UNIVERSITY OF UTAH
NUCLEAR REACTOR FACILITY
LICENSE NO. R-126
DOCKET NO. 50-407

LICENSE RENEWAL APPLICATION

UPDATED

SAFETY ANALYSIS REPORT

JUNE 2011

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

Nuclear Engineering
THE UNIVERSITY OF UTAH

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United States Nuclear Regulatory Commission
Docket Control Desk
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Rockville, Maryland 20852-2738

50-407
R-126

SUBJECT: SUBMISSION OF SAFETY ANALYSIS REPORT, TECHNICAL SPECIFICATIONS AND RESPONSES TO NRC RAIs

06/08/2011

Dear Mr Wertz:

We have completed the revision of the UUTR Safety Analysis Report and Technical Specifications for 100kW UUTR relicensing.

The enclosed please find:

- Responses to the NRC questions regarding the Safety Analysis Report for UUTR as submitted in ~~June 2009~~ April 2011
- Final UUTR Safety Analysis Report
- Responses to the NRC questions regarding the Technical Specification for the UUTR
- Final Technical Specification for the UUTR (which is a Chapter 14 in the revised Safety Analysis Report)

Sincerely,

Tatjana Jevremovic, PhD
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Notary Public
JOANN COOK
201 Presidents Circle, Room 201
Salt Lake City, Utah 84112
My Commission Expires
September 14, 2011
State of Utah

State of Utah / County of Salt Lake

On this 8th day of June, in the year 2011, before me JoAnn Cook, a notary public, personally appeared *Tatjana Jevremovic,* proved on the basis of satisfactory evidence to be the person whose name is subscribed to this instrument, and acknowledged *she* executed the same. Witness my hand and official seal.

Notary Public

A020
NRR

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UUTR

Answers to NRC RAIs

Submitted June 2011

RAI 13

Your response to RAI No. 13 provided an updated SAR Chapter 1.5 which indicated that the fuel was enriched to 20%. However. Table 4.2-1 and Table 1.5-1 indicates that the fuel enrichment is less than 20%. Please clarify the fuel enrichment value.

Please see the **UUTR SAR 1.5**, Page 18:

- 'enriched to 20% '
- → revised to 'less than 20%.'

Your response to RAI No. 13 provided updated SAR Chapter 4.2.1.13, 4.2.1.14, and 4.5.3.1, which mentioned the presence of erbium in the fuel. However, updated SAR Table 4.2-1 does not include erbium. Please clarify whether erbium is present in the UUTR fuel and provide the updated SAR description and tables.

- The **UUTR** TRIGA fuel elements do not have erbium.
- Therefore, the **UUTR SAR 4.2.1.13, 4.2.1.14,** and **4.5.3.1** were revised accordingly.

RAI 21

Your response to RAI No. 21 provided updated information regarding the reactor core configuration that would yield the highest power density and fuel temperature. However, the updated SAR Chapter 4.5.2.3 does not explicitly state if this is the limiting core configuration that does yield the highest power density and fuel temperature. Please provide limiting core configuration information, including the power level (100kW), limiting power density, and any other pertinent core parameters.

UUTR SAR 4.5.2.3 is revised accordingly.

RAI 22

Your response to RAI No. 22 provided updated SAR Chapter 4.5.3.3, Prompt Negative Temperature Coefficient, 4.5.3.4, Moderator Temperature Coefficient; and, 4.5.3.5, Void Coefficient, and indicated that an eigenvalue of 1.0 was used for the reactivity computations. However, neither the temperature nor reactivity condition at this value (k_{eff}=1.0) was specified. Please clarify the computational methods used to calculate the Fuel, Moderator, and Void temperature Coefficients for the UUTR.

UUTR SAR 4.5.3.3 including **Table 4.5-8, 4.5-9, 4.5-10, Figure 4.5-10** and **Figure 4.5-11** are revised accordingly.

Your response to RAI No. 22 provided an updated SAR, Chapter 4.5.2, Reactor Core Physics Parameter, Table 4.5-11, which listed various temperature reactivity coefficients. Our analysis could not replicate the calculations.

Please review the analysis and advise if numerical values are correct or if changes are needed.

Table 4.5-10 is revised, **UUTR SAR 4.5.5.5, P**age 117.

Your response to RAI No. 22 provided an updated SAR, Chapter 4.5.2, Reactor Core Physics Parameter, Table 4.5-10,which indicates an increase in the k_{eff} from 293 degrees Kelvin (K) to 333 degrees K. Our review finds this increase is not consistent with other TRIGA reactors. Please verify the K_{eff} have been calculated correctly.

UUTR SAR 4.5.3.5, Table 4.5-10, Page 116 is revised accordingly.

RAI 41

Your response to RAI No. 41 provided information of the radiological risks from experiments. However, we could not identify any information relative to experiments involving special nuclear material. If the UUTR plans to perform fueled experiments, please provide an analysis indicating the safety precautions and limitations, including appropriate Technical Specifications (TSs), to ensure that this type of experiment is bounded by the Maximum Hypothetical Accident (MHA), or otherwise adequately controlled to ensure the safety of the workers and public.

We do not plan to perform the experiments using nuclear fuel; **UUTR TS 3.8.2** , Page 394, is revised accordingly.

RAI 57

Your response to RAI No. 57 provided an updated SAR Chapter 13, Accident Analysis. During our review of updated SAR Chapter 13.2.2, Insertion of Excess Reactivity, we could not identify the source of the temperature coefficients used in the excess reactivity accident scenario. We noted different temperature coefficients were listed in updated SAR Chapter 4.5.3.1 and SCALE-generated values tabulated in updated SAR Chapter 13.2.2. Please provide a reference for the source of the temperature coefficients used in the analysis.

The temperature coefficients as obtained from MCNP5 simulations and listed in **UUTR SAR 13.2.2** were revised: **Figure 13.2-4**, Page 346.

Your response to RAI No. 57 provided an updated SAR Chapter 13, Accident Analysis. In our review of updated SAR, Chapter 13, various codes were used including PARET-ANL, MCNP5, and SCALE. Furthermore updated SAR Table 13.2.9 does not indicate very good agreement between SCALE 5.1 and MCNP5. Please indicate which codes were used for each updated SAR Chapter 13 accident, the basis for their use, and any validation information versus other codes or actual measurement of the UUTR. Additionally, please define the neutronic codes used in updated SAR Chapter 4, Section 4.5, Nuclear Design, and ensure their application and use is consistent and acceptable for inputs provided to the updated SAR Chapter 13 analyses.

MCNP5 and PARET-ANL were used to calculate the reactor power change in the case of reactivity insertion. Also, PARET-ANL was used to calculate the maximum centerline fuel temperature and coolant temperature with $1.20 reactivity insertion. **(UUTR SAR 13.2.2).** MCNP5 was used to calculate temperature coefficients, reactor power, and neutron flux, and other reactor core parameters, **UUTR SAR 4.5.**

Your response to RAI No. 57 provided an updated SAR Chapter 13, Accident Analysis. However, for reactor facilities that are integral to publicly-occupied buildings (e.g., engineering classroom building), the maximum exposed member of the public could be in the public space within building rather than outside. The MHA analyses presented in the updated SAR does not appear to discuss the dose to on-site non-occupational occupants of the Merrill Engineering Building (MEB) such as students, faculty, visitors, etc. More specifically:

1. Please provide a dose assessment for the maximum exposed individual member of the public in the unrestricted area of the MEB. Please describe the assumptions used and any systems, plans, procedures or stay times for which credit is taken in the analysis.

Four new locations (target areas) in the MEB each with two scenarios (ventilation system OFF or ON) were added in analyzing the consequences of the MHA, **UUTR SAR 13.2.1.2**, Page 336 ~ 344.

2. Unless the duration of exposure to members of the public is limited by evaluation in accordance with the facility emergency plan, assume that members of the public are exposed until the event ends. If an evacuation is credited to limit the dose to member of the public, please provide a basis for the evacuation (Emergency Plan, Procedure, etc) and an assessment of the evacuation time credited (such as results of drills, etc).

More detailed explanations were added in the **UUTR SAR 13.2.1.2,** Page 343.

Your response to RAI No. 57 provided an updated SAR Chapter 13, Accident Analysis. During our review, of updated SAR Chapter 13.2.1, Maximum Hypothetical Accident (MHA), we noted that the dose consequences of various accident scenarios were provided. However, it was not clear what assumptions were used regarding the dispersion of radioactive effluents by the ventilation system. Some information was provided in the updated SAR, but additional details are needed. More specifically:

1. Please provide an explanation of the normal operation of the ventilation system and the system's response to a high radiation condition. Include the instruments used to monitor the radiation levels, the alarm settings, the components that change state (dampers closing) in the limited intake mode, and the resulting parameters used in any dose calculations (flow or leak rates, etc.).

Detail description of the ventilation system is provided in **UUTR SAR 9.1.4.** Page 190.
The set points for ARM and CAM systems are described in **UUTR SAR 1.3.5.2.** Page 11.

2. For each scenario provided in updated SAR Chapter 13, please clearly state the status of the ventilation system (normal operation, limited intake mode, or secured) and the associated dose. Include the dose consequences should the ventilation system fail to actuate to the limited intake mode or be manually secured.

All ventilation conditions were specified in the **UUTR SAR 13.2.1.2**, Page 331 ~ 343.

3. If the ventilation system is not required to the dose requirement of 10 CFR Part 20, please provide a definitive statement supporting this conclusion.

The UUTR SAR provides estimates of the dose to public and workers for several scenarios assuming that the ventilation system is in operational mode, or in its limited intake mode or in when it is completely shutdown. For all these cases, the workers in the reactor room or people present at the event of MHA in the other places such as nearby classroom, laboratory (MEHL), hallway area and office at the 2nd floor have sufficient evacuation time for the dose to rich the 10CFR20 limit. The maximum evacuation time from MEB was approximately 5 minutes based on the annual training. The radiation dose to be accumulated during these 5 minutes is significantly lower than the dose limit for public.
See the **UUTR SAR 13. 2.1, Table 13.2-9** and **Table 13.2-10.**

RAI 3.1

Your response to RAI No. 3.1 indicated that a burnup limit was not necessary but did not provide any supporting information. NUREG-1537, Appendix 14.1, Section 3.1 (6) indicates a limit of 50% from the original concentration of uranium-235 in the fuel for TRIGA reactors. Please discuss the applicability of a burnup limit and whether it is appropriate for inclusion in the UUTR TS.

UUTR TS 3.1.6 (5) was added to explain the fuel burnup limitation.

RAI 3.5

Your response to RAI No. 3.5 provided a corrected core excess reactivity value of $1.20 from the previous value of $2.80. However, our review of the updated SAR found examples where the $2.80 value was still the referenced value (e.g., updated SAR, Chapter 4.2.2). Please provide a consistent SAR reference for the value for the core excess reactivity.

Core excess reactivity was replaced with $1.20 through the **UUTR SAR**. See **UUTR SAR 4.2.2** (see Page 86).

New RAIs:

1. NUREG-1537, Section 14, Technical Specifications, Section 5.4, Primary Coolant cleanup System, recommends a pH range from 5.5 to 7.5. UUTR updated SAR, Chapter 14, TS 3.3, indicates a pH range from 5 to 8. Please evaluate and justify the use of a larger pH range.

The pH range was changed from 5~8 to 5.5 ~7.5
See the **UUTR TS 3.3** and **4.3.**

2. NUREG-1537, Section 14, technical Specifications, requests specific information on acceptable safety limits for TRIGA fuel.

2.1 NUREG-1537, Section 14, Technical Specifications, Section 2.1, Safety limits, provides an acceptable limit of 500 degree Celsius for Aluminum clad fuel. The UUTR TSs, Section 2.1, Safety Limits, indicates a safety limit of 530 degree C for aluminum clad fuel. Please provide a justification for the use of the higher temperature limit.

The safety limit for aluminum and stainless steel fuel elements were revised. See the **UUTR TS 2.1.**

2.2 The updated SAR, Chapter 4.2.1.1, Reactor Fuel Description indicates a safety limit of 1150 degree Celsius for stainless steel fuel. The updated SAR, Chapter 14, technical specifications, Section 2.1, Safety Limit-Fuel Element Temperature, has a value of 1000 degree Celsius. Please clarify the correct value.

The **UUTR SAR 4.2.1.1** mentioned two fuel temperatures, 1,150 °C and 1,000 °C. The temperature 1,150 °C is a theoretical limit for yield strength, but the UUTR SAR mentioned that the safety limit for a stainless steel element is 1,000 °C.

See **UUTR SAR 4.2.1.1** on Page 67 (8th line from the top of the 1st paragraph).

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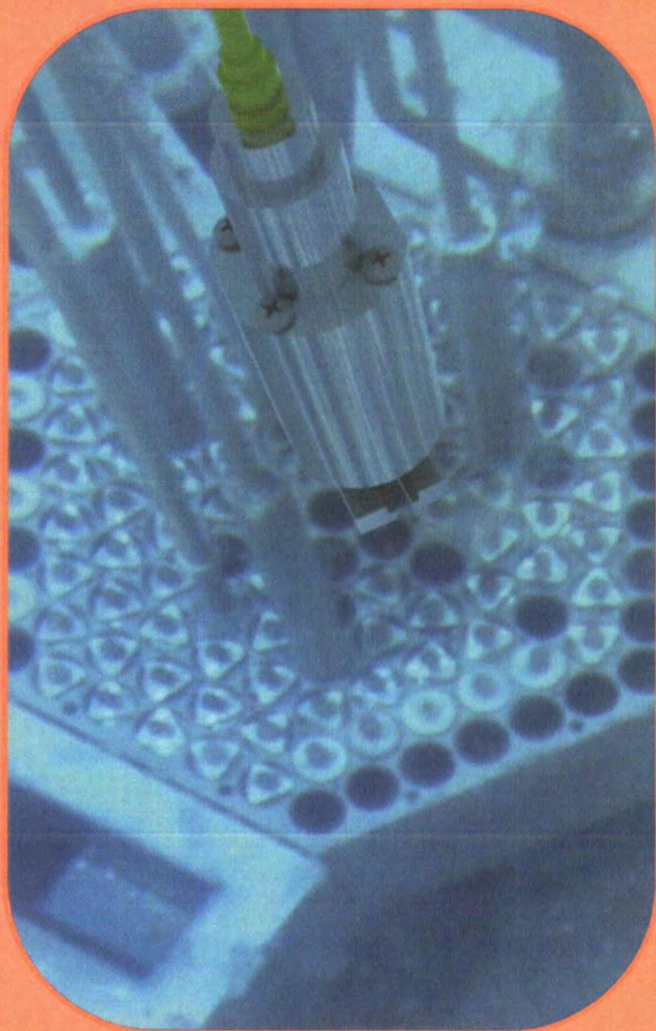
UUTR



Safety Analysis Report
2011

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1. THE UUTR FACILITY

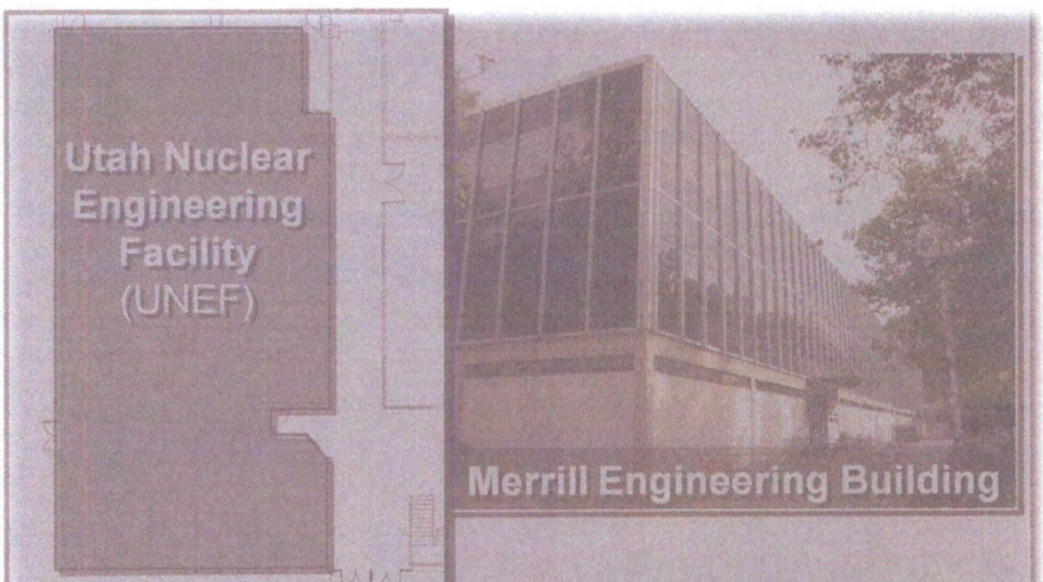


Figure 1.3-2 Location of the UUTR

1.1 Introduction

This Safety Analysis Report (SAR) document is prepared as part of an application to the Nuclear Regulatory Commission (NRC) from the University of Utah (U of U) for renewal of its TRIGA Modified Mark I Nuclear Reactor, License R-126, Docket 50-407.

The entire TRIGA reactor facility is contained within the Utah Nuclear Engineering Facilities (acronym - UNEF), and the TRIGA reactor is denoted herein as the University of Utah TRIGA Reactor (acronym - UUTR.) The reactor facility is located on the main campus of the University of Utah in Salt Lake City, Utah. The proposed license renewal will benefit all interested parties served by the facility within the regional area. The UUTR reactor core contains standard TRIGA fuel (enrichment <20%), is cooled by natural convection, and exhibits a large negative temperature coefficient of reactivity that provides inherent safety characteristic of all TRIGA reactors. The UUTR reactor is licensed, administered, and operated by the University of Utah for education, research, and reactor services since 1975. This SAR documents the University of Utah's intent to continue operation and maintenance of the UUTR reactor under NRC regulation. The UUTR is licensed to operate at 100kW steady-state power; however the UUTR never operated at the power above 90kW.

This document, supported by previous documents filed with the NRC including earlier SAR's, Technical Specifications, Safety and Security Plans, and Environmental reports, provides the description of the reactor system and its associated components' details, the general features, characteristics, and basic operation of the reactor. These documents reflect the as-built condition and the current administration and operation history of the facility. These documents include experience with the operation and performance of the reactor systems, radiation surveys, and personnel exposure histories related to operation of the UUTR at 100kW (i.e. 90kW). These analyses utilize conservative assumptions to provide ample operating safety margins. The descriptions and analyses contained in these reports is deemed to provide sufficient information to assure that the health and safety of the public is protected with continued operation of the reactor as proposed in this SAR.

1.2 Summary & Conclusions on Principal Safety Considerations

The physical operating safety of the UUTR reactor is achieved with TRIGA reactor fuel characteristics, the existing instrumentation, and the control and safety systems. The design basis parameters for the UUTR are: (1) power level, (2) fuel loading to achieve desired power level, and (3) control rods designs (number and position). The inherent safety of the UUTR reactor has been demonstrated by the extensive experience acquired from similar TRIGA systems throughout the world. This safety arises from the strongly negative prompt temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are terminated quickly and safely.

1.2.1 Safety Considerations

TRIGA fuel

The inherent safety of the UUTR lies primarily in the large, prompt negative temperature coefficient of reactivity characteristic of the TRIGA fuel (homogenized mixture of fuel and moderator provides that large prompt negative feedback, because there is no delay between fuel and moderator temperature variations); even if large sudden insertions of positive reactivity would occur with the reactor power rising in a short period of time, the prompt negative reactivity feedback produced by the increase in fuel temperature would cause the power excursion to be terminated before the fuel approaches its safety limit temperature and the fuel cladding is breeched. The UZrH fuel gives the reactor a prompt negative temperature coefficient of reactivity versus a delayed coefficient for research reactors utilizing aluminum clad plate-type fuel thus allowing TRIGA reactors to safely withstand events that would completely destroy plate-fueled reactor cores. The prompt shutdown and safety characteristics of TRIGA reactors fueled with TRIGA fuel have been demonstrated during transient tests conducted at General Atomics, Inc. in La Jolla, California as well as TRIGA pulsing facilities. This demonstrated safety has permitted the sitting of TRIGA fueled reactors in urban areas in buildings (without the pressure-type containment usually required for power reactors). The UUTR steady-state maximum power is therefore limited by the increase in fuel temperature.

TRIGA fuel cladding

Historically, analysis and testing of TRIGA fuel has demonstrated that fuel-cladding integrity is not challenged as long as stress on the cladding remains within yield strength for the cladding temperature. The UZrH is chemically stable. It can be safely quenched at 1,200 ^{o}C (1,473.15 ^{o}K) in water. The UZrH fuel material has superior retention of radioactive fission products; at about 650°C (923.15 ^{o}K) the UZrH retains more than 99% of fission products, even if the cladding is removed. The elevated TRIGA fuel temperatures release hydrogen from the zirconium matrix, with concomitant pressure buildup in the cladding and thus the strength of the cladding as a function of temperature establishes the upper limit on the reactor power. Power less than limiting values will ensure cladding integrity and, therefore, contain the radioactive materials that are produced by fission in the fuel elements of the UUTR core.

Heat removal

As a natural-convection cooled system, heat removal capacity is well defined as long as the primary coolant is sub-cooled, restricting potential for film boiling. Limiting the potential for film boiling assures that fuel and clad temperatures are not capable of challenging cladding-integrity. The maximum heat generated within a fuel element and the bulk water temperature determines the tendency for film boiling.

Experiments

All the experiments in the reference core of the UUTR are limited to reactivity worth less than $1.20. If the reactivity worth for an experiment is estimated to be more than $1, the experimental samples must be placed in an irradiator before reactor start-up and should not be removed until the shutdown of the reactor, thus assuring that the experiment failure cannot result in severe transients.

1.2.2 Consequences of Normal Operations

Operating experience, which documented radiation exposures to personnel working in the UUTR from both direct and airborne radiation during normal operation have been reviewed and assessed. These analyses and measurements show that the exposure rates for the power of 100kW are well within NRC accepted exposure limits. Under normal operating conditions, personnel are subjected to a maximum radiation field of less than 1 mR/hr. In actual practice, radiation exposures are lower since typical operation times are much less than the conservative assumptions indicate. All personnel entering the facility are closely monitored, exposures kept to as low as reasonable achievable (ALARA), and in no case will exposures be allowed to exceed 10CFR Part20 regulations or guidelines. **UUTR SAR 11**[1] shows that radiation dose rates directly above the reactor pool during expected operations at levels up to 100kW are within required levels for a radiation area as defined in 10CFR Part20. Installed monitoring systems provide information necessary to identify appropriate access controls.

As indicated in **UUTR SAR 11** radiation sources are discharged from the UUTR facility in the following forms:

- *Airborne radiation sources* consist mainly of Argon-41 and Nitrogen-16. Limits on Argon-41 are tabulated in Appendix B of 10CFR Part20. A general limit on off-site doses from gaseous effluents is listed in 10CFR20.1101. The nuclide Argon-41 is produced by thermal neutron absorption by natural Argon-40 in the atmosphere and in the air dissolved in the reactor cooling water. The activation product appears in the reactor pool and is subsequently released to the atmosphere through the ventilation system. Argon-41 has a 1.8-hour half-life. Nitrogen is the major contributor to radiation fields directly over the reactor pool during operation. Nitrogen-16 is produced by a fast neutron interaction with oxygen (as a natural component of water in the core). Nitrogen-16 has a 7.1-second half-life, and consequently does not remain at concentrations capable of contributing significantly to off-site dose. Operating experience, which documented radiation exposures to personnel working in the UUTR from both direct and airborne radiation during normal operation have been reviewed and assessed. Computational analyses and measurements for 100kW operating power of the UUTR show that the exposure rates are well within NRC accepted exposure limits. Under normal operating conditions, personnel will be subjected to a maximum radiation field of less than 1 mR/hr. In actual practice, radiation exposures will be lower since typical operation times are much less than the conservative assumptions indicated in

[1] This is the adapted notation of the UUTR SAR Chapters when referenced in this SAR.

UUTR SAR 11. All personnel entering the facility is and will be closely monitored, exposures kept to as low as reasonable achievable (ALARA), and in no case will exposures be allowed to exceed 10CFR Part20 regulations or guidelines.

- *No liquid radioactive material* is routinely produced by the normal operation of the UUTR. Dissolved minerals and metals are removed in the resin beds, characterized and transferred to Radiological Health Department (RHD) for disposal. Spent liquid samples are also characterized and transferred to RHD for disposal.
- *Solid radioactive sources* associated with the UUTR operation include low enriched fuel, start source, calibration sources used for instrumentation and samples. Solid radioactive sources are secured and inventoried.

The UUTR reactor water shielding and supporting containment provide biological shielding adequate to maintain personnel exposures ALARA and protect the reactor from natural and human disasters. The reactor room air handling system maintains the reactor room at a negative pressure with respect to surrounding areas to control and prevent the spread of airborne radioactive materials. The air from the reactor room passes through HEPA filters prior to being discharged to the atmosphere. In the event of a release of radioactive material within the reactor room, the reactor room air handling system automatically closes the inlet dampers effectively isolating the room by increasing the negative pressure and preventing the release of activity to surrounding offices. Sufficient air is discharged through the HEPA filters and Continuous Air Monitor (CAM) system to maintain negative air pressure in the reactor area.

1.2.3 Consequences of Potential Accidents

As described in **UUTR SAR 13** the following accidents are identified:

1. The maximum hypothetical accident (MHA);
2. Insertion of excess reactivity;
3. Loss of coolant accident (LOCA);
4. Loss of coolant flow;
5. Loss of pool water;
6. Mishandling or malfunction of fuel;
7. Experiment malfunction;
8. Loss of normal electrical power;
9. External events; and
10. Mishandling or malfunction of equipment.

Analyses demonstrate that the consequences of these reactor accidents are acceptable for 100kW, and that the doses to the public are well below limits established by 10CFR Part20.

1.3 General Description of the Facility

1.3.1 Geographical Location

The 1,167 acre campus is situated east of the city center on the foothills of the Wasatch Mountains. **Figure 1.3-1** shows the campus in relation to Salt Lake City and other towns and communities situated in the Great Salt Lake Valley.

Figure 1.3-1 Location of University of Utah Campus

1.3.2 Principal Characteristics of the Site

The reactor is located in the Utah Nuclear Engineering Facility (UNEF) on the ground floor of the Merrill Engineering Building (MEB) on the north end of the University of Utah campus, **Fig.1.3-2**. **Figure 1.3-3** shows the UNEF map, i.e. the layout of the rooms in the southwest corner of the MEB, which includes the UUTR laboratories and major equipment. The MEB is situated on high ground and rises about 100 ft above the nearest neighboring structures.

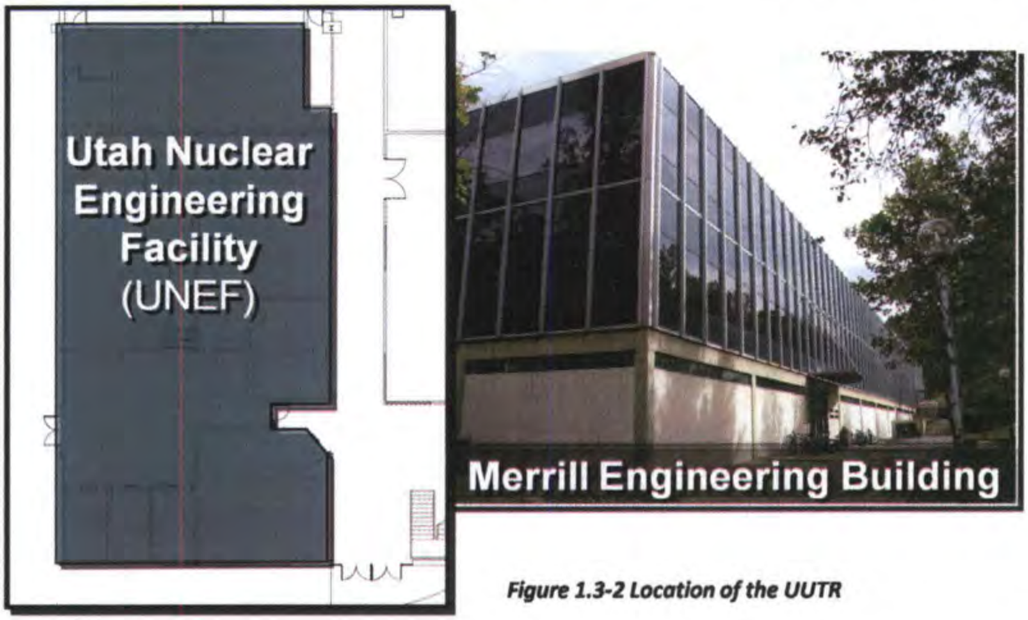


Figure 1.3-2 Location of the UUTR

The building is constructed with structural steel frames and reinforced concrete floors acting as diaphragms in distributing loads to vertically resisting elements. The MEB conforms to seismic zone 3 requirements of the Uniform Building Code. MEB was designed by Dean L. Gustavson Associates, Architects, and constructed by Alder Child Construction Company. The building has approximately 254,778 square feet of floor space assigned as follows: Classrooms - 24,859; Offices - 25,547; Teaching laboratories - 59,399; Research laboratories - 97,847; workshops, storerooms, corridors, etc. - 56,126. The areas immediately above the reactor room, on the second floor, are faculty and departmental offices. Radiation surveillance of the areas immediately above the UUTR has been performed by the Radiation Safety Officer at the University of Utah since the initial licensing of the UUTR. These areas will continue to be monitored. Residence by any person in this area can be controlled if necessary. The floor (or ceiling) of the area directly above the reactor core has four inches of concrete and 3/l6 inch of steel (floor support.) The third floor above the reactor area is comprised of office and laboratory space.

The reactor area comprises eight rooms including the reactor control room (1205D), prep-room for labs (1205), radiation measurement laboratory (1205J), radiochemistry laboratory (1205K), microscope room (1205H), fuel inspection room (1205F & G), radioactive source room (1205B), and reactor room (1205E) as indicated in **Fig. 1.3-3**. Entry to the reactor room from inside the building is restricted to one door in the control room. The reactor room has direct access to the outside loading area through a 12 ft wide overhead door. This overhead door and the doors between reactor and control room form the confinement enclosure. The reactor room and the laboratories will be treated as a single unit for ventilation and safety confinement purposes. The walls in the area are constructed of one-hour fire-resistant, standard, plaster and metal stud construction with the exception of the west wall, which is an exterior reinforced concrete wall. Large windows provide visibility to the reactor room from the administrative offices, control room and computational lab. Nonporous enamel paint finishes were used on all walls and ceilings of the reactor area for easy clean up.

[redacted]

Figure 1.3-3 UNEF Map

1.3.3 Principal Design Criteria, Operating Characteristics, & Safety Systems

The UUTR reactor is an open pool water moderated reactor built in 1975. The UUTR reactor tank is a double wall construction to resist earthquake; the reactor core rests below ground level in a containment 24 ft tank filled with water. The inner tank is made of aluminum that has a diameter of 8 ft. The outer tank is made of steel with diameter of 12 ft. This water is used as a moderator and as a biological shielding. The reactor core is of hexagonal prism shape; its height is 2 ft and the maximum diagonal distance is 2 ft.

The UUTR was licensed to operate at a steady-state thermal power of 100kW from 1975. The reference UUTR core consists of 78 fuel elements; each fuel element contains approximately [redacted] grams of U-235. The fuel element for UUTR has aluminum or stainless steel cladding fuel elements; each fuel element has 8.5 weight % of U-235 with less than 20% of uranium enrichment. The UUTR core has a [redacted] Pu-Be neutron start-up source attached to the top of the reactor pool with a stainless steel wire. The UUTR uses heavy water and graphite as a reflector.

[redacted]

Figure 1.3-4 Top view of the UUTR core

The UUTR fuel is a zirconium hydride matrix that exhibits a large negative temperature coefficient so that in the event of a prompt excursion the reactor will inherently and passively shut down the reactor without fuel damage. The UUTR core consists of seven rings, A through G. The A ring has no fuel and is used as a central irradiator. The UUTR has three control rods, a safety, shim, and regulation control rod. These three control rods control criticality; they are aluminum cladding boron-carbide (B_4C) rods. Shutdown margin and the excess reactivity are assured using these three control rods. The UUTR safety systems include multiple scram capabilities for elevated temperature or power, elevated radiation levels, loss of electrical power, high voltage interruption and low water level. Control rod interlocks limit the reactivity insertion rates. In addition to the scram systems, the ventilation system maintains a negative pressure in the facility during the UUTR operation. This ventilation system also operates in an emergency mode to mitigate a radionuclide release. **Figure 1.3-4** shows the core cross section indicating each fuel element, reflector, and irradiation positions.

1.3.4 Engineered Safety Features

The design of the UUTR TRIGA, licensed in 1975, imposed no requirements for engineered safety features. As discussed in **UUTR SAR 13,** and from previous analysis, neither forced-cooling flow nor shutdown emergency core cooling is required for operation at steady-state thermal power of 100kW.

The engineered safety features are used in case of severe accidents for reactors with relatively high power (research, test, and power reactors). However, low power reactors do not generally have engineered safety features (similar to the UUTR are the TRIGA at Reed College or the Oregon State University). The engineered safety features are usually divided into three categories, the containment systems (primary containment, containment heat removal, secondary containment). Additionally, they include the Emergency Core Cooling System (ECCS): high pressure core flooder (HPCF), low pressure flooder (LPFL), automatic depressurization system (ADS), [Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, U.S. Nuclear Regulatory Commission, 1987].

The UUTR does not have any of these systems.

1.3.5 Instrumentation and Control (I&C) and Electrical Systems

The UUTR reactor console consists of digital electronics for signal processing and display. The console is located in the reactor control room and supports all reactor operations including control rod movement operations, temperature and area radiation monitoring, safety settings, interlocks, and scram functions. The reactor instrumentation and control systems are manufactured by General Atomics and widely used in most of the NRC licensed facilities. As previously noted, there are no engineered safety features at the UUTR and, therefore, no associated instrumentation.

1.3.5.1 Reactor Control System

The reactor control system includes control rod drives, control rod positioning monitors, and SCRAM function. The control rod can be lowered or raised using each control rod button. No more than one control rod can be raised. In case of an emergency, an operator can use manual SCRAM bar to shut down the reactor. All three control rods are moved by manual mode only to regulate reactor power according to the linear and percent channel. The nuclear instruments of the reactor protection system are integrated into the reactor control system through the linear and percent power monitoring systems, fuel temperature monitoring system, pool water height monitor, high voltage interruption and magnet key interruption. If the reactor power exceeds maximum power or the fuel element temperature becomes higher than 200°C, the reactor will scram automatically. Also, the reactor will scram automatically when high voltage pass through the control system, pool water level is lower than 15.5 inches from the top of the reactor tank. It will scram the reactor also if the reactor key is removed during the reactor operation. The console display shows the information about the control rod position, power level, fuel temperature, reactor period, and other system parameters. The console also displays many other functions, such as monitoring reactor usage and water quality display and sample "in" and "out" buttons for pneumatic irradiator. Most of these system parameters can be monitored from the separate computer screen using Lab View™. In addition to the above instrumentation, two video cameras are located above the tank to provide full surveillance of the core and components. The monitors for this camera are located on the control console and North-South corner of the control room.

1.3.5.2 Radiation Safety Monitoring Systems

Radiation area monitoring systems are installed inside of the reactor room: one is near the top of the reactor tank, the second is located at the reactor room ceiling and the third is installed in the reactor radiochemistry lab (**Fig.1.3.-3**). All these three area radiation monitoring (ARM) systems have "low alarm set" at 1 mrem/h and "high alarm set" at 10 mrem/h. When high alarm set point is reached the ARM system will scram the reactor.

Continuous Air Monitor (CAM) system has alarm set point of 1,000 CPM for radioactive particulates such as iodine and noble gas and will provide the alarm sound to the control room. When CAM activates the alarm, meaning the reading is higher than the set point, reactor operator is required to investigate the reasons for the alarm.

ARM and CAM radiation levels can be read from the reactor console.

1.3.5.3 Electrical Power

Primary power for the UUTR including MEB is provided through the Rocky Mountain Power Co. Loss of the electrical power will cause the control rod to drop by gravity into the core in a less than 1 second placing the reactor in a subcritical condition. Because the UUTR reactor core is cooled by natural convection, no emergency power is required for reactor cooling system. In addition, electrical power to the console, emergency light, security system and reactor safety and radiation monitoring equipment has a battery back system (UPS) to insure safe operation in the event of loss of building electrical power.

1.3.5.4 Reactor Protection System

The UUTR has several reactor protection systems. The UUTR safety systems include multiple scram capabilities for elevated temperature or power, elevated radiation levels, loss of electrical power, high voltage interruption and low water level. There are two reactor power scram systems, linear power and percent power. Linear power channel will scram the reactor when the reactor power exceeds maximum designed level. The percent power channel will scram at 100 % of the licensed power of 100kW. High voltage pass through the control system or lower water level will scram the reactor also. If an operator removes the reactor key, the reactor will scram itself automatically. Control rod interlocks exist to limit the reactivity insertion rates. In addition to the scram systems, the ventilation system maintains a negative pressure in the facility during operation. This ventilation system also operates in an emergency mode to mitigate a radionuclide release.

1.3.6 Reactor Coolant and Other Auxiliary Systems

The reactor tank has a capacity of approximately 8,000 gallons of water with a depth measured from the surface to the top of the core of approximately 6.5 m (22 ft). The reactor core is cooled by the natural convection flow of light water within the reactor pool. Water purity is maintained by a dual bed demineralizing system including particulate filters and resin beds. Makeup water is added to the reactor tank from the main building water supply as necessary by passing the makeup water through the demineralizing system.

A shell and tube heat exchanger rated at seven-and-a-half ton capacity can be used to cool the pool water. The heat exchanger receives warm water from the pool and returns cooled water. Both the inlet and return lines to the pool have a small 1/4 inch hole approximately one foot below the normal pool water level. These holes prevent siphoning of the pool water below that level siphon inlet. The water pumped to the heat exchanger is circulated at the rate of about 20 gallons per minute by a centrifugal pump. The reactor pool temperature can be maintained at about 16 °C (289.15 °K), which is approximately the ambient ground temperature. For any operation at power above 25 kW, the heat exchanger-cooler has insufficient capacity to maintain the 16 °C (289.15 °K). When operating at 90kW, the pool temperature rises approximately 3 °K per hour based on the measurements. Administrative control will limit the maximum water temperature to 35 °C (308.15 °K). After the reactor is shutdown, the cooling system can lower the pool water temperature approximately 0.5 °K per hour based on measurements.

1.3.7 Radioactive Waste Management and Radiation Protection

The University of Utah has a radiological safety program on its campus. The University's Radiation Safety Officer provides radiological services and monitoring for the entire University including the University hospitals and School of Medicine, researchers employing radioactive materials in their work, and the UUTR. The University's Broad Scope Radiation License issued by the State of Utah (an NRC Agreement State) is administered by Radiological Health Department (RHD). All radioactive wastes generated within the UUTR facility are disposed of through RHD as specified in 10CFR61, and 71, and following the provisions as outlined under NRC agreement state status. In addition, RHD maintains oversight on all personnel dosimeters, area dosimeters including environmental dosimeters, and radionuclide transport as required in 10CFR20, and 49CFR171 to 178.

1.3.7.1 Gaseous Waste

The UUTR generates very small amount of Argon-41. For example ~11 mCi/year was generated in 2009 when the UUTR was operated at 90kW for 28.4 hours. The total of thermal energy generated in 2009 was 927.072kW-hours. The amount of Nitrogen-16 can be ignored because its half-life is very short and most of decay out in the reactor pool. To minimize release of the gaseous waste, the reactor room is kept at negative pressure all the time.

1.3.7.2 Liquid Waste

The UUTR facility does not regularly create or release the liquid waste.

1.3.7.3 Solid Waste

The UUTR generates a small amount of solid waste such as gloves, samples from experiments, resin bed parts, paper towels from fuel inspection, tools, and some lab-ware. Most of these solid wastes are stored in the storage room that is located inside of UUTR facility until they decay out. Solid waste shipments are coordinated with the University of Utah's RHD and Radiation Safety Officer (RSO).

1.3.8 Experimental Facilities and Capabilities

The UUTR has five different experimental or irradiation facilities: Fast Neutron Irradiation facility (FNIF), thermal irradiator (TI), pneumatic irradiator (PI), central irradiator (CI), and three vertical beam ports.

1.3.8.1 Central Irradiator

The UUTR central irradiator is located in the central fuel pin position of the UUTR core. A special tube has been constructed to accommodate insertion of the samples and can be placed in the central fuel pin position using a cable.

1.3.8.2 Pneumatic Irradiator

A pneumatic irradiator (PI) is available for rapid irradiations at the UUTR facility. The rabbit is installed within a 1.5 inch O.D. aluminum tube and is driven by dry, compressed helium gas. The pneumatic rabbit tube has a curved trajectory to prevent direct streaming of radiation from the core to the surface of the pool. PI is located at D4 position (**Fig. 1.3-5**).

1.3.8.3 Dry Tube

A dry tube thermal irradiator is available for use in a trapezoidal shaped D_2O tank attached to the side of the reactor core. The D_2O tank is made of aluminum. The samples are placed into polyethylene vials attached to fishing line and dropped into the irradiator through a curved PVC tube that extends to the top of the reactor pool.

1.3.8.4 Fast Neutron Irradiation Facility (FNIF)

The FNIF was specifically designed to provide a neutron irradiation environment with a quasi-fission energy spectrum and low photon component for neutron hardness assurance testing of electronic components. FNIF consists of aluminum can and lead inside to provide pure neutron irradiation environment. FNIF is attached to the west side of the reactor core.

1.3.8.5 Beam Ports

The reactor system permits the insertion of up to three diagonally directed beam tubes that extend from the reactor core to the reactor floor. There are no plans to open the diagonal beam ports. The upper terminus of each beam tube is adequately shielded with sand and secured with a 1/8 inch thick steel locked cap at the reactor floor level. The beam ports are presently closed and filled with sand and have not been installed or used to date. **Figure 1.3-5** shows CI, PI, FNIF, TI, beam ports and the structure of the UUTR pool.

[redacted]

Figure 1.3-5 UUTR pool and irradiation facilities

1.4 Shared Facilities and Equipment

The UUTR is housed by the Utah Nuclear Engineering Facility (UNEF) lead by the Utah Nuclear Engineering program (UNEP). UNEF and therefore the UUTR are integral part of the Merrill Engineering Building (MEB), as shown in **Fig. 1.3-2**, and thus share walls, water supplies, sewage, heating, cooling, and the main electrical supply. The ventilation system, electrical distribution, water distribution, and heating are all separate.

The water purification for the UUTR, is operated and maintained by the UUTR staff. University of Utah maintenance staff maintains the dedicated ventilation system for the facility. In the event of a loss of power, a battery backup system can provide emergency power to the CAM and ARM systems for at least 24 hours. Further details are provided in **UUTR SAR 5**.

1.5 Comparison with Similar Facilities

As of 2005, there were 19 operating TRIGA reactors in the US (**Fig. 1.5-1**), and 25 in the world excluding the US (**Fig. 1.5-2**).

Figure 1.5-1 TRIGA Facilities in the US
[http://www.ga-esi.com/triga/about/install_usa.pdf
http://en.wikipedia.org/wiki/list_of_nuclear_reactors
http://hepd5s.phys.utk.edu/nnbar/appendix/TRIGA_Attachment_1.pdf
http://standeyo.com/News_Files/NBC/NRC.Non-power.reactors.html]

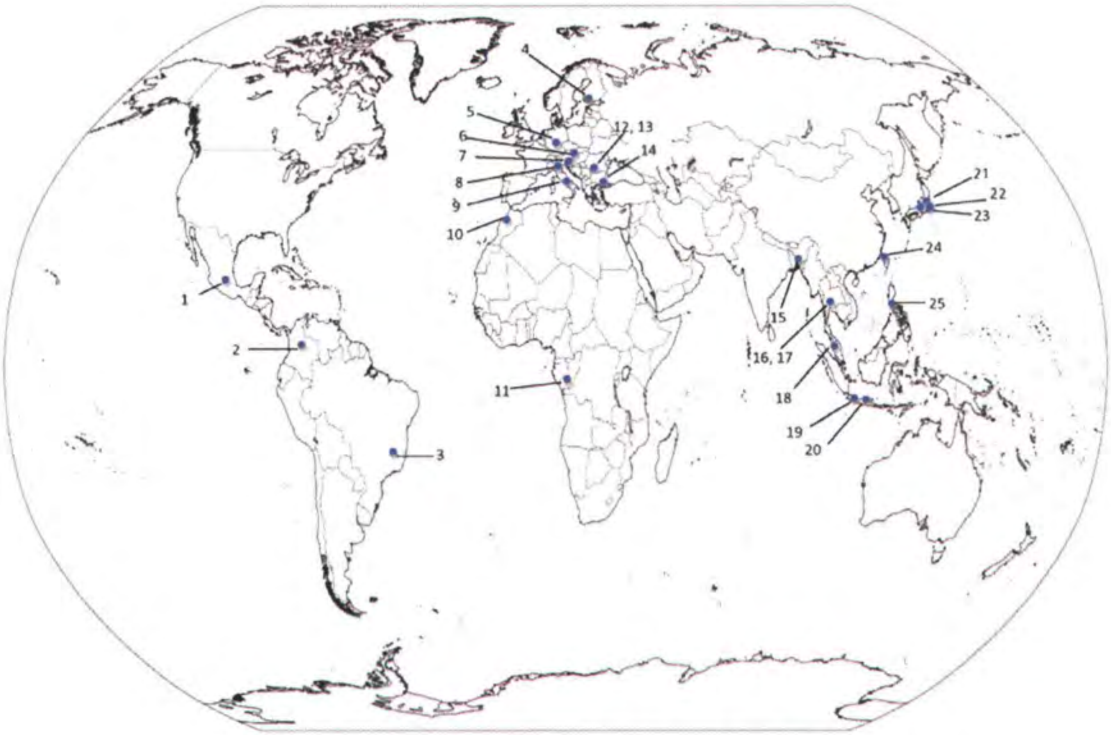


Figure 1.5-2 TRIGA Facilities in the world excluding the US
http://www.ga-esi.com/triga/about/install_inter.pdf
http://ansn.bapeten.go.id/download.php?fd=240&filename=Chapter1_Appendix1.pdf&down=1

Legend:

1. National Institute for Nuclear Research
   Mexico City, Mexico
   Mark II 1000 kW 1982
2. Institute of Nuclear Science and Alternative Energy
   Bogota, Columbia
   Mark I 100kW 1997
3. University of Minas Gerais
   Belo Horizonte, Brazil
   Mark I 100 kW 1960
4. The State Institute for Technical Research
   Helsinki, Finland
   Mark II 250 kW 1962
5. Jonhannes Gutenberg University
   Mainz, Germany
   Mark II 100 kW 1965
6. Federal Ministry of Education
   Vienna, Aust.
   Mark II 250 kW 1962
7. Jozef Stefan Nuclear Institute
   Ljubljana, Slovenia
   Mark II 259 kW 1966
8. University of Pavia
   Pavia, Italy
   Mark II 250 kW 1965
9. ENEA Cassaccia Research Center
   Rome, Italy
   Mark II 1000 kW 1960
10. National Center for Nuclear Sciences and Energy
    Rabat, Morocco
    Mark II 2000 kW 2008
11. Nuclear Science Commission
    Kinshasa, Zaire
    Mark II 1000 kW 1972
12. Institute for Nuclear Research
    Pitesti, Romania
    ACPR 500 kW 1979
13. Institute for Nuclear Research
    Pitesti, Romania
    MPR 16 14,000 kW 1979
14. Technical Institute of Istanbul
    Istanbul, Turkey
    Mark II 250 kW 1979
15. Atomic Energy Research Establishment
    Dhaka, Bangladesh
    Mark II 3000 kW 1986
16. Office of Atoms for Peace
    Bangkok, Thailand
    Conversion 1 MW 1977
17. Ongkharak Nuclear Research Centre
    Bangkok, Thailand
    MPR 10 10 MW
18. Malaysian Institute for Nuclear Technology
    Kuala Lumpur, Malaysia
    Mark II 1000 kW
19. National Atomic Energy Agency
    Bandung, Indonesia
    Mark II 2000 kW 1997
20. National Atomic Energy Agency
    Bandung, Indonesia
    Mark II 250 kW 1979
21. Musashi Institute of Technology
    Tokyo, Japan
    Mark II 100 kW 1963
22. Rikkyo Univeristy
    Yokosuka, Japan
    Mark II 100 KW 1961
23. Japan Atomic Energy Research Institute
    Tokai, Japan
    ACPR 300 kW 1975
24. National Tsing Hua University
    Taipie, Taiwan
    Conversion 1MW 1977
25. Philippine Atomic Energy Commission
    Quezon City, Philippines
    Conversion 3MW 1988

UUTR fuel is standard TRIGA fuel with 8.5% of uranium, by weight, enriched less than 20% in the uranium-235 isotope. TRIGA fuel is characterized by inherent safety and as described in **UUTR SAR 1.2.1,** the homogenized mixture of fuel and moderator provides that large prompt negative feedback because there is no delay between fuel and moderator temperature variations); even if large sudden insertions of positive reactivity would occur with the reactor power rising in a short period of time, the prompt negative reactivity feedback produced by the increase in fuel temperature would cause the power excursion to be terminated before the fuel approaches its safety limit temperature and the fuel cladding is breeched. The inherent safety of TRIGA reactors has been demonstrated by extensive experience acquired from similar TRIGA systems throughout the world and in the US.

Table 1.5-1 UUTR Design Parameters

PARAMETER	VALUE
Maximum steady-state thermal power	**100 kW**
Maximum excess reactivity	\$1.20
Number of control rods	3
Number of regulating rods	1
Number of shim rods	1
Minimum shutdown margin	\$0.50
Integral fuel-moderator material	D_2O, Graphite
Reactor cooling	Natural convection (water)
Number of fuel elements	78
Uranium enrichment	<20%
Uranium content	8.5 %
Shape	Hexagonal
Core length/width	2ft high, 2ft diagonal
Fuel assembly number and types	23-Al (8.91% burn-up) 36-SS (0.61% burn-up) 17-SS (8.77% burn-up) 2-SS Instrumental (8.77% burn-up)
Fuel assembly height/pitch	[redacted]
Fuel cladding outer/inner diameter	Al cladding-1.47"/1.41" SS cladding-1.45"/1,41"
External moderator	H_2O
Pool depth/diameter	24 ft/8 ft

The reference UUTR core consists of 78 fuel elements (23 with the aluminum cladding, 53 with the SS-304 cladding and two with the SS-cladding instrumental fuel elements). Each fuel element has almost the same geometry and the size, except for the upper and lower end fixtures. The fuel element is a solid, homogeneous mixture of uranium-zirconium hydride alloy. The hydrogen and zirconium ratio is 1.0 for aluminum cladding fuel elements and 1.60 for stainless steel cladding fuel elements. The UUTR 100 kW core consists of 78 elements with 12 heavy water and 12 graphite reflectors. The UUTR has hexagonal pitch (1.71 inch pitch) and contains approximately twice as many triflute coolant channels as there are elements. The design parameters for the UUTR TRIGA are listed in **Table 1.5-1**.

1.6 Summary of Operations

The UUTR provides a wide range of training, irradiation, and research services to education, research, and industry. The reactor has experimental facilities for the irradiation of materials. These facilities support the irradiation services that include sample irradiation for medical and industrial clients, neutron activation analysis, neutron damage testing of electronic components, ultra-sensitive detection of actinides, and educational and training support. Currently, the beam ports are sealed and not in use. To support the regional services, the UUTR typically operates about five to eight hours per week at 90 kW. The operational workload is not expected to increase significantly from this level.

1.7 Nuclear Waste Policy Act of 1982

In accordance with the Nuclear Waste Policy Act of 1982, the UUTR Licensee has signed a contract with the DOE for the return of the reactor fuel that is owned by the DOE to the DOE at cessation of the UUTR License.

1.8 Facility Modifications and History

Criticality of the UUTR was first achieved in 1975. The UUTR has original in-core detection systems such as one compensated ionization chamber, two uncompensated ionization chambers, and one fission chamber. Various upgrades have been made since 1975 with the major ones summarized in **Table 1.8-1**.

Table 1.8-1 Major UUTR facility modifications

Year	Activity (For all activities, 10CFR50.59 were reviewed)
October 1975	Construction completed, fuel loaded, initial criticality
1987	Decommissioned 5W AGN 201 reactor, located close to UUTR
1989	Added Fast Neutron Irradiation Facility (FNIF) (FNIF was installed on the west side of the reactor core; it provides 1 MeV equivalent fast neutron flux)
1991	Upgraded reactor control console (A new control console was installed with digital equipment)
1997	Replaced Continuous Air Monitoring recoding system (New recording system was installed)
August 1997	Add radiochemistry lab and class 100 clean room (Two fume hoods in the radiochemistry lab and class 100 clean room were added)
October 2001	Remodeled facility office area (Three offices for faculty and one student office area were added)
August 2005	Replaced hoist system in the reactor room (New hoist system which has 2 metric ton capacity was installed)
May 2007	Pneumatic irradiator on D-4 position was added (With compressed He gas)
December 2008	Replaced fume hood in the radio-chemistry laboratory and ventilation duct and pipe (Two old fume hoods in the radiochemistry lab were replaced with new fume hoods)
March 2010	Remodeled reactor room, control room area, and associated labs.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

The University of Utah's TRIGA reactor (UUTR) is located in Salt Lake City, Utah. Salt Lake City is located approximately 115 miles (185 km) east of Nevada and 84 miles (135.2 km) south of Idaho. **Figure 2.1-1** shows the location of UUTR in respect to the State of Utah, and **Fig. 2.1-2** shows the location of UUTR in respect to Salt Lake City.

2.1.1.1 Specification and Location

The UUTR is located in the Merrill Engineering Building (MEB) on the north end of the University of Utah Campus in Salt Lake City, Utah [redacted] The 1,167 acre campus is situated 2.4 miles (3.9 km) east of the Salt Lake City center on the foothills (1450 ft. elevation) of the Wasatch Mountains. The University of Utah campus is shown in **Figure 2.1-3** indicating a location of Merrill Engineering Building housing the UUTR.

2.1.1.2 Boundary and Zone Area Maps

Figures 2.1-1 and **2.1-2** illustrate the location of the UUTR with respect to the State of Utah and Salt Lake City indicating the zone areas maps. **Figures 2.1-3** and **2.1-4** show the location of the UUTR within the University of Utah campus.

2.1.1.3 Population Distribution

Salt Lake City is the largest city in the State of Utah with population of 181,743 (2000 census). Salt Lake County, which has an area of about 737.4 square miles (1909 km^2), has a population of about 898,387. The area surrounding the MEB that houses the UUTR is lightly populated with some residences located at about 150 yards (137 meters) west and 360 yards (330 m) north of the building. To the east there are the Primary Children's Medical Center, University Hospital, and Huntsman Cancer Hospital at the distances of around 740, 1000, and 1200 yards (677, 914, and 1097 m) respectively. Student residence halls are located approximately 1000 yards (914 m) to the south-east. South of the MEB, the campus extends for about one mile with former federal government land now owned by the University extending another half mile south. The nearest residences to the south are approximately 1.5 miles (2.4 km) from the building. According to the University of Utah the Radiological Health Department, the daytime campus population of faculty, students, and staff is estimated to be 25,000 to 30,000 people. The nighttime population is estimated to be 5,000 people. According to the

University of Utah Office of Residential Living, the number of students living on campus in residence halls is 1,250. The MEB has a total daytime occupancy of approximately 500, while the nighttime occupancy is approximately 30 people. **Table 2.1-1** summarizes the population data and **Fig. 2.1-5**.

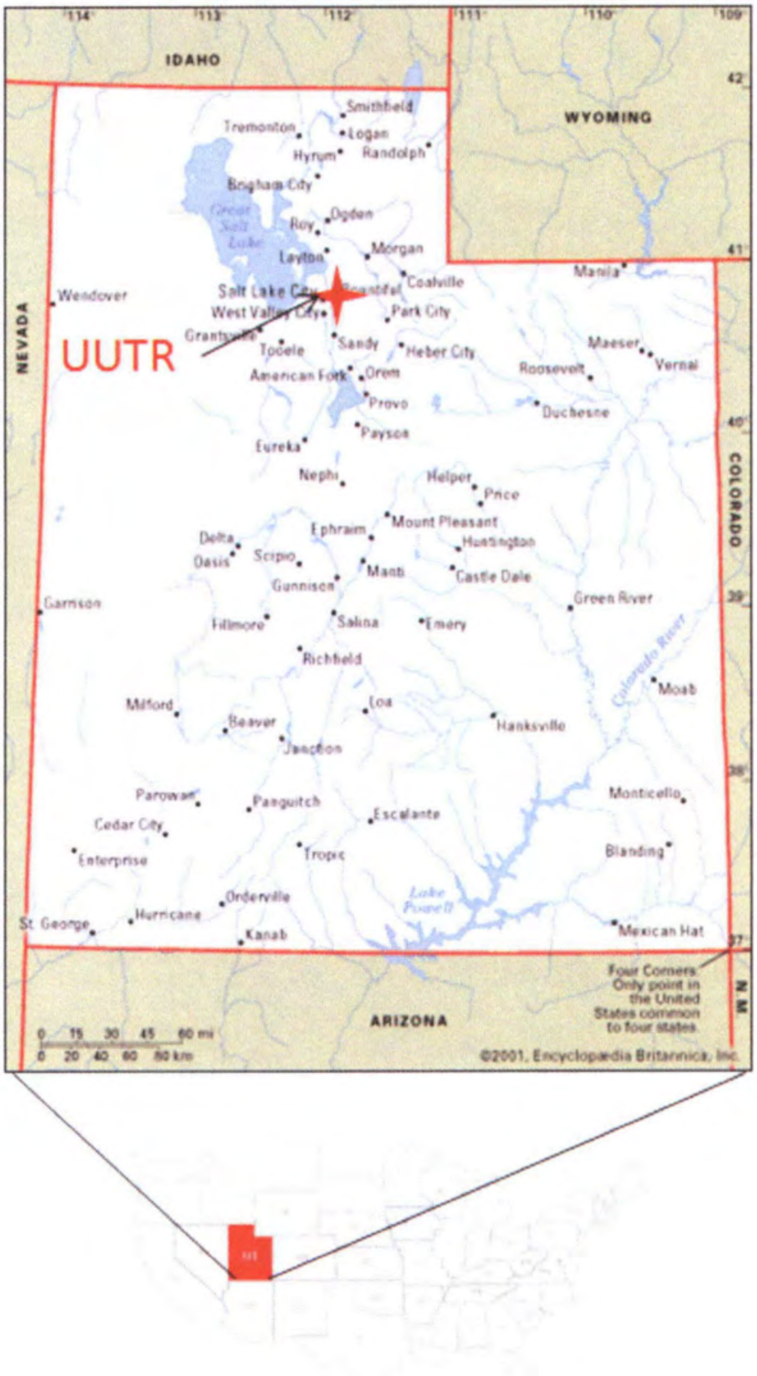


Figure 2.1-1 UUTR Location within Utah

Figure 2.1-2 UUTR Location Relative to Salt Lake City, Utah
[1mile, 2 miles, 4miles, 6miles and 8 miles distances are indicated]

Figure 2.1-3 University of Utah Campus

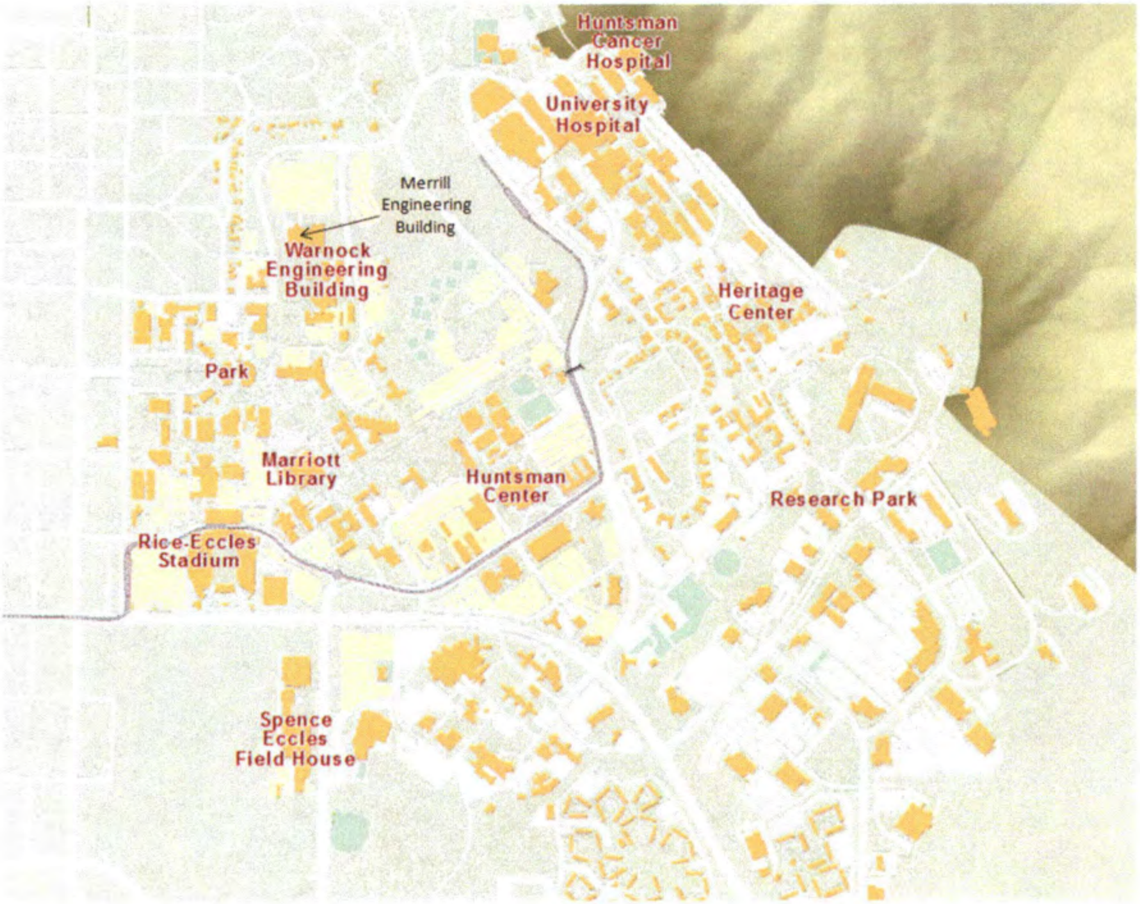
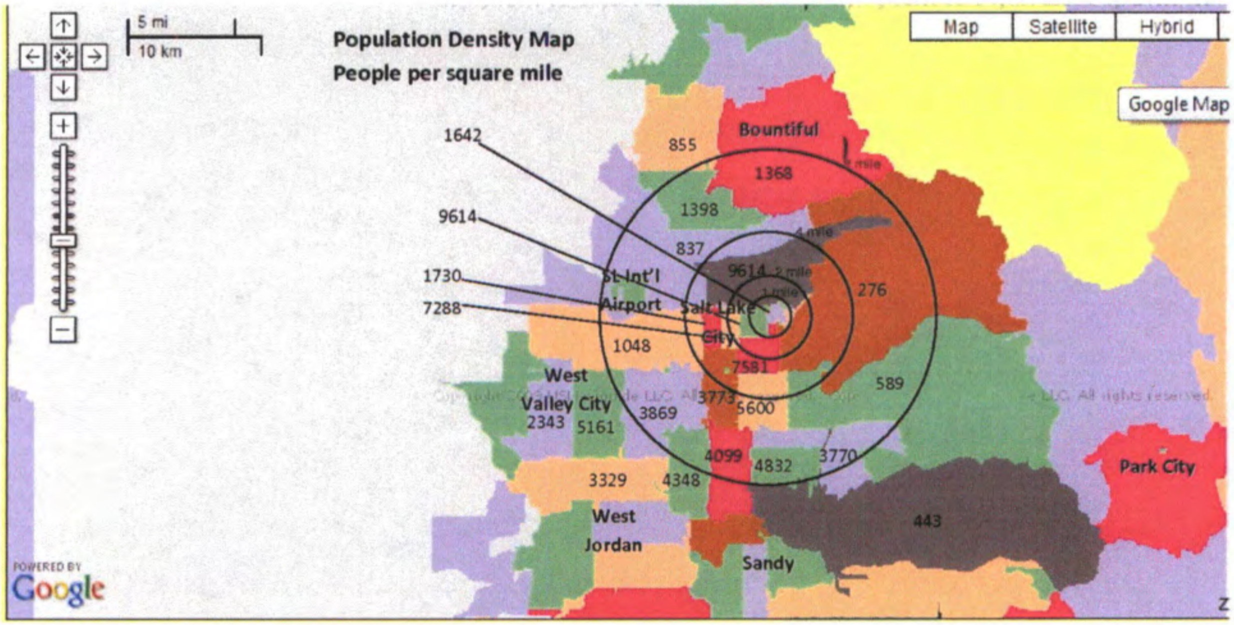


Figure 2.1-4 University of Utah Campus and Facilities



http://www.zipmap.net/Utah.htm

Figure 2.1-5 Population Density

Table 2.1-1 Population Data

Community	Direction from U of U Campus	Distance yards (m)	Population
Salt Lake City Metropolitan Area	SW	NA	1,130,293
Salt Lake City	W	2.4 miles (3.9 km)	181,742
University of Utah	S	NA	30,000
Student Residence Halls	SE	1009 (923)	1250
Merrill Engineering Building	NA	NA	500
University of Utah Hospital	E	740 (677)	425 Beds
Primary Children's Hospital	E	1000 (14)	252 Beds
Huntsman Cancer Hospital	E	1200 (1097)	50 Beds
Tooele	W	38 miles (61 km)	22,502
Provo-Orem Metropolitan Area	S	50 miles (80 km)	376,774
Ogden-Clearfield Metropolitan Area	N	40 miles (64 km)	442,646
Park City	SE	28 miles (45 km)	7,371
Heber	SE	40 (64km)	7,297
Midwav	SE	40.5 (67 km)	2.121

2.2 Nearby Industrial, Transportation, & Military Facilities

2.2.1 Locations and Routes

The UUTR is located on the northern side of the University of Utah campus. North, Northwest, and Northeast: located 440 yards (400 meters) from the facility are private residences; these houses are situated on quarter acre lots and extend north approximately one mile (1.6 km) before terminating in the mountains. Northwest: industrial freight train yard located 3.4 miles (5.5 km). West and Southwest: located 150 yards (137 m) from the facility are private residences; these houses are situated on quarter acre lots and extend north, west, and, south approximately two miles (3.2 km) before merging with the built up area of downtown Salt Lake City; beyond downtown lies the Salt Lake International Airport with an Air National Guard Transport Wing approximately 6 miles (9.7 km) west. East and Southeast: a TRAX light rail station lies 715 yards (654 m) east the Merrill Engineering Building. The University of Utah Medical Complex is east; this Complex has three hospitals and numerous medical facilities; the medical complex is located more than 740 yards (675 m) distance from UUTR, and this complex terminates to the east in the mountains; the medical complex extends south and merges into university dormitories (1000 yards or 923 m) and the U. S. Army reserve post, Fort Douglas (1500 yards or 1.4 km); beyond Fort Douglas lies University of Utah Research Park which is a complex of research laboratories and one hospital (1.25 miles or 2 km)., **Fig. 2.2-1**.

There is a gas pipeline located 1.3 miles (2.1 m) southeast of the facility. South: the University campus extends south for approximately 1090 yards (997 m) with dormitories on the far south and south southeast (870 yards or 800 m); beyond the campus to the south is a hospital and then residential housing. Only one road is located adjacent to the facility, North Campus Drive as shown in **Fig. 2.2-2.** This road moves traffic in an east west direction. It turns north where it passes closest to the facility (55 yards or 50 m), and then continues east west. North Campus Drive is a four-lane road (two lanes each direction) with a 35 mph (56 km/h) speed limit. Typical traffic consists of hospital and university employees, and students commuting to and from the campus roads.

Figure 2.2-1 Potential hazard locations close by to UUTR

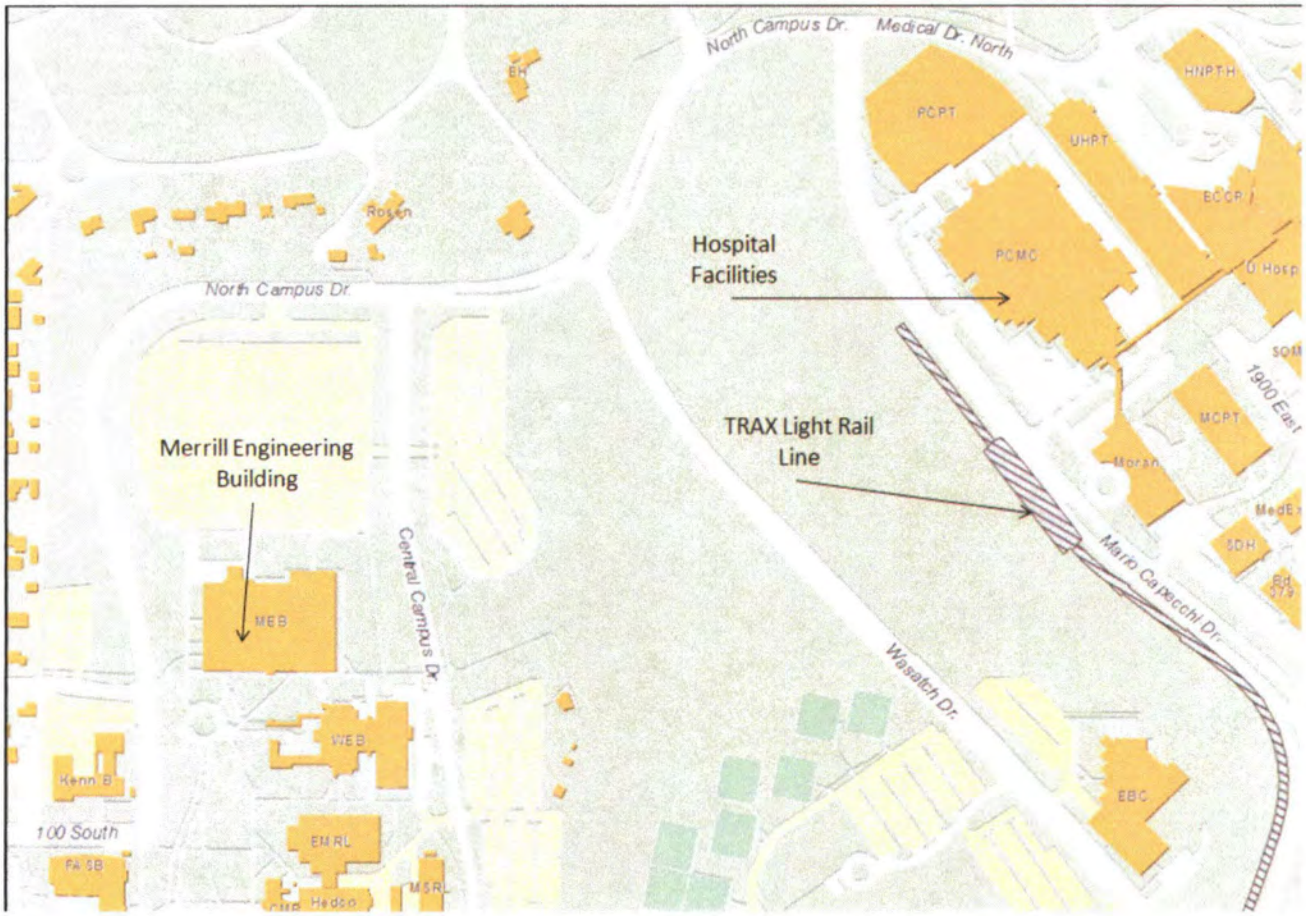


Figure 2.2-2 Location of UUTR housed in the Merrill Engineering Building

2.2.2 Air Traffic

The Salt Lake City International Airport (SLC) is located approximately 7 miles (11 km) west of the University of Utah. SLC aircraft operations are estimated to be 389,321 (takeoffs and landings) in 2008. The Salt Lake International airport does not have standard flight paths over the reactor site or in the vicinity of the university because due to the north/south runway trajectory and westward distance. Both the University of Utah Hospital and the Primary Children's Medical Center located 1 km east of the UUTR have their heliports. The airports are situated such that private flight routes over the UUTR are extremely rare, and the hospitals heliports are small and are used only for shock trauma patients. This limits the accidents to affect the safe operation of the UUTR. In addition, the UUTR's submerged design greatly reduces the risk of any credible external accident.

2.3 Meteorology

The UUTR is located in the Salt Lake Valley, which is located between the Wasatch Mountains and the Oquirrh Mountain Range. The climate is considered moderately mild, characterized by relatively cold, wet winters and warm, dry summers, and receives little precipitation. Under the Koppen climate classification, Salt Lake City has a semi-arid, continental climate [Köppen climate classification *BSk*]. The city has four distinct seasons, with a cold, snowy winter, a hot, dry summer, and comfortable, relatively wet transition periods. The Pacific Ocean is the primary influence on the weather, contributing storms from about October to May, with spring being the wettest season. Snow falls frequently during the winter, contributed largely by the lake-effect from the Great Salt Lake. The only source of precipitation in the summer is monsoon moisture moving north from the Gulf of California. Summers are hot, frequently reaching above 38°C (311.15 °K), while winters are cold and snowy. However, winters are warmer than one would expect at this elevation and latitude, due to the Rocky Mountains to the east and north that usually block powerful polar highs from affecting the state during the winter. Temperatures rarely fall below -18°C (255.15 °K), but frequently stay below freezing. Temperature inversions during winter can lead to thick overnight fog and daytime haze in the valley as cool air, moisture, and pollutants are trapped in the valley by surrounding mountains. [http://en.wikipedia.org/wiki/Climate_of_Salt_Lake_City]

2.3.1 General and Local Climate

2.3.1.1 Humidity

The Salt Lake Valley is a semi arid climate where summer humidity seldom exceeds 20%. The relative humidity by month in Salt Lake City is shown in **Table 2.3-1** (values are given for morning and afternoon) and in **Fig. 2.3-1.**

2.3.1.2 Wind Stability

Wind data taken from downtown Salt Lake Triad Center which is located ~5.6 miles from the UUTR are summarized in **Table 2.3-2**. **Figures 2.3.-2** and **2.3.-3** are showing the wind roses for ozone season and violation days, respectively [http://home.pes.com/windroses/] averaged over the period from 1980 to 1998[1]. The average wind speed and wind direction for the period 2005 to 2010 are shown in **Fig. 2.3-4** and **Fig. 2.3-5** respectively.

[1] These wind roses were prepared by Pacific Environmental Services, Inc. (PES) using the program WRPLOT that was developed by PES for the U. S. Environmental Protection Agency.

Table 2.3-1 Relative Humidity in Salt Lake City
[04/2010: http://www.cityrating.com/cityhumidity.asp?City=Salt+Lake+City]

Relative Humidity by Month (Percentage)

	Jan.	Feb.	Mar.	Apr.	May.	Jun.	Jul.	Aug.	Sep.	Oct.	Nov.	Dec.
Morning	79%	78%	70%	66%	66%	59%	52%	54%	62%	68%	75%	79%
Afternoon	69%	60%	46%	39%	34%	27%	22%	24%	30%	41%	59%	71%

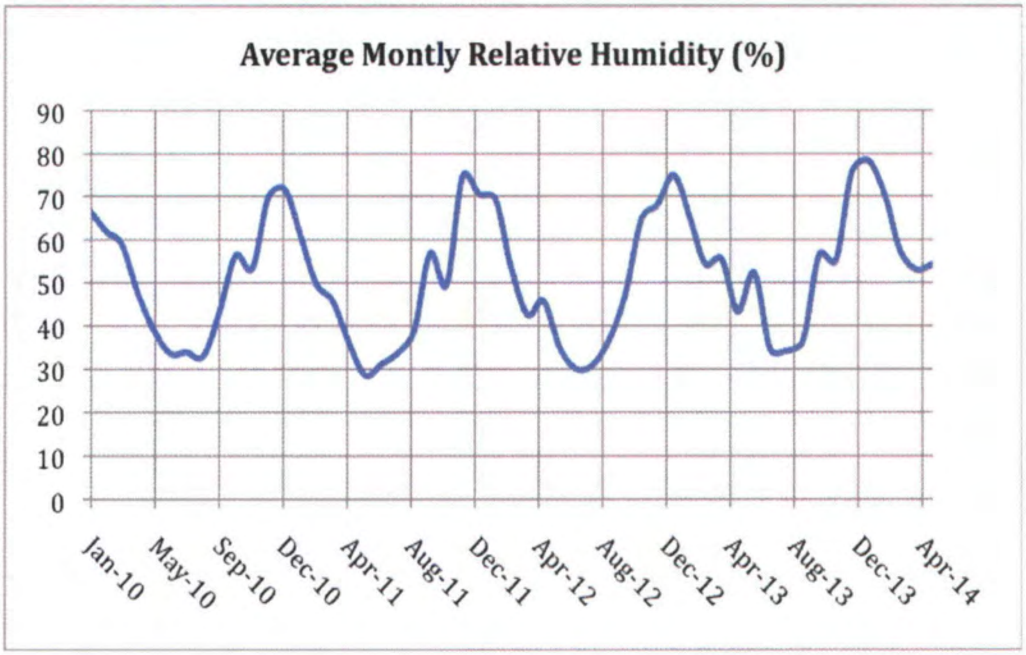
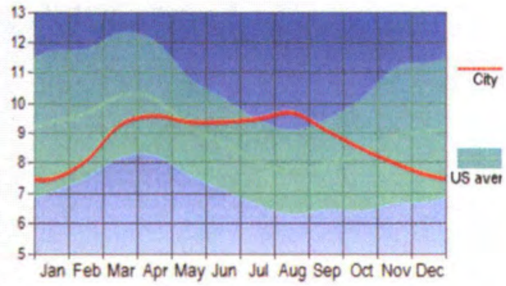


Figure 2.3-1 Average Monthly Relative Humidity in Salt Lake City (2005 to 2010)

Table 2.3-2 Wind Average Speed and Prevailing directions in Salt Lake City (Station 5.6 miles Away from the UUTR, 1930-2003) [Left: http://www.wrh.noaa.gov/slc/climate/slcclimate/SLC/table57a.php] [Right: http://www.city-data.com/city/Salt-Lake-City-Utah.html]

Month	Average Speed (mph)	Prevailing Direction*
January	7.6	SE
February	8.2	SE
March	9.5	SE
April	9.7	SSE
May	9.4	SSE
June	9.4	SSE
July	9.5	SSE
August	9.7	SSE
September	9.1	SE
October	8.4	SE
November	8.1	SE
December	7.7	SE



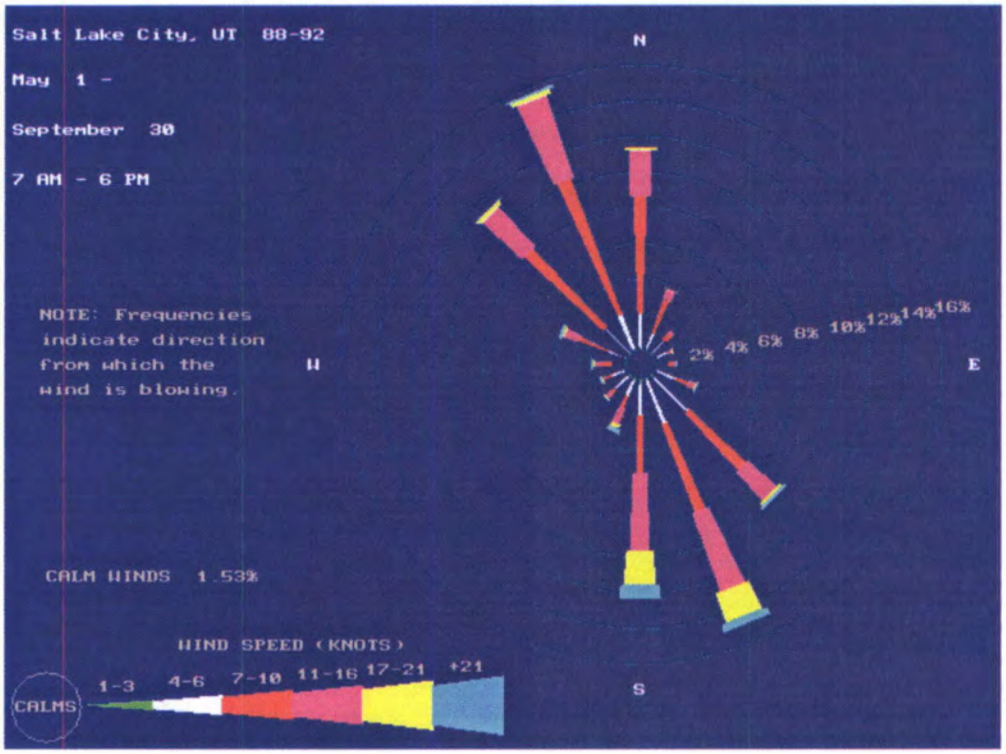


Figure 2.3-2Wind Rose for Ozone Season in Salt Lake City

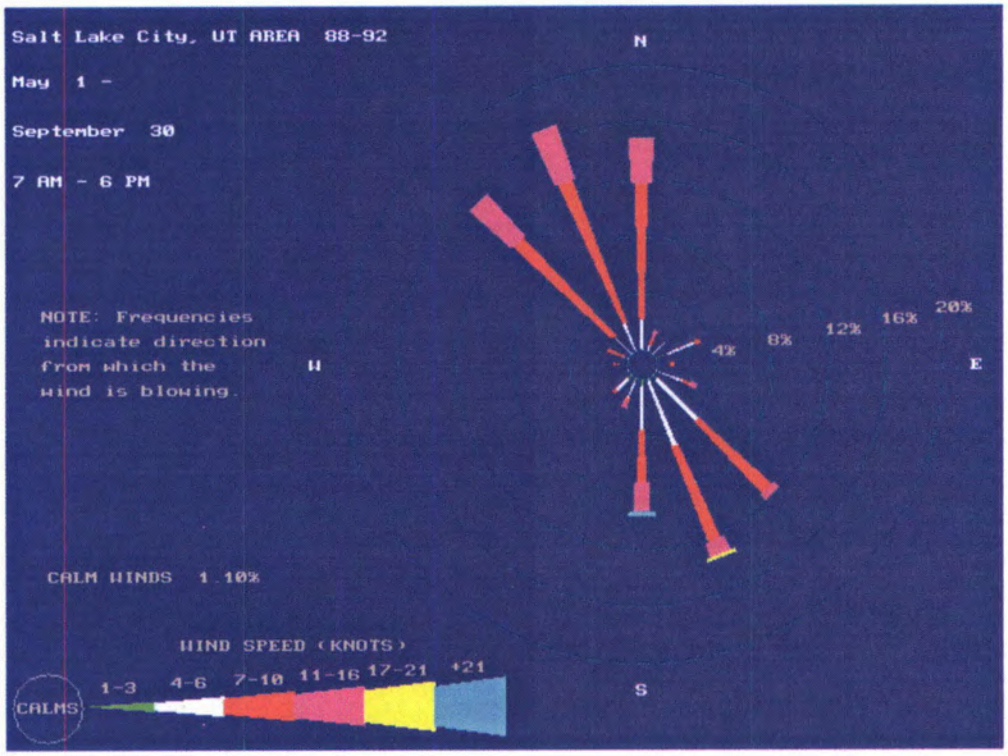


Figure 2.3-3 Wind Rose for Violation Days in Salt Lake City

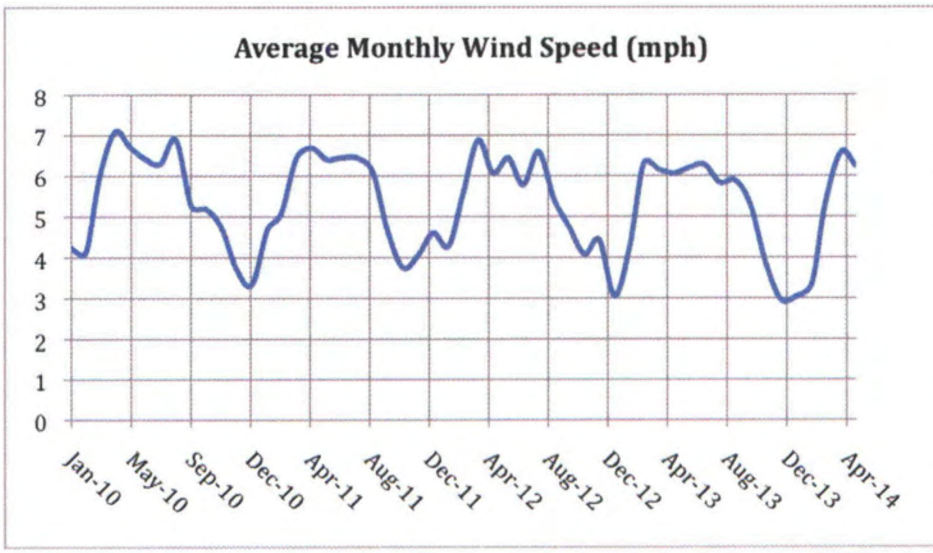


Figure 2.3-4 Average Monthly Wind Speed (mph) in Salt Lake City Area (2005 – 2010)

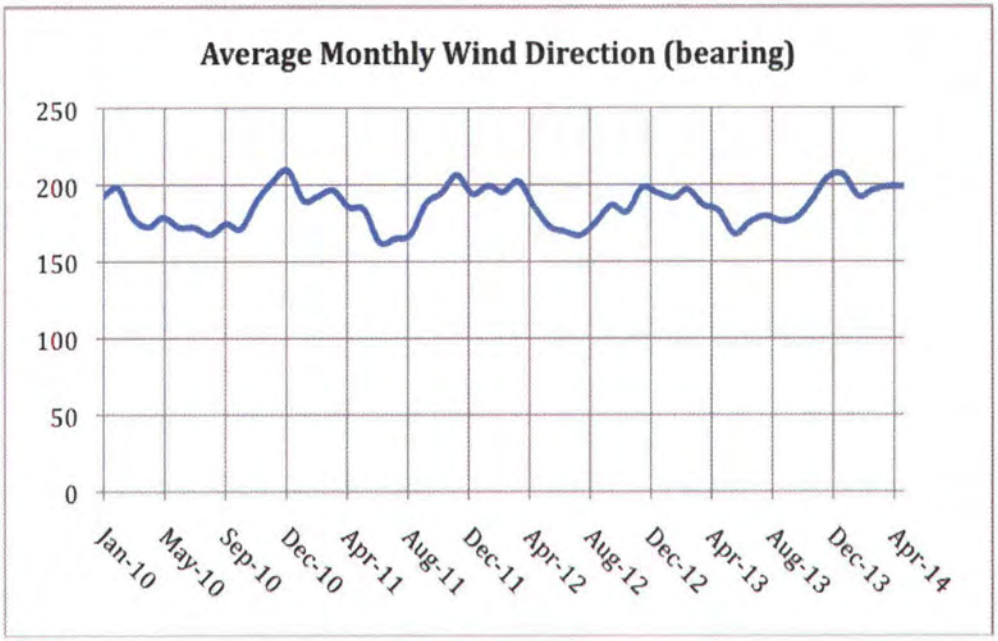


Figure 2.3-5 Average Monthly Wind Direction (mph) in Salt Lake City Area (2005 – 2010)

2.3.1.3 Temperatures

The average minimum daily temperature is -6.11 °C in January and the average maximum daily temperature is 32.78 °C in July. The extreme temperatures range from -34.44°C to 41.67 °C (**Table 2.3-3** and **Fig.2.3-6**).

Table 2.3-3 Historical Average and Extreme Monthly Temperatures in Salt Lake City (Station: International Airport) [http://en.wikipedia.org/wiki/Climate_of_Salt_Lake_City]

		Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Extreme Daily Maximum	°F	63	69	78	89	99	104	107	106	100	89	75	69
Average Daily Maximum	°F	37	43	53	61	71	82	91	89	78	64	49	38
Average Daily Minimum	°F	21	26	33	39	47	56	63	62	52	41	30	22
Extreme Daily Minimum	°F	-22	-30	2	14	25	35	40	37	27	16	-14	-21

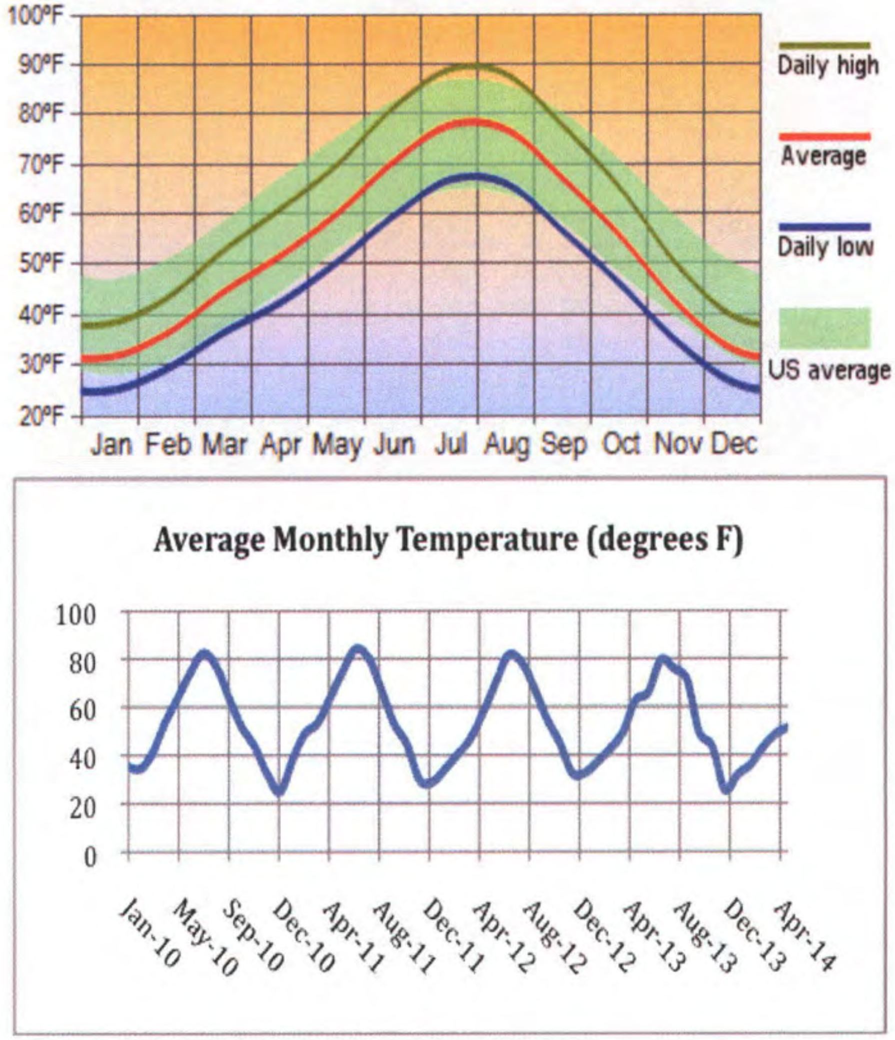


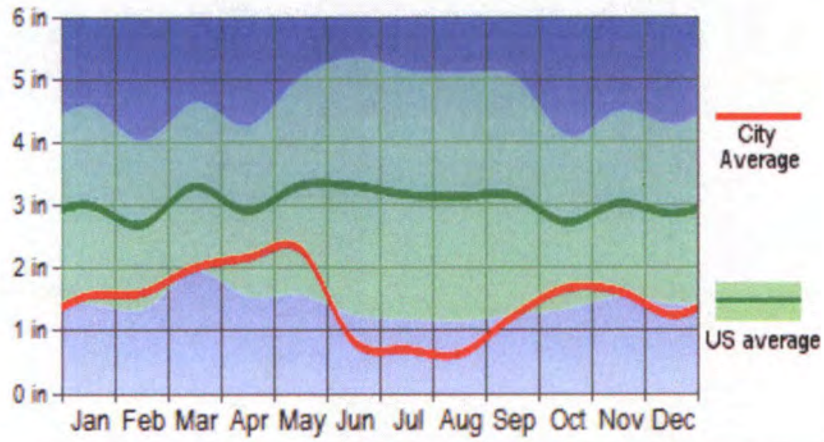
Figure 2.3-6 Average Monthly Temperatures in Salt Lake City [http://www.city-data.com/city/Salt-Lake-City-Utah.html]

2.3.1.4 Precipitation

Precipitation values are shown in **Table 2.3-4** and **Fig.2.3-7**. The average precipitation for the Salt Lake area is 15.27 inches per year, with a maximum of 24.26 inches recorded in 1983 and a minimum of 8.70 inches recorded in 1979.

Table 2.3-4 Average and Record Precipitation in Salt Lake City (Station: International Airport) [http://en.wikipedia.org/wiki/Climate_of_Salt_Lake_City]

		Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Record Precipitation	in.	3.23	4.89	3.97	4.90	4.76	3.84	2.57	3.66	7.04	3.91	3.34	4.37
Average Precipitation	in.	1.37	1.33	1.91	2.02	2.09	0.77	0.72	0.76	1.33	1.57	1.40	1.23
Average Snowfall	in.	13.6	9.9	9.1	4.9	0.6	0.0	0.0	0.0	0.1	1.3	7.0	12.0
Record Snowfall	in.	50.3	32.1	41.9	26.4	7.5	0.0	0.0	0.0	4.0	20.4	33.3	35.2



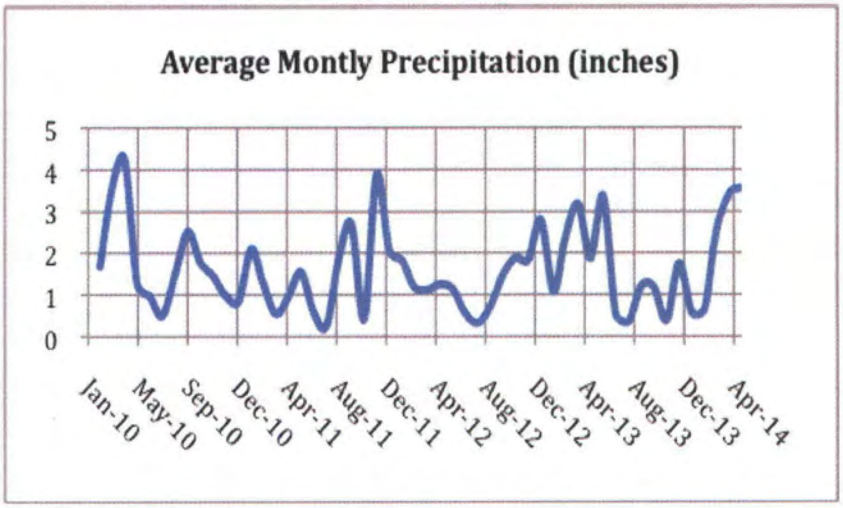


Figure 2.3-7 Average Precipitation in Salt Lake City [http://www.city-data.com/city/Salt-Lake-City-Utah.html]

2.3.1.5 Severe Weather Phenomena

Tornadoes in Utah are quite infrequent due to the relatively calm climate; Utah gets ~2 tornadoes a year. Between the years 1943 and 2002 there have been only 7 tornadoes which have measured as an F2 on the Fujita scale. Tornadoes of this magnitude have wind speeds between 113 and 157 mph and have the ability to tear roofs off some buildings, damage trees and mobile homes, and light objects are blown out. In the time period from 1950 to 2009, only 127 tornadoes were reported in the entire state of Utah (**Table 2.3-5**), most of them being of size FO, or a gale wind tornado, with wind speeds reaching up to 94 mph. Only one of these 127 tornadoes occurred within a 5-mile radius of the UUTR (**Table 2.3-6**). Because of the low frequency and low severity, tornadoes do not represent a significant hazard to the UUTR.

