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Office of the President

September 30, 1991

Mr. Alexander Adams, Jr.
Non-Power Reactor, Decommissioning, and
Environmental Project Directorate
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Project Number 677. Application for a Construction Permit/Operating License for the Arkansas Tech University TRIGA Reactor Facility

Dear Mr. Adams:

Enclosed are responses to questions concerning Arkansas Tech University's application for a construction permit/operating license for a TRIGA research reactor. Please contact Dr. Jack Hamm at (501) 968-0353 for additional information or clarification of these responses.

Sincerely,

Kenneth Kersh

Kenneth Kersh

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

ARKANSAS TECH UNIVERSITY

1. Introduction page 1.1;

   a. Please use the standard notation of ($\%\Delta k/k$), and (_____\$) throughout your documentation.

      This change is made on pages 1-1, 1-2 (7 places), 1-3, 3-29, 3-33, 3-34 (8), 3-35 (3), 3-36 (2), 4-7, 7-1, 7-2 (5), 7-5, 7-7 of the SAR and pages 1-3, 3-1, 3-2 (2), 3-3 (3), 3-4, 3-11 (3), 3-12, 6-10 of 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 is changed to read "Reactor experimental facilities will include a rotary specimen rack, a pneumatic transfer system, and a central core irradiation tube (Central Thimble).

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

      (1) The notation of $\%\Delta k/k$ and dollar. (See comment No. 1.)

      Changes are made.

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

      Zr/H corrected to H/Zr.

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

      0.11 $\%\Delta k/k$ corrected to 0.011 $\%\Delta k/k$

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

The void coefficient of reactivity is 0.001 %Δk/k/(1% void) [GA 2070].

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

Changes are 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.

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 (800 ft) to the reactor site. The Missouri-Pacific Railroad passes one half 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.

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 150 ft south west of the reactor site. The nearest permanent residence is located approximately

500 ft north of the reactor building. The maximum potential exposure to the public from argon-41 (routine operations) and fission products (accident situations) at these sites will be less than the exposures in the lee of the building. Dilution as well as decay of the isotopes will reduce the concentrations at these sites. The exposure is expected to be less than that in the lee of the building. As shown in sections 6.4 and 7.3 the exposures in the lee of the building is within the 10 CFR 20 limits.

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.

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

Corrected (page 3-1).

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 corrected paragraph is 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. This is the fuel we expect to use.

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 instrumented fuel elements will be present at the ATUR, and where will they be located in the core?

Instrumented fuel elements: There are two instrumented fuel elements. They are located at B5 and C11 (see figure 3-6 showing Typical core configuration). These are expected to be new non-gapped elements obtained from GA.

Neutron detectors: There are three neutron detectors; one fission and two ionization chambers. All detectors are expected to 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. The graphite elements are not expected to be placed at any other location. If a need arises for an alternate configuration, a detailed safety analysis will be performed for the new configuration and approval shall be obtained from the Reactor Safety and Utilization committee .

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 are the ones designed by GA and used earlier at the Michigan State reactor 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 ($1.88 \times 3.7 \times 10^{10} \times 70/10^6$).

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?

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 h = 7.62 m (25 ft) is 7.5 N. The stress on the tank wall of radius r = 1.524 m (5 ft) and thickness e = 0.0762 m (0.25 in.) is 17.9 MPa (2600 psi) [$\sigma = \gamma hr/e$]. This is less than the yield strength for aluminum which is 4000 psi. The bottom of the tank will be made of 0.5 in. aluminum. Other structural reinforcements will also be present. The tank will be designed and fabricated by GA

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.

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

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 corrected on page 3-13.

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 is added to the SAR. 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 reads 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.

The procedure will take into account the following points. (1) Avoid the movement or placement of the cask over the reactor or its control rods, if possible. (2) Move the cask from the edge of the tank to the top of the isotope removal tube over the shortest path, which will be over the central channel assembly. (3) Fasten the cask to a small crane to prevent the cask from falling into the tank. (4) Use additional reinforcements to the grating over the reactor tank.

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 last four sentences of the second paragraph on page 3-14 are removed from the SAR. The change for this paragraph is given below.

Several Experimental facilities are available in the reactor. For isotope production, a rotary specimen rack is located in a well in the reflector assembly. A pneumatically operated "rabbit" transfer system, which penetrates the reactor core lattice, is provided for the production of very short-lived radioisotopes. A central thimble that enters the center of the core lattice makes possible the extraction of a highly collimated beam of radiation or insertion of small samples into the region of maximum flux. There are no beam ports in this arrangement of the reactor. ~~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. Large samples in water tight containers may be lowered into the space around the reflector for irradiation. If necessary, the core tank may be used to store samples after irradiation.~~

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.

