cm
($2205\ cm^3/s \times 21.36\ s \div 49087\ cm^2$). The dose rate due to this thin disk at the pool surface was obtained as 670 μrad/hr (6.20). Larger values for the thickness of the disk will increase the dose rate since $E_2(\mu h)$ in equation 6.20 will become smaller. However, the decay of nitrogen-16 during the formation of a larger thickness disk will reduce the average concentration of nitrogen-16, N (given by 6.16), in the water. For the scenario described in 15.h, the flow velocity is 3.02 cm/s, the disk thickness is 1.86 cm and the dose rate at the pool surface reduces to 493 μrad/hr due to the combined effect. In addi-

tion, as shown in 15.h the decay of nitrogen-16 during the transit will be more significant. The effect of a larger disk would not be significant.

For the scenario described in 15.h, when all the above factors are taken into account the dose rate due to nitrogen-16 in the reactor room air (6.19) reduces to 1.6×10^{-11} rad/hr and the dose rate at the pool surface (6.20) reduces to 3.4×10^{-4} μrad/hr. Another way to look at this is given below. The transport time of 109.5 s was in the SAR. A 50% increase in the transport time will reduce the dose rate due to nitrogen-16 in the reactor room air (6.19) to 1.7×10^{-9} rad/hr and the dose rate at the pool surface (6.20) to 3.4 μrad/hr; both less than 1 %.

16. Chapter 7;

a. Section 7.1.1; The discussion of Safety Limits gives certain values for stainless steel rupture pressures. When you are reasonably certain of which fuel you will be using, please address what effect the history of the ATU fuel will have on such nominal parameters, and on the Safety limits for the proposed ATU fuel.

The fuel to be used for the initial core loading will consist of elements having burnups ranging from a fraction of a MW-day to roughly 4 MW-days. Due to the relatively low neutron fluence levels and temperatures experienced in TRIGA reactors, the effects of this prior exposure on the fuel cladding will be minimal. Therefore, it is assumed that the rupture pressures quoted for TRIGA fuel cladding will still be applicable for the fuel to be used in the ATUTR.

b. Section 7.1.2; Please provide the references for the equations and the calculational approach used to compute P_h, P_{fp}, P_{air}.

The reference for p_h is 3 in this chapter, GA-8129

The equation for P_{fp} is ideal gas law, $p_{fp} = nRT_K/V$, where R is the molar or universal gas constant. The number of moles of fission product gases, n, is given by f(n/E)E. The value for f is obtained from reference 4 (in the SAR), n/E = 0.00119 moles/MW-day is obtained from reference 5, and E was assumed to be three times the burnup for a standard TRIGA fuel element, viz 4.5 MW-days/element. The free volume occupied by gases is given by equation 7.9b

The equation for p_{air} is the ideal gas law, $p_{air} = RT_K/V_m$, where the molar volume of an ideal gas (air) at STP, V_m is equal to 22.4 liters/mole

c. Section 7.1.2, page 7-4; Please justify the statements made in the last two complete sentences on this page. Give primary references.

Total number of fissions required for 1 MW-day of energy generation is 2.7×10^{21}. About 27 atoms of fission product gases will be produced per 100 atoms fissioned [W.B. Lewis, Nucl. Appl., 2, 171 (1966)]. This translates to 0.729×10^{21} fission product gas atoms per MW-day or 0.00121 moles per MW-day. The value used in the SAR is within 2% of the above number.

d. Section 7.1.2, page 7-5; Please relate the assumed burn-up of "standard TRIGA fuel" to the history of the projected ATU fuel.

The burnup used for this calculation is a conservatively high value that will yield a conservative estimate for the pressure exerted by fission product gases. The maximum burnup that can be obtained from a standard TRIGA fuel is about 4.5 MW-days/element. Three times this value is used in the calculation. The resulting fission product pressure will be higher than that will be present in any fuel element that can be used in this reactor.

e. Section 7.1.2, page 7-5, next to last paragraph; Please cross reference the Technical Specification implementation of the not-immersed-fuel Safety Limit. Please discuss the Safety Limits of not-immersed fuel in terms of the pressure vs temperature diagram.

The temperatures 950 C and 920 C are for not-immersed fuel. The clad and fuel are assumed to be at the same temperature. The Technical Specification in 2.1 is changed to "The maximum temperature in a standard TRIGA fuel element shall not exceed 920 C when not immersed in water and 1150C when immersed in water."

TRIGA fuel with H/Zr equal to at least 1.65 has been pulsed to temperatures of 1150 C without damage to the clad. [Reference: Dee J.B., et. al., "Annular Core Pulse Reactor," General Dynamics, General Atomic Division Report GACD 6977 (supplement 2), 1966.] The peak adiabatic fuel temperature of 1000 C is possible since even at these temperatures the clad temperature is well below 500 C [Reference: "Safety Analysis Report, TRIGA Reactor Facility, Nuclear Engineering Teaching Laboratory, The University of Texas At Austin", November 1984.] At 500 C the ultimate strength of the clad is about 60000 psi (Figure 7-1) and at 1000 C the stress on the clad is about 46000 psi (equation 7.12). Therefore, the clad will not be ruptured at these temperatures. Calculations in the above reference also shows that peak fuel temperatures of 1250 C is possible as long as the clad temperature remains below 500 C.

f. Section 7.1.2, page 7-5, last paragraph; Please justify the values given for fuel temperatures for $\%\Delta k/k = 2.25\%$. Give primary references.

The value for the average fuel temperature is obtained from equation 7.5 and the peak fuel temperature is obtained from equation 7.7. The procedures are described on page 7-3. A pulse of $1.98 at

Michigan State University TRIGA reactor on January 26, 1984 resulted in a fuel temperature of 250 C as shown by the fuel temperature meter.

g. Section 7.2, page 7-7;

(1) Discuss the uncertainties in the results plotted in figure 7-2.

Will be addressed separately.

(2) Address whether the calculations are directly applicable to the proposed ATU fuel. For example, are the internal gap situations the same?

The ATU fuel is expected to be a standard non-gapped Triga fuel with 8.5 wt % Uranium, Hydrogen-to-Zirconium ratio of 1.65, and the clad is 0.02 in thick Stainless Steel 304. The Grid plates are standard grid plates designed by GA. Fuel and channel dimensions used are all typical values. We expect all calculations to be directly applicable. More information about the fuel (expected to be Berkeley fuel) will be given separately.

(3) Justify the statements and values given in the last two sentences of paragraph 1.

The last three statements are replaced with the following.

The values shown in Figure 7-2 are generated with the assumption that the core and its surroundings are at a temperature of 27 C when the core is uncovered. If the postulated coolant loss were to occur during reactor operations, the fuel may initially be at a higher temperature than 27 C, and so the results of the figure would not be as accurate. However, due to the design of the ATUTR, there is no credible process that would lead to loss of the pool water and uncovering of the core that would not take several minutes for the water to exit the pool. Since the reactor would be scrammed upon discovery of such a leak (following the low pool level alarm), then the fuel would cool in water for these several minutes prior to uncovering, thus allowing the fuel temperature to drop considerably from operating temperatures.

The following is added at the end of the first paragraph of Section 7.2.1.

If the full temperature difference between maximum operating temperature and that used to generate Figure 7-2 is assumed to be added to that predicted, the predicted temperature would only increase to 521 C. Referring to Figure 7-1, it can be seen that this temperature is still well below that required to cause clad rupture.

(4) On page 7-7, that peaking factor is 2. On page 7-3, Section 7.2.1, the peaking factor is 3.1. Explain the difference.

The overall core power peaking factor in the reactor is 2 (radial 1.6, axial 1.25). This value is used to obtain the maximum power density in an element at full power. This number is multiplied with 1.1 to take into account any uncertainties to obtain 2.2. The number 3.1 is obtained by multiplying 2.2 with the element peaking factor of 1.4. During a puise the maximum temperature occurs at the periphery of the fuel. This number 3.1 will yield the absolute peak temperature in any element during the pulse and is a pulse peaking factor.

h. Section 7.2.2, page 7-9, paragraph 1: Compare the thickness and material of the ATU reactor building roof with the assumed "thick concrete." Give a reasonable estimate of the factor by which the assumption over-computes the likely dose rate due to scattered radiation.

The dose in the reactor room due to scattered radiation was calculated assuming a thick concrete roof. The roof of the reactor room is composed of 5/8" sheet of gypsum board, 6" of insulation, 3/4" sheet of plywood, and 24 gage steel sheet with 1 mill of zinc coating. Neglecting the insulation the average density of the 1.4 "(3.556 cm) thick roof material is about 0.15 g/cm^3. The attenuation coefficient for gypsum and plywood is approximated to that for concrete and water respectively. The combined linear attenuation coefficient is about 0.042 cm^{-1} for 1 MeV and about 0.222 cm^{-1} for the 0.2 MeV (scattered) photons. For a thick vs actual roof ρ, μ_0, and μ_1 are the only parameters that change in equations 7.17 and 7.18 in the SAR. The quantity $\frac{\rho}{\mu_0 + \mu_1\left(\frac{\cos\theta_0}{\cos\theta_1}\right)}$ is equal to 4.9 for a concrete roof and is equal to 2.25 for the actual roof. Thus the factor by which the thick-concrete roof assumption over-computes the dose rate is at least 2.

i. Section 7.2.2, page 7-9, paragraph 2: Give reasonable estimates of the "optimistic" and "conservative" factors, and then lumped together, the net effect on the calculated dose rates. Discuss.

The equation 7.23 in the SAR for the solid angle will be replaced with the following from "Measurement and Detection of Radiation by Tsoulfanidis". This is more appropriate for a Disk source of radius R_s parallel to a circular aperture of Radius R_d and located at a distance d.

$$\Omega = \frac{\omega^2}{4}\left[1 - \frac{3}{4}\left(\psi^2 + \omega^2\right) + \frac{15}{8}\left(\frac{\psi^4 + \omega^4}{3} + \psi^2\omega^2\right)\right]$$

$$- \frac{35}{64}\omega^2\left(\frac{\psi^6 + \omega^6}{4} + \frac{3}{2}\psi^2\omega^2\left(\psi^2 + \omega^2\right)\right)$$

where

$$\psi = \frac{R_s}{d} \text{ and } \omega = \frac{R_d}{d}$$

where R_s is equal to the core radius (22.8 cm, same as r_c to be corrected in SAR), R_d is equal to the pool radius (150 cm), and d is equal to the distance from the core to the top of the pool (640 cm). With these corrections the data in Table 7.1 given in the SAR will change as given in the following corrected Table 7-1. This gives radiation dose rates from fission product decay gammas only.

The radiation resulting from the activation of the aluminum (6061) top and bottom grid plates, the reflector and Rotary specimen rack assembly and the steel (SS 304) in top end pieces and the steel clad of the fuel element and the steel in the Rotary specimen rack were calculated. Flux values for TRIGA Mark I from GA4361 and data from Tables for Neutron Activation Analysis by Michael D. Glascock, 1988 were used along with standard activation equations. The following Table 7-1A provides results from the calculations, which includes gamma radiation from fission product decay as well as activation of structural materials.

Table 7-1 Radiation Dose Rates for Loss of Shield Water (Fission Product Decay Gamma only)

	Radiation (rad/hr)			
	Direct		Scattered	
Operation Time→ / Decay time ↓	10 hr	1000 hr	10 hr	1000 hr
1 minute	665.0	819.0	0.241	0.297
10 minutes	326.0	479.0	0.118	0.174
1 hour	154.6	304.0	0.0561	0.110
1 day	14.5	114.0	0.00525	0.0412
1 week	1.68	46.9	0.00061	0.0170
1 month	0.3	17.4	0.00011	0.0063

Table 7-1A Radiation Dose Rates for Loss of Shield Water (Fission Product Decay and Activation of Strucural Materials)

	Radiation (rad/hr)			
	Direct		Scattered	
Operation Time→ / Decay time ↓	10 hr	1000 hr	10 hr	1000 hr
1 minute	1570.0	1730.0	0.569	0.626
10 minutes	383.0	538.0	0.139	0.195
1 hour	155.0	306.0	0.0564	0.111
1 day	14.6	115.0	0.00530	0.0416
1 week	1.69	47.4	0.000612	0.0172
1 month	0.31	17.8	0.00011	0.0065

The top grid plate and the top end pieces of the fuel elements provide shielding. The top grid plate is 0.75" aluminum with holes for the fuel elements. The volume of aluminum is about 1658 cm. The top

end pieces are made of SS-304 with an approximate volume of 2147 cm3. The effective thickness of 19.44" plate is 1.986 cm. The effective linear attenuation coefficient of the plate is about 0.34 cm^{-1}. The source from the fission product decay, the bottom grid plate, and the steel clad gets attenuated by this palate. The attenuation factor is $e^{0.34 \times 1.986} = 0.51$. Table 7-1B shows the dose rates when the effect of shielding is taken into account. This Table will be a final replacement for Table 7.1 in the SAR and the sentences "No accounting was made of sources other than fission product decay gammas (i.e., activation gammas from the steel cladding and aluminum grid plates) or attenuation through the fuel element end pieces and the upper grid plate. The first of these assumptions is optimistic, the second conservative; the net effect is conservative." will be replaced with the following.