2.3.1.6 Snowstorms and Ice storms

Utah can have some lengthy snow seasons, starting as early as September and going as late as May. The greatest amount of snowfall recorded in a give 24 hour period was on Oct 17-18, 1984 a total of 18.4 inches was recorded. And the greatest depth of snow on the ground recorded was 25 inches on Jan 12, 1993. On average Salt Lake City receives an annual snowfall of 62.7 inches (**Table 2.3-7**).

The most accumulated snow in a given time was 23.3 inches of snow from Jan 6-10 1993 (**Table 2.3-8**). Due to the rather dry conditions in the Utah, Salt Lake City area, there are not very many ice storms, most of the cold storms that the Salt Lake area receives are snowstorms. From 1950-present there is only one recorded day of having an ice storm, it occurred on January 26, 2005 and there was no property or crop damage nor any injuries or fatalities due to this storm.

2.3.2 Analysis of Potential Accidents

Accidents potentially taking place at Fort Douglas (1.4 km), Research Park (2 km), industrial rail yard (5.5 km), and gas pipeline (2km) would have minimal to no affect for reactor operation and safety due to distance from MEB (**Fig. 2.2-2**). The gas pipeline is situated far enough from the reactor facility to pose no significant threat (**Fig. 2.2-2**). Therefore, there are no nearby industrial, transportation, or material facilities that could experience accidents affecting the safety of the UUTR.

Table 2.3-5 Number of Tornadoes by Year in Salt Lake City [http://www.wrh.noaa.gov/slc/climate/tornado.php]

1950	0	1970	5	1990	4
1951	0	1971	1	1991	5
1952	0	1972	0	1992	4
1953	2	1973	0	1993	6
1954	1	1974	0	1994	0
1955	3	1975	0	1995	2
1956	0	1976	0	1996	3
1957	1	1977	0	1997	1
1958	0	1978	1	**1998**	**8**
1959	0	1979	0	1999	5
1960	0	1980	0	2000	7
1961	1	1981	2	2001	4
1962	1	1982	3	2002	4
1963	1	1983	0	2003	4
1964	1	1984	6	2004	2
1965	5	1985	0	2005	4
1966	2	1986	3	2006	2
1967	2	1987	3	2007	1
1968	4	1988	1	2008	0
1969	3	1989	6	2009	3
					Total 127

Table 2.3-6 Strongest Tornadoes Recorded in the State of Utah [http://www.wrh.noaa.gov/slc/climate/tornado.php]

F2	January 22, 1943	Young Ward
F2	June 3, 1963	Bountiful
F2	November 2, 1967	Emery
F2	August 14, 1968	West Weber
F2	May 29, 1987	Lewiston
F3	**August 11, 1993**	**Uinta Mountains**
F2	August 11, 1999	Salt Lake City
F2	September 8, 2002	Manti

Table 2.3-7 Maximum-Minimum seasonal snowfall for Utah [http://www.wrh.noaa.gov/slc/climate/slcclimate/SLC/table44.php][1928-present]

Maximum Seasonal Snowfall	Year		Minimum Seasonal Snowfall	Year
117.3"	1951-52	Normal Annual Snowfall 62.7"	16.6"	1933-34
110.8"	1973-74		18.5"	1939-40
98.7"	1992-93		22.3"	2002-03
98.0"	1983-84		27.9"	2004-05
91.3"	1943-44		30.1"	1940-41+
89.2"	1968-69		30.2"	1980-81
88.2"	1948-49		31.3"	1960-61

Table 2.3-8 Maximum duration of snowfall for Utah [http://www.wrh.noaa.gov/slc/climate/slcclimate/SLC/table50.php][1928-present]

Amount	Duration of Snowfall	
	Began	Ended
23.3"	1:10 pm January 6, 1993	11:05 am January 10, 1993
21.6"	March 12, 1944	March 15, 1944
20.4"	8:17 am February 24, 1998	12:30 pm February 25, 1998
19.8"	1:30 pm January 24, 1996	11:18 pm January 25, 1996
19.4"	8:00 am December 25, 2003	3:00 am December 28, 2003
18.4"	5:04 am October 17, 1984	10:35 am October 18, 1984
18.1"	1:03 pm December 28, 1972	1:30 pm December 29, 1972
17.4"	5:43 am March 1, 1977	3:35 am March 3, 1977
17.4"	6:02 pm April 9, 1974	8:20 pm April 10, 1974

2.4 Hydrology

Hazards from activation of ground water are considered remote. Ground water was not encountered in any of the borings made at the building site (more details are provided in **UUTR SAR 2.6.1**). The concrete and water shielding around the core minimize neutron activation of the surrounding soil.The Merrill Engineering Building which houses the reactor facility is located on the campus on high ground sloping westward with an average slope of about ten percent. Drainage characteristics around the building reactor site are favorable, and no threat of flooding or water damage exists.

The Great Salt Lake, Jordan River and other waterways and waterbeds are of no influence to the UUTR facility, **Fig. 2.4-1.**

Figure 2.4-1 Salt Lake City area map of waterways and waterbeds

2.5 Geology, Seismology, and Geotechnical Engineering

Salt Lake City lies in the northeastern part of the Jordan River Valley, which is surrounded on three sides by mountains and opens toward the northwest into the Great Salt Lake basin. The Oquirrh Range makes up the western boundary of the valley with an average height of 8,500 ft above mean sea level and peaks to 9,700 ft mean sea level. The southern boundary consists of the low east-west traverse range split by a water gap containing the Jordan River. The eastern boundary of the valley is formed by the Wasatch Range with higher summits near 12,000 ft mean sea level. A spur of mountain extends a short distance westward from the main Wasatch Range just north of Salt Lake City. Two main canyons break the Wasatch Range in this area. In addition, many smaller canyons open into the valley from all ranges. The valley floor consists of two parts: a broad, flat central plain at an elevation of approximately 4,225 ft mean sea level and a system of narrow Lacustrine terraces that intervene between the central flat and the bordering mountains. The plain and the surrounding terraces are part of the former lake bottom of prehistoric (late Pleistocene) Lake Bonneville.

Utah has a high earthquake frequency and contains several active fault zones. The Coast and Geodetic Survey (ESSA) of the U. S. government has classified the seismicity region through Central Utah, which includes Salt Lake County, as zone 3. **Figure 2.5-1** shows the major fault lines within the state of Utah [this figure is provided by the University of Utah Seismograph Station].

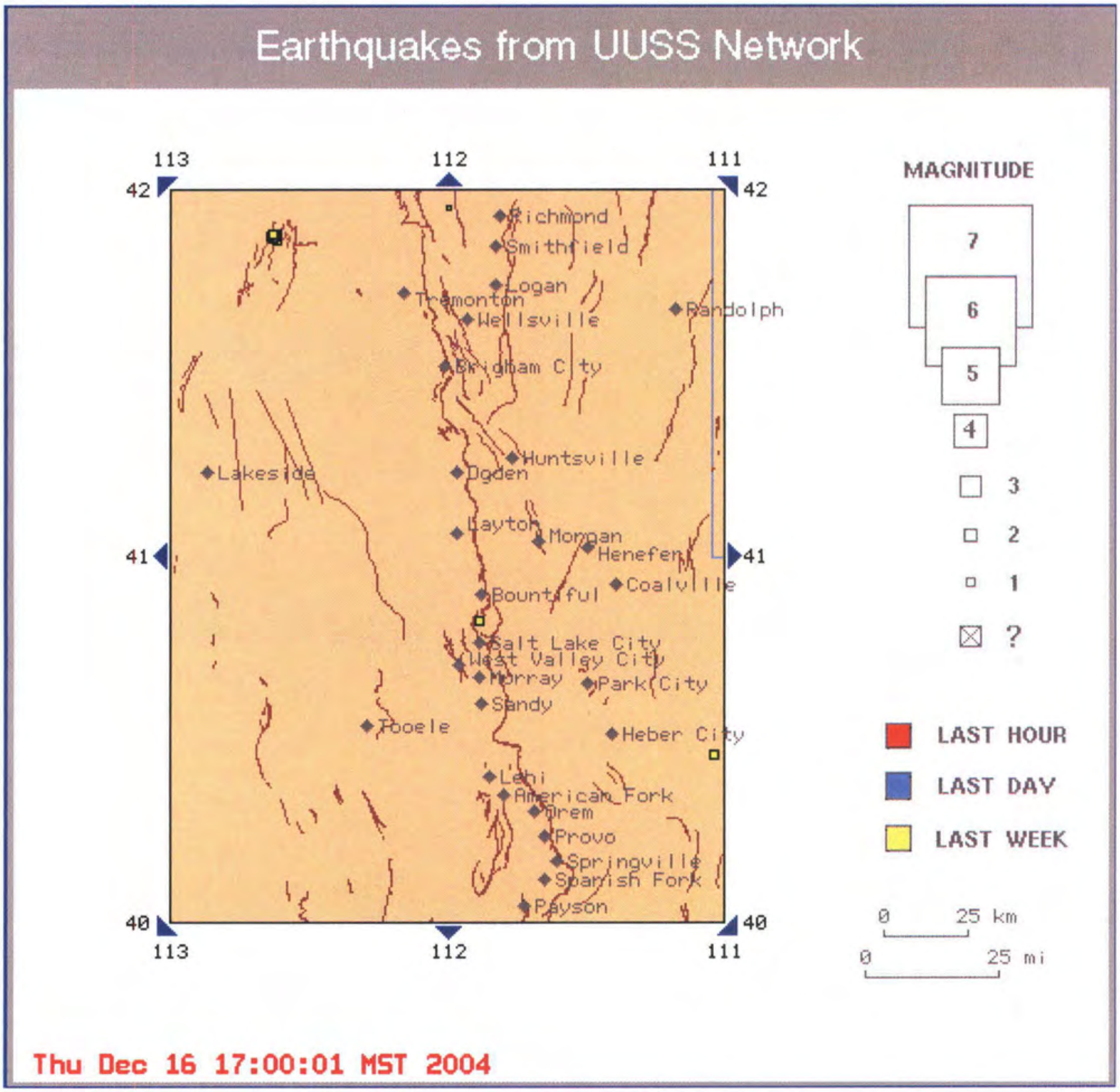


Figure 2.5-1 Fault lines on the Wasatch Front

2.5.1 Site Geology

Information on the foundation soil is available from the "Foundation Investigation for the Merrill Engineering Building", U/U Job No- SA-261, submitted to the Utah State Building Board in November 1957. **Figure 2.5-2** shows the general site geology. The results of three of the borings made at the site are given in **Fig. 2.5-3**. Boring No. 1 was made near the vicinity of the proposed reactor tank excavation. Soils at the site are predominantly granular soils with silty sand (fine) to clay silt binder. Coarser particles range from sand sizes to gravel, cobbles and boulders. Excavation of the fuel storage pits confirmed the bore data. Standard penetration tests were performed consecutively at intervals of approximately 5 ft in depth. The tests were performed by driving a standard 1 3/8" ID split-barrel penetration sampler one foot into the undisturbed soil by means of a 140 lb. drop weight falling 30 inches. Penetration resistance is logged in terms of blows per foot of penetration. This data is recorded to the right of each log boring in **Fig. 2.5-3**. Key to the boring log data is given in **Fig. 2.5-3,** and **Fig. 2.5-3** also gives the safe bearing capacity for footing design at the site. Calculations indicate that the load bearing capacity of foundation soil at the reactor site is more than adequate to support the reactor tank under operation conditions. Actual bearing stress at the reactor tank footing is about 1.9 lbs/ft^2 compared to a bearing capacity of well over 5 lbs/ft^2. No free ground water was encountered at any boring at or in the vicinity of the Merrill Engineering Building.

2.5.2 Seismicity and Maximum Earthquake Potential

Because of the design and construction specifics of the UUTR reactor as well as selection of the site, there is no substantial risk associated with the effects of seismic activity on the TRIGA reactor. The design specifics are as follows: the reactor is inground pool type located 16 ft below the ground level; the inner reactor pool tank is made of 5/16 inches thick aluminum followed by 2 ft sand and 3/16 inches of stainless steel shells. The emergency most likely to be caused by a severe earthquake is a possible breach or rupture of one of the pool tanks resulting in a reduction of water that shields the reactor core. The simultaneous failure of both tanks is not considered to be a credible outcome. A risk assessment of these scenarios was performed, and the results are reported in **UUTR SAR 13** of this document. Since 1964, The Utah Seismograph Station has recorded 86 earthquakes originating within a 5 mile (8.5 km) radius of the reactor. The largest Richter magnitude recorded for these 86 earthquakes is 2.7, with 98% less than 2. The historical record 1800 - 1964 indicate a further 44 earthquakes with a magnitude 7 earthquake occurring in the early 1900's. Data for the magnitude 7 earthquake is incomplete. Expanding the radius to 20 miles the number of earthquakes for the same time period is 1018. The largest magnitude recorded in this area is 5.2 with 95% below 2. The historical record for the 20-mile radius indicates that 58 earthquakes occurred including the 7 previously mentioned.

Only two damaging earthquakes have occurred in Salt Lake County during the past 146 years. A series of shocks with peak magnitude of 5.5 struck the city on 22 May 1910, and an earthquake of magnitude 4.3 occurred on 4 February 1955. Damage was minor in both instances. The most severe earthquake recorded in the state occurred on 14 November 1901 in Richfield, Utah (about 150 miles south of Salt Lake) and consisted of about 35 shocks of peak magnitude 6.7. The most costly earthquake (magnitude 3.8) occurred on 30 August 1962 in Logan, Utah (about 80 miles north of Salt Lake.) Property damage resulting from this earthquake was estimated at about $1,000,000 and consisted of broken windows, cracked plaster, toppled chimneys, etc. There is no record of any deaths directly related to an earthquake in the State of Utah. **Figure 2.5-4** shows earthquake locations in the Salt Lake City area. This seismic hazard map, from the USGS/NEIC PDE catalog for 1990 ~ 2006, shows the peak acceleration (%g) with 2% probability in 50 years site [NEHRP B-C boundary National Seismic Hazard Mapping Project, 2008].

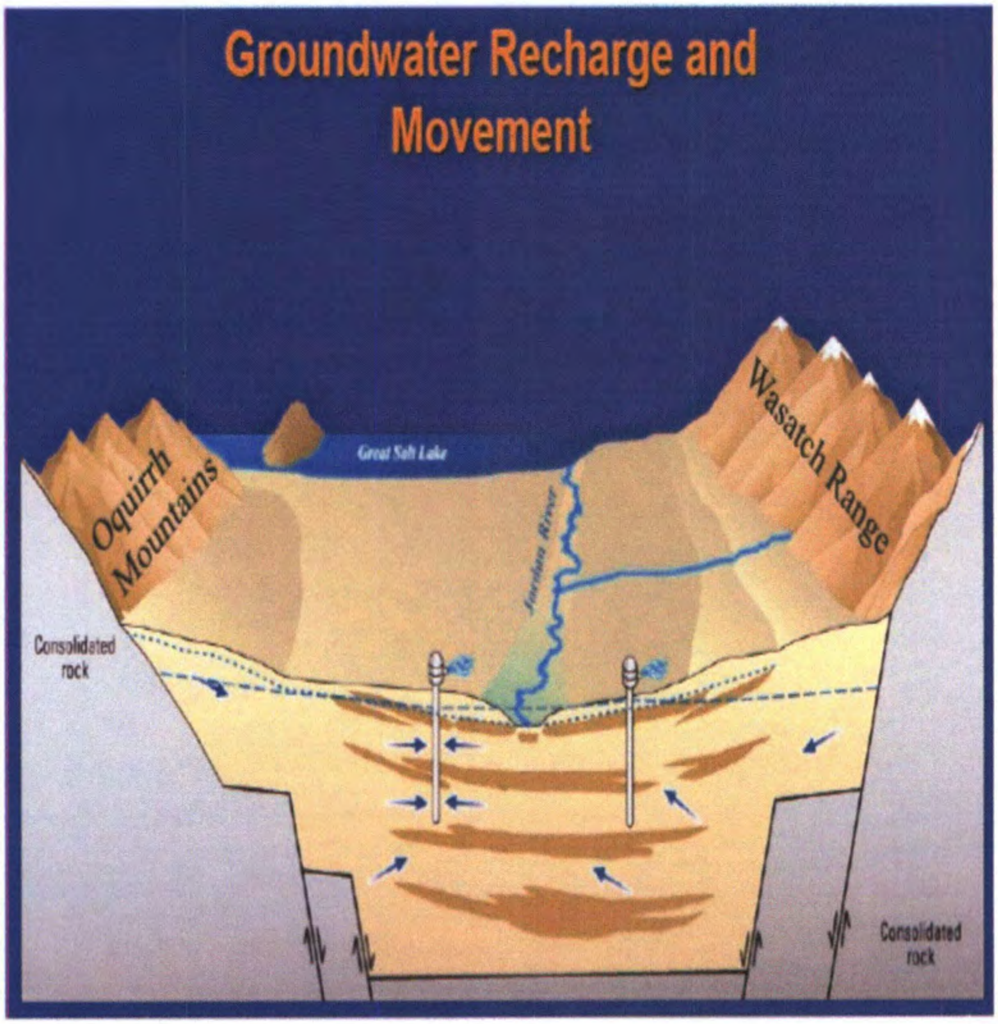


Figure 2.5-2 UUTR Site Geology

R. L. SLOANE ASSOCIATES LOG OF BOARINGS

Job No. 261 CLIENT Utah State Building Board DATE Oct. 03-31, Nov. 1, 1957

PROJECT University of Utah Engineering Building

REMARKS No free ground water encountered

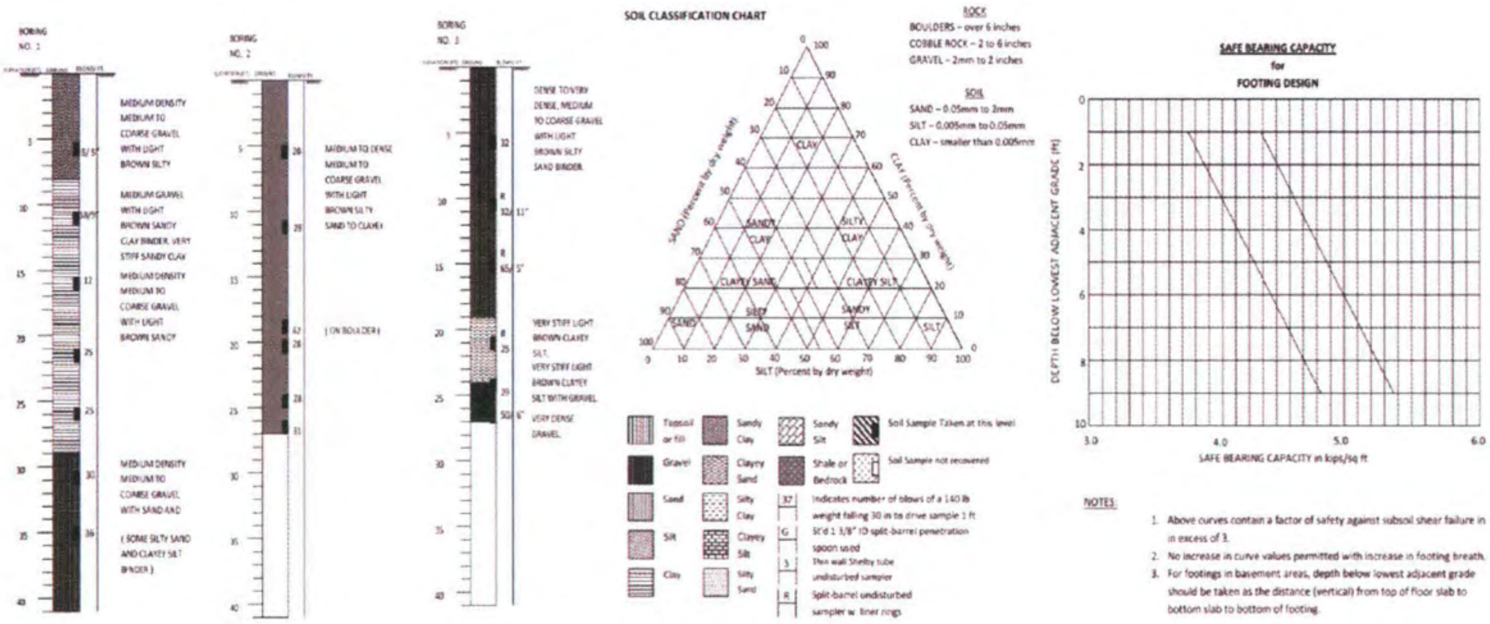


Figure 2.5-3 Log of Borings; Key to Log of Borings; Safe Bearing Capacity for Footing Design

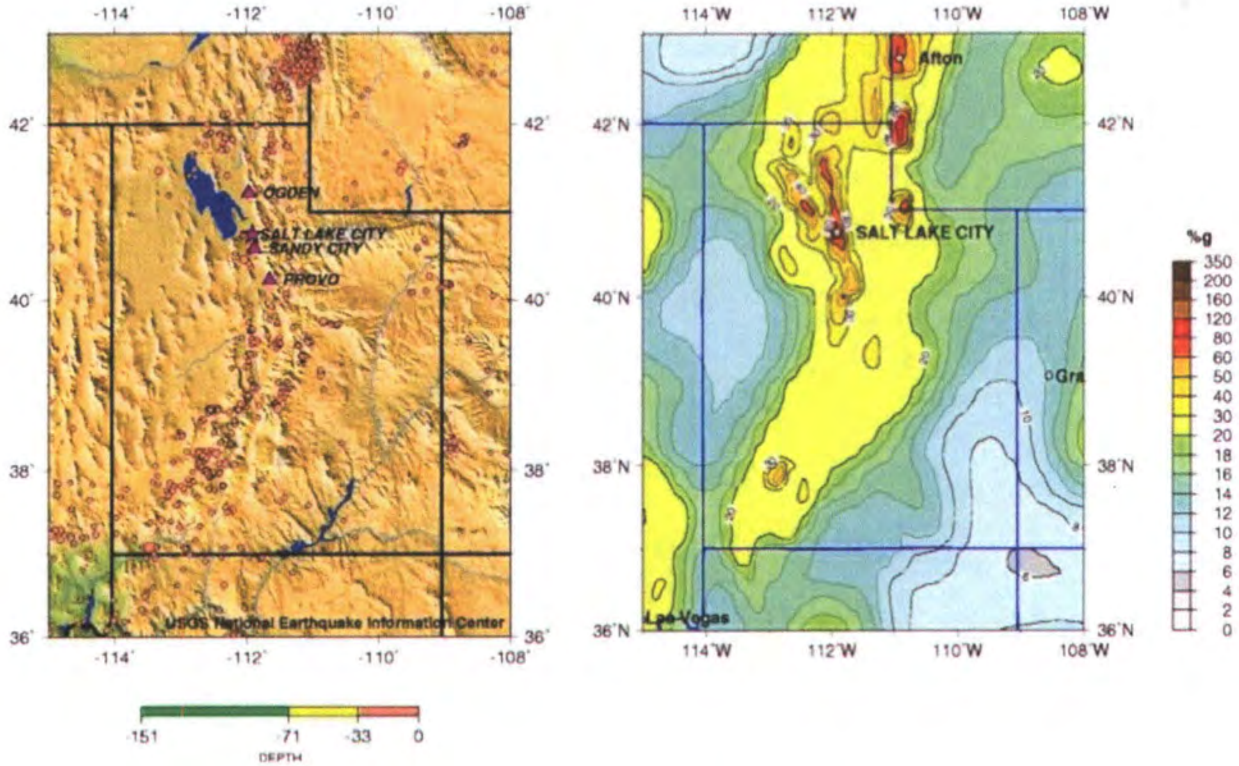


Fig. 2.5-4 Earthquake locations and seismic hazard map of Utah.

2.5.3 Surface Faulting

The local seismic conditions of the reactor site are generally favorable. The nearest fault is the splinter East Bench Fault, which lies about 2,000 ft (609.6 m) west of the reactor site. In the last 146 years only 6 earthquakes have had their origin at the East Bench Fault, and these have been mild and of local effect only. None had a Richter magnitude greater than 4.9. Thus, as specified by ANSI 15.7 the UUTR is farther than 400 meters from the surface location of a known capable fault.

2.5.4 Liquefaction Potential

To some extent, the type of ground on which a structure is built and the nature of the structure has a greater impact on the earthquake safety factors than proximity to a fault zone. In the case of the UUTR, the reactor site is a dry, compact mantle of earth covering bedrock. According to a 1994 study of liquefaction potential in the Salt Lake Valley conducted by the Utah Geological Survey, the probability of soil liquefaction occurring at the reactor site is very low (less than 5% given that an earthquake of sufficient magnitude to cause soil liquefaction will occur within the next 100 years). The reactor system is composed of dual containment tanks, which will move with the earth as a unit. The outer steel tank and its heavy reinforced concrete pad can withstand tensile, compressive, and shear forces of short duration. The inner aluminum tank, which contains the reactor core and shielding water, is sufficiently ductile to respond to earthquake motion without tank rupture. The construction of the reactor tank, concrete pad, footings, and structure conforms to zone 3 requirements of the Uniform Building Code.

3. DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Introduction

This chapter discusses the "General Design Criteria for Nuclear Power Plants" as set forth in 10 CFR 50, Appendix A, as they apply to the UUTR. These General Design Criteria were formulated for the purpose of establishing minimum requirements for principal design criteria of water-cooled nuclear power plants. Furthermore, they are to be applied to construction of new plants similar in design for which construction permits have been previously issued. Since the UUTR is a research and testing reactor, many of the power plant criteria cannot be logically applied. Therefore, the discussions of this chapter are oriented towards a separate consideration of each individual criterion, rather than toward identification of areas of noncompliance and corrective actions.

The nominal UUTR steady-state power level is 100 kW. Thus, the fission-product inventory is orders of magnitude less than those of conventional power reactors for which the General Design Criteria were primarily prepared. A conservative upper limit of energy released for an entire year of operation would be about 1 MW-day. These comparisons illustrate why the UUTR may be placed in a category of much lower risk and treated accordingly in a rigorous review of compliance with the General Design Criteria.

The accidents described in **UUTR SAR 13**, Accident Analysis, conservatively demonstrate that instrumented shutdown actions and building confinement are not necessary to ensure that radiological doses will not exceed allowable limits. **Table 3.1.1** presents a synopsis of the conclusions regarding the relevance of the General Design Criteria to the UUTR.

3.1.1 Overall Requirements (Criteria 1-5)

Criterion 1: Quality Standards and Records

Original structures, systems, and components important to safety were designed, fabricated, constructed, and/or tested to design specifications (MAN-NDI-86-03) and associated standards.

University of Utah engineers monitored the construction to assure that the construction work was in accordance with the specifications. Modifications have been made in accordance with existing standards and requirements.

Criterion 2: Design Bases for Protection Against Natural Phenomena

Hurricanes, tsunamis, and seiches do not occur in the Salt Lake Valley area. Flooding in the valley area could be caused by run-off from local rainstorm activity or snowmelt from the Wasatch Mountains. However, the UUTR is situated some 400 ft above the 100-yr. flood plain, and no local depressions are capable of holding floodwater.

Only a small number of tornadoes, one or two per year, have been reported in Utah. Based on the small probability of occurrences, postulated low intensity, the intermittent type of reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the UUTR structure. However, the buildings are designed to withstand the area wind loads.

The Salt Lake Valley area is classified as being in Seismic Zone 3 as defined in the Uniform Building Code. The UUTR structures have been designed and constructed in accordance with this code, see **UUTR SAR 13**. Seismic activity in the region has registered as high as 7 Richter scale in historical time, which indicates an upper limit on the most likely seismic event (refer to Section 2.5). Since the UUTR is designed to the Uniform Building Code for Zone 3, there is ample conservatism in the design for the maximum expected event. The UUTR structures may suffer some damage from a seismic event of the highest possible yield, but, as previously noted, low frequency of operation and low fission-product inventory minimize the risk of such an event, and resultant radiological doses would be within the ranges evaluated in **UUTR SAR 13**, Accident Analysis.

Criterion 3: Fire Protection

The reactor room and reactor control room structures, built of steel, concrete, and concrete block, are highly fire resistant. However, material inventories inside the rooms could include various flammable materials (paper, wood, etc.); and these, coupled with potential ignition sources, require that fires be considered.

Several features reduce both the likelihood and the consequences of a fire.

1. Periodic fire safety inspections are made by the Fire Marshall.
2. Periodic in-house inspections are made for the explicit purpose of reducing nonessential combustible material inventory.
3. Fire detection and suppression systems are installed in the facility.
4. A control room window into the reactor room permits the reactor operator to continuously observe the reactor room, so that immediate action can be taken to minimize the effects of a fire; established emergency procedures would be put into effect in the event of a fire.
5. The large volume of water in the reactor tank would protect the core from any conceivable fire.
6. The reactor is fail-safe and will shutdown should fire damage the instrumentation or control system.

If the fire alarm is tripped, the Salt Lake City Fire Department is automatically alerted and will respond to the UUTR within a few minutes.

Criterion 4: Environmental and Missile Design Bases

The construction of the facility precludes catastrophic rupturing of the reactor tank. There is no source within the reactor room for generating large, sustained, positive pressure differentials that could cause a breach of the reactor tank walls.

The amount of combustible materials allowed in the reactor facility has been limited to preclude damage to the reactor should the materials ignite. The closed beam tubes are angled upwards away from the core. This tubing does not penetrate the inner tank wall. There are no high-pressure systems within the reactor room. The piping systems are anchored and do not penetrate the tank walls, and they could not conceivably affect the reactor.

Criterion 5: Sharing of Structures, Systems, and Components

Electrical power and make up water constitute the only systems shared by the UUTR with other users. Sharing is based on the fact that the UUTR electric power is supplied by the power source for the Merrill Engineering Building (MEB). Loss of power results in the shutdown of the reactor since all control circuits are fail-safe, and no power is required for safe shutdown or to maintain safe shutdown conditions. An electric power failure at any point in the UUTR network will not detrimentally affect the reactor. Furthermore, battery backup power exists for security and radiation monitoring systems. Make up water is also supplied from the MEB's water system. This is not a significant safety issue because of the slow seepage rate from the reactor in the case of a tank wall rupture. (**UUTR SAR 13**)

3.1.2 Protection by Multiple Fission - Product Barriers (Criteria 10-19)

Criterion 10: Reactor Design

Accident analyses presented in **UUTR SAR 13** show that under credible accident conditions, the safety limit on the temperature of the reactor fuel would not be exceeded. Consequently, there would be no fission product release that would exceed 10 CFR Part 20 allowable radiation levels.

Criterion 11: Reactor Inherent Protection

Because of the fuel material (U-ZrH) and core design, there is a significant prompt negative temperature reactivity coefficient. Routine steady-state power operation is performed with the safety, shim, and regulating rods partially withdrawn, as described in **UUTR SAR 4** and **13**.

Criterion 12: Suppression of Reactor Power Oscillations

Not applicable. Due to the small dimensions of the core and low power levels, the reactor is inherently stable to space-time and xenon oscillations.

Criterion 13: Instrumentation and Control

The instrumentation and control system for the UUTR is a modified General Atomics TRIGA Reactor console. The console provides a safety channel (percent power with scram), a redundant, multi-range, linear power safety channel (source level to full power with scram), a wide-range log percent power channel (below source level to full power), a source count rate channel (fission chamber with control rod interlocks), period indication, two fuel temperature channels (C and D rings with scram), and two pool water temperature channels (bulk pool water and refrigeration water). Additional scrams are triggered by a loss of pool water level, a high voltage interrupt tripping the external security system, and airborne radiation levels in excess of the high area radiation monitor's limits. The UUTR can be operated only in manual mode. Percent withdrawal indicators show control rod position and movement. Interlocks prevent the movement of the rods in the up direction under any one the following conditions: scrams not reset, source level below minimum count, two UP switches depressed at the same time.

Criterion 14: Reactor Coolant Pressure Boundary

The reactor tank and cooling systems operate at low pressure and temperature. The vessel is open to the atmosphere, and there are no means for pressurizing the system. The reactor tank is of a double wall construction with an inner wall constructed of aluminum and the outer wall constructed of steel. The primary coolant system components are aluminum, stainless steel, or PVC. The system components outside the reactor tank have a low probability of serious leakage or of gross failure that could propagate. Furthermore, the design of the system is such that even though a line or component ruptures only a small amount of water would be removed from the tank (< 3 ft) by a siphon block (see **UUTR SAR 5**). Rupture of the reactor tank is virtually impossible since it is supported on the bottom by a reinforced concrete slab and the tank wall is buttressed by the surrounding the earth.

Criterion 15: Reactor Coolant System Design

The reactor tank is an open system and the maximum pressure in the primary system is due to the static head (about 23.5 ft). The primary cooling system, the secondary cooling system, and the purification system are pressurized by small capacity pumps.

The piping and valves in the primary and purification systems are PVC or aluminum, and of such size so as to provide adequate operating margins. The secondary system components are PVC and carbon steel. **UUTR SAR 5** describes the coolant system in detail.

Criterion 16: Containment Design

The structure surrounding the reactor constitutes a confinement building rather than providing absolute containment. Because of the low fission-product inventory, leakage from the structure can be tolerated.

Criterion 17: Electric Power Systems

The building (MEB) provides electrical power to the reactor console, continuous air monitor (CAM), and area radiation monitors (ARM), during normal reactor operations. With loss of building power, a battery backup system is available. THE UUTR is cooled by PASSIVE natural convection and there is no requirement to provide forced cooling flow for the removal of decay heat, so response time for the reactor operator to shutdown the reactor and confirm shutdown is adequate. The battery backup system provides twenty-four hours of power to the [redacted], CAM, and ARM.

Criterion 18: Inspection and Testing of Electric Power Systems

University of Utah electrical maintenance crews maintain the primary power distribution system supplying commercial power to UUTR. Routine inspections of the systems are performed. The UUTR can tolerate a total loss of electric power with no adverse effects on the safety of the facility. There are no electrical power (distribution) systems necessary to provide cooling to the UUTR during either normal or abnormal conditions (see **UUTR SAR 13**).

Criterion 19: Control Room

In the event of an accident when operating procedures require shutdown of the reactor, occupancy of the control room is not necessary since the reactor has been shut down and experiments are terminated. Exposure levels derived from an accident's radiation sources would be significantly reduced in magnitude (due to the location of the control room with respect to the reactor room). Consequently, control room radiation levels will not exceed allowable tolerance levels; nevertheless, the UUTR Emergency Plan describes actions for mitigating accident situations, which require control room evacuation.

3.1.3 Protection and Reactivity Control Systems (Criteria 20-29)

Criterion 20: Protection System Functions

The UUTR Reactor Protection System has been designed to initiate automatic actions to assure that fuel design limits are not exceeded by anticipated operational occurrences or accident conditions. Two power channels and two fuel temperature channels initiate the automatic actions. The Reactor Protective System automatically scrams the control rods when trip settings are exceeded (**UUTR SAR 7**). There are no other automatic actions required by UUTR systems to keep fuel temperature limits from being exceeded. The Reactor Protective System satisfies the intent of IEEE-323-1974 in the areas of redundancy, diversity, power-loss, fail-safe protection, isolation and surveillance.

Criterion 21: Protection System Reliability and Testability

The UUTR protection system is designed to be fail-safe. Any channel signal or functional loss that causes the channel to lose its ability to perform its intended function results in initiation of shutdown action. Protective action is manifested through several independent scram inputs arranged in series such that any event that interrupts current to the scram magnets results in shutdown of the reactor. Redundancy of channels is provided. In addition, a loss of any channel due to open circuit or loss-of-power will result in a scram. Scram action is, therefore, on a one-out-of-one basis. All instrumentation is provided with testing capability. The Reactor Protective System satisfies the intent of the IEEE-323-1974 standard. Additionally, all safety systems are tested during a reactor checkout prior to a run or control rod manipulation.

Criterion 22: Protection System Independence

The protective system satisfies the intent of IEEE-323-1974 "Criteria for Protective Systems for Nuclear Power Generating Stations." Protective functions are initiated through two independent power and two independent fuel temperature channels that provide a diversity of independent scram modes. Furthermore, the protective system is fail-safe upon loss of power.

The Reactor Protective System and the magnet power supply are, for the most part, physically and electrically isolated from the remainder of the control system. There is a separate conduit for each safety channel and one for the magnet power supply.

Criterion 23: Protection System Failure Modes

The reactor protective system is designed and constructed to be failing safe in the event of any failure of a safety channel. Failure of a safety channel will result in removal of power to the control rod magnets, dropping the control rods into the core. The reactor protective system contains no control functions. Therefore, loss of a protective function will not affect the operation of the reactor, such as initiating an uncontrolled reactivity insertion.

Criterion 24: Separation of Protection and Control Systems

The UUTR has four power indicating channels, two fuel temperature channels, and two water temperature channels. One of the power channels utilizes a fission chamber for startup count rates. This channel provides signals for both safety (source interlocks) and control action. The linear power channel utilizes a compensated ion chamber. This channel provides linear power for safety (scram) action as well as power monitoring capability. The third channel is a percent power channel. It is the primary safety channel with scram and power monitoring capabilities. The forth power monitoring channel (log channel) is not linked to a safety function. Fuel temperature is measured by thermocouples placed within the special instrumented fuel elements. The information from these channels is processed and displayed on the console independently, and connects directly to the safety system scram circuit. The ability of this configuration to meet the intent of protection system requirements for reliability, redundancy, and independence for TRIGA-type reactors has been accepted by the NRC.

Finally, the control and safety systems are fail-safe and will scram the reactor should they malfunction. No control or safety systems are required to maintain a safe shutdown condition.

Criterion 25: Protection System Requirement for Reactivity Control Malfunction

The UUTR Protection System is designed to assure that fuel temperature limits are not exceeded for any single malfunction of the reactivity control system. However, **UUTR SAR 13** shows that the simultaneous and accidental removal of all rods from the core at their normal drive speed will not exceed fuel temperature limits.

Criterion 26: Reactivity Control System Redundancy and Capability

The UUTR has three independent reactivity control rods: one shim rod, one regulating rod, and a safety rod. The shim and safety rods have nearly identical reactivity worths; therefore, for certain specialized applications, the role of these two rods is reversed. Both rods are capable of shutting the reactor down well below the shutdown margin. Each of the CONTROL rods has its own drive mechanism and control circuit and they can scram independently of each other. The regulating rod is used for fine power control and adjustment.

Upon receipt of a scram signal, all three rods are released from their drives and dropped into the core. Insertion of any two of the three rods ensures reactor shutdown.

Criterion 27: Combined Reactivity Control System Capability

Emergency core cooling is not required for the UUTR. Analyses (**UUTR SAR 13**) have shown that the worst conditions resulting in instant loss of coolant do not cause fuel-element temperatures to reach the safety limit.

Total worth of the rods is more than adequate to maintain the core at a subcritical level even with the most reactive rod removed from the core.

Criterion 28: Reactivity Limits

No conceivable malfunction of the reactivity control systems could result in a reactivity accident worse than the conditions encountered during an accidental maximum-yield pulse, as outlined in **UUTR SAR 13**. As shown in **UUTR SAR 13**, neither continuous rod withdrawal nor loss of coolant will cause undue heating of the fuel. Identified accidents will not result in significant movement of adjacent fuel elements or otherwise disturb the core so as to add reactivity to the system.

Since the coolant system operates at atmospheric pressure, control-rod ejection is not a credible event. The shim rod, the regulating rod, and the safety rod cannot drop out of the core because the barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. Travel out of the core in the downward position is therefore eliminated.

Criterion 29: Protection Against Anticipated Operational Occurrences

The protection and reactivity control system satisfy all existing design standards. Periodic checks (i.e., startup, shutdown, and maintenance procedures) of all reactor protective system channels and reactivity control systems demonstrate that they will perform their intended function. If there were a loss of electrical power, the reactor would scram due to the loss of current to the control rod electromagnets. Because the reactor is cooled by natural convection, and there is no requirement to provide forced cooling flow for the removal of heat, there is sufficient time for the reactor operator to shutdown the reactor and confirm shutdown.

3.1.4 Fluid Systems (Criteria 30-46)

Criterion 30: Quality of Reactor Coolant Pressure Boundary

The reactor tank is open to the atmosphere and is subjected only to ambient conditions. All components containing primary coolant (i.e. reactor tank, primary coolant systems and the purification system) are constructed of PVC, aluminum, and stainless steel, using standard codes for quality control. There is no requirement for leak detection in the primary coolant or purification loop since no conceivable leak condition can result in the tank water level lowering more than approximately three feet.

Criterion 31: Fracture Prevention of Reactor Coolant Pressure Boundary

Since the coolant system is open to the atmosphere, no reactor coolant pressure boundary is required. The reactor tank is of double wall construction with the aluminum inner wall surrounded by eighteen inches of sand and a steel outer wall, which prevents external forces from being directly transmitted to the tank, and precludes movement of the tank.

The tank wall cannot be inspected by any means other than visual observation through the water from inside the tank. All piping, valves, meters, etc. associated with the primary water system are located in open spaces and are readily accessible for periodic inspections.

The UUTR operates at relatively low powers and temperatures. Because of the moderate fluence levels and low temperature factors, no significant change in material properties is expected.

Criterion 32: Inspection of Reactor Coolant pressure Boundary

This criterion is not applicable.

Criterion 33: Reactor Coolant Makeup

The UUTR water purification system design includes a system for makeup of primary coolant water. This system is manually operated (See **UUTR SAR 5**)

Criterion 34: Residual Heat Removal

Natural convection cooling is adequate to dissipate the heat from the core. Many years of operations with TRIGA reactors have shown that natural convection will provide adequate flow for the removal of heat after several hours of maximum steady-state operation. Furthermore, calculations performed for loss of coolant (see **UUTR SAR 13**) show that fuel temperatures will not reach the safety limit even under loss-of-coolant conditions. Based on the above, there is no requirement to provide a residual heat removal capability.

Criterion 35: Emergency Core Cooling System

An emergency core-cooling system is not required for the following reasons:

1. The system is not pressurized and does not operate at high coolant temperatures.
2. Natural convection cooling will adequately dissipate generated core heat.
3. If all water is lost for any reason, air cooling (as shown by analysis in **UUTR SAR 13**) will satisfactorily dissipate heat and prevent exceeding the safety limits.

Criterion 36: Inspection of Emergency Core Cooling System

This criterion is not applicable.

Criterion 37: Testing of Emergency Core Cooling System

This criterion is not applicable.

Criterion 38: Containment Heat Removal

There are no systems, devices, equipment, experiments, etc., with sufficient stored energy to require a containment heat removal capability. This criterion is not applicable

Criterion 39: Inspection of Containment Heat Removal System

This criterion is not applicable.

Criterion 40: Testing of Containment Heat Removal System

This criterion is not applicable.

Criterion 41: Containment Atmosphere Cleanup

Post accident activities are not contingent upon maintaining the integrity of the building structure. Accident analyses (see **UUTR SAR 13**) have shown that downwind doses would not exceed 10 CFR Part 20 or ANSI 15.7 guidelines in any credible accident. Routine operations result in two isotopes of concern being produced: Argon-41 in the reactor room and Nitrogen-16 from the irradiation of oxygen in the tank water. Analysis in **UUTR SAR 11** shows concentrations to be below ANSI 15.7 guidelines for accident situations and below 10 CFR Part 20 guidelines for normal operations.

Criterion 42: Inspection of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 43: Testing of Containment Atmosphere Cleanup Systems

This criterion is not applicable.

Criterion 44: Cooling Water

A coolant system is utilized to cool reactor tank water during normal operation of the reactor. The UUTR requires no auxiliary cooling system for cooling of reactor tank water upon shutdown.

Criterion 45: Inspection of Cooling Water System

Cooling equipment used in normal operation of the reactor is located either in the reactor room, with adequate space provided to permit inspection and testing of all components. Operation of the bulk coolant and cooling system is checked on a daily basis prior to reactor operation. During this checkout, the performance of each system is monitored with emphasis on system flow rates and temperature.

Criterion 46: Testing of Cooling Water System

UUTR reactor cooling systems are routinely checked, tested, and maintained.

3.1.5 Reactor Containment (Criterion 50-57)

Criterion 50: Containment Design Basis

Under the conditions of a loss of coolant, it is conceivable that the temperature at the reactor room could increase slightly due to heating of the air flowing through the core.

However, since the building is not leak tight, it will not pressurize from the heating of the air. Furthermore, there is no requirement from a radiological exposure viewpoint for a containment structure; hence, only a confinement structure exists at the site. In addition, there is no source of energy (from an accident) that would provide a significant driving force if no corrective action was taken.

Criterion 51: Fracture Prevention Boundary

The confinement structure (the reactor room) is a building with reinforced filled concrete walls and the ceiling is a steel and concrete load-bearing floor. The entire structure is exposed to only normal external environmental conditions and internal environmental conditions are maintained at regulated conditions.

The structure will not be subjected to significant internal pressures during normal operations. Postulated accident conditions cannot result in significant changes in the pressure differential due to the non-leak tightness of the structure.

Criterion 52: Capability for Containment and Leakage Rate Testing

This criterion is not applicable.

Criterion 53: Provisions for Containment Testing and Inspection

The reactor room confinement capability is checked on a daily basis prior to reactor operation and routinely throughout reactor operations. This check involves monitoring the pressure differentials between the reactor room and the surrounding areas. The reactor room exhaust recirculation system is checked monthly to confirm proper operation.

Criterion 54: Piping Systems Penetrating Containment

Piping systems that involve penetrations through the reactor building walls have no effect on the safety of operation; therefore, isolation, redundancy, and secondary containment of these systems are not required.

Criterion 55: Reactor Coolant Pressure Boundary Penetrating Containment

The reactor room was not designed or constructed as a pressure containment structure, but does provide adequate airborne radioactivity confinement. As pointed out in the responses to previous criteria, there are no requirements for containment (or confinement) capabilities. The only systems that penetrate the reactor room are the ventilation system, primary coolant system, makeup water system, control and console monitoring cables, and the pneumatic transfer tube experimental facility.

Criterion 56: Primary Containment Isolation

Penetrations through the building walls have no effect on the safety of reactor operations; therefore, isolation systems are not required in the UUTR.

Criterion 57: Closed System Isolation Valves

The UUTR reactor building was designed to provide only confinement capability; isolation valves are not required.

3.1.6 Fuel Radioactivity Control (Criteria 60-64)

Criterion 60: Control of Release of Radioactive Materials to the Environment

There is no readily available path for liquid waste to be discharged directly to the environment. Liquids in the reactor room may result from spills, wash down of the floor, etc. These liquids are collected in a storage tank within the UUTR, analyzed for radioactivity, and disposed of accordingly (see **UUTR SAR 13**).

Criterion 61: Fuel Storage and Handling and Radioactivity Control

[redacted]

Criterion 62: Prevention of Criticality in Fuel Storage and Handling

Fuel-storage capability is provided by storage racks [redacted] The storage locations are criticality safe due to the geometry utilized and the limited quantity of fuel elements, which can be stored (**UUTR SAR**

9). Since only one fuel element can be handled at a time, handling does not present a criticality problem.

Criterion 63: Monitoring Fuel and Waste Storage

The reactor room and the UUTR fuel storage area radiation level are monitored with both an ARM system and a CAM system. No residual heat removal or temperature measuring capability is required for irradiated UUTR fuel elements. Fuel burn up is low; therefore, only a minimum decay heat source is present.

Criterion 64: Monitoring Radioactivity Releases

The radiation monitoring system for the UUTR consists of the ARM's and CAM's. ARM's monitor the reactor room and selected areas outside the reactor room for gamma activity. The UUTR exhaust stack is equipped with a CAM, which provides continuous readings of radiation from Ar-41 and beta/gamma particulate released from the facility. This CAM has local readouts and alarms as well as remote readouts and alarms in the reactor control room. Actions initiated to reduce the release of radioactivity if the set points of this instrument are exceeded are discussed in **UUTR SAR 9** and **UUTR SAR 11**. This CAM has local readouts and alarms as well as remote readouts and alarms in the reactor control room.

Table 3.1-1 General design criteria and the UUTR- Protection by multiple fission product barriers and Protection and Reactivity control systems

Criterion Number and Title	Compliance	Compliance not required	Conditional non compliance	Conditional compliance
1. Quality Standards and records	X			
2. Design Basis for Protections against natural phenomena	X			
3 Fire protection	X			
4. Environmental and Missile design basis	X			
5. Sharing of structures	X			
Protection by Multiple Fission Product Barriers				
10. Reactor design	X			
11. Reactor Inherent Protection	X			
12. Suppressions of Reactor Power Oscillations	X			
13. Instrumentation and Control	X			
14. Reactor Coolant Pressure Boundary	X			
15. Reactor Coolant system design	X			
16. Containment Design	X			
17. Electrical Power system	X			
18. Inspection and testing of electrical Power systems	X			
19. Control room	X			
Protection and Reactivity Control Systems				

20. Protection system function	X			
21. Protection system reliability and testability	X			
22. Protection system independence	X			
23. Protection system failure modes	X			
24. Separation of protection and control system	X			
25. Protection system Requirements for reactivity control malfunctions	X			
26. Reactivity control system redundancy	X			
27. Combined reactivity control system capability	X			
28. Reactivity limits	X			
29. Protection against anticipated operational occurrences	X			
Fluid Systems				
30. Quality of reactor coolant pressure boundary		X		
31. Fracture Prevention of Reactor coolant pressure boundary		X		
32. Inspection of reactor coolant pressure boundary	X			
33. Reactor coolant makeup	X			
34. Residual heat removal		X		
35. Emergency core cooling		X		
36. Inspection of emergency core cooling		X		
37. Testing of Emergency core cooling system		X		
38. Containment heat removal	X			
39. Inspection of Containment heat removal		X		
40. Testing of Containment heat removal		X		
41. Containment of Atmosphere Cleanup	X			
42. Inspection of Containment of Atmosphere Cleanup		X		
43. Testing of Containment atmosphere Cleanup system		X		
44. Cooling water	X			
45. Inspection of cooling waster system				X
46. Testing of Cooling water system				X
Reactor Containment				
50. Containment design basis	X			
51. Fracture prevention of Containment Pressure boundary	X			
52. Capability for Containment Leakage rate testing		X		
53. Provisions for containment testing and inspection		X		
54. System Penetrating Containment		X		
55. Reactor Coolant Pressure boundary penetrating containment				X
56. Primary containment isolation		X		
57. Closed system Isolation Valves		X		
Fuel and Radioactivity Control				
60. Control of Releases of Radioactive Materials to the Environment	X			
61. Fuel Storage and Handling and Radioactive	X			

control				
62. Prevention of Criticality in Fuel storage and Handling	X			
63. Monitoring Fuel and Waste storage	X			
64. Monitoring radioactivity Releases	X			

3.2 Meteorological Damage

The UUTR is protected from damage by high winds or tornadoes by virtue of the thick reinforced concrete structure surrounding the reactor tank. The UUTR is located below ground level and is within a double-wall tank, within the Merrill Engineering Building. The superstructure of the UUTR has been designed for area wind, rain, snow, and ice loads. The appropriate building codes were used. The UUTR has endured approximately 25 years of local weather conditions with no meteorological damage. Hurricanes, tsunamis, and seiches do not occur in the Salt Lake area.

Only a small number of tornadoes, approximately two a year, have been reported in Utah (as described in **Section 2.3**). Based on the small probability of occurrences, postulated low intensity, intermittent reactor operation and low fission-product inventory, no criteria for tornadoes have been established for the UUTR structure.

3.3 Water Damage

Ground Water: Ground water was not encountered in any of the borings made at the building site that houses the UUTR.

Precipitation: The precipitation in Salt Lake City area including rain and snow is discussed in **2.3.1,** *General and Local Climate*, indicating no excess in precipitation occurred.

Floods: The Merrill Engineering Building which houses the UUTR is located on the campus on high ground sloping westward with an average slope of about 10%. Drainage characteristics around the building reactor site are favorable, and no threat of flooding or water damage exists. There is no potential for flooding. However, even if flooding occurred, reactor safety would not be an issue since the core is located in-a water pool. The only equipment that can potentially be affected is pool water circulation pump. This pump can be replaced very easily and even in the worst case, the reactor pool water cannot be released from the pool and there is no possibility to contaminate the flooding water or ground water system. There are no pipes in the UUTR facility capable of flooding the reactor room to the first floor level. If, however, reactor area is flooded, the reactor will be shutdown safely.

There is no river small or large near the reactor facility. Furthermore, the lowest elevation in the UUTR, the reactor room floor, contains a sump. There was no flooding history in the University of Utah area since the existence of the University.

3.4 Seismic Damage

The UUTR site is in a UBC, Zone 3, risk area (see **UUTR SAR 2**). The UUTR building, reactor foundation, shielding structure, reactor tank, and core support structure are designed in accordance with UBC Zone 3 requirements. Meeting these requirements should ensure that the reactor could be returned to operation without structural repairs following an earthquake likely to occur during the plant lifetime. Furthermore, failure of the reactor tank and loss of the coolant in the event of a very large earthquake has been considered in **UUTR SAR 13** and the consequences found acceptable from the standpoint of public safety. Seismic considerations applicable to the UUTR facility are discussed in **UUTR SAR 2** and **UUTR SAR 13**.

3.5 Systems and Components

The systems and components used for safe operation and shutdown of reactor and their design basis are as follows:

- UUTR safety operation <u>follows the defense-in-depth mechanism</u> to protect the reactor personnel and the environment from radiation releases. The first barrier to stop the radiation and most importantly confine the fission products is the fuel cladding. The cladding acts as the first barrier for release of fission products. The next barrier preventing the release of radiation is the tank of water, which protects the environment and the personnel. Reactor tank, and the sand in between the tank and the concrete base, both serve as a shielding barrier to the environment.
- The operation of the reactor is <u>regulated with control rods</u>. As in any other TRIGA type reactors, the control rods are held in place on a bridge over the reactor tank. The shim and safety rods have the electrical-driven rack-and-pinion drives consisting of a two-phase reversible motor, a magnet rod-coupler, a rack-and-pinion gear system, and electromagnet and armature, a dashpot assembly, control rod extension shaft, and a potentiometer to identify the location of the rod. These are original TRIGA reactor control rod designs by General Atomics. During a SCRAM signal, the drop is done by gravity, which means the control rod, the rod extension, and magnet armature are detached from the electromagnet. The regulation rod is very similar to the safety and shim rods; however, the difference is in the reversible motor. A stepping motor and reduction gear are used instead of the reversible motor.

- Although there are no required engineered safety features for this reactor due to low operating power and good fission product retention in the fuel, a controlled ventilation system maintains a negative air pressure in the reactor area to reduce the consequences of airborne radiological release. A ventilation system is activated in a rare case of fission product release to protect the personnel and environment from release of such radiation. In the event that the airborne activity in the reactor room would exceed the pre-set level of 10 mrem/hr, the ventilation system will switch to a limited intake mode, which will provide some confinement of the air in the reactor room thus minimizing a spread of fission products and other radiation around the boundaries of the UUTR facility.
- The UUTR has several SCRAM functions: linear power, percent power, high voltage interrupted, pool water level, high radiation alarm, magnet key and manual SCRAM. The linear and percent power will SCRAM the reactor when the reactor power is equal to or higher than 100kW, and 110% of the licensed power, respectively. If the high voltage supply is lost, or radiation level in the reactor room is higher than 10 mrem/hr, the reactor will be scrammed.
- The UUTR has two different types of fuel element cladding, the aluminum and the stainless steel cladding. The B and C-rings contain only stainless steel fuel elements; D, E, F and G-rings have 23 aluminum fuel elements.
- Currently, four different neutron irradiation facilities are available at the UUTR. The FNIF (Fast Neutron Irradiator Facility) provides 1 MeV equivalent neutron flux. The PI (Pneumatic Irradiator) uses a pressurized He gas to place a sample in the core and remove it from the core. The TI (Thermal Irradiator) has well thermallized neutron beam. The CI (Central Irradiator) has the highest neutron flux in the core and provides a fast neutron flux.
- The UUTR has two instrumental fuel elements in the C and D-ring to measure fuel element temperature. The UUTR has one fission chamber, one compensated ionization chamber and two uncompensated ionization chambers. The fission chambers measure the neutron population in the core, while the ionization chambers measure linear, percent and log power of the reactor.

4. REACTOR CORE DESCRIPTION

[redacted]

4.1 Summary Description

The UUTR reactor is a standard design nominal 100kW (licensed 100kW but operated at 90kW), natural- convection-cooled TRIGA pool reactor with the graphite reflector providing accommodation to few pool irradiation facilities. The reactor core is located near the bottom of a water-filled aluminum tank that has a diameter of 8 ft and is about 24 ft deep. For personnel shielding, the tank is shielded radially by a minimum of 2 ft thick sand, 3/16 inches-thick steel and 3 ft thick ordinary concrete. The approximately 22 ft of water above the core provides adequate shielding at the top of the UUTR tank. The control rod drives are mounted at the top of the tank on a bridge structure spanning the diameter of the tank. The reactor can be operated in a steady-state mode by manual control.

TRIGA fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1,150 °C (1,423.15 °K). The inherent safety of the UUTR reactor has been demonstrated by the extensive experience acquired from similar TRIGA systems throughout the world. This safety arises from the strongly negative prompt temperature coefficient that is characteristic of uranium-zirconium hydride fuel-moderator elements. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. This results in a mechanism whereby reactor power excursions are terminated quickly and safely. The analyses that follow establish the safety limits for operation of the UUTR.

4.2 Reactor Core

The UUTR utilizes solid fuel elements, developed by General Atomics (GA), in which the zirconium-hydride moderator is homogeneously combined with enriched uranium. The unique feature of these fuel-moderator elements is the prompt temperature coefficient of reactivity, which gives the TRIGA reactor its built-in safety by automatically limiting the reactor power to a safe level in the event of a power excursion. The reactor core consists of a lattice of cylindrical stainless-steel-cladding U-$ZrH_{1.6}$ fuel-moderator elements (SS), and aluminum-cladding U-$ZrH_{1.0}$ fuel-moderator elements. The aluminum and SS fuel core elements are nominally [redacted] inches O. D., except for the upper and lower end fixtures, and [redacted] inches long tip to tip. Neutron reflection in the radial direction is provided by 12 graphite and 12 heavy water elements in an aluminum cladding. The height of the graphite and heavy water elements in the reflector is about 24 inches. Also in this tank at the outer perimeter, is 3.5 ft of water, which acts as a thermal shield to protect the aluminum tank structure from excessive nuclear heating, and also contributes to reducing the radiation dose to the aluminum tank.

The core components are contained between top and bottom aluminum grid plates. The top grid plate has 126 positions for fuel elements and control rods arranged in 6 concentric rings around a central port (used for high flux irradiations).

The power level of the UUTR reactor is accurately controlled with three control rods. Four instrumentation channels monitor and indicate the reactor neutron flux and power level on the console. The bulk water temperature and the reactor tank outlet and inlet water temperatures are indicated on the console. The water conductivity, measured at the inlet and outlet of the demineralizer, is displayed on a panel near the console. In addition, primary reactor water is routinely monitored to identify any significant increase in radioactivity.

4.2.1 Reactor Fuel

4.2.1.1 Reactor Fuel Description

The NUREG 1282 identifies the safety limit for high-hydride ($ZrH_{1.7}$) fuel elements with stainless steel cladding based on the cladding stress (resulting from hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding if fuel temperature is below 1,150 °C (1,423.15 °K) . A change in yield strength occurs for stainless steel cladding temperature of 500 °C (773.15 °K), but there is no scenario in TRIGAs for fuel cladding to achieve 1,000 °C (1,273.15 °K) while submerged; consequently the safety limit during reactor operation is conservatively set at 1,000 °C (1,273.15 °K) for stainless steel cladding fuel elements. Therefore, the important variable for a TRIGA reactor is the fuel element temperature. The NUREG 1537 Appendix 14.1 indicates that for reactors without instrumented fuel it is allowed to establish a power level that limits fuel cladding maximum temperature below safety limits.

During operation, fission product gases and dissociation of the hydrogen and zirconium builds up a gas inventory in internal components and spaces of the fuel elements. Limiting the maximum fuel temperature prevents an excessive internal pressure that could be generated by heating the gases. The temperature at which phase transitions may lead to cladding failure in aluminum-clad low-hydride fuel elements is reported to be 530 °C (803.15 °K), (Technical Foundations of TRIGA, GA-471, 1958, PP. 63-72). Fuel growth and deformation can occur during normal operation, as described in General Atomics Technical Report E-117-833. Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. The operating fuel temperature of 750 °C (1,023.15 °K) does not have significant effect on the fuel growth.

All aluminum and stainless steel (SS) cladding fuel elements and the optional graphite or heavy water loaded reflector elements have similar outside dimensions. The aluminum and SS fuel core elements are nominally [redacted] inches O. D., except for the upper and lower end fixtures, and [redacted] inches long tip to tip. The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing 8.5% by weight of uranium enriched to less than 20% in U-235. The hydrogen to zirconium ratio is 1.0 for the elements with aluminum cladding and 1.6 for the elements with stainless steel cladding. To facilitate hydriding, a 0.19 inch diameter hole is drilled through the center of the active fuel section. A zirconium rod is inserted in this hole after hydriding is completed. Each aluminum-clad element has a clad thickness of 0.030 inch-thick aluminum, and the stainless steel clad elements have a clad thickness of 0.020 inches. The active fuel portion is [redacted] inch nominal diameter by [redacted] inches long for aluminum fuel elements and [redacted] inches long for stainless steel elements. Each end has 4 inch-long sections of graphite. The lower end fixture supports the fuel-moderator element on the bottom grid plate. The upper fixture contains a knob for grasping with the fuel-handling tool. Also, the upper fixture has a triangular spacer, which permits cooling water to flow through the upper grid plate. **Figure 4.2-1** and **Table 4.2-1** show the details of the UUTR's two different types of fuel elements. Stainless steel fuel elements have same geometry as aluminum fuel elements except the fuel meat length, overall length and the clad thickness. The U-235 mass is different per element; the values as shown in **Table4.2-1** are the average U-235 mass values per fuel element based on the existing records from General Atomics fuel log. The active part of fuel element is shown in **Fig. 4.2-2**.

The UUTR has two instrumental fuel elements (IF). An instrumental fuel element has three thermocouples embedded in the fuel. The sensing tips of the IF element thermocouples are located halfway between the outer radius and the vertical centerline at the center of the fuel section and 1 inch above and below the horizontal center. The thermocouple leadout wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. The water-tight stainless steel conduit carrying the leadout wires to the top of the reactor tank. The stainless steel conduit is attached to the triangular supporter above the water surface. The IF has same dimension and shape except the three thermocouples and stainless steel conduit.

The exact burnup ratio of each fuel element in the UUTR is not known. In order to develop a full and as accurately as possible a numerical model of the UUTR reference core operating at 90kW using MCNP5, it was important to as accurately as possible, estimate the fuel burnup. The UUTR maximum operating power is 90kW. At that power the UUTR reactor operation record shows that during its operation from 1975 to 1998 the reactor generated cumulative of 191.643MWh and from 1998 to May 7, 2010 the UUTR generated cumulative of 14.221MWh, giving a total of 205.864MWh (for 2439.712 hours). According to General Atomic, the TRIGA fuel burn-up is 1.25g of U-235 per MWh. Thus, from 1975 to 1998, the mass of burned U-235 is 191.643MWh x 1.25g/MWh = 239.554g, and the average burned U-235 per fuel element is 239.554g / 78 elements = 3.07g/element. Similarly, from 1998 to 2010, the burned U-235 is 14.221MWh x 1.25g/MWh = 17.776g, and the average burned U-235 per fuel element is 17.776g / 78 elements = 0.228g/element. The estimated average burnup data are shown in **Table 4.2-2**.

Based on the data given in **Tables 4.2-1** and **4.2-2** it follows: the UUTR core has 17 SS cladding fuel elements and 2 instrumental fuel elements present in the core from 1975 with the average burn-up ratio of (3.07+0.228)/37.6 = 8.77%, 23 aluminum cladding fuel elements present in the core from 1975 with the average burn-up ratio of (3.07+0.228)/37 = 8.91%, and 36 SS cladding fuel elements present in the core from 1998 with the average burn-up ratio of 0.228 / 37.6 ≈ 0.61%.

[redacted]

Fig. 4.2-1 UUTR stainless steel element assembly. Aluminum element has same geometry as a stainless steel element except for the fuel section length, overall length and cladding thickness (Table 4.2-1)

Table 4.2-1 Specification summary for the UUTR fuel elements

Fuel-moderator material	**Aluminum Cladding**	**Stainless Steel (SS) Cladding**	**Un-irradiated New Fuel with Stainless Steel (SS) Cladding**
$UZrH_n$			
H/Zr atom ratio	1.0	1.6	1.6
Uranium content	8.5 wt%	8.5 wt%	8.5 wt%
U-235 enrichment	Less than 20 %	Less than 20 %	Less than 20 %
U-235 mass per element	[redacted]	[redacted]	[redacted]
Diameter of fuel meat	[redacted]	[redacted]	[redacted]
Length of fuel meat	[redacted]	[redacted]	[redacted]
Graphite end reflector			
Porosity	20 %	20 %	20 %
Diameter	3.556 cm	3.556 cm	3.556 cm
Length (top and bottom), each	10.16 cm	10.16 cm	10.16 cm
Cladding			
Material	Aluminum	Stainless steel 304	Stainless steel 304
Wall thickness	0.0762 cm	0.0508 cm	0.0508 cm
End fixture	Aluminum	Stainless steel 304	Stainless steel 304
Overall element			
Outside diameter	[redacted]	[redacted]	[redacted]
Length	[redacted]	[redacted]	[redacted]
Weight	[redacted]	[redacted]	[redacted]

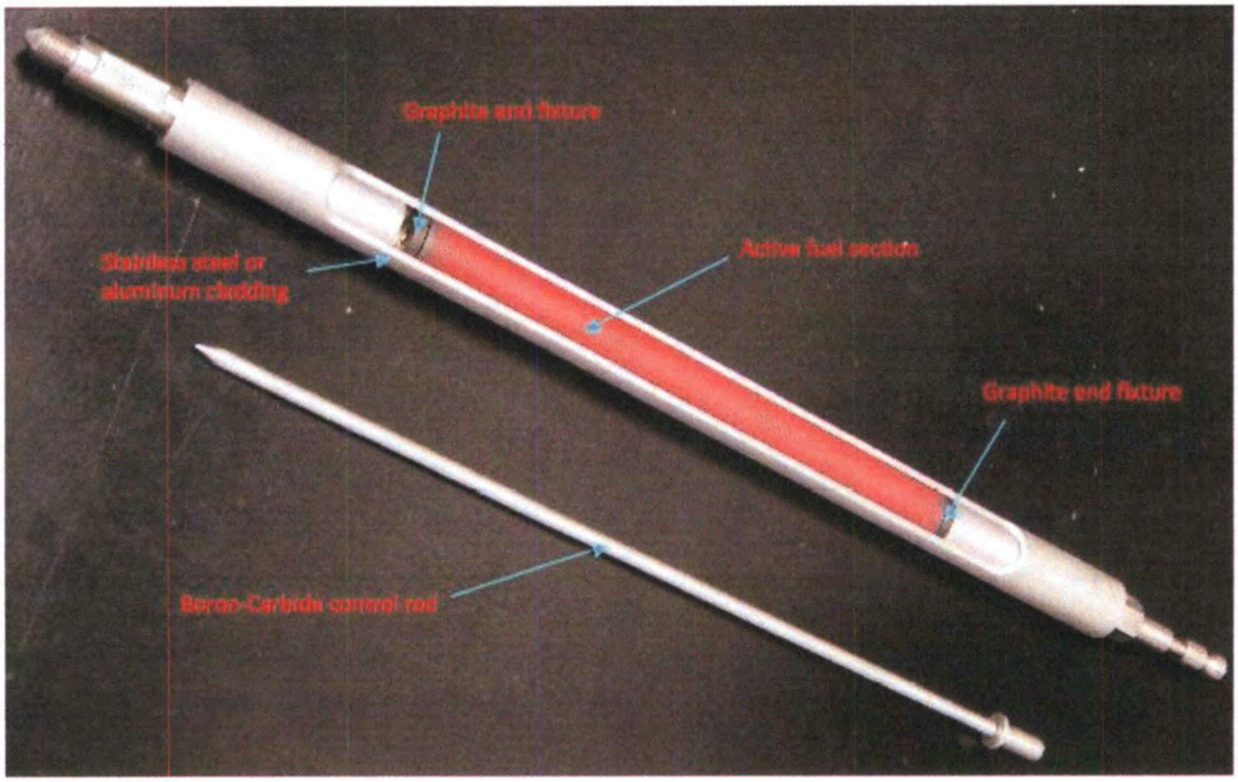


Figure 4.2-2 Active part of the fuel element

Table 4.2-2 Estimated average fuel element burn-up for UUTR operating at 90kW

Period	1975 - 1998	1998 - 2010
Burning history	191.643MWh	14.221MWh
Burned U-235	239.554 g	17.776 g
# of fuel elements in the active core	78	78
Average burnup	3.07 g/element	0.228 g/element

4.2.1.2 Mechanical Forces and Stresses

At 530 °C (803.15 °K), the equilibrium hydrogen pressure for the fuel is less than 689 Pa (0.1 psi), and distortion of the fuel by phase transformations will not occur. Thus, the cladding integrity is assured since the stress at 689 Pa on the 0.030 inch-thick aluminum cladding is only 16.88 kPa (2.45 psi) and 25.3 kPa (3.67 psi) for the 0.020 inch stainless steel cladding. The stress is calculated as follows:

$$S = P\frac{r}{t}$$

where r (radius) = 1.867 cm, t =cladding thickness (cm), P =internal pressure (Pa).

Example: For r=1.867cm, t=0.03in (0.0762cm), and P=689Pa, the stress can be calculated as follows:

$$S = P\frac{r}{t} = 689Pa \times \frac{1.867 \times 10^{-2} m}{0.0762 \times 10^{-2} m} = 16881Pa = 16.88kPa$$

4.2.1.3 Corrosion and Erosion of Cladding

The likelihood of a major fuel element cladding failure is considered small in TRIGA reactors because the elements must meet rigid quality control standards. In addition the pool water quality must be carefully controlled and much care is taken in handling fuel. Such cladding failures are, however, possible and the consequences of such a failure are analyzed and presented in **UUTR SAR 13**. The release of radioactivity by corrosion and leaching by the pool water has been measured previously showing that the potential release is easily controlled by isolating the leaking element in a container provided for that purpose.

The UUTR fuel elements are inspected every two years for swelling, discoloring, and streaks using underwater cameras. The calculated (based on PARET-ANL simulations) maximum fuel temperature of ~121.7 °C (394.85 °K) for 100 kW UUTR would not cause any swelling or radiation damage on the UUTR fuel elements (**UUTR SAR 4.6**).

4.2.1.4 Thermal Changes and Temperature Gradients

The PARET-ANL code is used to calculate the fuel average centerline, fuel average surface, and maximum fuel centerline temperature for 90kW (and 100kW) UUTR. The resulting values are shown in **Table 4.2-3.** The core inlet pool water temperature for the PARET-ANL calculations is selected as the bulk average temperature of the water in the tank before reactor start-up (measured to be 20.6 °C (293.75 °K). The maximum fuel centerline temperature in the hottest fuel element (Ring B) as calculated is 129.67 °C (402.82 °K) if UUTR would operate at 100 kW, with an average fuel centerline temperature of 74.06 °C (347.21 °K) and average fuel surface temperature of 58.9 °C (332.05 °K).

The **UUTR TS 2.1** specifies the safety limits for the maximum temperature of the reactor fuel:

1. The temperature in a stainless-steel clad, high hydride fuel element shall not exceed 1,000 °C (1,273.15 °K) under any conditions of operation;
2. The temperature in an aluminum clad low hydride fuel element shall not exceed 500°C (773.15 °K) under any conditions of operation.

Table 4.2-3 Thermal changes and temperature gradients for UUTR [Calculated values are obtained using PARET-ANL code]

Parameter		**UUTR**	
	Calculated 100kW	**Calculated 90kW**	**Measured 90kW**
Core inlet pool water temperature °C (°K)	20.6 (293.75)	20.6 (293.75)	20.6 (293.75)
T_s, average cladding surface temperature, °C (°K)	57.0 (330.15)	52.1 (325.25)	---
T_c, interior cladding temperature, °C (°K)	---	---	---
T_{fs}, average fuel surface temperature, °C (°K)	58.9 (332.05)	53.7 (326.05)	---
T_{fc}, max. fuel temperature, °C (°K)	129.67 (402.82)	121.70 (394.85)	---
Core exit pool water temperature, °C (°K)	21.95 (295.10)	21.75 (294.90)	23.04 (296.19)
Peak heat flux (kW/m^2)	49.057	49.054	---

The **UUTR TS 2.2** specifies the settings that prevent the safety limit from being reached as follows:

1. For a core composed entirely of stainless steel cladding fuel, high hydride fuel elements or a core composed of stainless steel cladding fuel, high hydride fuel elements with low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated:

Location of Instrumented Fuel Element	**Limiting Safety System Setting for SS Cladding**
B-hexagonal ring	800°C (1,073.15 °K)
C-hexagonal ring	755 °C (1,028.15 °K)
D-hexagonal ring	680 °C (953.15 °K)
E-hexagonal ring	580 °C (853.15 °K)

2. Or a core with low hydride fuel elements installed in other than the F or G hexagonal ring, limiting safety system settings apply according to the location of the instrumented fuel element:

Location of Instrumented Fuel Element	Limiting Safety System Setting for Al Cladding
B-hexagonal ring	460 °C (733.15 °K)
C-hexagonal ring	435 °C (708.15 °K)
D-hexagonal ring	390 °C (663.15 °K)
E-hexagonal ring	340 °C (613.15 °K)

4.2.1.5 Internal Pressure from Fission Products and the Production of Fission Gas

A number of experiments have been performed to determine the extent to which fission products are retained by UZrH (TRIGA) fuel; experiments on fuel with a uranium density of 0.5 g/cm^3 (8.5 wt-% U) were conducted over a period of 11 years and under a variety of conditions ["Safety Reports Series No.53, Derivation of the Source Term and Analysis of the Radiological Consequences of Research Reactor Accidents, IAEA, Vienna 2008]. Results prove that only a small fraction of the fission products are released, even in completely uncladded UZrH fuel. The release fraction varies from 1.5 x 10^{-5} for an irradiation temperature of 350 °C (623.15 °K) to ~10^{-2} at 800 °C (1,073.15 °K).

The experiments show that there are two mechanisms involved in the release of fission products from UZrH fuel, each of which predominates over a different temperature range. The first mechanism is that of fission fragment recoil into the gap between the fuel and cladding. This effect predominates in a fuel at temperatures up to ~400 °C (673.15 °K); the recoil release rate is dependent on the fuel surface-to-volume ratio but is independent of fuel temperature. Above ~400 °C, the controlling mechanism for fission product release from UZrH fuel is a diffusion-like process, and the amount released is dependent on the fuel temperature, the fuel surface-to-volume ratio, the time of irradiation, and the isotope half-life. The results of the UZrH experiments, and measurements by others of fission product release from Space Nuclear Auxiliary Power Program (SNAP) fuel, have been compared and found to be in good agreement.

The fractional release, ϕ, of fission product gases into the gap between fuel and cladding from a full-size standard UZrH fuel element is given by:

$$\phi = 1.5 \times 10^{-5} + 3.6 \times 10^{3} \exp\left(\frac{-1.34 \times 10^{4}}{T + 273}\right)$$

where T = fuel temperature, 0 – 1,600 °C (273.15-1,873.15 °K).

This relationship has also been found to apply to LEU TRIGA fuels ["Fission Product Releases from TRIGA-LEU Reactor Fuels," GA-A 16287, November 1980, Baldwin, Foushee, and Greenwood]. The first term is a constant for low temperature release; the second term is related to the high-temperature condition. The above equation applies to a fuel element, which has been irradiated for a time sufficiently long that all fission product activity is at equilibrium. Actual measured values of fractional releases fall well below that calculated by the above equation. Therefore, for safety considerations, this equation gives conservative values for the high temperature release from UZrH fuel. The results of the studies in the TRIGA reactor at GA on fission product release from fuel elements with high uranium loadings (up to 3.7 g U/cm^3, 45 wt-% U) agree well with data from older similar experiments with lower U loadings. As was the case with the lower U loadings, the release was determined to be predominantly recoil controlled at temperatures < 400 °C (673.15 °K) and controlled by a migration or diffusion-like process above 400°C. Low-temperature release appears to be independent of uranium loadings, but the high-temperature release seems to decrease with increasing weight fractions of uranium.

The correlation used to calculate the release of fission products from TRIGA fuel remains applicable for the high uranium loaded (TRIGA LEU) fuels as well as the 8.5 wt-% UZrH fuel for which it was originally derived. This correlation predicts higher fission product releases than measurements would indicate up to 1,100 °C (1,373.15 °K). At normal TRIGA operating temperatures (<750 °C (1,023.15 °K)) there is a safety factor of approximately four between predicted and experimentally deduced values. Thus, the UZrH fuel has been shown to retain a large fraction of the gaseous fission products. Assuming that all of the noble gases escape from the fuel matrix, 10% burn-up of the fuel loading in an element will create approximately 0.003 moles of noble gas atoms. Assuming that the noble gas atoms leave the fuel matrix and accumulate in 10 cm^3 of effective plenum area, the pressure and stress created on the cladding can be determined.

Table 4.2-4 shows the stress created from the fission gas pressure for a range of temperatures; about 3.3 cm^3 of void is built into the element as the clad-fuel gap. Other voids and interstices in the fuel probably account for much more than an effective 10 cm^3 of void. The use of 10 cm^3 as the characteristic volume is realistic since a large fraction of the noble gases will not escape from the matrix. From **Table 4.2-4** it is clear that the internal pressure and the resulting stresses are well below the yield strengths of the fuel cladding.

Typical stress vs. temperature data [Nuclear Engineering Handbook, H. E. Thorington, Editor, 1958] for aluminum and aluminum alloys show that the creep rate under these conditions is quite tolerable, i.e. less than 10^{-3} % per hour. Thus, 460 °C (733.15 °K) is an acceptable limit for cladding temperature with a fuel element burn-up as high as 10%. The maximum cladding temperature of the UUTR core operating at 100 kW is far below this limit, therefore there is no limit imposed on the fuel burnup.

4.2.1.6 Impact of Irradiation Effects Including Maximum Fission Densities and Fission Rates

Most of the irradiation experience to date has been with the uranium-zirconium hydride fuels used in the SNAP (containing about 10 wt% uranium) and TRIGA reactors. The presence of uranium influences the radiation effects because of the damage resulting from fission recoils and fission gases. Some significant conclusions may be drawn from these experiments. The uranium is present as a fine dispersal (about 1 μm in diameter) in the UZrH fuels, and hence the recoil damage is limited to small regions within the short (app. 10 μm) range of the fission recoils. The UZrH fuel exhibits high growth rate during initial operation, the so-called "offset" growth period, which has been ascribed to the vacancy-condensation type of growth phenomenon over the temperature range where voids are stable. The swelling of the UZrH fuels at high burnups is governed by three basic mechanisms:

Table 4.2-4 Pressure on the cladding due to fission product gases

Temperature, °C (°K)	Pressure		Tangential Stress	
	psi	kPa	psi	kPa
20 (293.15)	19.2	133	273	1885
100 (373.15)	24.5	169	348	2400
200 (473.15)	31.1	214	442	3044
300 (573.15)	37.6	259	534	3687
400 (673.15)	44.2	304	628	4330
500 (773.15)	50.8	350	722	4973
600 (873.15)	57.3	395	815	5617

1. The accommodation of solid fission products resulting from fission ofU-235. This growth is approximately 3% ΔV/V per metal atom % burnup. This mechanism is relatively temperature insensitive.
2. The agglomeration of fission gases at elevated temperatures (above 704.44 °C (977.59 °K)). This takes place by diffusion of the xenon and krypton to form gas bubbles.
3. A saturable cavity nucleation phenomenon, which results from the nucleation and growth of irradiation-formed vacancies into voids over a certain range of temperatures where the voids are stable. The saturation growth by this mechanism was termed offset swelling. It was deduced from the rapid decrease in fuel-to-cladding AT experienced during the early part of the irradiation. The saturation was reached in approximately 1,500 hr.

The burnup tests performed by GA have shown that TRIGA fuels may successfully be used without significant fuel degradation to burnups in excess of 50% of the contained U-235.

Thermal cycling tests have verified that the fuel matrix stability with respect to swelling or elongation, according to Sinmad ["The $UZrH_x$ Alloy: its properties and use in TRIGA Fuel," GA-4314, E-117-833, GA Technology, Inc., San Diego, CA, 1980.], showed no important change in length or diameter of the test sample in the temperature range 500 °C (773.15 °K) to 725 °C (998.15 °K). Small phase transition did occur at temperature of 653° C (926.15 °K). The test shows that for the temperature less than 200 °C (473.15 °K), there is no phase change or other transition to produce elongation or swelling in the fuel matrix. Sinmad also showed that hydrogen migration and accumulation of fission products in the fuel matrix is not important for the fuel temperatures below 500 °C (773.15 °K). A temperature of 500 °C (773.15 °K) is well above the fuel temperatures characteristics of a TRIGA reactor.

The specific characteristics that make TRIGA type fuels uniquely suited for use in extremely safe research-type reactors are:

- ZrH_x is single phase up to 649 °C (922.04 °K) [delta phase region];
- low hydrogen equilibrium disassociation pressure at normal fuel temperatures;
- high hydrogen retention;
- high heat capacity;
- low thermal expansion coefficient;
- relatively low reactivity in water;
- high fission product retention;
- very large negative prompt temperature coefficient of reactivity;
- high burnup possible by addition of burnable poison; and
- high loading of uranium possible with insignificant change in fuel material properties.

4.2.1.7 Dissociation Pressures

The hydrogen dissociation pressures of hydrides have been shown to be comparable in the alloys containing up to 75 wt-% U. The concentration of hydrogen is generally reported in terms of either weight percent or atoms of H/cm^3 of fuel (N_H). In the delta phase region, the dissociation pressure equilibria of the zirconium-hydrogen binary mixture may be expressed in terms of composition and temperature by the following relation

$$\log(P) = K_1 + \frac{K_2 \times 10^3}{T}$$

where

$K_1 = -3.8415 + 38.6433X - 34.2639X^2 + 9.2821X^3$
$K_2 = -31.2982 + 23.5741X - 6.0280X^2$
P = pressure, (atm)
T = temperature, (K)
X = hydrogen-to zirconium atom ratio.

The higher-hydride compositions (H/Zr > 1.5) are single phase (delta or epsilon) and are not subject to thermal phase separation on thermal cycling. For a composition of about $ZrH_{1.6}$, the equilibrium hydrogen dissociation pressure is 1 atm at about 760 °C (1,033.15 °K). The absence of a second phase in the higher hydrides eliminates the problem of large volume changes associated with a phase transformation at approximately 540 °C (813.15 °K) in the lower hydride compositions. Similarly, the absence of significant thermal diffusion of hydrogen in the higher hydrides precludes concomitant volume change and cracking. The clad material of stainless steel or nickel alloys provides a satisfactory diffusion barrier to hydrogen at long-term (several years) sustained cladding temperatures below about 300 °C (573.15 °K) ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, and "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002].

4.2.1.8 Hydrogen Migration

Under non-isothermal conditions, hydrogen migrates to lower-temperature regions from higher-temperature regions. The equilibrium dissociation pressure obtained when the redistribution is complete is lower than the dissociation pressure before redistribution. The dimensional changes of rods resulting from hydrogen migration are of minor importance in the delta and epsilon phases ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, and "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002].

Another important parameter that has to be considered is the hydrogen pressure inside the cladding. As shown in **Figure 4.2-3,** for UUTR operating temperature range for 100kW the hydrogen pressure is negligible for all UUTR fuel types.

4.2.1.9 Hydrogen Retention

The rates of hydrogen loss through 250-μm-thick stainless steel cladding are low at cladding temperatures characteristic of TRIGA fuel elements. A 1% loss of hydrogen per year occurs at about 500 °C (773.15 °K) clad temperature ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, and "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002].

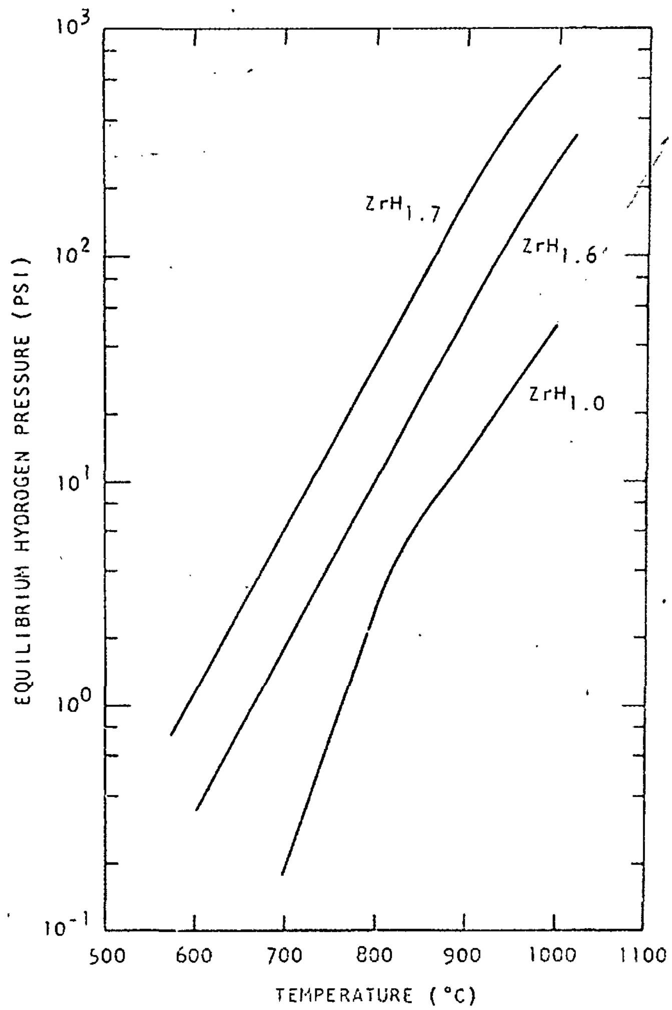


Figure 4.2-3 Equilibrium Hydrogen pressures over ZrH_x for various temperatures

4.2.1.10 Density

The density of ZrH decreases with an increase in the hydrogen content. The density change is quite high up to the delta phase (H/Zr = 1.5) and then changes little with further increases in hydrogen. The bulk density of massively hydrided zirconium is reported to be about 2% lower than the results from x-ray defraction analysis. For TRIGA fuel with a hydrogen-to-zirconium atom ratio of 1.6, the following relationships for the uranium density, $\rho_{u(A)}$ and weight fraction, ${}^{w}U$ in the $UZrH_{1.6}$ alloy apply:

$$\rho_{\mu(A)} = \frac{{}^{W}U}{0.177 - 0.125\,{}^{W}U}$$

$$^{w}U = \frac{0.177\rho_{\mu(A)}}{1+0.125\rho_{\mu(A)}}$$

The relationship between the uranium density and the volume fraction of uranium in the alloy is given by:

$$\rho_{\mu(A)} = 19.07V_f^{U(A)}$$

where $V_f^{U(A)}$ = volume fraction of uranium in the $UZrH_{1.6}$ alloy ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, and "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002].

4.2.1.11 Thermal Conductivity

Thermal conductivity measurements have been made over a range of temperatures at similar TRIGA facilities. Data from thermal diffusivity measurements taken by General Atomics along with the best available data for density and specific heat showed that the thermal conductivity is both independent of temperature and uranium content (NUREG-1282 op. cit.). A problem in carrying out these measurements by conventional methods is the disturbing effect of hydrogen migration under the thermal gradients imposed on the specimens during the experiments. This has been minimized at General Atomics by using a short-pulse heating technique to determine the thermal diffusivity and hence to permit calculation of the thermal conductivity. From the measurements at General Atomics of thermal diffusivity coupled with the data on density and specific heat, the thermal conductivity of uranium-zirconium hydride with an H/Zr ratio of 1.6 is 0.042 ± 0.002 cal/sec-cm-°. The UUTR reactor has H/Zr ratio of 1.6 for stainless steel and 1.0 for aluminum element. The thermal diffusivity is insensitive both to the weight fraction of uranium and to the temperature.

4.2.1.12 Heat Capacity

The heat content of zirconium hydride TRIGA fuel is a function of temperature and composition ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002, and "The UZrH, Alloy: Its Properties and Use in TRIGA Fuel," GA Report 4314, February 1980, Mt. Simnad]. The volumetric specific heat of 8.5 wt-% $UZrH_{1.6}$ is calculated by:

$$C_p = 2.04 + 4.17 \times 10^{-3} T \text{ [W-sec/cm}^3 \text{ °C]}$$

4.2.1.13 Chemical Reactivity

Zirconium hydride has a relatively low reactivity in water, steam, and air at temperatures up to about 600 °C (873.15 °K). Massive zirconium hydride has been heated in air for extended periods of time at temperatures up to 600 °C with negligible loss of hydrogen ["The UZrH, Alloy: Its Properties and Use in TRIGA Fuel," GA Report 4314, February 1980, Mt. Simnad]; an oxide film forms which inhibits the loss of hydrogen. The hydride fuel has excellent corrosion resistance in water. Bare fuel specimens have been subjected to a pressurized water environment at 298.89 °C (572.04 °K) and 1230 psi during a 400 hr period in an autoclave. The average corrosion rate was 350 mg/cm^2-month weight gain, accompanied by a conversion of the surface layer of the hydride to an adherent oxide film. The maximum extent of corrosion penetration after 400 hr was less than 2 mils.

In the early phases of development of the TRIGA fuel, water-quench tests were carried out from elevated temperatures. Fuel elements (1 -in. diameter) were heated to 800 °C (1,073.15 °K) and end-quenched to test for thermal shock and corrosion resistance. No deleterious effects were observed. Also, a 6-mm diameter fuel element was heated electrically to about 800 °C and a rapid stream of water was sprayed on it; no significant reaction was observed. Small and large samples were heated to 900 °C (1,173.15 °K) and quenched in water; the only effect observed was a slight surface discoloration. Finely divided UZrH powder was heated to 300 °C (573.15 °K) and quenched to 80 °C (353.15 °K) in water; no reaction was observed. Later, these tests were extended to temperatures as high as 1,200 °C (1,473.15 °K), in which tapered fuel elements were dropped into tapered aluminum cans in water. Although the samples cracked and lost hydrogen, no safety problem arose in these tests. Recently, the low-enriched TRIGA fuels have been subjected to water-quench safety tests at GA. Quench tests were performed on 20%-enriched TRIGA fuel samples to simulate cladding rupture and water ingress into the TRIGA reactor fuel elements during operation. These results indicate satisfactory behavior of TRIGA fuel for temperatures to at least 1,200 °C. Under conditions where the cladding temperature can approach the fuel temperature for several minutes (which may allow formation of eutectics with the clad), the results indicate satisfactory behavior to about 1,050 °C (1,323.15 °K). This is still about 50 °C (323.15 °K) to 100 °C (373.15 °K) higher than the temperature at which internal hydrogen pressure is expected to rupture the clad, should the clad temperature approach that of the fuel. It should be pointed out that thermocouples have performed well in instrumented TRIGA fuel elements at temperatures up to 650 °C (923.15 °K) in long-term steady-state operations, and up to 1,150 °C (1,423.15 °K) in very short time pulse tests ["Safety Analysis Report for the Oregon State University TRIGA Reactor," Oregon State University, July 2004, and "Safety Analysis Report for the Washington State University Modified TRIGA Nuclear Reactor," Washington State University, June 2002].

4.2.1.14 Prompt Negative Temperature Coefficient

The basic parameter, which provides the greatest degree of safety in the operation of a TRIGA reactor system, is the prompt negative temperature coefficient. This temperature coefficient allows great freedom in steady-state operation, since the effect of accidental reactivity changes occurring from experimental devices in the core is minimized.

The prompt negative temperature coefficient for TRIGA fuels is based on the neutron spectrum-hardening characteristic that occurs in a zirconium hydride fuel. Heating of the fuel-moderator elements causes the spectrum hardening. The rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, the thermal neutron spectrum in the fuel element shifts to a higher average energy (the spectrum is hardened), and the mean free path for neutrons in the element is increased appreciably. For a standard TRIGA element, the average chord length R which is defined by $R=4V/S$ where V is the volume of the fuel element and S is the surface area of the element, is comparable to a mean free path, and the probability of escape from the element before being captured is significantly increased as the fuel temperature is raised. In the water, the neutrons are rapidly thermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel in a standard TRIGA element thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in which the ratio of absorptions in the fuel to total cell absorptions decreases as fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity. More than 50% of the temperature coefficient for a standard TRIGA core comes from the temperature-dependent disadvantage factor, or cell effect, and 20% of each come from Doppler broadening of the U-238 resonances and temperature-dependent leakage from the core. These effects produce temperature coefficient of ~10^{-4} $\Delta k/k$/°K, which is essentially constant in respect to temperature.

The prompt-temperature coefficient for the UUTR operating at nominal power of 100kW, α_F, was calculated by varying the fuel temperature while keeping other core parameters unchanged. The MCNP5 model was used to simulate the reactor with all rods out at 293, 500, 600, 800, and 1,200 °K. The effective delayed neutron fraction of 0.00768 is used to convert the multiplication factor to reactivity. The MCNP5 model and calculated values are described in **UUTR SAR 4.5**.

4.2.2 Control Rods

The UUTR uses three motor-driven control rods (one regulating, one shim, and one safety). The control rods are positioned by standard TRIGA electrically powered rack and pinion drives. The position of each control rod is displayed on the console as a percentage of rod withdrawn (shown in **Fig. 4.2-4**). The control rods are held in place by electromagnets. When a scram is initiated the current is cut and the control rods drop by gravity into the core, shutting the reactor down.

The UUTR utilizes boron carbide control rods that are characteristic of most of the TRIGA reactors. The rods are enclosed into the aluminum tubes approximately 43 inches long and are 0.875, 0.875, and 0.25 inches in diameter (safety, shim, and regulator rods respectively) with a powder boron carbide neutron absorber filling insight of the rods. One rod is designated as a regulating rod and is used for fine control during the UUTR operation. The control rods pass through normal fuel positions in the UUTR core on the top and the bottom of the grid plates. Guide tubes ensure that the control rods remain in a proper position during their activated use. The safety, shim, and regulating control rods are located at D-7, D-13, and D-1 (**Fig. 1.3-4**). **Figure 4.2-4** shows also the geometrical dimensions of the three control rods. Each drive consists of a stepping motor, a magnet rod-coupler, a rack and pinion gear system, and a ten-turn potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw-tube which supports an electromagnet. The magnet engages a chrome-plated armature attached above the water level to the end of a connecting rod that fits into the connecting tube. The connecting tube extends down to the control rod. The magnet, its draw-tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. The lower section of the barrel contains an air snubber to dampen the shock of the scrammed rod. In the snubber section, the control rods are decelerated through a length of 3 in. The control rod can be withdrawn from the reactor core when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized and the armature is released. The speed of the rods is adjustable and rods are normally set to insert or withdraw the control rods at a nominal rate of 0.49 cm/sec.