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 are expected to 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 following sentence is added to the SAR on

page 3-24, "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.

This change is made on page 3-25.

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 sentence is changed to "The rod may be withdrawn from zero to a maximum of 15 inches from the core; however, a mechanical stop is used to restrict the travel so that the maximum permissible step insertion of reactivity (1.4 %Δk/k or 2.0 $) 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 four, 10 position racks in the tank. To facilitate extra storage, two racks may be attached to the same connecting rods by locating one rack at a different vertical level and offsetting the horizontal position slightly. A minimum of 8 ft of water above the racks will be maintained to provide shielding. The number and positioning of the fuel is such that the configuration will remain subcritical.

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 16 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.

Note: In section 6.2.2 . page 6-5 change the sentence "Nineteen elements may be stored in each pipe, and water added to provide radiation shielding." to "Nineteen elements may be stored in each pipe, and water may be added if necessary to provide radiation shielding."

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 are removed from this chapter and references to them are changed on pages 3-33 (1-3 to 1,2) and page 3-34 (4 removed).

   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.

   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. The 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. Correction made on pages 3-33 and 7-1.

7. Section 3.4;

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

      This is a typical value and 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 is changed as follows: "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.

This table and values are from Texas SAR and for their reactor.

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.

Change the paragraph as follows. 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 thoroughly 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?"

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.

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."

Yes. Sentence corrected to read "Maximum steady-state power level is set at 250 kW (thermal), and maximum fuel temperature is set at 500 C and if either 250 kW or 500 C is exceeded the reactor will scram.".

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.

9. Section 4.1;

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

Table 4-1 is changed as follows. The set point on the wide range log power channel is an interlock. The set points on the percent power channels #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 are changed to 100 % (250 kW).

Table 4-1 (2 places) shown above.

section 4.2.3 (1) " Scrams are set at 100 % of full power on both channels",

section 4.4, sentence changed as follows:

1. **Power on one of the two percent power (safety) channel exceeds 100% of the full power (250 kW) during steady state operation and power on one percent power channel exceeds 100% of full power during pulsing operation.**

The sentence " The reactor may be operated at power levels not to exceed 275 kW during short periods for test or calibration. " is removed from page 2-4 of Technical Specifications.

Table on page 3-6 of the Technical specifications.

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 full 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.

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.

If the bulk pool temperature goes above 50 C an alarm is produced which is visually or aurally annunciated at the console. Page 4-9. The selector switch position is changed in figure 4.3. Some of the Bis-table Trips (B/T) in figures 4.2 and 4.3 are located incorrectly. These figures are corrected.

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.

No. The reactor period signal does not provide a reactor scram. The temperature and power channels will provide the required redundant scrams.

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?

11. Section 4.3;

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

All control/safety rods are scramable. Control rod is also scramable.GA

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 the 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 the a scram signal is generated??? at the scram set point (100% of maximum power). The scram set point is the same in both steady-state and transient modes. The % power linear amplifier (ELC 266-2120, GA) provides a 10 V output for detector currents between 10^{-6} A and 10[illegible] 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.GA

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 is changed to read "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 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 the 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.

    Yes. The normal water level in the tank is 0.5 ft below the top edge of the pool tank. The top edge of the pool tank is 0.5 ft below the floor surface. The heat exchanger is mounted 5.0 ft above the floor on the south wall. The pump an he demineralizer are 1 .0 ft above the surface of the floor.

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

    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 maximum amount of water that will leak into the secondary system is about 950 gallons. The radiological hazard would not be

greater than that discussed in section 14.e.1. The pool water level indicator provides an alarm when the water is 0.5 ft below normal. At this point the amount of water lost would be only about 294 gallons. This reduces the radiological hazard. Also, the water leaking into the secondary will remain in the secondary system. This is a closed system with check valves preventing backflow into intake.

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 pool from a small tank through the auxiliary tap shown in Figure 5-1. 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.

The primary water level indicator provides alarm if water is lost from the system. Furthermore, the siphon break allows only 950 gallons of water from being lost.

A major failure of the tank is not expected. The concrete, shielding structure and a steel tank surrounding the pool tank will prevent such a failure. The steel tank will isolate the pool water from the ground water.

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.

f. How often is the radio-nuclide level in 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?

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. m of the normal operating level. Alarms are also generated if the water level is 0.5 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. 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 Keff for fuel storage pits containing 19 elements, both dry and water flooded. How does one transfer the fuel from the reactor; discuss procedures.

c. Section 6.2.3, Ventilation system; please discuss:

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

The dampers on the air ducts are normally closed due to the action of gravity and springs. They are held open by an electrically controlled actuator. The loss of electricity will close the dampers.