Sources other than fission product decay gammas (i.e., activation gammas from the steel cladding and aluminum grid plates) and attenuation through the fuel element end pieces and the upper grid plate were taken into account in the following calculation.

Table 7-1B Radiation Dose Rates for Loss of Shield Water (Fission Product Decay and Activation of Strucural Materials with shielding)

	Radiation (rad/hr)			
	Direct		Scattered	
Operation Time→ / Decay time ↓	10 hr	1000 hr	10 hr	1000 hr
1 minute	1240.0	1320.0	0.451	0.480
10 minutes	223.0	303.0	0.081	0.110
1 hour	79.5	157.0	0.0288	0.057
1 day	7.55	58.9	0.00274	0.0214
1 week	0.863	24.4	0.00031	0.0088
1 month	0.157	9.2	0.00006	0.0033

After 1600 hours of operation and a decay time of 1 minute the effect of including the structural activation is to increase the direct dose rate from 819 to 1320. When shielding from the top grid plate and the end pieces are included the dose rate reduces to 1320. The net effect is an increase in the dose by a factor of 1.6. But, after a decay of ten minutes the net effect is a decrease in dose by a factor of 1.6. The latter may be more reasonable since a loss of water may take more than several minutes.

j. Section 7.2.2, page 7-12, first paragraph: This states that scattered radiation outside the building would "not be too high." However, the calculations of scattered radiation do not apply directly to this location. Please address in a quantitative manner the projected dose rate at the nearest unrestricted area due to "loss of coolant" core radiation.

The dose rate from radiation scattered by air above the roof (7 m high) that reaches a point just outside the building (7.5 m) was calculated using equation 7.17 in the SAR. Data for air was used in the calculation. Attenuation of the incident radiation through the roof material was also taken into account. The source I_0C (photons/s) included the gammas from fission product decay and the activation of structural materials. The shielding provided by the top grid plate and the top end pieces was also taken into account. Results are tabulated in the following Table 16.j.

Table 16.j Radiation Dose Rates for Loss of Shield Water Outside the Building

	Radiation (rad/hr)	
Operation Time→ / Decay time ↓	10 hr	1600 hr
1 minute	0.148	0.157
10 minutes	0.0265	0.036
1 hour	0.00940	0.0187
1 day	0.000897	0.00700
1 week	0.000103	0.00289
1 month	0.000019	0.00110

k. Your analysis assumed 1.25 MW-yr burnup. What additional burnup will exist on your proposed fuel at the time of initial criticality of your reactor? Please adjust your calculations to include this additional burnup.

The 1.25 MW-yr is conservatively high since the nominal maximum burnup for a 70 element TRIGA core would be ~0.85 MW-yr (70 X 4.5 MW-days/element = 315 MW-days). The burnup assumed for the calculation is larger than the maximum burnup for an element that could be used in the reactor. The larger burnup will yield maximum dose rates.

l. Section 7.3.1;

(1) Table 7-3 and 7-4 should be more clearly labeled to indicate one relates to a semi-infinite volume (assuming the hemisphere referred to in step 4, page 7-15, is of infinite radius) and one relates to a finite radius volume. Note, in step 5, page 7-15, it is more usual practice to use the radius of a hemisphere rather than a sphere.

Please see the revised section 7.3 (Attachment #2).

Step 4 of 7.3.1 and Table 7.3 are deleted from the SAR. The calculation performed in step 4 was for a sphere of infinite radius. While this assumption over estimates the dose rate, it yields a maximum upper value. The same result could be obtained by setting $r \rightarrow \infty$ in equation 7.26. The 10 minute exposure in this case would be 15.9 mr. The result from step 4 (old step 5) represents the actual situation more appropriately and this is given in the SAR. The radius of the hemisphere rather than the radius of sphere with volume equal to the reactor room volume is used in the updated version.

(2) Please compare your results with results you would obtain using the methods of Regulatory Guide 1.109, as applicable to the ATU scenario.

Please see the revised section 7.3 (Attachment #2).

(3) Step 6, page 7-15: Are you calculating ingestion or inhalation dose here? Please discuss.

The inhalation dose is calculated here. Appropriate corrections are made in the SAR.

(8) Table 7-5; Your Rem/Ci factor for I-135 seems to be at least an order of magnitude high. Please adjust or explain.

This correction is made in Table 7-4 (old Table 7-5).

m. Section 7.3.2:

(1) Explain and discuss the reasons for the differences between the whole body doses of 1.9×10^{-3} mr and 4.7×10^{-2} mr in one hour. Give a single "most likely" value, and justify it.

These numbers were obtained by scaling the occupational exposure. Two numbers were obtained for the occupational exposure. The lower value was obtained from assuming immersion in a sphere with volume equal to the reactor room volume. The higher number was obtained by assuming immersion in a sphere of infinite radius. The correct procedure is outlined in step 6 of 7.3.2 and yields a one hour exposure of 0.031 mr.

(2) Is there also a range for the projected thyroid doses as for the total body doses? Explain. If so, please furnish the best estimate value.

The thyroid dose is obtained by scaling the occupational exposure which is a single value. There is no range for this value and the best estimate is given in step 7 of 7.3.2. The one hour thyroid dose is about 8.3×10^{-3} rems.

(3) Compare the ATU approach with the methods of NRC Regulatory Guide 1.109 for potential annual doses in the unrestricted area, as applicable. What is the location of maximum exposure in the unrestricted area?

Please see the revised section 7.3 (Attachment #2).

n. For the fission product release analyses, please discuss the ATU methods and results in comparison with the applicable guidance of ANSI/ANS 15.7, the ANS standard for research reactor site evaluation.

Please see the revised section 7.3 (Attachment #2).

The analysis of possible dose to members of the general public set forth here does not use the dilution calculations set forth in ANS/ANSI 15.7 directly due to the proximity of the nearest building to the release point. As noted in section 4 of ANS/ANSI 15.7, the criteria set forth there may not apply at distances of less than 200 m (660 ft).

o. The ATU Technical Specifications permit the irradiation of fueled experiments, and other experiments that could affect reactivity. Please provide safety analyses of potential accidents involving these types of experiments. Include the potential impact on the health and safety of the public of an accidental step reactivity insertion equal to the maximum licensed excess reactivity.

A new section 7.4 (Attachment #2) is added to the SAR analyzing the failure of fueled experiments. Insertion of maximum licensed excess reactivity is analyzed in section 7.1 of the SAR.

p. Section 7.4; Please provide additional detail that justifies the assumptions and shows the calculations for this section.

The information for this section was taken from Michigan State SAR. We do not have more information than is provided. This section should be deleted from the SAR.

17. Chapter 8

a. Section 8.1.7; Please provide additional information such as charter, quorums, minutes, and details of review and audit functions concerning the Reactor Safety and Utilization Committee.

This information is covered in section 6.2, Review and Audit, of the Technical Specifications. (TS ATUTR pages 6-4 and 6-5)

b. Section 8.2.4; Please describe the "special training" the Reactor Supervisor will receive to be qualified for this position.

The Reactor Supervisor will receive special training to be qualified as senior operator for the ATU reactor facility. Please see 17.c also. The following sentence will be added to this section.

The reactor supervisor will be certified by the licensing agency as a senior operator for the ATU reactor facility.

c. Please clarify what is acceptable experience. Is this experience in the nuclear field or directly with research reactors?

Experience that would enable the Reactor Supervisor to perform adequately the duties associated with facility activities is acceptable. Experience in the nuclear field is required and experience with research reactors is highly recommended. The paragraph of section 8.2.4 will be modified as given below.

A person with a combination of academic and nuclear experience in supervision of reactor operation and related functions will be designated as the Reactor Supervisor. The Reactor Supervisor will be certified by the licensing agency as senior operator for the ATU reactor facility. A minimum of three years nuclear experience will be required and experience with research reactors is highly recommended. Academic training in appropriate engineering or science may be substituted for up to two of the three years experience. [ANSI-15.4, 4.4]

d. Explain how persons with unescorted access to the facility will be trained to meet the requirements of 10 CFR Part 19 and the requirements of your Emergency Plan.

The last sentence of section 8.2.5 will be replaced with the following.

Other persons may receive unescorted access to the reactor facility after completion of a training program in accordance with the requirements of 10CFR19.

This training program will be implemented and maintained by the facility management and will include instruction in radiation safety and effects and proper response to emergency conditions including the facility Emergency Plan.

e. Section 8.3.1; You discuss the need for documented concurrence from a senior reactor operator for recovery from unplanned or unscheduled shutdowns. How does this relate to the requirements of 10 CFR 50.54(m)(l) to have a senior reactor operator present at the facility.

Recovery from an unplanned or unscheduled shutdown will require the presence of a senior operator and documented verbal concurrence from the senior operator. The second paragraph on page 8-6 of the ATU SAR will be modified as follows.

Movement of any fuel or control rods and relocation of any in-core experiment with a reactivity worth greater than one dollar will require the presence of a licensed senior operator. Recovery from unplanned or unscheduled shutdowns will require the presence of a senior operator and documented verbal concurrence from the senior operator [ANSI-15.1,6.1.3(3)].

f. Please provide additional detail on your staffing requirements for experiments.

The staffing requirement for each experiment will be specified in the experiment plan approved by the Reactor Safety and Utilization Committee. Each experiment will be designated as one of three classes. One class will consist of experiments that are routine in nature (e.g., reactor operation for calibration or instruction, irradiations such as neutron activation, etc.). This class of experiment will require only the presence of a reactor operator. Some of the calibration or irradiation experiments may require the presence of both a licensed operator and the experimenter and will be designated as a separate class of experiment. The third class of experiments will require the direct supervision of a licensed senior operator for such activities as relocation of in-core experiments with a reactivity worth greater than one dollar, fuel or control-rod relocations within the core region, or significant changes to shielding of core radiation.

g. Section 8.3.3; For a substantive change to an experiment, who will be responsible for making the determination that the change does not constitute an unreviewed safety question and thus subject to NRC review and approval?

This determination will be made by the Facility Management working with the Reactor Safety and Utilization Committee. The guidance provided by the "Review of experiments for research reactors" (ANS-15.6/ANSI N401) will be followed to approve an experiment. Those experiments which introduce risks beyond those analyzed in the Safety Analysis Report shall be submitted to the NRC for review and approval.

h. Section 8.4.2; What is the time limit for reporting violations of safety limits to the NRC?

The following will be added to section 8.4.2 as the second paragraph.

A safety limit violation will be reported promptly by telephone and confirmed in writing (telecopy) not later than the next working day to NRC. A follow-up report that describes the circumstances of the event will be submitted to the NRC within 14 days of the event.

i. Section 8.5.3; What plans do you have to retain information concerning events that may have a significant effect upon decommissioning of the facility?

The following will be added to section 8.5.3 as the second paragraph.

ATU will collect and retain for the lifetime of the reactor facility information that will have significant effect on the decommissioning of the facility (ANSI/ANS-15.10, 9.1 and 9.2). The following design/construction documentation should be collected and archived: (1) Complete as-built drawings, (2) Construction photographs with detailed captions, (3) Procurement records that identify types and quantities of materials used during construction, (4) Equipment/components specifications, including pertinent information, ie., supplier, weight, size, materials of construction, etc. The following documentation should be collected and archived during operational phase of the facility: (1) Safety Analysis Report(s), (2) Technical Manual(s), (3) Environmental Assessments, (4) Power History, (5) Radiological Survey Reports, (6) Operating and Maintenance Procedures, (7) Abnormal Occurrence Reports such as spills, (8) Deactivation Plans/Reports, (9) Technical Specifications, (10) Design Changes and Updated Drawings.