The control rods are designed to safely change the reactor power and/or shut the reactor down. According to the **UUTR TS 3.2.1**, the maximum rate of reactivity insertion by control rod motion shall not exceed \$0.30 per second. To determine the absolute maximum rate of reactivity insertion into the reference UUTR core, the maximum reactivity value determined for the safety rod of \$2.24 was used. Using the average safety rod rise time of 77 seconds (total travel distance is approximately 15 inches), the maximum reactivity insertion rate can be calculated from the following equation:

$$\frac{d\rho^{\$}}{dt} = \frac{d\rho^{\$}}{dx}\frac{dx}{dt} = \frac{\$0.00352}{unit}\frac{1000unit}{77seconds} = \frac{\$0.046}{second}$$

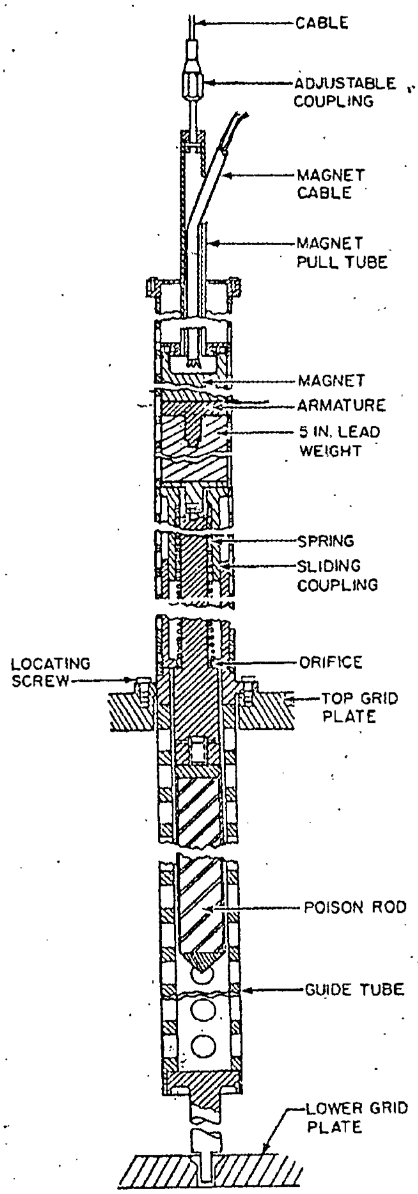
where

$\frac{d\rho^{\$}}{dt}$: maximum reactivity insertion rate

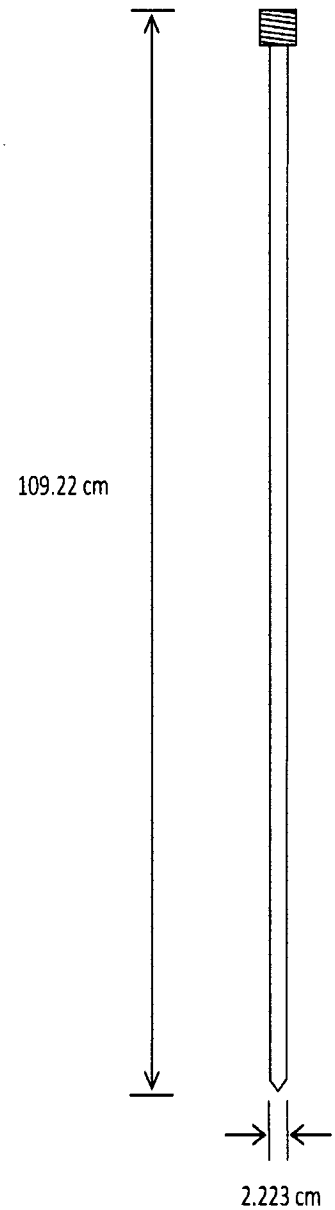
$d\rho^{\$}$: differential reactivity (it was assumed that the center of the control rod has the maximum rod worth)

dx: the control rod (15 inches) was divided by 1,000 equal units

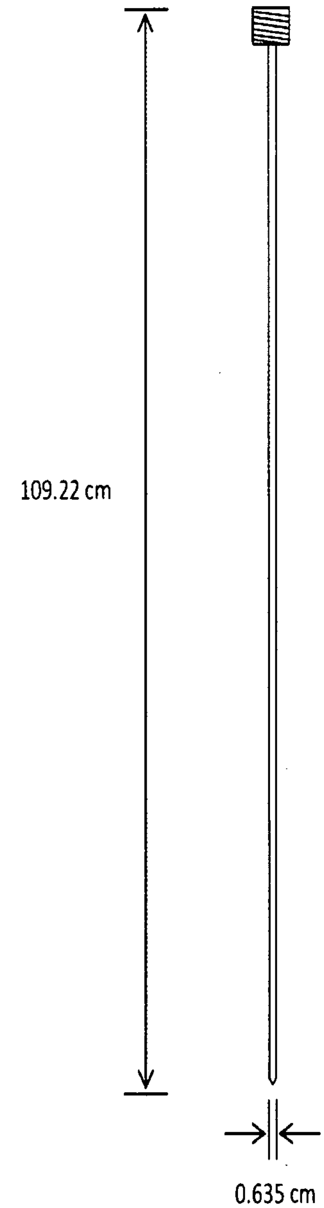
dt: rising time for the safety control rod (it was experimentally measured using a stop-watch)



Safety and Shim Control Rods



Regulating control rod



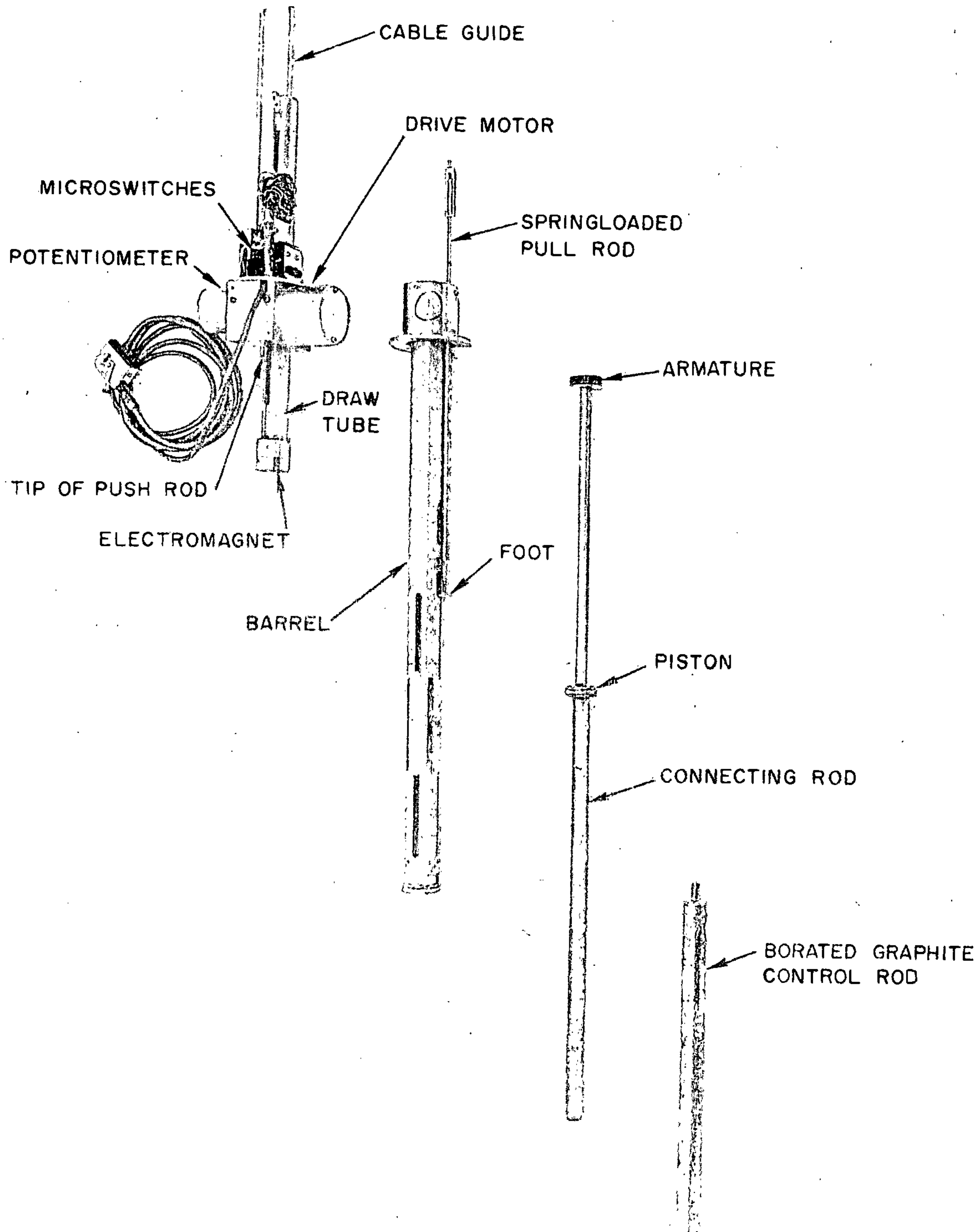


Figure 4.2-4 Schematics of the UUTR control rods and the rack-and-pinion control rod drive system

All three-control rods have the same rise time. Therefore, the obtained \$0.046/second is much lower than the Technical Specification requirement of \$0.30/sec [**UUTR TS 3.2.1**]. The shutdown margin must be greater than \$0.50 [**UUTR TS 3.1.2**] and the excess reactivity must be less than $1.20 [**UUTR TS 3.1.3**]. The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scram control rod reaches its fully inserted position shall not exceed 2 seconds. The SCRAM time specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds.

The 100kW UUTR's rod worth's for the safety, shim and regulating control rods are around $2.24, $1.55, and $0.287 with the total of about $4.08 (based on the measured values performed every six months; fluctuation of these measurements is ~10%). The shutdown margin for 100kW is approximately $1.018, and the excess reactivity is around $0.819. Current rod drop times are on the order of 1 second. The reactor shall not be operated unless the startup count rate interlock, and control element withdrawal interlocks are operable. The startup count rate interlock is described in **UUTR SAR 7.2**. The interlock exists to prevent the withdrawal of a control rod without some minimum count rate in the reactor. Control rod withdrawal interlocks prevent the withdrawal of more than one control rod at once.

The measured control rod worths, shutdown margins and excess reactivities for last eight years are shown in **Table 4.2-5**. These values are measured from semiannual control rod drops. Since 1975, the UUTR has been operated approximately 3,000 hours and at various power levels. Assuming that the reactor operated for 3,000 hours at 90 kW, the control rod worth change can be estimated by the following relationship:

$$R = n \cdot \sigma \cdot \phi$$

where

n = atomic density of Boron control rod (atoms/cm^3)
σ = thermal neutron absorption cross section (~759 barns)
ϕ = thermal neutron flux at 90 kW (~6 x 10^{11} neutrons/cm^2-sec)

The atomic density of boron is approximately 1.41 x 10^{23} atoms/cm^3. Thus the reaction rate for boron is R = 6.93 x 10^{20} atoms/cm^3 for 3,000 hours of reactor operation. This represents about 0.5% of the initial amount of boron atoms in the control rod and is therefore negligibly small. From **Table 4.2-5** it follows that the variations of the rod worth are less than 5% for safety and shim rods except for the measured value obtained on 04-02-2004. Based on **TS 3.2.1**, a SCRAM time less than 2 sec is required to ensure that the reactor will be promptly shut down when a scram signal is initiated. Operation experience and analysis have indicated that for the range of transients anticipated for a TRIGA rector, the specified scram time is adequate to ensure the safety of the reactor. The SCRAM times are checked semiannually. The start-up channel interlocks with the control rods ensure shutdown if neutron count rate is too low to provide meaningful startup information. When control rods are pull out the positions are filled in with the pool water.

Table 4.2-5 Control rod worth from the rod drop experiments

Date	Safety	Shim	Regulation	Shutdown margin	Excess reactivity
08-16-01	2.230	1.560	0.240	0.791	1.013
08-27-03	2.223	1.503	0.280	1.012	0.771
04-02-04	2.923	1.713	0.300	1.135	0.879
08-26-04	2.208	1.493	0.280	0.975	0.798
11-17-05	2.198	1.494	0.283	0.967	0.810
02-24-05	2.200	1.460	0.263	1.113	0.610
08-04-05	2.190	1.467	0.270	1.119	0.618
11-09-05	2.230	1.415	0.267	1.074	0.608
12-27-05	2.227	1.553	0.273	1.066	0.761
02-17-06	2.227	1.557	0.237	1.068	0.758
08-25-06	2.017	1.463	0.270	0.983	0.751
11-08-06	2.120	1.053	0.263	0.996	0.771
02-22-07	2.173	1.547	0.273	1.033	0.787
08-28-07	2.510	1.700	0.276	1.126	0.850
01-24-08	2.173	1.457	0.207	0.855	0.838
02-25-08	2.173	1.493	0.273	0.967	0.800
08-27-08	2.290	1.563	0.293	0.991	0.865
02-24-09	2.170	1.553	0.323	1.020	0.856
08-26-09	2.297	1.587	0.290	0.744	1.137
12-18-09	2.293	1.517	0.277	1.027	0.586
02-22-10	2.263	1.530	0.277	1.172	0.635
04-30-10	2.243	1.550	0.287	1.018	0.819

4.2.3 Neutron Moderator and Reflector

UUTR reflector consists of a single row of graphite or heavy water reflector elements. These are such that the equivalent volume fraction of the first 1.6 inches of reflector is approximately 30.5% water, 64.5% heavy water (concentrations of heavy water and regular water are 68% and 32% respectively), and 5% aluminum, or 95% graphite, and 5% aluminum (**Figures 4.2-5** and **4.2-6** for heavy water and graphite respectively). A TRIGA type reactor is a special type of an open pool reactor where the pool water represents a moderator and reflector. **Table 4.2-6** shows the specification of the heavy water and graphite reflector. Between each fuel element and heavy water/graphite reflector filled with light water, the water circulates through the reactor core by natural convection. **Figure 1.3-4** shows the locations of heavy water and graphite reflectors.

Water is kept from contact with the graphite by a welded aluminum cladding that encases the entire reflector. The graphite reflector elements are located near the thermal irradiator. Graphite elements have the same end structures at the top so that the fuel-handling tool can be used during fuel inspection. The heavy water elements have two different top-end fixtures: G8 through G18 has same top-end structure as a fuel element; screw is attached on the top as shown in **Fig. 1.3-4**.

Table 4.2-6 Specification of the heavy water and graphite reflectors in UUTR

	Heavy water reflector	**Graphite reflector**
Number of elements	12	12
Cladding material	Aluminum	Aluminum
Cladding thickness	0.076 cm	0.076 cm
Outside diameter	3.7465 cm	3.7465 cm
Overall length	72.2376 cm	72.2376 cm
Material	68% heavy water; 32% light water	100% graphite
Location (**Fig. 1.3-4**)	G-8,9,10,12,14,16,18; F-21,22,23,24,25	F-1,26,27,28,29,30; G-2,3,4,5,6,7

4.2.4 Neutron Startup Source

A plutonium-beryllium (Pu-Be) start up source is located in a special reflector element source holder placed in one of the fuel element positions on the west edge of the core (G-ring). **Figure 4.2-7** shows the position of Pu-Be source in the UUTR core and **Fig. 4.2-8** shows the startup neutron source geometry. The initial source strength was $8x10^6$ neutrons/sec; the overall length is 35.56 cm. The effective half-life of Pu-Be neutron source is approximately 142 years. The source generates neutrons of energy between 0 to 10 MeV. The UUTR has been operated for 35 years (since 1975), thus this source strength is changed and as of 2010 is about $6.74x10^6$ neutrons/sec. About 20% of the neutrons have the energy less than 1 MeV and broad intensity maxima at 4.0, 7.2 and 9.7 MeV (**Fig.4.2-9**). The absolute yield of Pu-Be neutron source for the neutron energy above 0.5 MeV is approximately 10^6 neutrons/sec.

Once the UUTR reactor is critical, the source is moved from its normal position at start up to a position at the side of the reactor pool. A source interlock indicator light visible to the operator indicates if the source is in or out of the core. Also the Startup Channel's fission chamber is located next to the Startup Source, and this channel's counts show the presence of the Startup Source. In addition, there is a video camera positioned above the reactor viewing the core with a monitor in the control room. Source position can be verified from this monitor.

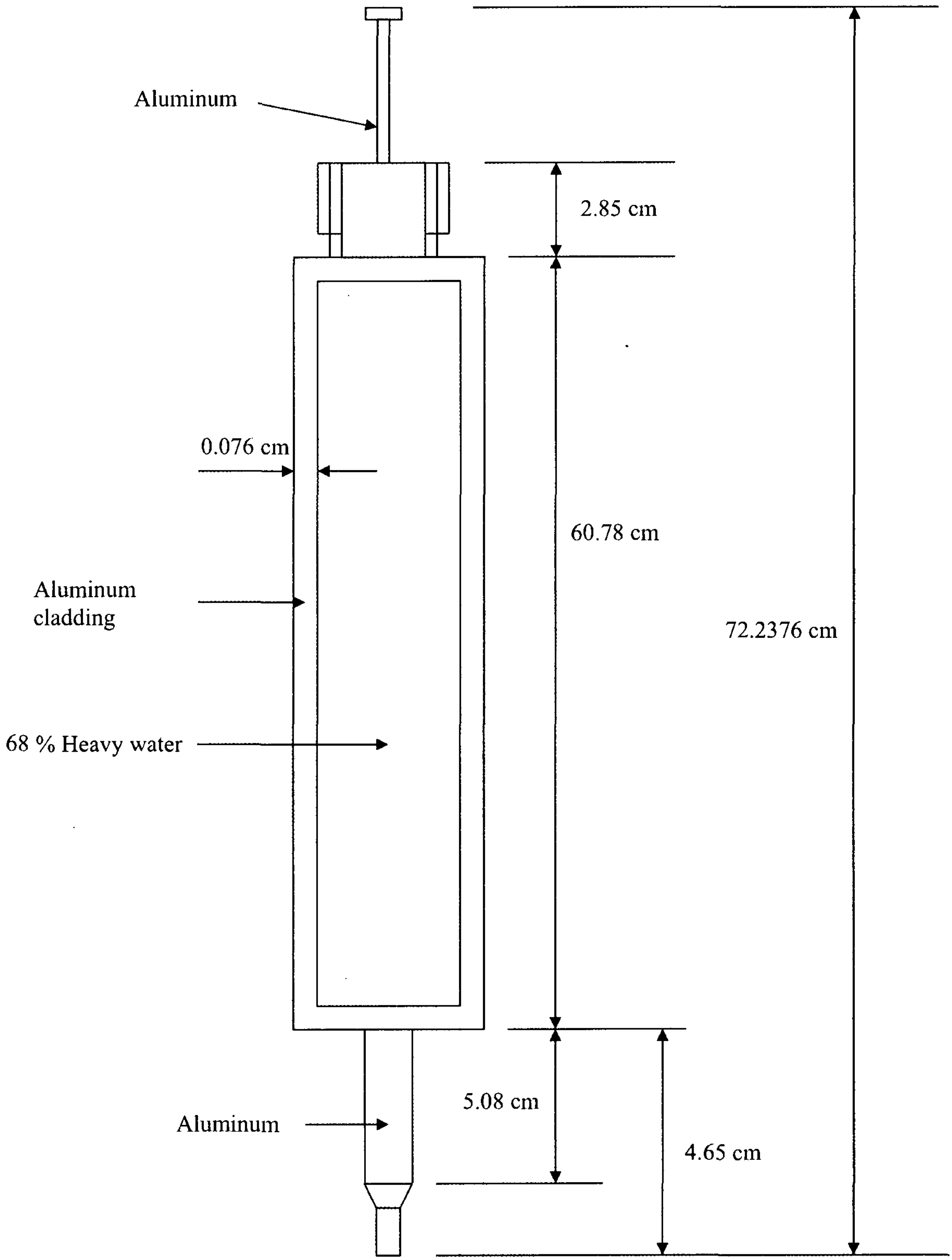


Figure 4.2-5 Schematics of the UUTR heavy water reflector

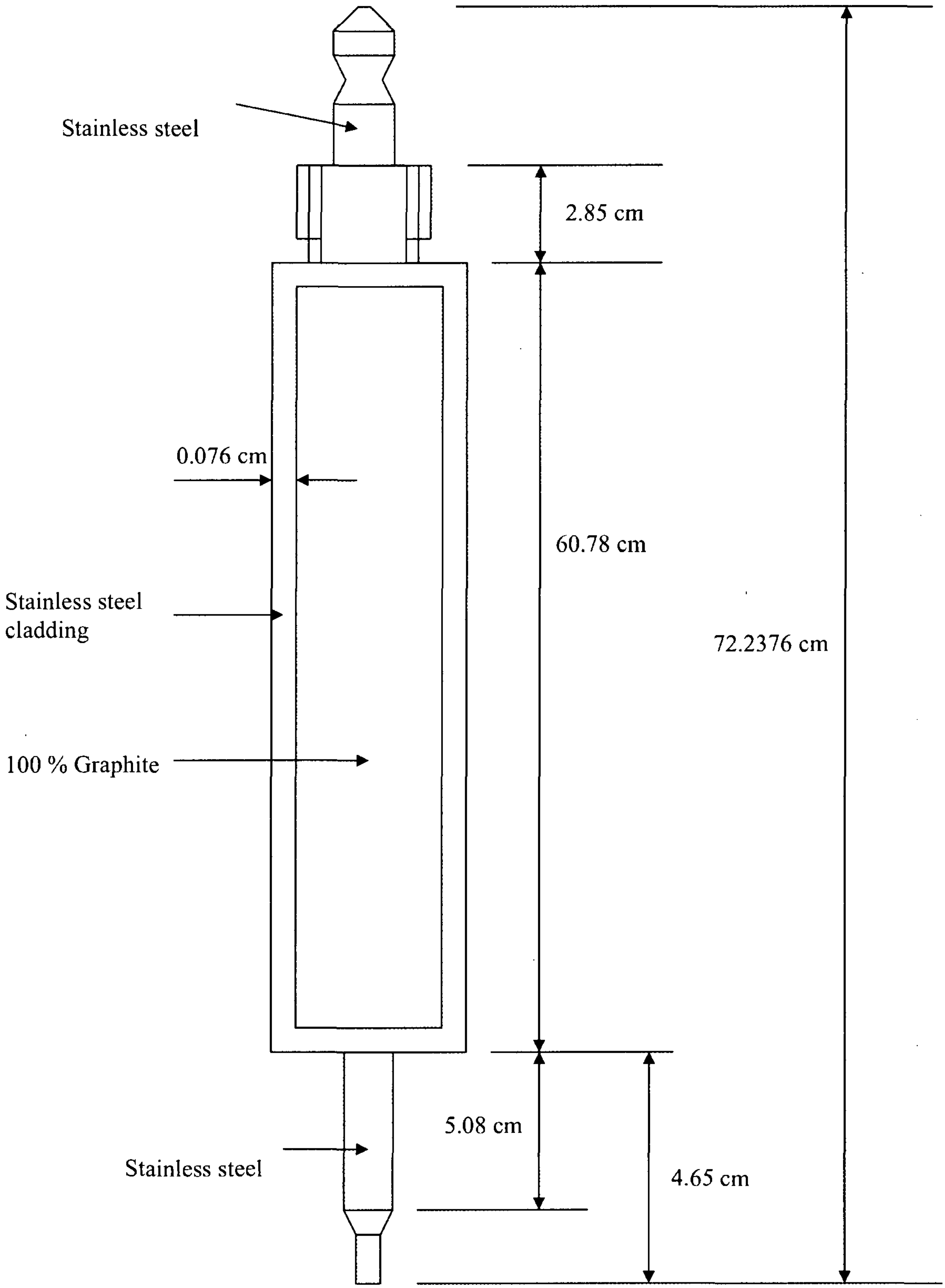


Figure 4.2-6 Schematics of the UUTR graphite reflector

A fission counter is used with a transistorized linear amplifier. The meaningful count rates range from approximately 10^{-3} W to about 2W (source level). These source levels are estimated to give count rates of about 5 counts/second and 10,000 counts/second, respectively. An interlock circuit is used to prevent rod withdrawal unless the source count level is above the required minimum value of at least 2 counts/second.

[redacted]

Figure 4.2-7 Plutonium-Beryllium neutron start-up source

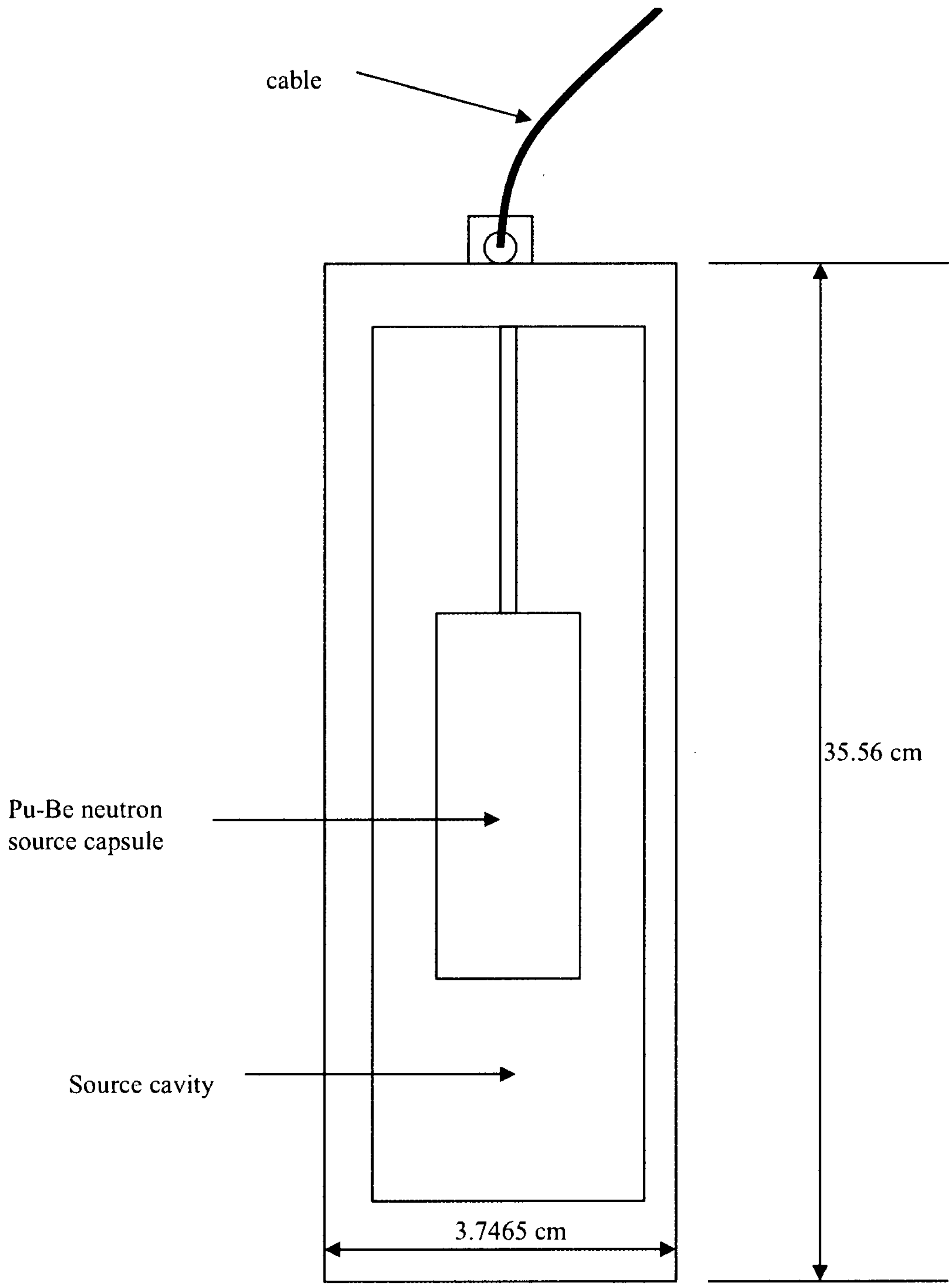


Figure 4.2-8 Pu-Be source geometry

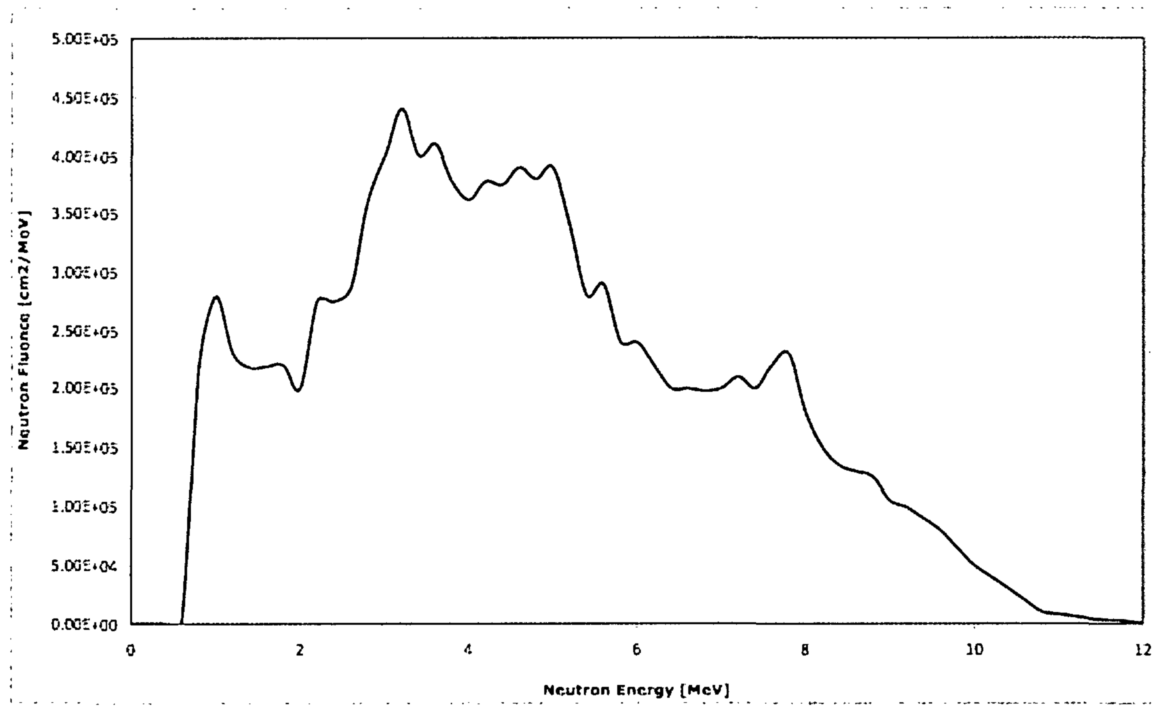


Figure 4.2-9 Plutonium-Beryllium source energy spectrum

4.2.5 Core Support Structure

The reactor tank is made of an inner liner, 5/16 inch-thick aluminum, with an outer diameter of 7 ft 8 inches, and an outer liner made of 3/16 inch-thick steel with a diameter of 12 ft. All outer surfaces of the steel tank that come into contact with the soil are painted to inhibit corrosion. The inner surface is painted with epoxy. The two-foot-space between the liners is filled with tamped sand to the concrete pad base. The bottom of the aluminum tank is a welded sheet of aluminum material the same as the side walls (6061-T6 aluminum). Both tanks are water tight, and welds on the inner aluminum tank have been made to meet ASME unfired pressure vessel standards. Since the bottom of the aluminum tank rests on a reinforced concrete pad that forms the bottom seal for the outer steel liner, the maximum stress at the bottom of the aluminum tank is the hoop stress at the bottom weld. Under a hydrostatic head of 24 ft of water, the hoop stress at the bottom of the inner tank is 24.1 MPa (3,500 psig), while the hoop stress on the outer steel tank, if subject to the same head, would be 68.9 MPa (10,000 psig), both well below the yield stresses of 246 MPa and 240 MPa, respectively, for these materials.

The reactor is located below ground level inside the shielded reactor tank. The general design of the tank and shield provides for shielding of the neutron flux from the earth by at least 2 ft of water in the tank and 2 ft of sand surrounding it. Approximately 20 ft of water and/or concrete and/or sand and earth above and to the side of the core provides the necessary shielding for personnel.

[redacted]

Figure 4.2-10 Side view of the UUTR structure

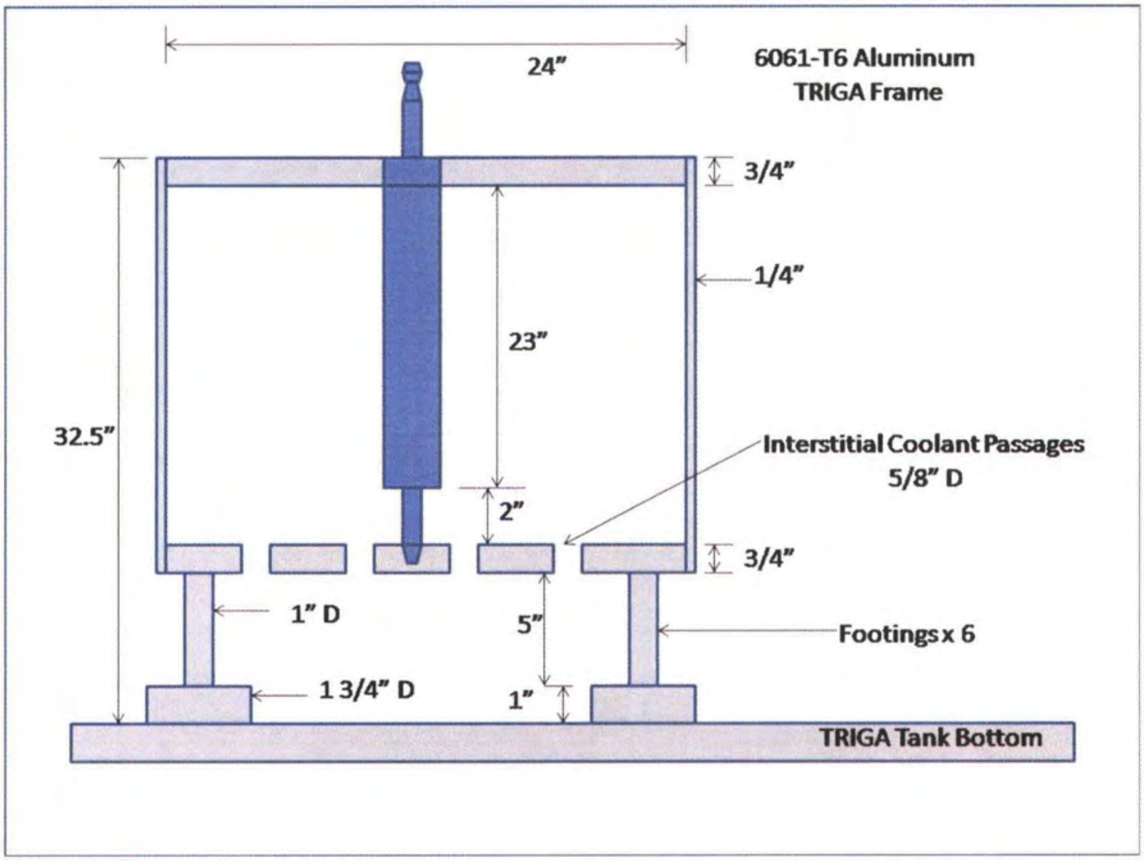


Figure 4.2-11 Grid plate and core support structure and its dimension

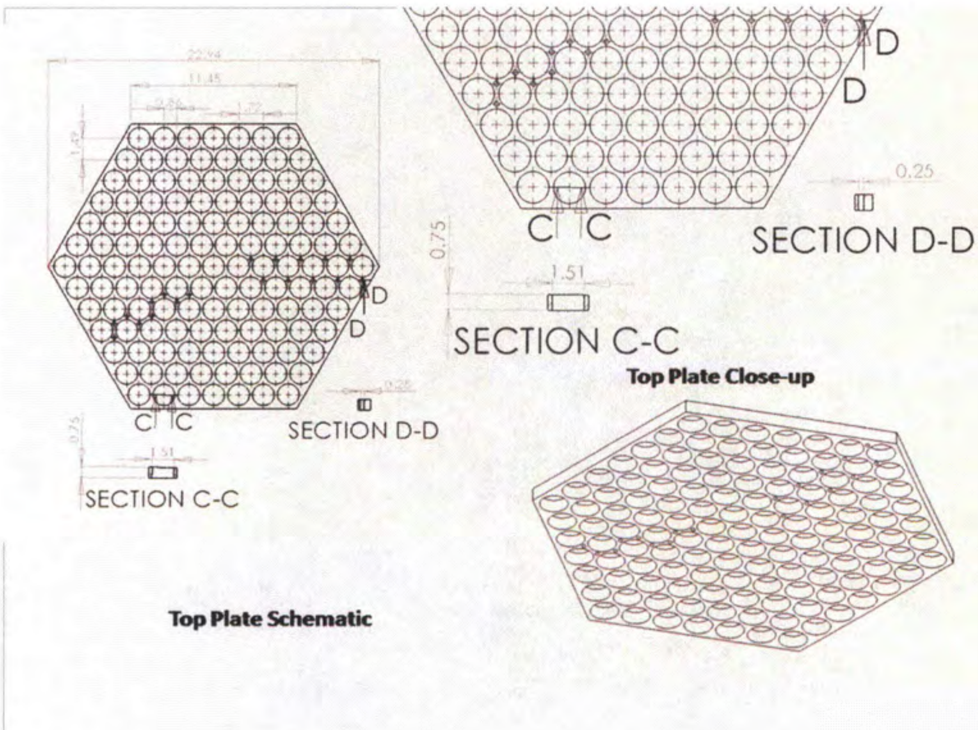


Figure 4.2-12 Top grid plate and its dimensions

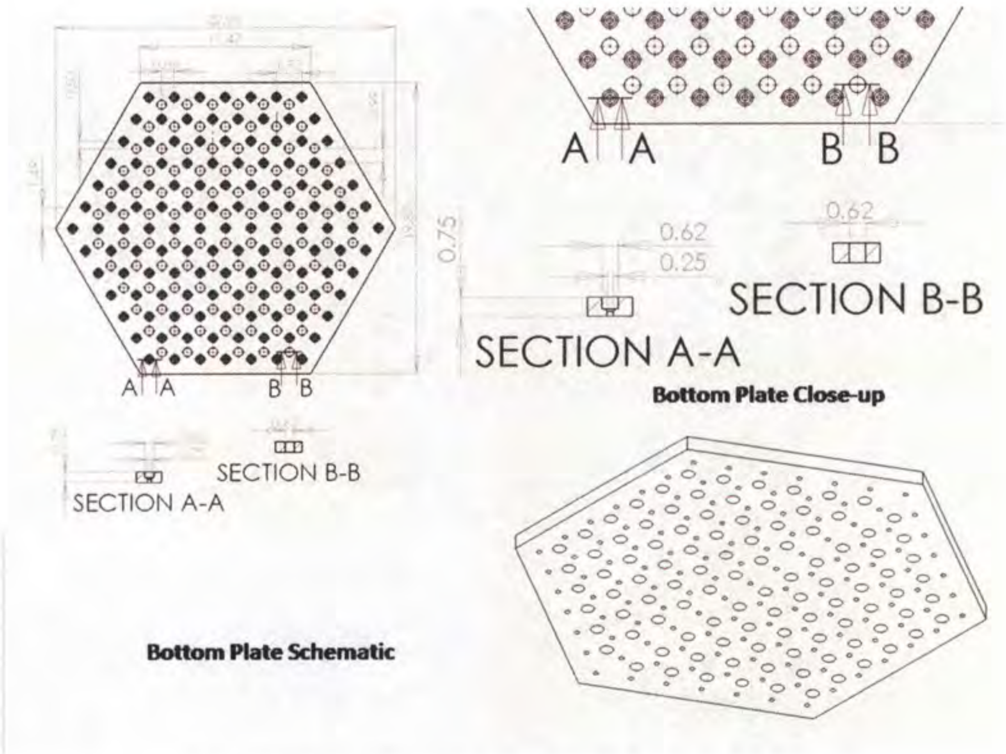


Figure 4.2-13 Bottom grid plate and its dimensions

[redacted]

Figure 4.2-14 UUTR core, Irradiators, and bottom section of the reactor tank

The reactor system contains provision for three diagonally directed beam tubes between the reactor core and the reactor room floor. Each tube is composed of two sections aligned along a common axis. The top tube section will be a 1 foot-diameter tube between the reactor floor and the wall of aluminum reactor tank. This tube will not penetrate the aluminum tank but will be sealed at the end where it butts against the tank. The upper beam tube will be adequately shielded with inner bags containing sand and capped at the reactor floor level with a 1/8-inch-thick steel cap for security, except when in use to extract a neutron beam for experimental purposes. **Figure 4.2-10** shows a side view of the UUTR reactor and **Fig. 4.2-11, 4.2-12** and **4.2-13** show the grid plates and its dimensions. **Figure 4.2-14** shows the UUTR core, irradiators, ionization chambers, and the bottom part of the reactor tank.

4.3 Reactor Tank or Pool

The inner tank of the UUTR is made of aluminum and is welded at joints. The welding on the tank are verified to be water proofed upon construction using X-ray testing, pressure testing, and soap-bubble leak testing. The water level is monitored by a sensor connected to an alarm. On the outside of the inner tank and inside of the outer tank, where sand is placed, two vertical columns are dug into the sand all the way down to the bottom of the tank. Cameras are used to monitor the sand at the bottom to verify no leakage takes place. The tank is placed on a 2 ft concrete block with stainless steel coating. The concrete is placed on several feet of clay. **Figure 4.3-1** shows the reactor, the pool and the three beam port positions. The beam ports do not penetrate into the inner aluminum tank. The beam ports are just placed between outer tank and inner tank with sand filled. The UUTR has three beam ports located at the outer tank. However, these beam ports has no interaction with the inner tank. The construction of these ports never finished and they are not being used. Therefore, there is no potential loss of water through these beam ports.

4.4 Biological Shield

The UUTR tank consists of an inner liner, with an outer diameter of 7 ft 8 inches, of 5/16-inch-thick aluminum and an outer liner of 3/16-inch-thick steel with a diameter of 12 ft. The 2 foot-space between the liners is filled with tamped sand to the concrete pad base. The bottom of the aluminum tank is a welded sheets of aluminum material the same as the side walls (6061-T6 aluminum). Both tanks are water tight, and welds on the inner aluminum tank have been made to meet ASME unfired pressure vessel standards. UUTR reactor pool contains 8,000 gallons of water that serves as a biological shield. Two feet-thick space between inner and outer reactor tanks filled with the sand provides additional radial biological shield. Since the bottom of the aluminum UUTR tank rests on a reinforced concrete pad that forms the bottom seal for the outer steel liner, the maximum stress at the bottom of the aluminum tank is the hoop stress at the bottom weld. Under a hydrostatic head of 24 feet of water, the hoop stress at the bottom of the inner tank is 24.1 MPa (3,500 psig), while the hoop stress on the outer steel tank, if subject to the same head, would be 68.9 MPa (10,000 psig), both well below the yield stresses of 246 MPa and 240 MPa, respectively, for these materials.

[redacted]

Figure 4.3-1 The UUTR tank

Three black circles on the wall represent three beam port positions. The inner tank is made of T-6061 aluminum and the beam ports are placed between the outer steel tank and the inner tank. The beam ports do not penetrate into the inner tank.

The reactor tank is built below the ground level and inside of the shielded reactor tank. The general design of the tank and shield are shown in **Fig. 4.2-10**. This design provides for biological shielding of the neutron and gamma flux from the earth by at least 2 feet of water in the tank and 2 feet of sand surrounding it in the radial direction. Approximately 20 feet of water and/or concrete and/or sand and earth above and to the side of the core provides necessary biological shielding for personnel.

In summary, the UUTR core is shielded:

- Radially, by 4ft water, 5/16 inch-thick aluminum tank, 2 ft sand, 3/16 inch-thick steel, 3 ft concrete and ground dirt.

- Axially, by 22 ft water above the core and 5/16 inch-thick aluminum, 2ft concrete and clay under the core structure.

4.5 Nuclear Design

The reactor design bases are established by the maximum operational capability for the fuel elements and configurations. The TRIGA reactor system has three major areas which are used to define the reactor design bases:

- fuel temperature,
- prompt temperature coefficient,
- control rod worths,
- thermal-hydraulics and heat transfer (pool water temperature), and
- reactor power.

The ultimate safety limit is based on fuel temperature, while the strongly negative temperature coefficient of reactivity contributes to the inherent safety of the TRIGA reactor. A limit on reactor power is set to ensure operation below the fuel temperature safety limit and pool water temperature limit. The following analysis and data are provided for UUTR operating at nominal power of 90kW (and licensed for 100kW).

4.5.1 Normal Operating Conditions

4.5.1.1 Critical Mass and Fuel Description

The 100kW UUTR consists of 78 fuel elements: 23 aluminum elements (8.91% burn-up), 36 stainless steel elements (0.61% burn-up), 17 old stainless steel elements (8.77% burn-up), and 2 stainless steel instrumental fuel elements (8.77% burn-up). [redacted] Aluminum element has H/Zr ratio of 1.0 and stainless steel element has H/Zr ratio of 1.6.Two instrumental fuel elements have same amount of U-235 as old stainless steel element. Graphite and heavy water elements are used as reflectors. Each fuel element is located B-ring through G-ring. A-ring is empty and used as a central irradiator.

4.5.1.2 Core Loading and Power per Fuel Element

Figure 1.3-4 shows the core loading for the operational 100kW UUTR. The power density was calculated using MCNP5 (described in **UUTR SAR 4.5.2**). The fuel element power distribution per fuel ring is obtained based on the MCNP5 model of the core loading. The highest power density, as expected, occurs in the "B" ring representing therefore the UUTR's hottest channel.

The effect of the graphite elements in the outer ring is to slightly increase the power density in the outer-fueled rings and to slightly reduce the power density in the inner-fueled rings. This again is expected as part of the reflector flux peaking. The average power per element per each UUTR ring is shown in **Table 4.5-1. Table 4.5-2** lists the maximum, minimum, and the average power per element, along with ratios of the maximum-to-minimum power and the maximum-to-average power per element.

4.5.1.3 Loss of Coolant Accident (LOCA) for 100kW UUTR Core

Under very rare conditions the water in the reactor tank can be drained out. Detailed analyses are presented in **UUTR SAR 13.2.3**. Once the coolant is drained, the core heat can be removed through natural air convection. Under the LOCA the fuel and cladding integrity will not be jeopardized. The radiation level on the top of the core was calculated to analyze how much time the supervisor and the personnel have to secure the area. The resulting analysis is showing that the personnel have several hours after the first leak from the reactor tank to secure the area.

Table 4.5-1 MCNP5 Calculated average power per fuel ring in 100kW UUTR

	# of fuel pins	Average power per fuel element P_{avg} (kW/element)
B Ring	6	1.979
C Ring	11	1.715
D Ring	14	1.475
E Ring	23	1.179
F Ring	19	0.947
G Ring	5	0.700
Total	78	1.282

Table 4.5-2 MCNP5 calculated maximum, minimum and average power per fuel ring and power ratios in 100kW UUTR

# of fuel pins in UUTR	Maximum power per fuel pin P_{max} (kW/pin)	Minimum power per fuel pin P_{min} (kW/pin)	Average power per fuel pin P_{avg} (kW/pin)	P_{max}/P_{min}	P_{max}/P_{avg}
78	2.022	0.609	1.282	3.318	1.577

If the decay heat production is sufficiently low or if there is a long enough interval between reactor shutdown and coolant loss, the convective cooling by air will be enough to maintain the fuel at a temperature, which will not damage the fuel elements. Texas A&M University analyzed this accident and arrived at the following results ["Amendment IX to the Safety Analysis Report - Texas A&M University Nuclear Science Center," November 1, 1972]:

- demonstration that standard fuel suspended in air can tolerate temperatures as high as 900 °C (1,173.15 °K) without damage to the cladding
- this temperature is not exceeded under the conditions of coolant loss if the maximum thermal power in an element is equal to or less than 21 kW for standard fuel and 24 kW for fuel if the reactor is operated for an infinite time prior to the accident
- if reactor operations are limited to 70 MWh per week, power levels up to 24kW/element for standard fuel will not cause element damage in the event of loss of coolant

Calculations performed by GA on the Torrey Pines Mark II reactor yielded similar results ["Final Safety Analysis Report - Stationary Neutron Radiography System," McClellan Air Force Base, Sacramento, CA, Prepared by Argonne National Laboratory-West, January 1992]. According to this calculation, after prolonged operation at 2,000kW (about 32kW per element), an instantaneous loss of water would produce a maximum fuel temperature of about 490 °C (763.15 °K), well below the temperature necessary to cause any failure of the cladding. It can be seen from **Table 4.5-2** that the maximum power level expected in the 100kW UUTR is around 2 kW per element therefore very much below 17kW per element. Also the operation of the UUTR in excess of 70MWh per week is not foreseen. (Typical averages are 50 hours per year). Therefore, with the maximum expected power level in the UUTR core of less than 3kW per element and a conservative prediction of a limiting element power of 24kW, there is a very substantial margin of safety against cladding failure and subsequent release of fission products in the event of a loss of cooling accident.

4.5.2 Reactor Core Physics Parameters

4.5.2.1 The 100kW UUTR Core Description

The UUTR has hexagonal shaped heterogeneous core that has 6 rings, A-ring inner position, B-ring through G-ring. The UUTR core configuration for 100 kW is shown in **Fig. 4.5-1.** In the UUTR operating at nominal maximum power of 100 kW, there are 78 elements and three control rods: safety, shim, and regulation. Experimentally measured total reactivity worth of all three control rods is ~$4.08. There are four different neutron irradiation positions available: Pneumatic Irradiator (PI), fast Neutron Irradiation Facility (FNIF), Thermal Irradiator (TI), and Central Irradiator (CI). The PI uses helium gas to place a sample into the core. The aluminum tube inserted into the central irradiator extends over the entire length of the core up to the upper grid plate. Thermal irradiator is filled with 68% of heavy water to thermalize the neutrons. The FNIF was designed to minimize the gamma irradiation of sample placed into the

FNIF. This fast neutron irradiator provides 1 MeV equivalent fast neutron irradiation environment. The UUTR has two instrumental fuel elements in the C-ring and D-ring as shown in **Fig. 4.5-1**. The UUTR core uses 12 heavy water (68 % concentration) elements and 12 pure graphite elements as a reflector.

[redacted]

Figure 4.5-1 Core configuration of 100kW UUTR representing limiting core configuration (LLC)

4.5.2.2 Calculational Methods and Verification

The UUTR core performances are assessed by modeling 100kW core using the MCNP5. The UUTR core never operated at the power level above 90kW, therefore all measured data are obtained at 90kW power. The MCNP5 model was developed for the UUTR core configuration shown in **Fig. 4.5-1**. The MCNP5 is a general purpose Monte Carlo transport code with capabilities for detailed neutronics computations of 3D reactor cores, and it is well suited to explicitly model the material and geometrical heterogeneities present in the UUTR core. The UUTR facility drawings were used to specify accurately the geometry of the UUTR core and surrounding structures. The geometry of fuel elements and control rods were based on the manufacturing drawings as well as real physical measurements.

Measurement data for the excess reactivity, shutdown margin and the worth of each control rod are available for 90kW UUTR core. The comparisons as presented in **UUTR SAR 4.5.2.3** verifies that the parameters used in the MCNP5 UUTR model are correct.

4.5.2.3 UUTR Core Calculation for 100kW UUTR Limiting Core Configuration

General Core Parameters. Criticality of 90kW UUTR is obtained with 78 standard 20% enriched TRIGA fuel elements and three water-filled control rod positions, as shown in **Fig. 4.5-1**. **UUTR SAR 4.1** and **4.2** describe the principal design parameters of the UUTR fuel pins and control rods. Measured and calculated operating condition's parameters of the 90kW UUTR are shown in **Table 4.5-3**.

Reactor Core Physics Parameters. The reactor core parameters are calculated using the MCNP5 (**Appendix 4.5.A**) and data are benchmarked against the real measurements showing favorable comparisons. **Table 4.5-4** summarizes calculated and measured excess reactivity, shutdown margin and control rod worth. The errors occurring in the experimental measurement of the control rod worths is ~ 10 %. The main sources of the error are due to the reading of control rod positions, power fluctuation during the rod drop and misreading of the rod drop graph. The excess reactivity and shut down margin are within the Technical Specification requirements with this error range. The error of MCNP5 calculation corresponds to one standard deviation, i.e. 68% confidence interval. Considering the measurement error and MCNP5 simulation confidence interval, the MCNP5 calculation results match with the measurement data. It verifies the MCNP5 model of the 100kW UUTR.

The effective delayed neutron fraction for the 100kW core was calculated with MCNP5 by utilizing the expression

$$\beta_{eff} = 1 - \frac{k_p}{k}$$

where k_p is the system eigenvalue assuming fission neutrons are born with the energy spectrum of prompt neutrons, and k is the system eigenvalue assuming fission neutrons are born with the appropriately weighted energy spectra of both prompt and delayed neutrons. The computed k=1.00649±0.00003, and k_p=0.99876±0.00003 are giving, β_{eff}=0.00768±0.00006.

Table 4.5-3 Parameters for UUTR normal operating condition

Power level		**90kW**	**100 kW**
Pool water temperature (average)		20~25 °C(293.15~298.15 °K)	20~25 °C(293.15~298.15 °K)
Bridge radiation Level		~20μR/hr	~20μR/hr
Fuel temperature as measured in two instrumental fuel elements[1]		D-ring: 95 °C (368.15 °K) C-ring: 110 °C (383.15 °K)	
Maximum calculated fuel temperature[2] Peak/Average fuel temperature Average fuel temperature		121.70 °C (394.85 °K) 1.72 70.93 °C (344.08 °K)	129.67 °C (402.82 °K) 1.50 86.70 °C (359.85 °K)
Pool cooling system			
	Flow rate in GPM	ΔT in °C	
Primary	5.2	5 °C	
Secondary	5.0	10 °C	
Excess reactivity at power		\$0.819	
Total rod worth		\$4.08	\$4.08

Table 4.5-4 Measured and calculated excess reactivity, shutdown margin and control rod worth for 90kW and 100kW UUTR core

UUTR		**Measurement average 2005 to 2009 at 90kW (fluctuations in measurement +-10%)**	**MCNP5 calculation for 100kW**
	Excess Reactivity (\$)	0.819	0.840±0.010
	Shutdown Margin (\$)	1.018	0.980±0.023
Control Rod Worth (\$)	Safety	2.243	1.924±0.035
	Shim	1.550	1.468±0.031
	Reg	0.287	0.294±0.022

Neutron lifetime. The prompt-neutron lifetime, l_p, was calculated using the MCNP5 model of the 100kW UUTR, and the $1/v$ absorber method, whereby a small amount of boron is distributed homogeneously throughout the reactor. The calculation of l_p is as follows:

$$l_p = \frac{1}{N_{B-10}\sigma_{a0}v_0}\frac{k_{ref} - k_p}{k_p}$$

where: k_{ref} is the eigenvalue of the original system,

k_p is the eigenvalue of the system with trace amounts of B-10,

$N_{B\text{-}10}$ = boron-10 number density [atoms/(barn-cm)],

v_0 = 220,000 cm/sec, and

σ_{a0} = 3837 barns = $\sigma_{a0}^{B\text{-}10}$ at 220,000 cm/sec

[1] Average temperature measured at three centerline locations inside of the instrumental fuel element

[2] Calculated using PARET-ANL (**Section 4.6**)

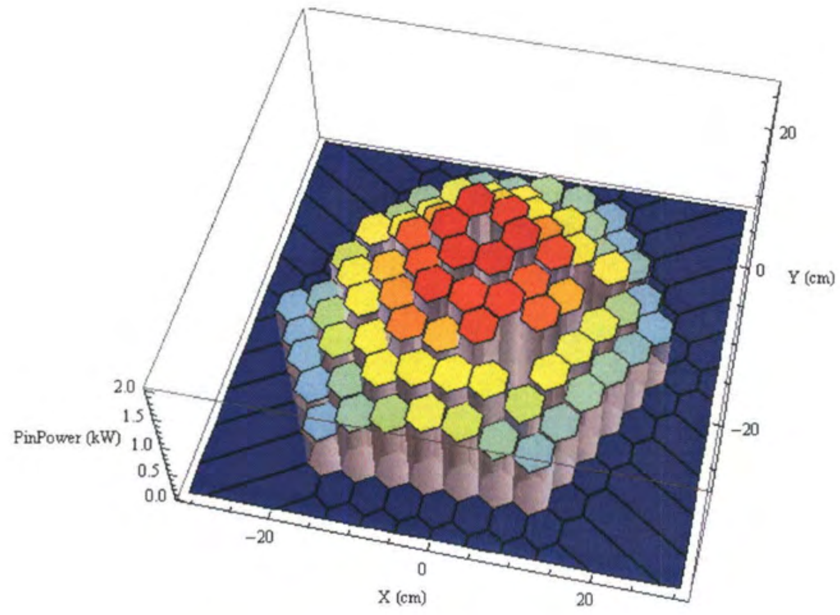
The limit as $N \to 0$ was found by perturbing the system with two different boron concentrations and then linearly extrapolating to N=0. The boron concentrations were 7.5×10^{-8}atoms/(barn-cm) (k_p =1.00504±0.00003, l_p =22.8±0.9 μs) and 1.5×10^{-7}atoms/(barn-cm) (k_p =1.00345±0.00003, l_p =23.9±0.5 μs). The resulted prompt-neutron lifetime for the 100kW UUTR core is 21.7±2.4 μs. The MCNP5 outputs the tally values per neutron, which is irrelevant to the actual operational power. The calculated fuel element power distribution can be normalized to specified core power levels. **Table 4.5-5** and **Table 4.5-6** list the fuel element power distribution per fuel rings at 100kW and 90kW power levels respectively. Fuel elements have less power at 90kW core level than at 100kW power level. **Figure 4.5-2** and **Figure 4.5-3** shows the UUTR fuel element power distributions at 100kW and 90kW total powers, respectively and indicating (a) 3D view of the fuel elements' power, (b) core fuel rings map, and (c) power per each fuel element. Details are presented in **UUTR SAR 4.5.3.6.**

Table 4.5-5 Fuel element power distribution per fuel ring at 100kW power level

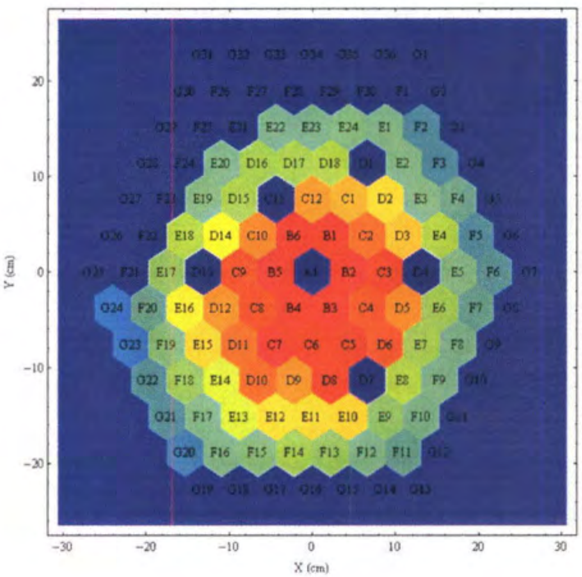
	# of fuel pins	Maximum power per fuel pin P_{max} (kW/pin)	Minimum power per fuel pin P_{min} (kW/pin)	Average power per fuel pin P_{avg} (kW/pin)	P_{max}/P_{min}	P_{max}/P_{avg}
B Ring	6	2.022	1.880	1.979	1.075	1.022
C Ring	11	1.895	1.497	1.715	1.266	1.105
D Ring	14	1.818	1.194	1.475	1.523	1.233
E Ring	23	1.445	0.930	1.179	1.554	1.226
F Ring	19	1.209	0.761	0.947	1.590	1.277
G Ring	5	0.788	0.609	0.700	1.293	1.124
Total	78	2.022	0.609	1.282	3.318	1.577

Table 4.5-6 Fuel element power distribution per fuel ring at 90kW power level

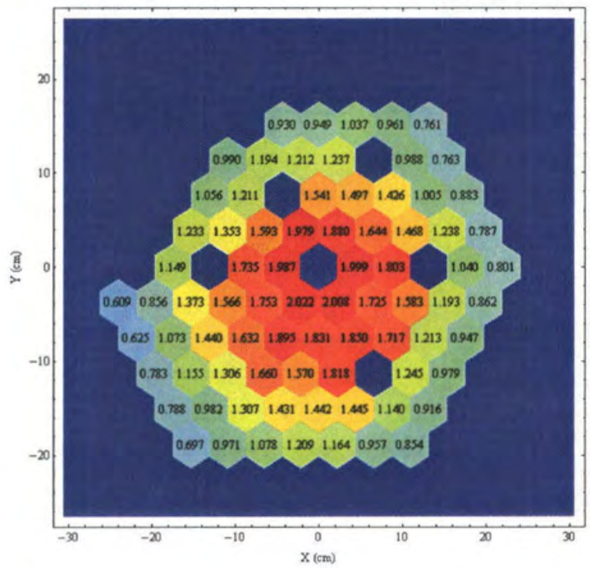
	# of fuel pins	Maximum power per fuel pin P_{max} (kW/pin)	Minimum power per fuel pin P_{min} (kW/pin)	Average power per fuel pin P_{avg} (kW/pin)	P_{max}/P_{min}	P_{max}/P_{avg}
B Ring	6	1.820	1.692	1.781	1.075	1.022
C Ring	11	1.705	1.347	1.544	1.266	1.105
D Ring	14	1.636	1.075	1.327	1.523	1.233
E Ring	23	1.300	0.837	1.061	1.554	1.226
F Ring	19	1.088	0.685	0.852	1.590	1.277
G Ring	5	0.709	0.548	0.630	1.293	1.124
Total	78	1.820	0.548	1.154	3.318	1.577



(a)



(b)



(c)

Figure 4.5-2 100kW UUTR power distribution obtained with MCNP5: (a) 3D view, (b) core fuel rings map, (c) power per fuel element (numbers are in kW)

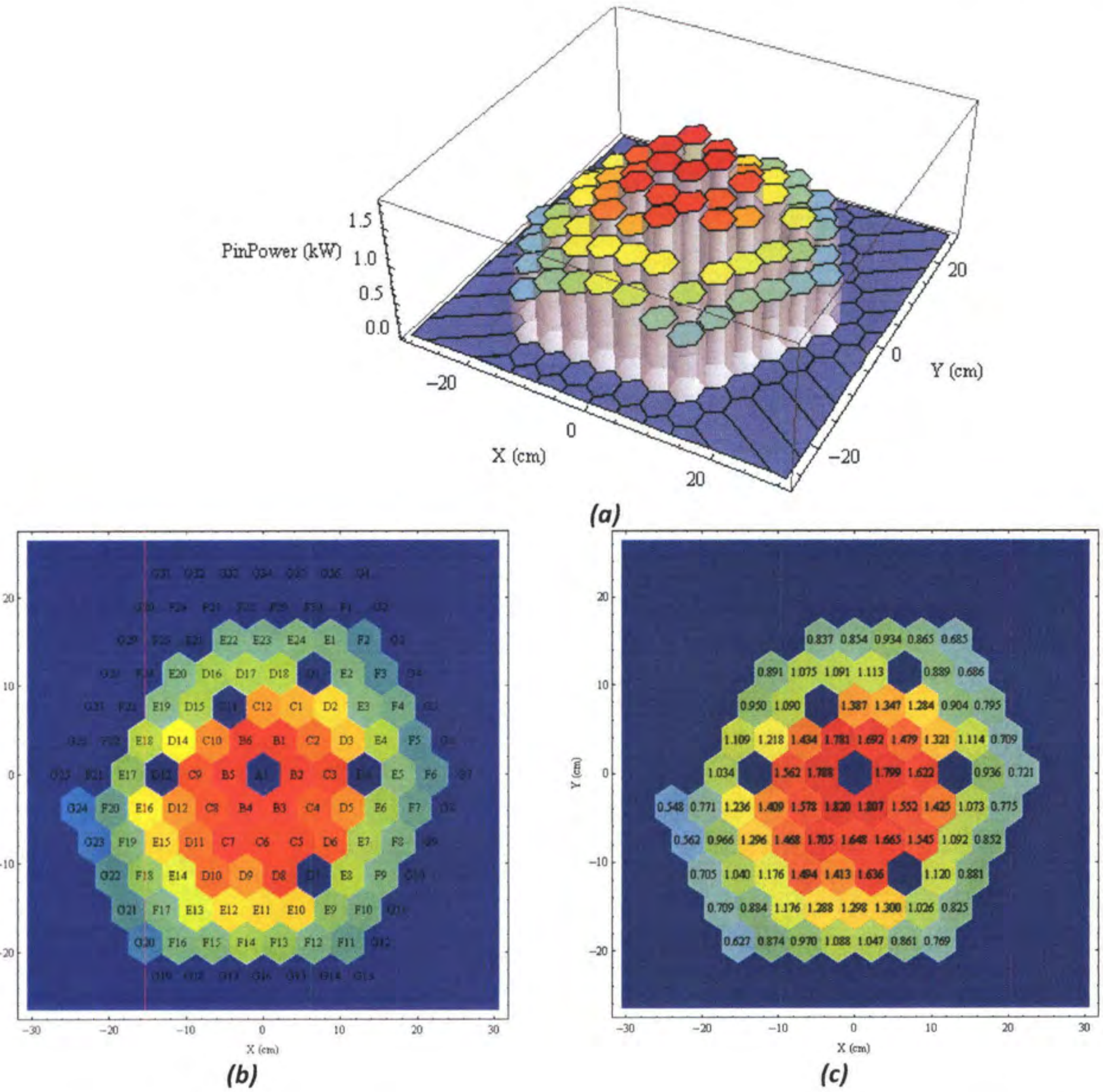


Figure 4.5-3 UUTR fuel elements power distribution at 90kW power level obtained with MCNP5: (a) 3D view, (b) core fuel rings map, (c) power per fuel element (numbers are in kW)

MCNP5 simulation gives also the peaking factor of each fuel element under the current core configuration, as shown in **Fig. 4.5-4**. They are in the range of 1.184 – 1.282 and are the same for 100kW and 90kW power levels. For the whole UUTR core, at 100kW, the maximum linear power density is 67.10 W/cm and the average linear power density is 34.32 W/cm, i.e. the peaking factor for the whole core is 1.955 (= 67.10 / 34.32); at 90kW power level, the maximum linear power density is 60.39 W/cm, the average linear power density is 30.89 W/cm, and the peaking factor for the whole core is same as for the 100kW operating power, i.e. 1.955.

The axial linear power density profile of the fuel element with the highest power is shown in **Fig. 4.5-5** for 100kW and **Fig. 4.5-6** for 90kW UUTR operation power; the origin point is set at the center of the fuel element. The increase of the linear power density at both ends of the fuel element is because of the graphite blocks present at both ends reflecting the neutrons.

Figures 4.5-7 and **4.5-8** show the neutron flux at the center plane of the 100kW and 90kW UUTR power level operation, respectively (X and Y coordinates are the index of mesh tally and the center of the mesh is at the center of the core). Both neutron flux distributions have the same shape but different magnitude. The thermal flux peaks at the position where there is no fuel element (ring is filled with the water only). Fast flux is the highest in the middle of the UUTR core.

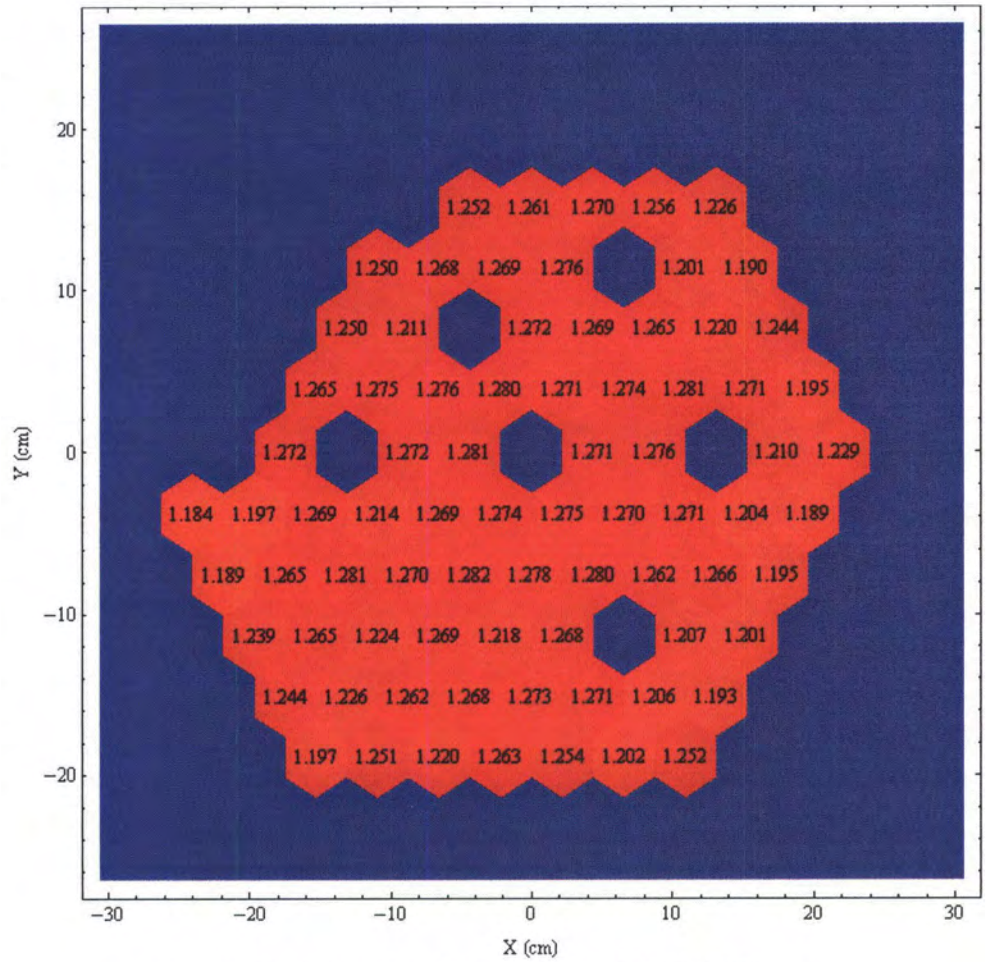


Figure 4.5-4 Peaking factors of the 100kW UUTR core

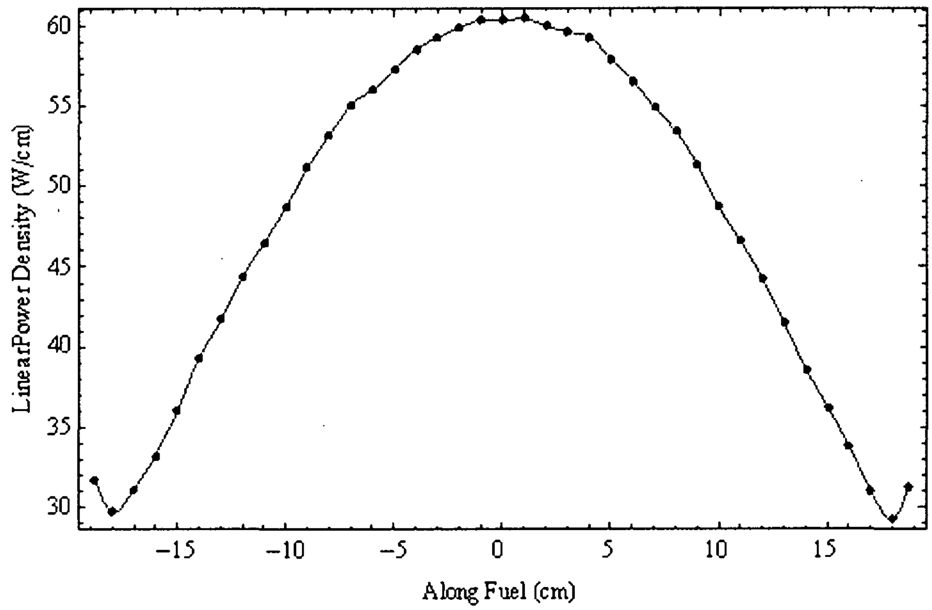


Figure 4.5-5 Axial power profile of the element with highest power for 100kW UUTR

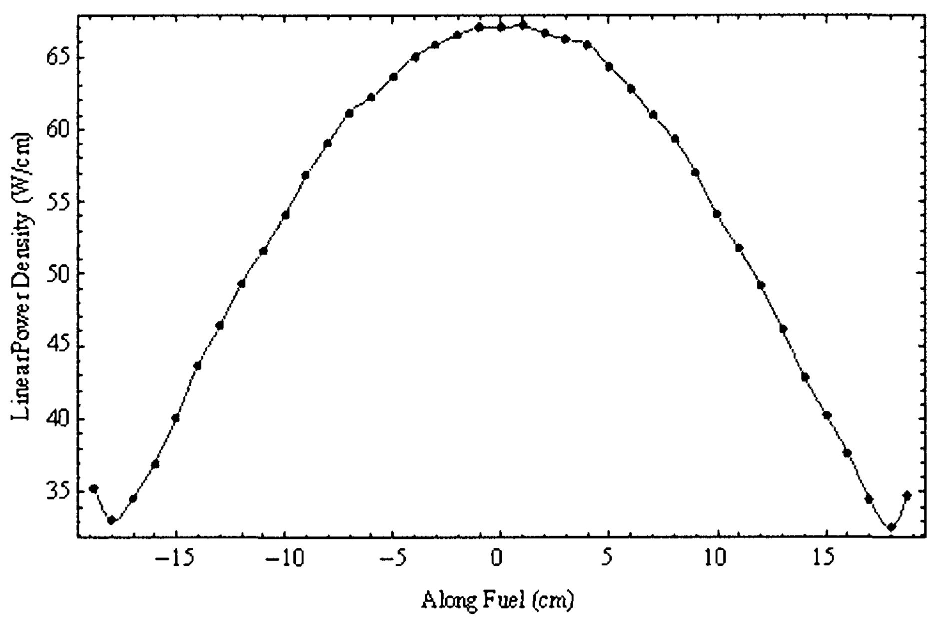
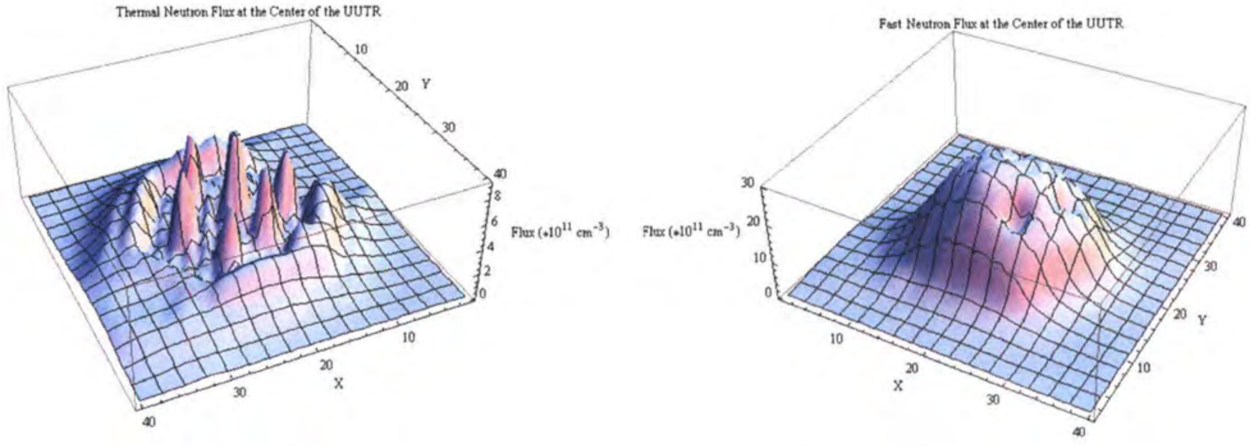
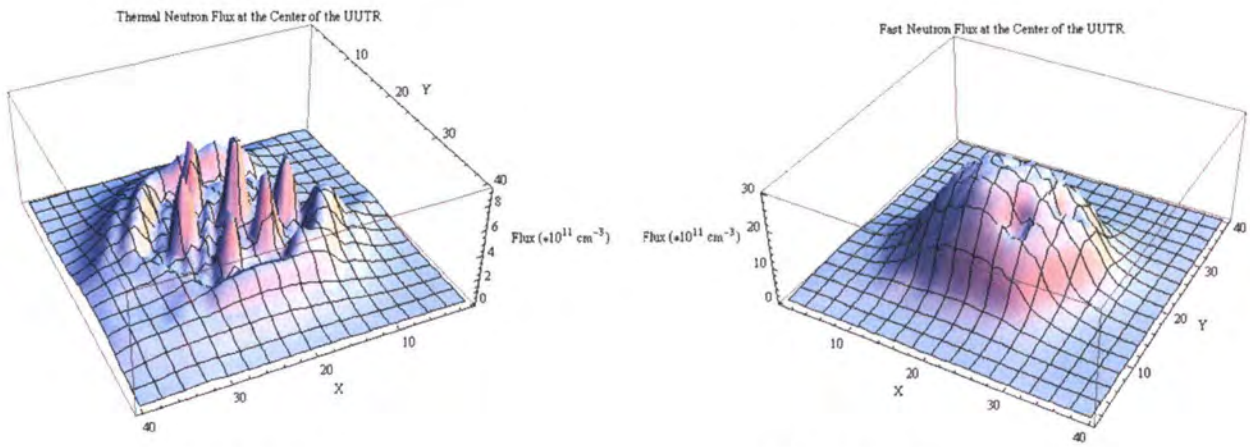


Figure 4.5-6 Axial power profile of the element with highest power for 90kW UUTR



(a) Thermal Neutron Flux **(b) Fast Neutron Flux**

Figure 4.5-7 Neutron flux at the center plane of the 100kW UUTR core from MCNP5 simulation



(a) Thermal Neutron Flux **(b) Fast Neutron Flux**

Figure 4.5-8 Neutron flux at the center plane of the 90kW UUTR core from MCNP5 simulation

4.5.3 Operating Limits

4.5.3.1 Reactor Fuel Temperature for 100kW UUTR

The basic safety limit for the TRIGA reactor system is the fuel temperature. UUTR contains two types of ZrH fuel, aluminum cladding fuel with $ZrH_{1.0}$ and SS cladding fuel with $ZrH_{1.6}$. **Figure 4.5-9** shows the phase diagram for ZrH matrix and indicates the points of the

UUTR operation range for two types of the UUTR fuels. The diagram indicates that the higher hydride compositions are single-phase and are not subject to the large volume changes associated with the phase transformations at approximately 530 °C (803.15 °K) in the lower hydrides. It has been shown that the higher hydrides lack any significant thermal diffusion of hydrogen. These two facts preclude concomitant volume changes. The effect of uranium presence in ZrH matrix shifts all the phases' boundaries of the ZrH phase diagram to slightly lower temperatures.

The results of General Atomics experimental and theoretical determinations show that fuel element integrity is not compromised for cladding temperatures at or less than 500 °C (773.15 °K). The limiting effect of fuel temperature is the hydrogen gas pressure causing cladding stress. The UUTR operates in a lower end fuel cladding temperature regimes and therefore it is not predicted that fuel integrity will be compromised; maximum fuel surface temperature is ~97 °C (370.15 °K), and maximum fuel centerline temperature is ~121 °C (394.15 °K) (**Table 4.6-1**).

4.5.3.2 Effective Delayed Neutron Fraction for 100kW UUTR

The effective delayed neutron fraction for the UUTR core operating at 100kW was calculated with the MCNP5 code and based on:

$$\beta_{eff} = 1 - \frac{k_p}{k}$$

where k_p is the system eigenvalue assuming fission neutrons are born with the energy spectrum of prompt neutrons, and k is the system eigenvalue assuming fission neutrons are born with the appropriately weighted energy spectra of both prompt and delayed neutrons. The computed values are summarized in **Table 4.5-7**.

The effective delayed neutron fractions are calculated therefore, as follows:

$$\beta_{eff}^{100kW} = 1 - \frac{0.99876}{1.00649} = 0.00768$$

Table 4.5-7 MCNP5 Calculated Effective Delayed Neutron Fraction for 100kW UUTR

k	k_p	β_{eff}
1.00649±0.00003	0.99876±0.00003	0.00768±0.00006

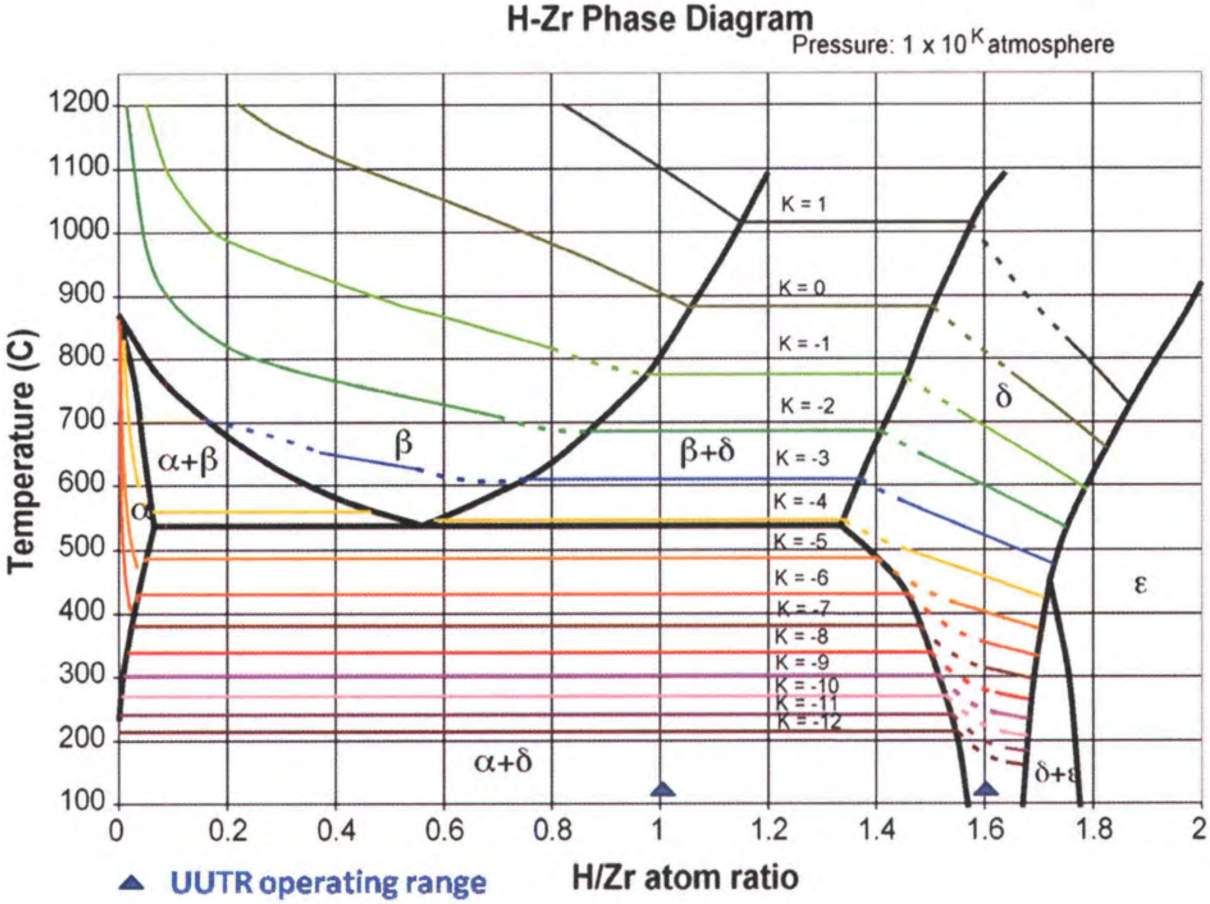


Figure 4.5-9 ZrH phase diagram and the UUTR operating range [Huang, J., et al., 2000. Hydrogen redistribution in zirconium hydride under a temperature gradient. J. Nucl. Sci. Technol. 37, 887; Olander, D., Ng, M., 2005. Hydride fuel behavior in LWRs. J. Nucl. Mater. 346, 98

According to the error propagation equation:

$$\Delta Y = \left|\frac{\partial F}{\partial X_1}\right|\Delta X_1 + \left|\frac{\partial F}{\partial X_2}\right|\Delta X_2 + \cdots + \left|\frac{\partial F}{\partial X_n}\right|\Delta X_n$$

if

$$Y = F(X_1, X_2,, X_n),$$

$$\Delta\beta_{eff} = k_p \frac{\Delta k}{k^2} + \frac{\Delta k_p}{k}$$

It follows:

$$\Delta\beta_{eff}^{100kW} = 0.99876\frac{0.00003}{1.00649^2} + \frac{0.00003}{1.00649} = 0.00006$$

$$\Delta\beta_{eff}^{250kW} = 1.00149\frac{0.00003}{1.00935^2} + \frac{0.00003}{1.00935} = 0.00006$$

4.5.3.3 Prompt Negative Temperature Coefficient of 100kW UUTR

The temperature coefficient of reactivity represents the change in reactivity per degree change in temperature. Because different materials in the reactor have different reactivity changes with temperature and the various materials are at different temperatures during reactor operation, several different temperature coefficients are used. Usually, the two dominant temperature coefficients are the moderator temperature coefficient and the fuel temperature coefficient.

The basic parameter which allows the TRIGA reactor system to operate safely with step insertions of reactivity is the strongly negative temperature coefficient associated with the TRIGA fuel and core design. This temperature coefficient allows a greater freedom in steady-state operation as the effect of accidental reactivity changes occurring from the experimental devices in the core is greatly reduced. This temperature coefficient arises primarily from a change in the fuel utilization factor resulting from the heating of the uranium-zirconium hydride fuel-moderator elements. The coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator, thus, fuel and solid moderator temperatures rise simultaneously. A quantitative calculation requires knowledge of the energy dependent distributions of thermal neutron flux in the reactor. The basic physical processes which occur when the fuel-moderator elements are heated are: the rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, their mean free path is increased appreciably. Since the average chord length in the fuel element is comparable with a mean free path, the probability of escape from the fuel element before capture is increased. In the water, the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel, thus, causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature dependent fuel utilization factor for the unit cell in the core, which decreases the ratio of absorptions in the fuel to total cell absorptions as the fuel element temperature is increased. This yields a loss of reactivity. The temperature coefficient then depends on spatial variations of the thermal neutron spectrum over distances of the order of a mean free path with large changes of the mean free path occurring because of the energy change in a single collision.

The prompt-temperature coefficient for the UUTR operating at nominal power of 100kW, α_F, was calculated by varying the fuel temperature while keeping other core parameters unchanged. The effective delayed neutron fraction (**Table 4.5-7**) is used to convert the multiplication factor to reactivity. The MCNP5 model of the reactor operation includes all control rods out. The results are shown in **Table 4.5-8**. The trend of reactivity change as a function of fuel temperature for the UUTR operating at 100kW is shown in **Fig. 4.5-10**.

The temperature coefficients are calculated as follows:

$$\alpha(\$/^{o}K)=\frac{\dfrac{k_2-k_1}{k_2\times k_1}}{(T_2-T_1)}\times\frac{1}{\beta}$$

Table 4.5-8 MCNP5 Calculated Prompt-Temperature Coefficients in 100kW UUTR

β=0.00768							
T °K (°C)	k	$(k_2-k_1)/k_1 x k_2$	\$Incr. $=(k_2-k_1)/(\beta x k_2 x k_1)$	\$Total	Δ°K (T_2-T_1)	α(\$/°K) $= [(k_2-k_1)/k_2 x k_1)]/[\beta x(T_2-T_1)]$	Applicable temperature range, °K (°C)
293 (19.85)	1.00649						
600 (326.85)	0.97869	-0.02822	-3.6748	-3.6748	307	-0.011970	293-600 (19.85-326.85)
800 (526.85)	0.95322	-0.0273	-3.5549	-7.2297 [\$Incr$_{600}$+\$Incr$_{800}$]	200	-0.01777	600-800 (326.85-526.85)
1200 (926.85)	0.91442	-0.0445	-5.7960	-13.0257 [\$Incr$_{600}$+\$Incr$_{800}$+ \$Incr$_{1200}$]	400	-0.01449	800-1200 (526.85-926.85)
The MCNP5 was used to model the UUTR core at the operating power of 100kW.						**-0.01436** = [(0.91442-1.00649) / (0.91442x1.00649)] / [0.00768x(1200-293)]	293-1200 (19.85-926.85)

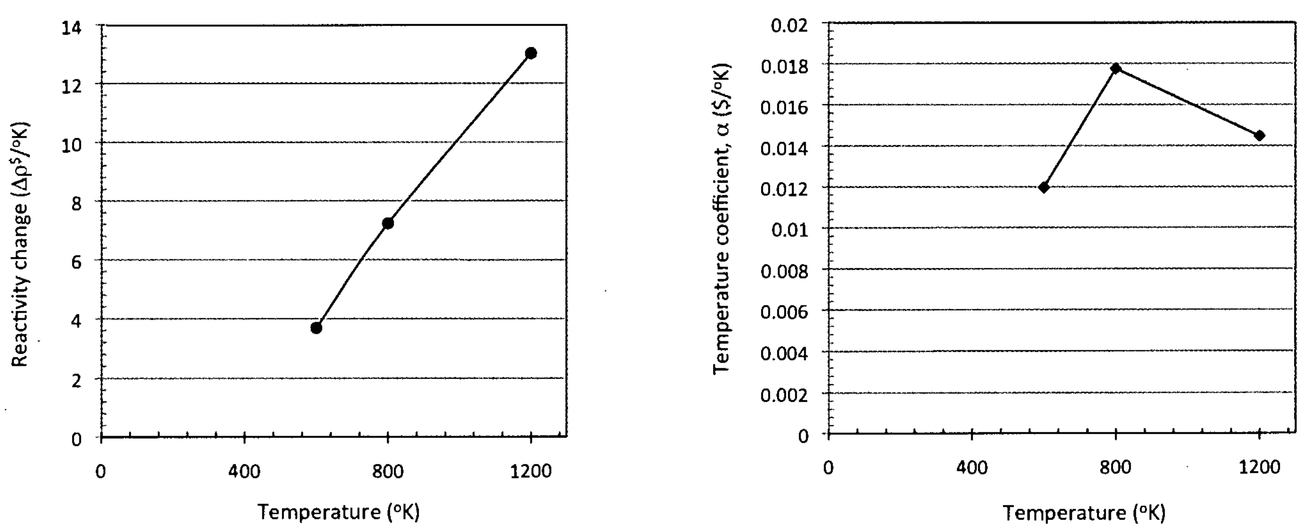


Figure 4.5-10 MCNP5 calculated fuel temperature coefficient and reactivity change as a function of temperature for 100kW UUTR

4.5.3.4 Moderator Temperature Coefficient of the 100kW UUTR

The moderator temperature coefficient represents the change in reactivity per degree change in moderator temperature. The magnitude and sign (+ or -) of the moderator temperature coefficient a function of the moderator-to-fuel ratio: the under moderated reactors have a negative moderator temperature coefficient; while the over moderated reactors have a positive moderator temperature coefficient. A negative moderator temperature coefficient is desirable because of its self-regulating effect: an increase in reactivity causes the reactor to produce more power which raises the temperature of the core and adds negative reactivity, which slows down the increase of reactor power.

The moderator temperature coefficient of reactivity, α_M, was determined by varying the moderator temperature using the MCNP5 model of the 100kW UUTR core; results are shown in **Table 4.5-9** where the α_M is defined with:

$$\alpha_M = \frac{\frac{k_2 - k_1}{k_2 \times k_1}}{\Delta T_M} \times \frac{1}{\beta}$$

4.5.3.5. Void Coefficient of the 100kW UUTR

The void coefficient of reactivity is defined as the change in reactivity per percent change in void volume. As the reactor power is raised to the point where the steam voids start to form, voids displace moderator from the coolant channels within the core. This displacement reduces the moderator-to-fuel ratio, and in an under moderated core, results in a negative reactivity addition, thereby limiting reactor power rise. The void coefficient is significant in water-moderated reactors that operate at or near saturated conditions.

Table 4.5-9 MCNP5 Calculated Moderator Temperature Coefficients in 100kW UUTR

β=0.00768							
T°K (°C)	k	$(k_2-k_1)/k_1 x k_2$	\$Incr. $(k_2-k_1)/(\beta x k_2 x k_1)$	\$Total	Δ°K (T_2-T_1)	α(\$/°K) =$[(k_2-k_1)/k_2 x k_1)]/[\beta x(T_2-T_1)]$	Applicable temperature range, °K (°C)
293 (19.85)	1.00649						
600 (326.85)	0.97565	-0.0314	-4.0893	-4.0893	307	-0.0133	293-600 (19.85-326.85)
MCNP5 (version 1.5) was used to calculate the moderator temperature coefficient only at 293 °K (19.85 °C) and 600 °K (326.85 °C) because the common temperature for S(α,β) treatment and light water cross section are available only for these two temperature values.						**-0.0133** =[(0.97565-1.00649) /0.97565x1.00649]/[0.00768x(600-293)]	293-600 (19.85-326.85)

The void coefficient of reactivity for the UUTR is determined using the MCNP5 model of the 100kW core. The voiding of the core was introduced by uniformly reducing the density of the water moderator in the core. Four water densities were selected to calculate the void coefficient. The results are summarized in **Table 4.5-10**. The void coefficient trend as a function of void (%) is shown in **Fig. 4.5-11**. The void coefficients are obtained from:

$$\alpha = \frac{\Delta\%Void}{\%Void_2 - \%Void_1} \times \frac{1}{\beta}$$

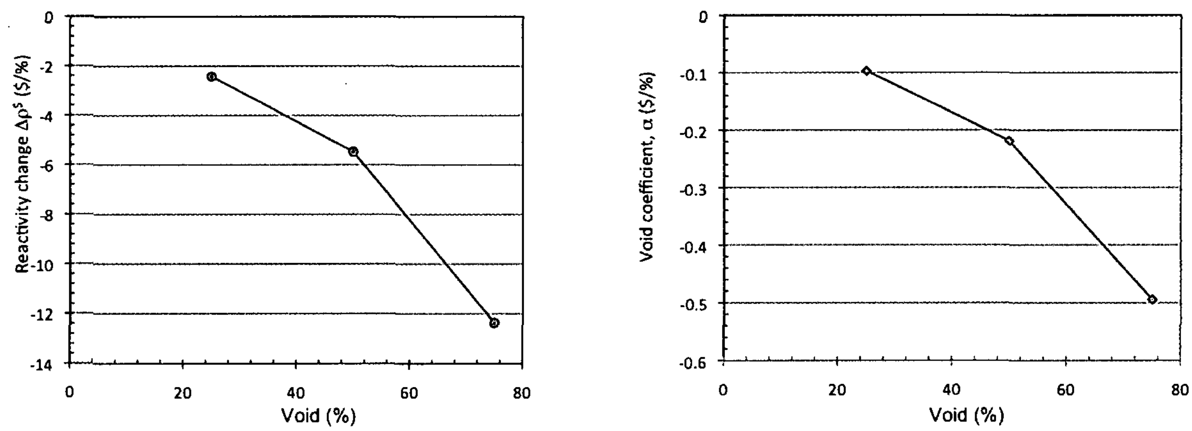


Figure 4.5-11 MCNP5 calculated void coefficient and reactivity change as a function of void (%) for 100kW UUTR

Table 4.5-10 MCNP5 Calculated Void Coefficients in 100kW UUTR

				β=0.00768			
Void (%)	*k*	$(k_2-k_1)/k_1 x k_2$	\$Incr. =(k_2-k_1) /$(\beta x k_2 x k_1)$	\$Total	*Δ%void (%void$_2$-%void$_1$)*	*α(\$/% void)* =[(k_2-k_1) /$k_2 x k_1$)]/ [βx((%void$_2$-%void$_1$)]	Applicable *%void* range
0	1.00649						
25	0.98794	-0.01866	-2.4291	-2.4291	25	-0.09716	0-25
50	0.94858	-0.04200	-5.4688	-7.8979 [\$Incr$_{25}$+\$Incr$_{50}$]	25	-0.21875	25-50
75	0.87018	-0.09498	-12.3672	-20.2651 [\$Incr$_{25}$+\$Incr$_{50}$+ \$Incr$_{75}$]	25	-0.49469	50-75
The MCNP5 was used to model the UUTR core at the operating power of 100kW.						**-0.2702** [(0.87018-1.00649) / 0.87018x1.00649]/ [0.00768x(75-0)]	0-75

4.5.3.6 Prompt-Neutron Lifetime for 100kW UUTR

The prompt-neutron lifetime, l_p, was calculated using the MCNP5 models of the 100kW UUTR core, and the $1/v$ absorber method, whereby a small amount of boron is distributed homogeneously throughout the reactor. The calculation of l_p is as follows:

$$l_p = \frac{1}{N_{B-10}\sigma_{a0}v_0}\frac{k_{ref} - k_p}{k_p}$$

where

k_{ref} is the eigenvalue of the original system,
k_p is the eigenvalue of the system with trace amounts of B-10,
$N_{B\text{-}10}$ = boron-10 number density [atoms/(barn-cm)],
v_0 = 220,000 cm/sec, and
σ_{a0} = 3837 barns = $\sigma_{a0}^{\text{B-10}}$ at 220,000 cm/sec.