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

The fresh air intake is at ground level on the south-east side of the reactor building (figure 6-1). The exhaust stack is at roof level at the center of the reactor building and about 10 m above the ground level.

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

A high radiation level signal from the air particulate monitor in the roof stack will sound an alarm in the reactor control room. This signal will activate a circuit that will remove electrical power to all the actuators holding the dampers 1 through 6 (Figure 6-3) open. Thus the dampers will be closed automatically. The dampers may also be closed by the operator manually turning the electrical power off.

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

Yes. The operating procedures will specify meeting this requirement.

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

The high radiation level resulting from the fuel element failure (Section 7.3) will initiate an alarm in the Control room. The signal will initiate the automatic shut down of all the dampers, and thus completely isolating the reactor room from the rest of the building. The reactor room doors and windows will remain closed during a fuel element handling operation. Doors to the control room from the reactor room may be opened briefly for personnel to evacuate. The reactor will be secured if it was not secured. Flow shall be diverted through the HEPA filter only after permission from the Reactor Safety and Utilization Committee is obtained. This committee shall issue such a permission only after consulting with the Radiation Safety Committee. It may be necessary to turn-on the exhaust fan to maintain a negative pressure in the reactor room. Flow through the HEPA filter will reduce the particulate contamination in the exhaust and the reactor room.

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?

The radiation sensor and the exhaust fan are located in the reactor room exhaust stack. Therefore, air from the control room is exhausted through the roof stack in the reactor room.

The receiver/sender end of the pneumatic transfer system is located in room 3. The air from room 3 is exhausted through the reactor room so that any radioactivity may be exhausted through the roof stack. The radiation monitor is also located in the roof stack.

(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?

Air from the main air supply duct enters reactor room through damper 1 (Figure 6-3). Air is distributed into the reactor room through 4 wall ventilators and into the control room through damper 2. Air from Control room is exhausted into the reactor room through damper 4. Air from the main duct enters room 3 and then exhausts into the reactor room. Air from control room, room 3 and the Reactor room is exhausted through the roof stack in the reactor room.

Air from the reactor room (including room 3 and control room) is exhausted through the roof stack. Since a negative pressure is maintained in the reactor room air will leak into the reactor room from the rest of the building. When the dampers are closed there is no air flow into or out of the reactor room. Therefore, reactor room air is not forced to 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?

Air is routed through the HEPA filter to reduce the contamination the reactor room air. It may be necessary to turn-on the exhaust fan to maintain a negative pressure in the reactor room. In this case the HEPA filter removes contamination from the exhaust air. Switch over to the HEPA system is not automatic. This is done manually if permitted by the Reactor Utilization Safety Committee and the Radiation Safety Committee and after evaluation of any radiological release hazards.

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

Yes, all facility (reactor building fans) will shut down upon reactor room isolation. However, it may be necessary to run the exhaust stack fan to maintain a negative pressure in the reactor room. This will be done with the permission of the Reactor Safety and Utilization Committee and after an evaluation of the health hazards.

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 detector is a GM-tube covering a range of at least 1 to 100000 counts per 30 s . It will be located in the roof stack. 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.

Calculations (section 6.4 of the SAR) show that the argon- 41 concentration in air released from the reactor room to the outside is 3.3×10^{-8} μCi/ml for a continuous single shift operation for a year. Actual operation time may be only about 20% of the single shift operation time. The above number reduces to 1.45×10^{-9} μCi/ml when averaged over a year. Also, the reactor may not be operated at full power all the time. These factors will reduce the actual production rate of argon-41 and therefore, will reduce the above number. The Argon-41 release rate is less than the MPC of 4×10^{-8} μCi/ml listed in 10 CFR Part 20. Experience at Michigan State University also shows that the release was roughly 40 % of the MPC without a pneumatic transfer system. Since there is sufficient reason to believe that the release rate from this facility will not exceed the 10 CFR part 20 appendix B limit this was not specified as a requirement in the Technical Specifications. However, a monitoring system will be available in the facility.

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. The sink in room 2 and the sink, hood, and shower in room 3 have individual containers below them that hold the waste water. The shower sits on a raised platform. The waste water will be held in these containers until proper disposal could be arranged. Water level in these containers will be checked at least monthly or more frequently as needed. The liquid waste will be disposed off into sewer drains only if approved by and under the supervision of health physics personnel. To dispose the waste liquid the containers are manually lifted and carried to a disposal sink and then drained.

(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. (The health physics personnel report directly to the Radiation Safety Committee.) The analysis will be performed on equipment to determine alpha, beta and gamma levels. The waste will be disposed off according to criteria specified in 10 CFR part 20, specifically 20.303.