18. Chapter 10; Please provide an updated SAR Chapter providing specific information on how your program meets the requirements of the regulations and any particular standards that you believe are applicable to your facility. Please consider the following issues in your update:

a. Section 10.1; Please provide a copy of the Arkansas Tech University ALARA policy statement.

b. Page 10-1; Neither the Introduction or the Policy and Organization sections mention that the requirements of 10 CFR Part 20 should be the minimum bases for an acceptable Radiological Protection Program.

c. Section 10.1.1; This section should be expected to assign the campus responsibility to an Office or organizational unit.

d. 10.1.2; It seems inappropriate to assign full implementation to the Reactor Supervisor. As a minimum, it seems that the Facility Director should hold that responsibility.

e. Section 10.1.2; What is the "special training" that the Reactor Supervisor will receive? Please provide additional detail.

f. Section 10.1.2; Please elaborate on the definition of academic training. Do you mean at least a B.S. degree? Justify how someone with training in biology or industrial engineering can substitute this training for nuclear experience.

g. Section 10.2; Discuss how you will meet the training requirements of 10 CFR Part 19.

h. Section 10.3; Please supply additional information on how the requirements discussed in Section 10.3 will be specifically applied to the material in Sections 10.3.1, 10.3.2 and 10.3.3.

i. Section 10.4, Radiation Monitoring; An environmental monitoring program should be established and it should be required in conjunction with the Construction Permit so that baseline data can be accumulated for at least a year before reactor operations start.

j. Section 10.4.1, Radioactive Effluent Monitoring; Monitoring of effluents is required unless you can clearly justify that there is no health and safety problem if they are not measured. Please discuss.

k. Section 10.4.2, Facility Monitoring; Requirements are set by 10 CFR Part 20 plus ALARA, and by Technical Specifications. Please relate the SAR to these requirements.

l. Provide details on monitoring of noble gas effluents, gaseous or airborne radioactive materials and liquid effluents. Include monitoring

equipment, set points, alarm actions, etc. How does this relate to Section 10.5?

m. Section 10.6; Please provide additional detail on the ALARA design features of the facility.

n. Section 10.6.2, Facility Operation; Various review functions seem to be assigned to the same office (Reactor Supervisor) as do the implementation functions. Review should be done at a level above that of implementation, no matter who it is.

o. Section 10.7; Please discuss your plans for retention of records concerning radiological events that can significantly impact decommissioning of the facility.

p. Section 10.8, Emergency Plan; This plan is required by 10 CFR Part 50, not by the Radiological Protection Plan. The Office responsible should be at least as high as Facility Director.

Chapter 10 is currently being revised to meet the requirements of the revised 10 CFR Part 20 and the guidelines provided by ANSI/ANS 15.11.

19. Chapter 11

Fire protection is normally considered as part of the facility design, with a description of the facility equipment and systems present to detect and minimize the effects of a fire. Please incorporate Chapter 11 into the SAR section on facility design.

This section will be incorporated as section 6.5 of the chapter 6.

20. Chapter 12

Please address the requirements of 10 CFR Part 55, and in particular 10 CFR 55.59.

A revised copy of Chapter 12 is attached (Attachment #3).

21. Chapter 13

a. Section 13.0; The NRC has the responsibility for the licensing of reactor operators, not General Atomics. Please correct your SAR.

The second paragraph on page 13-1 of ATU SAR will be modified as follows.

Training of university personnel associated with startup activities at the new facility will consist of training by GA Technologies and certification by NRC of at least two Senior operators. One or more of the certified operators shall have a bachelors or advanced degree in a field of engineering.

b. Please discuss your plans for monitoring construction activity to ensure that the facility is built in accordance with the SAR.

The Dean of the School of Systems Science will appoint a committee, which will include the Facility Director and the Reactor supervisor, to monitor the construction activities. The QA requirements specified in Chapter 9 of the SAR will be followed during the construction.

22. Chapter 14

Section 14.3; Please provide additional detail on the design features of the ATU reactor to accommodate decommissioning.

The reactor pool tank is placed inside a steel tank 14 ft in diameter and 27 ft deep. The steel tank will be surrounded by at least 1 ft of concrete. The space between the pool tank and the steel tank will also be filled with concrete. A larger pool tank, 10 ft dia as opposed to 6 ft dia at Michigan State University, will reduce by at least 50 % (1/r) the neutron population at the tank boundaries. More neutrons will be removed because of moderation and absorption in the larger amount of water. Samples of the materials used will also be retained to predict activities of these at the time of decommissioning.

23. The following questions apply to the Environmental Report;

a. Page 2, paragraph 2; Please be specific about the references to "other facilities" and their production of Ar-41. Please justify your quantities of "less than 50 Ci," and "less than 20 Ci." Please relate these quantities to your analyses in Section 6.4 of the SAR.

b. Page 2, paragraph 3; Please justify the statements you make about the quantities of hydrogen isotopes and liquid radioactive wastes released to the environment.

c. Page 2, paragraph 4; Please provide quantitative values and justify them in place of the statement "...expected to represent a fraction...".

d. Page 2, paragraph 4; Please explain the statement about "activation products are accumulated in an ion exchange resin...".

e. Page 3, paragraph 1; Please explain how liquid radioactive waste is stored and evaluated to ensure that releases remain "a fraction" of 10 CFR Part 20 constraints. What fraction?

f. Page 3, paragraph 2; The licensee will be responsible for potential environmental effects of irradiated fuel, until DOE actually takes possession, which might include packaging and shipping. Please discuss your plans in more detail.

g. Page 3; Because you have not discussed potential environmental impacts related to eventual decommissioning of your reactor at the end of its useful life, please discuss those effects in your Environmental Report.

h. Page 3, Section C; You dismiss potential environmental effects of accidents too briefly. Please discuss them, and justify your statement that they are "negligible".

i. Page 4, both paragraphs; The word "minimal" is used. Please be more specific and quantitative.

A revised copy of the Environmental Report addressing the above issues is attached (Attachment #4).

small activated components. Some of the reactor-based research results in the generation of solid low-level radioactive wastes. Solid wastes are stored by the university's health physics personnel and transported to an approved radioactive waste burial facility in accordance with applicable regulations.

6.3.2 Counting Laboratory

Air from the pneumatic transfer system is exhausted through the reactor ventilation system. A hood for handling radioactive materials, a sink for disposal of radioactive liquids, and a safety shower for decontamination are installed in this room.

6.4 CONTAINMENT DESIGN EVALUATIONS

Containment evaluation depends on the quantity of airborne radioactivity release possible from the air and water that are in the region of the reactor during operation. Calculation, measurement and experience of similar research reactors support the evaluation. Evaluation is limited to routine effluents and should be supplemented for experiment conditions that present specific release problems. Analysis of fission product releases are treated in another chapter.

Measurement and experience at other TRIGA facilities show that radiological contributions are caused mostly by argon-41 and nitrogen-16. Argon-41 is produced by the activation (n, γ) of argon-40 present either in the air in experimental facilities or the air dissolved in water. Calculations show an activity concentration in the reactor room of 8.1×10^{-8} μCi/ml from the pool water and 6.1×10^{-6} μCi/ml from the experimental facilities at 250 kW. The corresponding dose rates outside the building are 0.17 μrad/hr and 12.9 μrad/hr respectively. Nitrogen-16 is produced by the activation (n, p) of oxygen-16 in the reactor core region and has a very short half life. Calculations show an activity concentration of nitrogen-16 in the reactor room air of 5.5×10^{-9} μCi/ml resulting in an occupational dose rate of about 0.38 μrad/hr. The dose rate from the nitrogen-16 transported to the surface of the pool is about 670 μrad/hr.

6.4.1 Argon-41 Activity in the Reactor Room

6.4.1.1 Activation of Air Dissolved in the Pool Water

The release of argon-41 dissolved in reactor water depends on the gaseous exchange rate at the air-water interface and the change in gas solubility as a function of temperature. As the pool water circulates through the core, the equilibrium concentration of argon is depleted to the lowest solubility value. The release of argon

as a function of temperature and solubility approaches zero on a time scale comparable with the time required for the argon-41 activity to reach half the equilibrium value [1]. Argon atoms exchanged at the water-air interface depend on a water thickness depth that is small relative to the pool dimensions, therefore, a small fraction of the available saturated argon is exchanged with the air.

Evaluation of the water-air interface exchange rate for argon is related to an air and water thickness depth that depends on the argon atom diffusion coefficient. The total exchange rate is then a function of the pool surface area, A_s (7.3×10^4 cm^2), and an effective release volume $\dot{V_i}$. The two terms are related by

$$\beta_i A_s = f_{i \rightarrow j} \dot{V_i} , \qquad 6.1$$

where β_i ($cm\text{-}s^{-1}$) is a surface exchange coefficient, and $f_{i \rightarrow j}$ (s^{-1}) is the fraction of atoms exchanged from volume i to j. Estimates of β for argon vary considerably and a conservative value of 5.7×10^{-3} ($cm\text{-}s^{-1}$) [1-3] is assumed in the calculations.

During equilibrium conditions, and assuming no difference in the rates of escape fractions for argon-40 and argon-41, the number of argon atoms that escape from the water into the air equals the number of argon atoms that enter the water from air, i.e.,

$$f_{i \rightarrow j} \dot{V_i} N_i = f_{j \rightarrow i} \dot{V_j} N_j \qquad 6.2$$

where $N_j = 2.1 \times 10^{17}$ argon atoms per cm^3 of air and $N_i = 7.1 \times 10^{15}$ argon atoms per cm^3 of water at the core inlet temperature of 38 C [4].

The flow channel area per element is 0.0058 ft^2 [2] and with 70 elements the flow area is 0.406 ft^2 (377.2 cm^2). The heated length of the channel is 15 in. and the flow channel volume is 0.5075 ft^3 (14371 cm^3). At the full power operation of 250 kW the mass flow rate, w, through the reactor with natural circulation cooling is estimated (based on calculations in reference 2) to be about 2205 g/s (17500 lbm/hr) and the temperature rises from 38 C (100 F) to about 66 C (150 F) in passing through the core. The volume flow rate, v, is 2205 cm^3/s (w / density). The transit time through the core, t, is 6.52 s (flow channel volume / v). The pool cycle time, T, is equal to 2.52×10^4 s (volume of pool / v).

The changes in argon-41 concentration in the reactor water (subscript 1), in the pool water (subscript 2) and in the air of the room enclosure (subscript 3) are given by

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \varphi N_1^{40} \sigma^{40} - N_1^{41} \left(v + V_1 \varphi \sigma^{41} + \lambda^{41} V_1\right) + N_2^{41} v \quad 6.3$$

$$V_2 \frac{dN_2^{41}}{dt} = -\lambda^{41} N_2^{41} V_2 - v \left(N_1^{41} - N_2^{41}\right) - \left(f_{2\to3} N_2^{41} V_2' - f_{3\to2} N_3^{41} V_3'\right) \quad 6.4$$

$$V_3 \frac{dN_3^{41}}{dt} = \left(f_{2\to3} N_2^{41} V_2' - f_{3\to2} N_3^{41} V_3'\right) - N_3^{41}\left(\lambda^{41} V_3 + q\right) \quad 6.5$$

where the superscripts 40 and 41 are for argon 40 and 41 respectively. The volume of the different regions are, $V_1 = 14371$ cm^3, $V_2 = 55.6 \times 10^6$ cm^3 and $V_3 = 450 \times 10^6$ cm^3. The volume flow rate from room enclosure exhaust, q, is equal to 4.01×10^5 cm^3/s (850 cfm). The average thermal neutron flux , φ, is 4.0×10^{12} n/cm^2-s, the absorption cross sections, σ^{40} is equal to 0.47×10^{-24} cm^2, σ^{41} is equal to 0.060×10^{-24} cm^2 and the decay constant, λ^{41}, is 1.06×10^{-4} s^{-1}. The fraction of argon-41 atoms in region i that escape to region j per unit time, $f_{i\to j}V_i'$, is determined using equations 6.1 and 6.2. Since $v + V_1 \varphi N_1^{41} + \lambda^{41} V_1 \approx v$ equation 6.3 may be reduced to

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \varphi N_1^{40} \sigma^{40} - \left(N_1^{41} - N_2^{41}\right) v \quad 6.6$$

During equilibrium conditions the left hand side of the equations 6.4, 6.5 and 6.6 are set to zero. The resulting equations are solved to obtain an expression for the argon-41 concentration in the reactor room, N_3^{41} given below.

$$N_3^{41}\left[\frac{\lambda^{41} V_3 + q + f_{3\to2} V_3'}{f_{2\to3} V_2'} - \frac{f_{3\to2} V_3'}{\lambda^{41} V_2 + f_{2\to3} V_2'}\right] = \frac{V_1 \varphi N_1^{40} \sigma^{40}}{\lambda^{41} V_2 + f_{2\to3} V_2'} \quad 6.7$$

Solving for N_3^{41} yields 28.2 atoms/cm^3. The corresponding activity concentration is given by

$$A = \frac{\lambda^{41} \times N_3^{41}}{C} = 8.1 \times 10^{-8} \mu Ci/cm^3 \quad 6.8$$

where C = 3.7×10^4 dps/μCi.