The limit as $N \rightarrow 0$ was found by perturbing the system with two different boron concentrations and then linearly extrapolating to N=0. The results are summarized in **Table 4.5-11.** Based on **Table 4.5-11**, considering l_p is a function of N^{B-10}, a linear equation can be constructed from two calculated points: (N_1^{B-10}, l_1^p) and (N_2^{B-10}, l_2^p), as follows:

$$l_p = \frac{N_{B-10} - N_1^{B-10}}{N_2^{B-10} - N_1^{B-10}}\left(l_2^p - l_1^p\right) + l_1^p$$

Table 4.5-11 MCNP5 Calculated Prompt Neutron Lifetime in 100kW UUTR [B concentrations are the same as for the Oregon State TRIGA]

B concentration [atoms/(barn-cm)]	k_p	l_p [μs]
7.5×10^{-8}	1.00504±0.00003	22.79±0.94
1.5×10^{-7}	1.00345±0.00003	23.93±0.47

By setting $N_{B\text{-}10}$ = 0, the prompt-neutron lifetime of the core without boron can be computed by extrapolation method:

$$l_0^p = -N_1^{B-10}\frac{l_2^p - l_1^p}{N_2^{B-10} - N_1^{B-10}} + l_1^p$$

Therefore, it follows:

$$l_{100kW}^p = -7.5\times10^{-8}\frac{23.93 - 22.79}{1.5\times10^{-7} - 7.5\times10^{-8}} + 22.79 = 21.7\times10^{-6}$$

$$l_{250kW}^{p} = -7.5\times10^{-8}\frac{23.46-23.35}{1.5\times10^{-7}-7.5\times10^{-8}}+23.35 = 23.2\times10^{-6}$$

The error of l_1^p and l_2^p is defined with:

$$\Delta l_n^p = \frac{1}{N_n^{B-10}\sigma_{a0}v_0}\left(\frac{\Delta k_{ref}}{k_n^p}+\frac{k_{ref}\Delta k_n^p}{\left(k_n^p\right)^2}\right)$$

Thus giving:

$$\Delta^{100kW} l_1^p = \frac{1}{7.5\times10^{-8}\times3837\times220000}\left(\frac{0.00003}{1.00504}+\frac{1.00649\times0.00003}{1.00504^2}\right) = 0.94\times10^{-6}$$

$$\Delta^{100kW} l_2^p = \frac{1}{1.5\times10^{-7}\times3837\times220000}\left(\frac{0.00003}{1.00345}+\frac{1.00649\times0.00003}{1.00345^2}\right) = 0.47\times10^{-6}$$

$$\Delta^{250kW} l_1^p = \frac{1}{7.5\times10^{-8}\times3837\times220000}\left(\frac{0.00003}{1.00786}+\frac{1.00935\times0.00003}{1.00786^2}\right) = 0.94\times10^{-6}$$

The error of l_0^p is defined with:

$$\Delta^{250kW} l_2^p = \frac{1}{1.5\times10^{-7}\times3837\times220000}\left(\frac{0.00003}{1.00636}+\frac{1.00935\times0.00003}{1.00636^2}\right) = 0.47\times10^{-6}$$

$$\Delta l_0^p = \frac{N_1^{B-10}}{\left|N_2^{B-10}-N_1^{B-10}\right|}\left(\Delta l_2^p+\Delta l_1^p\right)+\Delta l_1^p$$

Therefore,

$$\Delta^{100kW} l_0^p = \frac{7.5\times10^{-8}}{1.5\times10^{-7}-7.5\times10^{-8}}(0.94+0.47)\times10^{-6}+0.94\times10^{-6} = 2.4\times10^{-6}$$

$$\Delta^{250kW} l_0^p = \frac{7.5\times10^{-8}}{1.5\times10^{-7}-7.5\times10^{-8}}(0.94+0.47)\times10^{-6}+0.94\times10^{-6} = 2.4\times10^{-6}$$

The prompt-neutron lifetime for 100 kW is 21.7±2.4 μs. The prompt-neutron lifetime depends on the buckling factor. For example, the prompt-neutron life-time for the Oregon

State University TRIGA HEU is 18.7 ± 2.8 μs, and the LEU TRIGA prompt-neutron lifetime is 22.6±2.9 μs.

4.5.3.7 Xe-Sm Worth in UUTR

The maximum reactivity loss due to temperature and xenon poisoning when operating at a steady power of 100kW is about 0.488% ($0.02). Xe-135 is produced directly in only 0.3% of all U-235 fissions. Total fission product yield of Xe-135 is 6.6%. About 95% of Xe-135 comes from I-135 decay [DOE Fundamentals Handbook: Nuclear Physics and Reactor Theory, Jan. 1993, Retrieved 2009]. Xenon is removed from the core by β decay to Cs-135 with half-life of 9.1 hours. Xe-135 can be removed from the reactor core by neutron absorption also (σ_a = 3.5×10^6 barns). Typically, about 90% of Xe-135 is removed by neutron absorption and 10% by β decay. The amount of Xe-135 in a core depends on the neutron flux. Generally, the Xe-135 approaches its equilibrium after 10 hours of reactor operation. The negative reactivity of xenon is usually less than 10% $\Delta k/k$. For reactors with very low thermal flux levels (~5×10^{12} neutrons/cm^2-sec or less), most of xenon is removed by the decay as opposed to removal through neutron absorption. The reactor shutdown does not cause any xenon-135 peaking effect. For the UUTR (thermal neutron flux is ~10^{11} neutrons/cm^2-sec), the concentration of Xe-135 is negligible and most Xe-135 will decay in less than 20 minutes after the shutdown of the reactor.

Sm-149 has an absorption cross section of 4.2×10^4 barns. Sm-149 is a stable isotope and total fission product yield is 1.4%. The effect of these fission product poisons is to reduce the thermal utilization factor, so these fission product poisons are regarded as sources of negative reactivity in the core.

4.5.3.8 Influence of the Experiments

The UUTR limits the worth of a single experiment to assure that sudden removal of the experiment will not cause the fuel temperature to rise above the critical temperature level of 500°C (773.15 °K) (**UUTR TS 3.8**). The limits are defined as follows:

- Any experiment with reactivity worth greater than $1.00 is secured to prevent unplanned removal from or insertion into the reactor
- The excess reactivity for reference core is less than $1.20
- The reactivity worth of an individual experiment is limit to no more than $1.20

By limiting the worth of all experiments in the reactor at one time to $1.20 it assures that the removal of the total worth for all experiments not to exceed the fuel element temperature limit of 773.15 °K (500°C) for an aluminum element and 1273.15 °K (1,000°C) for a stainless steel element (**UUTR TS 2.1**). Additionally, the TS for the UUTR specify the amount of any hazardous or explosive samples allowed to be used per experiment (**UUTR TS 3.8**). All samples in an experiment must be double encapsulated with a polyethylene vial and sealed with wax to prevent any material from leaking into the UUTR reactor pool tank. The UUTR experiments do not generate any Xe-135 or Sm-149. The UUTR does not allow samples to be put in or pulled out from the core during the reactor operation except if the reactor supervisor and/or the Director approve it. In general, the reactivity worth for such special experiments is less than \$0.30 so that the reactor power or fuel temperature will not fluctuate significantly.

4.5.3.9 Minimum Control Rod Worths and Stuck Rod Worths for All Allowed Core Conditions

The UUTR has three control rods: safety, shim, and regulation. According to the UUTR measurement, the rod worths for these three control rods are \$2.24, \$1.55, and \$0.287, respectively. Shutdown margin is approximately \$1.018 and excess reactivity is around \$0.819. In the control system, the control rods are partly or fully inserted in the core to suppress the excess reactivity loaded for fuel burn-up requirements. The postulated reactivity fault is based on one of these rods being withdrawn from its furthest insertion to the full out position. This same control rod is then assumed to be stuck in the fully withdrawn position. The requirement is then to demonstrate a safe shutdown reactivity balance with the remaining two control rods inserted in the presence of this positive reactivity fault. In the UUTR, the safety control rod that is highest worth control rod is fully out from the core. Shim and regulation control rods control the reactor power. In the case that safety rod is failed to insert, shim and regulation rods have enough negative reactivity to shut down the reactor. The reactivity of shim and regulation rods are greater than shutdown margin, the application of the stuck rod criterion ensures that the failure of the highest worth control rod will not prevent the control rod system from shutting down the reactor.

4.5.3.10 Transient Analysis of an Uncontrolled Rod Withdrawal

The total control rod worth of three control rods is \$4.08 (as measured and shown in **Table 4.5-4**). The control rod withdrawn time is 77 sec for all these three rods. Thus, the maximum reactivity insertion rate is \$0.053/sec, which is well below the **UUTR TS limit** of \$0.30/sec. In order to analyze this accident, one-delay group model was used with the prompt jump approximation based on "Dynamics of Nuclear Reactors" by David Hetrick, U. of Chicago Press, 1971. For linear increase in reactivity (a ramp input), this model gives:

$$\frac{P}{P_0} = e^{-\lambda t}\left(\frac{\beta}{\beta - \gamma t}\right)^{1+\frac{\lambda\beta}{\gamma}}$$

where:

P = final power level;
P_0= initial power level;
β= total delayed neutron fraction = 0.00768;

λ= one group decay constant (sec^{-1}) = 0.405 sec^{-1};
t = time (sec);
γ= linear insertion rate of reactivity ($\Delta k/k\ sec^{-1}$); and
$a = 1 + \lambda\beta/\gamma$.

The UUTR safety control rod worth (a rod with maximum worth) is \$2.243 (as measured and as shown in **Table 4.5-4**). This rod is located at position D7, as shown in **Fig. 1.3-4**. Based on **Fig. 4.5-3** the average power per fuel element surrounding the safety rod is 1.382kW ((1.545+1.12+1.026+1.3+1.636+1.665)/6); the average power per fuel element surrounding the shim rod, located at D13, is 1.261kW; the average power per fuel element surrounding the regulation rod, located at D1, is 1.072kW. Therefore, the safety control rod region is the most reactive region in the core compared to other two control rod regions.

Fully withdrawn safety control rod will take 77 sec resulting in \$0.029/sec linear reactivity insertion. For an initial power level of 0.1kW, and the trip setpoint at 100kW, the reactor power was calculated to reach the trip setpoint in about 22.74 seconds:

$$\frac{P}{P_0} = e^{-\lambda t}\left(\frac{\beta}{\beta - \gamma t}\right)^{1+\frac{\lambda\beta}{\gamma}}$$

$$\frac{100}{0.1} = e^{-0.405t}\left(\frac{0.00768}{0.00768 - 0.029 \times 0.00768t}\right)^{1+\frac{0.405\times 0.00768}{0.029\times 0.00768}}$$

$$1000 = e^{-0.405t}\left(\frac{1}{1-0.029t}\right)^{14.9655}$$

$$t \approx 22.74\,\text{sec}$$

Assuming that it takes 0.02 sec for the trip setpoint signal to cause a maximum release of the control rods and 2 sec for rod drop time which is a UUTR TS limit, the peak reactivity inserted will be \$0.72 (=0.029 × (22.74+2.02)). This is considerably less than the limiting reactivity insertion of \$1.2 as specified in the UUTR TS, and, thus, should produce no adverse safety effects. Assuming now an initial power of 90kW and the trip setpoint at 100kW, it is calculated that the trip setpoint will be reached in 2.36 sec. Allowing for total of 2.02 sec for the rods to be released (electronic response time + rod drop time), the peak reactivity inserted will be \$0.13 (=0.029 × (2.364+2.02)). This is well below the limiting reactivity insertion of \$1.2.

Because the UUTR control mechanism does not allow for multiple control rods to be withdrawn simultaneously, the accident of an uncontrolled withdrawal of all control rods is unlikely. However, this scenario is calculated anyway to assure that the limits even in an unlikely event will not be jeopardized. The total control rod worth measured is \$4.08 (as shown in **Table 4.5-4**). Withdrawal time of 77 sec produces \$0.053/sec linear reactivity insertion. Therefore, it will take 9.73 sec to reach 100kW assuming 0.1kW initial power and 0.27 sec to reach 100kW assuming 90kW initial power. Allowing total rod drop time of 0.92 sec (response time + rod drop time), the peak reactivity inserted will be \$0.56 for power from 0.1 kW to 100 kW and \$0.06 for power from 90 kW to 100 kW, respectively, which are both below the limitation of \$1.20.

4.5.3.11 Shutdown Margin Calculations for Limiting Core Condition

The shutdown margin of the UUTR configuration referred to the reference core, with the highest worth rod (safety rod) fully withdrawn is \$1.018 for 90kW obtained from measurement and \$0.980±\$0.023 for 100kW from the MCNP5 simulations. The shutdown margin is greater than the value as specified in **UUTR TS 3.1.2** (\$0.50).

Appendix 4.5.A

MCNP5 input file for the 100kW UUTR core

```
TRIGGA 3D Model
c New SS Fuel
100   1   -5.636 -2 11 -12                u=1    imp:n=1 $Fuel Meat
101   2   -1.70  -2 12 -14                u=1    imp:n=1 $Up Graphite
102   2   -1.70  -2 13 -11                u=1    imp:n=1 $Down Graphite
103   3   -7.92  (-1 15 -16) (2:-13:14)   u=1    imp:n=1 $Cladding
104   4   -1.0   1:-15:16 92 -93          u=1    imp:n=1 $H2O
c Old SS Fuel
110 like 100 but mat=12 rho=-5.636        u=2    imp:n=1 $Fuel Meat
111 like 101 but                          u=2    imp:n=1 $Up Graphite
112 like 102 but                          u=2    imp:n=1 $Down Graphite
113 like 103 but                          u=2    imp:n=1 $Cladding
114 like 104 but                          u=2    imp:n=1 $H2O
c Al Fuel
120   5   -6.143 -3 21 -22                u=3    imp:n=1 $Fuel Meat
121   2   -1.70  -3 22 -24                u=3    imp:n=1 $Up Graphite
122   2   -1.70  -3 23 -21                u=3    imp:n=1 $Down Graphite
123   6   -2.70  (-1 25 -26) (3:-23:24)   u=3    imp:n=1 $Cladding
124   4   -1.0   1:-25:26 92 -93                 u=3   imp:n=1 $H2O
c Instrumental Fuel
130 like 110 but                          u=4    imp:n=1 $Fuel Meat
131 like 111 but                          u=4    imp:n=1 $Up Graphite
132 like 112 but                          u=4    imp:n=1 $Down Graphite
133 like 113 but                          u=4    imp:n=1 $Cladding
134 like 114 but                          u=4    imp:n=1 $H2O
c Graphite
140   2   -1.70    -3 23 -24              u=6    imp:n=1 $Graphite
143    like 123 but                       u=6    imp:n=1 $Cladding
144    like 124 but                       u=6    imp:n=1 $H2O
c Heavy Water
150   7   -1.056  -3 23 -24               u=7    imp:n=1 $D2O
153    like 123 but                       u=7    imp:n=1 $Cladding
154    like 124 but                       u=7    imp:n=1 $H2O
c Water
160   4   -1.0     -1 92 -93              u=8    imp:n=1 $H2O
161   4   -1.0      1 92 -93              u=8    imp:n=1 $H2O
c Safety Control Rod
170   9   -2.52   -46 11 -93              u=10   imp:n=1 $B4C
171   6   -2.7     46 -47 11 -93          u=10   imp:n=1 $Al Cladding
172   4   -1.0    (47 -50 11 -93):(-50 -11 92)  u=10  imp:n=1 $H2O
173   6   -2.7     50 -1 92 -93           u=10   imp:n=1 $Al Tube
174   4   -1.0     1 92 -93               u=10   imp:n=1 $H2O
c Shim Control Rod
180   9   -2.52   -46 11 -93              u=11   imp:n=1 $B4C
181   6   -2.7     46 -47 11 -93          u=11   imp:n=1 $Al Cladding
182   4   -1.0    (47 -50 11 -93):(-50 -11 92)  u=11  imp:n=1 $H2O
183   6   -2.7     50 -1 92 -93           u=11   imp:n=1 $Al Tube
184   4   -1.0     1 92 -93               u=11   imp:n=1 $H2O
c Reg Control Rod
190   9   -2.52   -48 11 -93              u=12   imp:n=1 $B4C
191   6   -2.7     48 -49 11 -93          u=12   imp:n=1 $Al Cladding
192   4   -1.0    (49 -50 11 -93):(-50 -11 92)  u=12  imp:n=1 $H2O
193   6   -2.7     50 -1 92 -93           u=12   imp:n=1 $Al Tube
194   4   -1.0     1 92 -93               u=12   imp:n=1 $H2O
c Empty Control Rod
196   4   -1.0    -50 92 -93              u=5    imp:n=1 $H2O
```

```
197   6   -2.7      50 -1 92 -93           u=5   imp:n=1 $Al Tube
198   4   -1.0      1 92 -93               u=5   imp:n=1 $H2O
c Brand New SS Fuel, more U235
c 310 like 100 but mat=5  rho=-5.781         u=15   imp:n=1 $Fuel Meat
c 311 like 101 but                           u=15   imp:n=1 $Up Graphite
c 312 like 102 but                           u=15   imp:n=1 $Down Graphite
c 313 like 103 but                           u=15   imp:n=1 $Cladding
c 314 like 104 but                           u=15   imp:n=1 $H2O
c Lattice
200   4   -1.0   -101 102 -103 104 -105 106 92 -93    lat=2    u=9
                  fill=-7:7 -7:7 0:0
                  0 0 0 0 0 0 0 9 9 9 9 9 9 9 9
                   0 0 0 0 0 0 9 8 7 8 7 8 7 8 9
                    0 0 0 0 0 9 3 2 3 1 1 3 2 7 9
                     0 0 0 0 9 1 3 1 1 1 1 3 3 8 9
                      0 0 0 9 1 1 3 1 3 1 5 3 3 7 9
                       0 0 9 3 1 1 4 1 2 2 1 2 3 7 9
                        0 9 3 3 1 3 2 2 2 4 2 3 3 7 9
                         9 8 7 2 5 1 1 5 1 1 8 3 1 6 9
                          9 8 7 1 2 2 1 1 1 2 1 3 6 9 0
                           9 8 7 1 3 8 2 2 1 3 1 6 9 0 0
                            9 8 7 1 1 2 2 5 3 3 6 9 0 0 0
                             9 8 7 8 1 1 1 1 1 6 9 0 0 0 0
                              9 8 6 6 6 6 6 6 6 9 0 0 0 0 0
                               9 8 8 8 8 8 8 8 9 0 0 0 0 0 0
                                9 9 9 9 9 9 9 9 0 0 0 0 0 0 0 imp:n=1
201   4   -1.0   -111 112 -113 114 -115 116 92 -93  fill=9  imp:n=1 $Lattices
202   6   -2.7   (-121 122 -123 124 -125 126) 91 -94
                 (111:-112:113:-114:115:-116)              imp:n=1   $Al Wall
203   6   -2.7   -111 112 -113 114 -115 116 91 -92         imp:n=1   $Lower Al Plate
204   6   -2.7   -111 112 -113 114 -115 116 93 -94 41 43 45 imp:n=1   $Upper Al Plate
206   4   -1.0   -131 94 -97 41 43 45 imp:n=1                        $Top Water
207   4   -1.0   -131 96 -91                               imp:n=1   $Bottom Water
208   10  -2.30  -131 -96 95                               imp:n=1   $Bottom Concrete
301   9   -2.52  -40 93 -97       imp:n=1 $Safety Rod above core region
302   6   -2.7    40 -41 93 -97   imp:n=1
303   9   -2.52  -42 93 -97       imp:n=1 $Shim Rod above core region
304   6   -2.7    42 -43 93 -97   imp:n=1
305   9   -2.52  -44 93 -97       imp:n=1 $Reg Rod above core region
306   6   -2.7    44 -45 93 -97   imp:n=1
c FNIF
400   11  -0.00115   -141         imp:n=1 $ FNIF Air
401   8  -11.34  -140 141         imp:n=1 $ FNIF Pb
c Heavy water block
500   11  -0.00115   -159 160 -161 imp:n=1 $ Heavy water Air
501   6   -2.7   159 -158 160 -161 imp:n=1
502   7   -1.056 158 154 -155 156 157 160 -161  imp:n=1
503   6   -2.7   (-154:155:-156:-157)
                  150 -151 152 153 160 -161    imp:n=1
c
900   4   -1.0   -131 91 -94 140
                 (-150:151:-152:-153:-160:161)
                 (121:-122:123:-124:125:-126)   imp:n=1     $Water Around Core
999   0           131:-95:97   imp:n=0

C Surface Cards
1      cz      1.873  $Outer Radius
2      cz      1.82   $Inner Radius
3      cz      1.79   $Inner Radius for Aluminum Container
11     pz    -19.05   $SS Fuel Meat Bottom (7.5 inch * 2)
12     pz     19.05   $SS Fuel Meat Top
13     pz    -29.21   $SS Fuel Graphite Bottom (4 inch)
14     pz     29.21   $SS Fuel Graphite Top
```

```
15     pz    -30.39    $SS Cladding Bottom (1.18 cm)
16     pz     30.39    $SS Cladding Top (1.18 cm)
21     pz    -17.78    $Al Fuel Meat Bottom (7 inch * 2)
22     pz     17.78    $Al Fuel Meat Top
23     pz    -27.94    $Al Fuel Graphite Bottom (4 inch)
24     pz     27.94    $Al Fuel Graphite Top
25     pz    -29.12    $Al Cladding Bottom (1.18 cm)
26     pz     29.12    $Al Cladding Top (1.18 cm)
40     c/z     6.555 -11.354 1.00  $Safety Control Rod
41     c/z     6.555 -11.354 1.11  $Safety Control Rod Cladding
42     c/z   -13.11     0.0   1.00  $Shim Control Rod
43     c/z   -13.11     0.0   1.11  $Shim Control Rod Cladding
44     c/z     6.555  11.354 0.200 $Reg Control Rod
45     c/z     6.555  11.354 0.318 $Reg Control Rod Cladding
46     cz      1.00   $ Safety and Shim Rod in Unit
47     cz      1.11   $ Safety and Shim Rod in Unit Cladding
48     cz      0.200  $ Reg Rod in Unit
49     cz      0.318  $ Reg Rod in Unit Cladding
50     cz      1.750  $ Inner radius of Al tube for control rod
91     pz    -33.43    $Lower Plate Bottom
92     pz    -30.89    $Lower Plate Top (1 inch)
93     pz     30.89    $Upper Plate Bottom
94     pz     32.79    $Upper Plate Top (0.75 inch)
95     pz    -55.0     $Concrete Bottom
96     pz    -43.09    $Water Bottom (2 inch)
97     pz     50.0     $Water Top
C Lattice Cells
101    px     2.185
102    px    -2.185
103    p      0.5   0.8660254  0   2.185
104    p      0.5   0.8660254  0  -2.185
105    p     -0.5   0.8660254  0   2.185
106    p     -0.5   0.8660254  0  -2.185
c Frame Boundary
111    p      1.732038    1   0    50.460
112    p      1.732038    1   0   -50.460
113    p      1.732038   -1   0    50.460
114    p      1.732038   -1   0   -50.460
115    py   25.230
116    py  -25.230
c Al Wall
121    p      1.732038    1   0    54.270
122    p      1.732038    1   0   -54.270
123    p      1.732038   -1   0    54.270
124    p      1.732038   -1   0   -54.270
125    py   27.135
126    py  -27.135
C Reflector Surfaces
131    cz    65.0        $ Water reflector
c 131    p      1.732038    1    0  83.259682
c 132    p      1.732038    1    0 -83.259682
c 133    p      1.732038   -1    0  83.259682
c 134    p      1.732038   -1    0 -83.259682
c 135    py    41.629841
c 136    py   -41.629841
c FNIF
140    BOX -15.88 -26.77 -30.48   -15.24 26.40 0   -22.00 -12.7 0    0 0 60.96
141    BOX -22.82 -24.91 -30.48   -10.16 17.60 0    -8.80  -5.1 0    0 0 60.96
c Heavy water beside core
150    p      1.732038    1   0    54.270  $ Al outer
151    p      1.732038    1   0    84.670
152    py     0.0
153    p      1.732038   -1   0     0.0
```

```
154   p      1.732038    1   0    54.670  $ Al outer
155   p      1.732038    1   0    84.270
156   py     0.2
157   p      1.732038   -1   0     0.4
158   c/z    30.08 17.37 5.7  $ Air tub Al wall
159   c/z    30.08 17.37 5.5  $ Air tub
160   pz    -30.0            $ Heavy water top
161   pz     30.0            $ Heavy water bottom
201 pz -18.5
202 pz -17.5
203 pz -16.5
204 pz -15.5
205 pz -14.5
206 pz -13.5
207 pz -12.5
208 pz -11.5
209 pz -10.5
210 pz -9.5
211 pz -8.5
212 pz -7.5
213 pz -6.5
214 pz -5.5
215 pz -4.5
216 pz -3.5
217 pz -2.5
218 pz -1.5
219 pz -0.5
220 pz 0.5
221 pz 1.5
222 pz 2.5
223 pz 3.5
224 pz 4.5
225 pz 5.5
226 pz 6.5
227 pz 7.5
228 pz 8.5
229 pz 9.5
230 pz 10.5
231 pz 11.5
232 pz 12.5
233 pz 13.5
234 pz 14.5
235 pz 15.5
236 pz 16.5
237 pz 17.5
238 pz 18.5

mode n
kcode 100000 1.0 100 500
ksrc    -15.2950   -18.9227     0.0000
        -10.9250   -18.9227     0.0000
         -6.5550   -18.9227     0.0000
         -2.1850   -18.9227     0.0000
          2.1850   -18.9227     0.0000
          6.5550   -18.9227     0.0000
         10.9250   -18.9227     0.0000
        -17.4800   -15.1381     0.0000
        -13.1100   -15.1381     0.0000
         -8.7400   -15.1381     0.0000
         -4.3700   -15.1381     0.0000
          0.0000   -15.1381     0.0000
          4.3700   -15.1381     0.0000
          8.7400   -15.1381     0.0000
```

13.1100	-15.1381	0.0000
-19.6650	-11.3536	0.0000
-15.2950	-11.3536	0.0000
-10.9250	-11.3536	0.0000
-6.5550	-11.3536	0.0000
-2.1850	-11.3536	0.0000
2.1850	-11.3536	0.0000
10.9250	-11.3536	0.0000
15.2950	-11.3536	0.0000
-21.8500	-7.5691	0.0000
-17.4800	-7.5691	0.0000
-13.1100	-7.5691	0.0000
-8.7400	-7.5691	0.0000
-4.3700	-7.5691	0.0000
0.0000	-7.5691	0.0000
4.3700	-7.5691	0.0000
8.7400	-7.5691	0.0000
13.1100	-7.5691	0.0000
17.4800	-7.5691	0.0000
-24.0350	-3.7845	0.0000
-19.6650	-3.7845	0.0000
-15.2950	-3.7845	0.0000
-10.9250	-3.7845	0.0000
-6.5550	-3.7845	0.0000
-2.1850	-3.7845	0.0000
2.1850	-3.7845	0.0000
6.5550	-3.7845	0.0000
10.9250	-3.7845	0.0000
15.2950	-3.7845	0.0000
19.6650	-3.7845	0.0000
-17.4800	0.0000	0.0000
-8.7400	0.0000	0.0000
-4.3700	0.0000	0.0000
4.3700	0.0000	0.0000
8.7400	0.0000	0.0000
17.4800	0.0000	0.0000
21.8500	0.0000	0.0000
-15.2950	3.7845	0.0000
-10.9250	3.7845	0.0000
-6.5550	3.7845	0.0000
-2.1850	3.7845	0.0000
2.1850	3.7845	0.0000
6.5550	3.7845	0.0000
10.9250	3.7845	0.0000
15.2950	3.7845	0.0000
19.6650	3.7845	0.0000
-13.1100	7.5691	0.0000
-8.7400	7.5691	0.0000
0.0000	7.5691	0.0000
4.3700	7.5691	0.0000
8.7400	7.5691	0.0000
13.1100	7.5691	0.0000
17.4800	7.5691	0.0000
-10.9250	11.3536	0.0000
-6.5550	11.3536	0.0000
-2.1850	11.3536	0.0000
2.1850	11.3536	0.0000
10.9250	11.3536	0.0000
15.2950	11.3536	0.0000
-4.3700	15.1381	0.0000
0.0000	15.1381	0.0000
4.3700	15.1381	0.0000
8.7400	15.1381	0.0000

```
          13.1100    15.1381     0.0000
m1        1001.66c   -0.015896   $ new SS meat, H/Zr=1.6. 0.59% burn-up
         40000.66c   -0.899104
         92235.66c   -0.016728
         92238.66c   -0.068272
mt1       h/zr.60t
          zr/h.60t
m2        6000.66c    1.0        $ graphite
mt2       grph.60t
m3        6000.66c   -0.0004     $ ss cladding
         14000.60c   -0.0046
         24000.50c   -0.190
         25055.66c   -0.009
         26000.50c   -0.699
         28000.50c   -0.097
m4        1001.66c    2.0        $ H2O
          8016.66c    1.0
mt4       lwtr.60t
m5        1001.66c   -0.010   $ Al meat, H/Zr=1.0, 8.91% burnup
         40000.66c   -0.905
         92235.66c   -0.01533
         92238.66c   -0.06967
mt5       h/zr.60t
          zr/h.60t
m6       13027.66c    1.0        $ Al
m7        1001.66c    0.64       $ D20 (68% atom)
          1002.66c    1.36
          8016.66c    1.00
mt7       lwtr.60t
          hwtr.60t
m8       82000.50c    1.0        $ Pb
m9        5010.66c   -0.1566     $ b4c
          5011.66c   -0.6264
          6000.66c   -0.217
m10       1001.66c   -0.00619    $ Concrete
          6000.66c   -0.17520
          8016.66c   -0.41020
         11023.66c   -0.00027
         12000.66c   -0.03265
         13027.66c   -0.01083
         14000.60c   -0.03448
         19000.66c   -0.00114
         20000.66c   -0.32130
         26000.50c   -0.00778
m11       7014.66c    0.0000381259 $Air
          8016.66c    0.0000095012
         18000.59c    0.0000001664
m12       1001.66c   -0.015896   $ Old SS meat, H/Zr=1.6, 8.77% burnup
         40000.66c   -0.899104
         92235.66c   -0.015354
         92238.66c   -0.069646
mt12      h/zr.60t
          zr/h.60t
c
c New SS Fuel (u=1)
f17:n    (100<200[2  -5 0]<201)  (100<200[3  -5 0]<201)
         (100<200[-2 -4 0]<201)  (100<200[0  -4 0]<201)  (100<200[1 -4 0]<201)
         (100<200[2  -4 0]<201)  (100<200[3  -4 0]<201)
         (100<200[-3 -3 0]<201)  (100<200[-2 -3 0]<201)  (100<200[0 -3 0]<201)
         (100<200[2  -3 0]<201)
         (100<200[-3 -2 0]<201)  (100<200[-2 -2 0]<201)  (100<200[0 -2 0]<201)
         (100<200[3  -2 0]<201)
         (100<200[-3 -1 0]<201)
```

```
       (100<200[-2  0 0]<201)  (100<200[-1  0 0]<201)  (100<200[1  0 0]<201)
       (100<200[2   0 0]<201)  (100<200[5   0 0]<201)
       (100<200[-4  1 0]<201)  (100<200[-1  1 0]<201)  (100<200[0  1 0]<201)
       (100<200[1   1 0]<201)  (100<200[3   1 0]<201)
       (100<200[-4  2 0]<201)  (100<200[1   2 0]<201)  (100<200[3  2 0]<201)
       (100<200[-4  3 0]<201)  (100<200[-3  3 0]<201)
       (100<200[-3  4 0]<201)  (100<200[-2  4 0]<201)  (100<200[-1 4 0]<201)
       (100<200[0   4 0]<201)  (100<200[1   4 0]<201)
FS17 -201 -202 -203 -204 -205 -206 -207 -208 -209 -210 -211 -212 -213 -214
     -215 -216 -217 -218 -219 -220 -221 -222 -223 -224 -225 -226 -227 -228
     -229 -230 -231 -232 -233 -234 -235 -236 -237 -238
fq17 s
c
c Old SS Fuel (u=2)
f27:n  (110<200[0  -5 0]<201)  (110<200[5 -5 0]<201)
       (110<200[1  -2 0]<201)  (110<200[2 -2 0]<201)  (110<200[4 -2 0]<201)
       (110<200[-1 -1 0]<201)  (110<200[0 -1 0]<201)  (110<200[1 -1 0]<201)
       (110<200[3  -1 0]<201)
       (110<200[-4  0 0]<201)
       (110<200[-3  1 0]<201)  (110<200[-2 1 0]<201)  (110<200[2  1 0]<201)
       (110<200[-1  2 0]<201)  (110<200[0  2 0]<201)
       (110<200[-2  3 0]<201)  (110<200[-1 3 0]<201)
FS27 -201 -202 -203 -204 -205 -206 -207 -208 -209 -210 -211 -212 -213 -214
     -215 -216 -217 -218 -219 -220 -221 -222 -223 -224 -225 -226 -227 -228
     -229 -230 -231 -232 -233 -234 -235 -236 -237 -238
fq27 s
c
c Al Fuel (u=3)
f37:n  (120<200[-1 -5 0]<201)  (120<200[1 -5 0]<201)  (120<200[4 -5 0]<201)
       (120<200[-1 -4 0]<201)  (120<200[4 -4 0]<201)  (120<200[5 -4 0]<201)
       (120<200[-1 -3 0]<201)  (120<200[1 -3 0]<201)  (120<200[4 -3 0]<201)
       (120<200[5  -3 0]<201)
       (120<200[-4 -2 0]<201)  (120<200[5 -2 0]<201)
       (120<200[-5 -1 0]<201)  (120<200[-4 -1 0]<201) (120<200[-2 -1 0]<201)
       (120<200[4  -1 0]<201)  (120<200[5 -1 0]<201)
       (120<200[4   0 0]<201)
       (120<200[4   1 0]<201)
       (120<200[-3  2 0]<201)  (120<200[2  2 0]<201)
       (120<200[1   3 0]<201)  (120<200[2  3 0]<201)
FS37 -203 -204 -205 -206 -207 -208 -209 -210 -211 -212 -213 -214
     -215 -216 -217 -218 -219 -220 -221 -222 -223 -224 -225 -226 -227 -228
     -229 -230 -231 -232 -233 -234 -235 -236
fq37 s
c
c Instrumented Fuel (u=4)
f47:n  (130<200[-1 -2 0]<201)  (130<200[2 -1 0]<201)
FS47 -201 -202 -203 -204 -205 -206 -207 -208 -209 -210 -211 -212 -213 -214
     -215 -216 -217 -218 -219 -220 -221 -222 -223 -224 -225 -226 -227 -228
     -229 -230 -231 -232 -233 -234 -235 -236 -237 -238
fq47 s
c
f54:n  400    $ FNIF
E54 0 1E-9 5E-9 2.5E-8 1E-7 6.25E-7 2E-6 1E-5 1E-4 1E-3 0.01 0.1
      1 2 4 7 10 15 20
FS54  -217 -218 -219 -220 -221
fq54 s e
f64:n  500    $ TI
E64 0 1E-9 5E-9 2.5E-8 1E-7 6.25E-7 2E-6 1E-5 1E-4 1E-3 0.01 0.1
      1 2 4 7 10 15 20
FS64  -217 -218 -219 -220 -221
fq64 s e
c Mesh tally
FMESH84:n GEOM=rec ORIGIN=-40 -40 -40
```

```
          IMESH=40 IINTS=40
          JMESH=40 JINTS=40
          KMESH=40 KINTS=40
          EMESH=2.5E-8 6.25E-7 0.1 20 EINTS=1 1 1 1  OUT=ij
c CI
F94:n (196<200[0 0 0]<201)
E94 0 1E-9 5E-9 2.5E-8 1E-7 6.25E-7 2E-6 1E-5 1E-4 1E-3 0.01 0.1
      1 2 4 7 10 15 20
FS94  -217 -218 -219 -220 -221
fq94 s e
```

4.6 Thermal-Hydraulics Design

Very extensive thermal-hydraulic design studies and extensive actual performance tests have been done by General Atomics over the years on reactor cores utilizing TRIGA type fuel. This is well known [1-4] and is not repeated in this SAR but only the relevant results as they apply to the UUTR are presented.

Thermal-hydraulics modeling of the UUTR is based on the PARET-ANL code [5]. PARET (Program for the Analysis of REactor Transients) is primarily used for the design and analysis of thermal-hydraulics of the test and research reactors with pin and plate fuel types. The PARET-ANL code has been extensively compared to the SPERT I, and SPERT II experiments and has been validated [6-8]. A wide range of transient models has been analyzed using this code including the melting of the fuel cladding; the code has been used also for pulsing TRIGA reactors [5]. Additionally, PARET-ANL has been used by the PERTR Program for safety and thermal-hydraulics analysis of many research reactors redesigned for the reduced enrichment fuel.

4.6.1 PARET-ANL Thermal-Hydraulics Analysis of the 100kW UUTR

The UUTR thermal-hydraulics analysis is performed for the two-channel model: the hottest channel and the average channel representing the rest of the core. The hottest channel is located in the B-ring of the UUTR core, and is shown in **Fig. 4.6-1** by an X. The channel was divided into 19 axial regions; the peaking factors are shown in **Fig 4.6-2** for both the hottest channel and the average channel. These values are obtained from the UUTR neutronics calculations using MCNP5 (**UUTR SAR 4.5**). Thermal-hydraulics properties of the 100kW UUTR core are shown in **Table 4.6-1** in comparison to other TRIGA reactors. The PARET-ANL input data is listed in full in **Appendix 4.6.A**. In order to validate the results obtained from PARET-ANL, a set of measurements were performed in May of 2010 to obtain the average exit core temperature[3]. The measurement of the axial hottest channel temperature is not possible to obtain at the UUTR due to its core geometry. The comparison for average exit temperature for the moderator is shown in **Table 4.6-3**. A 5.6 % difference obtained between the measured and calculated value is in the range of measurement error. In the experiment the thermocouple was attached to an aluminum rod and placed at the top of the core. Due to the turbulences and the flow rate toward top of the core caused by the natural circulation of water through the core, the thermocouple was not stable at some times while the measurements have been taken. However, the data was acquired in each core rings (A through G) along each line from the center to each corner of the core as indicated in **Fig. 4.6-3**.

Table 4.6-2 shows the data points obtained for rings A through G for six diagonal measurement points. Dashed lines in the table indicate the locations where the measurements were not possible to obtain (difficulties in reaching these locations). The *"Ring Average"* refers

[3] All simulations correspond to the licensed power of 100kW; all measurements correspond to 90kW power.

to an average value for each fuel ring (A, B, C, and etc...). The entries in last column in Table 4.6-2 are the *"Ring Average"* values multiplied by the number of *"Elements"* in each ring. For example, since there are six (6) elements in the Ring B, the corresponding entry in the last column in Table 4.6.2 is 6 x 32.1=192.5. **Mean** in the last row of **Table 4.6-2** is defined as the sum of the entries in the last column divided by the total number of fuel elements in the core which yields the average channel outlet temperature over the whole core. Calculated centerline fuel temperature for the average and the hottest channels in 100kW UUTR are shown in **Fig. 4.6-4**; as expected the axial temperature profile follows the cosine distribution. Calculated axial moderator temperature profile is shown in **Fig. 4.6-5** indicating the expected trend.

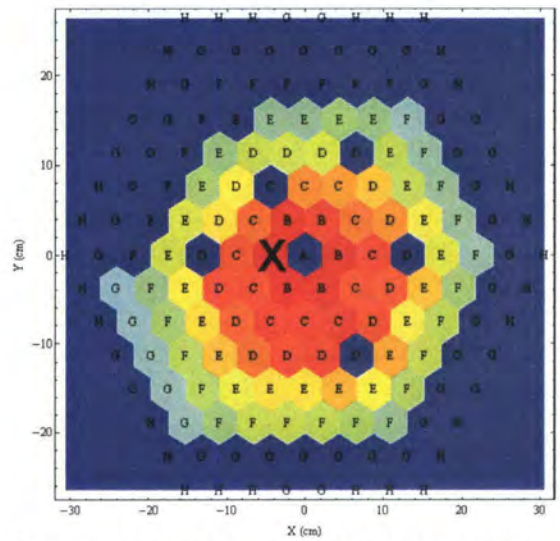


Figure 4.6-1 100kW UUTR core configuration indicating the location of the hottest channel (X)

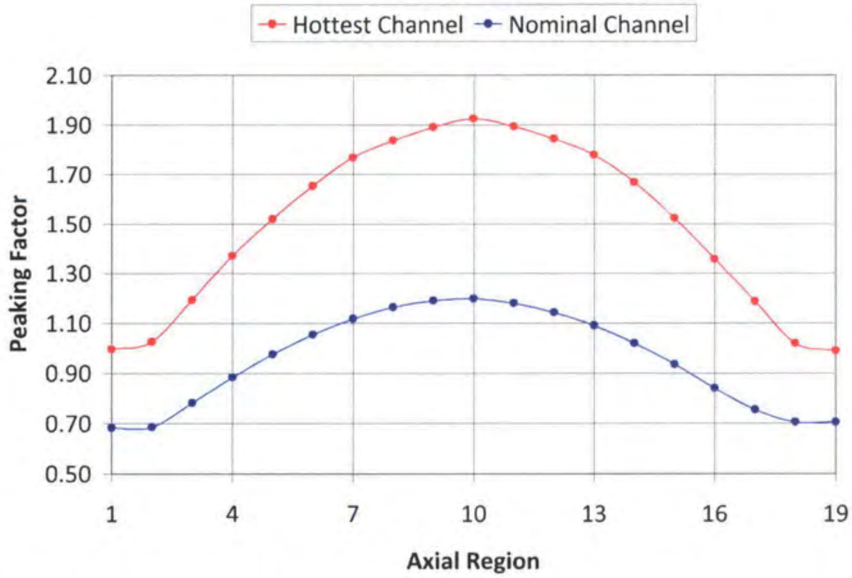


Figure 4.6-2 Axial peaking factors for hottest and the nominal channel of the 100kW UUTR core obtained from neutronics calculation using MCNP5

Table 4.6-1 Thermal-hydraulics properties of the 100kW UUTR in comparison to other TRIGA reactors [* indicates calculated values at 90kW UUTR]

Parameter	**UUTR**		**Standard TRIGA**	**WSU**
Lattice Type	Hexagonal		Hexagonal	7x9 Square 4-rod cluster
Power	90 kW	100 kW	1.5 MW	1 MW
Number of fuel elements	78		74	---
Diameter	[redacted]		[redacted]	[redacted]
Length	[redacted]		---	---
Flow area	0.0396 m^2		---	---
Inlet coolant temperature	20.60 °C (*293.75 °K)		15.6 °C (288.75 °K)	33.3 °C (306.45 °K)
Exit coolant temperature	21.95 °C (*295.10 °K)	25.3 °C (298.45 °K)	---	---
Coolant mass flow rate	*130 kg/s.m^2	115 kg/s.m^2	2794 kg/s.m^2	1204 kg/s.m^2
Average fuel temperature	70.93 °C (*344.08 °K)	86.70 °C (359.85 °K)	---	129.4 °C (402.55 °K)
Maximum fuel temperature (surface)	97.28 °C (*370.43 °K)	105.25 °C (378.40 °K)	---	---
Maximum fuel temperature (centerline)	121.70 °C (*394.85 °K)	129.67 °C (402.82 °K)	---	442.38 °C (715.53 °K)
Average heat flux	*24,272.63 W/m^2	38,144,26 W/m^2	---	---
Maximum heat flux	*49,054 W/m^2	49,057 W/m^2	891,760W/m^2	471,000W/m^2
Minimum DNB ratio Refer to **UUTR SAR 4.6.2**	*10.49	9.25	1.15	1.1

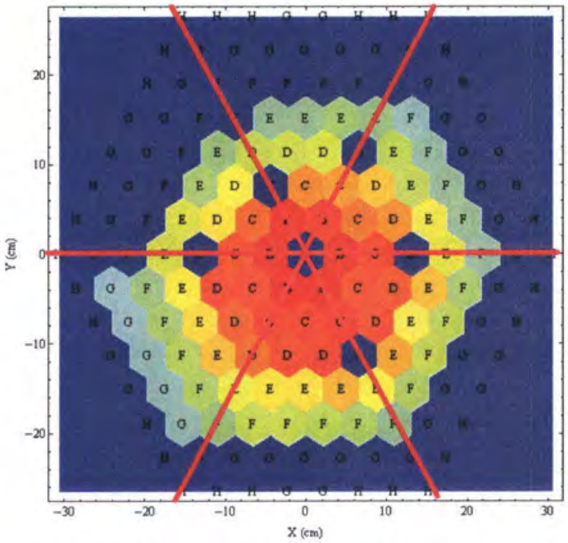


Figure 4.6-3 Locations of the moderator temperature measurements took on the top of the UUTR core operated at 90kW

Table 4.6-2 Moderator exit temperature measured for UUTR operated at 90kW (Refer to Fig. 4.6-3); calculated average exit temperature for 100kW is 21.95°C (295.10 °K) (Table 4.6-1) and for 90kW is 21.75°C (294.90 °K) (measured for 90kW is23.04°C (296.19 °K), Table 4.6-3)

Location	Temperature, °C (°K)						Elements	Ring Average, °C (°K)	Element * Ring Average, °C (°K)
	1	2	3	4	5	6			
A	30.6 (303.75)	---	---	---	---	---	1	30.6 (303.75)	30.6*(303.75)
B	32.0 (305.15)	31.1 (304.25)	33.1 (306.25)	32.1 (305.25)	---	---	6	32.1 (305.25)	192. 6 (465.75)
C	27.5 (300.65)	29.8 (302.95)	31.5 (304.65)	30.3 (303.45)	---	---	12	29.8 (302.95)	357.6 (630.75)
D	21.4 (294.55)	---	27.1 (300.25)	27.8 (300.95)	---	---	18	25.4 (298.95)	457.2 (730.35)
E	20.5 (293.65)	23.7 (296.85)	21.8 (294.95)	21.9 (295.05)	---	---	24	22.0 (295.15)	528.0 (801.15)
F	20.5 (293.65)	20.9 (294.05)	20.5 (293.65)	21.2 (294.35)	20.7 (293.85)	---	30	20.8 (293.95)	624.0 (897.15)
G	20.4 (293.55)	20.5 (293.65)	20.5 (293.65)	20.5 (293.65)	20.6 (293.75)	---	36	20.5 (293.65)	738.0 (1011.15)
								Mean	**23.1 (296.25)**

*This column gives the number of elements multiplied by the local average of each fuel ring.

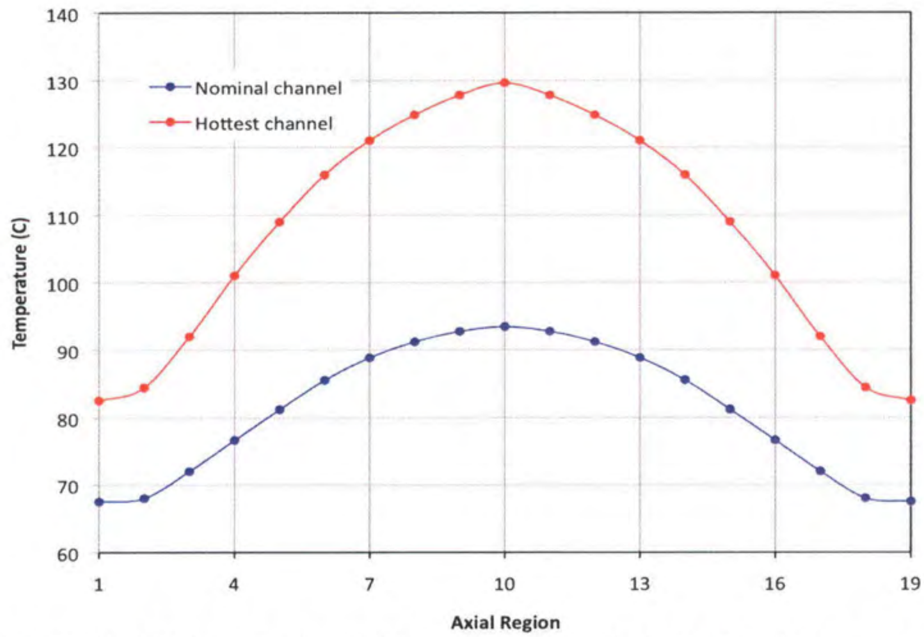


Figure 4.6-4 100kW UUTR centerline fuel temperature for the hottest and the nominal channel calculated using the PARET-ANL code

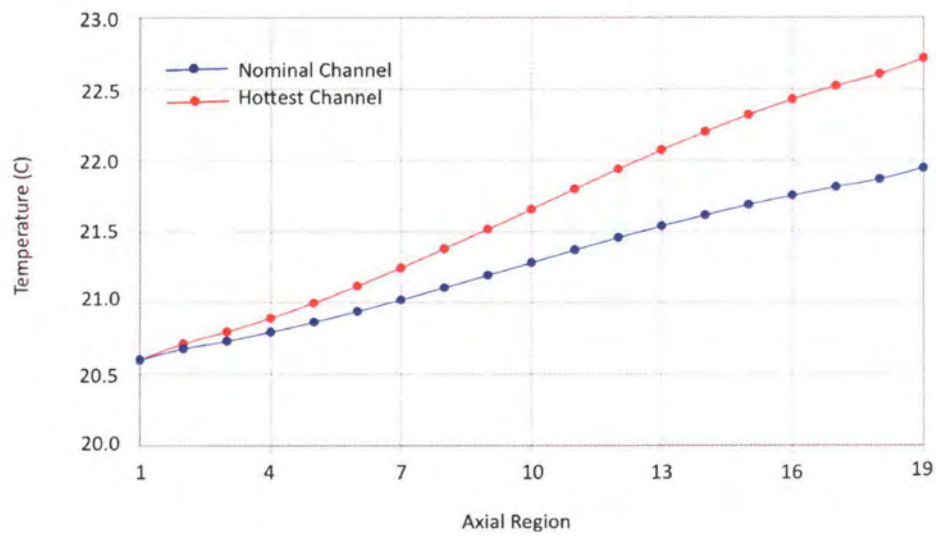


Figure 4.6-5 100kW UUTR axial moderator temperature profile obtained with the PARET-ANL

Table 4.6-3 Comparison of measured and calculated values for average moderator temperature in the UUTR operated at 90kW and 100kW

Average Temperature	Measured at 90kW	Calculated at 90kW	Difference (%)	Calculated at 100kW
°C (°K)	23.1 (296.25)	21.75 (294.9)	5.66	21.81 (294.96)

The fuel centerline temperature of the fuel rods in B-ring was calculated using PARET. **Figure 4.6-6** shows the temperature change vs UUTR power, from 10 kW to 110 kW. At 110 kW, the maximum fuel centerline temperature is estimated to a value of 136.43 °C (409.58 °K), which is far below the fuel temperature set point (200 °C).

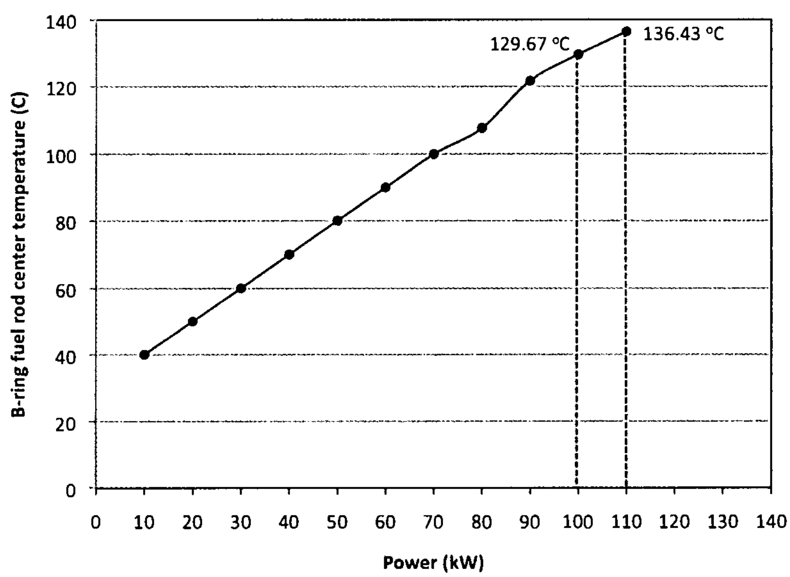


Figure 4.6-6 Maximum fuel centerline temperature vs UUTR power

4.6.2 Calculations to Determine the Departure from Nucleate Boiling Ratio (DNBR)

In fully developed nucleate boiling regime, the heat flux can be increased without significant increase in the surface temperature of the fuel element (cladding) up until the point of departure from nucleate boiling (DNB). In the subcooled boiling regime, the critical heat flux is a function of the following parameters:

- Coolant velocity
- Degree of subcooling
- Pressure

Bernath's correlation is specifically used for DNBR calculations of TRIGA reactors by General Atomics and Argonne National Laboratory [10]. Several correlations as described relate to TRIGA critical heat flux calculations as follows: Bernath's correlation, Groeneveld table look-up, and, Hall and Mudawar correlation. Bernath's correlation, for an 8 mm diameter, predicts the lowest CHF which implies that it is relatively a conservative correlation compared to others; the relations are depicted in **Fig. 4.6-7**. (Purdue correlation in this figure refers to Hall and Mudawar correlation for subcooled water flowing in a tube that is heated from the outside. The data for this correlation are proprietary and not available to the public as of August 2010). Traditionally, this relation has been used in STAT code (developed by General Atomics) for calculation of DNBR in TRIGA reactors. Due to the lack of measured data in the CHF regions of TRIGA reactors, none of the DNBR correlation can predict an accurate and/or definite value for the CHF. Therefore a conservative correlation must be selected such as Bernath correlation. The CHF data for four different TRIGA reactors are presented in **Table 4.6-4** indicating that the CHF calculated based on Bernath's correlation gives more conservative value compared to Groeneveld's correlation.

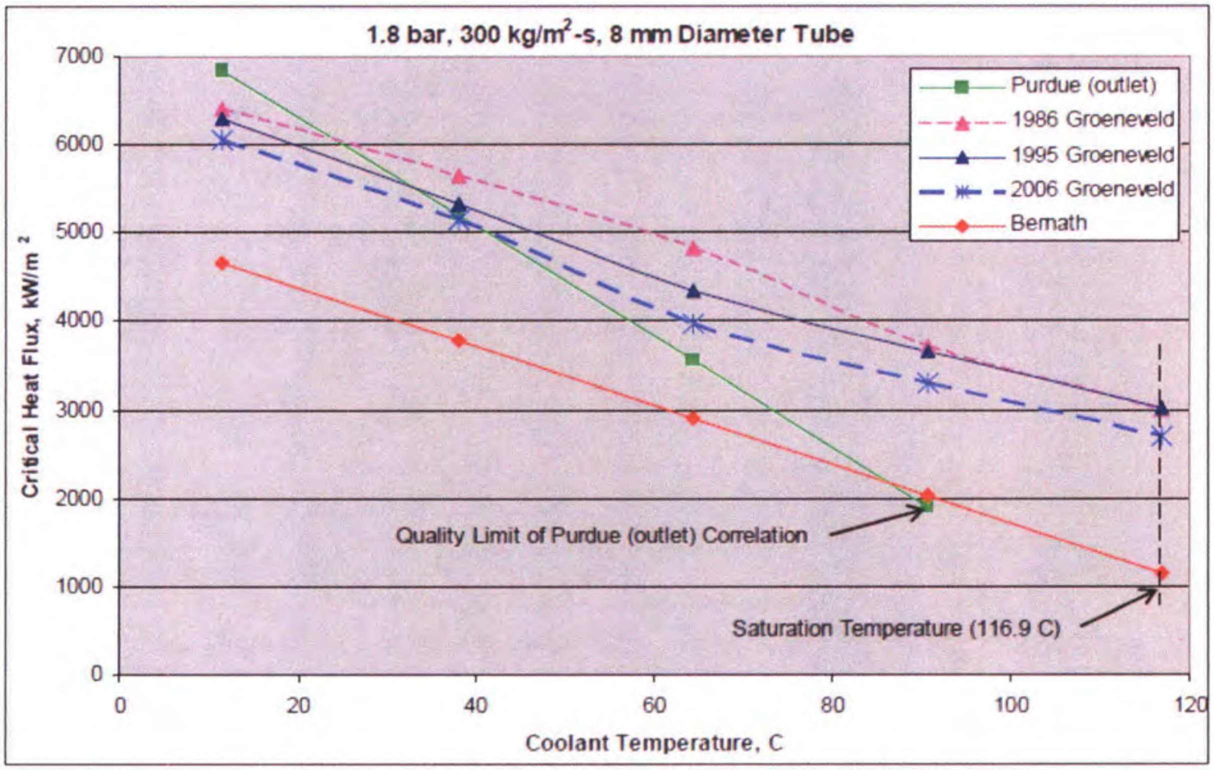


Figure 4.6-7 CHF calculation for 8mm diameter pipe for various inlet temperatures (1.8 bar pressure, 300 kg/m^2-s flow rate) [10]

Bernath's correlation is based on critical wall superheat condition at burnout and turbulent mixing convective heat transfer. Bernath's correlation gives the most conservative prediction for CHF among other correlations. Bernath's correlation is given as follows:

Table 4.6-4 CHF calculation for several TRIGA reactors [10]

Reactor	Bernath CHF, kW	2006 Groeneveld CHF, kW	
		Method A[4]	Method B[2]
Washington State University	47.6	61.2	64.8
Texas A&M University	49.4	62.4	71.8
Oregon State University	40.4	65.7	80.4
U.C. Davis McClellan Nuclear	56.5	70.5	71.0
Radiation Center	55.3	68.3	

$$CHF = h_{BO}\left(T_{WBO} - T_b\right)$$

$$h_{BO} = 10890\left(\frac{D_e}{D_e + D_i}\right) + \left[\text{SLOPE}\right]\cdot u$$

$$\begin{cases} \text{SLOPE} = \dfrac{48}{D_e^{0.6}} & D_e \le 0.1\,ft \\ \text{SLOPE} = 90 + \dfrac{10}{D_e} & D_e > 0.1\,ft \end{cases}$$

$$T_{WBO} = 57\ln P - 54\left(\frac{P}{P+15}\right) - \frac{u}{4}$$

Table 4.6-5 Parameters in Bernath'c Correlations

Quantity	Parameter	Unit
CHF	Critical Heat Flux (CHF)	$\frac{p.c.u^{*}}{hr \cdot ft^2}$
h_{BO}	Film coefficient at CHF	$\frac{p.c.u}{hr \cdot ft^2 \cdot {}^\circ C}$
T_{WBO}	Wall temperature at CHF	$^\circ C$
T_b	Bulk temperature	$^\circ C$
D_e	Hydraulic diameter	ft
D_i	Diameter of the heated surface	ft
P	Pressure	*psia*
u	Coolant Velocity	$\frac{ft}{s}$

*pound centigrade units (1 p.c.u. = 1.8 Btu)

[4] Method A, which provides a lower value of CHF power, evaluates the 2006 Groeneveld curve at the highest non-oscillatory flow calculated by RELAP5 and is the recommended method. Method B, which is not recommended, is based on the extrapolated RELAP5/MOD3.2 flow curves [11].

For the 100kW UUTR (78 fuel elements in the core), power per pin is shown in **Fig. 4.5.2-3 (c),** which is obtained from MCNP5 calculations. The pin with the highest power is located in the B-ring with 2.022kW power per pin. The most conservative geometry for a sub-channel analysis is shown in **Fig. 4.6-8**. Thermal-hydraulics parameters of the sub-channel are shown in **Table 4.6-6**.

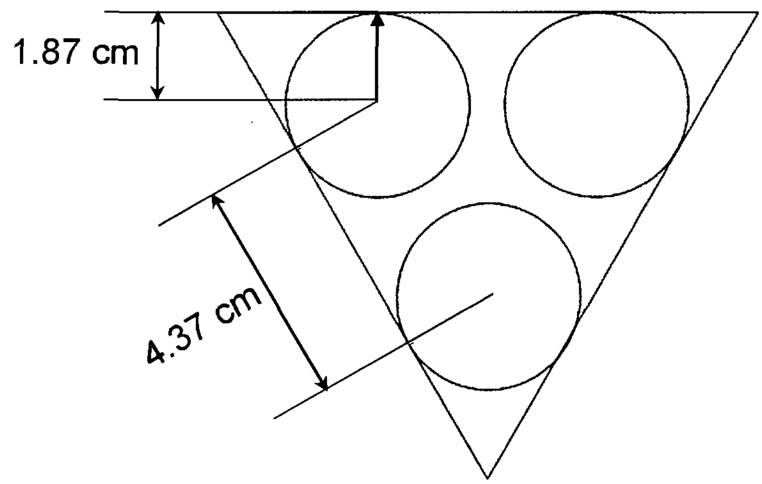


Figure 4.6-8 Sub-channel geometry with dimensions in cm

Table 4.6-6 Thermal-hydraulics parameters of the sub-channel

Parameter	Value	Unit
Area	8.0326	cm^2
Wetted Perimeter	64.4371	cm
Heated Perimeter	35.2487	cm
Hydraulic Diameter	0.4986	cm

The DNBR and CHF are shown in **Fig. 4.6-9** and **Fig. 4.6-10** respectively for various inlet coolant temperatures based on the expected inlet coolant temperatures for UUTR (described in **UUTR SAR 4.6.3**). The surface heat fluxes are obtained from the PARET-ANL calculations.

From **Figure 4.6-9** we can conclude: the **DNBR stays above 8.0** for various coolant inlet temperatures in the hottest channel. This DNBR value represents a safe region for the reactor operation in terms of fuel and clad integrity. DNBR of 8.0 means that the heat flux at the surface of the cladding to the coolant is 8.0 times lower than the CHF where the film boiling will occur. Based on the studies performed by General Atomics, the TRIGA reactors can safely operate until approximately 30kW/element, which assures that DNBR is above 1.0. The calculations are carried out for up to the pool temperature of ~50 °C (323.15 °K). Our limitations require that the UUTR reactor can be operated only for the pool temperature of up to 35 °C (308.15 °K). The calculations for the pool temperature (inlet coolant temperature) of 90 °C (363.15 °K) gives DNBR=5.11 which is high enough value for the safe operation of the

UUTR. This verifies that within the valid pool water temperatures at the atmospheric pressure, the DNBR will always stay in a reasonable range to prevent cladding damage due to film boiling.

Comparison between the Standard TRIGA, UUTR and WSU reactor are shown in **Table 4.6-1**; the hottest pin in the UUTR has 2.022kW of power while the average power per pin in the Standard TRIGA reactor is 20.3kW (1.5MW/74 rods). The ratio of the UUTR DNBR (at 20.6 °C (293.75 °K)) inlet coolant temperature the DNBR=10.49) to the Standard TRIGA reactor (DNBR=1.15) is approximately 10. This clearly verifies that the presented calculations for the UUTR are valid when compared to the Standard TRIGA reactors.

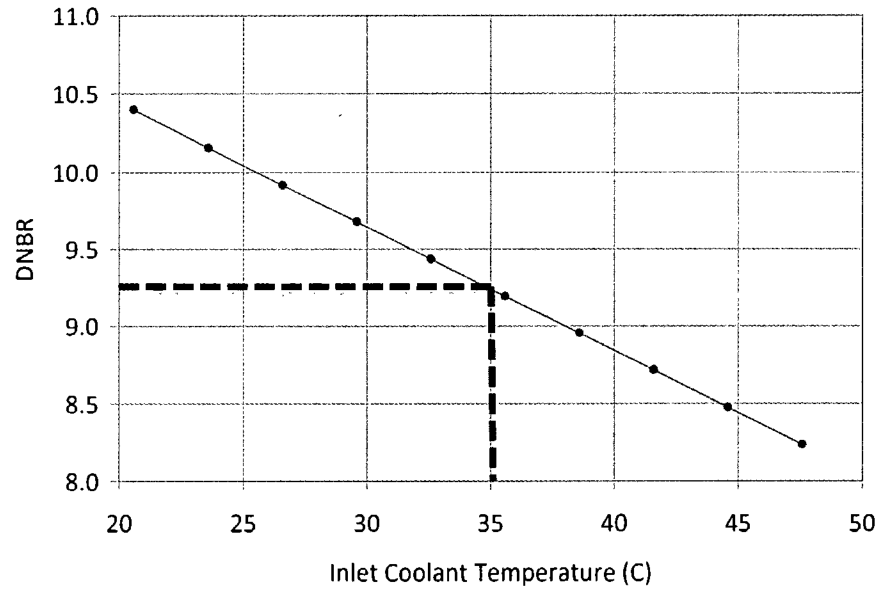


Figure 4.6-9 DNBR calculated using Bernath's correlation as a function of coolant inlet temperature

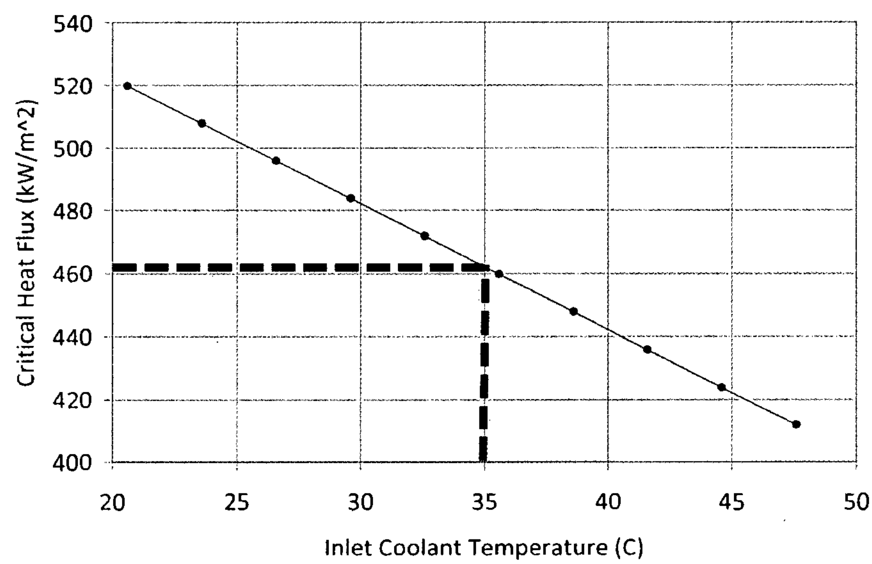


Figure 4.6-10 CHF calculated using Bernath's correlation as a function of coolant inlet temperature

Another important factor for the safe operation of the TRIGA reactors is the flow oscillations prior to CHF. This phenomenon is defined as the flow rate oscillation (increase and decrease of coolant flow rate in a cyclic fashion) in subcooled boiling. However, the tests done by GA [11] concluded that for a hexagonal lattice TRIGA core with 100 fuel elements of up to 2MW power the reactor operates with acceptable power stability. Therefore, for the UUTR at 100kW chugging phenomena is not expected to occur. During the excess reactivity insertion of \$1.2 (see **UUTR SAR 13**) the maximum power will reach 700kW (prompt jump power in a fraction of a second), which is a way below the limits obtained by GA [11]. Therefore chugging is not expected to occur under any operating or accident condition of the UUTR.

Another parameter of interest is the coolant velocity in the channel. CHF is plotted in **Fig. 4.6-11** for various coolant temperatures in the channel from 0.1 ft/s up to 3.0 ft/s. This is important to mention that the mass flow rate used in the hottest channel is the average coolant flow rate in the core. This is a very conservative assumption because the lower the mass flow-rate the higher the CHF. However, in order to verify this and determine a valid CHF region for the hottest channel, it is important to analyze the trend of the CHF versus the coolant mass flow rate. Considering energy balance in the hottest channel, the mass flow rate is approximately 223 kg/m^2-s, which is equivalent to 0.7 ft/s. Therefore, the DNBR calculation for the hottest channel is rather conservative.

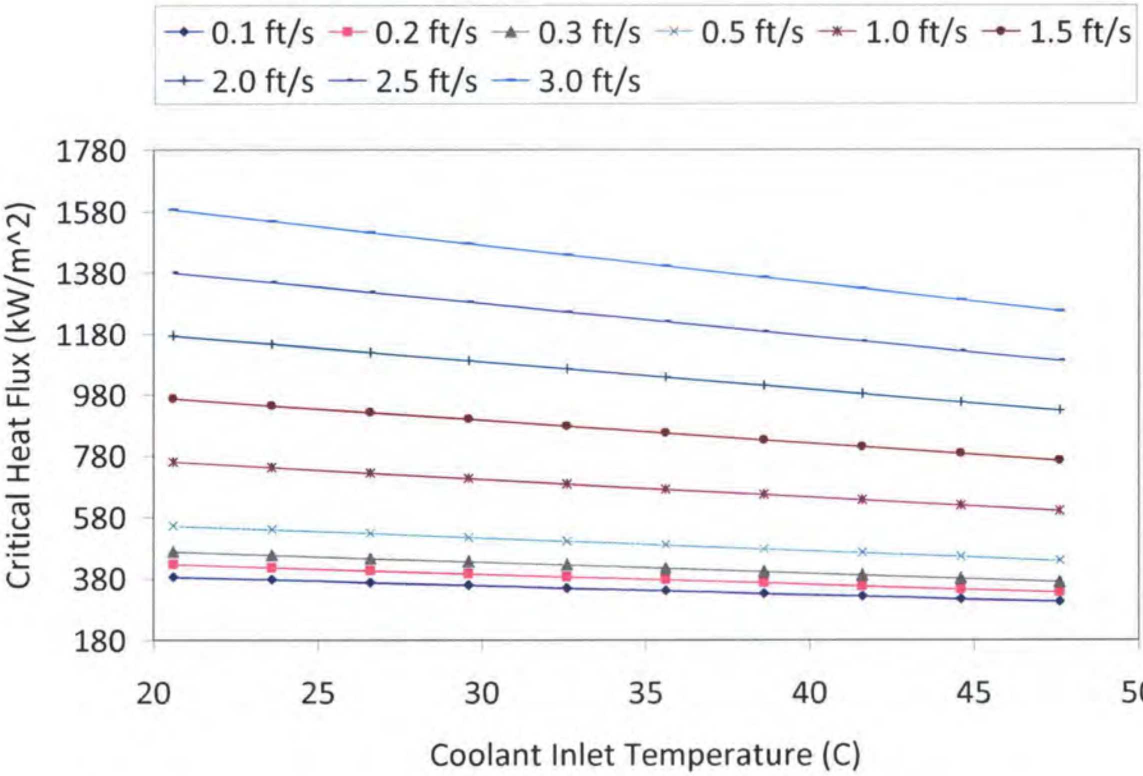


Figure 4.6-11 CHF calculated using Bernath's correlation as a function of coolant inlet temperature for various coolant velocities

4.6.3 Pool Water Temperature of the 100kW UUTR Core

Knowing that there are several mechanisms for the heat to be removed from the pool water such as the heat loss due to evaporation of water from surface, due to radiation, due to conduction of heat through walls, the average pool water temperature can be estimated using the following relation:

$$Q_{core} - Q^{*}_{evaporation} - Q^{*}_{wall} - Q^{*}_{radiation} = Q_{total}$$

The only heat source in the pool is the reactor core. The temperature change in the pool water depends on the heat generated in the medium:

$$Q = mc_p \Delta T$$

where:

m: mass of water in tank

c_p: specific heat capacity of water

The heat generated by the UUTR while operating at 90kW is:

$$\dot{Q} = 90{,}000 \frac{J}{s}$$

Therefore, the temperature change for X amount of time of the operation of the reactor is obtained as follows:

$$\Delta T = \frac{\dot{Q}X}{mc_p} = \frac{\left[\frac{J}{s}\right][s]}{[kg]\left[\frac{J}{kg \cdot K}\right]} = [K]$$

where:

ΔT: change in temperature

X: time during which UUTR is operated

$\dot{Q}$: reactor power

Heat loss rate through wall, radiation, and evaporation for open water tanks were obtained from data published for various water temperatures [EngineeringToolBox: http://www.engineeringtoolbox.com/heat-loss-open-water-tanks-d_286.html]. The modified equation is then:

$$\Delta T(\theta)=\frac{\left(\dot{Q}-\dot{Q}_{evaporation}(\theta)-\dot{Q}_{walls}(\theta)-\dot{Q}_{radiation}(\theta)\right)\cdot X}{mc_p}$$

where θ is the temperature of the water in the pool tank (heat losses are function of temperature inside the tank).

It follows that the heat losses through evaporations, walls, and radiation are temperature dependent. Therefore, the appropriate heat losses are selected based on the range of temperature. Using above correlation and amount of heat required heating up the water per hour, the relation between the pool (average) water temperature, core power and time of operation is shown in **Fig. 4.6-12**. The additional data are listed in **Table 4.6-7**.

Table 4.6-7 UUTR reactor tank dimensions

Radius	1.2192 m
Height	7.62 m
Volume	35.584 m^3
Inner Surface Area	58.3727 m^2
Top Surface Area	4.67 m^2
Mass of Water	30,584 Kg

This model is validated using measured data when the UUTR operated at 90kW. In the model as described, the steady state calculations were performed per each time step, assuming homogeneous temperature distribution across the pool tank. The measurements are obtained at four data points and averaged to obtain the pool water average temperature at each hour of UUTR operation at 90kW. This introduces an error because the heat source (a reactor) is constantly heating the pool water and natural circulation causes the mixing across the pool tank. Another source of error is associated with the instruments. In addition, the water level in the tank is not constant; in the theoretical model, the level of water in a tank was assumed to be constant. Measured and calculated values for pool average temperature when the UUTR operates at 90kW are shown in **Table 4.6-8**. The temperatures in the tank are measured at four positions as shown in **Fig. 4.6-13**. The average value of the tank temperature is not exactly equal to the average of the measurements obtained at these four locations; however, they provide a good estimate for the average temperature of the pool water. The percentage difference is also listed in **Table 4.6-8** between the calculated and measured values. In theoretical model the temperature losses in tank wall and loses due to evaporation are not explicitly taken into account; as temperature rises in pool tank, the rate of heat losses from walls and surface is changed, therefore the model fails to predict those values explicitly and the errors are expected to be larger for higher temperatures. However, within the range of operation of UUTR, the highest error obtained is 3.65% considered to be within an accepted range.

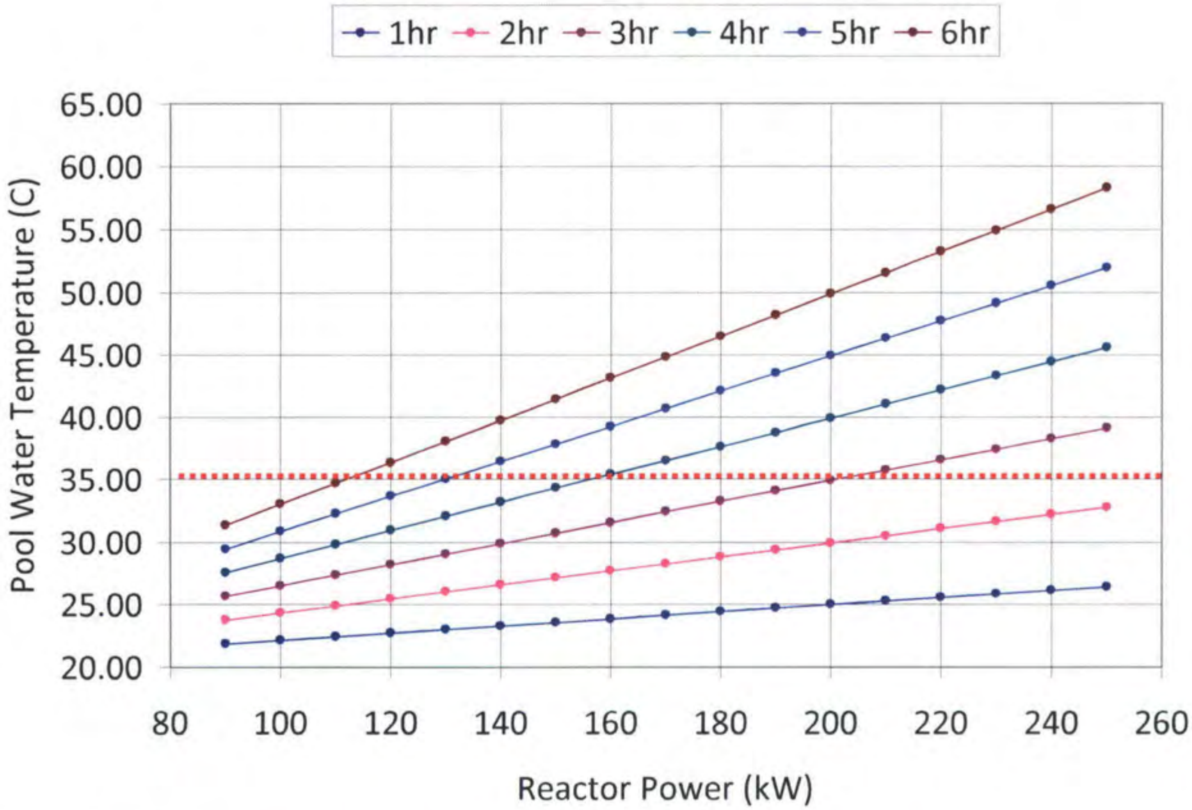


Figure 4.6-12 UUTR pool water temperature as a function of reactor power and hours of operation (dashed line indicates the UUTR TS limit)

Table 4.6-8 Comparison of measured and calculated pool water average temperature for 90kW UUTR [refer to Fig. 4.6-12]

Hour	Temperature (°C)						Difference (%)
	Measured at Location 1	Measured at Location 2	Measured at Location 3	Measured at Location 4	Average measured	Average calculated	
9:33am	20.3	20.4	20.3	20.3	20.3	20.3	-0.12
10:00am	21.7	21.8	21.5	20.4	21.4	21.2	-0.67
10:30am	23.0	23.0	22.8	21.9	22.7	22.2	-2.03
11:00am	24.3	24.2	24.1	23.1	23.9	23.2	-2.93
11:30am	25.4	25.4	25.3	24.5	25.2	24.2	-3.65

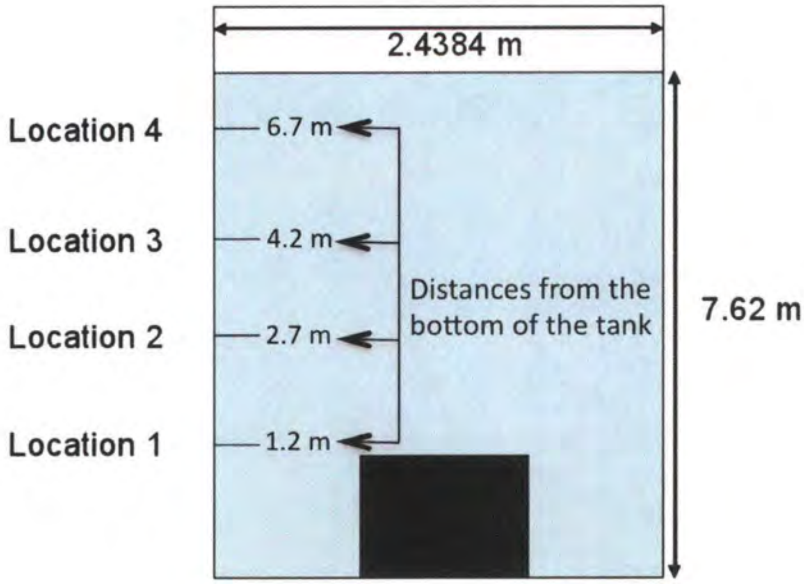


Figure 4.6-13 Locations of four thermocouples inside the UUTR reactor pool tank [not to scale; reactor indicated as a black box at the pool center]

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Appendix 4.6.A

PARET-ANL input file for the 100kW UUTR core

```
  0   200
*PARET:  The University of Utah TRIGA Reactor: 100kW (Steady State)
1001,     -2    19      7      1      0      1
1002,      0     0      6     -1      1     20
1003,    1.0-1        0.032540    1.66339+5 -18.00        1.79200-2
1004,    1.74110-2    1.74200-2   0.0         0.0         0.3810      0.1016
1005,    0.1016       0.0078     27.900-6    9.80664     0.00679
1006     0.00          0.80        1.0        998.63      -0.47296
1007,   -2.02010-3    1.15817-5   0.00        0.00        1.00         0.001
1008,    0.00         0.001       0.001       0.05        0.05         0.05
1009,    1.4          0.33
1111,    0.039556     1.00         1.00
1112,      0     1      1      6      0      0
1113,    3.81          0.2       10000.0      0.00
1114,   6.7056        0.6096
2001,    0.0          0.0        18.00        0.00        0.00
2002,    0.0          0.4170+4    2.0400+6    0.00       -273.0
2003,    0.0          0.0         0.199000    0.00        0.00
2004,    0.0          0.0         6.66340+2   0.00        0.00
2005,    0.0          0.0        16.8         0.00        0.00
2006,    0.0          0.0         3.975+6     0.00        0.00
3001,   4.35275-3       5      1  0.980
3002,        9.0-6      6      2  0.00
3003,      5.00-4       7      3  0.000
4001,    0.020053      19
5100,      1     0  0.02794       0.007874    0.5         0.55        1.0
5100,    1.00        0.00         0.00
5101,    0.6096      6.7056       4.38511-2   4.38511-2  1.3818       1.250
5102     1.0069       1.00        1.00        1.00
5103,    1.0395       1.00        1.00        1.00
5104,    1.2090       1.00        1.00        1.00
5105,    1.3770       1.00        1.00        1.00
5106,    1.5462       1.00        1.00        1.00
5107,    1.6773       1.00        1.00        1.00
5108,    1.7856       1.00        1.00        1.00
5109,    1.8708       1.00        1.00        1.00
5110,    1.9144       1.00        1.00        1.00
5111,    1.9346       1.00        1.00        1.00
5112,    1.9245       1.00        1.00        1.00
5113,    1.8686       1.00        1.00        1.00
5114,    1.7884       1.00        1.00        1.00
5115,    1.6728       1.00        1.00        1.00
5116,    1.5274       1.00        1.00        1.00
5117,    1.3727       1.00        1.00        1.00
5118,    1.2035       1.00        1.00        1.00
5119,    1.0244       1.00        1.00        1.00
5120,    0.9919       1.00        1.00        1.00
5200,      1     0  0.02794       0.992126     0.5        0.55         1.00
5200,    1.00        0.0          0.0
5201,    0.6096      6.7056       4.38511-2   4.38511-2 1.3818        1.250
5202,    0.6890       1.00        1.00        1.00
5203,    0.6928       1.00        1.00        1.00
5204,    0.7896       1.00        1.00        1.00
5205,    0.8920       1.00        1.00        1.00
5206,    0.9846       1.00        1.00        1.00
5207,    1.0647       1.00        1.00        1.00
5208,    1.1285       1.00        1.00        1.00
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5209,    1.1750    1.00    1.00    1.00
5210,    1.2012    1.00    1.00    1.00
5211,    1.2101    1.00    1.00    1.00
5212,    1.1913    1.00    1.00    1.00
5213,    1.1542    1.00    1.00    1.00
5214,    1.1018    1.00    1.00    1.00
5215,    1.0302    1.00    1.00    1.00
5216,    0.9449    1.00    1.00    1.00
5217,    0.8483    1.00    1.00    1.00
5218,    0.7622    1.00    1.00    1.00
5219,    0.7123    1.00    1.00    1.00
5220,    0.7111    1.00    1.00    1.00
9000,      2
9001,    1.0-1     0.000   1.0-1     0.15
10000,     2
10001,   114.00    0.00    114.00    1000.00
11000,     2
11001,   0.0       10.00   0.5       1000.0
12000,     2
12001,   5978.13   0.0     5978.13   1000.0
14000,     6
14001,   0.001     0.0     0.0001    0.20    0.00005   0.25
14002,   0.0005    0.28    0.001     1.00    0.005     1.50
16000,     6
16001,   0.1       50      0.0       0.02    10        0.20
16002,   0.005     1       0.25      0.10    10        0.28
16003,   0.50      25      1.00      1.00    10        1.50
17000,     2
17001,   1.0       0.0     1.00000   455.0
18000,     2
18001,   0.0       0.0     -4.00     0.381
```
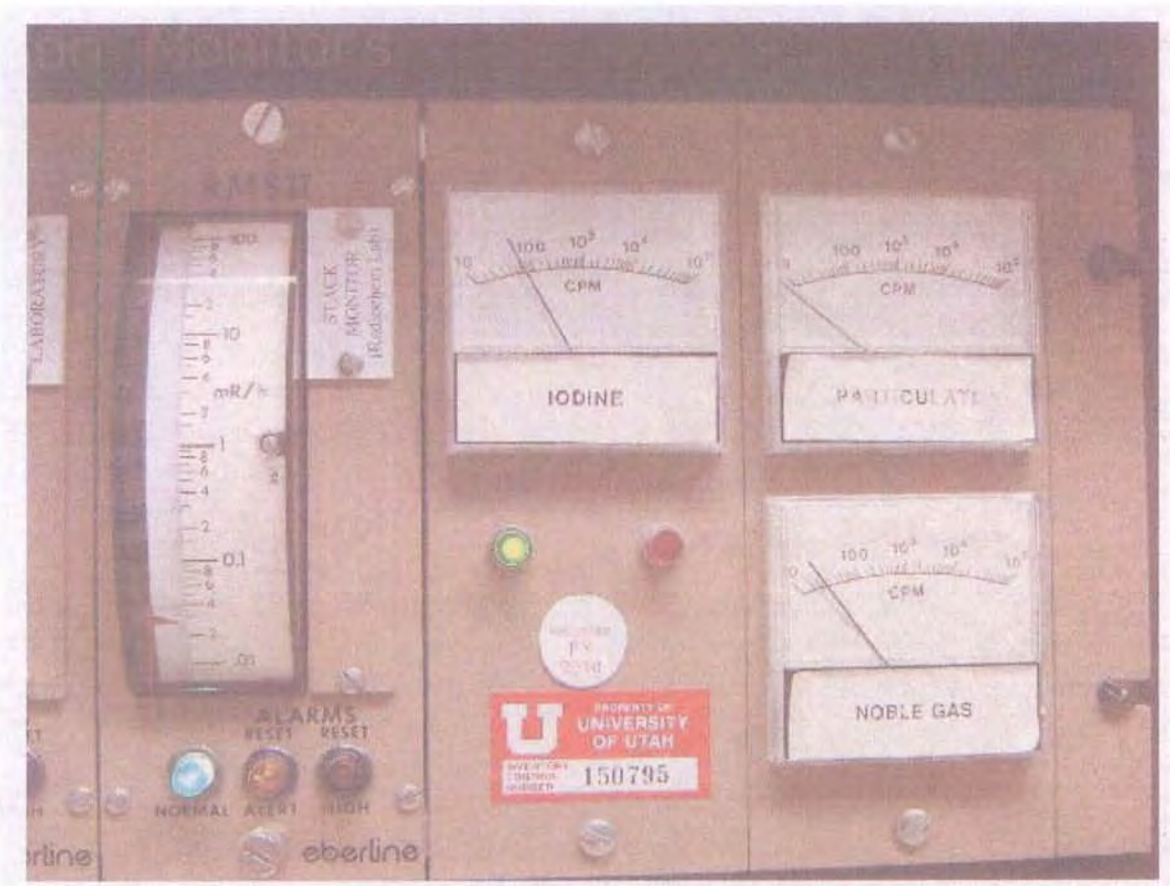

5. REACTOR COOLANT SYSTEMS



5.1 Summary Description

The UUTR is a natural convection water-cooled pool type reactor. Since the UUTR is small size and low power (100kW), the primary coolant system is not a necessary safety system of the facility, but is used for maintaining its efficient operation. The water in the reactor pool is used to moderate the reactor, to cool the fuel elements during reactor operation, and shield against the radiation coming from the operating reactor core. Therefore, the primary cooling system is used to remove the heat generated during operation, remove any particulate and soluble impurities, maintain low conductivity and pH in the water, maintain optical clarity of the tank water, and shielding radiation generated in the core. In the case of loss of water from the pool, the design analysis of TRIGA fuel shows that it may be cooled by natural convection in air without risk of fuel failure.

The reactor pool is open to the atmosphere, while the secondary coolant system is a R134A based shell and tube heat exchanger. This secondary coolant system can dissipate 25kW of heat; the system is located near the reactor pool. The heat exchanger receives warm water from the top of the pool and returns cooler water to the pool. The return line carries the water to a location several ft above the core where it is ejected into the pool water as a horizontal water jet in order to diffuse the Nitrogen-16 activity and prevent it from reaching the top of the pool as fast as it would otherwise (**Fig. 5.1-1**). The discharge temperature is about 10 ^{o}C (283.15 ^{o}K). This discharge temperature can be read from the reactor console. Both the inlet and return lines to the pool have a small 0.635 cm hole approximately 30 cm below the normal pool water level. These holes will stop siphoning action and prevent draining of the pool below that level should a siphoning drain be accidentally initiated. The tank water temperature increases approximately 3 ^{o}K /hour at 90kW power. The UUTR reactor core is cooled by natural convection alone. In the case of extended operation, the 25kW capacity of secondary cooling system without cooling tower cannot remove the heat efficiently. To protect the resin bed purifying system, the UUTR has administrative pool water temperature limit of 35 ^{o}C (308.15 ^{o}K).

The heat energy from the reactor operation is removed from the core by natural convection. However, after the reactor is shut down, the heat exchanger, shown in **Fig. 5.1-3**, can lower the pool temperature at approximately 0.5 ^{o}K /hour. The 25kW capacity heat exchanger will absorb the tank water heat. As shown in **Figures 5.1-2** and **5.1-4**, the absorbed heat is cooled down using portable water supply and this water is released to the sanitary system. The UUTR tank contains approximately 8,000 gallons of high-purity water and the core and fuel are clearly visible from the top (**Fig. 5.1-1**). The 22 ft of water over the top of the reactor core provides biological shielding for personnel in the reactor room. The reactor containment uses double walled construction with an aluminum inner tank filled with reactor coolant water surrounded by two ft of sand with a steel outer wall. This massive structure is imbedded more than 12 ft into the ground and rests on a concrete pad.