(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?

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.

Air-borne radioactive materials coming from experiments will be disposed off according to experimental plans pre-approved by the Radiation Safety committee. Air in room 3 is exhausted through the reactor room exhaust stack to monitor any contamination and thus prevent inadvertent release to the unrestricted environment.

One source of liquid radioactive effluents is from small quantities of liquid irradiated samples. This will not be disposed off in the sink but will be disposed off according to pre-approved experimental plans. Water from the sink, hood, and shower will be held in containers until proper analysis and disposal could be arranged (see section 16.e.1)

g. Section 6.4.1, Argon;

(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.

Most highly exposed individual in the lee of the building. The exhaust stack is located in the middle of the reactor building at roof level and at an elevation of 10. m above ground level. The calculations are given in attachment #1.

(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.

The maximum potential exposure to the public from argon-41 routine operations at the site of the nearest permanent residence and the nearest temporary residence will be less than the exposures in the lee of the building. Dilution as well as decay of the isotopes will reduce the concentrations at these sites. The exposure is expected to be less by at least a factor of 10 than that in the lee of the building. As shown in sections 6.4 and 7.3 the exposures in the lee of the building is within the 10 CFR 20 limits.

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

It is assumed that the activated air from the experimental facilities is removed at an exhaust rate of about 10 cfm. This is a conservative value considering the effective volume of the experimental facilities. The actual release rate will be reduced by a factor of 25 because of the fact that experiments will occupy more than 80 % of the volume (leaving only 20 % of the air to be irradiated) and that the reactor

will be operated at only about 20% of the continuous full power operation time in a typical year (see 15.g.6).

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

(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.

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

Please see attachment #1.

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

There are three experimental facilities described, the central thimble, the rotary specimen rack, and the pneumatic transfer system. When an experiment is located in one of these locations more then 80% of the air will be displaced, leaving only a maximum of about 20% of the air to be irradiated. The reactor is not expected to be operated at full power for less than 20% of the continuous single shift operation time. Three hours a day, three days a week, 4 weeks a month , and 10 months a year is 18% of the continuous single shift operation time and is only about 5% of the continuous full year operation. A conservative number of 20% was assumed. The above two factors reduces the argon-41 production rate by a factor of 25 ($1/(0.2 \times 0.2)$). Also, the central thimble is normally filled with water. The water will be removed and the region filled with air only when necessary. The exhaust rate from the pneumatic transfer system may be smaller then the number assumed when the system is not operating. The fluxes assumed are for the 250 kW operation and at the region very close to the reactor core. For regions of the central

thimble and the pneumatic system far away from the reactor core the fluxes may be smaller. These factors, if taken into account, will further increase the above factor of 25 reduction in the argon-41 production from the experimental facilities.

(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 room 3 is exhausted through the exhausted 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.

The 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.

Without the diffuser the transit time from the core to the pool surface is 109.4 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 the discharge of 11669 cm^3/s is spread into only 10% of the pool cross sectional area and then rises up through this cross section, the effective upward velocity would be approximately 1.6 cm/s (11669 $cm^3/s \div 7297\ cm^2$). The transit time for the water to reach the surface of the pool is 400 s (640 $\div$ 1.6 cm/s). In this time the nitrogen-16 decays by a factor of 5.6×10^{-17}. This estimate shows that the diffuser reduces the nitrogen-16 concentration by about 11 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 ($2[illegible]5\ cm^3/s \times 21.36\ s \div 49087\ cm^2$). 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. The decay of nitrogen-16 during the formation of a larger thickness disk will reduce the concentration of nitrogen-16 in the water. For example, in the scenario described in 16.h the decay of nitrogen-16 will be more significant since this reduces the concentration to negligibly small levels. The effect of a larger disk would not be important at all.

The average nitrogen-16 concentration in the water during the time it spreads into a thin disk is N = 524 atoms/ml (6.16). The transport time of 109.4 s was used to obtain this number and the resulting dose rate was about 600 μrad/hr. A 50% increase in the transport time will reduce the dose rate to about 3.6 μrad/hr (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.

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 section, 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, n/E = 0.00119 moles/MW-day is obtained from reference 5 (((obtain primary reference, Texas or GA))), 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.

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 maximum 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 (about 13.3%). Three times this value is used in the calculation. The resulting fis-

sion 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 950C 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.

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

We expect to use standard non-gapped fuel.

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

(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 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 pulse 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.

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.

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.

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.

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 burnup of 1.25 MW-yr corresponds to about 19.2% burnup. The maximum burnup for a standard TRIGA fuel element is 4.5 MW-days/element. This translates to about 13.3% burnup. 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.