These calculations show that argon-41 decays while in the water, and most of the radiation is safely absorbed in the water. The whole body gamma ray dose to a person immersed in a semi-infinite cloud of radioactive gases can be approximated by

$$D(\text{rad/hr}) = 900\ E\ A\ q\ \psi(x), \qquad 6.9$$

where E = the photon energy (MeV), A = activity concentration in the discharge (Ci/m^3), q = the building exhaust rate (m^3/s) and ψ(x) is the dilution factor at the distance x (s/m^3). If it is assumed that the discharge is at the roof line, the dilution factor in the lee of the building (x = 0), is given by [5],

$$\psi(0) = \frac{1}{CSU} = 4.5 \times 10^{-3}, \qquad 6.10$$

where C = 0.5, S = building cross sectional area normal to wind = 170 m^2 and U = wind velocity = 2.62 m/s. The average dose rate at ground level outside the building is

$$D = 900 \times 1.3 \times 8.1 \times 10^{-8} \times 0.401 \times 4.5 \times 10^{-3}.$$

$$= 1.7 \times 10^{-7} \text{rad/hour}. \qquad 6.11$$

Actual dose values for argon-41 release are anticipated to be substantially lower due to lower neutron fluxes, shorter operation times and larger dilution factors.

6.4.1.2 Activation of Air in the Experimental Facilities.

The central thimble, the rotary specimen rack, and the pneumatic transfer tube contain air. Of the radioisotopes produced, argon-41 (half-life = 110 min) is the most significant with respect to airborne radioactivity hazards. Nitrogen-16 (half-life = 7.11 s) and oxygen-19 (half-life = 26.9 s) are considerably less significant. The saturated concentration of argon-41 release from an experimental facility is calculated from

$$N_i\,(\text{atoms/ml}) = \frac{\Sigma_a\ \varphi_i}{\left(\lambda^{41} + q_i/V_i\right)} \qquad 6.12$$

where $\Sigma_a = 1.59 \times 10^{-7}\ \text{cm}^{-1}$, and Vi is the effective air volume of the facility. Since there are no clear flow paths or driving forces to move air from these facilities,

values of the volume exhaust rate, q_i, would be expected to be quite low. The reactor room exhaust fan effects one air change in the room roughly every 20 minutes. If we assume, in the absence of a driving force or clear exhaust path, that the experimental facilities have an air turnover rate of roughly one tenth that of the building, then this gives a q_i value for the rotary specimen rack of 33000 (200 × 60) = 2.75 cm^3/s. For simplification, this same rate will be used to analyze release from the PTS and the central thimble even though one air change per 200 minutes would give a much smaller value for these facilities. It should be noted that this assumed flow rate, from a relatively closed device, is roughly the same as that stated for each beam port at the TRIGA reactor facility at the University of Texas [2].

The effective air volume, V_i, conservative estimates of average thermal neutron fluxes for 250 kW operation, and the source rate from the experimental volumes, $N_i \times q_i$ (atoms/s), are given in Table 6-1.

Table 6-1 Activity of Argon-41 in the Experimental Facility

Region	Effective air volume ml	Average thermal flux at 250 kW (n/cm^2-s) × E+12	Source atoms/s $N_i \times q_i$
Central thimble	4.85E+2	5.4	4.09E+8
Rotary specimen rack	3.30E+4	1.7	3.93E+9
Pneumatic tube	1.60E+3	1.7	4.07E+8
Total			4.74E+9

The total estimated source of argon-41 from experimental facilities, S^{41}, is about 4.74×10^9 atoms/s (13.6 μCi/s). This is assumed to be released into the reactor room of volume, $V = 450 \times 10^6 cm^3$ and is removed by ventilation exhaust rate, $q = 4.01 \times 10^5 cm^3/s$, from the reactor room and decay (decay constant $\lambda^{41} = 1.06 \times 10^{-4} s^{-1}$). The rate of change of argon-41 concentration with time is equated to the source minus the removal rate and the resulting differential equation

is solved with zero initial conditions. Assuming saturated conditions the argon-41 concentration in the reactor room and the building exhaust air is given by

$$N^{41} = \frac{S^{41}}{(\lambda^{41} V + q)} = 1.06 \times 10^{4} \text{ atoms/ml} \qquad 6.13$$

The activity is calculated as outlined in the previous section and is obtained as $3.03 \times 10^{-5} \mu$Ci/ml in the reactor room and the building exhaust air. This calculation is based on each experimental facility being completely filled with air and operated to saturation levels of argon-41. Since experiments will replace much of the available air space in the facilities and since eight continuous hours of operation produce only 95% of saturation levels, it is assumed that only 20% of the experiment facility argon-41 is exhausted to the reactor room, and thus, the concentration level, during operations, in the reactor room is estimated to be no more than $6.1 \times 10^{-6} \mu$Ci/ml due to the experimental facilities. When the argon-41 estimated to be present from the pool water is added to this figure, the total estimated concentration of argon-41 in the reactor building is $6.2 \times 10^{-6} \mu$Ci/ml. The average dose rate at ground level outside the building is about 12.9×10^{-6} rad/hr. Actual dose values for argon-41 release are anticipated to be substantially lower due to lower neutron fluxes, smaller air volumes, shorter operation times, and larger dilution factors.

6.4.2 Nitrogen-16 Activity in the Reactor Room

The cross-section threshold for oxygen-16 (n, p) nitrogen-16 reactions is 9.4 MeV, however, the minimum energy of the incident neutrons must be about 10.2 MeV because of center of mass corrections. This high threshold limits the production of nitrogen-16 since only about 0.1% of all fission neutrons have energy in excess of 10 MeV. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the threshold. The effective cross-section of the reaction averaged over the TRIGA spectrum is 2.1×10^{-29} cm^2. This value agrees well with the value obtained from integrating the effective cross section over the fission spectrum.

The concentration of nitrogen-16 atoms per cm^3 of water as it leaves the reactor core is given by

$$N_2 = \frac{N_1 \sigma_1 \varphi_v}{\lambda_2} \times \left(1 - e^{-\lambda_2 t}\right) = 1.35 \times 10^{7} \text{ atoms/ml} \qquad 6.14$$

where N_1 = oxygen atoms per ml of water = 3.3×10^{22} atoms/ml, σ_1 = (n, p) cross section averaged over 0.6 - 15 MeV = 2.1×10^{-29} cm^2, φ_v = neutron flux (0.6 - 15 MeV) = 4×10^{12} n/cm^2-s, λ_2 = nitrogen-16 decay constant = 9.35×10^{-2} s^{-1}, and t = average time of exposure in reactor = 6.52 s.

The flow velocity in the core is 5.85 cm/s [volumetric flow rate (2205 cm^3/s) ÷ flow area (377.2 cm^2)]. Assuming that the water will rise with the same velocity, the transit time from the core to the surface is 109.5 s (640 cm ÷ 5.85 cm/s). This assumption is quite conservative as energy losses from the fluid stream resulting from turbulent mixing will reduce the velocity significantly. Furthermore, delays in transit time resulting from operation of the diffuser in the pool is sizable. Measurements of the dose rates at the pool surface of several TRIGA reactors show that the diffuser reduces the nitrogen-16 contribution to the surface dose rate by an order of magnitude depending on the size of the pool.

In 109.5 seconds the nitrogen-16 decays to 3.6×10^{-5} times the value of activity leaving the core. Thus the concentration of nitrogen-16 that reaches the surface of the pool, N_0, is estimated to be 485 atoms/ml.

Only a small portion of the nitrogen-16 atoms present near the pool surface are transferred into the air of the reactor room. When an N-16 atom is formed, it appears as a recoil atom with various degrees of ionization. For high purity water (approximately 2 μmho) practically all of the nitrogen-16 combine with oxygen and hydrogen atoms of the water. Most of it combines in anion form, which has a tendency to remain in the water [6]. It is assumed that at least one-half of all ions formed are anions. Because of its 7.1 s half-life, the nitrogen-16 decays before reaching a uniform concentration in the tank water. The activity will be dispersed over the surface area of the pool and much of it will decay during the lateral movement.

For the purpose of this analysis it is postulated that the water-bearing nitrogen-16 rises to the surface and spreads into a disk source of radius 125 cm. The time it takes for the nitrogen-16 to spread into this disk is

$$t_s = \frac{125 \text{ cm}}{5.85 \text{ cm/s}} = 21.4 \text{ s} \qquad 6.15$$

The average concentration during this time is [2]

$$N = \frac{N_0}{\lambda_2 t_s} \times \left(1 - e^{-\lambda_2 t_s}\right) = 210 \text{ atoms/ml.} \qquad 6.16$$

The number of nitrogen-16 atoms escaping into air is estimated as 1.9 atoms / cm^2-s [210 atoms/cm^3 × 0.009 cm/s] where 0.009 cm/s is the escape velocity [7]. The total source into the room is

$$S = 1.9 \frac{\text{atoms}}{\text{cm}^2\text{-s}} \times 4.91 \times 10^4 \text{ cm}^2 = 9.3 \times 10^4 \text{ atoms/s} \qquad 6.17$$

In the room, the activity is affected by dilution, ventilation, and decay. For saturation conditions the concentration is given by

$$N^{16} = \frac{S}{\lambda_2 V + q} = 2.2 \times 10^{-3} \text{ atoms/ml.} \qquad 6.18$$

where $V = 450 \times 10^6$ cm^3 is the reactor room volume, and $q = 4.01 \times 10^5$ cm^3/s is the ventilation exhaust rate. This corresponds to an activity concentration of 5.5×10^{-9} μCi/ml or about 0.015 Ci for continuous single shift operation at full power for one year. The gamma dose rate from nitrogen-16 of this concentration in the air is given by

$$D = \frac{3.7 \times 10^4 \frac{\text{photons}}{\text{s-}\mu\text{Ci}} \times 5.5 \times 10^{-9} \frac{\mu\text{Ci}}{\text{ml}} \times 600 \text{ cm}}{2 \times K}$$

$$= 3.8 \times 10^{-7} \text{ rad/hr.} \qquad 6.19$$

The effective radius of the room, 600 cm, is calculated based on a hemisphere of volume equal to the reactor room volume of 450 m^3. The flux to dose conversion factor, K, is equal to 1.6 x 10^5 photons/cm^2-s per rad/hr.

The dose rate at the pool surface arising from the nitrogen-16 near the surface is calculated using

$$D = \frac{\lambda_2 N}{2 \mu K} \times \left(1 - E_2(\mu h)\right), \qquad 6.20$$

where N = 210 atoms/ml, μ is the attenuation coefficient for 6 MeV photons in water = 0.0275 cm^{-1} and E_2 is the second exponential integral. The thickness of the nitrogen-16 bearing water, h, is equal to 0.97 cm [volumetric flow rate 2205 cm^3/s × t_s ÷ area of the disk surface]. This yields a dose rate of 670 μrad/hr. Transport time higher than that used in the calculation is normal and this would reduce the calculated dose substantially.

6.4.3 Evaluation of Gaseous Effluents

As discussed above, the only significant anticipated radioactive gaseous effluents produced by the reactor are argon-41 and nitrogen-16. Both of these radionuclides will contribute to doses received by personnel in the reactor room. However, due to its low concentration in the reactor room air and very short half-life, nitrogen-16 will not contribute measurably to doses received by personnel in the unrestricted area. Estimates of possible annual exposures to personnel in these two areas will now be made.

6.4.3.1 Estimated Doses to Reactor Room Personnel

As arrived at in previous sections, the calculated average concentrations of the two main radioactive gases present in the reactor room are

Ar-41	6.2×10^{-6} μCi/ml
N-16	5.5×10^{-9} μCi/ml.