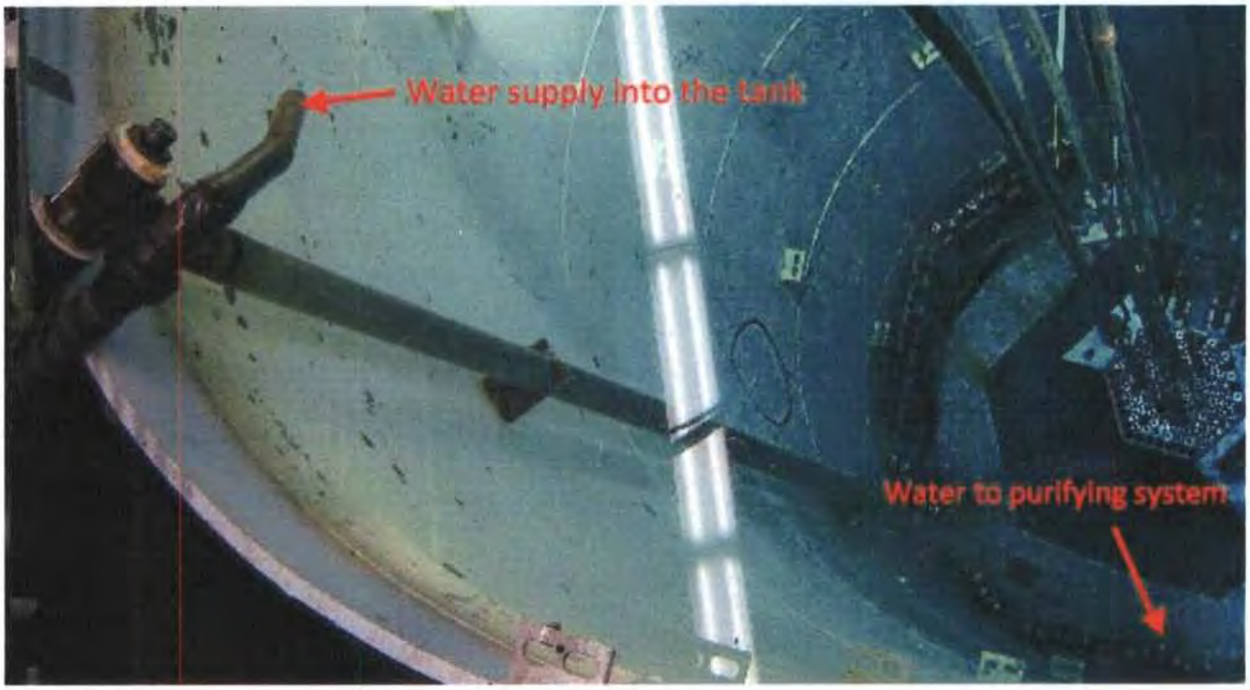


Figure 5.1-1 The UUTR pool and water supply system

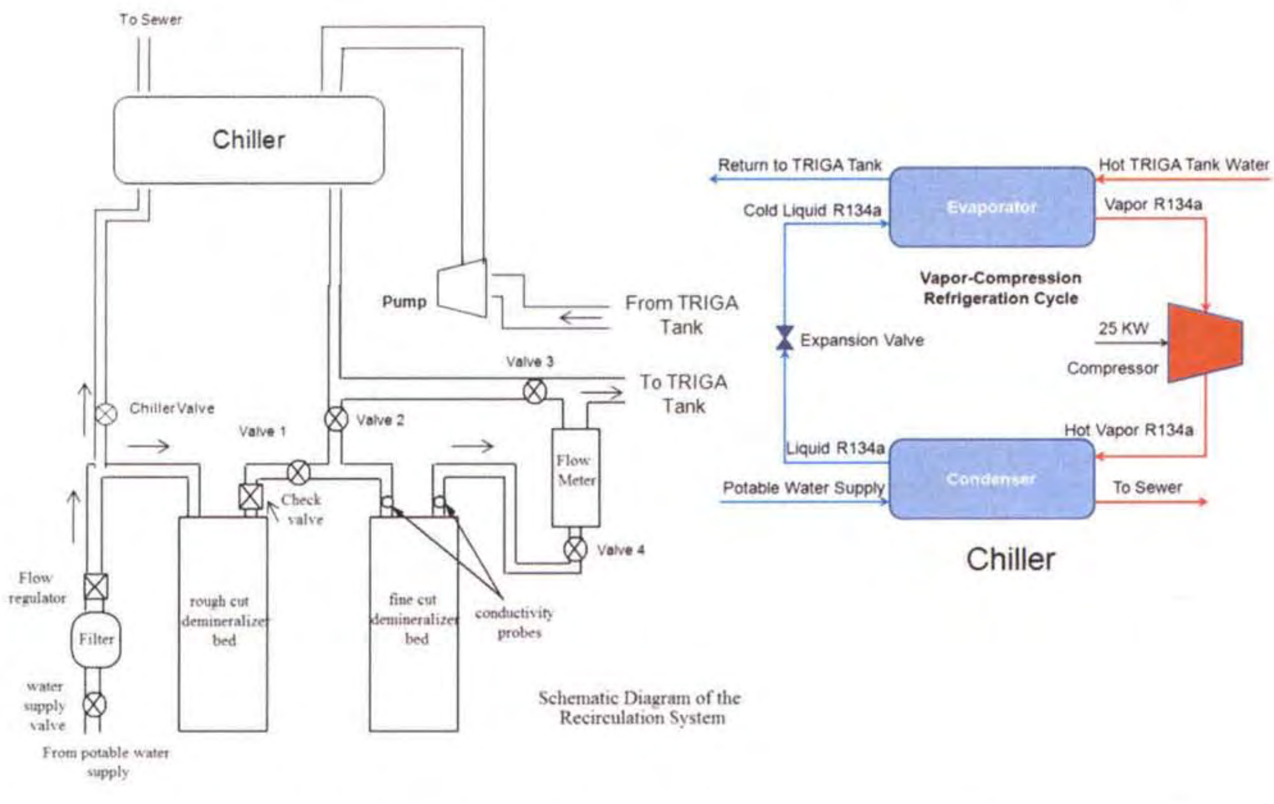


Figure 5.1-2 Schematics of the UUTR secondary cooling system

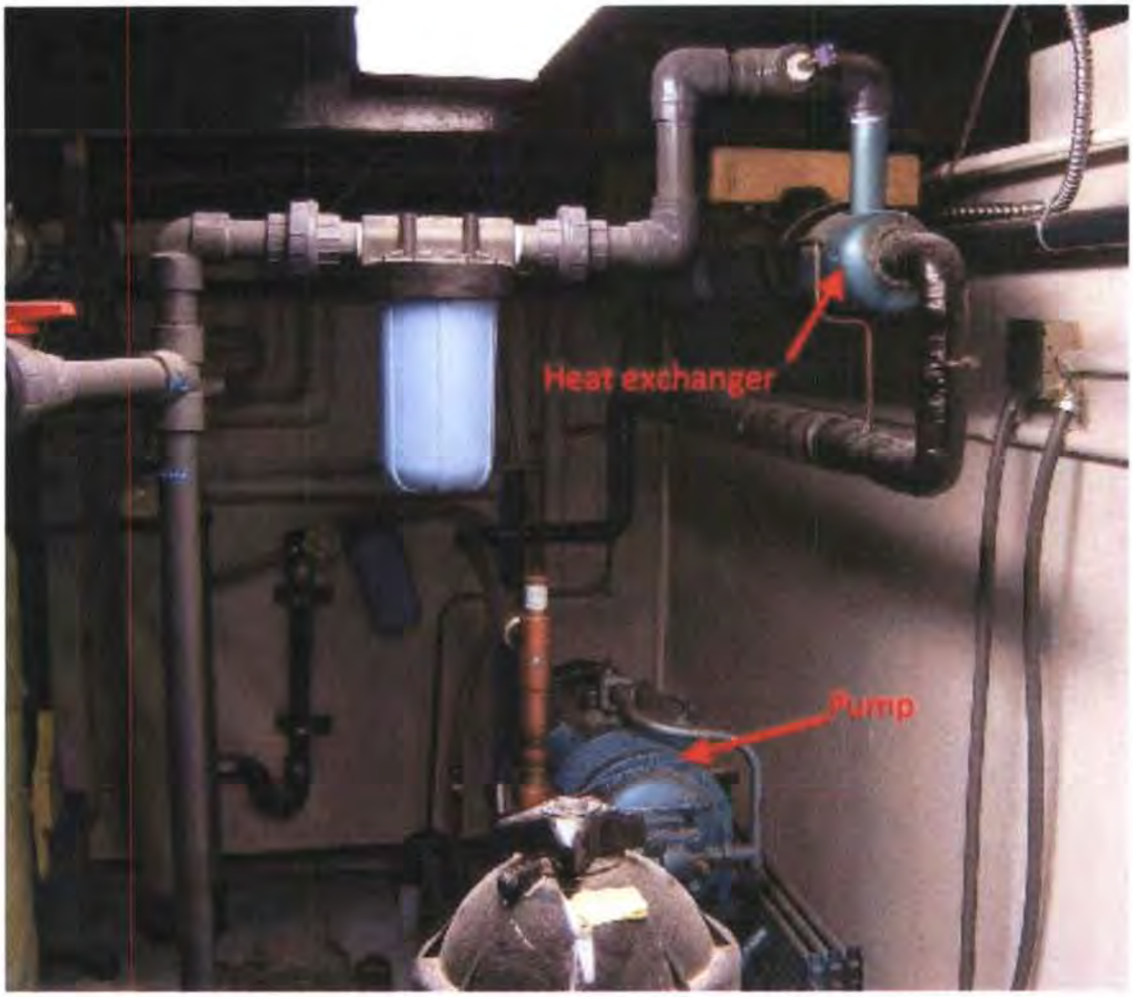


Figure 5.1-3 The UUTR heat exchanger system

5.2 Primary Coolant System

The UUTR core is cooled by natural circulation of the reactor tank water. The volume of the tank is 8,000 gallons; this tank represents the primary cooling system of the UUTR. The radiation shielding requirement is fulfilled by keeping at least 18 ft of water directly above the reactor core. In order to monitor water level of the pool, an ultra sonic sensor is installed in the reactor tank. The alarm and scram system will be activated if the water level is lower than 15.5 inches from the top of the reactor tank. The tank water temperature is maintained below 35 °C (308.15 °K) by limited run time at full core power. The pool water temperature in general is approximately 291.15 °K (18° C). The time to reach the temperature of 35 °C from 18 °C (291.15 °K) is about 8 hours operating the UUTR at 90 kW (estimates shown in **Fig. 4.6-11**). After the reactor shutdown, the pool water temperature decreases at the rate of ~ 0.5 °K /hour; it takes 34 hours to cool down to the temperature of 18 °C without any secondary cooling system involved. The UUTR primary coolant system consists principally of a pump, heat exchanger, demineralizer, deionizer, fiber cartridge filter, flow meter, and conductivity proves (**Fig. 5.2-1**).

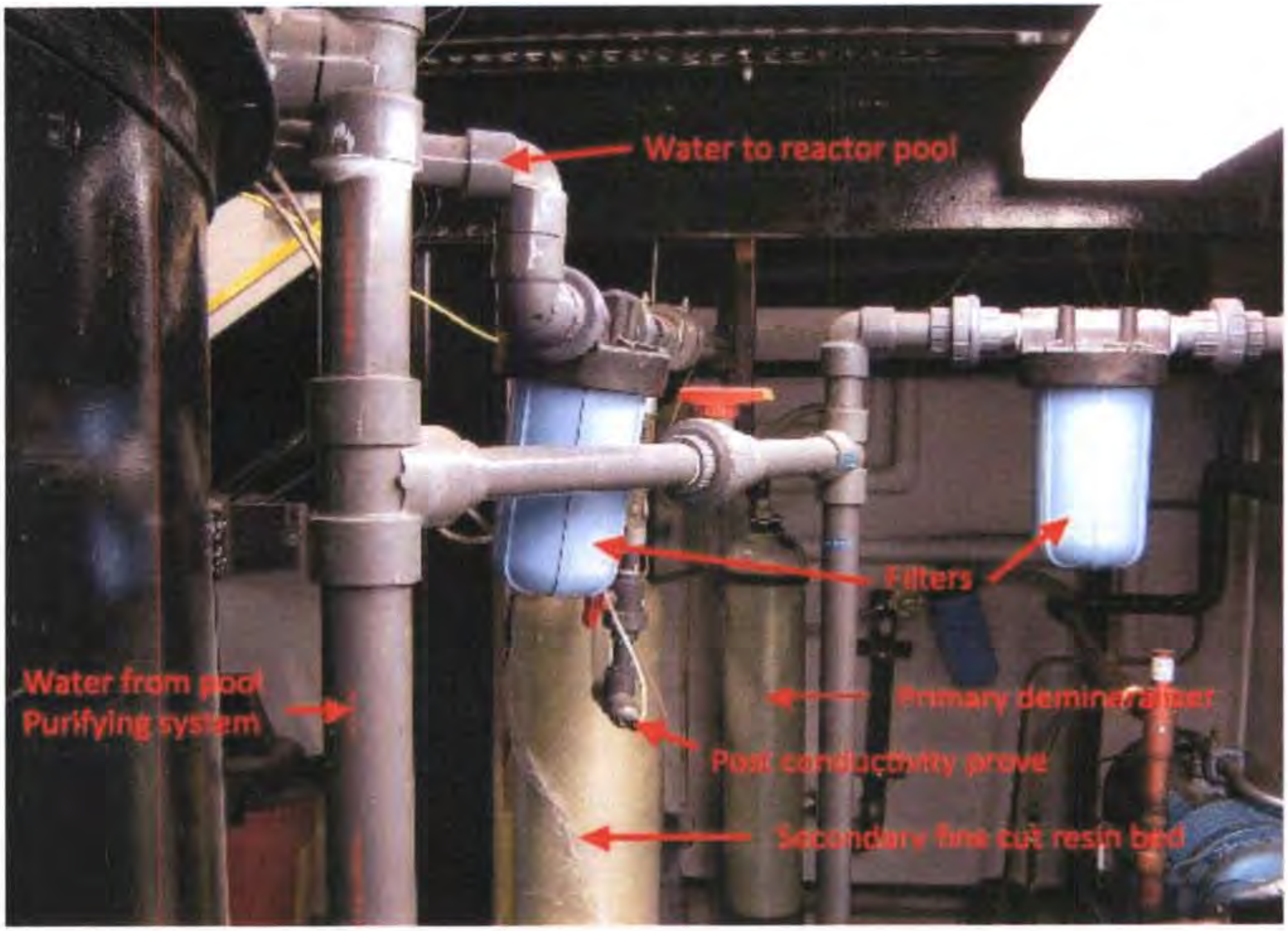


Figure 5.1-4 The UUTR secondary cooling system

The natural convective flow of the primary cooling system can remove 100kW of heat from the reactor fuel. An administrative control limit of 35 °C has been imposed on the bulk water temperature primarily to prevent possible skin scalding in resin-bed clean-up system. The manufacturer recommendation for current resin-bed system is to keep the pool water temperature below 40 °C (313.15 °K) to prevent melting of the resins.

In the emergency case, such as loss of the coolant water, the primary coolant system will be turned on. Also, there is another outlet in the west side of the reactor room wall. **Figure 5.2-2** shows the emergency water outlet in the reactor room. Also, the water from fuel inspection area can be used as an emergency water supplier. In the emergency case, some of the water will not pass through the filtering system.

The system parameters are displayed on the console. Visual and audible alarms are also located on the reactor control console, and are activated if the tank water level drops below present limits. There are also tank level indicators in the control room, and on the control rod support structure in the reactor room.

All system components that contact the primary water are made from PVC, aluminum, or stainless steel.

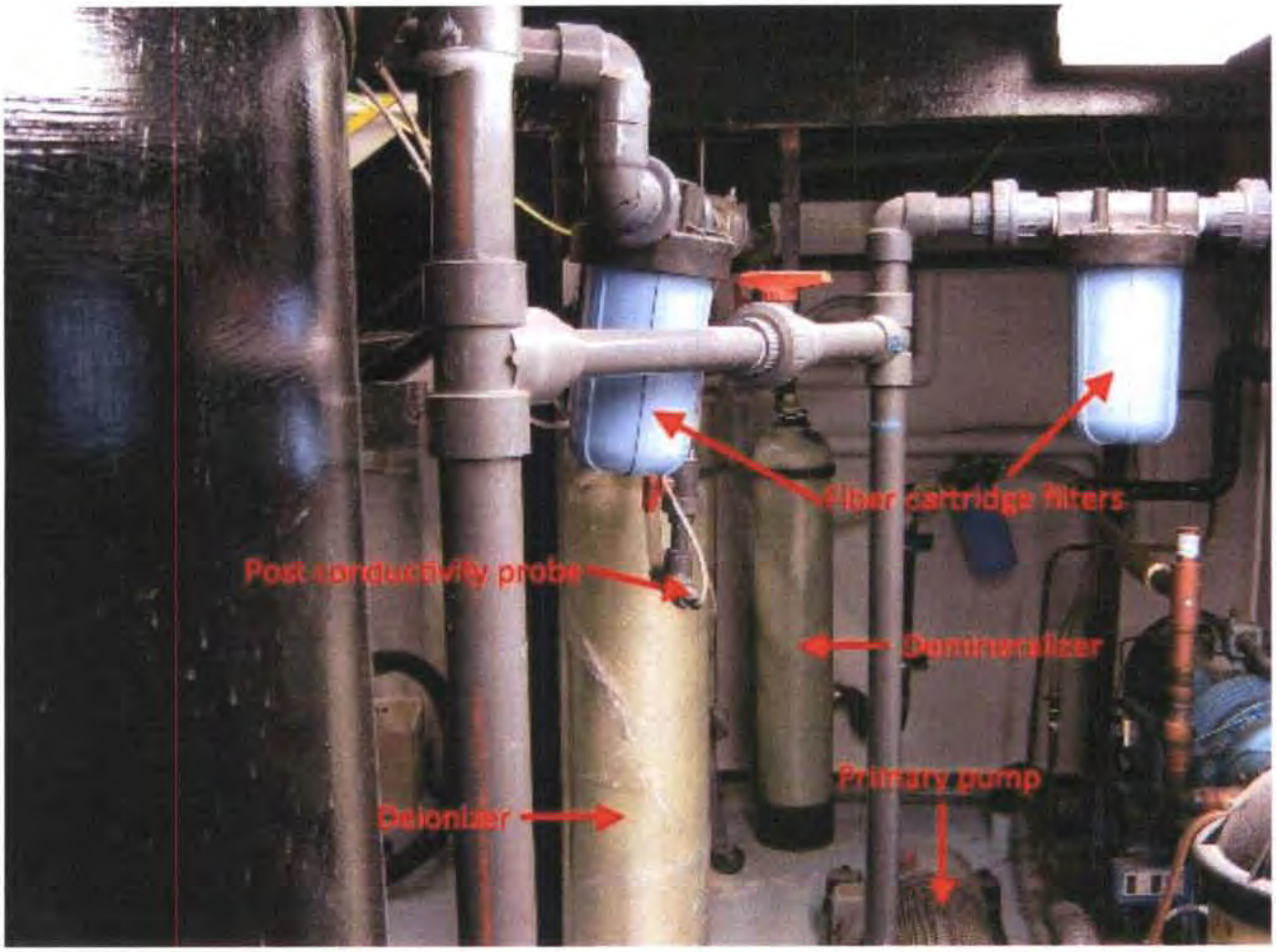


Figure 5.2-1 The UUTR's primary coolant system

5.2.1 Pump

The water system pump is a centrifugal-type with a stainless steel body and impeller. The pump is driven by a directly-coupled induction motor. The suction and the discharge of the pump are a 5 cm and a 3.8 cm flanged-pipe connection, respectively. **Figure 5.2-3** shows the primary coolant pump system.

5.2.2 Heat Exchanger

The UUTR has a 25kW capacity tube-type heat exchanger as a secondary cooling system (**Fig.5.1-3**). The heat exchanger uses R134a to cooling down primary coolant. The pressure of the secondary system is maintained at a higher level than the primary system to prevent cross contamination of secondary water should a leak develop in the heat exchanger. Butterfly valves allow isolation of inlet and outlet of each coolant system in case of heat exchanger damage to prevent mixing of cooling water. The heat exchanger was upgraded in 1996.

Figure 5.2-2 Emergency water supplier in the reactor room

5.2.3 Cleanup Loop

The UUTR has two resin beds cleanup system. The resin beds are fiberglass canisters of mixed-resin beds. A pre-filter is used to remove any particulate material prior to the water entering the deionizer. It uses replaceable fiber cartridges. Two independent conductivity cells are used to measure the conductivity of water entering the resin bed and the conductivity of the water exiting the resin beds subsequent to entering the reactor tank. There are readouts of the conductivity located on the reactor console. The prime function of a demineralizer is to maintain the conductivity of the reactor pool water at a sufficient low level (<5μmhos/cm) to prevent corrosion of the reactor components exposed to the water such as fuel element and control rod elements.

The demineralizer is a mixed-bed type that removes both positive and negative ions from the circulation water. Any contaminants in the water are absorbed and concentrated in the resin bed. Any radioactive ions in the water are absorbed and concentrated in the resin bed. A demineralizer will be slightly radioactive in a normal use. If the conductivity approaches 5μmhos/cm for both pre and post conductivity proves, the resin bed will be replaced based on the UNEP procedure (UNEP form 006R3).

The flow meter is mounted down stream from the demineralizer. It has a range of 0 to 30gpm. The meter is operated by the flow of water, which forces the flow rotor upward in the tube. The flow rate can be read in the control room's computer system with LabView™.

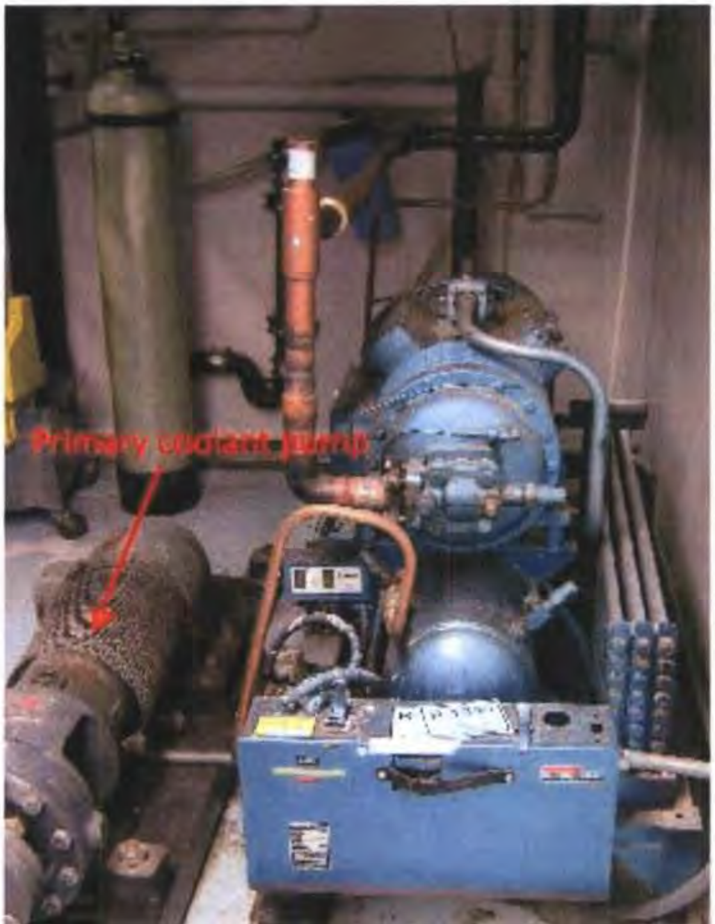


Figure 5.2-3 Primary coolant pump system

5.2.4 Piping and Valves

All piping and fittings in the water system are of PVC and have tefron packing. All of the water is returned to the pool through a deflector nozzle, which creates swirling current in the pool that increases the time it takes water to travel from reactor core to the surface, and thus ample time for the decay of radioactive Nitrogen-16.

Water level is normally kept approximately between 6 inches and 10 inches from the top of the reactor tank. To prevent a malfunction in the primary system from the draining the pool, the primary inlet is approximately 30 inches below water level. The alarm will be activated and the reactor will be scram if the water level is lower than 15.5 inches measured from the top of the reactor tank. The corresponding procedure is defined with the ***UNEP form 003R3.***

5.3 Secondary Coolant System

The UUTR has 25kW R134a based secondary cooling system. The flow rate of water in this system is approximately 4 gpm. The secondary cooling system circulates water from the reactor tank through heat exchanger and cooling system. The absorbed heat is removed using portable water supply and this water is released to the sanitary system as show in **Fig. 5.1-2**. The secondary cooling system is rarely used; the last time it was used was in 1996. This system is rarely used because the UUTR pool water temperature is kept below 20 °C (293.15 °K) in general, thus it does not exceed 35 °C when the UUTR operates at full power. The amount of portable water released into the sanitary system after absorbing heat from heat exchanger is less than 50 gallons/year as it was measured in 1996. The UUTR tank water temperature is administratively limited to lower than 35 °C to prevent damage to the demineralizer. At temperature above approximately 40 °C[1], the resin may break down and be dispersed into the reactor pool, which may cause a corrosion of the reactor components.

The major components of the secondary cooling system are the primary pump, secondary pump, tube type heat exchanger, water filter and control instrumentation. If the water temperature is below 10 °C (283.15 °K), the pump will be shut off automatically and thus be protected against freezing. One thermometer is installed to measure the water temperature that is monitored from the control room. The circulation pump for the primary coolant continues to operate while the refrigeration system is shut off. In operation, water leaves the pool through the suction pump at a point about 20 ft below the surface of the pool, passes through the primary pump and heat exchanger, passes through fine cut resin bed and returns to the pool through a distribution pipe located 2 ft below the water surface as shown in **Fig. 5.1-1**.

The UUTR tank water is routinely (every month) monitored for radionuclides. The water sample from the reactor pool is counted using a high purity germanium detector system to identify presence of any radionuclide. In the event of any radioactive spill, the reactor is

[1] From the private conversation with the manufacturer

immediately secured and the key is removed if the reactor is in operation, and the reactor supervisor must be called. Contamination of any surface near the resin bed area will be removed using standard decontamination techniques, with appropriate personnel in protective clothing monitoring the procedures. The reactor supervisor or a member of the UUTR staff will retain all decontamination fluids and solids for examination before the materials are discharged to the sanitary sewer system, as defined by the ***UUTR procedure form UNEP-032***. The ***UNEP-037 form*** must be completed after the cleanup of the contaminated area is completed.

The pump and heat exchanger system are both shown in **Fig. 5.1-3**. Water from heat exchanger is circulated through the resin bed at a flow rate ~4gpm through a manually adjustable bypass valve in the circulation loop. Conductivity probes located at the inlet and outlet of the demineralizer unit determines the effectiveness of the water purification system. The output of the fine-cut resin bed normally is of 0.1 μmhos/cm and a conductivity alarm occurs when the conductivity rises to 4 μmhos/cm. This alarm is located in the control room. Therefore, the conductivity of the pool water is maintained below 5μmhos/cm. The fine cut resin bed is replaced as needed, when the conductivity rises to 4 μmhos/cm. The pH of the pool water is maintained in the range of 5.5 to 7.5 in conformance with the **TS 3.3**.

5.4 Primary Coolant Cleanup System

The purity of the water in the pool is maintained by the primary UUTR cleanup system that extracts 5 gpm of primary coolant using a pump and passes it through two 25 micron cartridge filters and a resin bed to keep the primary water chemistry within operational limits. A detailed drawing of the clean up loop is shown in **Fig. 5.5-1** (Underwater cleanup system is shown in **Fig. 5.4-1** which is used for visible dirt filtering). The water recirculation schematics is shown in **Fig. 5.4-2**.

The primary coolant conductivity must be less than 5 μmhos/cm for both conductivity probes. The pH of the pool water must be between 5.5 and 7.5 (The pH value for the UUTR tank is varied between 5.9 to 6.3 which is measured by in-tank pH-meter shown in **Fig. 5.4-3**). The conductivity limit used by the licensee is a longstanding value for research reactors accepted by the U.S. NRC, which has been shown to be effective in controlling corrosion in aluminum and stainless-steel systems.

The UUTR resin bed (demineralizer) has siphon breaks to prevent a failure in the demineralizer from siphoning a large amount of primary coolant from the reactor tank. The UUTR contains water level detector, two conductivity probes, pH meter, and GM detector (shown in **Fig. 5.4-4**) near the reactor tank. The reactor room floor contains about 40 gallons liquid storage pit, which is 22 ft away from the reactor tank (**Fig. 5.4-5**). In the worst case, if any leaking exists in the reactor room floor area, the liquid can be stored in this liquid storage pit up to 40 gallons of liquid.

Figure 5.4-1 Under water cleanup system

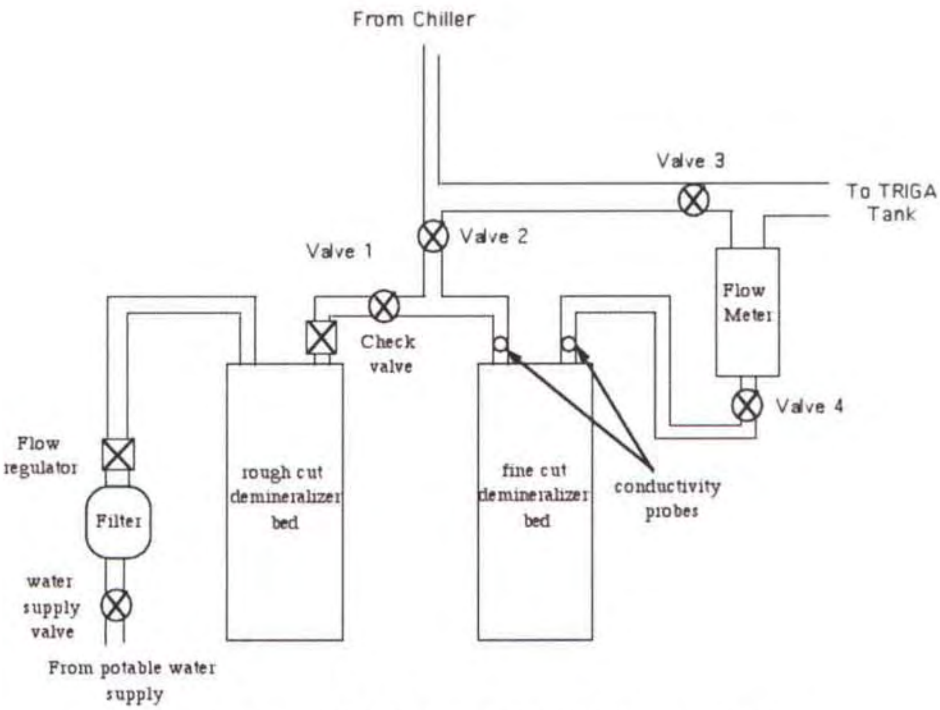


Figure 5.4-2 Schematics of the recirculation system

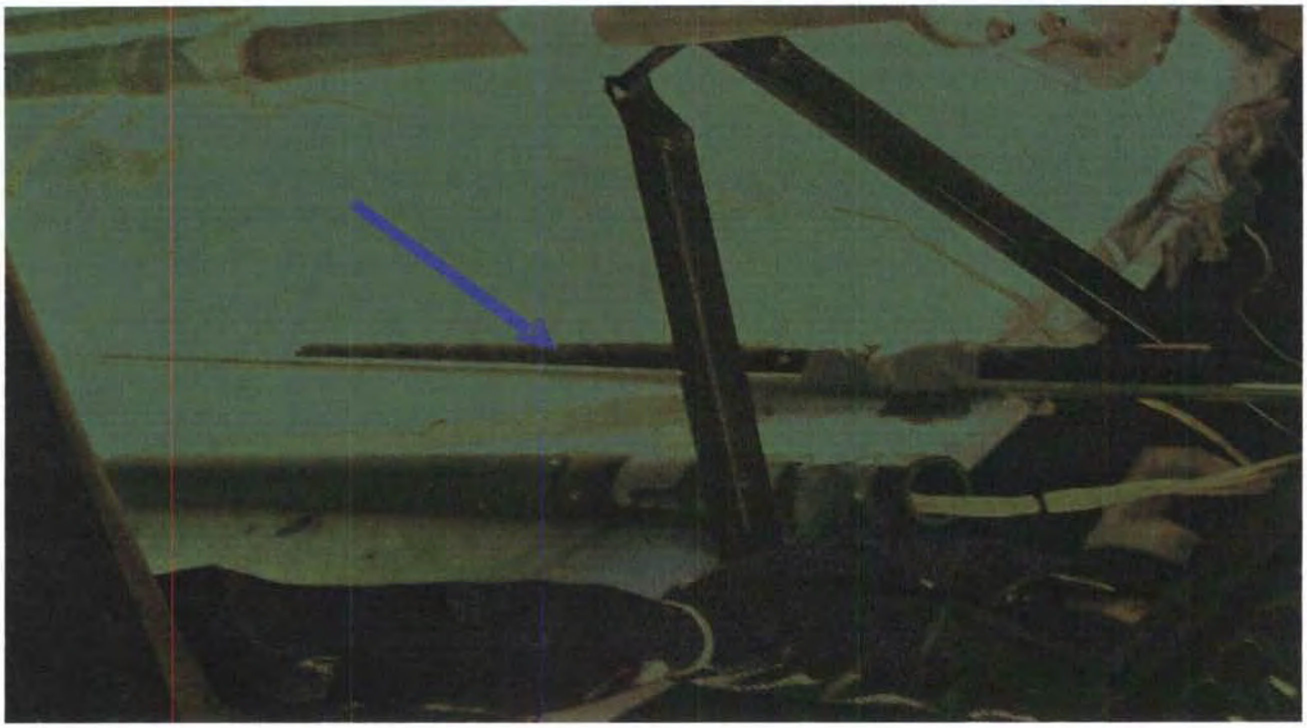


Figure 5.4-3 In-tank pH meter (indicated by arrow)

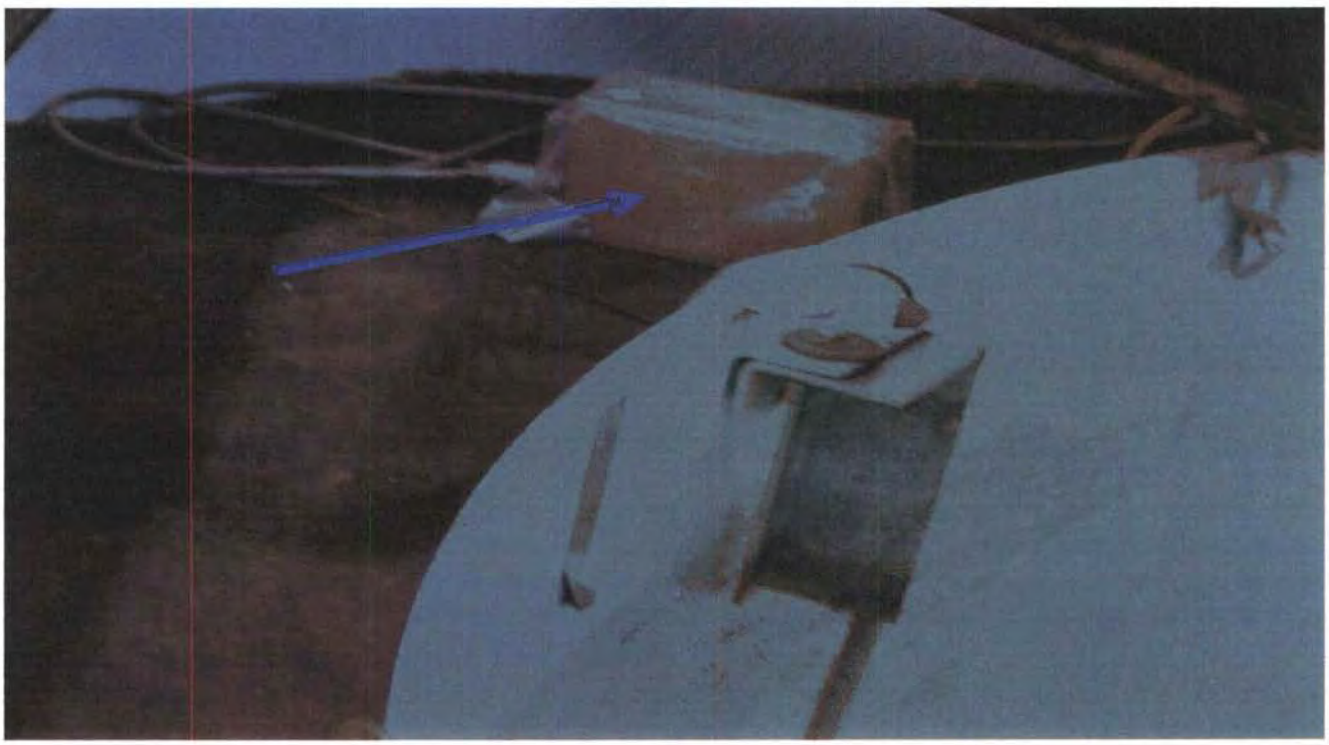


Figure 5.4-4 GM detector near the reactor tank (indicated by arrow)

Figure 5.4-5 A 40-gallon water storage (indicated by arrow)

A pool level alarm is provided to indicate a loss of coolant if the pool level drops more than 15.5 inches from the top of the reactor tank. In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool. Loss of coolant alarm after 15.5 inches of loss requires corrective action.

In order to insure that radioactive species do not build up in the pool water, the pool water is routinely monitored for the contained radionuclides. The monitoring involves the counting of a sample of pool water on a sensitive gamma ray spectrometry system and identifying the radionuclides present. We perform gamma spectroscopy on pool water for Cs-137 peaks and activities higher than µCi's in order to ensure it is within the ALARA limits. Our facility is committed to keep both occupational and public radiation exposure as low as is reasonably achievable (ALARA). This means, we assure the exposure to be no greater than 10% of the occupational limits and 50% of the public limits prescribed by the 10 CFR Part 20.

5.5 Primary Coolant Makeup Water Systems

The level of the water in the pool is maintained by the pool water make up system. A schematic diagram of the makeup water system is given in **Fig. 5.5-1**. The source of makeup water is the Merrill Engineering Building's culinary water system. This system feeds deionized water to the UUTR reactor tank. It supplies water with a conductivity of 1 μmho/cm and the deionizer beds are changed if the conductivity increased to 4.5 μmhos/cm (There is an alarm on the control panel which will go off if the conductivity exceeds 4 μmhos/cm. This water is passed through a mixed resin bed prior to entering the UUTR reactor tank. This mixed resin bed remains non-radioactive as it processes building water only. The outlet flow of the makeup system discharges to the purification system.

There is a float switch monitoring the pool water level attached to the side of the reactor pool that controls make up water for the pool. If the water level decreases below a certain level (15.5 inches from the top of the reactor tank), the alarm will sound and the reactor will scram if it is in operation. The tank water refilled manually approximately every two months. The UNEP form 008R4 shows the schematic of the tank water supply system and procedure of the UUTR tank water refill.

Figure 5.5-1 Primary coolant makeup water system (A: Deionizer B: Demineralizer C: pre-fiber Filters)

5.6 Nitrogen-16 Control Systems

Measurements at the prototype TRIGA MARK F reactor facility showed that that the diffuser results in a hold-up time increase by a factor of 3, so that the nitrogen-16 rise time is closer to 67 seconds [Armed Forces Radiobiology Research Institute Safety Report for 1,000 kW MARK F TRIGA]. In 67 seconds, the Nitrogen-16 decays to $7\text{x}10^{-7}$ of its initial value. Thus, the number of Nitrogen-16 atoms that reach the water near the UUTR pool surface is estimated to be about $2.86\text{x}10^{4}$ atoms/sec (for UUTR power of 100kW). Only a small portion of the Nitrogen-16 atoms present near the pool surface is transferred into the air of the reactor room. When a Nitrogen-16 atom is formed, it appears as a recoil atom with various degrees of ionization. For high-purity water (~2 μmho/cm) practically all of the Nitrogen-16 combines with oxygen and hydrogen atoms of the water. Most of it combines in an anion form, which has a tendency to remain in water [R.L. Mittl and M.H. Theys, "Nitrogen-16 concentration in EBWR", Nucleonics, March 1961, p. 81.]. It is assumed that at least one-half of all ions formed are anions. Because of its 7.4 sec half-life, the Nitrogen-16 will not live long enough to attain a uniform concentration in the tank water. With a diffuser in operation the Nitrogen-16 atoms are dispersed in the 1-foot of water at the top of the pool directly above the core. Actually, they are more likely dispersed over a wide area in the pool and decay before this lateral movement is completed. Directly above the core, the dominant contribution to the dose rate is the direct radiation from the core. Maximum fraction of Nitrogen-16 atoms that can escape from the water to the air per second is estimated to be ~15 atoms/sec. At a point 3 meters immediately above the surface of the tank water, the Nitrogen-16 concentration in a 91 cm diameter cylinder (assuming no mixing with the air of the room air) is ~$2\text{x}10^{-10}$ $\mu\text{Ci/cm}^3$ at 100kW power (see the calculations below). The dose rate due to this concentration is small compared to the dose rate from the core itself. In the rest of the room, the activity is affected by dilution, ventilation, and decay. Thus, the accumulated concentration of Nitrogen-16 in the whole room is $7.74\text{x}10^{-12}$ $\mu\text{Ci/cm}^3$ for 100kW.

The particulate radioactive materials such as Nitrogen-16 are monitored from Continuous Air Monitor (CAM). A reactor operator can monitor this number from the control console. **Figures 5.6-1** and **5.6-2** show the CAM system and monitoring window at control console. At 90 kW reactor power a typical reading is less than 100 CPM. The CAM system is located 3 meters behind the reactor tank. The setting point of the CAM system for particulate materials is 1,000 CPM. The UUTR can operate for 48 hours if CAM is out of order.

Calculation for 100 kW UUTR

Starting from:

$$N^N = \frac{\Phi_V N^0 \sigma^0}{\lambda^N}\left[1 - e^{-\lambda^N t}\right]$$

where

N^N = Nitrogen-16 atoms per cm^3 of water

Φ_V = virgin fission neutron flux = 3.54x10^{12}, neutrons/cm^2-sec at 100 kW

N^0 = oxygen atom per cm^3 of water = 3.3x10^{22} atoms/cm^3

σ^0 = absorption cross section of oxygen = 2x10^{-29} cm^2

λ^N = Nitrogen-16 decay constant = 9.35x10^{-2}/sec

t = average time of exposure in reactor (sec)

Assuming that t = 20 sec, then N^N is 2.11x10^7 atoms/sec. For the UUTR, it takes approximately 67 seconds for Nitrogen-16 to travel from the top of the reactor core to surface of the pool water that is 22 ft above the core upper surface. Therefore, about 4.02x10^4 atoms/sec will reach the water near the pool surface. The maximum fraction of Nitrogen-16 atoms that can escape from the water to the air per second can be assumed to be similar to the case of Argon. Thus,

$$f_{2\rightarrow3}^{N} \leq \frac{1}{2}\frac{3\times10^{-3}cm/\text{sec}}{30cm} = 5\times10^{-5}/\text{sec}$$

where f is the fraction of Nitrogen-16 atoms in region 2 that escape to region 3 per unit time, sec; the subscript 2 denotes the reactor tank water region external to the reactor core, and subscript 3 denotes the reactor room region. The number of nitrogen atoms entering the air is given by

$$f_{2\rightarrow3}^{N}N^{N}V = \frac{f_{2\rightarrow3}^{N}(4.02\times10^{4})}{f_{2\rightarrow3}^{N}+\lambda^{N}} = \frac{5\times10^{-5}(4.02\times10^{4})}{5\times10^{-5}+9.35\times10^{-2}} = 21.5 atoms/\text{sec}$$

At 3 meters immediately above the surface of the tank water, the Nitrogen-16 concentration in a 0.91 m diameter cylinder for 100kW UUTR is equal to

$$A = \frac{21.5}{3.7\times10^{4}\times300\times\pi\times45.5^{2}} = 2.98\times10^{-10}\frac{\mu Ci}{cm^{3}}$$

In the rest of the room, the activity is affected by dilution, ventilation, and decay. The accumulation of Nitrogen-16 in the room as a whole is given by

$$\frac{d\left[V_3N_3^{16}\right]}{dt} = S - \left[\lambda^N + \frac{Q}{V_3}\right]N_3^{16}V_3$$

where

S = number of Nitrogen-16 atoms entering the room from pool per second (atoms/sec)

V_3 = volume of the reactor room = 5.65x10^8 cm^3

Q = volume flow rate from the reactor room exhaust = 6.1x10^5 cm^3/sec

For saturation condition, it follows:

$$V_3 N_3^{16} = \frac{S}{\lambda^N + \frac{Q}{V_3}} = \frac{21.5}{9.35 \times 10^{-2} + 1.08 \times 10^{-3}} = 227 nuclei$$

The cumulative activity from Nitrogen-16 for the whole reactor room is then 1.09x10^{-11} μCi/cm^3.

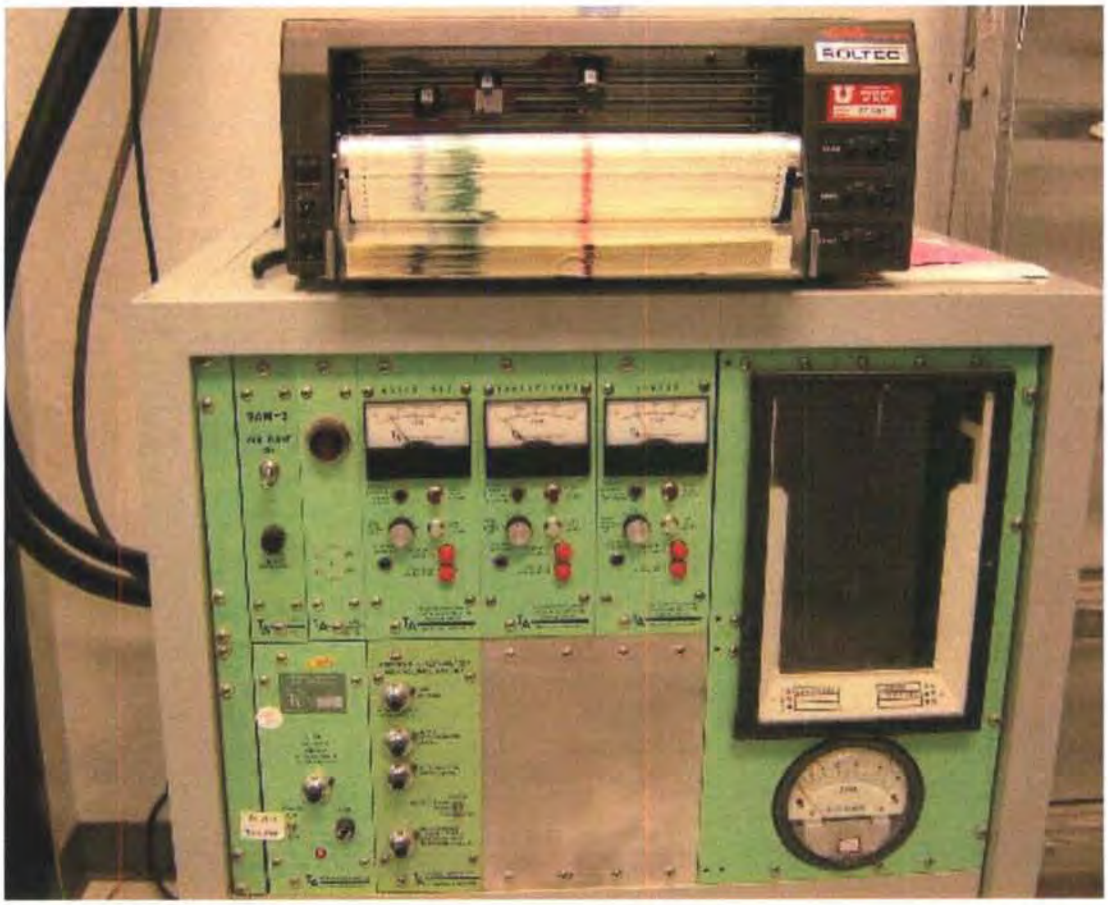


Figure 5.6-1 Continuous Air Monitor (CAM)

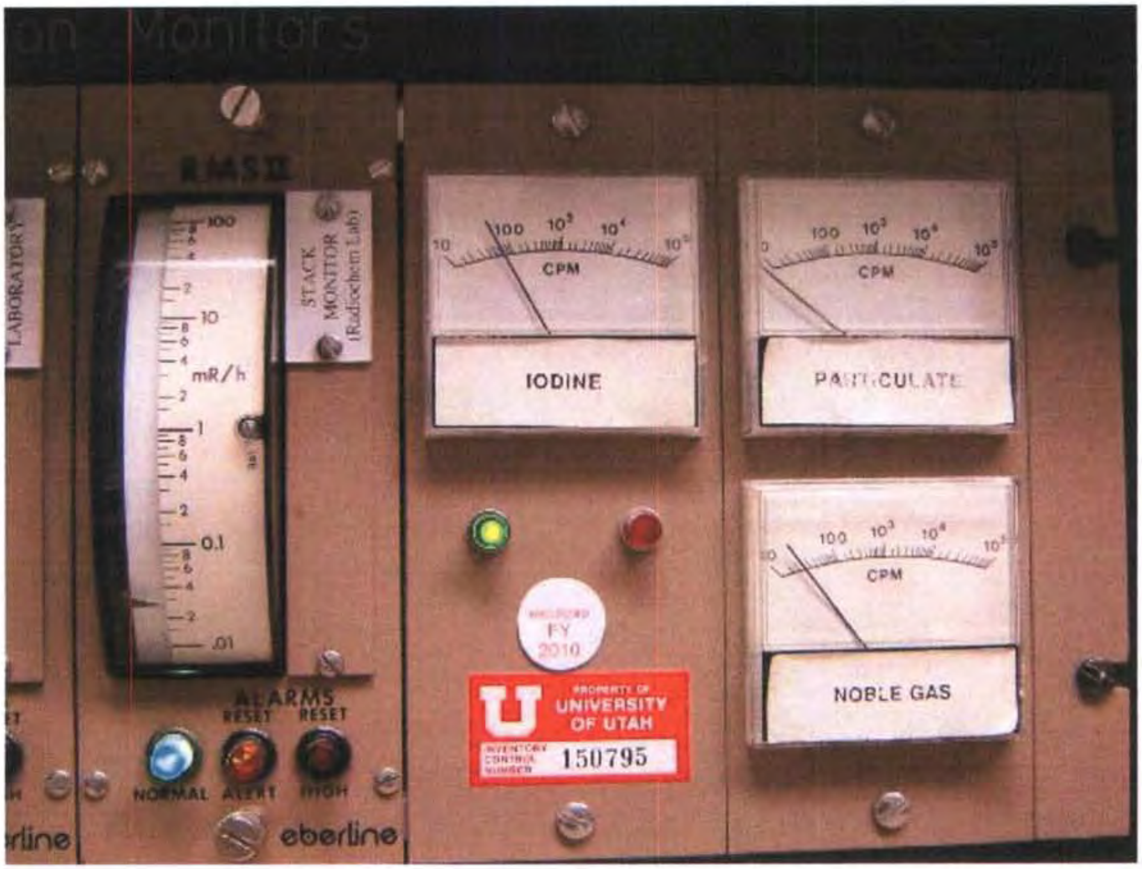
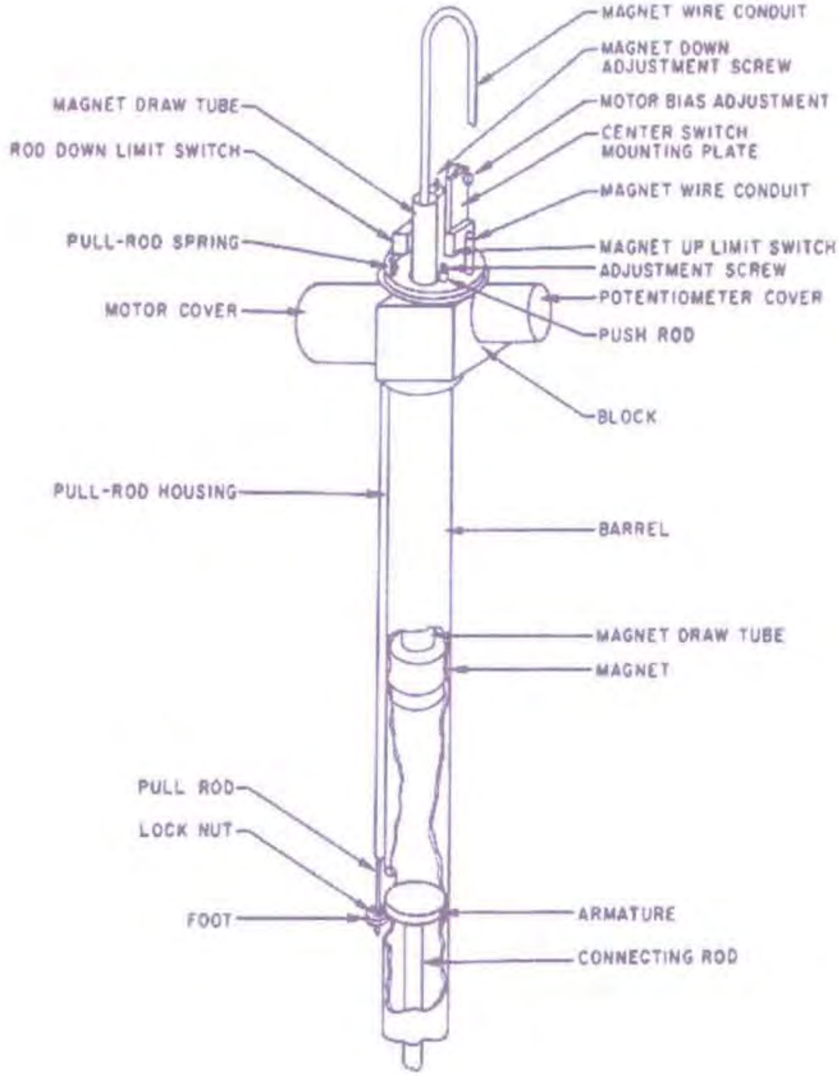


Figure 5.6-2 CAM monitor for iodine, noble gas and particulates

6. ENGINEERED SAFETY FEATURES

Chapter13, *Accident Analysis*, describes that there are no accidents whose consequences could be unacceptable without mitigation. Thus the UUTR reactor facility design does not include, have or need any engineered safety features. That is, there is no conceivable mode of operation, which could create a significant threat to the health and safety of the reactor staff and general public.

7. INSTRUMENTATION AND CONTROL SYSTEMS



7.1 Summary Description

The types and operational characteristics of the instrumentation used on this reactor are analyzed in order to demonstrate that these instruments meet all design criteria for proper operation and, more importantly, safety of operation. The UUTR is currently operated in a steady state mode only. The two main items used in the reactor's control system are the fuel and control rods. The design bases for and functional characteristics of the reactor core and its components are discussed in **UUTR SAR 4**. The **UUTR SAR 7** specifically addresses the design parameters of the instrumentation that monitors and controls conditions in the reactor core that ensure system components perform safely. The conditions monitored and controlled are: pool water level and chemistry and temperature, reactor power level, control rod position, fuel temperature, ventilation system and area radiation monitoring. Radiation monitoring is described in more details in **UUTR SAR 11**.

The UUTR is operated from a console located in the control room. Additional instrumentation is housed in cabinets on either side of the console.

7.2 Design of Instrumentation and Control System

Three independent power-measuring channels (linear power, percent power and log power) provide for a continuous indication of power from the source level to 100 kW. The reactor will scram at over power for linear and percent power channels. Fuel temperature is measured from two instrumental fuel elements for display as well as used by the reactor protection system. Other parameters such as water conductivity, pH, neutron population up to 1 kW, and water temperature are also monitored and displayed.

7.2.1 Design Criteria

The instrumentation and control system is designed to provide the following:

- Complete information on the status of the reactor and reactor-related system
- A means for manually withdrawing or inserting control rods
- Manual control of reactor power level
- Automatic scram in response to over power, high fuel temperature and high voltage Interruption
- Monitoring of radiation and airborne radioactivity levels in the reactor room

The **UUTR TS 5.2** requires that the reactor shall not be operated unless there is 18 ft water above the top of the reactor core. In addition, the UUTR is equipped with the tank water level alarm set to indicate loss of coolant if the water level drops 15.5 inches from the top of the UUTR water tank. The conductivity of the pool water is limited to 5 μmhos/cm (**UUTR TS**

3.3). The pH of the pool water must stay between 5.5 and 7.5. The conductivity and the pH of the primary coolant water must be measured monthly.

7.2.2 Design-Basis Requirement

The primary design basis for the UUTR is the safety limit on fuel temperature. To prevent exceeding the safety limit, the reactor will scram automatically for high fuel temperature and high power condition. The reactor will also scram when the control system loses high voltage to the power monitoring systems.

7.2.3 System Description

7.2.3.1 Reactor Power Measurements

Three separate detectors measure the UUTR power: two uncompensated ion chambers for percent and log powers, and one compensated ion chamber for linear power. The linear and percent power channels have digital readouts on the console in addition to the recorded graphical output. The integrated power channel uses the linear power output data to determine run integrated power. A fission chamber is used to determine the neutron start-up rate. If the neutron level is less than 2 cps the interlocks restrict movement of the control rods.

The linear power channel can display reactor power from 0.1 W to 1 MW. The numeric value displayed on the linear power channel is interpreted as a percentage of the power setting on the range switch. If the linear power exceeds 100% of the range switch setting, a reactor SCRAM is initiated. The output of the linear power channel signal is integrated and displayed so that the operator can monitor the run integrated power. A SCRAM will also occur if there is a loss of the ion chamber supply voltage. A compensated ion chamber is used with a transistorized linear amplifier and preamplifier to feed linear power circuits. Linear count rates are read on the linear recorder during reactor start-up, steady state power and shutdown. The compensated ion chamber (Westinghouse WL-6971) has a thermal neutron sensitivity of 4.0×10^{-14} amp/n/cm^2-sec and an uncompensated gamma sensitivity of 5.0×10^{-11} amp/R/hr.

The sensitivity of a linear power channel is variable, so that its output on either the linear recorder or the meter will read 3, 10, 30, 100, 300, 1000, 3,000, 10,000, 30,000, 100,000, 300,000, 1,000,000 W full scale depending on the setting of the range change switch on the control panel.

The power level amplifier, fed by the linear power channel, is a Burr-Brown model 3061/25 integrated circuit operational amplifier with an accuracy of $\pm$ 0.02%, and a drift of 0.03% of full scale for 8 hours within the temperature limits of 293.15 °K (20 °C) to 323.15 °K (50 °C). The integrated power channel uses the output of the linear power channel in conjunction with information from the range switch to exactly determine the reactor power level. This power level is integrated over time and fed to two digital counters, one of which can be reset by the operator to monitor the run integrated power.

The percent power channel provides the reactor operator with reactor power information on a scale of percent licensed power. This channel is capable of scramming the reactor at a set point at or below 110% of the licensed power. The percent power channel uses an uncompensated ion chamber (Westinghouse WL-6937) and has a thermal neutron sensitivity of $4.0\text{x}10^{-14}$ amp/n/cm^2-sec and a gamma sensitivity of $5.0\text{x}10^{-11}$ amp/R/hr. The percent power channel has a fixed sensitivity and its output is read on a meter indicating 0 to 150% of the full power. This channel feeds a meter and scram circuit directly through an attenuator.

The magnet scram amplifier is a General Atomics transistorized model AS-120. The scram is adjustable from 25% to 150% of the full power. The accuracy of the trip point is $\pm$ 5% of the set point with a maximum delay of 20ms. The amplifier will handle up to 4-rod magnets. The scram adjustment on percent power channel is set to scram the reactor at about 110% of full scale of the range switch. Scrams are also initiated for loss of ion chamber supply power.

The log power channel provides the reactor operator with reactor power on a log scale. This meter provides the reactor operator with an additional indication of how quickly the reactors power is changing. The logarithmic channel (Log-N) provides continuous indication of power covering 5 decades from 4×10^{-4} to 4×10^{-9} amp. The Log-N uses an uncompensated ion chamber located approximately on the mid plane of the core on the outside of the perimeter core shroud. The detector is a Westinghouse WL-6337 or equivalent, having a thermal neutron sensitivity of 4.0×10^{-14} amp/n/cm^2-s and a gamma sensitivity of 5.0×10^{-11} amp/R/hr.

The signal for the period monitoring channel is provided from the period amplifier of the Log-N channel. The period amplifier is a General Atomic transistorized model AP-130 having a range of -40 seconds to +7 seconds. The period meter is located on the control console in the Control Room.

The fission counter measures the number of neutrons in the core at start-up. An interlock is provided so that withdrawal of any control rods is not permitted, unless the source count is above the required minimum value of at least 2 counts per second. The fission counter shows the number of thermal neutrons that interact with the fission chamber every second. When a thermal neutron causes the U-235 lining of the fission chamber to fission, one of the fission products is propelled into the center of the chamber. The negative charge from the resulting ionization is collected on the center conductor. This pulse is amplified, discriminated from other ionization pulses (because it is much larger), and fed into a rate meter. If too few fission pulses are received, the source interlock is turned on.

The count rate channel provides information on the reactor at low power level of about 10^{-3} W (1 count per second) to 2 W (10^4 counts per second). The detector is saturated at high power level and is not used as a reference past about 2 W. The count rate channel uses a fission counter, Westinghouse WL-6971 or equivalent, as the detector. This detector has a sensitivity of about 0.14 count/neutron/cm^2. The detector is approximately positioned on the mid plane of the core on the outside of the perimeter shroud of the core.

Additional count rate channel equipment includes a preamplifier (transistorized model), a linear amplifier (General Atomic transistorized model), a log count rate meter (General Atomic transistorized model), and a linear count rate meter (General Atomic model CR-100). Detailed electronic descriptions of the power monitoring channels can be found in **Table 16.1-2.**

7.2.3.2 Temperature Measurements

The UUTR is equipped with two independent instrumented fuel elements that monitor the fuel temperature in the core. **Figure 7.2-1** shows the cross section of the instrumented fuel element. The fuel temperature is displayed on the reactor console. Exceeding the set point will initiate a SCRAM. The **UUTR TS 2.1** defines the safety limits.

The peak fuel temperature for a stainless steel cladding, high hydride fuel element shall not exceed 1,000 °C (1,273.15 °K) under any conditions of operation. The peak fuel temperature in an aluminum cladding low hydride fuel element shall not exceed 500 °C (773.15 °K) under any conditions of operation. The **UUTR TS 2.2** defines the limited safety system settings to ensure the safety limits are never reached. For a core composed entirely of stainless steel cladding, high hydride fuel elements or a core composed of aluminum cladding, low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in **Table 7.2-1**. For a core containing flux traps, the limiting safety system settings given in the above table must be applied to the anticipated hottest fuel element in the core. In addition, **TS 3.2.2 and 3.2.3** requires measuring one fuel temperature for operation with SCRAM settings at or below the limited safety system setting.

The fuel temperature monitoring channels consists of a K type thermocouple and an Omega CN9000A temperature controller. The useful range is 0 °C (273.15 °K) to 800 °C (1,073.15 °K) with a ±1 °K accuracy. The controllers are configured for on/off control. The output relays on two fuel temperature controllers that are connected in series in a normally energized closed circuit to provide the fuel temperature SCRAM function. Fuel temperatures are calibrated annually.If the thermocouple calibration procedure indicates that any temperature channel is incorrect by more than one or two degrees, the deviation can be corrected by adjusting the offset or gain constants internal to the controller [described in the CN9000A manual for this procedure]. Handling of the thermocouple wires and connectors may result in thermal gradients, which will slightly disturb the channel readings. Temperatures must be given time to stabilize (several minutes) after any handling of the wiring before calibration may proceed. Fuel temperatures channels are checked during each run and compared to previous runs of the same power level as early check for drift or temperature channel malfunction. Fuel temperature set-points for the SCRAM function are set at 200 °C (473.15 °K) for 100 kW operation.

The most important parameter for a TRIGA reactor is the fuel element temperature. A loss in the integrity of the fuel rod cladding may arise occur if there is a buildup of excessive pressure between the fuel moderator and the cladding, and the fuel temperature then exceeds the safety limit. Such pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel moderator. The fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy determine the magnitude of this pressure.

The safety limit for the high hydride TRIGA fuel is based primarily on experimental evidence obtained during high performance reactor tests on this fuel. These data indicates that the stress in the cladding due to hydrogen pressure from the disassociation of zirconium hydride will remain below the stress limit, provided that the temperature of the fuel does not exceed 1,150 °C (1,423.15 °K) and the fuel cladding is water-cooled (See GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227 for more detailed information).

[redacted]

Figure 7.2-1 Instrumented fuel element

Table 7.2-1 Location and limited safety settings for stainless steel cladding and aluminum cladding

Location of Instrumented Fuel Rod	Limiting Safety System Setting for stainless steel cladding	Limited safety system setting for aluminum cladding
B-hexagonal ring	800 °C (1,073.15 °K)	460 °C (733.15 °K)
C-hexagonal ring	755 °C (1,028.15 °K)	435 °C (708.15 °K)
D-hexagonal ring	680 °C (953.15 °K)	390 °C (663.15 °K)
E-hexagonal ring	580 °C (853.15 °K)	340 °C (613.15 °K)

The safety limit for the low hydride fuel elements depends upon avoiding the phase change in the zirconium hydride that might cause excessive distortion of a fuel element. This phase change takes place at 530 °C (803.15 °K) as shown by the phase diagram in **Figure 4.5-9**. Additional information is given in "Technical Foundations of the TRIGA" Report GA-471, pages 63-72, August 1958. After reviewing all safety analysis conducted for the preparation of **UUTR SAR 13** of this report (Accident Analysis), we conclude that no credible accident can cause melting in the fuel's cladding.

For the stainless steel cladding, high hydride fuel element, the limiting safety system settings that are indicated represent values of the temperature that, if exceeded, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by fuel elements designed for this purpose, the limiting settings are given for different locations in the fuel array. Under these conditions, it is assumed that the core is loaded such that the maximum fuel temperature is produced in the B-hexagonal ring.

The margin between the safety limit of 1,000 °C (1,273.15 °K) and the limiting safety system setting of 800 °C in the B-hexagonal ring was selected to ensure that conditions would not arise that would allow the fuel element temperature to approach the safety limit. The safety margin of 200 °C (473.15 °K) accounts for differences between the measured peak temperature and calculated peak temperature encountered during operation and for uncertainty in temperature channel calibration. The thermocouples that measure the fuel-moderator temperature are located approximately midway between the fuel axial center-line and the fuel edge.

During steady-state operations, the equilibrium temperature is determined by the power level, the physical dimensions and properties of the fuel element, and the parameters of the coolant. Because of the interrelationship of the fuel-moderator temperature, the power level, the changes in reactivity required to increase or maintain a given power level, any unwarranted increase in the power level would result in a relatively slow increase in the fuel-moderator temperature. The margin between the maximum setting and safety limit would assure a shutdown before conditions could result that might damage the fuel elements.

For low hydride fuel element, the 460 °C maximum limit for the safety system setting gives an ample margin that assures that the safety limit would not be reached through errors in measurement. Temperatures of 460 °C have been shown to be safe through extensive operating experience.

The UUTR meets the requirements for safe operation as required by the NRC by its utilization of the two scrammable fuel temperatures, both of which are verified as operational at startup. The fuel temperatures are also checked during each reactor run and compared to a previous reactor run at the same power level.

7.3 Reactor Control System

7.3.1 Control Rod Drives

The three control rods are positioned by control rod drives mounted on the top of the reactor center channel. The control rods are positioned by standard TRIGA electrically powered rack and pinion drives as shown in **Fig. 7.3-1 and Fig. 7.3-2**. All rods and rod drives are identical and operate at a nominal rate of approximately 24 inches per minute. Limit switches mounted on each drive assembly to stop the rod drive motor at the top and bottom of travel and provide switching for console indication that shows:

1. When the magnet is in the up position.
2. When the magnet (and thus the control rod) is in the down position.
3. When the control rod is in the down position.

A key-locked switch on the reactor console power supply prevents unauthorized operation of all control rod drives.

The rod drives are connected to the control rods through a connecting rod assembly. These assemblies contain a bolted connection at each end to accept the control rod at one end and the control rod drive at the other. The grid plates provide guidance for all control rods during operation of the reactor. No control rods can be inserted or removed by their drives in such a way that the rod would be disengaged from the grid plate.

Each drive consists of a stepping motor, a magnet rod-coupler, a rack and pinion gear system, and a ten-turn potentiometer used to provide an indication of rod position. The pinion gear engages a rack attached to a draw-tube which supports an electromagnet.

The magnet engages a chrome-plated armature attached above the water level to the end of a connecting rod that fits into the connecting tube. The connecting tube extends down to the control rod. The magnet, its drawtube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the control rod drive mounting plate with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod drive assembly. The lower section of the barrel contains an air snubber to dampen the shock of the scrammed rod. In the snubber section, the control rods are decelerated through a length of 3 in. The control rod can be withdrawn from the reactor core when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized and the armature is released.

The rod drive motors are stepping motors driven by a translator. The speed of the rods is adjustable and rods are normally set to insert or withdraw the control rods at a nominal rate

24 inches/min. The unique characteristics of a stepping motor/translator system are used to provide fast stops and to limit coasting or over-travel.

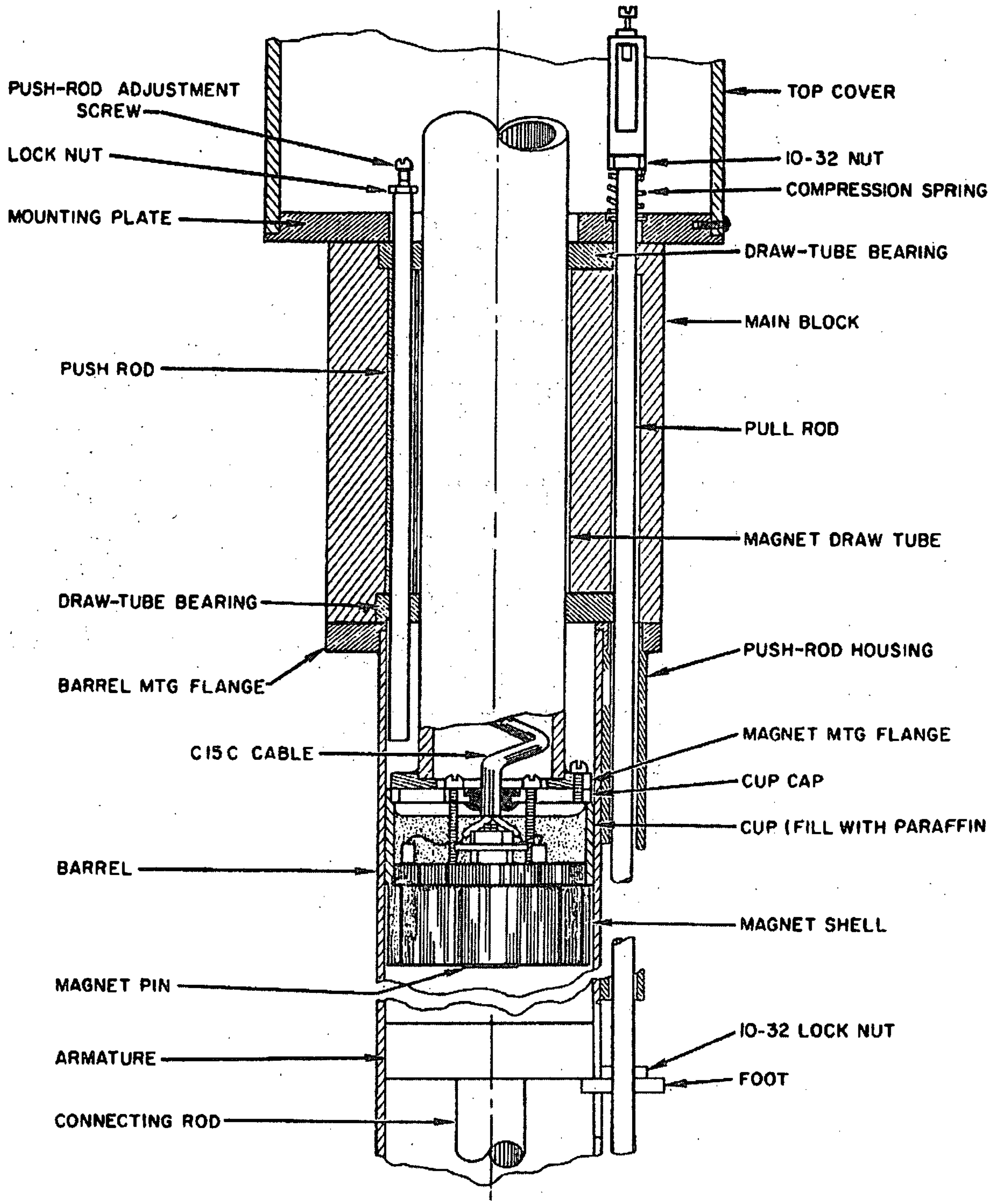


Figure 7.3-1 Control rod drive mechanism

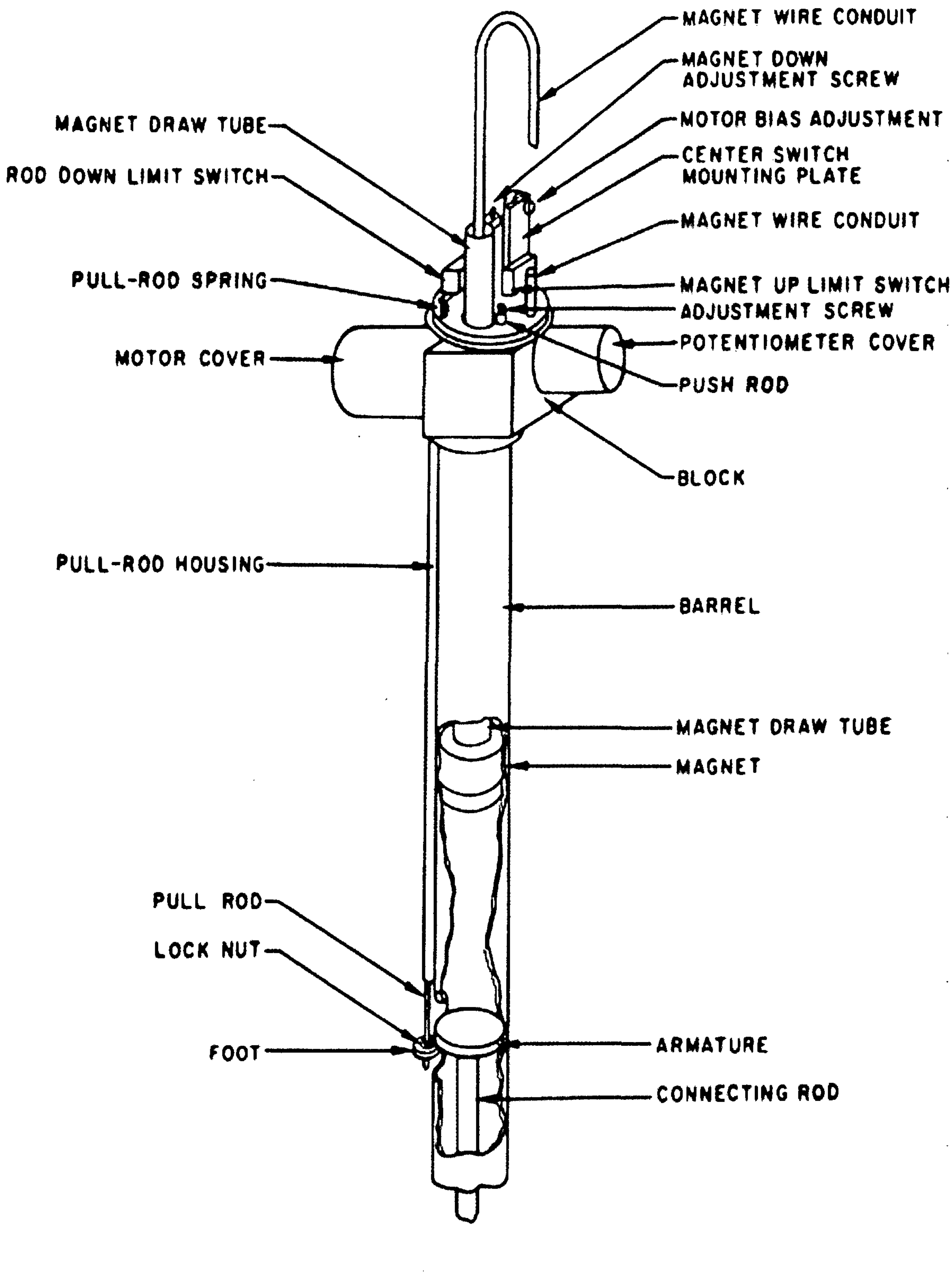


Figure 7.3-2 Control rod drive mechanism, showing components and adjustment locations

A 110 v, 60 cps, two-phase motor drives a pinion gear and a 10-turn potentiometer. The potentiometer may be employed to provide rod position indications. The pinion engages a rack attached to the magnet draw tube. An electromagnet, mounted on the lower end of the draw tube, engages an iron armature that screws into the end of a long connecting rod which terminates at its lower end in the control rod. The magnet, armature, and upper portion of the connecting rod are housed in a tubular barrel that extends well below the reactor water line. Located part way down the connecting rod is a piston equipped with a stainless steel piston ring. Whereas the upper portion of the barrel is well ventilated to allow free movement of the piston in the water, the lower 2 inches of the barrel has graded vent ports to restrict piston velocity.

Clockwise (as viewed from the shaft end of the motor) rotation of the motor shaft rotates the pinion, thus raising the magnet draw tube. If the magnet is energized, the armature and connecting rod will rise with the draw tube, so that the control rod is withdrawn from the reactor core. The piston moves up with the connecting rod. In the event of a reactor scram, the magnet will be de-energized and will release the armature. The connecting rod, piston, and control rod will then drop, thus reinserting the control rod into the reactor. Since the upper portion of the barrel is well ventilated, the piston will move freely through this range. However, when the connecting rod is within 2 inches of the bottom of its travel, the piston is restrained by the dash pot action of the restricted ports in the lower end of the barrel. This restraint cushions bottoming impact.

7.3.2 Limit Switch

A spring-loaded pull rod extends vertically through a housing and up through the block. This rod terminates at its lower end in an adjustable foot that protrudes through a window in the side of the barrel. The foot is placed so as to be depressed by the armature when the connecting rod is fully lowered. Raising the rod releases the foot, allowing the pull rod to be driven upward by the force of the compression spring. The top of the pull rod terminates in a fixture that engages the actuating lever on a micro switch. As a result, the micro switch reverses position according to whether or not the armature (and control rod) is at its bottom limit. This micro switch is the rod down switch.

A push rod extends down through the block into the upper portion of the barrel. It is arranged so as to engage the top surface of the magnet assembly when the magnet drawtube is raised to its uppermost position. The upper end of the push rod is fitted with an adjustment screw that engages the actuator of a second micro switch. Thus, this micro switch reverses position according to whether the magnet is at or below its full up position. This micro switch is the magnet up switch.

A bracket, fitted with an adjustment screw, is mounted on top of the magnet drawtube. A third micro switch is arranged so that its actuating lever is operated by the adjustment screw on the bracket. The switch will thus reverse position according to whether the magnet drawtube is at or above its completely depressed position. This micro switch is the magnet down switch.

The circuit associated with the three micro switches provides Limit contacts for the motor and the control-rod enunciator system, which consists of three indicator lamps for each rod drive.

During normal operation, two points receive line power through the normally closed control rod UP and DOWN pushbuttons, which provide the dynamic braking. Depressing the UP button opens the line at a single point. This permits the line current to flow through the DOWN button to the second point through a 1 mf phase-shifting capacitor. The phase difference at the motor windings causes the motor to rotate in a clockwise direction. Counter clockwise motion is obtained when the control rod DOWN button is depressed.

The unconventional circuit employed in the rod-drive system minimizes the number of switch contacts required. Therefore, relays, with their attendant reliability problems, are not required. It should be noted that all rod-drive units are identical both mechanically and electrically: they are, therefore, interchangeable.

The motor coupling is attached to the motor shaft by a single 8-32 dog-point setscrew. Both the pinion gear and the potentiometer coupling are pinned to the motor coupling. To prevent the follower potentiometer from supporting any of the pinion-gear load, the potentiometer coupling runs in an outrigger bearing. The follower-potentiometer shaft is connected to the potentiometer coupling by a single 6-32 setscrew. An oil-saturated, felt vapor seal restricts the entrance of water vapor into the follower potentiometer. Gravity loading of the rack against the pinion ensures minimum backlash between the rack and the follower potentiometer. Both motor and follower potentiometer are fully enclosed in metal case.

7.3.3 Interlock System

The push buttons for the control rods are interlocked in such a way that all three rods can be lowered at the same time in order to shut down the reactor quickly without a scram, yet only one of the control rods can be raised at a time due to an electric interlock between the three raise switches **(UUTR TS 3.2.3)**. This prevents excessive reactivity from being inserted into the reactor in a short amount of time.

The minimum source-count interlock relay prevents the withdrawal of all rods **(UUTR TS 3.2.3)**. The source interlock operates in the following manner. Part of the line is connected to the UP pushbuttons of each rod drive through a source relay. If minimum source count is reached, the relay is de-energized and the interlock makes the switches inoperative.

7.4 Reactor Protection System

7.4.1 Scram Circuit

Both the manual and automatic scram functions of the reactor console are designed to provide safe shut down of the reactor under any credible situation. The manual option allows the operators to initiate a scram at their own discretion. Likewise, should the console itself malfunction an automatic scram will be initiated, resulting in loss of console power and thus, of magnet power.

The **UUTR TS 3.2.3** states the reactor shall not be operated unless the safety system channels (Manual, and Key Scrams, Loss of Console Power Scram, Loss of High Voltage Scram) are operable. A channel check of each of the above reactor safety system channels shall be performed before each day's operation or before each operation extending more than 1 day, except for the pool level channel which shall be tested monthly or at intervals not to exceed six weeks. All reactor safety channels shall undergo a channel test and a channel check after any maintenance or modification.

The reactor SCRAMs are normally closed circuits that, when interrupted, cause the reactor control rods to be dropped back into the core thereby shutting the reactor down. In general, each SCRAM is a normally closed relay contact that is connected from terminal strip #1 to terminal strip #2. The relay for the linear channel SCRAM, for example, is located in the linear channel display controller. All SCRAM logic is executed at 120 VAC for historical reasons (the relays used have 120 VAC coils). Disconnect console power to service. Each SCRAM has its own relay that is set up to latch itself on with the current flowing through the normally closed SCRAM relay. Once a SCRAM has occurred, a contact of this relay serves to turn on the appropriate SCRAM indicator on the center console. The scrams can be reset by momentarily turning the Reset key to the Reset position. When the console power is first turned on all of the SCRAM lights will be activated so that the operator can check that none of the bulbs have burned out. All relays used in the SCRAM logic are normally energized so that a local loss of power or loose connection will cause a SCRAM rather than possibly preventing one.

7.5 Engineered Safety Features Actuation System

There are no engineered safety feature actuation systems.

7.6 Control Console and Display Instruments

7.6.1 Console Data Recorder

A Fluke 2620 multichannel voltmeter is connected to all of the temperature, power and rod drive indicator channels in the console. Physically the unit resides in the back of the console above the circuit bin. The data acquisition system provides an easy method for maintenance personnel to examine critical signals in the console, and it can also be connected to a computer via an IEEE-488 or RS232 link to record and display reactor operations data. The unit has 20 channels (plus a channel zero at its front panel banana plugs) that can be configured for VDC, VAC, resistance, frequency, current, and thermocouple measurements. For more information on the operation and programming of the 2620, see the manual in the equipment filing cabinet. Some of its maintenance uses are:

a) To double check the temperature monitor displays, using the thermocouple function
b) To check for a short in a thermocouple (particularly an instrumented fuel element) using the resistance function
c) To check for excessive noise on any of the power, temperature, or rod position lines using the VAC function
d) For general trouble shooting in the console by plugging the test leads into the front panel, and using channel zero.

7.6.2 Rod Condition Indication

LED readouts are provided to indicate the position of each of the control rods in the core. The readout indicates, by percentage, how far the control rod is out of the core. The push button switches are also lit to indicate the status of the control rod. The lower button (red) is lit when the control rod is fully inserted into the core. The raise button is lit when the control rod is in any position other than fully inserted.

7.6.3 Annunciator Panel

When the alarm is received at the annunciator panel, an audible signal will sound and an annunciator light will be on and the reactor will scram automatically. This annunciator can be reset in the normal fashion.

7.7 Radiation Monitoring Systems

7.7.1 Area Radiation Monitors

The UUTR has two important radiation detection systems to ensure that the reactor will operate within established guidelines **(UUTR TS 3.7.1)**. These systems are the Area Radiation Monitors System, and the Continuous Air Monitor System. The Area Radiation Monitors are scrammable channels. Readouts from two different radiation monitoring systems are a part of the reactor console. The Area Radiation Monitors (ARMs) display the radiation levels present at four strategic areas in the UUTR. Should the limiting safety system setting of 10mR/hr be exceeded, the high radiation alarm will be triggered. The Continuous Air Monitor (CAM) draws air from the facility ventilation system and tests it for radioactive noble gas, radioactive Iodine, and radioactive airborne particulates. Any alarm will sound at the console and at the CAM if the set points for this unit are exceeded.

The Area Radiation Monitor (ARM) is located on the TRIGA control console. Remote detectors are placed in areas of the UUTR where personnel will be working with radioactive material. These locations are: the reactor ceiling, reactor tank, the stack, and the counting lab. In the event that radiation levels exceed 10 mR/hr at any location, the high radiation alarm will be activated. This sends a signal [redacted] that will implement the following:

- SCRAMS the reactor (if in operation)
- [redacted]
- Closes the inlet air damper
- Activates audible and visual High Radiation alarms

Closing the inlet damper for the CAA (Control Access Area that includes control room, reactor room, and fuel inspection room) is done automatically by sending a 12 VDC pulse from the ADT system to the damper motor control box. The 12 VDC pulse energizes L-1 (24VAC). L-1 closes and latches LR-1 (latching relay 1). This results in the de-energizing of the damper motor, thus closing the UUTR facility ventilation. This resets LR-1 and closes the circuit to provide power to the damper motor, thus opening the damper. Once the alarm has been implemented, the enunciator can be secured by using the damper bypass key (this only stops the audible and visual alarm). In the event of main power failure, a battery backup system takes over for a limited amount of time. For extended periods of time, power is taken from the back up generator located on a concrete pad outside of the UUTR lab.

7.7.2 Continuous Air Monitor

The Continuous Air Monitor is located in the radiochemistry lab (**Fig. 5.6-1**). For a complete description of its function, see the manual in the equipment filing cabinet. A cable consisting of multiple twisted pairs connects the CAM with the displays on the console left front panel. Inside each of the three CAM modules, an op amp drives a current setting potentiometer that is in series with the respective current sensitive meter on the console. By holding the particular CAM module in the calibrate mode one can verify that the reading of the meter at the console agrees with that on the CAM. Calibrating the CAM should not change the agreement of the CAM meter and the console meter. Adjust the current setting potentiometer in the CAM if, for some reason the two do not agree. The CAM alarm relays have been configured to turn on the red light DS22 and sound an alarm BZ2 (both 120 VAC) at the console if any of the three CAM monitors exceeds its setpoint. An alarm and light will also be activated at the CAM itself. The green light DS23 is normally on and will only turn off if one of the CAM channels goes below its lower setpoint (usually indicative of a defective detector).

Equipment specifications are:

Area Radiation Monitors:

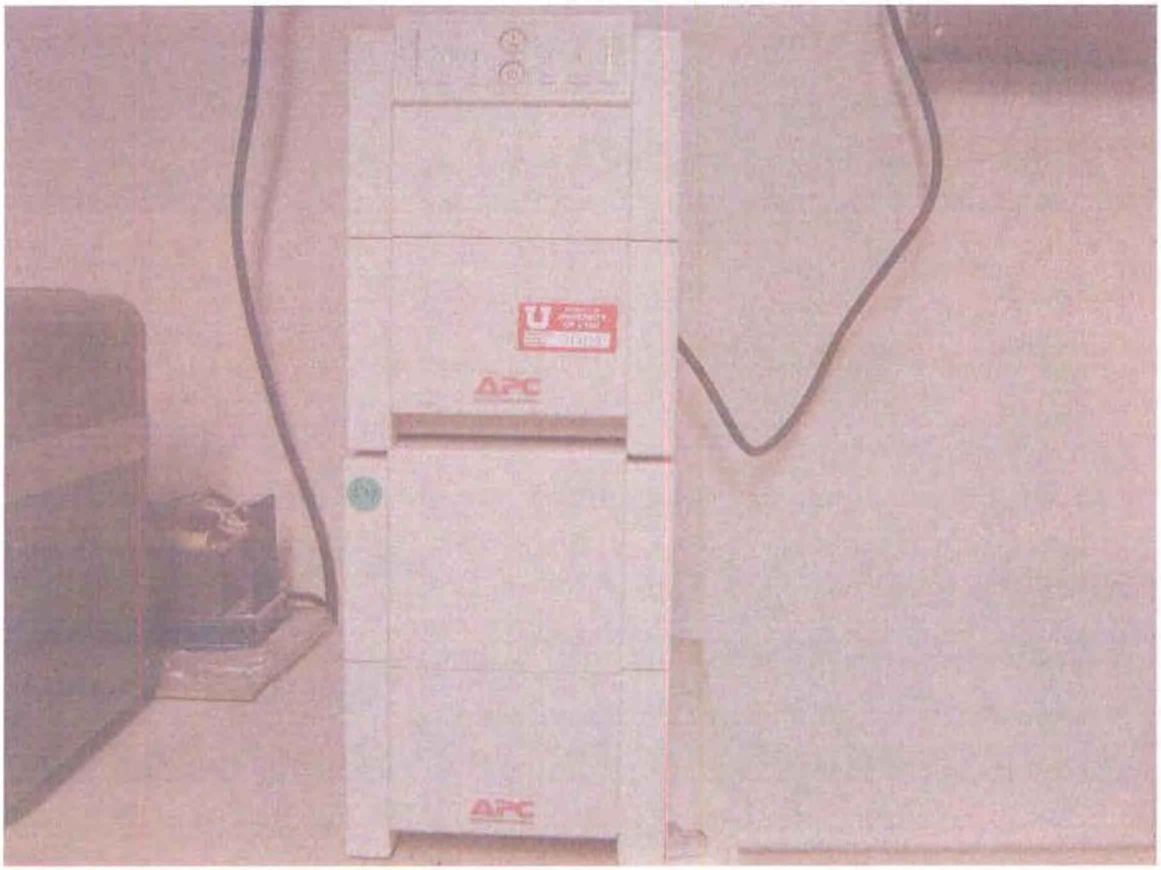
- Useful Range: 0.01 to 100 mR/hr
- Sensor Type: Geiger-Mueller Tube
- Accuracy: ±20% for gamma energies from 40 keV to 2.5 MeV
- Calibration Interval: Annual
- Safety Functions: Exceeding 10 mR/hr causes SCRAM (Channel can be bypassed)

Continuous Air Monitors:

- Useful Range: 1 to 10^5 CPM
- Sensor Type: Geiger Mueller Tube except for Iodine channel which is a NaI crystal
- Accuracy: ±10%
- Calibration Interval: Annual
- Safety Functions: Exceeding 1,000 CPM causes alarm

When the Continuous Air Monitor is operating, it samples the reactor room exhaust air prior to the HEPA filter located in the ventilation system, providing information on the levels of Ar-41, particulates, and Iodine in the reactor facility. The ARMs provide a continuous evaluation of the radiation levels in the reactor facility and provide warning alarms when the radiation levels exceed anticipated levels. The ARM located in the Counting Laboratory provides a continuous evaluation of the radiation level at that location and provides warning alarms when the radiation levels exceed anticipated levels. Experience has shown that monthly verification of area radiation and air-monitoring setpoints in conjunction with annual calibration is adequate to correct for any variation in the system caused by a change of operating characteristics over a long time span.

8. ELECTRICAL POWER



8.1 Normal Electrical Power Systems

The electrical power for the UUTR is supplied from the Merrill Engineering Building's electrical power system. The electrical power provided for building lighting and reactor instrumentation is single-phase, 60 Hz, 120/240 V. The reactor room has its own independent circuit panel that is controlled and monitored by reactor personnel. The design and safety equipment of the UUTR does not require building electrical power to safely shut down the reactor, nor does the UUTR require building electrical power to maintain acceptable shutdown conditions. **Figure 8.1-1** shows the electrical power system control box for the Merrill Engineering Building.

Figure 8.1-1 The control box for the electrical power system for the Merrill Engineering Building

8.2 Emergency Electrical Power Systems

The reactor will scram in the case of a building electrical power interruption. The emergency power is not required to maintain the reactor in a safe shutdown condition. The radioactive decay heat generated even after extended runs in the core following a scram is not sufficient to cause fuel damage. Power for the radiation monitors and the facility intrusion detectors is supplied by an uninterruptible power supply (UPS) within the reactor room. In the event of an electrical outage, this UPS supplies the necessary power for the operation of these instruments for a minimum of 24 hours. Battery-powered emergency lighting is also available to facilitate personnel movement during a power outage. **Fig. 8.2-1** shows the UPS system for emergency electrical power.

Figure 8.2-1 Uninterruptible Power Supply (UPS) for emergency electrical power

9. AUXILIARY SYSTEMS

[redacted]

9.1 Heating, Ventilation, and Air Conditioning Systems

University plant operations provide the reactor lab with electricity, potable water, heating, and air-conditioning. University maintenance personnel maintain these services. Heating and cooling systems are designed such that they do not cause the interchange of the atmosphere within the reactor lab and surrounding areas. Its role is to reduce the consequences of fission products released from the fuel or other experimental facilities. The controlled ventilation system is designed to prevent the spread of airborne radioactive particulates into occupied areas outside of the reactor area and labs. The system removes with high-efficiency filters assuring that all releases (of gaseous and particulate releases) are monitored and discharged through the stack. The objective of the structure surrounding the UUTR is to ensure that provisions are made to reduce the amount of radioactivity released into the environment by maintaining a negative pressure within the reactor area during operation. Automatic shutdown of the ventilation system confines the free air volume of the reactor area during emergency conditions. Remote monitoring of the conditions within the reactor area can be conducted.

9.1.1 Heating and Air Conditioning

The UNEF area has two separate HVAC systems, one for the reactor room area (MEB[1] 1205) and one for the non-reactor related area (MEB 1206 office area). The HVAC system can be controlled from a special panel in the main office in the Merrill Engineering Building (MEB). The University of Utah Plant Operation office maintains and controls all HVAC systems of the Merrill Engineering Building. The reactor room area and non-reactor area have their own temperature controllers that can control the temperatures within $\pm 2\ ^{o}K$. The cooling tower for the UNEF and all other areas HVAC system is located on the roof of the MEB. **Fig. 9.1-1** shows the cooling tower for the MEB HVAC system.

9.1.2 Ventilation

The reactor room (MEB 1205 D, E, F, and G) has its own ventilation system that is independent of all other areas in the MEB. Fresh air is supplied into the reactor room and two fume hoods in the MEB 1205 F and G replace and circulate the air with the ratio

[1] MEB: Merrill Engineering Building that houses the Utah Nuclear Engineering Facilities (UNEF) such as the UUTR and its associated facilities

of 4 room exchange/hour to keep the negative pressure within the reactor room. These fume hoods are operational all the time. Automatic actuation of the ventilation system into limited intake mode helps to confine the free air volume of the reactor room during emergency conditions (as explained in **UUTR SAR 9.1.4**). The air from the reactor room is monitored from the stack monitor that is located at the MEB 1205 F. This exhaust system is connected to the roof of the MEB (~40 ft above the ground level). **Fig. 9.1-2** shows two exhaust motors for the UNEF ventilation system.

Figure 9.1-1 The cooling tower for the MEB HVAC system

9.1.3 Reactor Area Confinement

The floor of the reactor room is a ~6-inch concrete slab placed on a 6-inch compacted granular base. The MEB consists of precast-prestressed exterior wall panel and poured-in-place pilaster. The outside wall of the MEB has approximately 12-inch thickness. The MEB has a structural steel roof frame and insulating concrete fill, as well as structural steel interior floor frame with metal-formed concrete slab. A bridge crane with a 2-ton capacity serves the reactor room area. The first floor of the MEB contains UNEF, Mechanical engineering laboratories and offices, classrooms, chemical engineering laboratories, and nano-fab laboratory. The second floor occupied mostly offices and laboratories mechanical engineering, electrical engineering and chemical engineering. The forth floor contains offices. The ceiling of the area directly above the reactor core has four inches of concrete and 3/l6 inch of steel (floor support.) The third floor above the reactor area is comprised of office and laboratory space.

Figure 9.1-2 Two exhaust motors for the UNEF ventilation system

9.1.4 Ventilation System Modes of Operation

There are two different operation modes of the ventilation system: operational mode and limited intake mode during the emergency situations. A damper system will be closed automatically during the emergency conditions such as when the high radiation alarm is initiated within the reactor room area. During such an emergency situation no fresh air will be supplied into the reactor room, but the exhaust air will go through the HEPA filters. However, there could be a situation where the operator could decide to manually close the flow of the exhaust air through the HEPA filters; if that happens, the ventilation system will be in a complete shutdown condition (or in "off" condition). This most unlikely situation is however considered as the most conservative for some of the accident analysis as presented in **UUTR SAR 13**.

9.1.4.1 Operational Mode

In the operational mode, the air in the controlled access area (reactor room area) is constantly being exchanged. The air leaving the facility has a volumetric flow rate of more than 100 CFM per each fume hood. Currently, the UNEF is equipped with two fume hoods located in the MEB 1205 F room, and MEB 1205 G room. The result of this is a negative pressure of greater than 0.01 inches of water in the reactor room. The UNEF ventilation system draws its supply air from the MEB's main ventilation air system, as shown in **Figures 9.1-3 and 9.1-4**. Prior to the entry into the reactor room, supplied fresh air passes through one of the two pre-filters. This reduces the concentration of dust in the air so that the pool water contamination can be kept at a minimum. Then, air mixes with the current air in the reactor room and migrates to the reactor room where it passes through one of the two pre-filters and then HEPA filters. At this point, a continuous air sample is diverted from the ventilation system to the Continuous Air Monitor (CAM). Here, airborne activity is determined and relayed to the TRIGA control console. The same data is also stored permanently on a chart recorder of the CAM system **(Fig. 5.6-1).**

9.1.4.2 Limited Intake Mode

In the event when the airborne activity in the UNEF reactor room exceeds the pre-set level of 10 mrem/hr, the ventilation system will switch to a limited intake mode. This mode provides an enhanced negative pressure to the reactor room of greater than 0.1 inches of water by closing the damper. In this mode, no fresh airflow through the ventilation system is allowed. **Figure 9.1-4** shows schematics of the UUTR's ventilation system indicating the location of a damper. In the limited intake mode, the damper will be closed ensuring that all potential airbone radionuclides will be contained within the facility. In this case, the exhaust air will then go through the HEPA filters. From there, the air is monitored by the CAM, which records the radiation level of any potential release of radioactive material.