A revised copy of section 7.3 is attached (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 $r \to \infty$ in equation 7.26. The 10 minute exposure in this case would be 18.2 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.

(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.

(4) 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). The Thyroid dose reduces by the factor 10 to 0.505 rems. This reduces the total Thyroid dose for 10 minutes to 5.16 rems. The one hour exposure to the general public in step 7 of 7.3.2 reduces to 9.9×10^{-3} rems.

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 0.035 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 9.9×10^{-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?

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.

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.

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. The following sentence is 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 modified paragraph of section 8.2.4 is given below.

"A person with special training to supervise 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.

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 is 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 ac-

tivities 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 Reactor Safety and Utilization Committee. Reactor Safety and Utilization Committee will follow guidance provided by the "Review of experiments for research reactors" (ANS 15-6/ANSI N401) 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?

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. Add the above two sentences to the second paragraph on page 8-8 of the SAR.

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?

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) Ab-

normal Occurrence Reports such as spills, (8) Deactivation Plans/Reports, (9) Technical Specifications, (10) Design Changes and Updated Drawings. (section 14.4)

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.

It is ATU's policy to establish a program that goes beyond minimum requirements. The ATU Radiation Safety Program is designed to provide maximum research and educational opportunities while minimizing personnel exposure to radiation and radioactive material. The ATU Radiation Safety policies are designed to keep personnel exposures as low as reasonably achievable (ALARA). The Radiation Safety procedures were developed to implement the policies for strict compliance with applicable federal and state regulations and to encompass ALARA principles.

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.

The first paragraph on page 10-1 is modified as follows.

"This section describes the elements of the Arkansas Tech University radiation protection program and establishes the guidelines to be applied to provide an acceptable level of radiation protection for personnel at the reactor facility and the general public. Regulatory requirements for there guidelines are specified in 10 CFR Part 20 and is consistent with keeping exposures and releases as low as reasonably achievable."

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

The office of the Radiation Safety Officer is responsible for the implementation of the radiation protection program for the university. The second paragraph on page 10-1 is modified as follows.

"The Arkansas Tech University radiation protection program is based on the requirements in 10 CFR Part 20 and the commitment to keep exposures to personnel and the general public as low as reasonably achievable (ALARA). This commitment forms one of the bases for the operating procedures and the procedures on radiation protection [ANSI-15.11, 3.1]. The radiation protection program for the university is implemented through the office of the Radiation Safety Officer who chairs the Radiation Safety Committee (Figure 8-1)."

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.

The first paragraph of page 10-2 is modified as follows.

"The Facility Director has the authority and the Reactor Supervisor has responsibility to implement the radiation protection program for the reactor facility. This responsibility includes the authority to act on questions of radiation protection, the acquisition of appropriate training for radiation protection, and the reporting to management of problems associated with radiation protection [ANSI-15.11, 3.2]."

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

Please see 17.b.

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.

Please see 17.c.

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.

ATU established an environmental survey program at the beginning of the second quarter of 1990. The program consists of 8 TLD's positioned around the vicinity of the proposed research reactor. (attach figure) The results of the program to date are to be used to establish a baseline radiation level prior to construction of the TRIGA reactor. This background is due to naturally occurring radionuclides and Entergy Inc. ANO units I and II.

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.

Review will be made by the Facility Director.

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

The following sentence is added to section 10.7

"ATU will collect and retain for the lifetime of the reactor information about radiological events that will have significant effect on the decommissioning of the facility during construction and operation of the facility. (see also 17.i)

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.

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.

The modified paragraph is given below.

"An Emergency Plan, as required by 10 CFR Part 50, will be established, maintained, and implemented by the Facility Director. The plan and the emergency response procedures will exist as a separate document. The Arkansas Tech University radiation protection program and emergency plan will be integrally related. A review and partial assessment of the emergency plan and the radiation protec-

tion program will occur each year such that a complete assessment occurs during a two year period."

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.

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 is 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.

Argon-41 is produced by the activation (n, γ) of argon-40 present in the air in experimental facilities or the air dissolved in water. Calculations (reference ATU SAR page 6-9) show that the activity concentration of argon-41 is 2×10^{-7} μCi/ml (= Ci/m^3) from the pool water and $9. \times 10^{-5}$ Ci/m^3 from experimental facilities at full power (250 kW). These are equilibrium concentrations of the activity released into the reactor room volume of 394 m^3. The building exhaust rate is assumed to be 0.401 m^3/s. For the release rate from the experimental facilities (central thimble, rotary specimen rack and pneumatic transfer system) an exhaust rate of 4.75×10^{-3} m^3 (about 10 cfm) is assumed. Experiments usually replace about 80% of the air in these facilities. This will reduce the activity concentration coming from the experimental facilities to 1.8×10^{-5} Ci/m^3 {$9. \times 10^{-5} \times (1-0.8)$}. The total concentration of argon-41 from the above two sources is about 1.82×10^{-5} Ci/m^3.