The dose rate from these concentrations is calculated as follows. A hemisphere of volume equal to that of the reactor room (450 m^3) will have a radius of R = 6.0 m. The exposure rate (R/hr) received by a person covered by a hemisphere of radius R containing and activity concentration of A (Ci / m^3) is given by

$$D(\text{rad/hr}) = A\,\Gamma\,2\,\Pi\left(\frac{1-e^{-\mu R}}{\mu}\right), \qquad 6.21$$

where the source strength, Γ = 0.66 R.m^2/hr.Ci for argon-41 and Γ = 1.8 R.m^2/hr.Ci for nitrogen-16 [8, 9]. The linear energy absorption coefficient, μ = 0.0035 m^{-1} for argon-41 and μ = 0.0022 m^{-1} for nitrogen-16. For the case when $\mu R << 1$ the above equation reduces to

$$D(\text{rad/hr}) = A\,\Gamma\,2\,\Pi\,R. \qquad 6.22$$

So the dose rates from the calculated average concentrations in the air of the reactor room are

Ar-41	1.51×10^{-4} rad/hr
N-16	0.37×10^{-6} rad/hr.

Thus, for a worker occupationally exposed for 40 hrs/week, 50 weeks/yr, the total annual exposure from gaseous radionuclides in the reactor room air would be 0.302 rad or ≈ 300 mrem since the exposure is chiefly from gamma rays. This ex-

posure is conservatively high due to several assumptions in its calculation. These conservative assumptions are

a. The reactor operates at full power 2000 hrs/yr
b. The worker is in the reactor room for this 2000 hrs
c. The Ar-41 concentration is always equal to the equilibrium value (in reality, it will be very nearly zero each morning and build to only 95% of the value assumed here by day's end)
d. No credit is taken for diffuser operation to reduce the concentration of N-16

Thus, the expected annual occupational dose to reactor personnel from exposure to Ar-41 and N-16 in the reactor room is well within regulatory guidelines.

6.4.3.2 Estimated Doses to Unrestricted Areas

Due to its low concentrations in the reactor room air and its short half-life, the contribution to dose in unrestricted areas from N-16 will be negligible. The maximum possible exposure to persons in an unrestricted area would be to a person standing on the ground just below the exhaust fan outlet of the reactor building. To make the most conservative estimate of the dose possible at this location, the average concentration of Ar-41 in the air at this location must be determined. The average concentration at a point on the ground just below the exhaust fan outlet can be determined from

$$A_{ground} = A_{bldg}\, q\, \psi(0) \qquad 6.23$$

where q is the exhaust rate from the building, 0.401 m^3/s, and $\psi(0)$ is the dilution factor in the lee of the building at zero distance from the building. The value of ψ is given by equation 6.10 as 4.5×10^{-3} s/m^3. Thus the average concentration of Ar-41 at the foot of the building during reactor operations and assuming equilibrium levels inside the building is

$$A_{ground} = 6.2 \times 10^{-6} \times 0.401 \times 4.5 \times 10^{-3}$$

$$= 1.11 \times 10^{-8}\ \mu Ci/ml\,. \qquad 6.24$$

Now since the reactor will only operate part of the time, this concentration may be averaged over one year by multiplying by the fraction of time that the reactor is in full power operation. A reasonably conservative estimate is that the reactor operates at full power for a single shift for 48 weeks out of the year (two weeks

vacation, two weeks maintenance per year). Then the yearly average concentration of Ar-41 in the unrestricted area in the lee of the building is

$$A_{avg} = 1.11 \times 10^{-8} \times \frac{1920}{8760} = 2.43 \times 10^{-9}\ \mu Ci/ml \qquad 6.25$$

which is roughly 1/16 of the limit for unrestricted areas set forth in 10CFR20, App. B. The dose from this concentration can be estimated as that from a semi-infinite cloud of radioactive gas as

$$D\ (rad/hr) = 900\ E\ A_{avg} = 900 \times 1.3 \times 2.43 \times 10^{-9}$$

$$= 2.84 \times 10^{-6}\ rad/hr \qquad 6.26$$

and, so, the annual dose to a person standing at the foot of the reactor building for one year's operations with the most unfavorable wind conditions would be

$$Dose = D\frac{rad}{hr} \times 8760\frac{hrs}{yr} = 2.49 \times 10^{-2}\ rad = 25\ mrad. \qquad 6.27$$

The nearest temporary residence to the reactor building is Jones Hall, a dormitory which is approximately 150 feet (45 m) southwest of the exhaust fan outlet. Under the worst wind and weather conditions, the average concentration of Ar-41 at this point could approach that in the lee of the building. However, the residents of the dorm will have a much lower tendency for occupancy during hours of reactor operation. If a conservatively high occupancy rate of 50% is assumed (4 hours of the normal 8 hour workday), then, even under the worst weather conditions, allowing for no dilution of the Ar-41 between the reactor building and the dorm over 100 feet away, the occupants of the dorm would receive a maximum annual dose of approximately 12.5 mrem.

The nearest permanent residence to the reactor building is a private house located approximately 500 feet (150 meters) north of the reactor building. To approximate the maximum concentration of Ar-41 at this site, use the relation [10]

$$C = \frac{Q}{\pi U \sigma_y \sigma_z} \quad \text{for } y = z = 0 \qquad 6.28$$

where Q is the source strength (Ci's) at release and U is the average wind speed. From the reference at a point 0.15 km downwind from the release point and given the most pessimistic weather conditions, we have U = 2.62 m/s, σ_y = 10, and σ_z = 7. Thus the concentration of Ar-41 at this point would be

$$C = \frac{6.2 \times 10^{-6} \times 0.401}{\pi \times 2.62 \times 10 \times 7} = 4.32 \times 10^{-9} \mu Ci/ml \qquad 6.29$$

and the dose, averaged over one year of operation would be

$$D\,(rad/hr) = 900\,E\,C = 900 \times 1.3 \times 4.32 \times 10^{-9} \times \frac{1920}{8760}$$

$$= 1.11 \times 10^{-6}\,rad/hr \qquad 6.30$$

Now, since this is a permanent residence, the dose must be calculated over a full year of exposure and so the annual dose would be

$$Dose = D\frac{rad}{hr} \times 8760\frac{hrs}{yr} = 9.7 \times 10^{-3}\,rad = 9.7\ mrad. \qquad 6.31$$

Thus it can be shown that using conservative estimates of dispersion of emissions that the maximum conceivable dose due to release of Ar-41 from 1920 hours of full power operation of the reactor to persons in the nearest occupied buildings in the unrestricted area will be on the order of 10 mrad/yr. The production rate of Ar-41 used in these calculations is also expected to be conservatively high. The figures used above indicate an annual release of $6.2 \times 10^{-6} \times 0.401 \times 1920 \times 3600 = 17$ Ci of Ar-41 from the reactor building. Actual measurements of Ar-41 releases from the Oregon State University TRIGA Mark I reactor, which operates approximately 100 MW-days per year, show an annual release of approximately 8 Ci/yr. Since the calculations above are based on operations of 20 MW-days/year, it appears that the estimated Ar-41 production, and thus the estimates of both occupational and unrestricted doses, are high by approximately a factor of 10. Monitoring of reactor room and exhaust air will assure that Ar-41 emissions are within applicable limits.

6.5 FIRE PROTECTION

REFERENCES

1. "Argon-41, Analysis of Concentration in Reactor Room from Pool Water Releases, Supplement to final questions, The University of Texas at Austin TRIGA Reactor", January 1985.

2. "Safety Analysis Report, TRIGA Reactor Facility, Nuclear Engineering Teaching Laboratory. The University of Texas at Austin", November 1984.

3. Dorsey, N. E., "Properties of Ordinary Water-substance, Reinhold Publishing Corp., New York. p 554.

4. Ibid., p 537.

5. Slade, D. H., (ed.), "Meteorology and Atomic Energy", USAEC Reactor Develop. and Tech. Div. Report TID-24190, DFSTI, Springfield, Virginia, 1968.

6. Mittl. R. L., and M. H. Theys, "N-16 Concentration in EBWR", Nucleonics P. 81 (1961).

7. Dorsey, N.E., op cit., p554.

8 Cember, H. "Introduction to Health Physics". Pergamon Press, page 167,164.

9. Radiological Health Handbook.

10 Sesonske, A. "Nuclear Power Plant Design Analysis", USAEC 1979.

side the building are caused by only scattered radiation. This is not expected to be too high to be a public hazard.

7.3 FISSION PRODUCT RELEASE FROM CLAD RUPTURE

In this analysis it is assumed that a fuel element in the region of highest power density fails in air after a long exposure at full power. The inventory of radioactive noble gases and halogens in the reactor core can be calculated [7,8] using

$$Q_i\,(Ci) = 0.21081 \times 10^6 \times P \times F_i \times \left(1 - e^{-\lambda_i t}\right) \qquad 7.24$$

where the constant 0.21081×10^6 has units of fissions/MW per disintegrations/Ci, P is the Power in MW, F_i is the cumulative yield of fission products [9], λ_i is the decay constant and t is the operating time. The core inventory after continuous operation at 0.25 MW for 5 years (1.25 MW-yr) is given in Table 7-2. Since the normally quoted maximum burnup for TRIGA fuel elements is 4.5 MW-days per element, then the nominal maximum burnup for a 70 element core will be 0.85 MW-yr. Thus,

Table 7-2 Noble Gases and Halogens in the Reactor

Isotope	$T_{1/2}$		F_i %	Q_i Ci/core
Kr-83m	1.86	H	0.53	1119.4
Kr-85m	4.36	H	1.31	2761.6
Kr-85	10.70	Y	0.29	168.0
Kr-87	76.00	M	2.54	5354.6
Kr-88	2.79	H	3.58	7547.0
Kr-89	3.18	H	4.68	9865.9
Xe-131m	12.00	D	0.04	84.3
Xe-133m	2.30	D	0.19	400.9
Xe-133	5.27	D	6.77	14271.9
Xe-135m	15.70	M	1.06	2234.6
Xe-135	9.13	H	6.63	13976.8
Xe-137	3.82	M	6.13	12922.7
Xe-138	14.20	M	6.28	13238.9
I-131	8.05	D	2.84	5987.0
I-132	2.26	H	4.21	8875.1
I-133	20.80	H	6.77	14271.9
I-134	52.30	M	7.61	16042.7
I-135	6.75	H	6.44	13576.2

the postulated fission product inventory here represents a credible maximum possible using standard TRIGA fuel. The postulated inventory here also would represent approximately 23 years of operation at an estimated 20 MW-days/year.

The release of fission products from U-ZrH fuel has been studied at some length. A summary report of these studies [4] indicates that the release is mainly through recoil into the fuel-clad gap at temperatures below 400 C and this process is independent of the operating temperature. Above this temperature the release is through a diffusion process and is temperature dependent. It is important to note that the release fraction in accident conditions is characteristic of the normal operating temperature and not the temperature during the accident conditions. This is because the fission products released are those that have collected in the fuel-clad gap during normal operation.

The following assumptions are used in the analysis:

a. One Fuel element in the region of highest power density fails in air after 1.25 MW-yr exposure and 100% of the noble gases and halogens in the gap are released. The release from a single element of a 70 element core in the region of highest power density with a peak to average flux of 2 is assumed. This fuel element produces 2.85% of the total power.
b. Peak fuel temperature is less than 400 C and the release fraction is estimated to be less than 1.5×10^{-5} (GA 4314). If a conservative value of 2.0×10^{-5} is assumed the fraction of noble gases or halogens released from the fuel element is obtained as 5.7×10^{-7} ($0.0285 \times 2. \times 10^{-5}$).
c. There is no plate-out of any released fission products.

7.3.1 Exposure to Reactor Room Occupants.

In order to calculate the exposure to reactor room occupants the following assumptions are made.