Figure 9.1-3 Main ventilation air supply system to the reactor room

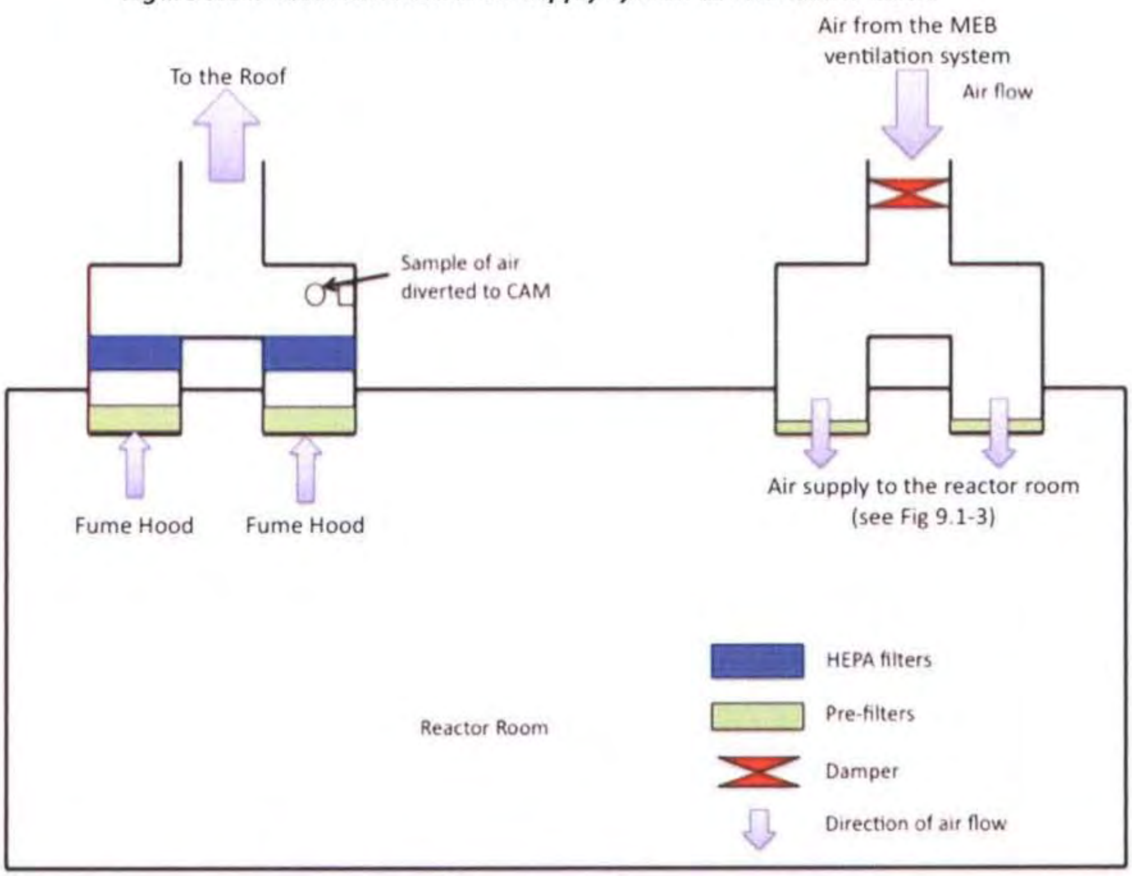


Figure 9.1-4 The UNEF ventilation system; in the limited intake mode, the damper will be closed

9.1.4.3 Complete Shutdown of the Ventilation System

The ventilation system shall work all the time even when the damper is closed. The ventilation system can be shutdown for the maintenance when replacing a motor or a fan belt. In this mostly unlikely situation for the UUTR facility, the damper will be closed and ventilation system will be completely shutdown (or in "off" condition). The ventilation system can be shutdown manually from the reactor control console by pushing ventilation switch or, again manually, by cutting the power from the mechanical room located on the 3rd floor near the roof in Merrill Engineering Building. There will be no fresh air into the reactor room and all air and gaseous radioisotopes will be contained within the reactor area. In this case, no exhaust air will go through the HEPA filters. This condition of ventilation system is considered as the most conservative in the analysis of some of the accidents as presented in **UUTR SAR 13**.

9.2 Handling and Storage of Reactor Fuel

9.2.1 Fuel Handling

The fuel handling system provides a safe and effective means for transporting and handling the reactor fuel from the time it enters the boundaries of the UUTR facility until it leaves. The fuel handling cycle within the UUTR consists of:

1. receiving fresh, unirradiated, fuel elements

[redacted]

[redacted]

9.2.2 Fuel Handling Equipment

9.2.2.1 Fuel Handling Tool

Tools are provided for handling individual fuel elements and for manipulating other core components. Individual fuel elements are handled with a flexible or rigid handling tool. The FEHT utilizes a locking ball-detent grapple to attach to the top end fitting of a fuel element.

9.2.2.2 Overhead Crane

The UUTR reactor room has 2-ton capacity overhead crane system; its use is administratively controlled. An overhead crane running on tracks provides the capability for movement of heavy objects (including the handling of the fuel element cask and removing steel weight from the fuel storage pit anywhere in the reactor room. The crane has a capacity of 2,000 kg and is locally controlled from a pendant box but can be computer controlled. The crane system ss shown in **Fig. 9.2-1** was upgraded in 2005. The loading test was performed after the crane was upgraded. A 2 ton-steel box was used to test the crane system. During this test, the bending of the mono-rail system on the ceiling and hoist cable strength were examined and the use of crane approved.

Figure 9.2-1 The UUTR overhead crane system

[redacted]

[redacted]

Manufacturer checks the crane system every year. The crane system is secured when it is not in use; only a reactor supervisor can access and control the crane system. When the crane system is in use, at least two SRO (Senior Reactor Operator) and at least two additional staff personnel are required to be present. The crane is operated in accordance to ANSI B30.11, Monorail Systems and under hanging Cranes.

[redacted]

Figure 9.2-3 Inside of the center fuel storage pit single fuel rack can hold up to 18 fuel elements

Figure 9.2-4 Fuel element cask

9.2.3 Administrative Controls Used to Ensure Safe Fuel Handling

All movement of the irradiated reactor fuel is done as determined by UNEP form-005 "*Core Change and Critical Fuel Loading*" and UNEP form-004 "*Biennial Fuel rod Inspection*." These procedures require that a detailed written and approved movement and/or change schedule be prepared prior to all core changes or fuel rod movement. It is inappropriate to include such procedures in a SAR since they change with time. The key points involved are:

1. Presence of a Senior Reactor Operator to supervise the operation **(UUTR TS 6.1.3)**
2. Insuring that all reactor control systems are on and functioning properly **(UUTR TS 3.2.2 and UUTR TS 3.2.3)**
3. The reactor will remain subcritical during the operation
4. An accurate log of the operation is maintained
5. Minimization of personnel radiation exposure
6. Stored fuel maintained in geometry with a multiplication factor of 0.9 or lower **(UUTR TS 5.4)**

[redacted]

7. Resultant core meets all license and Technical Specification requirements **(UUTR TS 3.1.2, UUTR TS 3.1.3 and UUTR TS 3.8.1)**.

Fuel not in the core is always stored in one of the storage racks that have been designed for "always safe geometry" with a k_{eff} of 0.9 or less when filled with fuel rods. All fuel rods are inscribed with suitable identifying numbers and an accurate log of the location of all fuel at the facility is maintained. [redacted] The facility maintains a computerized Special Nuclear Material (SNM) record system that keeps track of fuel including burnup. Semi annual material status /balance report and burnup ratio calculations are based on this computerized SNM record system.

All aspects of the use and control of SNM at the UUTR facility receipt of new fuel and the shipment spent fuel is covered in the UUTR *Special Nuclear Materials Accountability Plans**. The plan has been written to cover all the SNM related requirements in 10 CFR Parts 40, 70, and 150. The plan has been reviewed by the U.S. NRC and covers:

1. Introduction and definitions
2. Accountability responsibility
3. Fuel storage
4. Physical inventory
5. Core change log
6. Fuel rod records including identification number, fuel movement, and location
7. Semiannual material status/balance report
8. Transfer of SNM and
9. Loss or theft of SNM

9.2.4 Fuel Storage Racks

The in-tank fuel storage for the UUTR consists of seven, in-tank, aluminum fuel storage racks, with a combined capacity to accommodate [redacted] irradiated fuel elements, as shown in **Fig.9.2-6**. [redacted]

These storage racks are designed to meet the following criteria:

1. The UUTR in-tank fuel storage racks are designed to withstand earthquake loading to prevent damage and minimize distortion of the rack arrangement. The building housing the UUTR was designed to conform to Uniform Building Code Zone 3 criteria and the UUTR containment tanks and supporting structures also comply with Uniform Building Code Zone 3 criteria. The storage racks are parts of the UUTR core, so they are designed as well to withstand a Uniform Building Code Zone 3 earthquake (**Appendix 9.2.A**).
2. The UUTR in-tank fuel storage racks are designed with sufficient spacing between the fuel elements to ensure that the array, when fully loaded, will be substantially subcritical. (Storage requirements are $k_{eff}< 0.9$). The fuel storage rack is modeled in MCNP5 as shown in **Fig. 9.2-7** (**Appendix 9.2.B**) assuming the fuel storage rack is fully loaded with new SS fuel, which has the highest amount of U-235; this model is the most conservative giving the highest value for the k_{eff}. The MCNP5 model gives $k_{eff} = 0.43467 \pm 0.00020$. Therefore, the UUTR in-tank fuel storage racks are configured such that the criticality is not possible.

[redacted]

Figure 9.2-6 Fuel storage racks

[redacted]

Figure 9.2-7 MCNP5 model of fully loaded UUTR fuel storage rack

3. The UUTR in-tank fuel storage racks are [redacted] deep enough below the water surface – the water provides adequate radiation shielding. The radiation of fuel elements in the fuel storage racks comes from spontaneous fission of fuel and the possible fission products in the fuel elements. The stored fuels are not hot. Therefore the radiation of the stored fuel elements is very low.
4. In the unlikely event of total loss of reactor tank coolant water, the UUTR in-tank fuel storage racks modeled using the MCNP5 gives k_{eff} = 0.05537 ± 0.00005 (**Appendix 9.2.B**). Therefore, the UUTR in-tank fuel storage racks are configured such that the criticality is not possible.
5. In the unlikely event of earthquake that will "un-weld" the fuel storage racks and mounts together part of the existing fuel rods, the MCNP5 model as shown in **Fig. 9.2-8** (**Appendix 9.2.C**) predicts an extreme case when 28 fuel elements were stacked closely. The simulation gives k_{eff} = 0.79700 ± 0.00017. In case all water coolant leaked out of the pool tank, the MCNP5 model (**Appendix 9.2.C** - the water changed to air) gives k_{eff} = 0.44087 ± 0.00015. Therefore, the UUTR in-tank fuel storage racks are configured such that criticality is not possible.

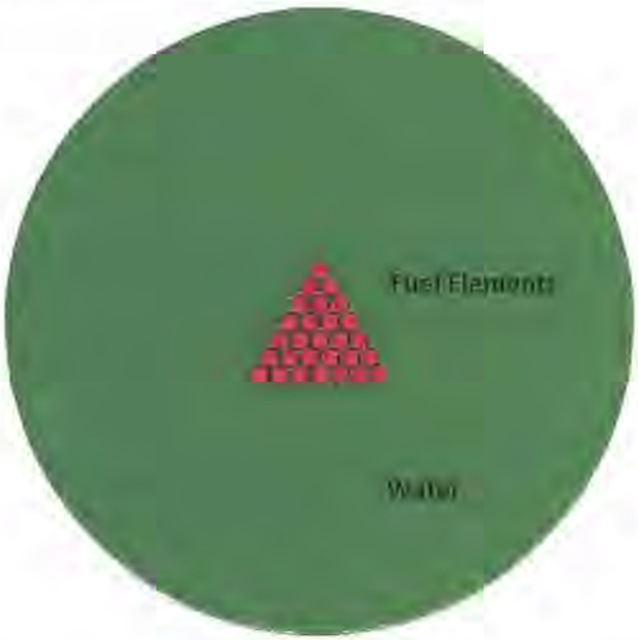


Figure 9.2-8 MCNP5 model of 28 stacked fuel elements

6. The UUTR in-tank fuel storage racks have a combined capacity for storage of a typical core loading of irradiated fuel elements.
7. The in-tank fuel storage racks are designed and arranged to permit efficient handling of fuel elements during insertion, removal, or interchange of fuel elements. The fuel elements are loaded into the in-tank fuel storage racks

from above. Each storage hole has adequate clearance for inserting or withdrawing a fuel element without interference. The weight of the fuel elements is supported by the lower plates of the racks. [redacted] This mounting arrangement prevents the racks from tipping or being laterally displaced.

9.2.5 Fuel Storage Pits

There is additional fuel storage at UUTR located in three fuel storage pit[redacted]

The storage pits are sufficiently spaced and shielded with hydrogenous material to insure that there is no possibility of neutron coupling between them. [redacted] **Figure 9.2-5** shows the storage rack that is used in the storage pit. Each pit has a liner and a lead-filled shield plug that is locked in place when fuel is not moved into or out of the pits. The pits have racks with holes for holding fuel elements. Each hole in the rack can only hold one fuel element. All storage pit material (liners, racks, plug casing, and pipes) that may contact either the fuel elements or the pit water is fabricated from aluminum or 304 stainless steel. This is the same type of material as used for the fuel element cladding and end fittings.

The fuel storage pits were designed with the following criteria:

1. The spent fuel storage pits are designed with sufficient spacing to ensure that the stored fuel array, EVEN when fully loaded, will be independent of each other and subcritical (satisfying the criteria for k_{eff}< 0.9). The simulation of three storage pits using the MCNP4C was performed based on the configuration at the facility. Two calculations were performed for each pit assuming existence of the water moderator and of the air moderator. The criticality calculation results are shown in **Table 9.2-1**. In every of the assumed scenarios the obtained value for k_{eff} is below required value of 0.9. For a fuel storage pit scenario assuming to be completely filled with water, control of spacing is not a criterion to limit the effective multiplication factor of the array (k_{eff}). An analysis shows that the largest k_{eff} for a pit is approximately 0.75 when all three pits are loaded to full capacity of 8.5 wt % fuel element[redacted] and are filled with water (0.218 when dry).

Furthermore, [redacted] elements are only 1/4 the number required to have a criticality and each pit is separated for about 1 ft, therefore, there should be no criticality issue related to the storage pits. Radiation levels at the reactor room floor level with either water in the storage pits or the lead plug in place are less than 3 μR/hr. Approximately 13 μR/hr are expected when the top shielding materials (0.5 inch-thick steel) were removed. This measurement is performed every month for regular "monthly checkout" using Ludlum Model 19 detector.

Table 9.2-1. Criticality calculations for three storage pits using MCNP4C

Moderator	South Pit	Center Pit	North Pit	99 % confidence
Air	0.21777 $\pm$0.00058	0.21777 $\pm$0.00058	0.21777 $\pm$0.00058	0.21124~0.21430
Water	0.74117 $\pm$0.00081	0.74117 $\pm$0.00081	0.74117 $\pm$0.00081	0.73904~0.74331

2. The spent fuel storage pits are designed to withstand earthquake loading to prevent damage and distortion of the pit arrangement. The spent fuel storage pits are designed to withstand horizontal and vertical accelerations due to earthquakes. Stresses in a fully loaded storage pit will not exceed stresses specified by the UBC Zone 3 seismic criteria.
3. The spent fuel storage pits are fabricated from materials compatible with the fuel elements and to provide adequate personnel shielding.
4. The spent fuel storage pits are designed and arranged to permit efficient handling of fuel elements during insertion or removal of fuel elements.
5. The spent fuel storage pits have shield plugs that can be locked in place.
6. The fuel elements are loaded into the racks from above. Each hole in the rack has adequate clearance for inserting or withdrawing a fuel element without interference. The lower plates of the racks that are supported by the pit liners support the weight of the fuel elements. Each rack is designed so that it is constrained by the pit liner and cannot tip or become laterally displaced.

The fuel temperature measurements at the UUTR operating at the power of 90kW showed that the average fuel temperature is approximately 65 °C (338°K), and that it would take about seven (7) days for the fuel elements to cool down to the temperature of close to 20° C (293.15 °K). Therefore, a significant decrease in fuel temperature will certainly take place in 10 days after shutdown prior to moving fuel elements to the storage. Therefore, allowing at least 10 days of cooling down time prior to transferring fuel to the storage pits will provide a safety margin.

9.2.6 Fuel Inspection

The UUTR performs the fuel inspection every TWO years (odd years as 2005, 2007, 2009, ...). The fuel inspection consists of the inspection of all fuel elements including storage pits, heavy water elements, graphite elements, thermal irradiator, and control rods.

9.2.6.1 Inspection of Fuel Elements

The biennial fuel inspection involves systematic removal, inspection and logging of each of the fuel elements. This procedure satisfies **UUTR TS 3.1.4, 4.1** and **5.1.3** for reactor operation. Therefore, it requires an operator at the console. During this procedure, the safety rod is withdrawn in order to be scrammed in the unlikely event of a reactivity transient that is indirectly detected through the reactor safety systems.

Minimal handling of the fuel is suggested because of the risk of dropping the fuel from the fuel handling tool to the tank floor where damage to the cladding or locator pin may occur. If a rod is dropped, the element would be re-inspected before moving to inspect another fuel element. A fuel rod may be lifted from the floor by using the "lasso" tool. To use this tool, the nylon cable must be tightened around the indented portion of the element directly below the fluted section. The fuel element is then immediately placed in a standard fuel location and then lifted for inspection using the fuel handling tool.

The fuel is inspected for defects including surface anomalies (spots or scratches of reddish brown, black, or white), cladding dents and bent pins. An element is considered damaged if it meets the criteria outlined in **UUTR TS 3.1.6.** Also, if an element releases bubbles directly from the cladding region during or after being raised near the surface for visual inspection, then it may be assumed that a pin hole leak was induced through depressurization. If a pin hole leak is suspected or observed, then the Reactor Supervisor will be notified and the procedures outlined below of this chapter shall be followed. After the fuel has been inspected and the results logged in the Operation Log, the results are transferred to the Fuel Log. It is important that accurate documentation is kept during this process, with complete entries and legible writing. **UUTR TS 3.1.6** states that the reactor will not be operated with damaged fuel and the leaking is considered to be a condition of damage. The only pathway for fission products to reach the reactor room is through the tank water. This pathway essentially scrubs the material and allows only a portion of the noble gases to escape the tank. Virtually all soluble materials are retained in the tank water.

A leaking fuel element is potentially very difficult to detect and recognize because of the low levels of activity associated with such a leak. There are two mechanisms of regular surveillance which may detect the release of active materials: the Continuous Air Monitor (CAM) and monthly spectroscopy of the tank water. The purpose of the CAM is to detect short-lived gaseous products from gross leakage. The purpose of the monthly spectroscopy is to detect and differentiate long-lived soluble products released in small quantities.

Gross Leakage. If the CAM readings indicate that a fuel element may be leaking, the reactor operator should contact the Senior Reactor Operator on duty and the RS. It is important to remember that an increase in the CAM reading may be caused by nearby sources, by temperature inversions capturing natural activity near the earth's surface or by low pressure systems causing a more rapid than normal release of natural active gases. To confirm the suspicion, the glass and charcoal filters from the CAM should be analyzed in the gamma spectroscopy system spectrum for the following noble gases or daughter products:

Isotope	Half-life	γ Energies (MeV)
Kr-85	4.4 hours	0.151; 0.304
Kr-87	76 minutes	0.402; 2.01
Kr-88	2.8 hours	0.196; 1.53
Xe-135	9.2 hours	0.249; 0.607

If the analysis yields positive results, the operations staff should perform the procedure described in the following section.

Pin Hole Leak. If the monthly water spectroscopy indicates that a fuel element may be leaking, the reactor operator should contact the senior reactor operator on duty and the reactor supervisor. It is important to remember that an increase in gross gamma counts may be caused by activation of short-lived trace contaminants. To confirm the suspicion, the gamma spectrum should be analyzed for the following isotopes:

Isotope	Half-life	γ Energies (MeV)
I-131	8 days	0.364; 0.640; 0.720
I-132	21 hours	0.500; 0.870; 1.400
I-133	54 minutes	0.860; 1.100; 1.780
I-134	6.7 hours	1.270; 1.800

If the analysis yields positive results, the operations staff should perform the procedure described in the following section.

Detection of a Leaking Fuel Element. Assuming a leaking fuel element has been positively identified, the operation staff will immediately inform the NRC of the existing condition, the basis for such a conclusion, and the procedures for resolving the problem. A standard detection scheme will be established to identify such a problem resolution. This scheme may involve operating the reactor for a period of two hours, then taking a sample of tank water to be analyzed in the gamma spectroscopy system for the content of the absorbed noble gases.

To identify the leaking element(s), the staff will obtain a peristaltic pump, 50 ft of 1/8" ID plastic tubing with a lead weighted 2" maximum diameter glass funnel securely fastened on one end. The funnel will be placed in the tank near the core. The other end is placed in the pump, wrapped around the germanium crystal approximately 5 times and terminated in a 5 gallon container. The peristaltic pump will move water through the tubing at a rate of approximately 1 foot/second to allow a delay time of approximately 30 seconds from core to detector.

The spectroscopy system will be tuned to count only the 151 KeV gamma from Kr-85m and the system will be operated in the MCS mode. While the reactor is operated, a member of the staff will move the funnel from element to element, documenting the exact time the funnel resided over each element. This data will be compared to the MCS data for identification of the element(s) releasing the highest Kr-85m activity.

The element(s) identified as leaking Kr-85m will be removed from the reactor core. The standard problem resolution scheme will then be performed to compare previous data with new data. The leaking element(s) will be removed from the tank at a convenient time and placed in the fuel storage pits. At the time of the next fuel shipment from the facility, the elements will be removed from the storage pits and sent to permanent storage.

The NRC will be informed of the problem resolution, the basis for the conclusion, and the supporting data. The reactor can then be placed into normal operation. The UNEP form 004 shows the procedure for fuel inspection.

9.2.6.2 Control Rod Inspection

It is necessary to perform this procedure in a careful manner in order to prevent reactivity transients, reduce personnel exposure, and prevent contamination. To inspect the control rods, it is required that the assemblies be removed from the tank and disassembled. It is important to note that removal of the negative reactivity associated with the absorber necessitates the removal of equal or greater positive reactivity in the form of fuel elements. After the fuel elements have been removed, the control rod guide tube must be detached from the grid plate. This is done using the allen head tank tool. The tool is very carefully placed in the screw head and turned counter-clockwise approximately 6-8 turns until the screw feels loose. While the screw is held captive in the guide tube, it is conceivable that if the screw is turned more than necessary, the threads may bite into the captive loop and be removed from the guide tube and lost within the grid structure. Under rare circumstances, the screw may rest on top of the grid plate and may be removed with a magnet.

The rod assembly is composed of an aluminum guide tube, an aluminum clad boron carbide absorber, an aluminum sliding coupling used as a shock damper, and stainless steel spring and locking screws. Each component is exposed to different flux levels. Experience shows that the aluminum, boron, and carbon that are exposed to the highest flux do not activate, or perhaps, the activity decays before the assembly is removed from the tank.

The highest activity components are the stainless steel spring and locking screws. These should be inspected immediately upon removal from the tank and the results must be recorded. The absorber is removed through the top of the guide tube by removing the cap. Once the absorber has been removed, the mechanism is inspected for corrosion pits, friction scratches, broken welds, and other defects. The device may then be assembled and checked for proper operation. The assembly is placed in the appropriate grid position, being careful to place the locator pin firmly in the lower grid plate, and then tightened to the grid. If the assembly appears to be inoperable any time before placing it back into operation and if the problem cannot be solved by the operations staff, then the Reactor Supervisor will be informed so that new components can be obtained. The UNEP form 002 shows the procedure for the control rod inspection.

9.2.6.3 Heavy Water and Graphite Element Inspection

The UUTR has 12 in-core heavy water elements and 12 graphite elements as a reflector. Five heavy water elements (F-21 through F-25) have the same end fixture as fuel element. Same fuel handling tool can be used to move these heavy water elements. Rest of the heavy water elements has a screw as an end fixture. A special tool for these kinds of elements was manufactured and is stored in the reactor room floor. The graphite elements have a same inspection procedure as fuel elements. The inspections for the graphite elements are performed at under water. Heavy water elements must be removed from the reactor pool using a rack as shown in **Figure 9.2-5**, and bring them to the fuel inspection room (MEB 1205 F). Maximum load of 6 elements in the rack are recommended. After visual inspection for outside surface of an element, the top screw must be opened very slowly because the inside of the heavy water element may be slightly pressurized. After open the lid refill the element with provided D_2O up to the top of the element. A reactor operator or a SRO must record all activities including element ID number, location, physical status, and the amount of heavy water refilled. When the inspection is completed, put the elements in the rack and lower the rack into the pool. Using the same handling tool put them into the original locations. The UNEP Form 020 shows the procedure for heavy water inspection.

9.2.6.4 Thermal Irradiator (TI) Inspection

The inspection procedure for TI is same as heavy water element. The TI is very heavy (~150 lbs), an operator or a SRO must make sure that the tool is engaged firmly on the TI. The first stage is to remove the dry tube from the TI and attach it on the side of the reactor tank. The dry tube should not be out of the water surface. When the TI is moved to fuel inspection area, unscrew the drain plug on the top of the TI and refill with provided D_2O. Carefully put the TI into the pool and place it on its original location. Dry tube can be inspected using under water camera system. After all inspection, the dry tube must be placed into the TI.

9.2.6.5 Summary of Biennial Fuel Inspection

The UUTR fuel inspection starts early December and it takes 2 weeks to complete. Control rod inspection requires special caution because it may be hot. During the control rod inspection, unnecessary personnel must stay away from the reactor room to avoid unexpected radiation dose. Three teams work in rotation to reduce their individual doses. All fuel elements and graphite inspection are performed under water and all heavy water elements inspection are done in the fuel inspection area that is located behind the reactor room. **Figure 9.2-9** shows the fuel inspection area. After fuel inspection is completed, thermal power calibration and control rods drop are required.

Figure 9.2-9 Fuel inspection area

Appendix 9.2.A

Typical Aluminum Angle Tab Connection For Support of the Fuel Rod Storage Racks

Location:
University of Utah
Nuclear Engineering Dept.
TRIGA Reactor Tank

Date: May 04, 2010
Under Direction of:
Prof. Tatijana Jevremovic
Evaluation Performed By:
Jason P. Rapich, SE (Graduate Student)

Summary:

The purpose of this summary is to address safety concerns as to the structural integrity of the support connection provided for the fuel storage rack[redacted]

Visual observation was performed on 05/03/2010 concerning the supportive connection of the aluminum angle tab connection provided for structural support of the fuel storage racks. The tab connection appeared to be of the approximate size and shape represented by the attached detail sheet. It was further observed that the connection of the tab was provided by a fillet weld on at least three of the four sides of the angle tab connection. The thickness of the angle tab appeared to be approximately ¼ inch. The effective throat thickness of the fillet weld appeared to be approximately 3/16 inch or greater. Both angles and associated welds appeared to be in good condition. No signs of damage or severe deterioration could be visually observed of the existing condition.

The actual rack size appeared to be approximately [redacted] The thickness appeared to be ¼ inch matching the thickness of the angle tab. The connection of the rack to the angle plate was provided by an approximate 3/16 inch thick fillet weld. The racks are provided with approximate 2 inch diameter holes approximately 4 inches on center for the length of the rack. The total weight of material support by the rack suspended in water is less than 50 lbs. The section of material provided for this support condition under a simple span scenario is sufficient for a load of 50 lbs or less.

The size of the welds provided and determined per the visual observation are within the guide lines given by AWS D1.2 for the thickness of welded material provided for the tab connection and rack for aluminum material.

The geological location of the reactor structure is currently defined by the IBC 2006 as a site design category 'D'. This is defined as a location that has the potential to experience large accelerations during a seismic event. As per the original documentation for the design of the TRIGA vessel tank structure the seismic location was determined to be a 'Zone III' per the UBC code governing during the early 1970's. This connection appears to be provided in accordance to applicable building and welding codes for the given time frame. Per the visual observation provided, no items were noted that would cause the connection to be insufficient per current IBC, and AWS construction codes.

Per the documentation of the construction of the TRIGA vessel the bottom of the vessel is provided 16 ft below top of grade of the TRIGA laboratory floor elevation. The TRIGA vessel is enclosed in a secondary steel vessel. There exists 2 ft of sand infill between the walls of the two vessels. Due to the depth of the embedment of the primary TRIGA vessel and the dampening effect of which the sand will play on any ground motion interaction due to seismic force effects, very little stress is anticipated for this connection within the TRIGA primary vessel. It is anticipated that the seismic force exerted on the vessel will dissipate though the opposing side of the vessel into the sand infill medium provided.

Due to the inherit dampening action of the vessels construction, it is reasonable to expect that the angle tab welded connection and support of the storage rack will perform well with out significant damage during a potential seismic effect.

Project No. ______ Page No. ______

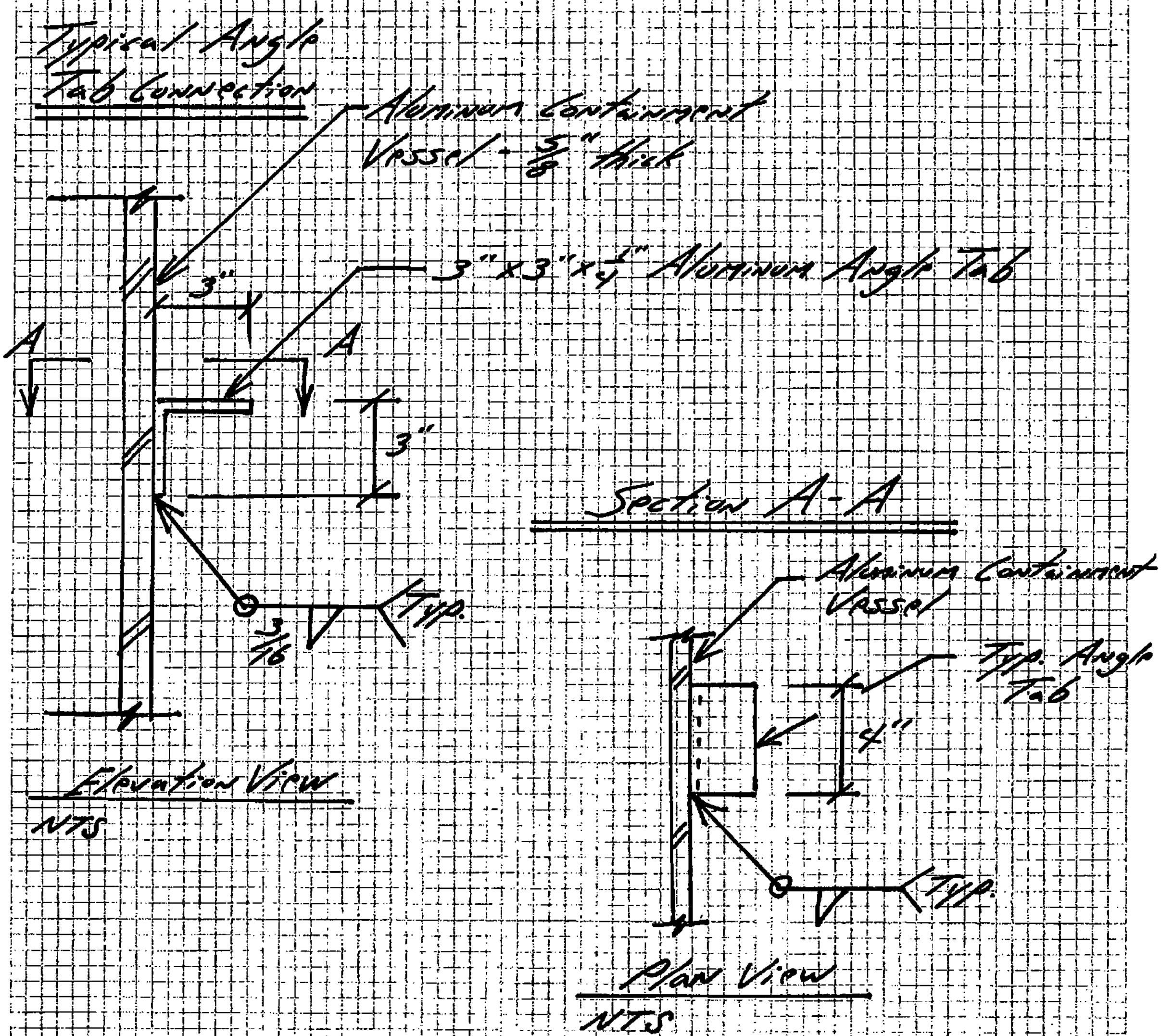
Subject: Typical Angle Tab Connection For Storage Rack Support within TRIGA Vessel

Computed by: JAR Date: 5-3-10

Checked by: ______ Date: ______

Approved by: ______ Date: ______

Sheet No. ______ of ______



Note:
All measurements given are approximations from visual observation of the existing condition.

SC 3020 R2 199

Appendix 9.2.B

MCNP5 input for the Model 1

```
Rack Model 1
100   1   -5.636 -2 11 -12     u=1   imp:n=1 $Fuel Meat
101   2   -1.70  -2 12 -14     u=1   imp:n=1 $Up Graphite
102   2   -1.70  -2 13 -11     u=1   imp:n=1 $Down Graphite
103   3   -7.92  (2:-13:14)    u=1   imp:n=1 $Cladding
104   0   -1 15 -16  fill=1  imp:n=1
105   like 104 but trcl (-0.101409  4.86692  0)     imp:n=1
106   like 104 but trcl (-0.40546  9.7254  0)     imp:n=1
107   like 104 but trcl (-0.911625  14.567  0)     imp:n=1
108   like 104 but trcl (-1.61902  19.3833  0)     imp:n=1
109   like 104 but trcl (-2.52643  24.166  0)     imp:n=1
110   like 104 but trcl (-3.63227  28.9067  0)     imp:n=1
111   like 104 but trcl (-4.93463  33.5972  0)     imp:n=1
112   like 104 but trcl (-6.43123  38.2294  0)     imp:n=1
113   like 104 but trcl (-8.11949  42.7953  0)     imp:n=1
114   like 104 but trcl (-9.99647  47.2868  0)     imp:n=1
115   like 104 but trcl (-12.0589  51.6963  0)     imp:n=1
116   like 104 but trcl (-14.3033  56.0161  0)     imp:n=1
117   like 104 but trcl (-23.5318  70.3218  0)     imp:n=1
118   like 104 but trcl (-26.542  74.1475  0)     imp:n=1
119   like 104 but trcl (-29.7089  77.8445  0)     imp:n=1
120   like 104 but trcl (-33.0271  81.4063  0)     imp:n=1
121   like 104 but trcl (-36.4908  84.8269  0)     imp:n=1
122   like 104 but trcl (-40.094  88.1001  0)     imp:n=1
123   like 104 but trcl (-43.8304  91.2205  0)     imp:n=1
124   like 104 but trcl (-47.6935  94.1825  0)     imp:n=1
125   like 104 but trcl (-51.6766  96.981  0)     imp:n=1
126   like 104 but trcl (-55.7729  99.6112  0)     imp:n=1
127   like 104 but trcl (-59.9752  102.068  0)     imp:n=1
128   like 104 but trcl (-64.2762  104.349  0)     imp:n=1
129   like 104 but trcl (-68.6684  106.448  0)     imp:n=1
130   like 104 but trcl (-84.6483  112.318  0)     imp:n=1
131   like 104 but trcl (-89.3548  113.561  0)     imp:n=1
132   like 104 but trcl (-94.109  114.608  0)     imp:n=1
133   like 104 but trcl (-98.9027  115.455  0)     imp:n=1
134   like 104 but trcl (-103.728  116.102  0)     imp:n=1
135   like 104 but trcl (-108.575  116.547  0)     imp:n=1
136   like 104 but trcl (-113.437  116.79  0)     imp:n=1
137   like 104 but trcl (-118.305  116.831  0)     imp:n=1
138   like 104 but trcl (-123.17  116.668  0)     imp:n=1
139   like 104 but trcl (-128.024  116.303  0)     imp:n=1
140   like 104 but trcl (-132.859  115.737  0)     imp:n=1
141   like 104 but trcl (-137.666  114.969  0)     imp:n=1
142   like 104 but trcl (-142.437  114.002  0)     imp:n=1
143   like 104 but trcl (-158.732  109.072  0)     imp:n=1
144   like 104 but trcl (-163.239  107.232  0)     imp:n=1
145   like 104 but trcl (-167.665  105.206  0)     imp:n=1
146   like 104 but trcl (-172.003  102.998  0)     imp:n=1
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149   like 104 but trcl (-184.414  95.3168  0)     imp:n=1
150   like 104 but trcl (-188.326  92.4193  0)     imp:n=1
151   like 104 but trcl (-192.114  89.3614  0)     imp:n=1
152   like 104 but trcl (-195.771  86.1483  0)     imp:n=1
153   like 104 but trcl (-199.291  82.7857  0)     imp:n=1
```

```
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155   like 104 but trcl (-205.895  75.6355  0)     imp:n=1
156   like 104 but trcl (-215.941  61.8914  0)     imp:n=1
157   like 104 but trcl (-218.433  57.7096  0)     imp:n=1
158   like 104 but trcl (-220.749  53.4277  0)     imp:n=1
159   like 104 but trcl (-222.884  49.0531  0)     imp:n=1
160   like 104 but trcl (-224.835  44.5933  0)     imp:n=1
161   like 104 but trcl (-226.599  40.0561  0)     imp:n=1
162   like 104 but trcl (-228.173  35.4493  0)     imp:n=1
163   like 104 but trcl (-229.553  30.781  0)     imp:n=1
164   like 104 but trcl (-230.737  26.0593  0)     imp:n=1
165   like 104 but trcl (-231.724  21.2924  0)     imp:n=1
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167   like 104 but trcl (-233.097  11.6559  0)     imp:n=1
168   like 104 but trcl (-233.482  6.80317  0)     imp:n=1
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170   like 104 but trcl (-232.706  -15.0585  0)     imp:n=1
171   like 104 but trcl (-231.978  -19.8717  0)     imp:n=1
172   like 104 but trcl (-231.05  -24.6505  0)     imp:n=1
173   like 104 but trcl (-229.924  -29.3865  0)     imp:n=1
174   like 104 but trcl (-228.602  -34.0715  0)     imp:n=1
175   like 104 but trcl (-227.086  -38.6973  0)     imp:n=1
176   like 104 but trcl (-225.378  -43.2559  0)     imp:n=1
177   like 104 but trcl (-223.482  -47.7395  0)     imp:n=1
178   like 104 but trcl (-221.401  -52.1402  0)     imp:n=1
179   like 104 but trcl (-219.138  -56.4504  0)     imp:n=1
180   like 104 but trcl (-216.698  -60.6626  0)     imp:n=1
181   like 104 but trcl (-214.085  -64.7695  0)     imp:n=1
182   like 104 but trcl (-203.64  -78.2133  0)     imp:n=1
183   like 104 but trcl (-200.307  -81.761  0)     imp:n=1
184   like 104 but trcl (-196.829  -85.1668  0)     imp:n=1
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192   like 104 but trcl (-164.56  -106.651  0)     imp:n=1
193   like 104 but trcl (-160.076  -108.546  0)     imp:n=1
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900   4   -1.0   -131 98 -99   #104 #105 #106 #107 #108 #109 #110 #111
      #112 #113 #114 #115 #116 #117 #118 #119 #120 #121 #122 #123 #124
      #125 #126 #127 #128 #129 #130 #131 #132 #133 #134 #135 #136 #137
      #138 #139 #140 #141 #142 #143 #144 #145 #146 #147 #148 #149 #150
      #151 #152 #153 #154 #155 #156 #157 #158 #159 #160 #161 #162 #163
      #164 #165 #166 #167 #168 #169 #170 #171 #172 #173 #174 #175 #176
      #177 #178 #179 #180 #181 #182 #183 #184 #185 #186 #187 #188 #189
      #190 #191 #192 #193 #194   imp:n=1
901   12  -1.60  -132 131 98 -99    imp:n=1
902   0   132:-98:99  imp:n=0

C Surface Cards
1   c/z   116.84  0  1.87
2   c/z   116.84  0  1.82
11    pz   -19.05   $SS Fuel Meat Bottom (7.5 inch * 2)
12    pz    19.05   $SS Fuel Meat Top
13    pz   -29.21   $SS Fuel Graphite Bottom (4 inch)
14    pz    29.21   $SS Fuel Graphite Top
15    pz   -30.39   $SS Cladding Bottom (1.18 cm)
16    pz    30.39   $SS Cladding Top (1.18 cm)
91    pz   -29.21   $Cladding Bottom (11.5 inch * 2)
92    pz    29.21   $Cladding Top
```

```
98    pz   -40.0
99    pz   120.0
C Reflector Surfaces
131   cz   121.92      $ Water reflector
132   cz   182.88      $ Sand wall

mode n
kcode 25000 0.4 100 500
ksrc  116.84  0  0
      116.739  4.86692  0
      116.435  9.7254  0
      115.928  14.567  0
      115.221  19.3833  0
      114.314  24.166  0
      113.208  28.9067  0
      111.905  33.5972  0
      110.409  38.2294  0
      108.721  42.7953  0
      106.844  47.2868  0
      104.781  51.6963  0
      102.537  56.0161  0
      93.3082  70.3218  0
      90.298  74.1475  0
      87.1311  77.8445  0
      83.8129  81.4063  0
      80.3492  84.8269  0
      76.746  88.1001  0
      73.0096  91.2205  0
      69.1465  94.1825  0
      65.1634  96.981  0
      61.0671  99.6112  0
      56.8648  102.068  0
      52.5638  104.349  0
      48.1716  106.448  0
      32.1917  112.318  0
      27.4852  113.561  0
      22.731  114.608  0
      17.9373  115.455  0
      13.1125  116.102  0
      8.26493  116.547  0
      3.40303  116.79  0
      -1.46479  116.831  0
      -6.33006  116.668  0
      -11.1843  116.303  0
      -16.0192  115.737  0
      -20.8263  114.969  0
      -25.5972  114.002  0
      -41.8918  109.072  0
      -46.3988  107.232  0
      -50.8253  105.206  0
      -55.1635  102.998  0
      -59.4059  100.611  0
      -63.5453  98.0489  0
      -67.5743  95.3168  0
      -71.4861  92.4193  0
      -75.2737  89.3614  0
      -78.9307  86.1483  0
      -82.4507  82.7857  0
      -85.8275  79.2794  0
      -89.0554  75.6355  0
      -99.1012  61.8914  0
      -101.593  57.7096  0
      -103.909  53.4277  0
```

```
        -106.044  49.0531  0
        -107.995  44.5933  0
        -109.759  40.0561  0
        -111.333  35.4493  0
        -112.713  30.781  0
        -113.897  26.0593  0
        -114.884  21.2924  0
        -115.671  16.4885  0
        -116.257  11.6559  0
        -116.642  6.80317  0
        -116.392  -10.2191  0
        -115.866  -15.0585  0
        -115.138  -19.8717  0
        -114.21  -24.6505  0
        -113.084  -29.3865  0
        -111.762  -34.0715  0
        -110.246  -38.6973  0
        -108.538  -43.2559  0
        -106.642  -47.7395  0
        -104.561  -52.1402  0
        -102.298  -56.4504  0
        -99.8581  -60.6626  0
        -97.2445  -64.7695  0
        -86.8002  -78.2133  0
        -83.4669  -81.761  0
        -79.9887  -85.1668  0
        -76.3717  -88.4248  0
        -72.6221  -91.5293  0
        -68.7465  -94.4749  0
        -64.7515  -97.2565  0
        -60.6441  -99.8693  0
        -56.4315  -102.309  0
        -52.1209  -104.571  0
        -47.7198  -106.651  0
        -43.2359  -108.546  0
        -38.6769  -110.253  0
m1       1001.66c    -0.015896    $ new SS meat, H/Zr =1.6, 0% burnup
        40000.66c    -0.899104
        92235.66c    -0.01683
        92238.66c    -0.06817
mt1      h/zr.60t
         zr/h.60t
m2       6000.66c     1.0         $ graphite
mt2      grph.60t
m3       6000.66c    -0.0004      $ ss cladding
        14000.60c    -0.0046
        24000.50c    -0.190
        25055.66c    -0.009
        26000.50c    -0.699
        28000.50c    -0.097
m4       1001.66c     2.0         $ H2O
         8016.66c     1.0
mt4      lwtr.60t
m11      7014.66c     0.0000381259 $ Air  0.00115g/cm3
         8016.66c     0.0000095012
        18000.59c     0.0000001664
m12      8016.66c     2.0           $ Sand: SiO2 1.60g/cm3
        14000.60c     1.0
```

Appendix 9.2.C

MCNP5 input of Model 2

```
Rack Model 3
100   1   -5.636 -2 11 -12    u=1   imp:n=1 $Fuel Meat
101   2   -1.70  -2 12 -14    u=1   imp:n=1 $Up Graphite
102   2   -1.70  -2 13 -11    u=1   imp:n=1 $Down Graphite
103   3   -7.92  (2:-13:14)   u=1   imp:n=1 $Cladding
104   0   -1 15 -16  fill=1  imp:n=1
105   like 104 but trcl (-1.9  -3.2909  0)     imp:n=1
106   like 104 but trcl (1.9  -3.2909  0)     imp:n=1
107   like 104 but trcl (-3.8  -6.58179  0)     imp:n=1
108   like 104 but trcl (0.  -6.58179  0)     imp:n=1
109   like 104 but trcl (3.8  -6.58179  0)     imp:n=1
110   like 104 but trcl (-5.7  -9.87269  0)     imp:n=1
111   like 104 but trcl (-1.9  -9.87269  0)     imp:n=1
112   like 104 but trcl (1.9  -9.87269  0)     imp:n=1
113   like 104 but trcl (5.7  -9.87269  0)     imp:n=1
114   like 104 but trcl (-7.6  -13.1636  0)     imp:n=1
115   like 104 but trcl (-3.8  -13.1636  0)     imp:n=1
116   like 104 but trcl (0.  -13.1636  0)     imp:n=1
117   like 104 but trcl (3.8  -13.1636  0)     imp:n=1
118   like 104 but trcl (7.6  -13.1636  0)     imp:n=1
119   like 104 but trcl (-9.5  -16.4545  0)     imp:n=1
120   like 104 but trcl (-5.7  -16.4545  0)     imp:n=1
121   like 104 but trcl (-1.9  -16.4545  0)     imp:n=1
122   like 104 but trcl (1.9  -16.4545  0)     imp:n=1
123   like 104 but trcl (5.7  -16.4545  0)     imp:n=1
124   like 104 but trcl (9.5  -16.4545  0)     imp:n=1
125   like 104 but trcl (-11.4  -19.7454  0)     imp:n=1
126   like 104 but trcl (-7.6  -19.7454  0)     imp:n=1
127   like 104 but trcl (-3.8  -19.7454  0)     imp:n=1
128   like 104 but trcl (0.  -19.7454  0)     imp:n=1
129   like 104 but trcl (3.8  -19.7454  0)     imp:n=1
130   like 104 but trcl (7.6  -19.7454  0)     imp:n=1
131   like 104 but trcl (11.4  -19.7454  0)     imp:n=1
900   4   -1.0   -131 98 -99 #104 #105 #106 #107 #108 #109 #110 #111
      #112 #113 #114 #115 #116 #117 #118 #119 #120 #121 #122 #123 #124
      #125 #126 #127 #128 #129 #130 #131   imp:n=1
902   0   131:-98:99  imp:n=0

C Surface Cards
1   c/z   0  9.71681  1.87
2   c/z   0  9.71681  1.82
11    pz   -19.05   $SS Fuel Meat Bottom (7.5 inch * 2)
12    pz    19.05   $SS Fuel Meat Top
13    pz   -29.21   $SS Fuel Graphite Bottom (4 inch)
14    pz    29.21   $SS Fuel Graphite Top
15    pz   -30.39   $SS Cladding Bottom (1.18 cm)
16    pz    30.39   $SS Cladding Top (1.18 cm)
98    pz   -60.0
99    pz    60.0
C Reflector Surfaces
131   cz    60.0      $ Water reflector

mode n
kcode 50000 0.8 100 500
ksrc  0  9.87269  0
      -1.9  6.58179  0
```

```
        1.9  6.58179  0
        -3.8  3.2909  0
        0.  3.2909  0
        3.8  3.2909  0
        -5.7  0.  0
        -1.9  0.  0
        1.9  0.  0
        5.7  0.  0
        -7.6  -3.2909  0
        -3.8  -3.2909  0
        0.  -3.2909  0
        3.8  -3.2909  0
        7.6  -3.2909  0
        -9.5  -6.58179  0
        -5.7  -6.58179  0
        -1.9  -6.58179  0
        1.9  -6.58179  0
        5.7  -6.58179  0
        9.5  -6.58179  0
        -11.4  -9.87269  0
        -7.6  -9.87269  0
        -3.8  -9.87269  0
        0.  -9.87269  0
        3.8  -9.87269  0
        7.6  -9.87269  0
        11.4  -9.87269  0
m1       1001.66c    -0.015896   $ new SS meat, H/Zr =1.6, 0% burnup
        40000.66c    -0.899104
        92235.66c    -0.01683
        92238.66c    -0.06817
mt1      h/zr.60t
         zr/h.60t
m2       6000.66c     1.0        $ graphite
mt2      grph.60t
m3       6000.66c    -0.0004     $ ss cladding
        14000.60c    -0.0046
        24000.50c    -0.190
        25055.66c    -0.009
        26000.50c    -0.699
        28000.50c    -0.097
m4       1001.66c     2.0        $ H2O
         8016.66c     1.0
mt4      lwtr.60t
m11      7014.66c     0.0000381259 $ Air  0.00115g/cm3
         8016.66c     0.0000095012
        18000.59c     0.0000001664
m12      8016.66c     2.0          $ Sand: SiO2 1.60g/cm3
        14000.60c     1.0
```

9.3 Fire Protection Systems and Programs

The purpose of the fire protection system is to provide detection and notification capability which will mitigate loss of property and life in the event of a fire. The UNEF (Utah Nuclear Engineering Facilities) does not contain any potential source of fire in the reactor room or the control room. However, there are several fire detection and suppression systems installed in accordance with National Fire Protection Code in the UUTR facility. There is a fire extinguisher and a fire hose in the reactor room and radio-chemistry laboratory. The fire extinguishers and detection system are regularly inspected by University operations.

In the case of detection of smoke/fire visual and audible alarms will go off and a signal will be sent to the Salt Lake City Fire Department [redacted] These detection and protection systems are maintained by the University Plant operations.

The fire protection plan is followed by the reactor personnel prior to the arrival of the Fire Department Personnel arrive (obtained from the UNEP Emergency/Security Plans and Procedures)

In case of fire in the facility rooms:

- Immediately scram the reactor and/or suspend other laboratory operations. Secure the reactor and remove the key.
- Use the lab fire extinguisher as soon as possible. Evacuate all non-essential personnel immediately.
- Start the emergency call list in action by calling the campus dispatcher (5-2677).
- If the fire is deemed out of immediate local control, sound the building fire alarm box outside the door or call 9-911.

In case of fire in the Reactor Room Area:

- Start the emergency call list in action by calling the campus dispatcher (5-2677). Ask the dispatcher to call Salt Lake City Fire Department and request the use of foam.
- Drop the sources, if possible.
- Use the existing fire extinguisher as soon as possible. Evacuate all non-essential personnel immediately.

- Provide radiation surveys for the firemen when they arrive. Instruct the firemen to flood the facility with foam, dry chemicals, or water until the fire is extinguished.
- If the fire is caused by an electrical power malfunction, terminate all electrical power to the irradiator room as soon as possible.
- Complete Safeguard Event Log (NUREG-031)

9.4 Communications

Communications within the laboratory are provided by the University telephone system. There are four separate lines that enable the control room to telephone the office of the Reactor Supervisor and the offices of the Senior Reactor Operators. The telephone sets are also equipped with an intercom function.

9.5 Possession and Use of Byproduct, Source and Special Nuclear Material (SNM)

The UUTR staff has materials used to maintain and operate the TRIGA reactor in compliance with the facilities operating license No. R-126 pursuant to 10 CFR parts 30, 50 and 70. This facility is licensed to receive, possess, and use up to 4.9 kilograms of U-235 and a [redacted] Curie sealed Pu-Be source. Additionally, the facility is licensed to receive and possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the reactor.

Samples and sources are stored in shielding materials that reduce the activity levels at the surface of the container. Storage containers range from small lead "pigs" to large brick vaults. These containers may be located in the reactor room, the radiochemistry labs, or radiation measurement lab. Radioactive samples and sources are clearly labeled and secured. Access to rooms containing radioactive sources is restricted to trained personnel and those accompanied by trained personnel. Access to the reactor room and control room is restricted to licensed operators and those accompanied by licensed operators.

Samples, experimental devices, reactor components, or by products of normal reactor operations can only be released to holders of a radioactive materials license. Irradiated samples must undergo a materials release survey. The container type, dose rates, license identification, and other appropriate information must be noted on form UNEP-027.

The UUTR facilities include four compensating fume hoods that operate with a minimum air velocity of 100 cfm per fume hood. Each fume hood can be retrofitted with the appropriate shielding as required. Additionally, each lab is equipped with separate waste collectors for radioactive and hazardous waste. Administrative controls prohibit processing or mixing of waste that creates "mixed waste". Hazardous chemicals are characterized and disposed of through the University of Utah's Environmental Health and Safety Department. Radioactive waste is characterized and disposed of through the University of Utah's Radiological Health Department.

9.6 Cover Gas Control in Closed Primary Coolant Systems

The UUTR is an open pool type reactor with the primary coolant loop open to the atmosphere. As a result, no cover gas control system is necessary for this reactor.

9.7 Other Auxiliary Systems

The UUTR facility two auxiliary systems that do not fall into standard format categories: The facility lighting and the access control.

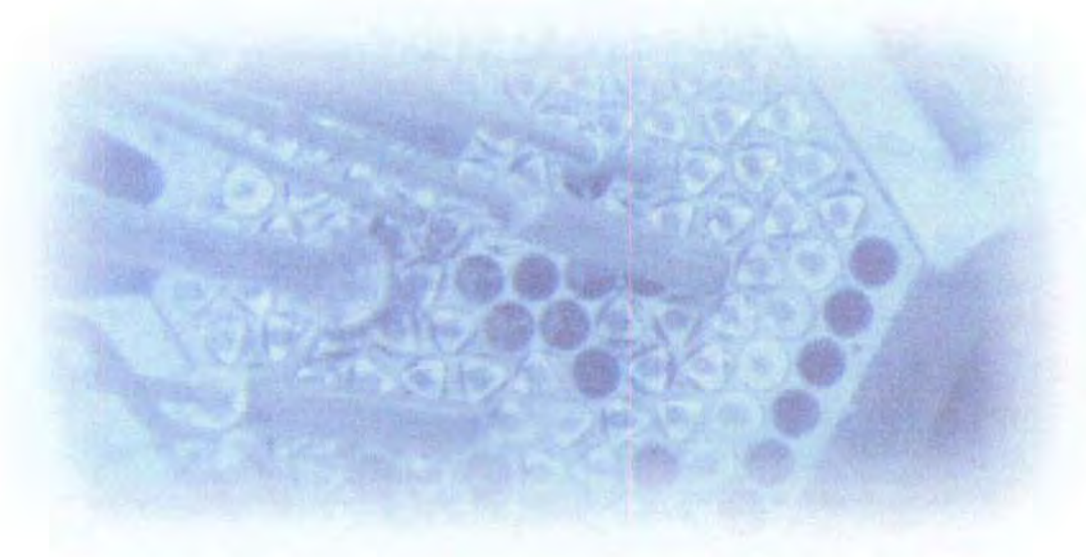
9.7.1 Lighting Systems

Fluorescent lighting is used in all of the rooms associated with reactor operations. University maintenance personnel service this lighting. High intensity lights are located at the top of the reactor to illuminate the core for visual inspections and various other operations including movement of experiments, movement of the source, and maintenance of the core and its associated fixtures. There are three, battery-powered, emergency lighting units located in the reactor facility to provide lighting in the event of power failure. University maintenance personnel also service the emergency lighting. (The maintenance personnel must be accompanied for such activities at UNEF).

9.7.2 Access Control

The reactor room is equipped with security system in accordance with the requirements of the UNEF security plan. Access to the UNEF labs and facilities is controlled by the Reactor Supervisor and UNEP Director. Access to the reactor room and control room is restricted to licensed operators and those accompanied by licensed operators. Persons entering the reactor room are required to wear personal dosimeters or be accompanied by an individual wearing dosimeter. Emergency exit may be made through the radiochemistry laboratory or through the control room.

10. EXPERIMENTAL FACILITIES AND UTILIZATION



10.1 Summary Description

Utilizations and experimental features. The UUTR provides research, educational and training services to support the scientific curriculum at The University of Utah Nuclear Engineering Program (UNEP), and the education of the community about nuclear engineering and science, and radiation and health physics. Principal experimental features of the UUTR facility include:

- central irradiator
- three beam ports
- pneumatic tube irradiator
- fast neutron irradiator and
- thermal irradiator

To date, no beam tube irradiations have been performed at UUTR. Monitoring of experimental activities is preformed from the reactor console location using reactor control instrumentation, visual observation, voice, intercom, and remote camera.

Design requirements for the experiment and the review and approval process. All experiments that are performed at the UUTR fall into one of the following three categories: routine, modified routine, and new. All experiments began as new experiments. New experiments must be reviewed and approved by the Reactor Safety Committee prior to implementation. Modified routine experiments are similar in nature to experiments that are routinely performed. They require the review of a Senior Reactor Operator (SRO) prior to implementation. Routine experiments are those experiments that have been approved and documented and are performed on a regular basis. A list of currently approved experiments is maintained at the facility, and includes evolutions such as normal reactor operation and routine use of experimental facilities. Any evolution not included on the current list of approved experiments must be reviewed and approved in accordance with **UUTR SAR 10.3** prior to performance. These practices do not depend upon the UUTR core power.

Limiting experimental characteristics. Regular experiments include the neutron activation analysis and irradiation of various material samples. In a typical year, NAA is performed on the order of 100 samples and irradiation of materials in the order of 300 samples. The NAA samples are counted on Ge(Li) detectors so the activity level is very low in terms of radiation exposure to individuals. On average, 95% of all experiments involve the samples irradiation in the fast neutron irradiator and thermal irradiator. The experiment limit for reactivity insertion is $0.30/sec. Any explosive or hazardous material will be prohibited to irradiate unless the mass is less than 25 milligrams.

Additionally, at the UUTR, UNEP regularly provides a year-long training for the NRC issued SRO licenses. On average to date, UNEP trains four to seven students a year. For that purpose, the UUTR is run every week for 3 to 5 hours, for 12 months each year.

10.2 Experimental Facilities

The UUTR facility has been in operation from 1975, which is for over 35 years at the time of this documentation, during which time no significant problem or excessive personnel exposure has occurred associated with the operation and use of the reactor and the associated experimental programs.

The Radiation Monitoring Systems and the Radiation Safety Program as well as the procedures described in this SAR have proven to be quite adequate to insure the safe usage of the UUTR facility. According to the University of Utah RHD Dosimetry report, the highest exposure that has been recorded by any user or reactor operational personnel over the past 10 years has been 7 mR/year. The radiation monitoring program includes using film badges to measure the maximum possible exposure for gamma and neutron radiation in a variety of locations around the facility. Several points in the UUTR facility were surveyed for last 30 years. The hottest spot is found to be on the control rods bridge on the top of the reactor pool with gamma exposure of 4 μR/hour and minimal exposure to neutron radiation of ~ 1mR/month without reactor operation. With reactor operation at 90 kW, the gamma exposure increased up to ~20 μR/hour. This measurement was performed using Ludlum Model 19 gamma detector during every reactor operation. Recent measurement was performed on February 19, 2010. The UUTR's annual operation hour changes every year, but for last 10 years, the longest continuous operation was 52 hours. This operation included power of the UUTR of 1 kW, 10 kW through 90 kW. The gamma dose near the control rods bridge area is approximately 0.02 mR/hr at 90 kW; it is generally 0.004 to 0.006 mR/hr after the reactor shutdown. Typical operation time for the UUTR is 52 hours per year. Under the assumption that the reactor power is 90 kW during these 52 hours, the maximum annual possible gamma exposure to any individual is ~54 mR under the assumption that this individual spends 24 hours a day 365 days a year on the bridge.

10.2.1 In-Core Irradiators

The UUTR has been designed with multiple in-core irradiation facilities to facilitate a broad range of potential experimental activities. These facilities consist of a central cavity, the pneumatic transfer tube, and individual fuel element locations.

10.2.1.1 Central Irradiation Facility (CIF)

The central irradiation facility (CIF) is located in the central fuel pin position, **Fig. 10.2-1**. A special tube has been constructed to accommodate samples and can be placed in the central fuel pin position (the A fuel ring) by means of a cable. The dimensions of this assembly are the same as a fuel pin. Because this facility is an in-core irradiator located in the center of the core, there are special restrictions on the reactivity of samples placed in this irradiator. Additionally, a cutout is available where both the A and B rings are removed to accommodate a larger irradiator, capable of holding multiple samples. This irradiator has two special features associated with it. One of these is a sealed interior that holds heavy water. The other is a motor that rotates the sample holder to spatially average the neutron fluence in the assembly. This cutout facility is currently not in the UUTR core and will require an appropriate analysis, experimental review and approval by the reactor safety committee prior to implementation.

10.2.1.2 Pneumatic Transfer System (PTS)

A pneumatic transfer system (PTS) is available for use at the UUTR facility. The PTS is installed within a 1.5 inch O. D. tube and is driven by the force of dry, compressed helium. As shown in **Fig. 10.2-2** the PTS has a slight curve in its tube in order to prevent direct streaming of neutrons from the core to the surface of the pool. The UUTR PTS is designed to quickly transfer individual specimens into and out of the reactor core. The specimens are placed in a small polyethylene holder, "rabbit," which in turn is placed into the receiver. The "rabbit" is an enclosed polyethylene holder (**Fig. 10.2-3**). It travels through aluminum and PVC tubing to the terminus at reactor core centerline, and returns along the same path to the receiver. Directional gas flow moves the "rabbit" between receiver and terminus. A compressed gas system supplies helium to the system, and a set solenoid valve directs flow. Controls to operate the compressed gas and solenoid valve are on the console. The key system elements and their functions are described below. Experiments are inserted into the rabbit and contained by a screw cap on one end. Available space inside the rabbit is approximately 0.625 inches in diameter and 4.5 inches in length.

The receiver positions the "rabbit" for transfer to the terminus and receives the rabbit after irradiation. Two transfer lines connect the receiver to the terminus: one allows the rabbit to travel between the receiver and terminus, the other controls gas flow direction. The receiver is located in the counting laboratory. The exhaust is released into the ventilation stack and prevents uncontrolled release of airborne radioactivity. The exhaust fans maintain the negative pressure with respect to the surrounding room. The PTS exhaust passes through a pre-filter, a HEPA filter, before continuing up the stack. The stack monitor and the CAM sample the exhaust air in the ventilation system including the exhaust released from the PTS.

[redacted]

Figure 10.2-1 UUTR Central Irradiation Facility (CI Facility)

The terminus consists of two concentric tubes, which extend into the reactor core. The inner tube is perforated with holes (which are smaller than the sample container diameter). The bottom of the inner tube contains a stainless steel spring shock absorber to lessen the impact of the rabbit when it reaches this end of the transfer line, which is approximately at the mid-plane of the core. When air flows to the terminus, the capsule rests in the bottom of the inner tube: when air flows to the receiver, the capsule moves out of the inner tube by air flowing through the tube's holes. The outer tube supports the inner tube and provides a path for the air to flow through. The outer tube bottom support is shaped like the bottom of a fuel element and can fit into any fuel location in the core lattice. Both tubes that extend to the top of the reactor tank are offset to reduce radiation streaming. A weight has been installed to counteract the buoyancy of the air-filled tubes and keep the terminus firmly positioned in the core. A set of solenoid valves direct flow through the transfer-line-loop sending the rabbit either to the terminus or to the receiver depending on valve position.

[redacted]

Figure 10.2-2 UUTR Pneumatic Transfer System (PTS)

10.2.1.3 Vacant Fuel Positions

A total of 20 reactor grid positions are vacant for fuel elements and may be utilized for the irradiation of materials (**Fig. 10.2-4**). The very center vacant position (CI) is used as a central irradiator from the time reactor was built. These in-core irradiation facilities or the positioning of a single experiment in a fuel element vacancy grid position shall meet the requirements of the Technical Specifications for design, safety evaluation, restrictions and approvals.

Figure 10.2-3 "Rabbit" holder

[redacted]

Figure 10.2-4 Vacant fuel positions in the UUTR core

10.2.2 In-Reflector Irradiators

The UUTR has been designed with multiple in-reflector irradiation facilities to facilitate a broad range of potential experimental activities. These facilities consist of a dry tube thermal irradiator and a fast irradiator.

10.2.2.1 Dry Tube Thermal Irradiator

A dry tube thermal irradiator is available for use in the trapezoidal shaped D_2O tank attached to the side of the core, **Fig. 10.2-5**. The samples are placed into polyethylene vials attached to a line and dropped into the irradiator through a curved PVC tube that extends to the top of the reactor pool.

10.2.2.2 Fast Neutron Irradiation Facility (FNIF)

The fast neutron irradiator is designed to provide sample exposure to neutrons with minimal moderation, **Fig. 10.2-6**. The entire device has two pieces: a stand and a sample holder. All structures were fabricated from Al-5052, which is a material compatible with the TRIGA reactor system. The irradiator consists of two pieces, the outer box and the inner box. The outer box is loaded with lead bricks and sealed by bolting the inner box to the outer box with aluminum bolts. Graphite gasket material ("Grafoil") was used on the contact surfaces. Grafoil is radiation resistant and has been proven in high radiation applications at other reactor facilities. The irradiator is loaded with standard lead shielding bricks on one side. This side is placed next to the core face for additional gamma shielding. Also, thermal neutron absorbers may be placed in any position to decrease sample exposure to thermal neutrons. The irradiator is about 500 pounds sub-buoyant.

The sample holder contains one lead brick above the test volume and one lead brick below the test volume. The amount of shielding may be adjusted as necessary. The sample holder was sealed with neoprene gasket material. It is about 35 pounds sub-buoyant when loaded with the two lead bricks inside. The device stand rests on the floor of the reactor tank, and the irradiator is located on the device stand. The assembled device aligns with the reactor core the western most hexagonal face. All aluminum surfaces were anodized to prevent corrosion. The stand and the irradiator will remain in the pool during their useful lifetime and may be moved to change the irradiation position or may be temporarily removed from the pool for maintenance of the radiation conditioning materials. Only the sample holder is moved on a regular basis to access samples.

[redacted]

Figure 10.2-5 UUTR dry tube thermal irradiator

[redacted]

Figure 10.2-6 UUTR Fast Neutron Irradiation Facility (FNIF)

The irradiator is located on the outside of the grid structure. The irradiator affects the flux by changing the moderation and reflectivity in the region where the irradiator replaces light water with air, lead, and aluminum. As installed the irradiator reduces core reactivity by no more than $0.10. The device is considered to be a secured experiment. The sample holder is also considered to be secured and is estimated to change the reactivity by no more than $0.05 when placed in position. These reactivities cannot result in a prompt critical condition if the devices are accidentally separated from the core or flooded by water. The irradiator, located outside the grid structure, does not interfere with fuel cooling channels; therefore, it does not affect fuel and cladding temperatures or fuel internal pressure. Use of the fast neutron irradiator does not increase the probability of personnel exposure because it introduces no additional mechanisms for exposure, nor does it increase the probability of radioactive material release because it introduces no additional mechanisms for release. Use of the fast neutron irradiator does not increase the probability of pool water leakage because it does not introduce any new mechanisms for failure of the tank. Because of the low reactivity worth of the sample holder, the holder may be conservatively removed while the reactor is either critical or shutdown. However, the sample holder shall be inserted and removed from the irradiator only while the reactor is shutdown in order to carefully control neutron exposure.

Samples to be irradiated in the sample holder will be in compliance with **UUTR TS 3.8.1** and will have less than $1.0 of reactivity. The hazards associated with the fast neutron irradiator device have been reviewed by the UUTR staff and it was concluded that the installation of the fast neutron irradiator did not constitute a change in the Technical Specifications and has no unreviewed safety issues.

10.2.3 External Irradiators

The UUTR facility has only one experimental facility for irradiation that is external to the tank and biological shielding supplied by the tank water, the three beam ports. The UNEP has no immediate plans to open any of these three beam ports. Future use still require analysis, documentation, review and approval of utilization of the beam ports by the Reactor Safety Committee.

The reactor system contains three diagonally directed beam ports that extend from the reactor core to the reactor floor. The upper beam port is adequately shielded with sand and capped at the reactor floor level with a 1/8 inch thick steel cap for security. When the beam port is employed, the sand will be removed and a sealed aluminum beam tube (carefully weighted so as to have a net density greater than water) will be installed between the inner tank wall and the reactor core shroud along the common axis of the upper beam port. There are three aluminum beam tubes (one for each port) currently in storage. Each beam tube is composed of two sections aligned along a common axis. The top tube section is a 1 ft diameter tube that will be inserted in the port between the reactor floor and the wall of the aluminum reactor tank. If installed, this tube would not penetrate the aluminum tank because it is sealed at the end where it makes contact with the tank. During beam port use, both the lower and upper beam tube sections can be sealed against air flow to ensure that no radiation hazards arising from Argon-41 or nitrogen-16 buildup will be present. Furthermore, penetration of the reactor tank is not necessary to install and utilize the beam tubes. Only inner tubes have contact with the reactor tank, and thus there is no increased risk of inner tank water leakage. **Figure 10.2-7** shows one of the three UUTR's beam ports. When the beam port is employed, appropriate shielding at the reactor floor level will be constructed with dense concrete blocks and other available shielding materials.

10.3 Experiment Review

Prior to conducting any new irradiation experiment, a technical and safety review of the proposed experiment must be performed. The following procedures apply to the review and approval of all experiments utilizing any of the UUTR irradiation facilities. These procedures outline the prerequisites to experimental approval, the review and approval of new, modified routine and routine experiments. Additionally, the procedures that outline documentation, and personnel training requirements are presented as follows.

10.3.1 Prerequisites to Experiment Approval

No new experiment shall be implemented until the following criteria are met:

1. A hazards analysis of the proposed experiment has been performed and the results have been reviewed for compliance with the limitations on experiments (**UUTR TS 3.8.2**) by the Reactor Safety Committee (RSC)
2. An experiment review has been completed by the operating staff and has received RS approval. Minor modifications to a reviewed and approved experiment may be made at the discretion of the SRO. If the SRO determines that the changes do not constitute a significantly new or different safety risk greater than the approved original experiment, then the modified experiment may be conducted without further approval. The SRO must document all such decisions **(UUTR TS 6.5)**.

Figure 10.2-7 One of three beam ports; UUTR has three beam ports that are never used

3. The reactivity worth of a proposed experiment has been calculated and does not exceed one specified by **UUTR TS 3.8.1.** If the reactivity worth of an experiment has been measured in a similar core position at equal neutron flux, then an estimate of the reactivity worth is acceptable.
4. The quantity of known explosives is less than 25 milligrams of TNT equivalent **(UUTR TS 3.8.2)** and the pressure produced in the experiment container upon accidental detonation of the explosive has been experimentally determined to be less than the design pressure of the container.

10.3.2 Review and Approval of New, Modified Routine, and Routine Experiments

The approval procedure applied to an experiment or class of experiments shall depend on whether a proposed experiment is evaluated as a new, modified routine, or routine experiment.

The review and approval of all experiments must be initiated by the experimenter who shall complete and submit the required form, Irradiation Request and Performance, to the RS. The RS shall verify the completeness of the requested information and pass the form to Operating Staff for evaluation. The Operating Staff shall evaluate the form and shall determine if the experiment requires initial RSC approval in the form of an approved Experiment Authorization or if the experiment can be reviewed and approved by UNEF Staff under an existing experimental authorization.

10.3.2.1 New Experiment

A new experiment is any proposed activity utilizing the UUTR that does not directly conform to an existing Experiment Authorization and therefore requires RSC approval. This approval is initiated by the submittal of a completed Experiment Authorization UUTR staff. The Experiment Authorization will then be evaluated by UNEF staff. The US NRC requires that new experiments that entail unreviewed safety questions must be approved by their organization as well. The UNEF staff shall then review the request at a staff meeting. Final approval comes from the RSC after reviewing the Experiment Authorization, TRIGA reactor EA form, and the minutes of the UNEF staff meeting. If approved, then the approved new experiment shall be passed to the RS and operating staff for scheduling. The experimenter must then submit a Form UNEP-027 and the new experiment shall receive final approval by the signature of the RO on this form.

10.3.2.2 Modified Routine Experiment

A modified routine experiment is one where planned or desired changes to the experiment could result in an increase of a safety hazard previously identified for the routine experiment. But the modified experiment remains within the parameters of an existing Experiment Authorization since the experiment presents no new and unreviewed safety hazards. These modifications can be approved by an SRO.

10.3.2.3 Routine Experiment

A routine experiment is one that has an existing approval from the RSC and has an existing Experiment Authorization and TRIGA Reactor experiment authorization form. To perform a routine experiment, the approval signature of a RO on Form UNEP-027 is required.

10.3.3 Documentation

This section describes the forms that are required for the documentation of experiment review and approval.

10.3.3.1 Experiment Authorization

An Experiment Authorization must be submitted for any new experiment. The authorization shall include a description of the experimental devices and general procedures that will be used to conduct the proposed experiment. It must also include an analysis of safety hazards. The authorization shall be submitted to UNEF staff for screening before it is submitted to the RSC for approval. This is a standard form for staff and RSC review of an experiment. The RS must sign this form. For new and modified experiments, additional authorization signatures are required from the RA and the RSO or their designees. This form is used as a training tool and a checklist to insure the experiment integrates with Technical Specifications, Experimental Authorization, and UNEF Procedures as well as other applicable documents. Completion of this form with all appropriate signatures shall give approval to UNEF operations to perform the proposed experiment.

10.3.3.2 Irradiation and Performance Request Form: UNEP form-027

The submittal UNEP form –027 is required in order to request irradiation or services of an approved experiment. The purpose of this form is to describe the isotopes to be produced, the activity expected, and information regarding the handling of radioactive materials. This form must be signed and approved by an RO.

The sample, its encapsulation, the procedures for handling it, and the method of positioning it in the reactor must satisfy the following criteria in order to be approved for irradiation. This criterion is established to provide the individuals reviewing Form UNEP-027 an adequate technical basis for making their decision.

1. Encapsulation must ensure sample containment in order to prevent contamination of the reactor pool, handling areas, and any laboratories involved.
2. The induced sample activity can be safely handled using available equipment.
3. If a dimensional change of the sample is expected, adequate expansion space must be left in the irradiation capsule.
4. The expected reactivity change due to insertion and removal must be within acceptable limits.
5. Significant reactivity variations due to sample movement (sway, bobbing, or rotation) must be prevented.

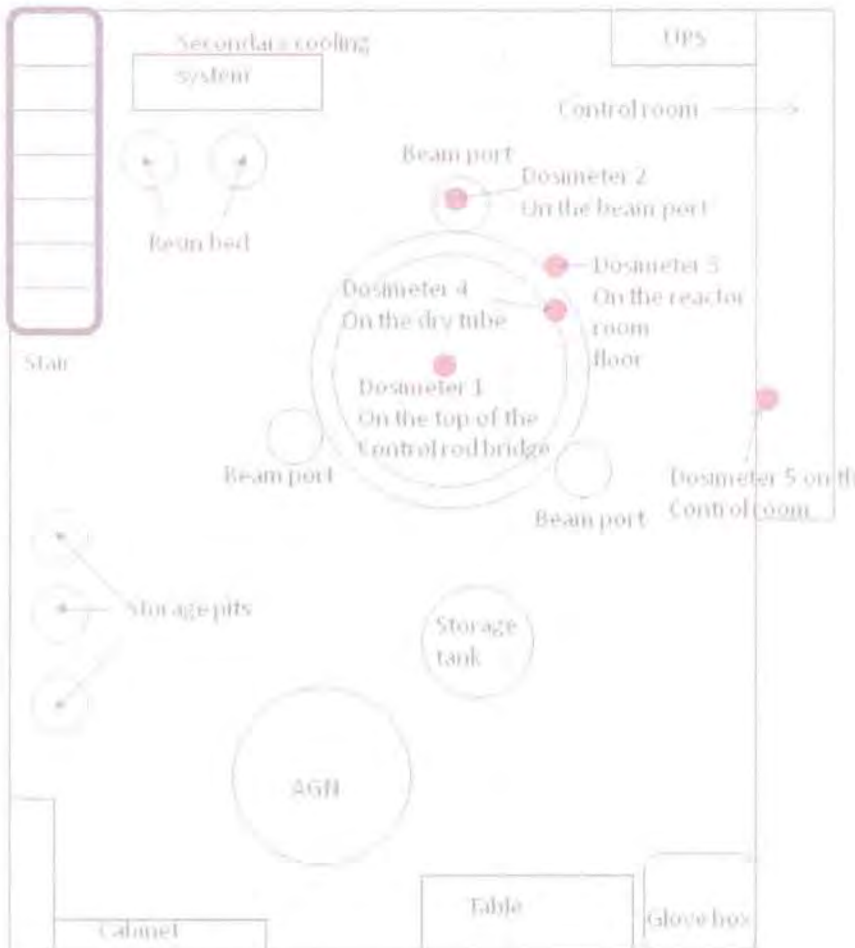
6. Expected activity calculation must be figured as either the total activity of all samples or a per batch amount from which the total activity can be calculated. The activity should be categorized according to the major contributing isotopes.
7. Review and approval documentation for the experiment must be on file in the control room.

Activity level should be calculated at the end of irradiation (i.e., decay time equals zero).

10.4 Personnel Training Requirements

The training and requalification of UNEF and UNEP students and staff for conducting an experiment is documented in the RO Requalification Program. Non-operator personnel who are involved with an experiment while it is being conducted must be briefed prior to an experiment and closely supervised at all times by a qualified licensed reactor operator.

11. RADIATION PROTECTION AND WASTE MANAGEMENT



This chapter deals with the UUTR radiation protection program and the corresponding program for management of radioactive waste. Specifically addressed are the radiation sources that will be present during normal operation and the expected radiation exposures due to normal operation. This Chapter also describes the facility radiation protection programs used to monitor and control these sources and exposures.

11.1 Radiation Protection

In order to ensure safety and productivity, the use of radiation-producing machines and radioactive materials must be conducted in strict accordance with established federal and state safety standards in order to minimize unnecessary radiation exposure to the users and to members of the general public. The objective of the UUTR Radiation Protection Program is to keep radiation exposures at the RC and to the general public "as low as reasonably achievable", ALARA.

11.1.1 Radiation Sources

The radiation sources present at the UUTR can be categorized as air-borne, liquid, or solid sources. Each of these categories is discussed in the following sections.

The airborne sources at the UUTR consist mainly of Argon-41 (Argon-41, half-life:1.83 hrs), due largely to neutron activation of air (contains about 1% argon) dissolved in the reactor's primary coolant, and Nitrogen-16 (N-16, half-life:7.13 sec), due to neutron interactions with oxygen (O-16) in the primary coolant. Liquid sources are quite limited at the UUTR and include some activation samples, and liquid scintillation samples. No routine liquid effluent or liquid waste is anticipated to be generated at 100kW. Solid sources are more diverse, as they are associated with most TRIGA reactor operations. Solid sources are more diverse, as they are associated with most TRIGA reactor operations. In addition, other solid sources are the neutron startup source, small fission chambers for use with nuclear instrumentation, items irradiated as part of normal reactor use, and small instrument check and calibration sources. Solid waste is yet another solid source, but is expected to be very limited in volume and curie content as past history at the UUTR has demonstrated. Solid waste is expected to result almost exclusively from activation.

11.1.1.1 Airborne Radiation Sources

11.1.1.1.1 Radiological Standards

Appendix B of 10CFRPart 20 lists the allowable Derived Air Concentration (DAC) for Argon-41 as $3x10^{-6}$ $\mu Ci/cm^3$. For 2,000 hours exposure this will produce the 50 mSv (5 rem) maximum permissible annual exposure. Appendix B of 10 CFR Part 20 lists the allowable Effluent Concentration (EC) for argon-41 as 10^{-8} $\mu Ci/cm^3$. For 8,760 hours exposure this will produce the 0.5 mSv (0.05 rem) annual exposure for a member of the public.

Maximum concentrations of waste gases allowed to be released into unrestricted and restricted areas as stipulated in 10CFR20 are defined as follows:

- Argon-41: 1×10^{-8} (μCi/ml) Unrestricted
- Argon-41: 3×10^{-6} (μCi/ml) Restricted
- Nitrogen-16: *
- Nitrogen-16: *

*denotes no 10CFR20 limits established; hemispherical immersion model used.

11.1.1.1.2 System Parameters

Calculations for Argon-41 and Nitrogen-16 releases during normal operations are based on the UUTR system parameters shown in **Table 11.1-1**.

11.1.1.1.3 Reactor Core Parameters

UUTR core parameters are determined based on measurements and the MCNP5 based core models. The actual core geometry was used to calculate radiation shielding. The UUTR 100kW core consists of 78 fuel elements and 12 graphite elements, and 12 heavy water elements. The UUTR core has aluminum and stainless steel cladding fuel elements. The fuel has 8.5 w% of uranium and less than 20% enrichment of U-235. Aluminum elements have a H/Zr ratio of 1.0 and stainless steel elements has the H/Zr ratio of 1.6. Steel density is 7,900 kg/m^3. Heavy water elements consist of 68% heavy water and 32% of light water. Density of graphite is 1,700 kg/m^3. UUTR 100 kW, 23 aluminum elements, 36 new stainless steel elements, 17 old stainless steel elements, and 2 stainless steel instrumental elements made the core.

Table 11.1-1 UUTR system parameters at 100kW

Parameter	Symbol	Value
Reactor steady power	P	100kW
Core coolant mass flow rate	w	130 kg/m^2-s
Core coolant density	ρ	1.0 g/cm^3
Core avg. thermal neutron flux at full power sec,	ϕ_{th}	9.4x10^{11} n/cm^2-sec
Core avg. fast neutron flux at full power sec	ϕ_f	2.6x10^{12} n/cm^2-sec
Fuel element heated length	L	0.381 m
Total flow cross sectional area	A	395.6 cm^2
Mass flow rate per fuel element .	h	65.9 g/s
Reactor tank depth		7.32 m (24 ft)
Reactor tank water depth above core		6.71 m (22 ft)
Coolant volume in reactor tank	V	3.42x10^7 cm^3
Reactor room air volume (20,000 ft^3)	V_{bav}	5.65x10^8 cm^3
Decay constant for Argon-41	λ_r	1.06x10^{-4} /s
Thermal absorption cross section for Ar-40	σ_{40}	0.53 barns
Volume flow rate from the reactor room exhaust		2.20x10^9 cm^3/hr
Reactor room air changes per hour (λ_v)	λ_v	4/hr = 1x10^{-3} /sec

11.1.1.1.4 Reactor Area Parameters

For purposes of radiation dose calculations within the reactor room, the volume of the reactor room is assumed to be 5.65x10^8 cm^3. The air exhaust rate is 6.1x10^5 cm^3/sec.

11.1.1.1.5 Radiological Assessment of Argon-41 and Nitrogen-16 Production in Experimental Facilities

During normal operation of the UUTR, there are two sources of airborne radioactivity, that are generated, Argon-41 and Nitrogen-16.

The nuclide Argon-41 is produced by thermal neutron absorption by natural Ar-40 in the atmosphere and in the air dissolved in the reactor cooling water. The activation product appears in the reactor pool and is subsequently released to the atmosphere through the ventilation system.

The nuclide Nitrogen-16 is produced by fast neutron interactions with oxygen. The only source of Nitrogen-16 in the reactor that requires consideration results from interactions of fast neutrons with oxygen in the cooling water as it passes through the reactor core. Any interaction with oxygen in the atmosphere is relatively insignificant and is neglected in this analysis. A portion of the Nitrogen-16 produced in the core is eventually released from the top of the reactor tank into the outside of the reactor pool. The half-life of Nitrogen-16 is only 7.13 seconds, so its radiological consequences outside the reactor pool are insignificant.

The cladding of a fuel element could fail during normal operations as a result of corrosion or manufacturing defect. If a failure occurs, a fraction of the fission products, essentially the noble gases and halogens, would be released to the reactor tank and, in part, ultimately become airborne thus released to the atmosphere via building ventilation. Because of the large inventory of coolant water in the UUTR tank (~8,000 gallons), migration of fission product gasses to the air is very limited and slow. This operational occurrence is addressed in **Chapter 13** as the maximum hypothetical accident (MHA) for the UUTR. Similarly, the fuel element failure, although very unlikely, can occur while the reactor is operating normally, abnormally, or when it is shut down. Although this type of failure could occur during normal operation, its occurrence would be evident from the pool tank radiation monitors and reactor room air monitors. No further reactor operation would take place after such an event until the situation is evaluated and eliminated (i.e., the failed element located and removed from the core).

Production of Argon-41 and Nitrogen-16 may be different than the routine operations for some experimental configuration and possible other scenarios; these scenarios do not produce long term, routine radioactive effluent, but are assessed to determine if the amount of radioactive effluent is so high as to impact the annual exposure that might result from routine operations.

Production of Argon-41. The UUTR reactor core has Fast Neutron Irradiator Facility (FNIF) and Pneumatic Irradiator (PI). The FNIF was designed to provide 1 MeV equivalent neutron flux and is used for sample irradiation at the reactor core power of 1 kW only. At that power, the total neutron flux in the irradiator is about 2.1×10^9 neutrons/cm^2-sec. The thermal neutron component in the FNIF is less than 10% of the total neutron flux (~2.1×10^8 neutrons/cm^2-sec). The volume of the FNIF container is approximately 14,000 cm^3. The actual air volume of the container is about 10,000cm^3 because two lead bricks, flux map plate, and other shielding materials occupy the inside of the container. The total usage of this irradiator for last 10 years was less than 50 minutes/year. The concentration of Argon-41 produced in the air at 1 kW for time *t* is given with:

$$X = \frac{N(40)\phi\sigma(40)}{3.7 \times 10^4}\left[1 - e^{-\lambda(40)t}\right]\mu Ci/cm^3$$

where: ϕ = neutron flux (2.1×10^8 neutrons/cm^2-sec)
$N(40)$ = Number of Ar-40 atoms/cm^3 = 2.5×10^{17} atoms/cm^3
$\sigma(40)$ = cross section for Argon-41 production = 0.53×10^{-24} cm^2
$\lambda(41)$ = decay constant for Argon-41 = 1.06×10^{-4}/sec

Using these values, the maximum possible amount of Argon-41generated per hour in the FNIF container will be 2.05 μCi/hr. The normal reactor room exhaust is

$6.1x10^5$ cm^3/sec ($2.196x10^9$ cm^3/hr). Thus, on average over one hour, the Argon-41 concentration in the exhaust will be:

$$\frac{2.05\frac{\mu Ci}{hr}}{2.196\times10^{9}\frac{cm^3}{hr}}=9.33\times10^{-10}\frac{\mu Ci}{cm^3}$$

This is the most conservative estimate. The actual use of the UUTR's FNIF is 15 minutes per 4 months. In this case, the activity of Argon-41 will decrease to $2.8x10^{-10}$ μCi/cm^3. Therefore, the generated amount of Argon-41 from the FNIF container is well below the 10CFR20 limits.

Production of Nitrogen-16. About 21% of the FNIF container volume (approximately 10,000 cm^3) is occupied with oxygen. Using same methodology as the one described in **Section 5.6**, the number of Nitrogen-16 N^N can be calculated from this equation:

$$N^N=\frac{\Phi_V N^0\sigma^0}{\lambda^N}\left[1-e^{-\lambda^N t}\right]$$

where

N^N = Nitrogen-16 atoms per cm^3 of water

Φ_V = virgin fission neutron flux = $2.1x10^9$ neutrons/cm^2-sec at 1 kW

N^0 = oxygen molecule per cm^3 of air = $5.323x10^{18}$ molecules/cm^3

Container volume = 10,000 cm^3

σ^0 = absorption cross section of oxygen = $2x10^{-29}$ cm^2

λ^N = Nitrogen-16 decay constant = $9.35x10^{-2}$/sec

t = average time of exposure in reactor = 600 sec/experiment

From these values, the calculated number of Nitrogen-16 in the FNIF container is 2.39 Nitrogen-16 molecules/cm^3. Thus, the activity from N^{16} is $2.23x10^{-2}$ Bq/cm^3 ($6.04x10^{-4}$ μCi/cm^3). The FNIF container is kept in the reactor pool water for 30 minutes after irradiation. During these 30 minutes, the Nitrogen-16 activity will decay to $2.61x10^{-81}$ μCi/cm^3. After 30 minutes decay time, the total activity of Nitrogen-16 in the FNIF container will be $2.61x10^{-77}$ μCi/cm^3. Thus, the Nitrogen-16 activity in the FNIF container is negligible.

The UUTR has a Pneumatic Irradiator (PI) which uses pressurized Helium-gas to send a sample into the core and out from the core. Helium is a stable gas with the zero (n,γ) cross section. In the worst case, if any Helium-5 is generated from Helium-4, it will not escape from the water surface because of the Helium-5 half life of only $7.6x10^{-22}$ sec.

Neutron interactions with structural and control materials, including cladding, as well as materials irradiated for experimental purpose result in the formation of activation products. These products are in the nature of fixed sources and are mainly a source of occupational radiation exposure. Administrative controls preclude the significant formation of airborne activation products, other than the aforementioned Argon-41.

Two other experimental facilities, the dry tube irradiator and the beam ports, when used will produce the Argon-41. There is no plan to use the beam port at the UUTR. And therefore, there is no possibility to generate Argon-41 from these beam ports. Argon production in the dry tube irradiator is very small because the effective volume of the dry tube is very small. The approximate volume of dry tube sample holder is 35 cm^3. The thermal neutron flux at 90kW in the dry tube is 3.6×10^{11} neutron/cm^2-sec (measured from experiment). Using the variables shown in **UUTR SAR 11.2.1.2**, the activity of Argon-41 from the dry tube for 90kW per hour will be:

$$X = \frac{N(40)\phi\sigma(40)}{3.7 \times 10^4}\left[1 - e^{-\lambda(40)t}\right]\mu Ci/cm^3 = 0.409\mu Ci/cm^3 - hr$$

Using experimentally measured thermal neutron flux of 3.6×10^{11} neutrons/cm^2-sec, the activity from the dry tube will be 0.409 μCi/cm^3-hr for 90kW. Thus, on average per one hour, the Argon-41 concentration form the dry tube in the exhaust will be:

$$\frac{0.409\dfrac{\mu Ci}{cm^3 - hr} \times 35cm^3}{2.196 \times 10^9 \dfrac{cm^3}{hr}} = 6.52 \times 10^{-9}\frac{\mu Ci}{cm^3}$$

The Argon-41 transport from the tube must occur by diffusion into the reactor room before entering the exhaust stacks. The beam ports remained capped and there are no current plans to open the existing beam ports. Before future use of these beam ports, review and approval for such an application must be obtained from the UUTR Reactor Safety Committee and the NRC if deemed necessary.

11.1.1.1.6 Radiological Assessment of Argon-41 from the Pool Water

Argon-41 in the reactor room appears because it evolves from the primary coolant into the air of the room. This evolution results from the reduced solubility of argon in water as the water temperature increases. Detailed calculations addressing the production and evolution of Argon-41 from the primary coolant are shown as follows: variables as listed and the data from **Table 11.2-1** are used in calculation of Argon-41 concentrations in the core region, in the reactor tank outside of the core, and in the reactor room area:

V_{core} = volume of the active core = 3.14x25^{2}x40 = 7.85x10^4 cm^3
V_{fuel} = 78x3.14x40x1.865^2 cm^3 = 3.41x10^4 cm^3
$V_{heavy\ water\ element}$ =12x3.14x1.865^{2}x40 = 5.24x10^3 cm^3
$V_{graphite\ element}$ = 12x3.14x1.865^{2}x40 = 5.24x10^3 cm^3
$V_{pneumatic\ irradiator}$ = 1x3.14x1.865^{2}x40 = 4.37x10^2 cm^3
$V_{neutron\ source}$ = 1x3.14x1.865^{2}x40 = 4.37x10^2 cm^3
$V_m = V_{fuel} + V_{heavy\ water\ element} + V_{graphite\ element} + V_{pneumatic\ irradiator} + V_{neutron\ source}$
= 4.55x10^4 cm^3
V_{water} = volume of water in active region = $V_{core} - V_m$ = 3.3x10^4 cm^3
$N(40)$ = Argon-40 atomic density (atoms/cm^{-3}) in coolant
ϕ_{th} = 9.40x10^{11} n/cm^2-sec for 100 kW

It is assumed that the reactor core can be approximated as a cylinder with the radius of 25 cm and height of 40 cm. Saturated concentration of Argon-40 in water at the coolant inlet temperature of 293.15 °K (20 °C) and 1 atm is approximately 6.7x10^{-5} g/cm^3 [Gevantman, LH. Solubility of selected Gases in Water In:Lide, Dr, editor, CRC Handbook of Chemistry and Physics, 84th ed. Boca Raton: CRC Press, 2003. P 8-88]. If it is assumed that air is saturated with water vapor above the water tank (17.5 mmHg vapor pressure at 20° C) and that the mole fraction of argon in dry air is 0.0094, the partial pressure of argon in air above the tank is 0.0094(760-17.5) = 7.0 mmHg. By Henry's law, the concentration of Ar-40 in water at the inlet temperature of 20 °C is obtained with: 6.7x10^{-5}(7.0/760) = 6.17x10^{-7}g/cm^3. From this result, the Argon-40 density in the coolant is calculated and is equal to $N(40)$ = 9.3x10^{15} atoms/cm^3. The number of atoms per second of Argon-41 produced in 100kW core is $N(40)\times V_{water}\times\sigma_{40}\times\phi_{th}$ = 1.4x10^8 atoms/sec. Assuming that 100% of the Argon-41 atoms escape to the reactor room area, the equilibrium Argon-41 concentration in the reactor room can be calculated using the following equation:

$$\frac{Bq}{cm^3} = \frac{N(40)V_{water}\sigma_{40}\phi_{th}\lambda_\gamma}{V_{reactorroom}(\lambda_\gamma + \lambda_v)}.$$

Based on the variables listed in **Table 11.2-1**, the equilibrium Argon-41 concentration during full power steady state at 100kW in the reactor room area would be 0.024 Bq/cm^3 (=6.4 x 10^{-7} µCi /cm^3). Appendix B of 10CFR Part20 Derived Air Concentration (DAC) for a semi-infinite cloud of Argon-41 is 3 x 10^{-6} µCi/cm^3. Therefore, under realistic conservative calculations, equilibrium Argon-41 concentration during full power steady state operation at 100 kW is lower than DAC. Since personnel do not stay in the reactor room for extended periods when UUTR is at power, this does not present a restriction. Actual measurements of Argon-41 in the reactor room after reactor operation for about 4.0 hours at 90 kW (reactor room exhaust system on) showed that the Argon-41 concentrations averaging about 2.67×10^{-8} μCi/cm^3 for areas that are occupied during normal work in the room.