The release rate to the environment is about 7.2×10^{-6} Ci/s (1.82×10^{-5} Ci/m^3 $\times$ 0.401 m^3/s) . The total release for continuous single shift operation for a year is 50 Ci (multiply above number by 1920×3600 s/year). Actual operation time may be less than 20 % of the single shift operation time, and this would give a release rate less than 10 Ci. The fact that the reactor may not be operated at full power during this time is not taken into account. Michigan State reported an argon-41 release rate of 400 μCi/year. They did not have the pneumatic transfer system. OSU reactor (1 MW) reported a release of 7.78 Ci/year.

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...".

Total volume of all solid radioactive waste is projected to be 1 to 2 m^3 per year. Total volume produced at the university may be about 4 to 5 m^3 per year. (More like one-half of university volume.)

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

A small portion (3-10 gpm) of the reactor coolant is diverted through a purifying loop which contains an ion-exchange demineralizer. Activated impurities are removed from the coolant by the resin. The resin with activation products accumulated in it is replaced yearly or as needed. This resin is a solid radioactive waste which will be disposed off according to the requirements of 10 CFR Part 20.

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?

Liquid waste from sinks and shower will be stored in individual containers. Representative samples from these containers would be collected and analyzed by standard techniques. (15.e) When the concentration of radioactive materials in the waste are less than the guidelines values of 10 CFR 20.303, the liquids may be discharged directly to the sewer. However, to meet ALARA requirements one would try to wait till concentrations are less than the guideline values. A number like 10 % of the guideline values may be used if cost effective and is possible.

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.

ATU will be responsible for storing, packaging and shipping of irradiated fuel. Irradiated fuel elements will be stored in the reactor

core, storage racks or the storage pipes until shipped. ATU will maintain all required monitoring for a special nuclear material.

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.

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

At the time of decommissioning, Arkansas expects to ship low level radioactive waste to a disposal site in Nebraska (approx 600 mi). One truck load (40 ft long van) would be adequate for the shipment of all radioactive waste associated with decommissioning.

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.

Occupational Exposures

Radiological contributions are caused mostly by argon-41 and nitrogen-16. Argon-41 is produced by the activation (n, γ) of argon-40 present in the air in experimental facilities or the air dissolved in water. Calculations (reference ATU SAR page 6-9) show that the activity concentration of argon-41 is 2×10^{-7} μCi/ml (=Ci/m^3) from the pool water and $9. \times 10^{-5}$ Ci/m^3 from experimental facilities at full power (250 kW). These are equilibrium concentrations of the activity released into the reactor room volume of 394 m^3. The building exhaust rate is assumed to be 0.401 m^3/s. For the release rate from the experimental facilities (central thimble, rotary specimen rack and pneumatic transfer system) an exhaust rate of 4.75×10^{-3} m^3 (about 10 cfm) is assumed. Experiments usually replace about 80% of the air in these facilities. This will reduce the activity concentration coming from the experimetal facilities to 1.8×10^{-5} Ci/m^3 $\{9. \times 10^{-5} \times (1-0.8)\}$. The total concentration of argon-41 from the above two sources is about 1.82×10^{-5} Ci/m^3.

The dose rate from this activity concentration in the reactor room is calculated as follows. A hemisphere of volume equal to the reactor room volume has a radius R = 5.7 m. The exposure rate (R/hr) or approximately » dose rate rad/hr received by a person covered by a hemisphere of radius R (m) and which contains an activity concentration of A (Ci/m^3) is given by (reference Cember, Introduction to Health Physics, page 167)

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

where the source strength, Γ = 0.66 R.m^2/hr.Ci for argon-41 (reference Radiological Health Handbook, and Cember, page 164) and the linear energy absorption coefficient, μ = 0.0035 m^{-1}. For $\mu R << 1$ the quantity in parenthesis reduces to R and the above equation reduces to

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

The dose rate from argon-41 is 4.3×10^{-4} rad/hr. The dose from continuous single shift operation at full power for 1 year (1920 hr/year) is 826 mrad. Actual reactor operation time is estimated to be less than 20% of the above time. This will reduce the total occupational dose from argon-41 to about 165 mrad per year. This number will be further reduced by lower power levels (neutron fluxes), smaller air volumes, shorter operation times, and larger dilution factors.