1. Noble gases and the halogens are released into the reactor room rapidly at the fraction given in (b) above. The concentration, qi (μCi/ml), of the radioisotope in the room which has a volume of 450 m^3 is given by equation 7.25 and the calculated values are shown in Table 7-3.

$$qi\left(\frac{\mu Ci}{ml}\right) = \frac{Qi \times 10^{6}\,(\mu Ci) \times 5.7 \times 10^{-7}}{450 \times 10^{6}\,cm^{3}}. \qquad 7.25$$

2. Ventilation in the room is assumed to be zero.

Table 7-3 Exposure to Occupant (step 4 of 7.3.1)

Isotope	qi μCi/ml	E_γ Mev/dis	k Mev/cm^2/s	μ cm^{-1}	Exposure mr/10 min whole body
Kr-83m	1.42E-06	2.60E-03	3.80E+01	3.00E-01	1.00E-03
Kr-85m	3.51E-06	1.60E-01	6.00E+05	1.70E-04	1.65E-03
Kr-85	2.14E-07	2.20E-03	2.30E+01	5.00E-01	1.26E-04
Kr-87	6.81E-06	7.80E-01	5.30E+05	2.00E-05	1.84E-02
Kr-88	9.59E-06	2.00E+0	6.30E+05	5.80E-05	5.54E-02
Kr-89	1.25E-05	1.50E+0	6.00E+05	6.50E-05	6.07E-01
Xe-131m	1.07E-07	2.00E-02	3.00E+04	1.00E-03	9.94E-05
Xe-133m	5.09E-07	4.10E-02	2.40E+05	3.20E-04	1.46E-04
Xe-133	1.81E-05	4.60E-02	3.30E+05	2.80E-04	4.31E-03
Xe-135m	2.84E-06	4.30E-01	5.10E+05	1.20E-04	4.27E-03
Xe-135	1.78E-05	2.50E-01	5.60E+05	1.50E-04	1.40E-02
Xe-137	1.64E-05	1.60E-01	6.00E+05	1.70E-04	7.70E-03
Xe-138	1.68E-05	1.10E+0	5.60E+05	7.80E-05	5.97E-02
I-131	7.61E-06	3.80E-01	5.20E+05	1.26E-04	9.91E-03
I-132	1.13E-05	2.20E+0	6.50E+05	5.60E-05	6.95E-02
I-133	1.81E-05	6.10E-01	5.20E+05	1.05E-04	3.82E-02
I-134	2.04E-05	2.60E+0	7.00E+05	5.00E-05	1.38E-01
I-135	1.73E-05	1.50E+0	6.00E+05	6.70E-05	7.82E-02
Total Whole body					5.61E-01

3. The occupant remains in the room for 10 minutes while evacuation takes place.

4. The whole body dose from each isotope, D_h, was also calculated using data from the Radiological Health Handbook [10]. The exposure for a person immersed in a hemispherical cloud of finite radius is given by

$$D_h\left(\frac{R}{hr}\right) = q_i \times 3.7 \times 10^4 \times \left(E_\gamma\right) \times \left(\frac{1 - e^{-\mu r}}{2 k \mu}\right) \qquad 7.26$$

where E_γ (Mev/dis) for each isotope is obtained from [11], and k is the energy fluence rate to give 1 R/hr for each isotope , μ is the linear absorption coefficient for each isotope and r is the radius of a hemisphere with volume equal to reactor room volume = 599 cm. The results for 10 minute exposure is given in Table 7-3.

Table 7-4 Thyroid Dose

Isotope	q_i μCi/ml	I_i 10 min μCi	Effectivity Factor rem/Ci	Thyroid Dose rems
I-131	7.61E–06	1.58E+00	1486000	2.35E+00
I-132	1.13E–05	2.35E+00	52880	1.24E–01
I-133	1.81E–05	3.78E+00	395100	1.49E+00
I-134	2.04E–05	4.25E+00	25380	1.08E–01
I-135	1.73E–05	3.59E+00	123100	4.42E–01
Total Thyroid Dose .				4.52E+00

The total whole body 10 minute exposure, immediately following a fuel element rupture in air, is 0.56 mr. This value is well within the requirements of 10 CFR Part 20 Since the actual burnup rate is expected to be much lower than that postulated here, the exposure calculated here serves as a conservative estimate of that possible.

5. For calculating the internal dose from the halogens, mostly iodine, the activity inhaled must be found. This value, denoted by I_i (Ci) in Table 7-4, is calculated based on the "working" inhalation rate [12] of 3.47×10^{-4} m^3/s and is given by:

$$I_i = q_i \times 3.47 \times 10^{-2} \frac{m^3}{s} \times 10 \text{ min} \times 60 \frac{s}{min} . \qquad 7.27$$

The dose from this amount of each isotope is calculated using the inhalation dose factors from Table E-7 of NRC Reg Guide 1.109 [13] and tabulated in Table 7-4. Since the thyroid is the critical organ for radioiodines, the dose for the thyroid is the only one calculated here. The resultant thyroid dose of 4.52 rem for 10 minutes is high, but it is based on several conservative assumptions regarding fuel burnup, halogen plate-out, and evacuation time.

7.3.2 Exposure to General Public.

Some of the radioisotopes from a fuel element rupture would be released to the environment and the purpose of this analysis is to calculate the exposure to the general public. In so doing, a dilution factor for the gases escaping the reactor

building must be found. Guidance for determining this dilution factor can be found in both NRC Reg Guide 1.111 and in ANSI/ANS 15.7. The equations used to compute dilution factors in both of these documents are basically identical. However, in both cases, the equations and values of certain variables are meant for situations in which the point of interest is at least 100-200 meters from the point of release. Since the nearest occupied building to the reactor building is only 40-50 meters away, an alternative form for determining the dilution factor must be used. The form that is chosen here is the same as that used in Chapter 6 in evaluating possible doses due to argon-41 release from the reactor building. That is, the dilution factor used here is that applicable in the lee of the building and no further dilution is assumed to occur between the reactor building and the nearest exposed individual.

The estimate of possible doses due to off-site release of fission product gases is now carried out with the following assumptions:

1. Following rupture of the fuel element discussed in section 7.3.1, the air ventilation and radiation monitoring system fail allowing the exhaust fan to vent unfiltered air from the reactor building to the outside at the rate of 850 cfm (0.401 m^3/s) and this exhaust continues for one hour.
2. The concentration of radioisotopes in the reactor room does not decrease with time as it would in a real accident due to removal of radioisotopes and decay.
3. The concentration of radioisotopes in the unrestricted area is equal to that in the lee of the building. That is, the maximum exposed individual is standing next to the reactor building during the accident.
4. The release occurs just below roof level, approximately ~ meters above ground and the dilution factor of 4.5×10^{-3} s/m^3 as found in Chapter 6 will be used.
5. The exposure to the maximum exposed individual is calculated by the methods set forth in NRC Reg Guide 1.109 [13] as:

$$D^{\gamma} \text{ or } D^{\beta} = 3.17 \times 10^{4} \sum_{i} Q_i \left[\frac{\chi}{Q}\right]^{D} \left(DF_i^{\gamma} \text{ or } DF_i^{\beta}\right) \qquad 7.28$$

   The results of these calculations are given in Table 7.5 for the gamma and beta dose.

6. The internal dose to other organs due to inhalation is not computed due to the thyroid being the critical organ for iodine.

Table 7-5 Exposure to Maximum Individual (step 5 of 7.3.2)

Isotope	Qi	DF^{γ}	DF^{β}	D^{γ}	D^{β}
	Ci/yr	$\frac{mrad-m^3}{pCi-yr}$		mrad	mrad
Kr-83m	2.05E-3	1.93E-5	2.88E-4	5.64E-6	8.42E-5
Kr-85m	5.07E-3	1.23E-3	1.97E-3	8.90E-4	1.42E-3
Kr-85	3.09E-4	1.72E-5	1.95E-3	7.58E-7	8.60E-5
Kr-87	9.83E-3	6.17E-3	1.03E-2	8.65E-3	1.44E-2
Kr-88	1.38E-2	1.52E-2	2.93E-3	2.99E-2	5.77E-3
Kr-89	1.80E-2	1.73E-2	1.06E-2	4.44E-2	2.72E-2
Xe-131m	1.54E-4	1.56E-4	1.11E-3	3.43E-6	2.44E-5
Xe-133m	7.35E-4	3.27E-4	1.48E-3	3.43E-5	1.55E-4
Xe-133	2.61E-2	3.53E-4	1.05E-3	1.31E-3	3.91E-3
Xe-135m	4.10E-3	3.36E-3	7.39E-4	1.97E-3	4.32E-4
Xe-135	2.57E-2	1.92E-3	2.46E-3	7.04E-3	9.02E-3
Xe-137	2.37E-2	1.51E-3	1.27E-2	5.11E-3	4.29E-2
Xe-138	2.43E-2	9.21E-3	4.75E-3	3.19E-2	1.65E-2
Totals				1.31E-1	1.22E-1

The above referenced Reg Guide also sets forth similar calculations for "Total Body" and "Skin" doses received from noble gas releases. An investigation of the equations and dose conversion factors for these two doses shows that the total body dose conversion factor for each of the isotopes listed above is lower than the associated exposure conversion factor. Therefore, the total body dose due to gamma and beta exposure from the postulated fission product release would be less than the exposures listed in Table 7-5. For the skin dose, the conversion factor listed in Table B-1 of Reg Guide 1.109 for beta skin dose is, again, lower than the factor for beta exposure. So, allowing for the 1.11 multiplication factor for gamma exposure in the equation for computing skin dose, the estimated skin dose from the postulated release would be less than 0.27 mrem.

For radioiodines, the chief dose received is due to inhalation of the halogens and the critical organ for iodine is the thyroid. Therefore, the inhalation dose due to radioiodines released during the postulated accident will now be estimated. The procedure for this estimation is set forth in Reg Guide 1.109 and is given by:

Table 7-6 Thyroid Dose to Maximum Individual

Isotope	X_i pCi/m^3	DFA_{ia} mrem/pCi	D_a^A mrem
I-131	4.87E−1	1.49E-3	5.81E+0
I-132	7.22E−1	1.43E-5	8.26E−2
I-133	1.16E+0	2.69E-4	2.50E+0
I-134	1.30E+0	3.73E-6	3.88E−2
I-135	1.10E+0	5.60E-5	4.93E−1
Total Thyroid Dose			8.92E+0

$$D_a^A = R_a \sum_i X_i DFA_{ia} \qquad 7.29$$

where R_a is the inhalation rate for individuals in age group a, X_i is the average concentration of the ith isotope in air, and DFA_{ia} is the inhalation dose factor for isotope i and age group a (where the organ of interest is the thyroid). Table 7-6 below summarizes these factors for the radioiodines released in the postulated accident.

Since the nearest occupied building to the reactor building is a dormitory, the maximum exposed individual was taken to be an adult for the above calculations. Also, in order to attempt to make a more accurate estimate of the thyroid dose possible, the average concentration was calculated in a different manner than for the noble gases. The average iodine concentrations were calculated from:

$$X_i\left(\frac{pCi}{m^3}\right) = \frac{Q_i(Ci) \times 4.5 \times 10^{-3}\left(\frac{s}{m^2}\right) \times 1 \times 10^{12}\left(\frac{pCi}{Ci}\right)}{3600\left(\frac{s}{hr}\right) \times 8760\left(\frac{hr}{yr}\right)} \qquad 7.30$$

Thus, the average concentration of each isotope of iodine was found from assuming that the total iodine inventory released in the accident was exhausted from the building in one hour and averaging this concentration over a year's time. The

equivalent thyroid dose would be found by using the instantaneous concentration and multiplying by an $\frac{8000}{8760}\left(\frac{m^3}{hr}\right)$ inhalation rate.

In conclusion, it is seen that, even using conservative estimates of fission products released and ignoring radioactive decay and the ventilation system's safety features, that the estimated dose to persons in the unrestricted area from the postulated failure of a single fuel element are within the target guidelines set forth in NRC Reg Guide 1.109. In fact, the dose due to noble gases would be within the stated limits even if all fuel elements failed simultaneously. Due to the tendency of iodine gases to "plate out" on surfaces and the automatic actions of the ventilation system, the thyroid doses estimated here are conservatively high. Thus, no credible accident would be likely to cause a member of the general public to receive more than 10 mrem total body dose or 15 mrem thyroid dose.

7.4 SAFETY ANALYSIS FOR FAILURE OF FUELED EXPERIMENT

The reactivity limit on an experiment is 1.4 %Δk/k. Accidental movement of the experiment with this maximum worth when the reactor is operating at full power 250 kW is considered here. This is a conservative analysis since with the 1.4 %Δk/k experiment loaded the excess reactivity available is not sufficient to reach the maximum power level. Analysis similar to the one given in section 7.1 shows that the resulting maximum fuel temperature is only 524 C, which is less than the temperature due to the insertion of maximum excess reactivity from low power. The corresponding stress on the clad is 1572 psi. Insertion of the maximum excess reactivity from full power is not possible, but the following is given as an information. If the maximum excess reactivity is inserted from full power (somehow!) the maximum fuel temperature is 837 C and the corresponding stress on the clad is 3620 psi. The values do not exceed equilibrium element conditions and pressure.