11.1.1.1.7 Radiological Assessment of Argon-41 Outside the Operations Boundary

The Argon-41 from the reactor room is discharged from the UUTR through the facility's exhaust stack, which is 40 ft (12.19 m) above ground level. An atmospheric dilution will reduce the Argon-41 concentration considerably before the exhaust plume returns to ground level locations that could be occupied by personnel. The flow rate from the reactor room exhaust is 2.2×10^9 cm^3/hr (=1,300 CFM). At the steady state concentration as computed in the previous section, the release rate at 100 kW would be:

$$Q = 3.27 \times 10^{-6} \frac{\mu Ci}{cm^3} \times 6.11 \times 10^5 \frac{cm^3}{\sec} = 2\mu Ci/\sec$$

Maximum downwind concentration (μCi/m^3), at grade, may be computed using the Sutton formula [Slade, D. H.(ed.), "Meteorology and Atomic Energy," Report TID-24190, U.S. Atomic Energy Commission, 1968]:

$$C_{\max} = \frac{2Q}{e\pi\bar{u}h^2}\frac{C_z}{C_y}$$

where:

$\bar{u}$ = mean wind speed (m/sec)

e = 2.718

C_z and C_y are diffusion parameter in the crosswind and vertical directions respectively.

Maximum concentration downwind occurs at distance d (m) as follows:

$$d = \left[\frac{h}{C_z}\right]^{\frac{2}{2-n}}$$

where:

n = parameter associated with the wind stability condition

h = 12.19 m

Based on the MNRC SAR for n and C_z [Facility Safety Analysis Report, Rev. 2, McClellan Nuclear Radiation Center Reactor, April 1998], the atmospheric dispersion calculations for 100kW is shown in **Table 11.1-2**.

The 10CFR20 Appendix B lists that the Effluent Concentration (EC) for Argon-41 is 1×10^{-8} μCi/cm^3 for 50 mrem to the public exposed for a full year of 8,760 hours. This number corresponds to 5.7×10^{-3} mrem/hour per μCi/cm^3. The average operating time of the UUTR for last 10 years is ~50 hours per year, which is less than 1% of 8,760 hours/year.

Over the full range of conditions indicated in **Table 11.1-2**, the peak downwind concentration is substantially below the DAC of 3×10^{-6} $\mu Ci/cm^3$ established in 10CFR20 Appendix B and less than the permissible effluent concentration of 1×10^{-8} $\mu Ci/cm^3$ for all meteorological conditions.

Table 11.1-2 Atmospheric dispersion calculations for 100 kW UUTR

Pasquill stability class	u (m/s)	n	C_y ($m^{n/2}$)	C_z ($m^{n/2}$)	d(m)	C_{max} ($\mu Ci/cm^3$)
Extremely unstable (A)	1.6	0.2	0.31	0.31	59.13	1.97×10^{-9}
Slightly Unstable (C)	4.0	0.25	0.15	0.15	152.32	7.87×10^{-10}
Slightly stable (E)	3.5	0.33	4 C_z	0.075	444.46	1.69×10^{-11}
Extremely stable (G)	0.77	0.5	8 C_z	0.035	2450.46	1.79×10^{-11}

11.1.1.1.8 Radiological Assessment of Nitrogen-16 Sources

Nitrogen-16 is generated by the reaction of fast neutrons with oxygen and the only significant source results from reactions with oxygen in the reactor pool water. Nitrogen-16 has a half-life of 7.13 seconds and emits 6.14 MeV gamma rays. The effective cross-section σ_{np} for the Oxygen-16 to Nitrogen-16 reaction, averaged over the fast neutron spectrum is 0.02 mbarns (2×10^{-29} cm^2). The atomic density N_N of the nuclide as it leaves the reactor core is given in terms of oxygen density in water N_o = $3.34\times10^{22}/cm^3$, as

$$N_N = \frac{\phi_f N_o \sigma_{np}}{\lambda_{16}}\left(1 - e^{-\lambda t}\right)$$

where t represents the time in the core. The averaged fast neutron flux for the UUTR is 2.6×10^{12} neutrons/cm^2-sec for 100 kW. From the above equation, it follows that the number of Nitrogen-16 is 1.47×10^{7} atoms/cm^3. As the coolant leaves the core, it passes through the actual flow area of 395.6 cm^2. Operation at power requires primary cooling. Primary coolant enters the pool through coolant supplier approximately 2 ft below the water surface. The coolant outlet is located about 2 ft above the core. Core exit is at 22 ft below the pool surface. Exit flows are a small fraction of mixing flow, and under these conditions it is considered adequate to use a nuclide concentration reduced by the ratio of the total core exit surface area (approximately 395.6 cm^2 for 78 elements) and the pool (surface area = 4.67×10^4 cm^2); mixing reduces the concentration of Nitrogen-16 from the core exit by 0.0046 [Reed Research Reactor Safety Analysis Report, August 2007]. Thus, above the reactor core, the actual Nitrogen-16 concentration will be approximately 6.76×10^4 Nitrogen-16/cm^3. Because the characterization of the flow velocity of Nitrogen-16 is very difficult and complicated, a flow rate from the core to the surface is conservatively assumed as core exit flow rate for dose rate calculation.

In this calculation, two major gamma energies, 6.13 MeV and 7.11 MeV, were used. The total dose rate in the reactor room area was calculated based on:

$$\dot{D} = \frac{\lambda \left[N^{16}\right]_w}{2\mu K}\left[1 - E_2(\mu h)\right]$$

where

λ= Nitrogen-16 decay constant (=0.0971/sec)
$[N^{16}]_w$ = concentration of Nitrogen-16 in the water
μ = linear attenuation factor for 6 MeV gamma, cm^{-1} (0.0277/cm)
K = flux-to-exposure rate conversion (1.65×10^5 gammas $R^{-1}hcm^{-2}s^{-1}$)
h = thickness of Nitrogen-16 bearing water, and
$E_2(\mu h)$ = exponential function (~ 0).

For this calculation, the second order exponential integral function was conservatively assumed to be zero. In the reactor room area, Nitrogen-16 activity is affected by dilution, ventilation, and decay. The accumulation of Nitrogen-16 in the reactor room under equilibrium conditions is determined by:

$$\left[N^{16}\right] = \frac{\left[N^{16}\right] v_e A}{\lambda V_r + q}$$

where

v_e = escape velocity of Nitrogen-16 from the water surface [0.009 cm/sec: Dorsey, N.E., "Properties of Ordinary Water-Substances," pp. 537-544, Reinhold Publ. Co., New York, New York]
A = area of water surface (4.67×10^4 cm^2)
V_r = room volume (5.65×10^8 cm^3)
q = exhaust rate (6.11×10^5 cm^3/sec)

Using these variables and reducing factor of 00046, the concentration of $[N^{16}]_w$ near the water surface will be 0.51 Nitrogen-16/cm^3 for 100 kW and 0.7 Nitrogen-16/cm^3. The gamma exposure rate due to immersion from an equilibrium concentration of Nitrogen-16 in the air is calculated as follows:

$$\dot{X} = \frac{\left[N^{16}\right]_a B\lambda\left(1 - e^{-\mu_s R_0}\right)}{2\mu_s g}$$

where

B = dose buildup factor (~1)

$[N^{16}]_a$ = Nitrogen-16 concentration in the reactor room air

μ_s = linear absorption coefficient (3.03x10^{-5} /cm for air at 6 MeV)

R_o = radius of reactor room sphere (~5 m)

g = dose conversion factor (160 Bq cm^{-2}mR^{-1}h).

From the equation above, the dose rate in the reactor room area will be 0.077 mR/hr for 100 kW UUTR. During the reactor operation at 90 kW, Area Radiation Monitor reads no more than 0.05 mR/hr all the time.

11.1.1.2 Liquid Radioactive Sources

11.1.1.2.1 UUTR Pool Water Activity

The two thirds of the UUTR pool are below ground level. During normal operating condition, there is no leakage of coolant from the primary or secondary cooling system; there is no liquid radioactive material produced. The only liquid radioactive material generated would come from the neutron activation products in the primary coolant. The demineralizer resin bed removes most of these radioactive materials. The UUTR's monthly survey shows that total activity of the pool water is below 1 nCi/liter. The concentration of any radionuclide in the primary coolant depends on the UUTR power and the operating time. The UUTR is operated at 90 kW in general about 50 hours/year. The pool water activity for last couple of years is shown in **Table 11.1-3.** As shown in **Table11.1-3**, the activity of the pool water and the effluent volumes are both very small, and thus not considered to represent a radiation source.

In the reactor room floor, there is a small liquid storage that can hold approximately 40 to 50 gallons of liquid. In the past 35 years of operation history, this storage was not used as storage for any liquid effluent or any liquid radionuclide sources or releases. This storage may be used to hold liquid radioactive releases if such a release occur during the experiments independent of the reactor, and keep the liquid until radionuclides contained in decay out.

11.1.1.2.2 Tritium Activity of the UUTR Pool Water

The ventilation system of the UUTR facility discharges 0.61 m^3/sec of air from the reactor room into the atmosphere. A small amount of tritium is produced in the UUTR pool water through the neutron activation of the deuterium present in the pool water. The calculations (as shown in details below) for the reactivity of the UUTR pool water gave the value of 4.12 x 10^{-10} Ci/liter. The pool evaporation rate is approximately 11 liters/day and the reactor room exhaust discharge is approximately

5.27×10^{10} cm^3/day. Under the assumption that the tritium content of the pool water and the evaporated water are the same (this is most conservative assumption), the reactor room exhaust would contain 8.6×10^{-14} μCi/cm^3 of tritium. The 10 CFR20 limit for tritium is 1×10^{-7} μCi/cm^3 and EPA limit for this isotope is 1.5×10^{-9} μCi/cm^3. Therefore, the tritium generation from the UUTR pool water is significantly below the limits defined in both, the 10CFR20 and the EPA.

Table 11.1-3 The UUTR monthly pool water survey results for UUTR operating at 90 kW

Year Month	2008 (nCi/liter)	2009 (nCi/liter)	2010 (nCi/liter)
January	0.820	0.268	0.845
February	0.327	0.473	0.605
March	0.345	0.452	0.760
April	0.432	0.520	0.758
May	0.568	0.232	
June	0.742	0.281	
July	0.725	0.232	
August	1.115	0.348	
September	0.667	0.467	
October	0.510	0.430	
November	0.481	0.492	
December	0.390	0.663	

<u>Calculations:</u>

Assumptions: The UUTR holds 8,000 gallons of water (30,240 liters=3.024×10^{7} cm^3)
The UUTR operates on average 50 hours/year at 90kW
Heavy water density = 1.1056 g/cm^3
1 mole of D_2O = 22grams,
D_2O concentration in water = 1/3,200
Actual volume of water that interacts with thermal neutrons
= 1.017×10^{6} cm^3
Thermal neutron flux near the side of the core = 3×10^{11} n/cm^2-sec,
Absorption cross-section of D_2O = 0.52 mbarns,
N_A = 6.023×10^{23} molecules/mole, and
Volume of D_2O = 318 cm^3.

The mass M of D_2O in the pool water is

$$M = \rho \times V = 1.1056\frac{g}{cm^3} \times 318cm^3 = 336grams$$

meaning that these 336 grams of D_2O corresponds to 15.3 moles of D_2O that contains 9.2 x 10^{24} D_2O molecules. The reaction rate *R* for tritium generation is given by

$$R = N \times \sigma \times \phi = \frac{9.2 \times 10^{24} atoms}{318cm^3} \times 0.52 \times 10^{-3} \times 10^{-24} cm^2 \times 3 \times 10^{11} \frac{neutrons}{cm^2 - \sec}$$
$$= 4.51 \times 10^6 molecules/cm^3 - \sec$$

For 50 hours of normal reactor operation, the total number of tritium in the pool water is

$$4.51 \times 10^6 \frac{molecules}{cm^3 - \sec} \times 318cm^3 \times 50hours \times \frac{3600\sec}{hour} = 2.51 \times 10^{14} D_2O$$

The total activity of tritium in the pool water is therefore,

$$Ac = \frac{\ln 2}{T_{\frac{1}{2},tritium}} \times 2.51 \times 10^{14} = 4.61 \times 10^5 Bq = 1.24 \times 10^{-5} Ci$$

If this amount of tritium is mixed with the reactor pool water (3.024×10^7 cm^3), the tritium reactivity per unit volume in the pool water will be

$$\frac{1.24 \times 10^{-5} Ci}{3.024 \times 10^7 cm^3} = 4.12 \times 10^{-13} \frac{Ci}{cm^3} = 4.12 \times 10^{-7} \frac{\mu Ci}{cm^3}$$

The reactor room exhaust discharge would contain

$$\frac{4.12 \times 10^{-10} \frac{Ci}{liter}}{5.27 \times 10^{10} \frac{cm^3}{day}} \times 11 \frac{liters}{day} = 8.60 \times 10^{-14} \frac{\mu Ci}{cm^3}.$$

This activity is significantly less than limits specified by the 10CFR20 or the EPA.

11.1.1.2.3 Tritium from Heavy Water Element and Thermal Irradiator (TI)

The UUTR core contains 12 heavy water reflector elements and thermal irradiator (TI). TI contains approximately 1.53×10^4 cm^3 (1.53×10^4 grams) of D_2O. Each heavy water element contains approximately 454 grams of heavy water. Heavy water has the thermal neutron absorption cross-section of 0.52 mbarn. Each heavy water element contains 1.24×10^{25} molecules of D_2O. Under the assumption that the UUTR has a maximum thermal neutron flux of 3×10^{11} neutrons/cm^2-sec near the heavy water elements (this is a measured thermal neutron flux at the thermal irradiator), then the number of tritium molecules from each heavy water element will be approximately 1.93×10^9/sec. The total amount of tritium from 12 heavy water elements after 50 hours of reactor operation will be 4.18×10^{15} atoms. The total activity of tritium atoms from 12 heavy water elements for 50 hours operation is 7.45×10^6 Bq (2.01×10^{-4} Ci).

The D_2O in the TI will generate tritium also. The number of D_2O molecules in TI is 6.16×10^{26}. Thus, the TI will generate ~9.61×10^{10} tritium atoms per second. After 50 hours of the UUTR operation per year, the number of tritium atoms in TI will be 1.73×10^{16}. Thus, the total number of tritium atoms from 12 heavy water elements and TI after 50 hours reactor run become 2.15×10^{16}. The total activity from the tritium is 3.83×10^7 Bq (1.04×10^{-3} Ci).

Both the TI and heavy water elements tubes made of aluminum were leak tested by pressurizing to 30 psig (206 kPa).The leaking from the heavy water elements or TI is not possible but under the worst case scenario that all tritium leak from the containers and is suddenly mixed with 8,000 gallons of the UUTR pool water, the tritium activity per unit volume after 50 hours per year operation will be

$$\frac{1.04 \times 10^{-3}\,Cl}{3.024 \times 10^{7}\,cm^3} = 3.42 \times 10^{-11}\,\frac{Ci}{cm^3} \quad \text{for 90kW UUTR}$$

Assuming that the tritium content of the pool water and evaporated water are same, the tritium activity from the reactor room exhaust would be 7.14×10^{-12} $\mu Ci/cm^3$, thus:

$$\frac{3.42 \times 10^{-8}\,\frac{Ci}{liter} \times 11\,\frac{liter}{day}}{5.27 \times 10^{10}\,\frac{cm^3}{day}} = 7.14 \times 10^{-18}\,\frac{Ci}{day} = 7.14 \times 10^{-12}\,\frac{\mu Ci}{cm^3}$$

Even in this worst case scenario, the tritium activity in the reactor room from the pool water and all heavy water elements including TI will be 7.23×10^{-12} $\mu Ci/cm^3$ for 100 kW UUTR, which is significantly lower than 10CFR20 or EPA limits, therefore the UUTR does not produce the tritium to be counted as a source term.

11.1.1.3 Solid Radioactive Sources

Solid sources consist of reactor fuel, a startup neutron source, 1.8 Ci Am-Be neutron source, and fixed radioisotope sources such as those used for instrumentation calibration. Solid wastes include: ion-exchange resin used in reactor-water cleanup, irradiated samples, labware and anti-contamination clothing associated with reactor experiments and surveillance or maintenance operations. The solid radioactive sources associated with reactor operations are summarized in **Table 11.1-4** (The data based on the UUTR fuel log book and record from the General Atomics Data sheet). Because the actual inventory of fuel and other sources continuously changes in normal operation, the information in the table is to be considered representative rather than an exact inventory.

Table 11.1-4 Elemental composition of 100kW UUTR core

Fuel zone		Reflector zone		Grid plate	
Element	Mass fraction	Element	Mass fraction	Element	Mass fraction
Fe	0.0450	Fe	0.0150	Al	1.00
Al	0.0020	Al	0.2000		
C	0.0600	C	0.7100		
Cr	0.0115	Cr	0.0010		
Ni	0.0050	Ni	0.0002		
Mn	0.0012	Mn	0.0001		
Zr	0.7800	H	0.0737		
H	0.0140				
U	0.0740				
V	0.0073				

The main solid radioactive source related to the UUTR is the fission products generated in the reactor fuel. Typical UUTR fuel element releases 50 R/hr when in the air at 3 ft distance (after it is removed from the UUTR core). The fuel elements always stay 22 ft under the water in the UUTR pool, therefore this source will not present any public hazard or hazard to personnel at the UUTR. Typical radiation level near the control bridge area when the UUTR operates at the nominal power of 90 kW is 20 μR/hr. This radiation level was measured using Ludlum Model 19 detector during the reactor operation on February 19, 2010. Another possibility of solid radioactive sources to be found at the UUTR facilities may come from the irradiation of the various solid samples. The expected activity of a sample is calculated before its irradiation in one of the UUTR ports; generally the sample is stored in the pool until most of its radioactive isotopes decay out. The ARM (Area Radiation Monitor) system in the control room has two setting points: 1 mR/hr for low dose setting and 10 mR/hr for the high dose setting. In this type of radiation exposure, the main isotopes are aluminum and sodium. These irradiated samples are stored in the water before any measurement and stored in a lead cage after the experiment. Typical gamma activities from these samples are typically less than 1 μCi.

Radioactive solid waste is generally considered to be any item or substance no longer of use to the UUTR facility, which contains or is suspected of containing radioactivity above the background level. Volume of solid waste at the UUTR is small, and the nature of the waste items is limited and of known characterization. Consumable supplies such as absorbent materials or protective clothing are declared radioactive waste if radioactivity above background is found to be present. The RHD at the University of Utah is responsible for the administration of radioactive waste disposal for the UUTR facility. The University of Utah has a Broad Scope Medical and Academic License granted through the Utah Division of Radiation Control. Utah is an Agreement State.

When possible, solid radioactive waste is initially segregated at the point of origin from items that are not considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the future need for the items and materials involved.

11.1.1.4 Gamma Dose Rate from the 100kW UUTR Core

The UUTR maintains a program of area monitoring in controlled and uncontrolled areas in addition to regular personnel monitoring. The data in **Tables 11.1-5** and **11.1-6** were obtained while the reactor was operating at a peak level of 90 kW. **Table 11.1-5** summarizes average personnel radiation doses for the past six years. These averages reflect personnel doses during typical UUTR activities including normal reactor operations, fuel inspections, source handling, and various research protocols. These summaries validate UUTR compliance with 10CFR20 dose limits for radiation workers and members of the public.

Table 11.1-5 Average UUTR personnel dose (All "M" values recorded at 1 mrem)

Year	Average (mrem)	Stdev	N	Max	Min
2004	4.9	5.2	8	14	1
2005	3.0	2.7	11	9	1
2006	6.9	5.5	15	21	1
2007	2.9	2.1	19	9	1
2008	6.5	7.4	14	24	1
2009	1.7	1.1	10	4	1

Table 11.1-6 summarizes area radiation dosimeter results for a few key indicator areas. These results were obtained through a variety of operating conditions, which reflect the range of radiation use at the UUTR facility. The area dosimeter labeled "NE Floor" is attached directly to the reactor tank, "Control Room" is located in the reactor control room near the core viewing window, and "2126 B Board" is located directly above the reactor room in an uncontrolled hallway (**Fig. 11.1-1**). These data additionally demonstrate radiation levels in controlled non-controlled areas are at levels

To further ensure compliance with radiation dose limits, the UUTR estimates photon dose from the core using a model based on UUTR MCNP 5 modeling and point-source geometry. The selection of a mathematical model to estimate radiation dose is more correctly based on the extended-source characteristics of the core. However, when deriving a model it was desirable to develop a model that provided reasonable estimates of photon dose from the core throughout the facility, but also be relatively simple to use and be reproducible across a wide range of applications. Thus, even though the core would more correctly be modeled with using extended source geometry at distances near the water tank, a model based on point-source geometry was selected because of the ease of use and reproducibility and wide applicability. As is shown in the **Appendix 11.2.A**, this model is representative of observed area dosimeter values.

Table 11.1-6 Summary results of area dosimeters for three key areas

Year	Time of 90kW reactor operation in a year	"NE Floor (Dosimeter 3)"	"Control Room (Dosimeter 5)"	"2126 Bboard"
	(hour)	(mrem)	(mrem)	(mrem)
2004	24.833	48	5	1
2005	51.915	96	1	1
2006	31.467	119	8	7
2007	52.463	66	3	7
2008	36.445	87	9	8
2009	28.400	42	2	3

In creating the model, some generalizations about the reactor were made. Generally, the size of the core can be described as a right circular cylinder of radius 30 cm and height of 60 cm. The core is surrounded by a right circular cylinder of water of radius 183 cm and a height of 729 cm. The water is surrounded by an additional cylinder of sand (ρ=1.6 g/cm^3) with thickness of 61 cm. The core is situated in the bottom, center of the water cylinder, with the top of the core approximately 653 cm below the surface of the water.

Based on these general assumptions, the following model for estimating dose in air from photons has been derived:

$$\phi(E)\left(\frac{\gamma}{cm^2 s}\right) = \frac{S_A(E)N}{4\pi a^2} B(E) e^{-\mu(E)_1 x_1} B(E) e^{-\mu(E)_2 x_2}$$

ϕ *(E)* (γ/cm^2-sec) is the photon fluence rate at the point of interest for photon energy E
S_A *(E)* (γ/cm^2-n) is the photon flux density per neutron determined from MCNP 5 modeling for photon energy E
N (n/sec) is the total core neutron flux determined from MCNP5 modeling
a (cm) is the distance from the core to the point of interest
B (E) is the photon dose buildup factor for photon energy E
μ *(E)* (cm^{-1}) is the linear attenuation coefficient of materials 1 and 2 (Water and Sand) for photon energy E
x (cm) is the thickness of the attenuating materials 1 and 2
Dose in the air can then be determined using the following formula:

$$\dot{D}(E)\left(\frac{Gy}{hr}\right) = \phi(E)\frac{\mu_{en}}{\rho}(E)CF$$

$\dot{D}(E)(Gy/hr)$ is the dose rate in air for photon energy E at the point of interest
$\phi(E)(\gamma/cm^2 - \sec)$ is the photon fluence rate at the point of interest for photon energy E
$\frac{\mu_{en}}{\rho}(E)(cm^2/g)$ is the mass attenuation coefficient in air for photon energy E
CF is the product of all conversion factors to derive the appropriate units
The total dose is simply the sum of all $\dot{D}(E)$

$$\dot{D}\left(\frac{Gy}{hr}\right) = \sum_E \dot{D}(E)\left(\frac{Gy}{hr}\right)$$

The total dose for the year can be determined by multiplying $\dot{D}(E)$ by the total operating time for the entire year.

An example of the use of this formula and the agreement between the results and observed area dosimetry data is shown as follows. Photon linear attenuation coefficients were calculated by the National Institute of Standards and Technology's XCOM calculator (http://www.nist.gov/physlab/data/xraycoef/index.cfm). Buildup factors were obtained from the *Radiological Health Handbook* (US Department of Health, Education, and Welfare, *Radiological Health Handbook,* 1970).

As demonstrated through calculations, the dose levels from operation of the UUTR reactor at 90 kW (and 100kW) are and will be within the dose limits established in 10CFR20; the dose limits to occupational workers, members of the public, and dose levels in non-controlled areas are within the established limits.

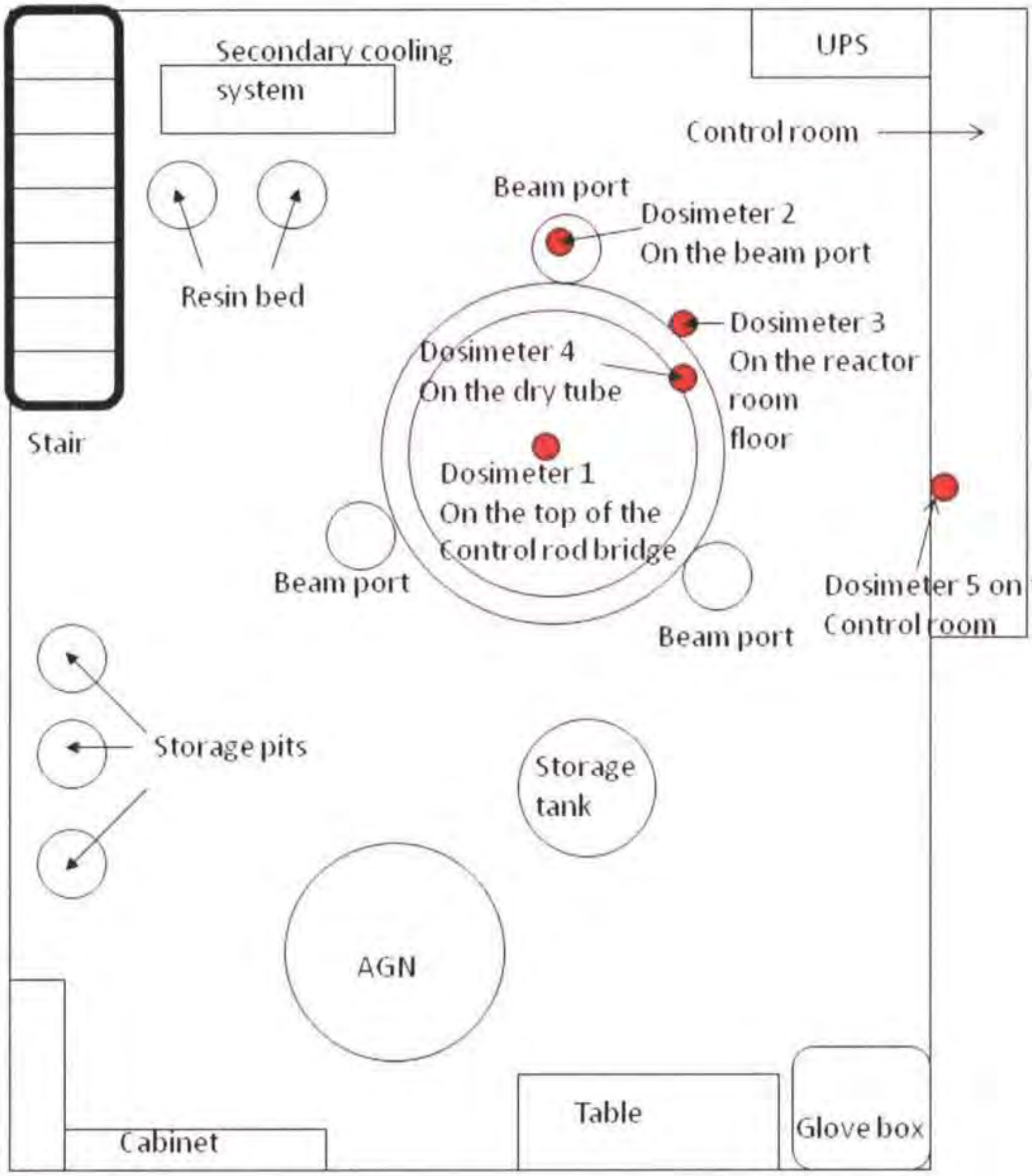


Figure 11.1-1 various locations throughout the facility where area monitoring is employed

From 8/2008 through 6/2009, the UUTR reactor operated for 13.45 hours with a total thermal power generation of 891.2 kW-hr. The measured dose from the area dosimeter attached to the side of the reactor tank was 55 mrem. Using the calculation method described above (calculation spreadsheet is included in **Table 11.2.A-1**), and a dose in air to effective dose equivalent conversion (Ferdeal Guidance Report 12, *External Exposure to Radionuclides in Air, Water and Soil*) of 0.75 Sv/Gy, 65 mrem was estimated. This example demonstrates the above described model is reasonably representative of an expected dose rate from the operating reactor.

11.1.2 Radiation Protection Program

The Radiation Protection Program was prepared by personnel of the UUTR and the University of Utah Radiological Health Department (RHD) in response to the requirements of 10CFR Part20. The goal of the Program is the limitation of radiation exposures and radioactivity releases to a level that is As Low As Reasonably Achievable (ALARA) without seriously restricting operation of the Facility for purposes of education and research. The Program is executed in coordination with the RHD. It has been reviewed and approved by the Reactor Safety Committee (RSC) for the UUTR Facility.

Production and use of radioactive materials within the reactor lab are subject to the guidelines issued by the University's RHD. In addition, the UUTR Facility follows internal procedures that fall within the guidelines of the University of Utah, the Utah State Division of Radiation Control, and federal regulations, the American National Standard *Radiation Protection at Research Reactor Facilities*, and Regulatory Guides issued by the NRC.

11.1.2.1 Management and Administration

Preparation, audit, and review of the Radiation Protection Program are the responsibility of the Director of the UUTR facility (Utah Nuclear Engineering Facilities inclusive of the UUTR and other facilities and labs present). The Reactor Safety Committee (RSC) reviews the activities of the Director and audits the Program. Surveillance and record-keeping are the responsibility of the Reactor Supervisor who reports to the Director. ALARA activities, for which record keeping is the particular responsibility of the Reactor Supervisor, are incumbent upon all radiation workers associated with the UUTR Facility. Substantive changes in the Radiation Protection Program require approval of the RSC. Editorial changes, or changes to appendices, may be made on the authority of the Director. Changes made to the Radiation Protection Program apply automatically to operating or emergency procedures; corresponding Program changes may be made without further consideration by the RSC.

11.1.2.2 Training

Implementation of training for radiation protection is the responsibility of the Reactor Supervisor. Personnel who need access to the facility, but are not reactor staff, are either escorted by trained personnel or provided facility access training. Radiation training for licensed operators and staff is integrated with the training and requalification program. The goal of facility access training is to provide knowledge and skills necessary to control personnel exposure to radiation associated with the operation of the nuclear reactor. Specific training requirements of 10 CFR Part 19, 10 CFR Part 20, the Radiation Protection Plan, and the Emergency Plan are explicitly addressed. A facility walkthrough is incorporated. All persons granted unescorted access to the reactor facilities must receive the training and must complete without assistance a written examination over radiation safety and emergency preparedness. Examinations must be retained on file for audit purposes for at least three years. The reactor staff accomplishes health physics functions at the reactor following approved procedures. Therefore, procedure training for the licensed reactor staff training includes additional radiological training. Examinations for reactor staff training are prepared and implemented in accordance with the Requalification Plan. All personnel and students are required to attend web-based training and 3-hour hands-on training at the RHD and pass the exam in order to be eligible to attend the lab classes or be accepted into the Senior Reactor Operator training classes. In summary, the general levels of training include the following:

- *Radiation/Radioactive Material User Orientation* - All personnel permitted unescorted access to the UUTR Reactor Building shall receive training in radiation protection as required by 10CFR19.12. Initial training shall cover the following areas in sufficient depth for the work being done:
    - storage, transfer, and use of radiation and/or radioactive material in portions of the restricted area, including radioactive waste management and disposal;
    - health protection problems and health risks (including prenatal risks) associated with exposure to radiation and/or radioactive materials;
    - precautions and procedures to minimize radiation exposure (ALARA);
    - purposes and functions of protective devices;
    - applicable regulations and license requirements for the protection of personnel from exposure to radiation and/or radioactive materials;
    - responsibility of reporting potential regulatory and license violations or unnecessary exposure to radiation or radioactive materials;
    - appropriate response to warnings in the event of an unusual occurrence or malfunction that involves radiation or radioactive materials, and
    - radiation exposure reports which workers will receive or may request.
- *Examination to demonstrate understanding of the material required for each section of training*:
    - reactor area unescorted access orientation - all personnel permitted unescorted access to the UUTR vital area shall receive additional training to include the following:
        - a reactor area access control rules;
        - emergency evacuation procedures for the reactor area;
        - dosimetry requirements for the reactor area;
        - reactor control room entry procedures and control requirements;

- keys are issued to the Director and Senior Reactor Supervisor only;
- reactor top security;
- location and use of communication systems;
- security door requirements;
- general checkout procedures when exiting the reactor area;
- emergency equipment location and use.

11.1.2.3 Procedures and Document Control

Operation of the Radiation Program is carried out under the direction of the Director of the UUTR facilities and the Senior Reactor Supervisor using formal RHD Procedures. The original copy of the procedures is maintained by the Senior Reactor Supervisor, who is also responsible for the distribution of the reproduced copies. While not intended to be all inclusive, the following list provides an indication of typical radiation protection procedures used in the UUTR program:

- UUTR: testing and calibration of area radiation monitors, facility air monitors,
- RHD: laboratory radiation detection systems, and portable radiation monitoring instrumentation;
- UUTR: working in laboratories and other areas where radioactive materials are used;
- RHD: facility radiation monitoring program including routine and special surveys, personnel monitoring, monitoring and handling of radioactive waste, and sampling and analysis of gaseous effluents released from the facility;
- RHD: monitoring radiation exposure in the environment surrounding the facility;
- RHD and UUTR: administrative guidelines for the facility radiation protection program, including personnel orientation and training;
- RHD: receiving of radioactive materials at the facility and unrestricted releasing of materials and items from the facility;
- RHD: leak testing sealed sources containing radioactive materials;
- RHD: safe transporting of radioactive materials;
- UUTR: general and personnel decontamination procedures;
- RHD and UUTR: personnel exposure investigation procedures;
- UUTR: personnel access procedures for the reactor area;
- RHD and UUTR: spill procedures; and
- RHD and UUTR: ALARA procedures.

11.1.2.4 Audits

The auditing of the UUTR Radiation Protection Program is performed by the Reactor Safety Committee (RSC). The RSC provides objective and independent reviews, evaluations, advice and recommendations on matters affecting nuclear safety at the UUTR. With respect to health physics activities, the RSC is responsible for auditing all procedures, personnel radiation doses, radioactive material shipments, radiation surveys, and radioactive effluents released to unrestricted areas.

Specific auditing responsibilities and requirements are defined in **UUTR SAR 12**.

11.1.3 ALARA Program

11.1.3.1 Policy and Objectives

Management of the UUTR Facility is committed to keeping both occupational and public radiation exposure as low as reasonably achievable (ALARA). The specific goal of the ALARA program is to assure that actual exposures are no greater than 10 % of the occupational limits and 50 % of the public limits prescribed by 10 CFR Part 20.

The Director of the UUTR Facility has the ultimate responsibility for the ALARA program, but has delegated this responsibility to the Senior Reactor Operator.

11.1.3.2 Implementation of the ALARA Program

Planning and scheduling of operations and experiments, education and training are the responsibilities of the Reactor Supervisor and/or the Director of the UUTR facilities. Any action that, in either of their opinions, might lead to a dose of 5 mrem to any individual requires a formal Radiation Work Permit.

The ALARA policy *(It is the policy of the facility administration to keep radiation exposures As Low as Reasonably Achievable (ALARA))* specified in facility Technical Specification has the purpose therefore to reduce radiation exposure to as low a level that is socially, technically, and economically practical. Three general principles followed to reduce the exposures are:

1. Minimize Time. The less time spent near a radioactive source, the less the exposure. One typical way this principle can be implemented is to carefully plan and practice a procedure before the actual implementation with radioactive sources.
2. Maximize Distance. When the distance between the body and the radioactive source is increased, the exposure decreases. Sometimes this can be effectively accomplished with tongs or other devices to hold a source away from the hands

and body. Store and use of radioactive materials far from locations are used for other purposes.

3. Use of Proper Shielding. When an appropriate shielding material is placed between the body and the radioactive source, the amount of radiation exposure is reduced. Use storage pigs and shielding blocks when possible is recommended.
4. Control of Contamination. Minimize exposure by:
   a) Wearing lab coats, gloves, booties & safety glasses/goggles where appropriate
   b) Changing gloves frequently
   c) No eating, drinking, smoking, chewing, or application of cosmetics in radioactive work space
   d) No mouth pipetting
   e) Washing hands at completion of radioactive work
   f) Monitoring hands, clothes and work area regularly

11.1.3.3 Exposure Limits at the UUTR

The radiation exposure limits at the UUTR TRIGA reactor facility shall not exceed the limits specified in the Code of Federal Regulations Title 10, Part 20 entitled "Standard for Protection Against Radiation." The important exposure limits are:

1. Occupational Dose Limits -(10 CFR 20.1201)
   a) Whole body total summation -5 rem/year
   b) Extremities excluding lens of eye -50 rem/year
   c) Lens of the eye -15 rem/year
   d) Skin (shallow dose) -50 rem/year
   e) Airborne exposure not to exceed limits of table 1, appendix B.
2. General Public Dose Limits -(10 CFR 20.1301/20.1302)
   a) Total individual public limit -0.1 rem/year
   b) Unrestricted area maximum dose rate -0.002 rem/hour
   c) Unrestricted area dose limit -0.05 rem/year
3. Minor and Pregnant Women Dose Limits -(10 CFR 20.1502)
   a) Minors and declared pregnant women -10% of 20.1201 limits
   b) Embryo/fetus -0.5 rems
4. Surveys and Monitoring
   a) A system of procedures shall be in place for routine surveys and monitoring of the radiation levels in the facility during reactor operations.

b) A permanent record system shall be maintained of all survey and monitoring activities.

5. Personnel Dosimetry
   a) All personnel at the facility shall be assigned a personnel monitoring device that shall be worn at all times while at the facility.
   b) The responsibility for processing personnel dosimetry devices data and maintaining a permanent record of all exposure shall be the responsibility of the RHD.
   c) Personnel exposure data shall be processed on a monthly basis

11.1.3.4 Review and Audit

Implementation of the ALARA Program is audited annually by the Director as part of the general audit of the Radiation Protection Program.

11.1.4 Radiation Monitoring and Surveying

The radiation monitoring program for the UUTR reactor is structured to ensure that all three categories of radiation sources - air, liquid, and solid - are detected and assessed in a timely manner. Additionally the purpose of the radiation survey program is to assure radiological surveillance over selected UUTR work areas in order to provide current as well as characteristic data on the status of radiation conditions in such areas. Such information is used to confirm that safe radiation working conditions exist within the various operational areas under surveillance. The objectives are:

- to assure that the monitoring program is organized such that routine radiation level and contamination level surveys of specific areas and activities within the facility are performed, and special radiation surveys necessary to support non-routine facility operations are also performed
- to make frequent on-the-spot personal observations (along with recorded data) of radiation work areas; these observations may provide advance warning of needed corrections in order to ensure safe use and handling of radiation sources and other radioactive materials
- to use the information which has been gathered through completion of the first two objectives in order to ensure (and document) that all phases of the UUTR operational and radiation protection programs are in line with the goal of keeping radiation doses to personnel and releases of radioactivity to the environment ALARA.

11.1.4.1 Surveillances

Radiation monitoring surveillance requirements are imposed by the RSC through the Radiation Protection Program (independent of the Emergency Plan) and include:

- Daily surveys - performed in areas where radioactive materials are frequently used. Such surveys involve direct radiation level measurements in areas known to contain constant or changing radiation fields and contamination surveys of the floors and other surfaces in the affected area(s). The areas requiring daily surveys are determined at the discretion of the Senior Reactor Operator and the Director of the Facility;
- Weekly surveys - performed in the areas where radioactive materials are less frequently used. These surveys involve direct radiation level measurements in the areas known to possess constant or changing radiation fields and contamination surveys of the floors and other surfaces in the affected area. The areas requiring weekly surveys are determined at the discretion of the Senior Reactor Operator and the Director of the Facility;
- Monthly surveys - performed in the areas where radioactive materials are infrequently used. The monthly surveys involve direct radiation level measurements in the areas known to possess constant or changing radiation fields and contamination surveys of the floors and other surfaces in the affected area. The areas requiring monthly surveys are determined at the discretion of the Senior Reactor Operator and the Director of the Facility;
- Receipt radiation surveys - required of all incoming packages of radioactive material;
- Special surveys – any non-routine radiation survey requested by any member of the UUTR staff when needed;
- Release surveys - required of any object that is removed from a designated radiation/contamination area prior to it being released from that area.

11.1.4.2 Radiation Monitoring Equipment

Radiation monitoring equipment used in the reactor program is summarized in **Table 11.1-7**. Because equipment is updated and replaced as technology and performance requires, the equipment in the table should be considered as representative rather than exact.

11.1.4.3 Instrument Calibration

Radiation monitoring instrumentation is calibrated according to written procedures. NIST traceable sources are used for the calibration. The Director AND Senior Reactor Supervisor are responsible for calibration of the instruments on site, **Table 11.1-**

7. Calibration records are maintained by the facility staff and audited annually by the RSC. Calibration stickers containing pertinent information are affixed to instruments. Area Radiation Monitor, Continuous Air Monitor, HPGE and LSC are calibrated at the UUTR facility. All portable alpha and gamma detectors are calibrated at RHD.

Table 11.1-7 Radiation Monitoring Equipment Used in the UUTR Radiation Protection Program

Instrument	Location	Function
Area Radiation Monitors (4)	-Stack Effluent Monitor -Reactor Room Tank -Reactor Room Ceiling -Counting Room (Laboratory)	Measure Radioactivity In Stack Effluent, And Measure Gamma Radiation Fields In Reactor and Counting Room (Laboratory)
Continuous Air Monitors (3)	-Stack Effluent Monitors	Measure Radioactivity In Stack Effluent
Micro R Meter (1)	-Control Room	Auxiliary Meter During Operation, And Monthly Radiation Level Surveys
Portable GM Survey Meter (4)	-Reactor Room -Control Room	Personnel Contamination Survey
Pancake GM Survey Meter (1)	-Reactor Room	Beta Gamma Dose Rates
Ion Chamber (1)	-Reactor Room	Beta Gamma Dose Rates
Pressurized Ion Chamber (1)	-Facility Entrance	Emergency Survey Meter
Portable Alpha Detector (1)	-Counting Laboratory →Radio-Chemistry Laboratory	Personnel Contamination Survey
Liquid Scintillation Detector (1)	-Counting Laboratory	Personnel Contamination Survey
Neutron Detector (1)	-Reactor Room	Measure Neutron Dose Rates
HPGe (2)	-Counting Laboratory	Gamma Spectroscopy
Portable NaI	Counting Laboratory	Gamma Spectroscopy
Air Flow Velocity meter (1) →Pressure gauge (1)	-Control Room	Measure Air Flow Rate → Measure pressure differences between facility and outside

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on many different factors including facility design features, operating procedures, training, proper equipment, etc. Training and procedures have been discussed in **UUTR SAR 11.1.2.** This section describes the design features such as shielding, ventilation, containment, and entry control for high radiation areas, protective equipment, personnel exposure, and estimates of annual radiation exposures for specific locations within the facility. Dosimetry records and trends are also included.

11.1.5.1 Shielding

The water around the UUTR TRIGA reactor is the principal design feature for control of radiation exposure during operation. The shielding is based on TRIGA shield designs used successfully at many other similar reactors (General Atomics has developed source terms to serve as a basis for reactor shielding design analysis). UUTR reactor is designed so that radiation from the core area can be extracted via vertical ports for research and educational purposes. The radiation exposure is controlled by restricting access to areas of elevated radiation fields.

11.1.5.2 Personnel Exposure

Regulation 10 CFR20.1502 requires monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits prescribed in 10 CFR20.1201.The regulation also requires monitoring of any individuals entering a high or very high radiation area within which an individual could receive a dose equivalent of 0.1 rem in one hour. **Table 11.1-8** lists results of average values for the last 5-year survey of deep dose equivalent (DDE) occupational exposures at the UUTR. From the table, it is evident that all occupational doses can be maintained far below the regulatory limits given in 10 CFR 20. There have been no instances of any exposures in excess of 10 % of the applicable limits. Monitoring of workers and members of the public for radiation exposure required by the RSC is described in the Program.

11.1.5.3 Authorization for Access

Personnel who enter the control room or the reactor area either holds the authorization for unescorted access, or is under direct supervision of an escort (i.e., escorted individuals can be observed by the escort) who holds authorization for unescorted access.

The UUTR control room and the reactor room are designated as restricted areas for security. Outer doors are locked to prevent unauthorized entry. Access to the reactor facilities is tightly controlled. Only personnel trained in radiation protection and security procedures are issued room keys. All persons must enter these rooms through the inner doors with a key or accompanied by an authorized individual who has been issued a key. Upon entering the restricted area, an individual without key access must complete the sign-in procedure and either wear a dose monitor (electronic portable dosimeter) or be accompanied by an individual wearing a dose monitor at all times. Occasionally, some areas may be posted as radiation areas; and, in rare cases, an area may be posted as a high radiation area. Appropriate restrictions and precautions are observed in all such cases. The sign-in procedure required for access to the reactor lab documents specific information from all persons admitted to the facility. Visitor's cards are maintained as part of the permanent records of the reactor lab.

Table 11.1-8 PLS Typical radiation levels at various UUTR locations at 90 kW operation power of the UUTR

Facility/Location	Typical Dose Equivalent Rate at Contact (mr/hr)	Typical Dose Equivalent Rate at 30 cm (mr/hr)
Reactor Top	0.025	0.02
Fuel Storage Pit	0.012	0.012
Demineralizer Tank	0.022	0.013
Primary Water Pipe	0.013	0.012
Beam Tube (1)	0.013	0.013
Beam Tube (2)	0.015	0.015
Beam Tube (3)	0.016	0.015
Control Room	0.013	0.013
Reactor room Floor	0.014	0.014
AGN reactor Area	0.013	0.013

Personnel, students, visitors and anyone before entering the UUTR area (called the Utah Nuclear Engineering Facilities, UNEF) is required to leave outside the UNEF area the following: cell phone, laptops or similar computer-based devices, bags, books, notebooks, personal belongings, metal objects, sharp objects, drinks and any other object that Senior Reactor Operator or Reactor Supervisor or Director of the facility finds inappropriate to bring into the UNEF areas.

11.1.5.4 Access Control during Operation

When the UUTR TRIGA reactor is operating, the licensed reactor operator (or senior reactor operator) at the controls is responsible for controlling access to the control room and the reactor area.

11.1.5.5 Exposure Records for Access

Personnel who enter the reactor area must have a record of accumulated dose measured by a gamma sensitive individual monitoring device, either a personal dosimeter or a self-reading dosimeter. Normally no less than two individual monitoring devices may be used for a group of visitors, all spending the same amount of time in the reactor area.

Typical personnel monitoring devices used at the UUTR are listed in **Table 11.1-9**.

Table 11.1-9 Typical personnel monitoring devices used at the UUTR

Type	Dose	Radiation Measure	Reading frequency
Portable Electronic dosimeter	Deep Dose Equivalent	Gamma	As needed
TLD	Deep Dose Equivalent Eye Dose Equivalent Shallow Dose Equivalent	Gamma, Beta	Monthly
TLD Finger Ring	Extremity Dose equivalent	Gamma, Beta	Monthly

Before any work with radioactive materials begins, the RHD staff will evaluate the likely or possible doses that an individual may receive. Personnel dosimeters are issued for any of the following conditions:

- before entry into high or very high radiations areas;
- when the deep dose equivalent could exceed 50 mrem in any one month;
- when the shallow dose equivalent could exceed 500 mrem in any one month;
- when the total effective dose equivalent to minors or a declared-pregnant worker could exceed 50 mrem in a year.

Internal dosimetry evaluation is limited to two bioassay methods. The RHD may analyze urine for the presence of ^{3}H using a liquid scintillation counter. Iodine uptake in the thyroid may be analyzed *in vivo* through a thyroid counting program established by RHD. In emergency situations, arrangements would have to be made with RHD, for whole body *in vivo* counting using gamma spectroscopy. Bioassay is required under the conditions determined by the RHD.

An ALARA investigation is performed by the RHD staff if a personnel dosimeter shows a reading that exceeds the stated ALARA investigation level or is unacceptable from an ALARA point of view. This involves documenting the abnormal reading, investigating the cause, and evaluating how it might be mitigated in the future. ALARA investigation is automatically performed when the dose in any reporting period exceeds 1% of the applicable regulatory limit for occupational workers in 10 CFR 20. Additionally, an ALARA investigation is initiated by the SRO and/or Director of the UUTR (and UNEF) for individuals designated as visitors when a measured dose exceeds 10 mrem.

11.1.5.6 Record Keeping

Although the UUTR is potentially exempt from federally required record keeping requirements of 10CFR20.2106(a), certain records are required in confirmation that personnel exposures are less than 10 percent of applicable limits.

11.1.5.7 Records of Prior Occupational Exposures

These records (NRC Form 4) are requested from previous sites by RHD, and then maintained permanently by the RHD. This is not normally done for students since they typically do not have any prior occupational exposure.

11.1.5.8 Records of Occupational Personnel Monitoring

RHD permanently maintains exposure records and each personnel get his/her annual dose report from RHD annually.

11.1.5.9 Records of Doses to Individual Members of the Public

Self-reading dosimeter records are kept in a logbook maintained by the UUTR. Such records are kept permanently by the UUTR. Results of measurements or calculations used to assess accidental releases of radioactive effluents to the environment are to be retained on file permanently at the UUTR.

Bioassay is required for any individual who is likely to receive an annual intake of all nuclides combined of 0.1 of the Annual Limit of Intake (ALI) or more.

11.1.5.10 Ventilation

The UUTR ventilation system is described in details in **UUTR SAR 3**. This section discusses only those ventilation design features that apply to radiation protection. The reactor area ventilation system:

- maintains Argon, Iodine and Nitrogen levels at concentrations in the reactor area consistent with keeping occupational doses below the limits as specified in 10 CFR 20;
- is balanced such that the reactor area is negative to both the laboratories and offices found in the corridor and to the outside atmosphere; and
- has HEPA filters on all ducts originating from irradiation or sample handling facilities.

11.1.5.11 Radioactive Materials

The rules for working with radioisotopes and radiation fields in a safe manner are governed by good judgment and common sense for safe laboratory practices and by a thorough knowledge of the nature of the experiment and the equipment being used. Experiments require careful planning from the first to last step. It is inevitable that certain steps in the experimental procedure are more accident-prone than other steps resulting in spillage and spread of radioactive material. These problems must be anticipated in designing the experiments. Some "excessive" caution is necessary in dealing with radioisotopes. A set of guidelines is tabulated in the **UUTR SAR 11.1.5.12** in order to minimize external radiation exposure, to minimize internal radiation exposure by avoiding ingestion, inhalation, and absorption of radioactive material, and to prevent the spread of contamination in the event of a spill or other accident.

11.1.5.12 General Laboratory Requirements

The laboratory requirements for the safe utilization of radioactive materials are not fundamentally different from those for use of other potentially hazardous materials. A properly designed laboratory gives due consideration to the movement of personnel and materials, the comfort and convenience of personnel, the required utilities, waste disposal, illumination, fire prevention and security, as well as to the minimization of potential hazards and to minimizing the probability of the creation of hazardous working conditions. The potential hazards that are unique to the utilization of radioactive materials in a laboratory are those of external radiation exposure, internal radiation exposure, and the spreading of radioactive contamination to other areas. Thus a radioisotope laboratory must, in addition to the usual safety considerations, provide due consideration for adequate shielding against external radiation, containment of volatile radioactive materials, minimization of contamination, and provision for ease of decontamination. The RHD, in granting authorization to use radioactive materials, will consider the laboratory facilities in relation to the proposed use. The specific laboratory requirements are dependent upon the type of experiment, quantity of radioactive material to be used, and the hazard rating of the radionuclides being used. It is difficult to establish precise laboratory requirements for the wide varieties of utilization that occur at a university. However, radionuclides are classified into hazard groups, and a laboratory classification scheme for purposes of monitoring is predicated on these hazard groupings. At the UUTR the RHD monitors the UNEF radiation levels monthly as a cross check to the monitoring done by the reactor staff.

11.1.5.13 Basic Laboratory Practices

It is essential that all personnel who work with radioactive materials become familiar with the radiation protection program at the University of Utah. The guidelines for proper procedures as well as requirements in handling radioactive materials are summarized in this section. It is essential to point out that these guidelines pertain to all use of radioactive materials. It is expected that the individual will use the utmost care always to ensure safe use of radioisotopes in order to avoid endangering his or her colleagues in the laboratories. All experimenters must therefore:

- Wear personal dosimeters (e.g., film badge, ring badge, or pocket dosimeter)
- Wear protective clothing, such as lab coats, full-length slacks, overshoes, and safety glasses or goggles
- Protect the hands by wearing plastic gloves. (Consider the outer part of the gloves to be contaminated and limit the use of gloves to the immediate experimental area. Do not use gloves in the "inactive" regions of the laboratory, where it is normally allowed to use bare hands [e.g., doorknobs, light switches, fume hood doors, and telephones])
- Prohibit drinking, eating, smoking, and application of cosmetics in a radioactive materials laboratory. (Even if parts of the laboratory are "inactive," it is necessary to depart from the laboratory for drinking, eating, smoking, or application of cosmetics.) The presence of empty food or drink containers will be considered a violation of these regulations, since it will be inferred that consumption occurred on the premises. Food or drink may be transported (expeditiously) through a radioactive materials laboratory only if in a completely closed container
- Prohibit pipetting radioactive solutions using mouth, licking gummed labels, or combing hair in radioactive materials laboratory
- Monitor hands, feet, clothing, and shoes, before leaving the laboratory
- Use suitable monitoring equipment such as portable survey meters in laboratories. (These instruments give exposure rate to radiation in mR/hr, or the observed rate of decay of radioisotopes in counts/min.)

- Survey the laboratory area before commencing an experiment using radioisotopes. (This precaution will ensure that the laboratory is uncontaminated when starting the work. Allocate a smaller portion of the surveyed area for experimental work. In case of an accident, it will be relatively easy to contain the radioactivity and to decontaminate that area.)
- When using volatile materials, always work in fume hoods. (For extremely high activity levels, a glove box is preferred. Inasmuch as feasible, avoid open bench top experiments.)
- Ensure that the fume hood is in satisfactory condition (e.g., strippable or washable paint on exposed area, proper air-flow, and unclogged drains). EHS personnel are required to conduct an annual inspection of each fume hood to ensure proper operational characteristics. (It is preferable to use glove boxes with pressure inside the box slightly less than atmospheric pressure.)
- Use a large porcelain or stainless steel tray lined with absorbent paper and carry out the experiments on top of this tray. (In case of an accident it is an easy matter to decontaminate the tray.)
- Line adjacent porous surfaces with absorbent paper or equivalent material.
- Store and transport radioactive material in closed containers. (Do not transport open containers from one part of the laboratory to another.)
- Label all containers of radioactivity properly with date, radioisotope, quantity of radioactivity, and your name. (Regulations require that each container be clearly marked as to its contents.)
- Use radiation shields if measured radiation levels at the body will result in a dose equivalent in excess of about 20 mrem (0.20 mSv). (Remember that the maximum permitted radiation exposure is 100 mrem/week (1.0mSv/week). In shielding samples, do not forget that the back or sides of the hood may face an adjacent laboratory; it will be necessary to consider exposure to this area as well.)
- Survey the work area. Decontaminate the work area as necessary and clean up all equipment immediately after use. (Check the area with survey equipment to ensure the adequacy of the cleanup. Consult with the RHD if you are not able successfully to clean up the area.)
- Properly post notices to designate areas containing radioactive materials. (Areas where radiation exposure rates would result in a dose equivalent in excess of 5 mrem (0.05 mSv) in one hour should be posted as a Radiation Area and those in excess of 0.1 rem (1 mSv) in one hour as a High Radiation Area. Signs are available for these designations. Remove all signs or markings when the hazard is removed.)
- Rope off radiation areas and contaminated areas to restrict access and post signs to indicate the hazard. (The barriers should not be removed without prior consultation with the RHD.)
- Report all accidents promptly to the RHD on the Radioactive Materials Incident and/or Accident Report (Accidents can occur in the best-planned experiments.)

11.1.5.14 Security, Control and Safe Handling of Radioactive Materials

Stored radioactive materials must be secured from, or controlled in such a manner as to prevent, unauthorized removal from the place of storage. Radioactive materials which are neither in storage nor in restricted area must be tended under the constant surveillance and immediate control of the authorized user.

The basic approach to safe handling of radioactive materials is to focus on avoidance of spills, escapes, or other avenues to contamination by the material being handled. Thus the container must be suitable for the material contained, both from the integrity standpoint and the shielding standpoint. In all types of handling, which usually involves a change in position or location of the radioactive materials, one must never allow his or her attention to wander from the procedure at hand. Moreover, because of the nature of radioactive materials and the attendant dangers of exposure or contamination, extra precautions for safe handling must be adopted. For example, in transporting radioactive materials from one laboratory space to another, even in the same building and on the same floor, the mode of transport must include (at least) double containment, so that there is a second barrier to dispersion should the first barrier fail.

11.1.6 Contamination Control

At the UUTR the potential contamination is controlled by trained personnel following written procedures of how to control radioactive contamination, and by a monitoring program designed to detect contamination in a timely manner. There are no areas within the reactor facility with continuing removable contamination.

The most likely sites of contamination and control of contamination are:

- The reactor room floor area and reactor grid area. All personnel including gloves and shoes must be surveyed before they leave the reactor room.
- The fuel inspection area. Heavy water reflector elements are brought in this room to inspect the elements.
- Sample receiver in the counting lab. A sample from the Pneumatic Irradiator (PI) is transferred to this room using pressurized He gas.
- All personnel required to wear personnel dosimeter and lab coat in the control room and reactor room even when the reactor is not operational.
- Removal of irradiated samples from the reactor pool requires the presence of a proper detector.
- All contamination events are documented.

While working at this or other potentially contaminated sites, workers wear protective gloves, and, if necessary, protective clothing and footwear. Workers are required to perform surveys to assure that no contamination is present on hands, clothing, shoes, etc., before leaving workstations where contamination is likely to occur. If contamination is detected, then a check of the exposed areas of the body and clothing is required, with monitoring control points established for this purpose. Materials, tools, and equipment are monitored for contamination before removal from contaminated areas or from restricted areas likely to be contaminated. Upon leaving the reactor area, hands and feet are monitored for removable contamination. The UUTR staff and visiting researchers are trained on the risks of contamination and on techniques for avoiding, limiting, and controlling contamination. **Table 11.1-10** lists sample locations for routine monitoring of surface contamination control measures. On a biweekly basis, 100 cm^2 swipe tests are analyzed for contamination. Acceptable surface contamination levels for unconditional release are no more than 1,000dpm/100 cm^2 beta-gamma radiation.

Table 11.1-10 Representative Contamination Sampling Locations

Reactor area
Clean sample-preparation fume hood
Floor between reactor room and control room
Floor near entrance to reactor area
Fume hoods
Floor in NW comer of reactor area and fuel storage area
Table in north side wall of reactor room
Stairway between reactor room floor and upper grid plate
Outside reactor area
Floor between control room and computational room
Floor in the counting laboratory
Exit hallway floor
Table for rabbit sample preparation in the counting laboratory
Office area

The basic components of contamination control are:

- Wear fully protective clothing, including gloves, a laboratory coat, wrist guards, full-length slacks, shoes (preferably overshoes) that cover the feet and possibly the ankles, and safety glasses or goggles
- Designate a specific area for work with radioactive materials
- Label all containers and tools properly
- Use trays and absorbent papers
- Prohibit smoking, drinking, eating, or application of cosmetics in the radioactive materials laboratory
- Change gloves frequently so as to avoid contaminating various laboratory articles, fixtures, and surfaces
- Use transfer pipettes and prohibit any mouth-pipetting

- Work with volatile compounds only in operational fume hoods
- Use traps to absorb volatiles (Guidance on disposal of chemical traps should be obtained from the RHD)
- Provide for regular monitoring of clothing, shoes, and the work area
- Avoid all interruptions and distractions once the procedure has been commenced, and especially those which might cause contamination of laboratory articles or furniture (e.g., telephone calls)

At the UNEF the main method of contamination control is embodied in the procedures used to irradiate samples and to handle irradiated samples. All samples are irradiated in a plastic container of some type and lowered down into UUTR pool water containing irradiation tube of the reactor by one of the reactor staff or a qualified experimenter. At the end of the irradiation the sample is pulled up in the irradiation tube and out of the core. Depending upon the expected dose rate from the sample, it may be immediately withdrawn from the core to the bridge level and the sample's exposure rate immediately measured or allowed to cool for a while before being removed from the top of the irradiation tube.

Samples removed from the core are passed down to a hood in the Radiochemistry Laboratory which is on the floor below the reactor operating area via a 4 inch plastic tube that leads to the hood. Documentation on the sample irradiation is then made out by the person removing the sample including the dose rate of the sample. Samples with a dose rate of over 100 mr/hr are not allowed to be transferred to the Radiochemistry Laboratory.

In the Radiochemistry Laboratory a person wearing plastic gloves processes the sample as needed and again monitors the sample or samples after they have been removed from the irradiation container. Only low level samples for Neutron Activation Analysis or irradiated samples in a shield are allowed to be removed from the Radiochemistry Laboratory. The reactor room and radio-chemistry room have three Area Radiation Monitors (near the reactor tank, reactor room ceiling and radio-chemistry lab). If the reactor room has a radiation level higher than 10 mR/hr or somebody attempts to remove a hot sample from the reactor pool, the radiation alarm will be sounded. A radiation monitor is also located at the entrance of the computer room area.

11.1.7 Environmental Monitoring

Environmental monitoring is required to assure compliance with Subpart F of 10CFR Part 20 and with the UUTR Technical Specifications. Installed monitoring systems include area radiation monitors and airborne contamination monitors. The facility has maintained a comprehensive environmental and facility monitoring program for the last

35 years. This program has been very effective in quantifying the fact that the operation of the facility has had an insignificant impact on local environmental radiation levels and radiation exposure in and about the facility. The components of the program are described as follows.

11.1.7.1 Area Radiation Monitors (ARM)

Area radiation monitors are required for reactor operation. Radiation area monitor calibration is accomplished as required by Technical Specifications in accordance with facility procedures. With the exception of Argon-41, there are virtually no pathways for radioactive materials from the UUTR to enter the unrestricted environment during normal facility operations.

11.1.7.2 Airborne Contamination Monitors

The facility has one required air monitoring system in the reactor area. Two additional systems monitor air from the exhaust stack. Airborne contamination monitor calibration is accomplished as required by Technical Specifications in accordance with facility procedures.

The UUTR may impose additional requirements through the Radiation Protection Program.

The average quarterly radiation exposure in mRem in and about the facility is monitored using high sensitivity TLD type dosimeters. The monitoring locations include the unrestricted areas adjacent to the UUTR facility, the closest off-site point of continuous occupancy, and a number of off-site locations. The monitoring program includes at least 20 sampling locations and the dosimeters are changed at the first week of every month. The exposure data are analyzed at least monthly in order to insure compliance with 10 CFR 20.1301. If the exposure rate in an unrestricted area adjacent to the facility is found to be above the level of 3 times higher than background, action should be taken to determine the cause of the exposure in the unrestricted area and to reduce the exposure to the area. Three control dosimeters that are stored in the RHD are provided to measure the background radiation levels. Annually the exposure to the closest off-site point of continuous occupancy is analyzed to insure compliance with the established ALARA criteria found in the facility's Technical Specifications.

11.1.7.3 Contamination Surveys

Contamination monitoring requirements and surveillances addressed in **UUTR SAR 11.1.6** prevent track-out of radioactive contamination from the reactor facilities to the environment. As required by 10CFR20.1501, contamination surveys are conducted to ensure compliance with regulations reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive material, and potential radiological hazards.

11.1.7.4 Radiation Surveys

Quarterly environmental monitoring is conducted, involving measurement of both gamma-ray doses within the facility and exterior to the facility over the course of the quarter using fixed area dosimeters. Gamma-ray exposure-rate data, based on quarterly measurements over the most resent 5-year period is indicated in **Table 11.1-11**. Source terms are related to reactor power levels; therefore maximum radiation levels during operation at 100 kW should not exceed-twice the maximum historical values.

Table 11.1-11 Representative Environmental Exposures for 90kW UUTR

Year	Highest Annual Dose Inside UUTR Area	Highest Annual Dose Outside UUTR Facility
2005	1205B South wall: 237 mrem	Energy & Geosciences Institute (ERG, 1.5 miles from the reactor facility): 48 mrem
2006	1205B: 308 mrem	Background: 47 mrem
2007	1205B: 255 mrem	ERG: 50 mrem
2008	1205B: 307 mrem	ERG: 40 mrem
2009	1205B: 275 mrem	ERG: 47 mrem

11.1.7.5 Monitoring for Conditions Requiring Evacuation

An evacuation alarm is required in the reactor area. Response testing of the alarm is performed in accordance with facility procedures.

11.2 Radioactive Waste Management

The reactor generates very small quantities of radioactive waste. Training for waste management functions are incorporated in operator license training and requalification program.

11.2.1 Radioactive Waste Management Program

The objective of the radioactive waste management program is to ensure that radioactive waste is minimized, and that it is properly handled, stored and disposed of. The Radiological Health Department (RHD) staff is responsible for administering the radioactive waste management program, which also includes any records associated with the program. All records are retained for the life of the facility.

All radioactive waste or waste materials contaminated with radioactive materials may be disposed off, only in accordance with the practices and procedures established by the RHD and enforced by the RHD. The specific procedures, which may change with time, include provision for handling the radioactive wastes as described below. All disposal (and use) are conducted in a manner consistent with environmental monitoring requirements that are met by the RHD.

11.2.2 Radioactive Waste Controls

Radioactive solid waste is generally considered to be any item or substance no longer of use to the UUTR facility, which contains or is suspected of containing radioactivity above background levels. Volume of waste at the UUTR is small, and the nature of the waste items is limited and of known characterization. Consumable supplies such as absorbent materials or protective clothing are declared radioactive waste if radioactivity above background is found to be present. When possible, solid radioactive waste is initially segregated at the point of origin from items that are not considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the future need for the items and materials involved. Solid wastes are either allowed to decay in storage to background, or are transferred to the University of Utah RHD to disposal.

Although argon-41 is released from the UUTR, this release is not considered to be waste in the same sense as liquid and solid wastes; it is an effluent, which is a routine part of the operation of the facility. Typically, this gas simply mixes with reactor room and other facility air and is discharged along with the normal ventilation exhaust.

Liquid wastes are not customarily released from the UUTR.

11.2.3 Release of Radioactive Waste

The UUTR does not have a policy of releasing radioactive waste to the environment as effluent. If contaminated liquids are produced, they are contained locally, added to absorbent, and transferred to a waste barrel in preparation for transfer to RHD. Solid waste is likewise routinely contained on-site.

11.2.4 Routine Waste Collection

The Radiological Health Department routinely collects properly packaged and tagged solid and liquid materials from laboratories on campus including the UUTR Reactor Facility. It is the user's responsibility to properly tag the waste container and accurately estimate the specific radionuclide content in the waste.

11.2.5 Radioactive Waste Storage

The RHD accumulates and stores properly tagged waste materials in an appropriate location prior to transfer to a permanent, licensed disposal site.

11.2.6 Disposal to Sanitary Sewer System

Authorized users only may release to the sanitary sewer small quantities of non-alpha emitting radionuclides not to exceed the limits established in 10 CFR 20.2003 with the approval of the RHD. In order to ensure that total university releases do not exceed the appropriate limits, individual user release limits may not exceed those listed in the **Table 11.2-1**. (Hazardous [chemical] wastes with a radioactive component [officially labeled "Mixed wastes'] are subject to additional regulatory control. Proper disposition of these wastes must be determined by prior consultation with the RHD). **Table 11.2-1** is an example from 10 CFR 20. 2003 and when the facility needs to release any radioactive isotope such as in the **Table 11.2-1**, it is required to coordinate with the University Radiological Health Department.

11.2.7 Radioactive Waste Packaging and Labeling

The RHD collects to dispose only packaged and labeled radioactive waste materials. The basic packaging and labeling requirements are:

Dry waste such as paper, gloves, and plastics, should be placed in the standard Low Specific Activity box (LSA box), which has been lined with a plastic bag. No biological waste whatever is allowed in dry waste. Glass pipettes, broken glass, needles and any other sharp items should be placed in a strong inner package, which is placed in the larger box. Damp material and other waste that will give off vapors or fumes should be contained in small, well-sealed plastic bags or containers before they are placed in the box. (Animal carcasses, blood and tissue, and larger amounts [10 grams] of waste that will putrefy should be frozen and disposed of according to animal waste procedures given below.)

Table 11.2-1 User release limit for several isotopes [10CFR20.2003]

Radionuclide	Activity concentration for exempt material (Bq/g)	Activity limit for exempt consignment(Bq)	Activity limit for exempt consignment (Ci)
H-3	$1.0x10^{6}$	$1.0x10^{9}$	$2.7x10^{-2}$
C-14	$1.0x10^{4}$	$1.0x10^{7}$	$2.7x10^{-4}$
P-32	$1.0x10^{3}$	$1.0x10^{5}$	$2.7x10^{-6}$
S-35	$1.0x10^{5}$	$1.0x10^{8}$	$2.7x10^{-3}$
Ca-45	$1.0x10^{4}$	$1.0x10^{7}$	$2.7x10^{-4}$
Cr-51	$1.0x10^{3}$	$1.0x10^{7}$	$2.7x10^{-4}$
I-125	$1.0x10^{3}$	$1.0x10^{6}$	$2.7x10^{-5}$

Liquid waste that cannot be disposed of via the sewer system must be collected and stored in an appropriate container. Some liquid wastes must be absorbed on floor-dry and disposed of as solid waste. The RHD should be consulted for more specific assistance relating to liquid waste collection.

Liquid scintillation fluid vials. The waste from liquid scintillation counting deserves special mention because of both the volume and the requirement to use non-hazardous and non-toxic media:

- *when vials are reused*: the spent cocktail should be emptied into a liquid waste container and treated as liquid waste. This operation should be conducted in an operating fume hood
- *when vials are disposed of with contents*: Vials should be collected in their original carton or packed carefully in a separate box or bag. Filled scintillation vials must not be mixed with dry waste. The RHD should be consulted for more specific assistance as required
- *when empty vials are disposed of*: empty vials should be placed in a dry waste bag

11.2.8 Records and Labeling

The university license and state regulations require that inventory and control methods cover all aspects of work with radioactive material. Therefore, all packages and containers of radioactive waste must be labeled with a radiation symbol and a description of the contents. The label should indicate the radionuclide, the activity in kBq or MBq, and the name of the authorized user. Containers should be securely closed, both top and bottom, with strong tape or other appropriate devices. Flaps must not be tucked one under the other; this method is not as strong as properly folded and taped flaps. Radioactive label tape is not strong. Both the tops and bottoms of boxes must be secured. Plastic bags should be tied or taped closed. It is especially important to avoid inadvertent collection of radioactive waste by custodians. This goal will be achieved by proper and prominent labeling of radioactive waste containers in the laboratory. As waste is being accumulated in a container, a record of each addition should be made. The record sheet must be summarized and be properly labeled with the waste tag.

11.2.9 Release of Radioactive Waste

As mentioned in **UUTR SAR 11.2.3**, the UUTR does not have a policy of releasing radioactive waste to the environment as effluent. The UUTR release only a sample, which satisfies some specific conditions. Non-routine liquid radioactive waste is generated from decontamination or maintenance activities. Possible gaseous radioactive isotopes that are generated from the UUTR facility are Argon-41 and Nitrogen-16. The amount of Argon-41 satisfies 10 CFR 20 limits. Nitrogen-16 generation is very small and during the reactor operation, the ARM near the reactor tank and reactor room ceiling do not exceed 0.05 mR/hr. All solid waste or experimental samples are surveyed using portable GM counter, ion chamber, 16 channel alpha spectrometer and high purity germanium gamma spectrometer. If a sample satisfies a specific limit, then it can be released or disposed. RPR 54 and 55 specifies the specific procedures for the disposal, release and shipment of radioactive waste. The limits for specific isotopes are outlined in 49 CFR 173.425, and within 49 CFR 173.433 (d) (2) for mixed radionuclides. The exempted amount is specified in 10 CFR 30.71. **Figure 11.2-1** shows one example of release from the UUTR to US Air Force.

Appendix 11.2.A

For **Table 11.2.A-1**,
$\gamma(E)$ is the photon energy in MeV,
$S_A\,(E)$ is the MCNP 5 determined flux density for photon energy E
$\frac{\mu_{H2O}}{\rho}(E)$ is the mass attenuation coefficient for water for photon energy E
$\frac{\mu_{SiO\,2}}{\rho}(E)$ is the mass attenuation coefficient for sand for photon energy E
$\frac{\mu_{en}}{\rho}(E)$ (cm^2/g) is the mass attenuation coefficient in air for photon energy E
$B(E)$ is the photon dose buildup factor for photon energy E
$\phi_a(E)$ is the photon fluence rate at point a
$\dot{D}(E)$ (Gy/hr) is the dose rate in air for photon energy E at the point of interest

Table 11.2.A-1 Calculations results for the 90 kW UUTR

at 90 kW	7.53E+15	n/sec						
a=	183	$\rho(H_2O)$	1	g/cm^3	$x(H_2O)$ cm	92		
Phi		$\rho(SiO_2)$	1.6	g/cm^3	$x(SiO_2)$ cm	61		
γ(E)	$S_A(E)$	$\mu(E)/\rho$ H_2O	$\mu(E)/\rho$ SiO_2	$\mu_{en}(E)/\rho$ Air	B(E)	$\phi_a(E)$	$D_a(E)$	$D_a(E)$
(MeV)	(γ/cm^2-n)	(cm^2/g)	(cm^2/g)	(cm^2/g)		(γ/cm^2-sec)	(Gy/sec)	(Gy/hr)
0.40	8.032E-05	0.106	0.095	0.030	2.6	7.71E-03	1.46E-14	2.30E-12
0.80	1.945E-05	0.079	0.071	0.029	2.3	2.51E-01	9.28E-13	1.26E-10
1.20	9.173E-06	0.065	0.058	0.029	2.3	1.50E+00	8.30E-12	1.13E-09
1.60	5.731E-06	0.056	0.050	0.029	2.3	4.63E+00	3.42E-11	4.64E-09
2.00	3.993E-06	0.049	0.045	0.024	1.8	9.67E+00	7.28E-11	8.04E-09
2.40	1.476E-05	0.045	0.041	0.024	1.8	8.22E+01	7.43E-10	8.20E-08
2.80	1.572E-06	0.041	0.038	0.024	1.8	1.62E+01	1.71E-10	1.88E-08
3.20	1.522E-06	0.038	0.035	0.024	1.7	2.59E+01	3.12E-10	3.16E-08
3.60	1.063E-06	0.036	0.033	0.024	1.7	2.69E+01	3.64E-10	3.69E-08
4.00	7.104E-07	0.034	0.032	0.019	1.6	2.55E+01	3.05E-10	2.90E-08
4.40	7.864E-07	0.032	0.030	0.019	1.6	3.71E+01	4.89E-10	4.64E-08
4.80	7.070E-07	0.031	0.029	0.019	1.6	4.26E+01	6.13E-10	5.81E-08
5.20	3.856E-07	0.030	0.028	0.019	1.6	2.86E+01	4.46E-10	4.23E-08
5.60	2.933E-07	0.029	0.027	0.019	1.6	2.63E+01	4.41E-10	4.18E-08
6.00	2.207E-07	0.028	0.027	0.017	1.5	2.30E+01	3.65E-10	3.18E-08
6.40	2.853E-07	0.027	0.026	0.017	1.5	3.40E+01	5.75E-10	5.00E-08
6.80	1.197E-07	0.026	0.025	0.017	1.5	1.63E+01	2.92E-10	2.54E-08
7.20	7.260E-08	0.025	0.025	0.017	1.5	1.10E+01	2.10E-10	1.83E-08
7.60	2.704E-07	0.025	0.025	0.017	1.5	4.52E+01	9.09E-10	7.90E-08
8.00	1.294E-06	0.024	0.024	0.015	1.4	2.36E+02	4.62E-09	3.77E-07
8.40	2.182E-08	0.024	0.024	0.015	1.3	4.33E+00	8.91E-11	6.73E-09
8.80	8.493E-08	0.023	0.023	0.015	1.3	1.82E+01	3.92E-10	2.96E-08
9.20	8.018E-08	0.023	0.023	0.015	1.3	1.83E+01	4.13E-10	3.12E-08
9.60	3.434E-08	0.023	0.023	0.015	1.3	8.30E+00	1.95E-10	1.48E-08
10.00	1.594E-08	0.022	0.023	0.015	1.3	4.08E+00	9.47E-11	7.16E-09
10.40	1.126E-10	0.022	0.022	0.015	1.3	3.02E-02	7.29E-13	5.51E-11
10.80	3.616E-10	0.022	0.022	0.015	1.3	1.02E-01	2.55E-12	1.93E-10
11.20	7.514E-11	0.021	0.022	0.015	1.3	2.21E-02	5.76E-13	4.35E-11
11.60	3.870E-11	0.021	0.022	0.015	1.3	1.18E-02	3.19E-13	2.41E-11
12.00	2.420E-10	0.021	0.022	0.015	1.3	7.68E-02	2.14E-12	1.62E-10
12.40	2.212E-11	0.021	0.022	0.015	1.3	7.23E-03	2.08E-13	1.57E-11
12.80	0.000E+00	0.020	0.022	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
13.20	3.210E-11	0.020	0.021	0.015	1.3	1.12E-02	3.43E-13	2.60E-11
13.60	0.000E+00	0.020	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
14.00	1.806E-11	0.020	0.021	0.015	1.3	6.67E-03	2.17E-13	1.64E-11
14.40	2.271E-11	0.020	0.021	0.015	1.3	8.62E-03	2.88E-13	2.18E-11
14.80	0.000E+00	0.020	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
15.20	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
15.60	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
16.00	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
16.40	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
16.80	1.809E-11	0.019	0.021	0.015	1.3	7.69E-03	3.00E-13	2.27E-11
17.20	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
17.60	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
18.00	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
18.40	0.000E+00	0.019	0.021	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
18.80	0.000E+00	0.018	0.020	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
19.20	0.000E+00	0.018	0.020	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
19.60	0.000E+00	0.018	0.020	0.015	1.3	0.00E+00	0.00E+00	0.00E+00
20.00	0.000E+00	0.018	0.020	0.013	1.3	0.00E+00	0.00E+00	0.00E+00
Total	1.430E-04					7.46E+02	1.22E-08	1.07E-06
							t(h/yr)	13.45
							Gy/yr	8.65E-04
							Sv/yr	6.49E-04

12. CONDUCT OF OPERATIONS

This chapter describes and discusses the Conduct of Operations at the University of Utah TRIGA Reactor (UUTR). The Conduct of Operations involves the administrative aspects of facility operations, the facility emergency plan, the security plan, the reactor operator selection and requalification plan, and environmental reports.

12.1 Organization

The formal licensee of the UUTR is the President, The University of Utah. However, the UNEP Director is responsible for licensing and reporting information to the NRC. The President is informed of license issues via the normal university reporting channel, originating from the UNEP Director through the Vice Provost for Research to the President.

12.1.1 Structure

The administrative organization of the UUTR is summarized in **Fig. 12.1-1.** The lines of responsibility and the lines of communication (consultation) are to be strictly followed **(UUTR TS 6.1.2).** Failure to follow this organizational structure is a reportable occurrence as defined in **UUTR TS 6.1.1**. The organizational chart is essentially a set of three columns and three rows. The columns are defined by common functions: operations, reactor safety, and radiation safety. The rows are defined by levels of responsibility: administration, management, and staff. Administration is responsible for the overall operation of the facility including coordination of operations, reactor and radiation safety sections, budget authorization, and interfacing with regulatory organizations. Management is responsible for assisting administration when necessary and supervising day-to-day operations. Staff is responsible for executing day to day operations tasks.

The UUTR maintains a small staff of Reactor Operators (RO) and Senior Reactor Operators (SRO). There are no specialized staff solely responsible for implementing the radiation safety function, but each staff member has been trained in radiation safety and is responsible for the implementation of a radiation protection program. The direct line of responsibility for operation of the reactor is as follows: RO's, SRO's, Reactor Supervisor (RS), and manager of the facilities, Vice President for Research and President (U of U).

The Department of Radiological Health and the Reactor Safety Committee are separate support entities, which provide independent radiation surveys and audits, respectively, for the UUTR. Both of these committees are directly responsible to the

University of Utah Vice President for Research and not to management or staff of the UUTR; thus the independent characteristic of radiation safety reviews and audits is guaranteed.

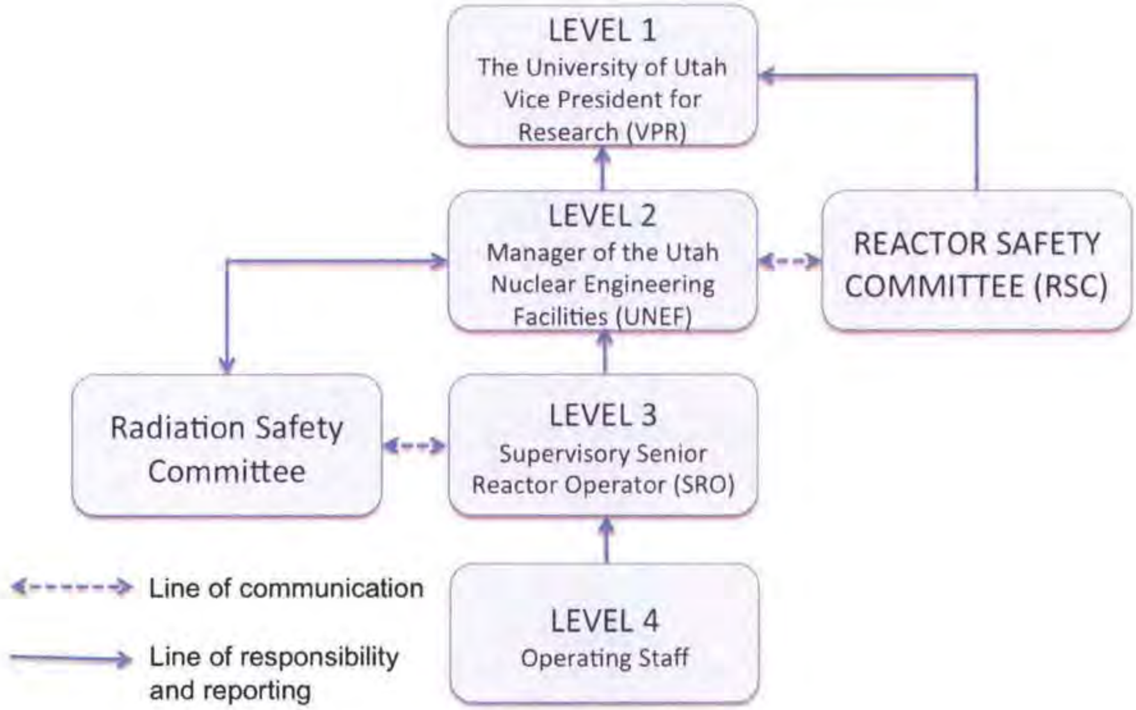


Figure 12.1-1 University of Utah Administrative Organization for Nuclear Reactor Operations [Clarified - December 2009; in compliance with ANSI/ANS 15.1-2007]

As indicated in **Fig. 12.1-1**, the Reactor Safety Committee shall report to Level 1. Radiation safety personnel shall report to Level 2. Additional description of levels follows:

a. *Level 1*: Individual responsible for the reactor facility's licenses, i.e., the Associate Vice President for Research in the Office of Vice President for Research; The Vice President for Research will assign which of the Associate Vice Presidents for Research will be the responsible Level 1 individual.
b. *Level 2*: Individual responsible for reactor facility operation, i.e., the Utah Nuclear Engineering Facility (UNEF) Manager shall be the Director of the Utah Nuclear Engineering Program (UNEP).
c. *Level 3*: Individual responsible for day-to-day operation or shift shall be the reactor supervisor (RS). This person shall be a senior reactor operator (SRO).
d. *Level 4*: Operating staff shall be senior reactor operators, reactor operators, and trainees.

12.1.2 Responsibility

- President, the University of Utah – chief executive officer of the university;
- Vice President for Research – research programs associated with the university are administered through the research office. The UNEP is administered through the research office;
- Radiation Safety Committee – The RSC is responsible for radiological safety on the entire university campus and controls the movement and use of all radioisotopes and radiation producing machines on campus. The RS in full consultation with the RSO enforces the regulations of this committee within the reactor area.
- Reactor Safety Committee – The reactor safety committee is responsible for reviewing proposed experiments, modifications and procedures, and changes thereto, with respect to the TS, CFR, SAR, and ANSI Standards, the Experiment Authorization and Modification Authorization procedures established and general common sense. The RSC is also responsible for auditing operation, operational records, any operating abnormalities, and the expected performance of facility equipment effecting nuclear safety. The RSC makes recommendations to UUTR through the appropriate channels based on their reviews of these items. In addition, the RSC reviews any reported safety-limit violations and assists in preparation of required reports, as necessary. In order to fulfill these responsibilities, the RSC shall meet at least semi-annually and on call of the Chairman. Documentation of the activities of the RSC shall be maintained. The documentation shall include the names and qualifications of the members, the agenda and approved minutes of all RSC meetings, RSC actions, and copies of all correspondence and reports to or from the RSC. RSC documentation is transmitted annually to the University Archives.
- The UNEF Manager – the director is accountable for ensuring all licensing requirements, including implementation and enforcement, in accordance with the NRC codes and guides. This is a level 2 position. The UNEF Manager with the most senior a Senior Reactor Operator (SRO) is responsible for liaison with the NRC regarding technical and emergency matters and for enforcement of all regulations. The UNEF Manager under all conditions shall be the UNEP Director. In the absence of the UNEP Director, The University of Utah shall appoint a person to serve as the Interim UNEF Manager. Therefore, Director has final authority and ultimate responsibility for the reactor facility and, within the limitations established by the facility license, makes final policy decisions on all phases of reactor operation, appoints personnel to all positions that report to the Director, and is advised in matters concerning radiation safety by the Radiation Safety Committee and

matters concerning safety by the Reactor Safety Committee. This individual holds a Senior Reactor Operator's License;

- Reactor Supervisor – the reactor supervisor is responsible to the UNEP director for directing the activities of reactor operators and for the day-to-day operation and maintenance of the reactor. The reactor supervisor shall be certified as a senior reactor operator. This is a level 3 position;
- The reactor operator reports – to the reactor supervisor and is primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operations and maintenance of reactor related equipment. A reactor operator shall be certified as either a senior reactor operator or a reactor operator. This is a level 4 position; and
- The RSO – is an experienced health physicist and is responsible for the radiological health and safety of the University community. The RSO works with the RS to ensure the radiological safety of operations within the reactor facility and is responsible for the transfer of radioisotopes outside the UNEF. The RSO is available for consultation in the event of any emergency. The RSO may perform only those duties specified for the RSO or a health physicist.
- Activities at the facility – will always be under direct control of an NRC licensed SRO designated by the RS. An SRO must be on call (but not necessarily on site) when the reactor is not secure. The SRO will be responsible to the RS for the overall facility operation including safe operation and maintenance of the facility and its associated equipment. Temporary changes to procedures that do not change their original intent may be made by the SRO (**UUTR TS 6.4**). The SRO may perform any duties not specified here and duties specified for RO's. RO's licensed by the NRC can legally operate the TRIGA reactor. RO's may directly supervise trainees when manipulating core reactivity. All RO's and trainees are under the direction of an SRO.

12.1.3 Staffing

All reactivity changes shall be made by, or in the presence and under the direction of the licensed operator of record at the time the reactivity changes are made.

1. When the reactor is not secured, the minimum staff shall consist of **(UUTR TS 6.1.3)**:
    a. A licensed reactor operator in the control room, e.g., Reactor Operator (RO) (may be the SRO or RS).
    b. A designated Senior Reactor Operator (SRO) on call but not necessarily on site.

    c. Another person present at the facility complex who is able to carry out prescribed written instructions.
2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:
    a. management personnel,
    b. radiation safety personnel,
    c. other operations personnel;
3. Events requiring the presence at the facility of the senior reactor operator are:
    a. initial startup and approach to power,
    b. all fuel or control-rod relocations within the reactor core region,
    c. relocation of any experiment with reactivity worth greater than one dollar;
    d. recovery from unplanned or unscheduled shutdown or significant power reduction.

12.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard "Selection and Training of Personnel for Research Reactors," ANSI/ANS-15.4-2007, Sections 4 through 7.

The Reactor Supervisor shall be responsible for the facility's Requalification Training Program and Operator Training Program. RO's and SRO's must have a degree in Nuclear Engineering or related field, or be working towards the completion of a degree.

The UUTR maintains a RO/SRO qualification and requalification program to ensure the competence of its operators. The training program covers basic nuclear phenomena, health physics, and reactor operations and involves both lectures and hands-on experience. The requalification program presents the same material. All licensees must take part in an annual requalification exam. The Reactor Supervisor is responsible for conducting the training and requalification programs. Topics covered of this training include:

- Organization;
- Security and access control;
- Radiation occupational safety; and
- Emergency procedures and responses

12.1.4.1 General Training for the UNEF Personnel

Training and orientation for all personnel within the UNEF who work with radioactive materials is specifically discussed in **UUTR SAR 11**.

12.1.4.2 Initial Reactor Operator Training

Either a reactor operator or a senior reactor operator trainee shall be signed up for two-semester training courses. For initial training for either a reactor operator or a senior reactor operator, the following documents shall be studied and knowledge demonstrated:

Training session 1: Basis nuclear reactor physics, regulations, radiation safety and protection, emergency procedures and 10 CFR codes;

Training session 2: reactor operating procedures, instrumentations, Technical specifications, physical securities, license and experiments.

12.1.5 Radiation Safety

This is covered in **UUTR SAR 11**.

12.2 Review and Audit Activities

An independent oversight committee referred to as the Reactor Safety Committee (RSC) conducts the review and audits on a biennial basis. Appointments to the Reactor Safety Committee are initiated by the RSC itself and are annually reviewed and approved by the President of the University.

The Reactor Safety Committee has been assigned approval authority for review and audit activities.

The Reactor Safety Committee (RSC) shall function to provide an independent review and audit of the facility's activities including **(UUTR TS 6.2)**:

1. reactor operations
2. radiological safety
3. general safety
4. testing and experiments
5. licensing and reports

6. quality assurance

12.2.1 Composition and Qualifications

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members shall collectively represent a broad spectrum of expertise in the appropriate reactor technology.

The members of the committee shall include the Reactor Supervisor and faculty and staff members designated to serve on the committee. The University's Radiation Safety Officer shall be an ex officio member of the RSC.

Members and alternates shall be appointed by and report to Level 1 management **(UUTR TS 6.2.1)**. Individuals may be either from within or outside the operating organization (University qualified and approved alternates may serve in the absence of regular members).

12.2.2 Charter and Rules

The RSC shall conduct its review and audit functions in accordance with a written charter. This charter shall include provisions for **(UUTR TS 6.2.2)**:

- meeting frequency;
- voting rules;
- quorums; method of submission and content of presentations to the committee;
- use of subcommittees; and
- review, approval and dissemination of meeting minutes.

12.2.3 Review Function

The responsibilities of the RSC or designed subcommittee(s) thereof shall include, but is not limited to, the following **(UUTR TS 6.2.3)**:

1. review and approval of all new experiments utilizing the reactor facility
2. review and approval of all proposed changes to the facility license by amendment, and to the Technical Specifications

3. review of the operation and operational records of the facility
4. review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that effect nuclear safety
5. review and approval of all determinations of whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or on unreviewed safety questions as defined by 10 CFR 50.59
6. review of reportable occurrences and the reports filed with the Commissions for said occurrences
7. review and approval of all standard operating procedures and changes thereto
8. biennial review of all standard procedures, the facility emergency plan, and the facility security plan.

A written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 and the review and audit group members in a timely manner after the review has been completed.

12.2.4 Audit Function

The Reactor Safety Committee (RSC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. Minutes, findings or reports of the RSC shall be presented to Level 1 and Level 2 management within ninety (90) days of completion. (**UUTR TS 6.2**).

According to **UUTR TS 6.2.4**), the RSC or a Subcommittee thereof shall audit reactor operations at least annually. The annual audit shall include at least the following:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions;
2. The retraining and requalification program for the operating staff;
3. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety; and
4. The Emergency Response Plan and implementing procedures.

12.3 Procedures

Written procedures shall be prepared and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed by the Reactor Supervisor, and approved by the UNEP Director. The procedures are reviewed by the RSC annually to ensure that they are appropriate. The procedures shall be adequate to assure the safe operation of the reactor, but will not preclude the use of independent judgment and action should the situation require.

The utmost care shall be taken to keep thorough, accurate records of UUTR operations, including detailed documentation of standard operating procedures for all routine activities and the methods of review for proposed and existing procedures. These documents, known collectively as the "UNEP Surveillance and Procedures Log" apply only to activities performed in the UNEF and UNEP at the U of U. Facility users are directly accountable for any activities they may undertake within the facility which are not outlined in these documents. This section lists the forms and procedures included in the "UNEP Surveillance and Procedures Log" as well as the approval process for new standard procedures. It also describes UUTR policy in the event that a deviation from standard procedure is necessary during reactor operation.

12.3.1 Reactor Operations

According to **UUTR TS 6.4** written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

1. Startup, operation and shutdown of the reactor;
2. Fuel loading, unloading, and movement within the reactor;
3. Maintenance of major components of systems that could have an effect on reactor safety;
4. Surveillance checks, calibrations, and inspections required by the technical specifications or those that have an effect on reactor safety;
5. Radiation protection;
6. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;
7. Shipping of radioactive materials;
8. Implementation of the Emergency Response Plan.

Substantive changes to the above procedures shall be made only after review by the RSC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the UNEP director and documented by the UNEP director within 120 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved prior to implementation by the RSO and documented by the RSO (Radiation Safety Officer) within 120 days of implementation. Temporary deviations from the procedures may be made by the responsible senior reactor operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the UNEP director.

12.3.2 Health Physics

The following activities will typically require written procedures:

- testing and calibration of area radiation monitors, facility air monitors, laboratory radiation detection systems, and portable radiation monitoring instrumentation;
- working with radioactive materials;
- facility radiation monitoring program including routine and special surveys, personnel monitoring, monitoring and handling of radioactive waste, and sampling and analysis of solid and liquid waste and gaseous effluents released from the facility;
- monitoring radioactivity in the environment surrounding the facility; personnel orientation and training;
- receipt of radioactive materials at the facility, and unrestricted release of materials and items from the facility; and
- transportation of radioactive materials.