Nitrogen-16 is produced by the activation (n, p) of oxygen-16 in the reactor core region. It takes the water approximately 109 s (reference ATU SAR page 6-14) to rise from the core region to the top of the pool. Due to the very short half life (7.11 s) the concentration of nitrogen-16 is significantly reduced before it leaves the pool. The saturation concentration of nitrogen-16 in the reactor room is estimated to be 1.68×10^{-8} Ci/m^3 (reference ATU SAR page 6-14). The diffuser which discharges water above the core region in the pool greately increses the transit time and this further reduces the concentration of nitrogen-16. The source strength, $\Gamma \approx 1.8$ R.m^2/hr.Ci for nitrogen-16 (reference Radiological Health Handbook using an averge gamma energy of 6 MeV). From equation 2 the dose rate from nitrogen-16 in the reactor room is obtained as 1.08×10^{-6} rad/hr. The annual occupational dose for continuous single shift operation at full power is 2.1 mrad. Accounting for the actual operation time of less than 20% reduces the occupational dose to less than 0.4 mrad.

The combined occupational annual dose from argon-41 and nitrogen-16 is about 166 mrad.

Public Dose

The radioisotope that is significant is the argon-41 carried by the exhaust discharge. The building exhaust rate is $q = 0.401$ m^3/s and the dilution factor in the lee of the building is given by (refernce Slade, D.H, (ed.), "Meteorology and Atomic Energy", USAEC Reactor Develop. and Tech. Div. Report TID-24190, DFSTI. Springfield, Virginia, 1968)

$$\psi(0) = \frac{1}{CSU} = 4.5\times10^{-3}\ \frac{s}{m^3} \tag{3}$$

where $C = 0.5$, S = building cross sectional area normal to wind = 170 m^2, and U = wind velocity = 2.62 m/s. The activity concentration in the outside air is given by $A_o = A \times q \times \psi(0) = 3.3 \times 10^{-8}$ Ci/m^3. For a person immersed in a semi-infinite cloud of radioactive gases equation 1 with $A = A_o$ reduces to

$$D(rad/hr) = A_o\,\Gamma\,2\,\Pi \left(\frac{1}{\mu}\right). \tag{4}$$

The dose rate obtained from equation 4 is about 3.89×10^{-5} rad/hr. The annual dose for continuous single shift operation at full power is 74.7 mrad. The actual operation time of less than 20% reduces the annual public dose from argon-41 to about 15 mrad.

Table 7-1 Radiation Dose Rates for Loss of Shield Water

	Radiation (rad/hr)			
	Direct		Scattered	
Operation Time →	10 hr	1000 hr	10 hr	1000 hr
Decay time ↓				
1 minute	6.6E+2	8.2E+2	2.1E−1	2.5E−1
1 hour	1.5E+2	3.0E+2	4.8E−2	9.4E−2
1 day	1.6E+1	1.1E+2	4.5E−3	3.5E−2
1 week	1.7E+0	4.7E+1	5.2E−4	1.4E−2
1 month	3.0E−1	1.7E+1	3.0E−5	5.4E−3

in the reactor room would be high but tolerable even immediately after loss of coolant and that emergency operations could be carried out with limited time of action. Since the direct radiation would be collimated upward, radiation levels outside 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. This inventory is conservative since actual burnup after 5 years is expected to be less than 22% of 1.25 MW-yrs.

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

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

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 above (b). The concentration, q_i (μCi/ml), of the radioisotope in the room which has a volume of 394 m^3 is given by equation 7.25 and the calculated values are shown in Table 7-3.

$$q_i\left(\frac{\mu Ci}{ml}\right) = \frac{Q_i \times 10^6\ (\mu Ci) \times 5.7 \times 10^{-7}}{394 \times 10^6\ cm^3} \qquad 7.25$$