In evaluating the possible failure of a fueled experiment, limits are set on the maximum activity of such experiments due to radioiodines and strontium-90. The restriction placed on such experiments is that the concentration of the radionuclides in question in the reactor room air do not exceed the maximum allowable DAC set forth in 10CFR20. For the purposes of this evaluation, the following assumptions are used:

a. One-fifth (20%) of the total iodine inventory of a fueled experiment is I-131 which has the most restrictive DAC. This is a conservative figure, since I-131 comprises just over 10% of total iodine (I-131 through I-135) by analysis of fission yields.

b. One-half (50%) of the total strontium inventory of such an experiment Sr-90 which has the most restrictive DAC. This is also conservative, since Sr-90 comprises approximately 25% of total strontium yield.

c. The iodine and strontium are evenly distributed in the air of the reactor room which has a volume of 450 m^3.

d. Upon failure of such an experiment, evacuation will take place within 15 minutes, following alarms on the building air monitor(s).

e. The concentration of radioiodine and strontium are averaged over one calendar quarter (13 weeks at 40 hours per week).

If a given activity of radionuclide is released into the reactor building, the concentration of that nuclide is found by simply dividing the activity released by the

Table 7-7 Maximum Allowed Fueled Experiment Inventories

Element	DAC $\frac{\mu Ci}{ml}$	Activity mCi
Iodine	2×10^{-8}	94.0
Strontium	1×10^{-9}	1.9

volume of the room. Therefore, if we take the DAC limits as our maximum allowable concentration, we can work backwards to find the activity released that would produce this concentration by:

$$A_{lim} = \frac{DAC \times 10^{-3}\left(\frac{mCi}{ml}\right) \times V_{room}\,(ml) \times 13\,\text{weeks} \times 40\left(\frac{hrs}{week}\right)}{\text{yield} \times 15\,\text{min} \times \frac{1\,hr}{60\,\text{min}}} \qquad 7.31$$

where the yield represents the assumed portion of total iodine and strontium that are I-131 and Sr-90 as set forth above and the 15/60 factor accounts for the time spent in the room after the release. Using this equation, we arrive at the limits of total iodine and strontium that, when released into the reactor room, will produce one DAC of I-131 or Sr-90 respectively. The results of these calculations are given in Table 7-7 below.

Due to the conservative nature of these calculations, the limits thus set are 100 mCi of total iodines (I-131 thru I-135) and 2 mCi of total strontium for fueled experiments. It should also be noted that these calculations take no credit for en-

gineered safeguards on such experiments. If a fueled experiment with such safety features is to be installed, it may be eligible for a higher activity limit following an individual safety analysis and approval by the NRC.

As a final check on the activity figures given above, the concentrations of these radioisotopes in the unrestricted area surrounding the reactor building is calculated. In performing this calculation, the concentration in the lee of the building is found by the same methods used earlier in this Chapter. Thus, using the 0.401 × m^3/s building exhaust rate and the 4.5 × 10^{-3} s/m^3 dilution factor, the concentration outside the reactor building would be 3.9 × 10^{-11} μCi/ml for I-131 and 1.9 × 10^{-12} μCi/ml for Sr-90. Both values are below the allowable air concentrations set forth in 10CFR20 for unrestricted areas.

REFERENCES

1. West, G. B., W. L. Whittemore, J. R. Shoptaugh Jr., J. B. Dee, C. O. Coffer, "Kinetic Behavior of TRIGA Reactors", General Atomic Report GA-7882, 1967.

2. Simnad, M. T., "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuels", General Atomic Report E-117-833, 1980.

3. Simnad , M. T. and J. B. Dee, "Equilibrium Dissociation Pressures and Performance of Pulsed U-ZrH Fuels at Elevated temperatures", Gulf General Atomic Report GA-8129, 1967.

4. Foushee, F. C., and R. L. Peters, "Summary of TRIGA Fuel Fission Product Release Experiments", Gulf Energy and Environmental Systems Report Gulf-EES A10801, 1971 p. 3.

5. "Safety Analysis Report, TRIGA Reactor Facility, Nuclear Engineering Teaching Laboratory, The University of Texas at Austin", November 1984.

6. Way, K. and E. P. Wigner, "Radiation from Fission Products", Physics Review, 70 p. 115, 1946.

7. Lamarsh J. R., "Introduction to Nuclear Engineering ", Addison Wesley, 1977.

8. "TRIGA Mark I Safety Analysis Report", Michigan State University, June 1984.

9. Meek and Ryder, "Compilation of fission product yields", NEDO 12154, 1974.

10. Radiological Health Handbook, U. S. Department of Health, Education, and Welfare, 1970.

11. USNRC Regulatory Guide 3.34, 1979.

12. ANSI/ANS 15.7

13. USNRC Regulatory Guide 1.109, 1977.

12. TRAINING AND REQUALIFICATION OF OPERATORS

This section describes the program applied to training and requalification of personnel who are to be certified as operators by the U.S. Nuclear Regulatory Commission. This program is based on guidance given in ANSI/ANS 15.4-1988, "Selection and Training of Personnel for Research Reactors" and on the requirements of 10CFR55.

12.1 INTRODUCTION

This requalification program is applicable to all personnel holding valid licenses for the ATUTR except as set forth in the program below. There are two license classes which are covered by this plan; the Reactor Operator (RO) and the Senior Reactor Operator (SRO). License qualification by written and operating test, and license issuance, renewal or removal are the responsibility of the U.S. NRC. No rights of such a license may be assigned or otherwise transferred and the licensee is subject to and shall observe all rules, regulations and orders of the Commission. Initial training provides for the skills and knowledge required to initially obtain a license while the requalification program maintains the skills and knowledge of RO's and SRO's during the period of the license.

12.1.1 Operator License Status

Active status of any licensee shall require the performance of the functions of their license for a minimum of four hours each four months. If this condition for an active license status is not met in any four month period, active status may be restored by; (1) completion of all requalification training for the current quarter, if any, (2) completion of a minimum of six hours of license functions, properly supervised by a qualified SRO, and (3) certification, by the Facility Director, that the above conditions have been met and that all other conditions and qualifications of the licensee are current and valid. A licensee whose license is inactive shall not perform any activities for which a license is required, except those set forth in item (2) above, until such time as the license is restored to active status.

In addition to the above minimum performance requirements all licensees must maintain continued medical qualification. This will be determined by biennial medical examination and certification of physical fitness by the Facility Director.

12.2 REQUALIFICATION PROGRAM

The requalification program consists of training personnel by lectures, instruction, discussion and self-study. In the case that the number of licensed individuals is three or less, then the majority of training will be accomplished through discussion and self-study.

12.2.1 Requalification Program Bases

The guidance bases for the requalification program arise from two main sources: (1) the regulatory requirements set forth in 10CFR55 for the licensing of RO's and SRO's and the requirements for requalification set forth in section 55.59, and (2) the standards for selection and training of research reactor personnel available in ANSI/ANS 15.4. In addition, 10CFR50.54 and 10CFR55.53 set forth conditions of the facility and individual licenses which also provide guidance bases for the program.

12.2.2 Schedule

The requalification program will be completed on a two year cycle. Lectures or self-study assignments from the topics listed in 12.2.3 will be given on a four month basis, thus allowing completion of all six topics in the two year cycle.

In addition to the lectures noted above, each licensee shall be required to complete the necessary hours of license activities prescribed in section 12.2 during each four month period. On an annual basis, each licensee must perform a minimum of two reactivity manipulations. For the purposes of this program, the following manipulations will serve to satisfy this requirement: reactor startup, reactor shutdown, power level changes of greater than 50% of initial reactor power, reactor pulsing, simulated emergency condition. In addition, at least one of these manipulations must be a reactor pulse and at least one manipulation must not be completed as part of the annual evaluation described in section 12.2.

12.2.3 Requalification Program Training Subjects

The six topics to be covered during each two year requalification program are:

a. Principles of Reactor Facility Operation - example topics may include thermodynamics, heat transfer, fluid flow, basic reactor theory and radiation protection.

b. Facility Design and Operating Characteristics - example topics may include safety systems, design features, experiment and test facilities.

c. Instrumentation and Control - example topics may include nuclear instruments, process instruments, control systems, radiological instruments, and experiment and test facility instrumentation and control.

d. Procedures and Technical Specifications - example topics may include normal, abnormal and emergency procedures, radiological control, and administrative controls.

e. Radioactive Materials Handling - example topics may include special nuclear material handling, radioactive materials handling, disposal, and safe practices.

f. Regulations - example topics may include facility management controls, rules, applicable regulations, and license conditions.

12.2.4 On-the-job Training

On-the-job training that is required as part of the requalification program will include the aforementioned minimum hours of license activities and reactivity manipulations. In addition, the following activities shall be completed as prescribed below as part of the on-the-job training.

a. Each licensed operator shall review the following items on a biweekly basis: Scram Log, Maintenance Log, Required Reading folder.

b. Each licensed operator shall review all facility design or operating procedures changes on a timely basis following final approval of such changes. Normally, this information will be included in the Required Reading folder and also distributed separately when the information warrants.

c. Each licensed operator shall review the Emergency Plan (and associated procedures), and the Security Plan (and associated procedures) on an annual basis. This review may be accomplished as one of the requalification program lectures.

12.3 EVALUATION AND RETRAINING

Knowledge of each licensed operator (RO or SRO) will be evaluated by the requalification program exams described in section 12.2.2 and given every four

months throughout the training cycle. Each subject on such exams will be graded on a 100 point basis with an average score of 80% as the acceptance criteria. An overall score of less than 60% on any subject will result in an immediate suspension of license duties until the licensee is restored to active status by the Facility Director. Proficiency will be demonstrated by retraining and additional examination. A score between 60% and 80% on any subject will require retraining as needed in that subject area and a demonstration of acceptable knowledge by oral or written exam prior to recertification. License holders who are required to undergo such remedial training will not be suspended from licensed duties during the time of retraining, and such retraining should be completed within one month of the date of initial examination.

The performance and competency of each licensed operator will be evaluated on an annual basis by the Reactor Supervisor or Facility Director. This evaluation may take the form of an annual operating exam monitored by the Facility Director, Reactor Supervisor, or a designated SRO. This operating exam will normally consist of at least two reactivity manipulations as described in section 12.2.3 above. In addition, the licensee's knowledge of such items as status of experiments, log or checklist entries, and facility design, procedure, or license changes will be evaluated by oral or written examination as part of this annual evaluation.

12.4 RECORDS

Records for each licensed operator will consist of the documentation for the requalification program activities within the two year training cycle. Records shall be kept for both the current and previous cycle. The documentation for each licensee will include:

a. Education, experience, employment history and medical/physical evaluation.
b. Record of attendance at training lectures or other completion of training programs/topics.
c. Copies of initial qualifying exam(s) and all requalification exams during the appropriate training cycles.
d. Records of annual operator evaluations.
e. Records of completion of required reactivity manipulations and hours of licensed activities.

REFERENCES

1. "Selection and Training of Personnel for Research Reactors", [ANSI/ANS-15.4-1988].

ENVIRONMENTAL IMPACT

This document deals with the environmental effects that are expected from the operation of a TRIGA Mark I research reactor at Arkansas Tech University. The reactor is part of the Center for Energy Studies.

A. Environmental Effects of Facility Construction

The TRIGA reactor will be located in the Center for Energy and Environmental Studies at Arkansas Tech University. Design of the building is intended for the reactor facility and other laboratories and offices associated with applications and research in nuclear technology. Approximately 22% of the 1068 square meter facility is designated for the reactor facility.

Utility construction projects such as chilled water, heated water, electricity, road construction, and other support facilities for the Center are provided and proposed. Utility requirements and construction activities for the Center and its related facilities including the reactor facility will not be different substantially from those required during standard construction projects. Construction activities of the facility should have no impact on areas beyond the Center site.

Utilities required by the facility are communications, electricity, chilled water, domestic water and sanitary sewer. Some utilities such as hot water, chilled water, and compressed air are generated at the reactor facility. All utilities at the Center are maintained by the University.