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	1.63E–06	2.60E–03	3.80E+01	3.00E–01	1.14E–03
Kr-85m	4.01E–06	1.60E–01	6.00E+05	1.70E–04	1.80E–03
Kr-85	4.20E–07	2.20E–03	2.30E+01	5.00E–01	1.44E–04
Kr-87	7.77E–06	7.80E–01	5.30E+05	2.00E–05	2.01E–02
Kr-88	1.10E–05	2.00E+0	6.30E+05	5.80E–05	6.04E–02
Kr-89	1.43E–05	1.60E+0	6.00E+05	6.50E–05	6.62E–01
Xe-131m	1.22E–07	2.00E–02	3.00E+04	1.00E–03	1.10E–04
Xe-133m	5.81E–07	4.10E–02	2.40E+05	3.20E–04	1.60E–04
Xe-133	2.07E–05	4.60E–02	3.30E+05	2.80E–04	4.71E–03
Xe-135m	3.24E–06	4.30E–01	5.10E+05	1.20E–04	4.67E–03
Xe-135	2.03E–05	2.50E–01	5.60E+05	1.50E–04	1.53E–02
Xe-137	1.88E–05	1.60E–01	6.00E+05	1.70E–04	8.42E–03
Xe-138	1.92E–05	1.10E+0	5.60E+05	7.80E–05	6.52E–02
I-131	8.69E–06	3.80E–01	5.20E+05	1.26E–04	1.08E–02
I-132	1.29E–05	2.20E+0	6.50E+05	5.60E–05	7.58E–02
I-133	2.07E–05	6.10E–01	5.20E+05	1.05E–04	4.17E–02
I-134	2.33E–05	2.60E+0	7.00E+05	5.00E–05	1.51E–01
I-135	1.97E–05	1.50E+0	6.00E+05	6.70E–05	8.54E–02
Total Whole body					6.13E–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 hemisherical 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 =573 cm. The results for 10 minute exposure is given in Table 7-3.

The total whole body 10 minute exposure, immediately following a fuel element rupture in air, is 0.61 mr. This value is well within the requirements of 10 CFR Part 20. Since the actual burnup is expected to be less than 22 % of 1.25 MW-yr the actual exposure is not expected to be larger than 0.13 mr for 10 minutes

5. The Isotope Inhalation, I_i (Ci) is calculated assuming a respiration rate of 3.47×10^{-4} m^3/s [12] and is given by

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

Table 7-4 shows the average gamma ray energy and internal dose effectivity for each fission product isotope. The iodine isotopes inhaled would concentrate in the thyroid and the thyroid dose, D_{th}, is calculated by multiplying the iodine ingested by the corresponding internal dose effectivity factor. These are also shown in Table 7-4. The resultant thyroid exposure of 5.16 rems for 10 minutes is reasonable based on the conservative assumptions made. The actual value will be less than 22% of this value based on actual burnup.

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. The following assumptions are made for the analysis.

Table 7-4 Thyroid Dose

Isotope	q_i µCi/ml	I_i 10 min µCi	Effectivity Factor rem/Ci	Thyroid Dose rems
I-131	8.69E–06	1.81E+00	1486000	2.69E+0
I-132	1.29E–05	2.68E+00	52880	1.42E–01
I-133	2.07E–05	4.31E+00	395100	1.70E+0
I-134	2.33E–05	4.85E+00	25380	1.23E–01
I-135	1.97E–05	4.10E+00	123100	5.05E–01
Total Thyroid Dose				5.16E+0

1. When the stack radiation monitors detect a higher radiation level ventilation air flow is diverted through an absolute filter. Air intakes to the room would be isolated and a negative pressure is maintained in the reactor room to prevent radioactivity release except through the exhaust vent. Air flow through the exhaust under these conditions is 150 cfm (0.071 m^3/s).
2. The concentration of the radioisotopes would be reduced over time from their removal by the ventilation and also by the decay of the isotopes. The concentration also decreases due to dilution over distance. For the purpose of this analysis these factors are not taken into account.
3. The contribution to the dose rates of the decay products of the isotopes released are small and not taken into account.
4. The concentration, q_i (µCi/ml) multiplied by the stack flow rate (0.071 m^3/s) is assumed to be released through the stack (release rate).
5. The release occurs at roof level, about 10.0 m above grade and a dilution due to wind will be encountered. The dilution factor, $\psi(0)$, given by

$$\psi(0) = \frac{1}{CSU} = 4.5 \times 10^{-3} \frac{s}{m^3} \qquad 7.28$$

where C is a shape factor taken as 0.5 (experimentally determined to be between 0.5 and 0.67), S is the cross section of the building and is taken as 170 m^2 since the prevailing winds near the building are from east and east-northeast, and U the average wind speed is about 2.62 m/s. The release rate from step 4 i[illegible] diluted at the rate of $\psi(0)$ and the concentration in the environment is obtained by multiplying the release rate by the dilution factor.

6. The exposure to the general public may be calculated as outlined in step 4 of section 7.3.1, but assuming immersion in a semiinfinite cloud for the concentration released to the environment obtained above. Over a period of one hour the total whole body exposure is 0.035 mr.
7. A one hour thyroid dose is calculated by scaling the results of the occupational exposure by 60/10 (time correction) and 3.2×10^{-4} (dilution correction, $0.071 \times 4.5 \times 10^{-3}$). This is about 9.9×10^{-3} rems.

These calculations indicate that the exposure to the general public as a result of the fuel element failure in air after extended reactor operations would not be significant.

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