B. Environmental Effects of Facility Operation

The TRIGA reactor facility is designed for 250 kW(thermal) steady-state operation. Environmental effects of the operation will include waste heat disposal, production of liquid and gas radioactive effluents, and the generation of liquid and solid radioactive wastes. None of these waste items are considered significant with respect to environmental impact although each is treated appropriately. No resources for the facility are considered significant to the environment.

Heat disposal from the reactor pool is provided by heat exchange with a central chilled water supply. Estimated chilled water requirements for dissipation of the peak facility heat load is 250 kW for the reactor and 66 kW for the building. A chilled water facility with a capacity of about 300 kW would provide the ultimate heat rejection source for the reactor.

Radioactive gas effluents produced by the reactor are argon-41 and nitrogen-16. Production and release of argon-41 is a function of the reactor power level, operation time and quantity of a. exposed to the reactor with some contribution from dissolved air in the coolant. Production and release of nitrogen-16 is related primarily to the reactor power level and coolant flow through the reactor. Occupational exposure to these gases inside the building will be controlled by monitoring and limiting production if necessary. The very short half-life of nitrogen-16 eliminates any significant environmental release of this gas The release of argon-41 from the facility is monitored and production and release of this gas will be limited as necessary to meet applicable regulations. Conservative calculations of argon-41 production suggest a maximum annual release of less than 20 Curies (17 Curies - ATU SAR). Actual measurements of releases from similarly designed, but higher power reactors indicate that actual releases should be less than 5 Curies (Oregon State University Reactor annual report).

Production of activation products in the coolant water consists of activation of low levels of impurities and very small quantities of tritium from natural deuterium in the water. The activation of short-lived gaseous products of oxygen and nitrogen are not considered significant environmental effects due to their rapid decay. The levels of tritium that may be produced in the coolant water are well below regulatory limits for tritium levels in water in unrestricted areas. Liquid waste releases will be held for monitoring and decay prior to release to the environment, through the sanitary sewer, in accordance with applicable regulations.

Solid wastes produced by the facility will include both standard waste for disposal with the University's solid waste, and radioactive waste. Non-radioactive solid and liquid waste will be disposed of through the University's normal waste stream and are expected to represent less than 2% of the total waste stream. Radioactive solid wastes will be mostly comprised of relatively short-lived isotopes such as Na-24, Mg-27, Al-28, Cl-38, Cr-51, Mn-56, Ni-65, As-76 and others. Small amounts of longer-lived radioisotopes such as Co-60 are also generated. Waste disposal will include gloves, paper, containers, samples and resin. Activation products from the coolant water are accumulated in an ion exchange resin which is removed and disposed of on a periodic basis. The annual volume of resin required to control

pool water quality is estimated to be less than 0.1 cubic meters while the estimate of other solid waste is 1-2 cubic meters per year.

Processing and disposal of fuel elements is not considered a significant activity of this facility. Projections from the operation schedules of similar types of reactor facilities indicate that less than 750 MW-days of burnup would be accumulated after 40 years. Intermediate storage of irradiated fuel shall be accomplished on-site in approved storage wells with sufficient shielding to insure that no noticeable environmental effects result from such storage. Ultimate disposal and processing of the fuel is a function of the Department of Energy.

C. Decommissioning Impacts

Studies such as NUREG CR-1756 contain detailed information for the radionuclide inventories expected after operation of a typical research reactor facility. Major isotopes of concern identified are Co^{60}, Zn^{65}, and C^{14}, although several other isotopes and such rare earth radionuclides as Eu^{152} are expected to be present in reactor materials and shield concrete.

Based on data from MSU reactor decommissioning, ATU estimates that less than 1000 ft^3 of radioactive waste would require disposal at the time of decommissioning. This waste would primarily consist of reactor structural components located inside the pool. By enlarging the pool diameter, ATU expects to eliminate the need to remove concrete from around the pool. MSU had a small pool and was required to remove a relatively small ($<<$ 1000 ft^3) amount of concrete for disposal as low level radioactive waste.

ATU would be responsible fro decommissioning and dismantling of the reactor. A decommissioning fund has been established, per State Board of Higher Education requirements, to set aside $47000 per year for the next ten years. It is expected that all low-level radioactive waste produced through the decommissioning efforts will be able to be removed from ATU in one shipment. This shipment would be made to an authorized, licensed low-level radioactive waste disposal site by a licensed carrier in accordance with the rules and regulations of the State of Arkansas and the Nuclear Regulatory Commission. ATU expects that the regional low level radioactive waste compact disposal site proposed for Nebraska (approximately 600 mi.) would be accepting waste by that time. One truckload (40 ft. long van) would be adequate for the shipment of all low level radioactive waste associated with decommissioning.

The only high-level radioactive waste associated with decommissioning of the reactor would be the reactor fuel elements, which remain the property of DOE. Final disposition of the fuel elements during decommissioning would therefore be a DOE responsibility under tne University Reactor Fuel Assistance Program. Additional NEPA reviews, if required, would be undertaken at that time to address decommissioning options and impacts.

D. Environmental Effects of Accidents

Accidents ranging from failure of experiments to the largest core damage and fission product release considered possible result in doses of only a small fraction of 10 CFR Part 20 guidelines and are considered negligible with respect to the environment. Credible accident analysis for TRIGA and TRIGA fueled reactors are presented in NUREG CR-2387 (PNL-4028).

The following accident scenarios were evaluated in the ATU SAR. Each has a probability of occurrence less than one in one million, and does not have the potential for catastrophic consequences.

Reactivity Insertion Accidents. During pulsing operation, reactivity is inserted rapidly into the reactor and is a designed feature of the fuel performance. The U-ZrH (H/ZR = 1.6) fuel used in the TRIGA reactor has a strong, prompt negative temperature coefficient. This temperature coefficient terminates the nuclear excursion. The maximum reactivity that may be inserted by the pulsing operation is limited through the use of mechanical stops to limit the pulse rod withdrawal. Movement of the pulse rod above 1 kW is prevented by circuit design.

There would be no loss of clad integrity, damage to the fuel, or radiological consequences as a result of the maximum reactivity (2.25 %Δk/k) insertion. Substantial volume changes associated with phase transformations of the fuel above 1250 C could result in a cladding failure. The peak fuel temperature of 662 C at the conclusion of the maximum reactivity pulse is well below the temperature (1250 C) at which Zr-H becomes unstable, eliminating the potential for fuel cladding rupture. The clad could also fail as result of high internal gas pressures produced by the hydrogen released from the fuel, the fission product gases, and the expansion of air at the time of the peak fuel temperature. However, at 662 C the ultimate tensile strength of the 0.02 in. thick, type 304 stainless steel cladding is above 41500 psi, and the yield strength is 28500 psi, which far exceeds the stress produced by the total internal pressure for the H-Zr fuel (1893 psi).

Loss of Reactor Coolant. Loss of reactor coolant water is prevented by installation of siphon breakers (1/2 in. diameter holes located 1 ft. below the inlet and outlet water lines) and by the use of a recirculating pump which does not have sufficient suction head to drain the tank. Due to the design of the ATUTR there is no credible process that would lead to loss of the pool water and uncovering of the core. The aluminum pool tank will be surrounded by 2 ft of reinforced concrete and this is enclosed by a steel tank 14 ft in diameter and 27 ft deep. The steel tank will be surrounded by at least 1 ft of concrete. Ground water level above the bottom of the tank would also prevent the water from being completely drained out of the tank in the case of a tank failure.

In the TRIGA reactor water is a major moderator of neutrons, and the loss of water will terminate the chain reaction. If the coolant is lost during reactor operation, fission product decay heat would be removed by natural convective air flow through the core. If the coolant was lost immediately at shutdown (zero cooling time), the maximum fuel temperature would reach 329 C, which would not result in any damage to the fuel. At this temperature, the stress on the clad would be 1186 psi, which as previously discussed is well within the clad yield strength, and would not result in any damage to the cladding. This would contain the fission products preventing a release to the environment. The fuel temperature could reach 900 C without substantial yielding of the clad, which would require operation at 1540 kW, well above the 250 kW proposed operation level.

The radiological hazard associated with the loss of shielding water scenario has been calculated for a direct radiation location 6.4 m above the unshielded reactor core near the top of the reactor tank, and for a scattered radiation location at floor level. The calculations for the second location assume that the radiation is reflected by a thick concrete ceiling 7 m above the top of the reactor tank (actual roof structure would give less backscattering; at least by a factor of two). Table 1 shows the occupational dose rates for two cases: loss of shielding water after reactor operation for 10 hours, and loss of shielding water after reactor operation for 1000 hours. Gamma radiation from fission product decay and activation of structural materials and attenuation of gamma radiation through fuel element end pieces and the upper grid plate were taken into account in this calculation.

Based on these calculations, the greatest potential dose of 1320 rem/hour at the top of the tank would exceed the NRC occupational limit of 5 rem/year in 14 seconds of exposure. However, anyone observing the loss of shielding water from the top of the tank would know to leave the building immediately, thereby greatly limiting exposure. The maximum, worst case scattered dose rate inside the reactor

Table 1 Radiation Dose Rates for Loss of Shield Water (Fission Product Decay and Activation of Strucural Materials with shielding)

	Radiation (rad/hr)			
	Direct		Scattered	
Operation Time→ / Decay time ↓	10 hr	1000 hr	10 hr	1000 hr
1 minute	1240.0	1320.0	0.451	0.480
10 minutes	223.0	303.0	0.081	0.110
1 hour	79.5	157.0	0.0288	0.057
1 day	7.55	58.9	0.00274	0.0214
1 week	0.863	24.4	0.00031	0.0088
1 month	0.157	9.2	0.00006	0.0033

room is about 480 mrem/hour. Dose rate received by general public even for the worst case would be an order of magnitude smaller than the above value.

Fission Product Release from Clad Rupture. The SAR also evaluated the potential for release of fission products to the environment due to clad rupture during fuel handling. As the "maximum hypothetical accident" the SAR assumed that a fuel element from the region of highest power density failed in air after a long exposure at full power. The fission products released would be those that collected in the fuel-clad gap during normal operation. The inventory of radioactive noble gases and halogens in the fuel element were calculated assuming continuous operation at 25 kW for 5 years. The analysis assumed release of 100% of noble gases and halogens in the gap, zero ventilation in the room, and occupant exposure in the room for 10 minutes while evacuation takes place.

The total whole body 10 minute exposure for a reactor room occupant was calculated to be 0.56 mr, well within the 1250 mrem/quarter occupational dose limit of 10 CFR Part 20. Thyroid exposure of 4.52 rem was calculated based upon isotope inhalation for 10 minutes (compared to the NRC standard of 30 rem/year, which is based on Appendix B concentrations).

Exposure to the public from the above accident was calculated assuming release through the exhaust vent at a rate of 850 cfm for an hour at roof level, with dilution, and that the concentration of radioisotopes in the reactor room does not

decrease with time. Under these conditions, the total body dose due to gamma and beta exposures was calculated as 0.25 mrem (compared to the NRC standard of 50 mrems/yr). The one hour thyroid dose was obtained as 8.92 mrems (compared to the NRC standard of 1500 mrem/yr).

E. Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation involves the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of the unavoidable effects.

F. Alternatives to Construction and Operation of the Facility

There are no suitable or more economical alternatives which can accomplish both the educational and the research objectives of this facility. These objectives include the training of engineering students and power plant operators in the operation of nuclear reactors, the operation as a source of neutrons for neutron activation analysis or neutron radiography and other activities related to education and radioisotope applications.

G. Long-Term Effects of Facility Construction and Operation

The long-term effects of a research facility such as the Center for Energy and Environmental Studies are considered to be beneficial as a result of the contribution to scientific knowledge and training. The impact on the environment associated with this facility is estimated to be the annual release of 5-20 Curies of argon-41 gas, 1-2 cubic meters of solid LLRW, and small quantities (less than 1,000 gallons) of liquid radioactive waste.

H. Cost and Benefits of Facility and Alternatives

The cost for this facility is projected at $1.25 million with an annual impact on the environment as outlined in G. above from the operation of the facility. The

benefits include, but are not limited to: the applications of neutron activation analyses, production of neutron beams for research and or application, production of short-lived radioisotopes, education of students and public, and training of operating personnel. Some of these activities could be conducted using particle accelerators or radioactive sources, but these alternatives are also costly and less effective and in some cases not applicable. There is no reasonable alternative to a nuclear research reactor of this type for conducting the broad spectrum of activities previously mentioned.