12.4 Required Actions

This is covered in the UUTR Technical Specifications.

12.5 Reports

This is covered in the UUTR Technical Specifications.

12.6 Records

This is covered in the UUTR Technical Specifications.

12.7 Emergency Planning

The UUTR Emergency Response Plan contains detailed information concerning the UUTR response to emergency situations. The UUTR Emergency Response Plan is written to be in accordance with ANSI/ANS 15.16, *Emergency Planning for Research Reactors.*

The Emergency Response Plan is designed to provide response capabilities to emergency situations involving the UUTR. Detailed implementing procedures are referenced in this plan. This approach provides the UUTR emergency staff the flexibility to cope with a wide range of emergency situations without requiring frequent revisions to the plan. The objective of the Emergency Plan is to provide a basis for action, to identify personnel and material resources, and to designate areas of responsibility for coping with any emergency at the UNEF that could impact public health and safety. This plan identifies both on-site and off-site support organizations that are required to be contacted for specialized assistance depending upon the nature of the emergency. Implementation of the plan on a day-to-day basis is the responsibility of the Reactor Supervisor and UNEP Director who serve as the Emergency Coordinators in an emergency. Provisions for reviewing, modifying, and approving the emergency implementation procedures are defined in the plan to assure that adequate measures to protect the staff and general public are in effect at all times.

12.8 Security Planning

The UUTR physical security plan ensures the protection of special nuclear materials on the premises. All such material associated with the UUTR NRC License R-126 is classified in the category of low strategic significance. The plan contains detailed information concerning the UUTR security measures. The plan provides the UUTR with criteria and actions for protecting the facility from such acts as intrusion, theft, civil disorder and bomb threats. Primary responsibility for the plan and facility security rests with the UNEP director. Implementation of the plan on a day-to-day basis during hours of operations is also the responsibility of the UNEP director.

12.9 Operator Training and Requalification

This reactor operator training and requalification program is designed to satisfy the requirements of the NRC's rules contained 10 CFR 55. It also generally complies with the requalification program in ANSI/ANS 15.4, Selection and Training of Personnel for Research Reactors.

12.9.1 Responsibility

The responsibility for this program rests with the reactor supervisor. This responsibility shall cover the following items:

- Selecting knowledgeable individuals to give classroom lectures and to supervise retraining operations;
- Certifying to the UNEP director that each individual has successfully completed the requalification program; and
- Granting of exemptions to the requalification program as provided for in this plan.

12.9.2 Schedule

The requalification program shall be conducted on a cycle not to exceed two years. Upon conclusion, it will promptly repeated.

12.9.3 Content

The requalification program shall consist of preplanned lectures, written examinations, an annual operating examination, and routine reactor operations.

12.9.4 Annual Lectures

Lectures shall be given annually (not to exceed 15 months) which cover the following:

- Emergency response Plan; and
- Physical Security Plan.

12.9.5 Biennial Lectures

Lectures shall be given biennially (not to exceed 30 months) which cover the following:

- Facility design, characteristics, instrument control and safety system
- Reactor principles
- Operating procedures, Technical Specifications, and administrative procedures; and
- Radiation protection.

12.9.6 Written Exam

Written exam shall be given covering the lecture material. The individual giving the lecture on a particular subject shall formulate, administer, and grade the written examination on that subject. Any licensed individual preparing and grading an examination is exempt from taking that examination. All written examinations will be proctored by the individual administering the exam, or by either appointed representative, but shall not be an individual taking the exam.

A grade equal to or greater than 70% will constitute a passing grade. Failure to achieve a passing grade will result in an accelerated retraining program in the subject area failed. This accelerated retraining program will be left to the discretion of the reactor supervisor.

12.9.7 Quarterly Operating Requirements

To maintain current license, each calendar quarter, a reactor operator shall operate the reactor for a minimum of four hours and perform a supervised reactor startup, including a core excess measurement and increase to power. For senior reactor operators, direct supervision of these operations may be considered equivalent to actual performance.

If a licensed reactor operator or senior reactor operator has not met these quarterly operating requirements, then before resumption of licensed duties, the operator or senior operator shall:

Satisfactorily perform the annual operating exam; and

Operate the reactor for a minimum of six hours under the direction of a senior reactor operator.

12.9.8 Annual Operating Exam

To maintain a current license, each calendar year, each reactor operator or senior reactor operator shall successfully complete an annual operating exam to be administered by the reactor supervisor. Successful completion is left to the discretion of the reactor supervisor. The annual operating exam for the reactor supervisor shall be administered by a senior reactor operator. The annual operating exam shall include the following:

- Reactor startup;
- Core excess measurement;
- Increase power to 90 kW;
- Change in power level > 10% in manual;
- Record a reactor log book;
- Respond to any annunciators;
- Shut down the reactor;
- State responses, or respond to, all of the following situations:
  1) loss of coolant;
  2) loss of electrical power;
  3) loss or male function of a nuclear instrumentation;
  4) rod drop
  5) inability to drive control rods;
  6) fuel cladding failure; and
  7) response to the high radiation alarm .

12.9.9 Medical Certification

All reactor operators and senior reactor operators shall undergo a medical examination by a physician biennially, not to exceed 30 months. The physician should be conversant with the medical requirements of this program. Following completion of the medical examination, an NRC form 396 shall be signed by the UNEP Director.

12.9.10 Records

Required documents and records pertaining to the requalification program for a reactor operator or senior reactor operator shall be maintained until the respective operator's license is renewed or surrendered **(UUTR TS 6.8.2).**

12.10 Startup Plan

This is not applicable.

12.11 Environmental Report

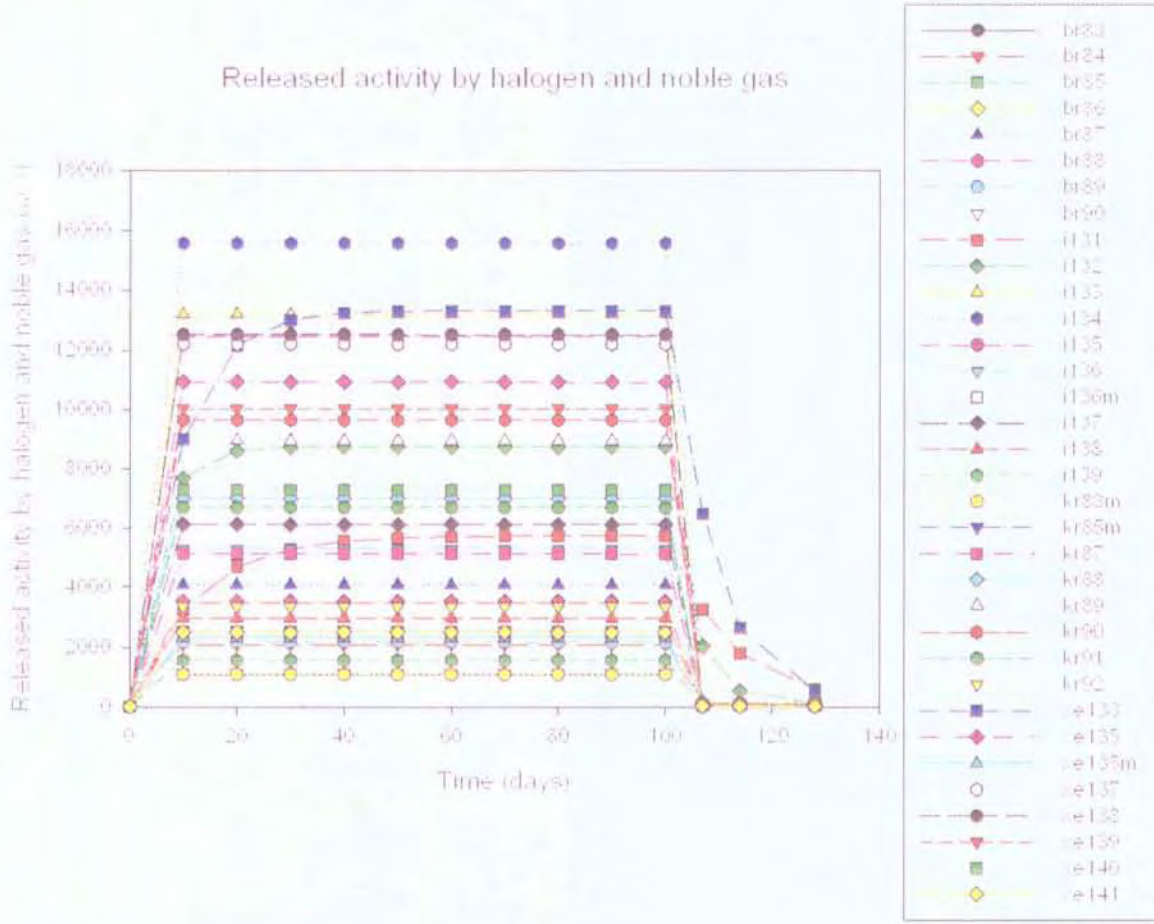
This shall be submitted as a separate document.

Approved UNEP Forms

Form No.	Title	RSC Approval
Monthly		
UNEP-001R10	TRIGA Prestart Checklist (3 sheets)	04/02/04
UNEP -020R12	Monthly Inspection Checksheet (3 sheets)	04/02/04
Semi-Annual		
UNEP -003R6	Semi-Annual Control Rod Calibrations (2 sheets)	03/29/00
UNEP -011R2	Calibration of Temperature Monitoring Channels (4 sheets)	03/12/97
UNEP -012R3	Semi-Annual Thermal Power Calibration (2 sheets)	03/18/98
UNEP -015R3	Emergency Kit Check	09/17/03
UNEP -038R2	Semi-Annual Ventilation System Maintenance and Verification (3 sheets)	03/26/02
Annual		
UNEP -023R4	Annual Maintenance and Calibration of the Area Radiation Monitors (ARMs) and Continuous Air Monitor (CAM) (3 sheets)	12/17/97
Biennial		
UNEP -002R3	Biennial Control Rod Inspection/Control Rod Movement or Repair (2 sheets)	05/23/02
UNEP -004R1	Biennial Fuel Rod Inspection (2 sheets)	12/17/97
UNEP -009R2	Tank Inspection Procedure (2 sheets)	12/17/97
UNEP -010R1	Heavy Water Reflector Element Inspection Procedure (2 sheets)	12/17/97
Unscheduled		
UNEP -005R4	Core Change and Critical Fuel Loading (2 sheets)	03/29/00
UNEP -006R3	Procedure for Changing Filters in the TRIGA Pool Water Refrigeration/Purification System (4 sheets)	12/17/97
UNEP -007	Procedure to Change Central Irradiator	05/25/88
UNEP -008R4	Procedure for Adding Water to the Reactor Tank (2 sheets)	12/17/97
UNEP -013R4	Adjustment of Power Monitoring Channels (3 sheets)	09/18/02
UNEP -016	Agreement for Off-Hours Access	07/27/88
UNEP -017R1	Familiarization Checksheet	05/13/98

Form No.	Title	RSC Approval
UNEP -018	Fuel Element Inventory Sheet	05/25/88
UNEP -021R21	UNEP Emergency Call List ~ revised 06/10/04 NA	
UNEP -022R2	Maintenance Log (2 sheets)	09/21/94
UNEP -024	Replacement of Ion-exchange Resin (2 sheets)	11/30/88
UNEP -025	Requalification Program Progress Checklist (2 sheets)	NA
UNEP -027R4	TRIGA Reactor Irradiation Request and Performance (2 sheets)	03/26/96
UNEP -028R1	Experimental Facility Reactivity Worth Determination Procedure	03/12/97
UNEP -030	Cf-252 Irradiation Request and Performance	03/21/90
UNEP -031	Safeguard Event Log	03/21/90
UNEP -032	Liquid Effluent Discharge Authorization (2 sheets)	03/19/92
UNEP -033	UNEP Security Alarms	05/29/91
UNEP -035R1	Audit and Review Program Checklist (2 Sheets)	06/09/93
UNEP -036	Calibration of pH Meter	06/08/95
UNEP -037	Radiological Emergency Classification Checklist	12/14/94

13. ACCIDENT ANALYSES



This chapter provides information and analysis to demonstrate that the health and safety of the public and workers are protected in the event of equipment malfunctions or other abnormalities in reactor behavior. The analysis demonstrates that facility design features, limiting safety system settings, and, limiting conditions for operation ensure that no credible accident could lead to unacceptable radiological, consequences to people or the environment.

13.1 Introduction

In about 1980, the U.S. Nuclear Regulatory Commission requested an independent and fresh overview analysis of credible accidents for TRIGA and TRIGA fueled reactors. Such an analysis was considered desirable since safety and licensing concepts had changed over the years. The study resulted in NUREG/CR-2387, *Credible Accident Analysis for TRIGA and TRIGA's-Fueled Reactors* [Reed College SAR]. The information developed by the TRIGA experience base, plus appropriate information from NUREG/CR-2387, serve as a basis for some of the information presented in this chapter. The reactor physics and thermal-hydraulic conditions in the UUTR at power level of 100 kW are established in **UUTR SAR 4**.

The fuel temperature is the limit on operation of the UUTR. This limit stems from the outgassing of hydrogen from UZrH fuel and the subsequent stress produced in the fuel element cladding material. Calculations performed by General Atomics and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as approximately 530 °C (803.15 °K) for low-hydride-type ($UZrH_{1.0}$), aluminum-clad elements, [Reed College SAR] and 1,175 °C (1,448.15 °K) for high-hydride-type ($UZrH_{1.60}$), stainless-steel-clad elements.[Reed College SAR]. Cladding damage in the high-hydride-type, stainless-steel fuel is caused by a pressure buildup in the element as a result of the evolution of hydrogen produced by dehydriding of the fuel with increasing temperature. The pressure internal to the fuel element reaches the point where the cladding fails. Cladding damage in the low-hydridetype, aluminum-clad fuel is caused by a phase change in the fuel matrix that occurs at about 803.15 °K (530 °C). The phase change causes the fuel to swell that causes the cladding to fail. For a core containing only aluminum clad fuel or containing both aluminum and stainless steel clad fuel, a fuel temperature limit of 500 °C (773.15 °K) is determined by the cladding damage threshold temperature of the low-hydride-type, aluminum-cladding fuel elements. For a future core with only stainless steel clad fuel, fuel temperature limits of 1,100 °C (1,373.15 °K) (with clad < 500 °C) and 930 °C (1,203.15 °K) (with clad > 500 °C) for UZrH with a H/Zr ratio less than 1.70 have been set to preclude the loss of clad integrity, [Reed College SAR].

Ten credible accidents for research reactors were identified in NUREG-1537 [NUREG-1537, Guidelines for-Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content, *Report NUREG-1537 Part 1, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1996.*] as follows:

1. The maximum hypothetical accident (MHA);
2. Insertion of excess reactivity;

3. Loss of coolant accident (LOCA);
4. Loss of coolant flow;
5. Loss of pool water;
6. Mishandling or malfunction of fuel;
7. Experiment malfunction;
8. Loss of normal electrical power;
9. External events; and
10. Mishandling or malfunction of equipment.

In normal operation a TRIGA type non-power reactor does not in and of itself constitute a threat to the health and safety of the facility staff or the general public. In this Chapter the analyses of postulated accidents that have been categorized into one of the above nine groups are presented: some categories do not contain accidents that appeared applicable or credible for the UUTR; some categories contain an analysis of more than one accident even though one is usually limiting in terms of impact. Any accident having significant radiological consequences was analyzed. For those events that do not result in the release of radioactive materials from the fuel, only a qualitative evaluation of the event is presented. Events leading to the release of radioactive material from a fuel element were analyzed to the point where it was possible to reach the conclusion that a particular event was, or was not, the limiting event in that accident category.

The MHA for TRIGA reactors is the cladding failure of a single irradiated fuel element in air with no radioactive decay of the contained fission products taking place prior to the release.

13.2 Accident Initiating Events and Scenarios, Accident Analysis, and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (MHA)

The failure of the encapsulation of one fuel element, in air, resulting in the release of gaseous fission products to the atmosphere is considered to be the Maximum Hypothetical Accident (MHA) for TRIGA and therefore the UUTR. Administrative controls prevent removal of fuel from the reactor pool during fuel handling, but one could postulate fuel failure in air during fuel transfer circumstances. Potential consequences of fuel failure in air including inhalation by the public are considered as the MHA scenario.

13.2.1.1 Accident Initiating Events and Scenarios

A single fuel element could fail at any time during normal reactor operation or while the reactor is in a shutdown condition, due to a manufacturing defect, corrosion, or handling damage. This type of accident is very infrequent, based on many years of operating experience with TRIGA fuel, and such a failure would not normally incorporate all of the necessary operating assumptions required to obtain a worst-case fuel-failure scenario.

For the UUTR, the MHA has been defined as the cladding rupture of one highly irradiated fuel element with no radioactive decay followed by the instantaneous release of the noble gas and halogen fission products into the air. For this accident, the fuel cladding type makes no difference as either element would contain the same amount of uranium-235 and, hence, the same inventory of fission products. The following assumptions and approximations were applied to this analysis:

1. Calculation of long-lived radionuclides inventory in fuel is based on continuous operation for 35 years prior to fuel failure at the power level of 100 kW with an average of 70 hours of annual reactor operation time (70 hours is a conservative assumption derived from the 35 years long operating history of the UUTR).

2. For short-lived radionuclides, the calculations of radionuclide inventory in fuel are based on the operation at the full thermal power of 100 kW for 100 days prior to fuel failure. (The operation time should be long enough for radionuclides inventory to reach its saturation; numerical modeling and simulations as presented in proceeding sections show that 100 days of a continuous full power operation of the UUTR is long enough time to model the radionuclides inventory at the saturation point.)

3. Radionuclide inventory in one "worst-case" fuel element is based on a 78-fuel elements core. The hottest channel is in the stainless steel fuel element, which contains [redacted] grams of uranium per element; value of 2.0 for the ratio of the maximum pin power in the core to the average pin power is selected as a conservative value (details of the MCNP5 modeling are given in **UUTR SAR 4.5**). Thus, it assumed that the worst case is defined as follows: a fuel element is exposed to generation of a thermal power of 2x(0.799/78) = 20.49 W (0.1097 MW/TU); thus for the 100 days of full-power operation, it follows: 2(100/78) = 2.56kW (13.71 MW/TU).

4. The fraction of noble gases and iodine contained within the fuel that is actually released is 1.0×10^{-4} selected as a conservative value based on the NUREG 2387; this value may be compared to 1.5×10^{-5} that was measured at General Atomics ["Fuel Elemerits for Pulsed TRIGA Research Reactors," *Nuclear Technology 28, 31-56 (1976), Simnad, M.T., F. C. Foushee, and G.B. West*] and used in SARs for other reactor facilities [NUREG-1390, Safety Evaluation Report Relating to the Renewal, of the Operating License for the TRIGA Training and Research Reactor at the University of Arizona, *Report NUREG-1390, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1990.*].

5. The fractional release of particulates (radionuclides other than the noble gases and iodine) is 1.0×10^{-6}, a very conservative estimate used in NUREG-2387 [NUREG/CR-2387 (PNL-4028), Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors, *Report Pacific Northwest Laboratory, Richland Washington, 1982., Hawley, S. C., and R.L. Kathren*].

13.2.1.2 Accident Analysis and Determination of Consequences

Radionuclide Inventory Buildup and Decay. A mass of uranium-235 yields thermal power P (kW) due to thermal neutron induced fission. The fission rate (fissions per second) is equal to the product of thermal power (kW) and a factor k = 3.12 x 10^{13} (fissions per second per kW). If a fission product is produced with a yield Y and decays with the rate constant λ, it is easily shown that the equilibrium activity A_∞ of the fission product, which exists when the rate of production by fission is equal to the rate of loss due to its decay, is given by $A_\infty = kPY$. The power must be small enough or the uranium mass large enough that the depletion of the uranium-235 becomes negligible. Starting at time t=0, the buildup of activity is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t}\right)$$

For the times much greater than the half-life of the radionuclide, $A \approx A_\infty$, and for the times much less than the half-life, $A(t) = A_\infty \lambda t$. If the fission process ceases at time t_1, the specific activity at later time t is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t_1}\right)e^{-\lambda(t-t_1)}$$

For example, fission product ^{131}I has a half-life of 8.04 days (λ =0.00359 h^{-1}) and a chain (cumulative) fission product yield of about 0.031. At thermal power of 1 kW, the equilibrium activity is about A_∞ = 9.67 X 10^{11} Bq (26.1Ci). After only four hours of operation, though, the activity is only about 0.37 Ci. For equilibrium operation at 100 kW, distributed over 78 fuel elements, the average activity per element would be 26.1×100/78 = 33.46 Ci per fuel element. The worst-case fuel element would contain twice this activity. With a release fraction of 1.0 X 10^{-4}, the activity available for release would be about (30.12)(2)(l.0 x10^{-4}) =6.024x10^{-3} Ci. This type of calculation is performed by the TRITON [M. D. DeHart, TRITON: a two-dimensional transport and depletion module for characterization of spent nuclear fuel, ORNL/TM-2005/39, Version 6, Vol. I, Sect. T1, Jan 2009] and KENO6 [D. F. Hollenbach, L. M. Petrie, S. Goluoglu, N. F. Landers and M. E. Dunn, KENO-VI: a general quadratic version of the keno program, ORNL/TM-2005/39, Version 6, Vol. II, Sect. F17, Jan 2009] codes in SCALE6 code package for hundreds of fission products and for arbitrary times and power levels of operation as well as arbitrary times of decay after reactor operation.

Data from TRITON and KENO6 in SCALE6 Calculations. SCALE6-TRITON code is a two-dimensional transport and depletion module for characterization of spent nuclear fuel, and in the following depletion calculation, TRITON is coupled with SCALE6-KENO6, a three dimensional Monte Carlo code to obtain the isotope concentration and activities for various UUTR's operational times and powers. The fuel pin that has the highest thermal power is the stainless steel pin; therefore the 2D/3D stainless steel fuel unit cell depletion models are developed for TRITON and KENO6 respectively. One million neutrons were used in KENO6 simulations.

For simulations assuming 100 days of a continuous full power operation, the red dashed line in **Fig. 13.2-1** shows the total activity in the stainless steel fuel pin at various times, while the black solid line displays the potential released total reactivity if cladding crack accident occurs at each time point (represented by red triangles). According to this simulation the radioactive nuclide inventories will reach the saturation point at around 60 days. The long-lived radionuclides concentration will continue to build up slowly, but this effect will not have an impact on the released activity, because the released activity is dominated by the halogen and noble gases whose half-life is very short (halogen and noble gas half-lives are listed in **Table 13.2-5**). Because the gas release factor of 10^{-4} is 100 times larger than the release factor for other nuclides (10^{-6}), the radioactive gas has a great contribution to the total released activity. **Figure 13.2-2** shows the released activity of halogen and noble gases (only 34 nuclides whose released activities are larger than 1000 μCi are shown in the figure). Their inventory reaches the saturation at around 60 days as well. Therefore, a 100 days continuous full power operation is conservative assumption to simulate the maximum hypothetic accident.

Table 13.2-1 shows halogen and noble gas activities to potentially be released in a maximum hypothetical accident at the UUTR for short-lived radionuclides simulation obtained using SCALE6 while **Table 13.2-2** shows particulate activities to potentially be released in a maximum hypothetical accident at the UUTR for short-lived radionuclides simulation obtained using **SCALE6** both modeled assuming the UUTR operation at 12.35 MW/TU thermal power for 100 consecutive days. **Table 13.2-3** shows halogen and noble gas activities to potentially be released in a maximum hypothetical accident at the UUTR for long-lived radionuclides simulation obtained using SCALE6 while **Table 13.2-4** shows particulate activities to potentially be released in a maximum hypothetical accident at the UUTR for long-lived radionuclides simulation obtained using SCALE6 over 40 years of operation assuming 0.5014MW/TU thermal power. **Tables 13.2-1** shows halogen and noble gas activities to potentially be released in a maximum hypothetical accident for short-lived radionuclides obtained using SCALE6, **Table 13.2-2** shows particulate activities to potentially be released in a maximum hypothetical accident at the UUTR for short-lived radionuclides using SCALE6, **Tables 13.2-3** shows halogen and noble gas activities to potentially be released in a maximum hypothetical accident for long-lived radionuclides obtained using SCALE6 and **Table 13.2-4** shows particulate activities to potentially be released in a maximum hypothetical accident for long-lived radionuclides obtained using SCALE6; these values are expressed in μCi, per nuclide, immediately after 7, 14, and 28 days of the reactor shutdown. In these tables, the activities per element are multiplied by the release fractions, thus yielding maximum activities to potentially be released in a maximum hypothetical accident.

SCALE6 was asked to output the top 500 nuclides that have the highest activity, but only the nuclides with a released activity larger than 0.1 μCi are presented in **Table 13.2-1** to **Table 13.2-4**.

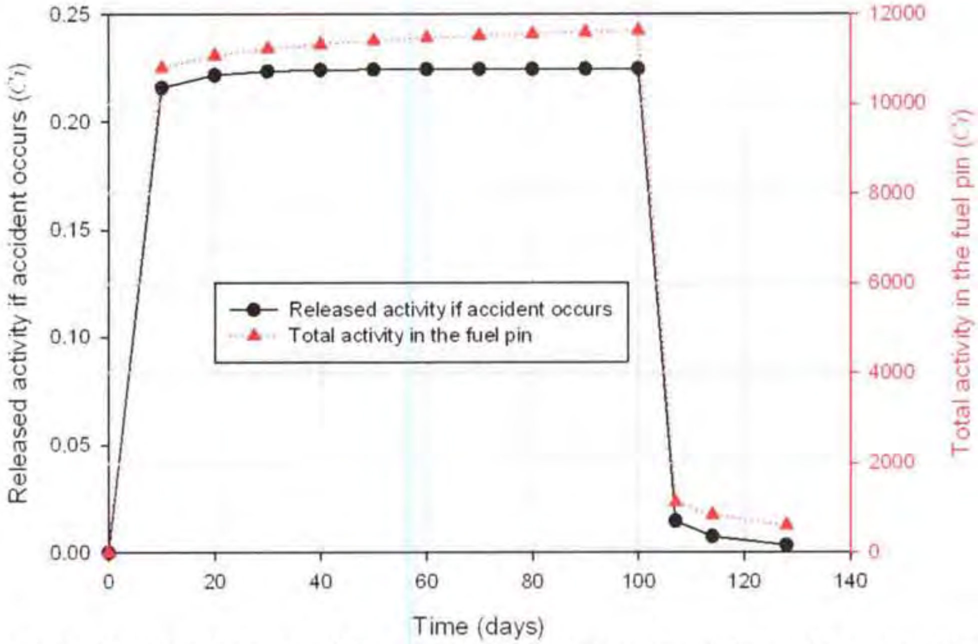


Figure 13.2-1 Total activity in the fuel pin and potential released activity in the UUTR obtained using SCALE6 (UUTR at 100kW for 100 days of continuous operation)

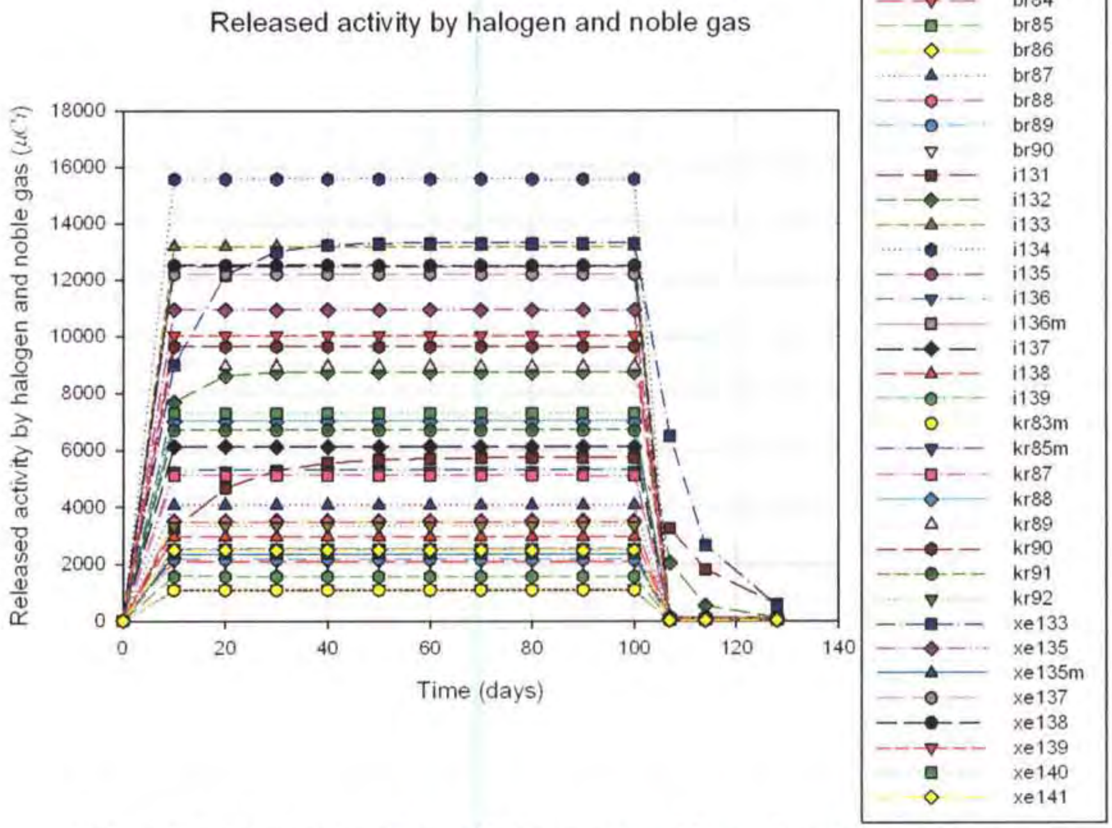


Figure 13.2-2 Released activity of halogen and noble gas from the UUTR obtained using SCALE6 (UUTR at 100kW for 100 days of continuous operation)

Table 13.2-1 Halogen and noble gas activities potentially released in maximum hypothetical accident at the UUTR for short-lived radionuclides obtained using SCALE6

Element	Potentially released activity (μCi) at time after reactor shutdown - in days			
	0 days	7 days	14 days	28 days
br83	1072.6	8.7	8.7	8.7
br84	2098.9	17.0	17.0	17.0
br84m	33.2	0.3	0.3	0.3
br85	2510.1	20.3	20.3	20.3
br86	3510.0	28.4	28.4	28.4
br87	4081.9	33.1	33.1	33.1
br88	3483.8	28.2	28.2	28.2
br89	2162.4	17.5	17.5	17.5
br90	1125.3	9.1	9.1	9.1
br91	448.4	3.6	3.6	3.6
br92	54.4	0.4	0.4	0.4
br93	6.4	0.1	0.1	0.1
he6	6.1	0.0	0.0	0.0
i131	5739.7	3250.2	1799.8	571.0
i132	8728.2	2040.9	515.1	93.1
i132m	18.5	0.1	0.1	0.1
i133	13183.9	156.6	106.9	107.0
i133m	852.6	6.9	6.9	6.9
i134	15561.3	126.1	126.1	126.1
i134m	725.5	5.9	5.9	5.9
i135	12473.7	101.1	101.1	101.1
i136	5293.0	42.9	42.9	42.9
i136m	2487.6	20.2	20.2	20.2
i137	6107.9	49.5	49.5	49.5
i138	2966.1	24.0	24.0	24.0
i139	1557.8	12.6	12.6	12.6
i140	304.3	2.5	2.5	2.5
i141	82.1	0.7	0.7	0.7
i142	11.8	0.1	0.1	0.1
kr83m	1072.4	8.7	8.7	8.7
kr85	10.2	10.2	10.2	10.2
kr85m	2517.5	20.4	20.4	20.4
kr87	5119.2	41.5	41.5	41.5
kr88	6991.9	56.6	56.6	56.6
kr89	8935.7	72.4	72.4	72.4
kr90	9629.1	78.0	78.0	78.0
kr91	6677.9	54.1	54.1	54.1
kr92	3338.0	27.0	27.0	27.0
kr93	974.9	7.9	7.9	7.9
kr94	176.5	1.4	1.4	1.4
kr95	15.0	0.1	0.1	0.1

kr96	75.1	0.6	0.6	0.6
kr98	3.2	0.0	0.0	0.0
xe131m	61.9	56.9	46.6	26.6
xe133	13305.4	6466.7	2646.5	507.4
xe133m	386.5	71.7	10.7	3.2
xe134m	66.0	0.5	0.5	0.5
xe135	10918.7	105.1	105.0	93.9
xe135m	2313.8	18.7	18.7	18.7
xe137	12195.2	98.8	98.8	98.8
xe138	12505.5	101.3	101.3	101.3
xe139	10017.8	81.2	81.2	81.2
xe140	7274.1	58.9	58.9	58.9
xe141	2483.9	20.1	20.1	20.1
xe142	878.6	7.1	7.1	7.1
xe143	53.7	0.4	0.4	0.4
xe143m	53.7	0.4	0.4	0.4
xe144	12.7	0.1	0.1	0.1
Subtotal	214752.0	13474.5	6556.9	2728.5

Table 13.2-2 Particulate activities potentially released in maximum hypothetical accident at the UUTR for short-lived radionuclides obtained using SCALE6

Element	**Potentially released activity (μCi) at time after reactor shutdown - in days**			
	0 days	**7 days**	**14 days**	**28 days**
ag109m	0.7	0.0	0.0	0.0
ag111	0.3	0.2	0.1	0.0
ag111m	0.3	0.0	0.0	0.0
ag112	0.3	0.0	0.0	0.0
ag113	0.3	0.0	0.0	0.0
ag113m	0.1	0.0	0.0	0.0
ag114	0.2	0.0	0.0	0.0
ag115	0.2	0.0	0.0	0.0
ag115m	0.1	0.0	0.0	0.0
ag116	0.3	0.0	0.0	0.0
ag117	0.1	0.0	0.0	0.0
ag117m	0.1	0.0	0.0	0.0
ag118	0.1	0.0	0.0	0.0
ag118m	0.1	0.0	0.0	0.0
ag119	0.2	0.0	0.0	0.0
ag120	0.1	0.0	0.0	0.0
ag121	0.1	0.0	0.0	0.0
as77	0.2	0.0	0.0	0.0
as78	0.4	0.0	0.0	0.0
as79	0.9	0.0	0.0	0.0
as80	2.5	0.0	0.0	0.0

as81	3.9	0.0	0.0	0.0
as82	5.2	0.0	0.0	0.0
as82m	0.5	0.0	0.0	0.0
as83	6.8	0.1	0.1	0.1
as84	2.5	0.0	0.0	0.0
as84m	2.0	0.0	0.0	0.0
as85	5.2	0.0	0.0	0.0
as86	10.1	0.1	0.1	0.1
as87	1.0	0.0	0.0	0.0
as88	2.5	0.0	0.0	0.0
ba137m	0.7	0.7	0.7	0.7
ba139	127.5	1.0	1.0	1.0
ba140	123.0	84.4	58.0	27.6
ba141	115.6	0.9	0.9	0.9
ba142	114.2	0.9	0.9	0.9
ba143	110.1	0.9	0.9	0.9
ba144	87.3	0.7	0.7	0.7
ba145	38.4	0.3	0.3	0.3
ba146	18.4	0.1	0.1	0.1
ba147	4.9	0.0	0.0	0.0
ba148	0.5	0.0	0.0	0.0
cd115	0.2	0.0	0.0	0.0
cd117	0.2	0.0	0.0	0.0
cd118	0.2	0.0	0.0	0.0
cd119	0.2	0.0	0.0	0.0
cd119m	0.1	0.0	0.0	0.0
cd120	0.2	0.0	0.0	0.0
cd121	0.1	0.0	0.0	0.0
cd121m	0.1	0.0	0.0	0.0
cd122	0.3	0.0	0.0	0.0
cd123	0.2	0.0	0.0	0.0
cd124	0.3	0.0	0.0	0.0
cd125	0.1	0.0	0.0	0.0
cd126	0.2	0.0	0.0	0.0
cd127	0.2	0.0	0.0	0.0
cd128	0.1	0.0	0.0	0.0
cd130	1.7	0.0	0.0	0.0
cd131	0.3	0.0	0.0	0.0
ce141	102.0	88.5	76.4	56.9
ce143	117.9	4.4	1.1	1.0
ce144	23.6	23.2	22.8	22.1
ce145	78.1	0.6	0.6	0.6
ce146	59.5	0.5	0.5	0.5
ce147	37.5	0.3	0.3	0.3
ce148	31.7	0.3	0.3	0.3
ce149	15.5	0.1	0.1	0.1

ce150	8.0	0.1	0.1	0.1
ce151	2.0	0.0	0.0	0.0
ce152	0.4	0.0	0.0	0.0
cs136	0.1	0.1	0.1	0.0
cs136m	0.1	0.0	0.0	0.0
cs137	0.8	0.8	0.8	0.8
cs138	133.5	1.1	1.1	1.1
cs138m	4.4	0.0	0.0	0.0
cs139	126.1	1.0	1.0	1.0
cs140	113.8	0.9	0.9	0.9
cs141	82.7	0.7	0.7	0.7
cs142	54.0	0.4	0.4	0.4
cs143	29.0	0.2	0.2	0.2
cs144	8.5	0.1	0.1	0.1
cs145	1.5	0.0	0.0	0.0
cs146	0.2	0.0	0.0	0.0
eu156	0.4	0.3	0.2	0.1
eu157	0.1	0.0	0.0	0.0
eu158	0.1	0.0	0.0	0.0
ga76	0.1	0.0	0.0	0.0
ga77	0.1	0.0	0.0	0.0
ga78	0.3	0.0	0.0	0.0
ga79	0.4	0.0	0.0	0.0
ga80	0.2	0.0	0.0	0.0
ga81	0.2	0.0	0.0	0.0
ga82	0.1	0.0	0.0	0.0
ga84	0.2	0.0	0.0	0.0
ge77m	0.1	0.0	0.0	0.0
ge78	0.4	0.0	0.0	0.0
ge79	0.6	0.0	0.0	0.0
ge79m	0.3	0.0	0.0	0.0
ge80	2.3	0.0	0.0	0.0
ge81	2.6	0.0	0.0	0.0
ge81m	0.1	0.0	0.0	0.0
ge82	2.6	0.0	0.0	0.0
ge83	1.0	0.0	0.0	0.0
ge84	0.5	0.0	0.0	0.0
ge86	12.4	0.1	0.1	0.1
in115m	0.3	0.0	0.0	0.0
in117	0.2	0.0	0.0	0.0
in117m	0.2	0.0	0.0	0.0
in118	0.2	0.0	0.0	0.0
in119	0.1	0.0	0.0	0.0
in119m	0.2	0.0	0.0	0.0
in120	0.2	0.0	0.0	0.0
in121	0.1	0.0	0.0	0.0

in121m	0.1	0.0	0.0	0.0
in122	0.3	0.0	0.0	0.0
in123	0.2	0.0	0.0	0.0
in123m	0.1	0.0	0.0	0.0
in124	0.3	0.0	0.0	0.0
in125	0.2	0.0	0.0	0.0
in125m	0.1	0.0	0.0	0.0
in126m	0.2	0.0	0.0	0.0
in127	0.9	0.0	0.0	0.0
in127m	0.2	0.0	0.0	0.0
in128	0.3	0.0	0.0	0.0
in128m	0.3	0.0	0.0	0.0
in129	0.6	0.0	0.0	0.0
in129m	0.5	0.0	0.0	0.0
in130	1.8	0.0	0.0	0.0
in130m	0.1	0.0	0.0	0.0
in131	0.4	0.0	0.0	0.0
in131m	0.1	0.0	0.0	0.0
in132	0.1	0.0	0.0	0.0
la140	123.1	96.0	66.6	31.7
la141	116.0	0.9	0.9	0.9
la142	116.1	0.9	0.9	0.9
la143	117.6	1.0	1.0	1.0
la144	108.6	0.9	0.9	0.9
la145	76.4	0.6	0.6	0.6
la146	33.2	0.3	0.3	0.3
la146m	14.8	0.1	0.1	0.1
la147	17.7	0.1	0.1	0.1
la148	7.2	0.1	0.1	0.1
la149	1.6	0.0	0.0	0.0
la150	0.2	0.0	0.0	0.0
mo99	121.7	21.6	4.5	1.1
mo101	103.1	0.8	0.8	0.8
mo102	84.9	0.7	0.7	0.7
mo103	58.9	0.5	0.5	0.5
mo104	35.8	0.3	0.3	0.3
mo105	18.5	0.1	0.1	0.1
mo106	7.6	0.1	0.1	0.1
mo107	2.5	0.0	0.0	0.0
mo108	0.6	0.0	0.0	0.0
mo109	0.3	0.0	0.0	0.0
mo110	0.1	0.0	0.0	0.0
nb100	117.4	1.0	1.0	1.0
nb100m	6.3	0.1	0.1	0.1
nb101	99.4	0.8	0.8	0.8
nb102	56.3	0.5	0.5	0.5

nb102m	15.7	0.1	0.1	0.1
nb103	38.2	0.3	0.3	0.3
nb104	7.5	0.1	0.1	0.1
nb104m	5.7	0.0	0.0	0.0
nb105	5.1	0.0	0.0	0.0
nb106	0.3	0.0	0.0	0.0
nb107	0.1	0.0	0.0	0.0
nb95	54.1	57.8	60.2	62.1
nb95m	0.9	0.9	0.9	0.7
nb97	120.0	1.1	1.0	1.0
nb97m	113.5	1.0	0.9	0.9
nb98	113.3	0.9	0.9	0.9
nb98m	0.8	0.0	0.0	0.0
nb99	72.4	0.6	0.6	0.6
nb99m	48.4	0.4	0.4	0.4
nd147	44.6	28.8	18.6	7.9
nd149	21.5	0.2	0.2	0.2
nd151	8.3	0.1	0.1	0.1
nd152	5.3	0.0	0.0	0.0
nd153	3.0	0.0	0.0	0.0
nd154	1.3	0.0	0.0	0.0
nd155	0.4	0.0	0.0	0.0
nd156	0.1	0.0	0.0	0.0
np239	211.2	28.6	5.2	1.8
pd109	0.7	0.0	0.0	0.0
pd111	0.4	0.0	0.0	0.0
pd112	0.3	0.0	0.0	0.0
pd113	0.3	0.0	0.0	0.0
pd114	0.2	0.0	0.0	0.0
pd115	0.2	0.0	0.0	0.0
pd116	0.2	0.0	0.0	0.0
pd117	0.2	0.0	0.0	0.0
pd118	0.1	0.0	0.0	0.0
pd120	0.1	0.0	0.0	0.0
pm147	2.6	2.8	2.9	3.0
pm148	0.1	0.0	0.0	0.0
pm149	21.3	2.6	0.4	0.2
pm151	8.3	0.2	0.1	0.1
pm152	5.3	0.0	0.0	0.0
pm153	3.2	0.0	0.0	0.0
pm154	1.4	0.0	0.0	0.0
pm154m	0.1	0.0	0.0	0.0
pm155	0.6	0.0	0.0	0.0
pm156	0.2	0.0	0.0	0.0
pm157	0.1	0.0	0.0	0.0
pr143	117.4	91.3	64.4	32.0

pr144	23.6	23.2	22.8	22.1
pr144m	0.3	0.3	0.3	0.3
pr145	78.0	0.6	0.6	0.6
pr146	59.6	0.5	0.5	0.5
pr147	44.7	0.4	0.4	0.4
pr148	32.5	0.3	0.3	0.3
pr148m	0.8	0.0	0.0	0.0
pr149	21.4	0.2	0.2	0.2
pr150	12.4	0.1	0.1	0.1
pr151	6.8	0.1	0.1	0.1
pr152	2.5	0.0	0.0	0.0
pr153	0.8	0.0	0.0	0.0
pr154	0.1	0.0	0.0	0.0
rb100	0.7	0.0	0.0	0.0
rb88	70.4	0.6	0.6	0.6
rb89	93.4	0.8	0.8	0.8
rb90	88.1	0.7	0.7	0.7
rb90m	25.6	0.2	0.2	0.2
rb91	110.9	0.9	0.9	0.9
rb92	95.7	0.8	0.8	0.8
rb93	70.5	0.6	0.6	0.6
rb94	32.8	0.3	0.3	0.3
rb95	15.4	0.1	0.1	0.1
rb96	4.1	0.0	0.0	0.0
rb97	0.8	0.0	0.0	0.0
rb98	0.1	0.0	0.0	0.0
rh103m	50.1	44.3	39.2	30.7
rh104	0.1	0.0	0.0	0.0
rh105	18.5	0.9	0.2	0.2
rh105m	5.5	0.0	0.0	0.0
rh106	1.4	1.4	1.4	1.3
rh107	3.1	0.0	0.0	0.0
rh108	1.2	0.0	0.0	0.0
rh109	0.7	0.0	0.0	0.0
rh109m	0.3	0.0	0.0	0.0
rh110m	0.5	0.0	0.0	0.0
rh111	0.4	0.0	0.0	0.0
rh112	0.3	0.0	0.0	0.0
rh113	0.3	0.0	0.0	0.0
rh114	0.1	0.0	0.0	0.0
rh115	0.1	0.0	0.0	0.0
ru103	50.2	44.4	39.3	30.8
ru105	19.5	0.2	0.2	0.2
ru106	1.4	1.4	1.4	1.3
ru107	3.1	0.0	0.0	0.0
ru108	1.2	0.0	0.0	0.0

ru109	0.6	0.0	0.0	0.0
ru110	0.5	0.0	0.0	0.0
ru111	0.3	0.0	0.0	0.0
ru112	0.2	0.0	0.0	0.0
ru113	0.1	0.0	0.0	0.0
sb127	3.1	0.9	0.3	0.0
sb128	0.4	0.0	0.0	0.0
sb128m	6.7	0.1	0.1	0.1
sb129	10.9	0.1	0.1	0.1
sb130	15.7	0.1	0.1	0.1
sb130m	19.1	0.2	0.2	0.2
sb131	50.8	0.4	0.4	0.4
sb132	25.9	0.2	0.2	0.2
sb132m	29.1	0.2	0.2	0.2
sb133	47.7	0.4	0.4	0.4
sb134	7.5	0.1	0.1	0.1
sb134m	7.2	0.1	0.1	0.1
sb135	2.9	0.0	0.0	0.0
sb136	0.2	0.0	0.0	0.0
sb137	1.5	0.0	0.0	0.0
se79m	0.9	0.0	0.0	0.0
se81	4.0	0.0	0.0	0.0
se81m	0.3	0.0	0.0	0.0
se83	4.9	0.0	0.0	0.0
se83m	5.5	0.0	0.0	0.0
se84	20.6	0.2	0.2	0.2
se85	20.4	0.2	0.2	0.2
se86	26.0	0.2	0.2	0.2
se87	15.6	0.1	0.1	0.1
se88	7.3	0.1	0.1	0.1
se89	1.0	0.0	0.0	0.0
se90	0.3	0.0	0.0	0.0
sm153	3.2	0.3	0.0	0.0
sm155	0.6	0.0	0.0	0.0
sm156	0.3	0.0	0.0	0.0
sm157	0.1	0.0	0.0	0.0
sm158	0.1	0.0	0.0	0.0
sn121	0.2	0.0	0.0	0.0
sn123m	0.3	0.0	0.0	0.0
sn125	0.2	0.1	0.1	0.0
sn125m	0.5	0.0	0.0	0.0
sn127	1.9	0.0	0.0	0.0
sn127m	1.1	0.0	0.0	0.0
sn128	6.6	0.1	0.1	0.1
sn128m	3.2	0.0	0.0	0.0
sn129	5.6	0.0	0.0	0.0

sn129m	4.0	0.0	0.0	0.0
sn130	12.1	0.1	0.1	0.1
sn130m	11.4	0.1	0.1	0.1
sn131	9.3	0.1	0.1	0.1
sn131m	8.8	0.1	0.1	0.1
sn132	12.0	0.1	0.1	0.1
sn133	2.8	0.0	0.0	0.0
sn134	0.4	0.0	0.0	0.0
sr100	0.8	0.0	0.0	0.0
sr101	0.1	0.0	0.0	0.0
sr89	70.0	63.7	57.9	47.9
sr90	0.8	0.8	0.8	0.8
sr91	115.8	0.9	0.9	0.9
sr92	117.9	1.0	1.0	1.0
sr93	123.9	1.0	1.0	1.0
sr94	120.3	1.0	1.0	1.0
sr95	104.7	0.8	0.8	0.8
sr96	74.6	0.6	0.6	0.6
sr97	34.8	0.3	0.3	0.3
sr98	16.2	0.1	0.1	0.1
sr99	2.7	0.0	0.0	0.0
tc100	0.1	0.0	0.0	0.0
tc101	103.1	0.8	0.8	0.8
tc102	85.1	0.7	0.7	0.7
tc102m	0.2	0.0	0.0	0.0
tc103	60.6	0.5	0.5	0.5
tc104	37.6	0.3	0.3	0.3
tc105	19.5	0.2	0.2	0.2
tc106	8.2	0.1	0.1	0.1
tc107	3.0	0.0	0.0	0.0
tc108	1.1	0.0	0.0	0.0
tc109	0.6	0.0	0.0	0.0
tc110	0.3	0.0	0.0	0.0
tc111	0.1	0.0	0.0	0.0
tc99m	107.1	20.8	4.3	1.0
te127	2.8	1.1	0.5	0.3
te127m	0.2	0.2	0.2	0.2
te129	10.0	1.0	0.9	0.7
te129m	1.7	1.5	1.3	1.0
te131	51.1	0.5	0.4	0.4
te131m	8.1	0.2	0.1	0.1
te132	85.4	19.8	5.0	0.9
te133	74.2	0.6	0.6	0.6
te133m	67.4	0.5	0.5	0.5
te134	138.6	1.1	1.1	1.1
te135	66.4	0.5	0.5	0.5

te136	26.8	0.2	0.2	0.2
te137	9.0	0.1	0.1	0.1
te138	1.3	0.0	0.0	0.0
te139	0.1	0.0	0.0	0.0
te140	0.3	0.0	0.0	0.0
u237	0.5	0.2	0.1	0.0
u239	211.9	1.7	1.7	1.7
y100	12.2	0.1	0.1	0.1
y101	5.7	0.0	0.0	0.0
y102	5.3	0.0	0.0	0.0
y103	0.1	0.0	0.0	0.0
y90	0.7	0.8	0.8	0.8
y91	81.1	75.5	69.5	59.1
y91m	67.3	0.5	0.5	0.5
y92	119.3	1.0	1.0	1.0
y93	125.7	1.0	1.0	1.0
y93m	43.9	0.4	0.4	0.4
y94	128.1	1.0	1.0	1.0
y95	126.6	1.0	1.0	1.0
y96	79.1	0.6	0.6	0.6
y96m	40.1	0.3	0.3	0.3
y97	60.8	0.5	0.5	0.5
y97m	36.4	0.3	0.3	0.3
y98	38.2	0.3	0.3	0.3
y98m	22.0	0.2	0.2	0.2
y99	41.6	0.3	0.3	0.3
zn77	0.1	0.0	0.0	0.0
zn78	0.1	0.0	0.0	0.0
zr100	111.0	0.9	0.9	0.9
zr101	61.3	0.5	0.5	0.5
zr102	40.5	0.3	0.3	0.3
zr103	10.1	0.1	0.1	0.1
zr104	1.7	0.0	0.0	0.0
zr105	2.3	0.0	0.0	0.0
zr95	85.4	79.3	73.6	63.4
zr97	119.7	1.1	1.0	1.0
zr98	111.0	0.9	0.9	0.9
zr99	112.1	0.9	0.9	0.9
subtotal	9483.8	979.2	767.1	576.0

Table 13.2-3 Halogen and noble gas activities potentially released in maximum hypothetical accident at the UUTR for long-lived radionuclides obtained using SCALE6

Element	Potentially released activity (μCi) at time after reactor shutdown - in days			
	0 days	**7 days**	**14 days**	**28 days**
br83	8.6	0.1	0.1	0.1
br84	16.8	0.2	0.2	0.2
br85	20.1	0.2	0.2	0.2
br86	28.1	0.3	0.3	0.3
br87	32.6	0.3	0.3	0.3
br88	27.8	0.3	0.3	0.3
br89	17.3	0.2	0.2	0.2
br90	9.0	0.1	0.1	0.1
br91	3.6	0.0	0.0	0.0
br92	0.4	0.0	0.0	0.0
br93	0.1	0.0	0.0	0.0
i131	45.7	25.9	14.4	4.6
i132	68.6	16.4	4.3	0.9
i132m	0.1	0.0	0.0	0.0
i133	106.4	1.5	1.1	1.1
i133m	6.8	0.1	0.1	0.1
i134	124.4	1.3	1.3	1.3
i134m	5.8	0.1	0.1	0.1
i135	99.7	1.0	1.0	1.0
i136	42.3	0.4	0.4	0.4
i136m	19.9	0.2	0.2	0.2
i137	48.8	0.5	0.5	0.5
i138	23.7	0.2	0.2	0.2
i139	12.5	0.1	0.1	0.1
i140	2.4	0.0	0.0	0.0
i141	0.7	0.0	0.0	0.0
i142	0.1	0.0	0.0	0.0
kr83m	8.6	0.1	0.1	0.1
kr85	4.3	4.3	4.3	4.3
kr85m	20.1	0.2	0.2	0.2
kr87	40.9	0.4	0.4	0.4
kr88	55.9	0.6	0.6	0.6
kr89	71.4	0.7	0.7	0.7
kr90	77.0	0.8	0.8	0.8
kr91	53.4	0.5	0.5	0.5
kr92	26.7	0.3	0.3	0.3
kr93	7.8	0.1	0.1	0.1
kr94	1.4	0.0	0.0	0.0
kr95	0.1	0.0	0.0	0.0
kr96	0.6	0.0	0.0	0.0
xe131m	0.5	0.5	0.4	0.2

xe133	102.7	50.4	20.8	4.2
xe133m	3.1	0.6	0.1	0.0
xe134m	0.5	0.0	0.0	0.0
xe135	103.7	1.1	1.1	1.0
xe135m	18.5	0.2	0.2	0.2
xe137	97.5	1.0	1.0	1.0
xe138	100.0	1.0	1.0	1.0
xe139	80.1	0.8	0.8	0.8
xe140	58.1	0.6	0.6	0.6
xe141	19.8	0.2	0.2	0.2
xe142	7.0	0.1	0.1	0.1
xe143	0.4	0.0	0.0	0.0
xe143m	0.4	0.0	0.0	0.0
xe144	0.1	0.0	0.0	0.0
Subtotal	1733.0	113.8	59.5	29.5

Derived Quantities. The raw data shown in **Tables 13.2-1 to 13.2-4** are the activities potentially released from a single worst-case fuel element that has experienced a cladding failure. This activity may itself be compared to the annual limit of intake (ALI) to gauge the potential risk to an individual worker. By dividing the activity by a conservative value of 4.59×10^8 cm^3 free volume[1] of the reactor area to allow for equipment present in the room, one obtains an air concentration (specific activity) that may be compared to the derived air concentration (DAC) for occupational exposure as given in 10 CFR Part 20 or in EPA federal guidance.

Comparison with the DAC and the ALI. The ALI is the activity that, if ingested or inhaled, would lead to either (a) the maximum permissible committed effective dose equivalent incurred annually in the workplace, nominally 5 rem, or (b) the maximum permissible dose to any one organ or tissue, nominally 50 rem. The DAC is the air concentration that, if breathed by reference man for one work year (2,000 hours), would result in the intake of the ALI. ALI does not apply to noble-gas radionuclides.

Potential activity releases are compared to ALIs, and air concentrations in the reactor area are compared to DACs in **Tables 13.2-5** and **13.2-6**. For certain radionuclide, the maximum activity in **Tables 13.2-1 to 13.2-4** is taken, and the radionuclides with lower than $10\mu Ci$ are not included in **Tables 13.2-5** and **13.2-6**. However, there is no credible scenario for accidental inhalation or ingestion of the undiluted radioiodine released from a fuel element.

When DACs are compared with the potential airborne concentration of radionuclides in the reactor area, br85, br86, br87, br88, br89, br90, br91, br92, i131, i132, i133, i133m, i134, i134m, i135, i136, i136m, i137, i138, i139, i140, i141, kr88, kr89, kr90, kr91, kr92, kr93, kr94, kr96, xe134m, xe135, xe137, xe138, xe139, xe140, xe141, xe142, xe143, xe143m, ba143, ba144, ce144, ce145, ce146, cs138, cs139, cs140, cs141, cs142, la144, la145, mo102, mo103, nb100, nb101, nb102, nb97m, nb99, nb99m, pr146, rb90, rb91, rb92, rb93, sr89, sr93, sr94, sr95, sr96, tc102, tc103, te132, te135, y91, y96, y97, zr100, zr101, zr95, zr98 and zr99 are of the potential

[1] This volume excludes the reactor area chem.-lab (the volume including this lab is $5.65 \times 10^8 cm^3$)

consequence. However, annual dose limits could be attained only with a constant air concentration over a long period of time. The iodine-133 released in the failure of a single element, for example, would decay with a half-life of 20.8 hours. Thus, even the undetected failure of a fuel element would not be expected to lead to the violations of the occupational dose limits expressed in 10 CFR Part 20 or in other federal guidance.

Table 13.2-4 Particulate activities potentially released in maximum hypothetical accident at the UUTR for long-lived radionuclides obtained using SCALE6

Element	Potentially released activity (μCi) at time after reactor shutdown - in days			
	0 days	7 days	14 days	28 days
as83	0.1	0.0	0.0	0.0
as86	0.1	0.0	0.0	0.0
ba137m	0.6	0.6	0.6	0.6
ba139	1.0	0.0	0.0	0.0
ba140	1.0	0.7	0.5	0.2
ba141	0.9	0.0	0.0	0.0
ba142	0.9	0.0	0.0	0.0
ba143	0.9	0.0	0.0	0.0
ba144	0.7	0.0	0.0	0.0
ba145	0.3	0.0	0.0	0.0
ba146	0.1	0.0	0.0	0.0
ce141	0.9	0.8	0.7	0.5
ce143	0.9	0.0	0.0	0.0
ce144	0.9	0.9	0.8	0.8
ce145	0.6	0.0	0.0	0.0
ce146	0.5	0.0	0.0	0.0
ce147	0.3	0.0	0.0	0.0
ce148	0.3	0.0	0.0	0.0
ce149	0.1	0.0	0.0	0.0
ce150	0.1	0.0	0.0	0.0
cs137	0.6	0.6	0.6	0.6
cs138	1.1	0.0	0.0	0.0
cs139	1.0	0.0	0.0	0.0
cs140	0.9	0.0	0.0	0.0
cs141	0.7	0.0	0.0	0.0
cs142	0.4	0.0	0.0	0.0
cs143	0.2	0.0	0.0	0.0
cs144	0.1	0.0	0.0	0.0
ge86	0.1	0.0	0.0	0.0
la140	1.0	0.8	0.5	0.3
la141	0.9	0.0	0.0	0.0
la142	0.9	0.0	0.0	0.0
la143	0.9	0.0	0.0	0.0
la144	0.9	0.0	0.0	0.0
la145	0.6	0.0	0.0	0.0

la146	0.3	0.0	0.0	0.0
la146m	0.1	0.0	0.0	0.0
la147	0.1	0.0	0.0	0.0
la148	0.1	0.0	0.0	0.0
mo101	0.8	0.0	0.0	0.0
mo102	0.7	0.0	0.0	0.0
mo103	0.5	0.0	0.0	0.0
mo104	0.3	0.0	0.0	0.0
mo105	0.1	0.0	0.0	0.0
mo106	0.1	0.0	0.0	0.0
mo99	1.0	0.2	0.0	0.0
nb100	0.9	0.0	0.0	0.0
nb100m	0.1	0.0	0.0	0.0
nb101	0.8	0.0	0.0	0.0
nb102	0.4	0.0	0.0	0.0
nb102m	0.1	0.0	0.0	0.0
nb103	0.3	0.0	0.0	0.0
nb104	0.1	0.0	0.0	0.0
nb95	1.0	1.0	1.0	1.0
nb97	1.0	0.0	0.0	0.0
nb97m	0.9	0.0	0.0	0.0
nb98	0.9	0.0	0.0	0.0
nb99	0.6	0.0	0.0	0.0
nb99m	0.4	0.0	0.0	0.0
nd147	0.4	0.2	0.1	0.1
nd149	0.2	0.0	0.0	0.0
nd151	0.1	0.0	0.0	0.0
np239	1.6	0.2	0.0	0.0
pm147	0.4	0.4	0.4	0.4
pm149	0.2	0.0	0.0	0.0
pm151	0.1	0.0	0.0	0.0
pr143	0.9	0.7	0.5	0.3
pr144	0.9	0.9	0.8	0.8
pr145	0.6	0.0	0.0	0.0
pr146	0.5	0.0	0.0	0.0
pr147	0.4	0.0	0.0	0.0
pr148	0.3	0.0	0.0	0.0
pr149	0.2	0.0	0.0	0.0
pr150	0.1	0.0	0.0	0.0
pr151	0.1	0.0	0.0	0.0
rb88	0.6	0.0	0.0	0.0
rb89	0.7	0.0	0.0	0.0
rb90	0.7	0.0	0.0	0.0
rb90m	0.2	0.0	0.0	0.0
rb91	0.9	0.0	0.0	0.0
rb92	0.8	0.0	0.0	0.0

rb93	0.6	0.0	0.0	0.0
rb94	0.3	0.0	0.0	0.0
rb95	0.1	0.0	0.0	0.0
rh103m	0.5	0.4	0.4	0.3
rh105	0.2	0.0	0.0	0.0
rh106	0.1	0.1	0.1	0.1
ru103	0.5	0.4	0.4	0.3
ru105	0.2	0.0	0.0	0.0
ru106	0.1	0.1	0.1	0.1
sb128m	0.1	0.0	0.0	0.0
sb129	0.1	0.0	0.0	0.0
sb130	0.1	0.0	0.0	0.0
sb130m	0.2	0.0	0.0	0.0
sb131	0.4	0.0	0.0	0.0
sb132	0.2	0.0	0.0	0.0
sb132m	0.2	0.0	0.0	0.0
sb133	0.4	0.0	0.0	0.0
sb134	0.1	0.0	0.0	0.0
sb134m	0.1	0.0	0.0	0.0
se84	0.2	0.0	0.0	0.0
se85	0.2	0.0	0.0	0.0
se86	0.2	0.0	0.0	0.0
se87	0.1	0.0	0.0	0.0
se88	0.1	0.0	0.0	0.0
sn128	0.1	0.0	0.0	0.0
sn130	0.1	0.0	0.0	0.0
sn130m	0.1	0.0	0.0	0.0
sn131	0.1	0.0	0.0	0.0
sn131m	0.1	0.0	0.0	0.0
sn132	0.1	0.0	0.0	0.0
sr89	0.7	0.7	0.6	0.5
sr90	0.6	0.6	0.6	0.6
sr91	0.9	0.0	0.0	0.0
sr92	0.9	0.0	0.0	0.0
sr93	1.0	0.0	0.0	0.0
sr94	1.0	0.0	0.0	0.0
sr95	0.8	0.0	0.0	0.0
sr96	0.6	0.0	0.0	0.0
sr97	0.3	0.0	0.0	0.0
sr98	0.1	0.0	0.0	0.0
tc101	0.8	0.0	0.0	0.0
tc102	0.7	0.0	0.0	0.0
tc103	0.5	0.0	0.0	0.0
tc104	0.3	0.0	0.0	0.0
tc105	0.2	0.0	0.0	0.0
tc106	0.1	0.0	0.0	0.0

tc99m	0.9	0.2	0.0	0.0
te129	0.1	0.0	0.0	0.0
te131	0.4	0.0	0.0	0.0
te131m	0.1	0.0	0.0	0.0
te132	0.7	0.2	0.0	0.0
te133	0.6	0.0	0.0	0.0
te133m	0.5	0.0	0.0	0.0
te134	1.1	0.0	0.0	0.0
te135	0.5	0.0	0.0	0.0
te136	0.2	0.0	0.0	0.0
te137	0.1	0.0	0.0	0.0
u239	1.6	0.0	0.0	0.0
y100	0.1	0.0	0.0	0.0
y90	0.6	0.6	0.6	0.6
y91	0.9	0.9	0.8	0.7
y91m	0.5	0.0	0.0	0.0
y92	1.0	0.0	0.0	0.0
y93	1.0	0.0	0.0	0.0
y93m	0.4	0.0	0.0	0.0
y94	1.0	0.0	0.0	0.0
y95	1.0	0.0	0.0	0.0
y96	0.6	0.0	0.0	0.0
y96m	0.3	0.0	0.0	0.0
y97	0.5	0.0	0.0	0.0
y97m	0.3	0.0	0.0	0.0
y98	0.3	0.0	0.0	0.0
y98m	0.2	0.0	0.0	0.0
y99	0.3	0.0	0.0	0.0
zr100	0.9	0.0	0.0	0.0
zr101	0.5	0.0	0.0	0.0
zr102	0.3	0.0	0.0	0.0
zr103	0.1	0.0	0.0	0.0
zr95	1.0	1.0	0.9	0.8
zr97	1.0	0.0	0.0	0.0
zr98	0.9	0.0	0.0	0.0
zr99	0.9	0.0	0.0	0.0
Subtotal	79.7	13.5	11.7	9.9

Comparison with the Effluent Concentration. Effluent concentration, listed in the last columns of **Tables 13.2-5** and **13.2-6**, are defined as continuous exposure (8,760 hours per year) rather than 2,000 hours per year occupational exposure. Exposure to a constant airborne concentration equal to the effluent concentration for one full year results in the annual dose limit of 100 mrem to members of the public. As is apparent from **Tables 13.2-5** and **13.2-6**, the reactor area average concentrations immediately after fuel element failure significantly exceed the effluent concentrations for several radionuclides. Thus, only for these radionuclides is it necessary to consider radioactive decay and atmospheric dispersal after release in estimating potential risk to members of the public. For posting purposes, concentrations relative to DACs are additive. For dosimetry purposes, products of concentrations and times, relative to DAC-hours, are additive.

Table 13.2-5 Comparison of halogen and noble gas potential released activities immediately after the reactor shutdown with ALIs and reactor area concentrations with DACs and effluent concentrations (>10 μCi)

Element	Half-life	Potentially released activity (μCi)	Inhalation ALI (μCi)	DAC ($\mu Ci/cm^3$)	Effluent conc. Limit ($\mu Ci/cm^3$)	Reactor room conc. ($\mu Ci/cm^3$)	Ratio to DAC	Ratio to effluent conc. limit
br83	2.40h	1072.6	6.0×10^{4}	3.0×10^{-5}	9.0×10^{-8}	2.3×10^{-6}	0.1	26.0
br84	31.80min	2098.9	6.0×10^{4}	2.0×10^{-5}	8.0×10^{-8}	4.6×10^{-6}	0.2	57.2
br84m	6.0min	33.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.2×10^{-8}	0.7	72.4
br85	2.90min	2510.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.5×10^{-6}	54.7	5468.6
br86	55.1s	3510.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.6×10^{-6}	76.5	7647.0
br87	55.65s	4081.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.9×10^{-6}	88.9	8893.0
br88	16.29s	3483.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.6×10^{-6}	75.9	7590.0
br89	4.40s	2162.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.7×10^{-6}	47.1	4711.2
br90	1.91s	1125.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.5×10^{-6}	24.5	2451.7
br91	0.541s	448.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	9.8×10^{-7}	9.8	976.8
br92	0.343s	54.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-7}	1.2	118.6
i131	8.02d	5739.7	5.0×10^{1}	2.0×10^{-8}	2.0×10^{-10}	1.3×10^{-5}	625.2	62524.0
i132	2.30h	8728.2	8.0×10^{3}	3.0×10^{-6}	2.0×10^{-8}	1.9×10^{-5}	6.3	950.8
i132m	1.39h	18.5	8.0×10^{3}	4.0×10^{-6}	3.0×10^{-8}	4.0×10^{-8}	0.0	1.3
i133	20.8h	13183.9	3.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.9×10^{-5}	287.2	28723.2
i133m	9s	852.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-6}	18.6	1857.6
i134	52.5min	15561.3	5.0×10^{4}	2.0×10^{-5}	6.0×10^{-8}	3.4×10^{-5}	1.7	565.0
i134m	3.52min	725.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.6×10^{-6}	15.8	1580.7
i135	6.57h	12473.7	2.0×10^{3}	7.0×10^{-7}	6.0×10^{-9}	2.7×10^{-5}	38.8	4529.3
i136	83.4s	5293.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-5}	115.3	11531.6
i136m	46.9s	2487.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.4×10^{-6}	54.2	5419.7
i137	24.13s	6107.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-5}	133.1	13307.0
i138	6.23s	2966.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	6.5×10^{-6}	64.6	6462.1
i139	2.28s	1557.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.4×10^{-6}	33.9	3393.9
i140	0.86s	304.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	6.6×10^{-7}	6.6	662.9
i141	0.43s	82.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.8×10^{-7}	1.8	178.8
i142	0.2s	11.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.6×10^{-8}	0.3	25.8
kr83m	1.83h	1072.4	----	1.0×10^{-2}	5.0×10^{-5}	2.3×10^{-6}	0.0	0.0
kr85	10.78y	10.2	----	1.0×10^{-4}	7.0×10^{-7}	2.2×10^{-8}	0.0	0.0
kr85m	4.48h	2517.5	----	2.0×10^{-5}	1.0×10^{-7}	5.5×10^{-6}	0.3	54.8
kr87	76.3min	5119.2	----	5.0×10^{-6}	2.0×10^{-8}	1.1×10^{-5}	2.2	557.6
kr88	2.84h	6991.9	----	2.0×10^{-6}	9.0×10^{-9}	1.5×10^{-5}	7.6	1692.6
kr89	3.15min	8935.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-5}	194.7	19467.7
kr90	32.32s	9629.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.1×10^{-5}	209.8	20978.4

kr91	8.57s	6677.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.5×10^{-5}	145.5	14548.9
kr92	1.84s	3338.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.3×10^{-6}	72.7	7272.4
kr93	1.29s	974.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.1×10^{-6}	21.2	2123.9
kr94	0.21s	176.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.8×10^{-7}	3.8	384.4
kr95	0.11s	15.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.3×10^{-8}	0.3	32.6
kr96	0.08s	75.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.6×10^{-7}	1.6	163.5
xe131m	11.9d	61.9	----	4.0×10^{-4}	2.0×10^{-6}	1.3×10^{-7}	0.0	0.1
xe133	5.25d	13305.4	----	1.0×10^{-4}	5.0×10^{-7}	2.9×10^{-5}	0.3	58.0
xe133m	2.19d	386.5	----	1.0×10^{-4}	6.0×10^{-7}	8.4×10^{-7}	0.0	1.4
xe134m	0.29s	66.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.4×10^{-7}	1.4	143.7
xe135	9.14h	10918.7	----	1.0×10^{-5}	7.0×10^{-8}	2.4×10^{-5}	2.4	339.8
xe135m	15.3min	2313.8	----	9.0×10^{-6}	4.0×10^{-8}	5.0×10^{-6}	0.6	126.0
xe137	3.82min	12195.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-5}	265.7	26569.1
xe138	14.08min	12505.5	----	4.0×10^{-6}	2.0×10^{-8}	2.7×10^{-5}	6.8	1362.3
xe139	39.68s	10017.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.2×10^{-5}	218.3	21825.4
xe140	13.6s	7274.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.6×10^{-5}	158.5	15847.8
xe141	1.73s	2483.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.4×10^{-6}	54.1	5411.5
xe142	1.22s	878.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-6}	19.1	1914.2
xe143	0.51s	53.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-7}	1.2	117.1
xe143m	0.96s	53.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-7}	1.2	116.9
xe144	0.39s	12.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.8×10^{-8}	0.3	27.7

Table 13.2-6 Comparison of particulate potential released activities immediately after reactor shutdown with ALIs and reactor area concentrations with DACs and effluent concentrations (>10 μCi)

Element	Half-life	Potentially released activity (μCi)	Inhalation ALI (μCi)	DAC ($\mu Ci/cm^3$)	Effluent conc. Limit ($\mu Ci/cm^3$)	Reactor room conc. ($\mu Ci/cm^3$)	Ratio to DAC	Ratio to effluent conc. limit
as86	0.95s	10.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.2×10^{-8}	0.2	22.0
ba139	80.06min	127.5	3.0×10^{4}	1.0×10^{-5}	4.0×10^{-8}	2.8×10^{-7}	0.0	6.9
ba140	12.75d	123.0	1.0×10^{3}	6.0×10^{-7}	2.0×10^{-9}	2.7×10^{-7}	0.4	134.0
ba141	18.27min	115.6	7.0×10^{4}	3.0×10^{-5}	1.0×10^{-7}	2.5×10^{-7}	0.0	2.5
ba142	10.6min	114.2	1.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	2.5×10^{-7}	0.0	1.2
ba143	14.5s	110.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	239.9
ba144	11.5s	87.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-7}	1.9	190.2
ba145	4.31s	38.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.4×10^{-8}	0.8	83.7
ba146	2.22s	18.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.0×10^{-8}	0.4	40.0
ce141	32.5d	102.0	7.0×10^{2}	3.0×10^{-7}	1.0×10^{-9}	2.2×10^{-7}	0.7	222.2
ce143	33.04h	117.9	2.0×10^{3}	8.0×10^{-7}	3.0×10^{-9}	2.6×10^{-7}	0.3	85.6
ce144	284.9d	23.6	3.0×10^{1}	1.0×10^{-8}	4.0×10^{-11}	5.1×10^{-8}	5.1	1285.7
ce145	3.01min	78.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.7×10^{-7}	1.7	170.2
ce146	13.52min	59.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	129.7
ce147	56.4s	37.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.2×10^{-8}	0.8	81.8
ce148	56s	31.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	6.9×10^{-8}	0.7	69.0
ce149	5.3s	15.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.4×10^{-8}	0.3	33.8

cs138	33.41min	133.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.9×10^{-7}	2.9	290.8
cs139	9.27min	126.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-7}	2.7	274.7
cs140	63.7s	113.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.5×10^{-7}	2.5	248.0
cs141	24.84s	82.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.8×10^{-7}	1.8	180.3
cs142	1.69s	54.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-7}	1.2	117.7
cs143	1.79s	29.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	6.3×10^{-8}	0.6	63.2
ge86	150ns	12.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-8}	0.3	27.1
la140	1.68d	123.1	1.0×10^{3}	5.0×10^{-7}	2.0×10^{-9}	2.7×10^{-7}	0.5	134.1
la141	3.9h	116.0	9.0×10^{3}	4.0×10^{-6}	1.0×10^{-8}	2.5×10^{-7}	0.1	25.3
la142	91.1min	116.1	2.0×10^{4}	9.0×10^{-6}	3.0×10^{-8}	2.5×10^{-7}	0.0	8.4
la143	14.2min	117.6	9.0×10^{4}	4.0×10^{-5}	1.0×10^{-7}	2.6×10^{-7}	0.0	2.6
la144	40.8s	108.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	236.5
la145	24.8s	76.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.7×10^{-7}	1.7	166.5
la146	6.27s	33.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.2×10^{-8}	0.7	72.2
la146m	10.0s	14.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.2×10^{-8}	0.3	32.2
la147	4.02s	17.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.9×10^{-8}	0.4	38.6
mo101	14.61min	103.1	1.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	2.2×10^{-7}	0.0	1.1
mo102	11.3min	84.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.8×10^{-7}	1.8	185.0
mo103	67.5s	58.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	128.4
mo104	60s	35.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.8×10^{-8}	0.8	77.9
mo105	35.16s	18.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.0×10^{-8}	0.4	40.2
mo99	2.75d	121.7	1.0×10^{3}	6.0×10^{-7}	2.0×10^{-9}	2.7×10^{-7}	0.4	132.5
nb100	1.5s	117.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.6×10^{-7}	2.6	255.7
nb101	7.1s	99.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.2×10^{-7}	2.2	216.5
nb102	1.3s	56.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.2×10^{-7}	1.2	122.6
nb102m	4.3s	15.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.4×10^{-8}	0.3	34.2
nb103	1.5s	38.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.3×10^{-8}	0.8	83.3
nb95	35.0d	54.1	1.0×10^{3}	5.0×10^{-7}	2.0×10^{-9}	1.2×10^{-7}	0.2	59.0
nb97	72.1min	120.0	7.0×10^{4}	3.0×10^{-5}	1.0×10^{-7}	2.6×10^{-7}	0.0	2.6
nb97m	52.7s	113.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.5×10^{-7}	2.5	247.3
nb98	2.86s	113.3	5.0×10^{4}	2.0×10^{-5}	7.0×10^{-8}	2.5×10^{-7}	0.0	3.5
nb99	15.0s	72.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.6×10^{-7}	1.6	157.7
nb99m	2.6min	48.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.1×10^{-7}	1.1	105.5
nd147	10.98d	44.6	8.0×10^{2}	4.0×10^{-7}	1.0×10^{-9}	9.7×10^{-8}	0.2	97.1
nd149	1.73h	21.5	2.0×10^{4}	1.0×10^{-5}	3.0×10^{-8}	4.7×10^{-8}	0.0	1.6
np239	2.36d	211.2	2.0×10^{3}	9.0×10^{-7}	3.0×10^{-9}	4.6×10^{-7}	0.5	153.4
pm149	53.1h	21.3	2.0×10^{3}	8.0×10^{-7}	2.0×10^{-9}	4.6×10^{-8}	0.1	23.1
pr143	13.57d	117.4	7.0×10^{2}	3.0×10^{-7}	9.0×10^{-10}	2.6×10^{-7}	0.9	284.3
pr144	17.28min	23.6	1.0×10^{5}	5.0×10^{-5}	2.0×10^{-7}	5.1×10^{-8}	0.0	0.3
pr145	5.98h	78.0	8.0×10^{3}	3.0×10^{-6}	1.0×10^{-8}	1.7×10^{-7}	0.1	17.0
pr146	24.15min	59.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	129.8
pr147	13.4min	44.7	2.0×10^{5}	8.0×10^{-5}	3.0×10^{-7}	9.7×10^{-8}	0.0	0.3
pr148	2.29min	32.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.1×10^{-8}	0.7	70.7
pr149	2.26min	21.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.7×10^{-8}	0.5	46.6
pr150	6.19s	12.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-8}	0.3	27.0
rb88	17.77min	70.4	6.0×10^{4}	3.0×10^{-5}	9.0×10^{-8}	1.5×10^{-7}	0.0	1.7

rb89	15.15min	93.4	1.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	2.0×10^{-7}	0.0	1.0
rb90	158s	88.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-7}	1.9	191.9
rb90m	258s	25.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.6×10^{-8}	0.6	55.7
rb91	58.4s	110.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	241.6
rb92	4.49s	95.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.1×10^{-7}	2.1	208.4
rb93	5.84s	70.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.5×10^{-7}	1.5	153.7
rb94	2.70s	32.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.2×10^{-8}	0.7	71.5
rb95	0.38s	15.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.4×10^{-8}	0.3	33.6
rh103m	56.1min	50.1	1.0×10^{6}	5.0×10^{-4}	2.0×10^{-6}	1.1×10^{-7}	0.0	0.1
rh105	35.36h	18.5	6.0×10^{3}	2.0×10^{-6}	8.0×10^{-9}	4.0×10^{-8}	0.0	5.0
ru103	39.26d	50.2	6.0×10^{2}	3.0×10^{-7}	9.0×10^{-10}	1.1×10^{-7}	0.4	121.5
ru105	4.44h	19.5	1.0×10^{4}	5.0×10^{-6}	2.0×10^{-8}	4.2×10^{-8}	0.0	2.1
sb129	4.40h	10.9	9.0×10^{3}	4.0×10^{-6}	1.0×10^{-8}	2.4×10^{-8}	0.0	2.4
sb130	39.5min	15.7	6.0×10^{4}	3.0×10^{-5}	9.0×10^{-8}	3.4×10^{-8}	0.0	0.4
sb130m	6.3min	19.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.2×10^{-8}	0.4	41.7
sb131	23.03min	50.8	2.0×10^{4}	1.0×10^{-5}	6.0×10^{-8}	1.1×10^{-7}	0.0	1.8
sb132	2.79min	25.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.6×10^{-8}	0.6	56.5
sb132m	4.15min	29.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	6.3×10^{-8}	0.6	63.4
sb133	2.5min	47.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.0×10^{-7}	1.0	103.9
se84	3.1min	20.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.5×10^{-8}	0.4	44.9
se85	31.7s	20.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.5×10^{-8}	0.4	44.5
se86	15.3s	26.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.7×10^{-8}	0.6	56.6
se87	5.5s	15.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.4×10^{-8}	0.3	34.0
sn130	3.72min	12.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.6×10^{-8}	0.3	26.3
sn130m	1.7min	11.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.5×10^{-8}	0.2	24.9
sn132	39.7s	12.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.6×10^{-8}	0.3	26.1
sr89	50.57d	70.0	1.0×10^{2}	6.0×10^{-8}	2.0×10^{-10}	1.5×10^{-7}	2.5	762.5
sr91	9.63h	115.8	4.0×10^{3}	1.0×10^{-6}	5.0×10^{-9}	2.5×10^{-7}	0.3	50.5
sr92	2.66h	117.9	7.0×10^{3}	3.0×10^{-6}	9.0×10^{-9}	2.6×10^{-7}	0.1	28.5
sr93	7.42min	123.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-7}	2.7	269.9
sr94	75.3s	120.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.6×10^{-7}	2.6	262.1
sr95	23.90s	104.7	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.3×10^{-7}	2.3	228.1
sr96	1.07s	74.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.6×10^{-7}	1.6	162.6
sr97	0.429s	34.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.6×10^{-8}	0.8	75.9
sr98	0.653s	16.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	3.5×10^{-8}	0.4	35.3
tc101	14.22min	103.1	3.0×10^{5}	1.0×10^{-4}	5.0×10^{-7}	2.2×10^{-7}	0.0	0.4
tc102	5.28s	85.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.9×10^{-7}	1.9	185.4
tc103	54.2s	60.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	132.0
tc104	18.3min	37.6	7.0×10^{4}	3.0×10^{-5}	1.0×10^{-7}	8.2×10^{-8}	0.0	0.8
tc105	7.6min	19.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.2×10^{-8}	0.4	42.4
tc99m	6.0h	107.1	2.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	2.3×10^{-7}	0.0	1.2
te129	69.6min	10.0	6.0×10^{4}	3.0×10^{-5}	9.0×10^{-8}	2.2×10^{-8}	0.0	0.2
te131	25min	51.1	5.0×10^{3}	2.0×10^{-6}	2.0×10^{-8}	1.1×10^{-7}	0.1	5.6
te132	3.2d	85.4	2.0×10^{2}	9.0×10^{-8}	9.0×10^{-10}	1.9×10^{-7}	2.1	206.8
te133	12.5min	74.2	2.0×10^{4}	9.0×10^{-6}	8.0×10^{-8}	1.6×10^{-7}	0.0	2.0
te133m	55.4min	67.4	5.0×10^{3}	2.0×10^{-6}	2.0×10^{-8}	1.5×10^{-7}	0.1	7.3

te134	41.8min	138.6	2.0×10^{4}	1.0×10^{-5}	7.0×10^{-8}	3.0×10^{-7}	0.0	4.3
te135	19.0s	66.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.4×10^{-7}	1.4	144.6
te136	17.63s	26.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	5.8×10^{-8}	0.6	58.3
u239	23.45min	211.9	2.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	4.6×10^{-7}	0.0	2.3
y100	0.735s	12.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.7×10^{-8}	0.3	26.7
y91	58.5d	81.1	1.0×10^{2}	5.0×10^{-8}	2.0×10^{-10}	1.8×10^{-7}	3.5	883.4
y91m	49.7min	67.3	2.0×10^{5}	7.0×10^{-5}	2.0×10^{-7}	1.5×10^{-7}	0.0	0.7
y92	3.54h	119.3	8.0×10^{3}	3.0×10^{-6}	1.0×10^{-8}	2.6×10^{-7}	0.1	26.0
y93	10.18h	125.7	2.0×10^{3}	1.0×10^{-6}	3.0×10^{-9}	2.7×10^{-7}	0.3	91.3
y93m	0.82s	43.9	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	9.6×10^{-8}	1.0	95.7
y94	18.7min	128.1	8.0×10^{4}	3.0×10^{-5}	1.0×10^{-7}	2.8×10^{-7}	0.0	2.8
y95	10.3min	126.6	1.0×10^{5}	6.0×10^{-5}	2.0×10^{-7}	2.8×10^{-7}	0.0	1.4
y96	5.3s	79.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.7×10^{-7}	1.7	172.3
y96m	9.6s	40.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.7×10^{-8}	0.9	87.3
y97	3.75s	60.8	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	132.5
y97m	1.17s	36.4	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	7.9×10^{-8}	0.8	79.3
y98	0.55s	38.2	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.3×10^{-8}	0.8	83.1
y98m	2.0s	22.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	4.8×10^{-8}	0.5	47.9
y99	1.47s	41.6	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	9.1×10^{-8}	0.9	90.6
zr100	7.1s	111.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	241.9
zr101	2.3s	61.3	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	1.3×10^{-7}	1.3	133.6
zr102	2.9s	40.5	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	8.8×10^{-8}	0.9	88.3
zr103	1.3s	10.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.2×10^{-8}	0.2	22.0
zr95	64.0d	85.4	1.0×10^{2}	5.0×10^{-8}	4.0×10^{-10}	1.9×10^{-7}	3.7	465.2
zr97	16.74h	119.7	1.0×10^{3}	5.0×10^{-7}	2.0×10^{-9}	2.6×10^{-7}	0.5	130.3
zr98	30.7s	111.0	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	241.8
zr99	2.1s	112.1	2.0×10^{2}	1.0×10^{-7}	1.0×10^{-9}	2.4×10^{-7}	2.4	244.3

Comparison between UUTR MHA released activity and other facility's. REED TRIGA MHA is simulated at a thermal power 5.2 times of UUTR MHA thermal power, and REED total released activity from radioactive gas is 845214/214752= 3.94 times of UUTR's. REED total released activity from radioactive particulate is 32686/9483.8=3.45 times of UUTR's. REED total released activity is (845214+32686)/(214752+9483.8)=3.92 times of UUTR. The released activity is roughly proportional to the thermal power. Therefore, UUTR source term calculation has the correct order [Ref. Reed research reactor safety analysis report, August 2007].

Dose to individuals from the release of radionuclides. In order to estimate doses to individuals from the hypothetical release of the above described source inventory, four scenarios were examined:

Scenario A: In this scenario the west wall (with the effective surface area of $100m^2$) of the reactor room is suddenly disappeared. The fission products and gasses are released into the reactor room and mixed with the air instantaneously. The air in the reactor room is moved out through the missing wall at the wind speed of 1 m/s. This is a ground level release. It will take 4.7 seconds for the air to leave the reactor room at the rate of $100m^3/sec$.

Scenario B: The nuclide inventory shown in the tables above was completely released and instantly mixed homogenously with the existing air in the reactor room. The ventilation system was assumed to be working normally (operational mode) and the only effluent was through the exhaust stack at the roof of the building. It was assumed that the entire source inventory was exhausted through the ventilation system with one complete room-air change. The standard exhaust rate is $6.11 \times 10^5 cm^3/sec$; one room-air change was expected to take 12.6 minutes. The physical height of the exhaust stack is 40 ft; no modifications were made to the stack height based on exhaust velocity. The sub-scenario assumes that air is to be discharged at ground level for 12.6 minutes, and no credit is to be taken for an elevated release.

Scenario C: The nuclide inventory was completely released and instantaneously homogenously mixed with the existing air in the reactor room. It is assumed that the entire source inventory was leaked through the cracks in walls. The leak rates were assumed to be $1.69 \times 10^4 cm^3/sec$ (from literature) and a very conservative rate of $6.15 \times 10^3 cm^3/sec$. Respectively, it is expected to take 7.54 hours and 20.7 hours for one reactor room exchange (shown in **Table 13.2-8**). These are the exposure times for the individuals outside the reactor room. This is the ground level release.

Scenario D: The nuclide inventory was completely released and instantaneously homogenously mixed with the existing air in the reactor room. It is further assumed that there is no crack or leaks from the reactor room, and the ventilation system is "off" (i.e., completely shutdown as described in **UUTR SAR 9.4.3**). The worker is exposed to radiation in the reactor room for 2 and 5 minutes. This scenario is the most conservative assumption for the worker inside the reactor room and the only one calculated for the worker inside the reactor room.

For these scenarios, the dose to an occupational worker who is assumed to remain in the reactor room as well as the dose to an individual member of the public downwind from the reactor building were estimated. For these both individuals, the assumed breathing rate was $0.02 m^3/min$ [NRC Regulatory Guide 1.183, July 2000]. Nuclides in the source inventory listed in **Tables 13.2-5** and **13.2-6** were evaluated in the dose estimates.

The calculations of atmospheric relative concentration (χ/Q) are based on the NRC Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" [February 1983] and are presented in **Table 13.2-7** for ground level and stack release at 40ft (12.192 m).

The following correlations are used to calculate the committed dose equivalent (CDE) to the thyroid and the committed effective dose equivalent ($CEDE$) for members of the general public for each isotope of concern at a known distance from the UUTR facility (scenarios A, B, and C-internal):

$$(CDE \text{ or } CEDE)_D = \sum_i \left\{ \frac{\left(\frac{\chi}{Q}\right)_D \cdot BR \cdot DCF_{\text{int},i} \cdot A_i \lambda_v \left(e^{-\lambda_i t_1} - e^{-\lambda_i t_2}\right)}{\lambda_i} \right\}, \ \lambda_v = \frac{R_v}{V}$$

where,

$\left(\frac{\chi}{Q}\right)_D$: atmospheric dispersion factor at a given distance D, [sec/m^3] ; *NRC Regulatory Guide 1.145[P. 1.145.3]*

BR: breathing rate, [0.00033 m^3/sec]

$DCF_{int,i}$: internal dose conversion factor for isotope i, [mrem/μCi]

A_i: initial activity of isotope i released into the reactor room, [μCi]

R: ventilation or leakage of air from the reactor area,

100 m^3/sec for scenario A

0.611 m^3/sec for scenario B for both stack and ground release

0.0169 m^3/sec for scenario C: it takes 7.54 hours for one room air change

0.00615 m^3/sec for scenario C: it takes 20.7 hours for one room air change

V: reactor room volume, [459 m^3]

λ_v: ventilation constant, [1/sec]

λ_i: decay constant for isotope i, [1/sec]

t_1: time when plume first arrives at the receptor point, [sec]

t_2: time when plume has passed the receptor point, [sec]

Table 13.2-7 Atmospheric relative concentration for various distances

Distance (m)	σ_y (m)	σ_z (m)	$\frac{\chi}{Q}$ (sec m^{-3})	
			Ground Level Release	Stack Release
10	1.29	1.04	5.93x10^{-2}	3.41 x10^{-31}
50	2.45	1.20	2.71 x10^{-2}	4.16 x10^{-24}
100	3.90	2.20	9.27 x10^{-3}	7.95 x10^{-09}
150	6.18	3.22	4.00 x10^{-3}	1.23 x10^{-05}
200	8.21	4.13	2.35 x10^{-3}	1.20 x10^{-04}
250	10.21	4.98	1.57 x10^{-3}	3.13 x10^{-04}

In order to calculate the deep dose equivalent (DDE) for both the thyroid and the whole body for the member of the public, the following correlation is used for a known distance from the UUTR facility (scenarios A, B and C-external):

$$(CDE \text{ or } CEDE)_D = \sum_i \left\{ \frac{\left(\frac{\chi}{Q}\right)_D \cdot DCF_{ext,i} \cdot A_i \lambda_v \left(e^{-\lambda_i t_1} - e^{-\lambda_i t_2}\right)}{\lambda_i} \right\}$$

where, $DCF_{ext,i}$ is the external dose rate conversion factor for isotope i, [mrem-m^3/μCi-yr]. For the personnel in the reactor room area for a given period of time being exposed (stay time, ST), the CDE and $CEDE$ are calculated as follows (scenario D-internal):

$$(CDE \text{ or } CEDE)_{ST} = \sum_i \left\{ \frac{DCF_{\text{int},i} \cdot A_i \cdot BR\left(1 - e^{-\lambda_{eff} ST}\right)}{\lambda_{eff} V} \right\}$$

$\lambda_{eff} = \lambda_i + \lambda_v$

t_{ST}: stay-time of personnel in the reactor room exposed [sec]

DDE to personnel in the reactor room for a given period of time for both the thyroid and the whole body is obtained from the following equation (scenario D-external):

$$\left(CDE_{Thyroid} \text{ or } \mathrm{DDE}_{whole\_body}\right)_{ST} = \sum_i \left\{ \frac{DCF_{ext,i} \cdot A_i \left(1 - e^{-\lambda_{eff} ST}\right)}{\lambda_{eff} V} \right\}$$

The following values are used in above equations:

- The leak rate from the reactor room: 0.00615 m^3/sec
- Reactor room ventilation exhaust rate: 2.2x10^{9}cm^3/hr
- Reactor room volume: 4.59x10^{8}cm^3 = 459 m^3
- Receptor breathing rate: 3.3x10^{-4}m^3/sec (NRC "light work" rate).

The total effective dose equivalent (TEDE) is the summation of CDE and DDE. **Table 13.2-8** summarizes the dose to an occupational worker in the reactor room during release and to an individual member of the public downwind from the reactor building. The dose for thyroid is for both CDE and DDE. The estimated doses are much less than the applicable dose limits for an occupational worker (5,000 mrem) or for an individual member of the public (100 mrem). The scenarios described represent conservative conditions that are not likely to occur and estimate doses less than applicable dose limits. Assumptions that result in conservative estimates include:

- Source inventory based on exaggerated reactor run times
- Complete release of source inventory release into reactor room air
- Scenario A represent an unlikely rapid, ground-release condition
- Exposure times that equal the entire release of the source term
- Atmospheric conditions that exceed average values observed on the University of Utah campus

Table 13.2-8 Summary of dose estimates for MHA scenarios to public and workers at UUTR

Scenario Summary	Applicable Total Dose Limit (mrem)	Estimated Thyroid and TEDE* Dose (mrem)

Scenario A

Scenario Summary	Applicable Total Dose Limit (mrem)	Downwind distance (m)	Thyroid Dose	TEDE Dose
Estimated dose to the member of the public: Release of radiation to the outside of the reactor room through the missing west wall of the reactor room.	100	10	6.9	8.7
		50	3.1	3.9
		100	1.1	1.3
		150	0.5	0.6
		200	0.3	0.3
		250	0.2	0.2

Scenario B

Scenario Summary	Applicable Total Dose Limit (mrem)	Downwind distance (m)	Thyroid Dose		TEDE Dose	
			Ground release	Stack release	Ground release	Stack release
Estimated dose to the member of the public: Release of radiation through the exhaust stack (ventilation system in operational mode) and the ground release	100	10	6.2	$4x10^{-29}$	7.1	$4x10^{-29}$
		50	2.8	$4x10^{-22}$	3.2	$5x10^{-22}$
		100	0.97	$8x10^{-7}$	1.1	$9x10^{-7}$
		150	0.4	$1x10^{-3}$	0.5	$1x10^{-3}$
		200	0.2	$1x10^{-2}$	0.3	$1x10^{-2}$
		250	0.2	$3x10^{-2}$	0.2	$4x10^{-2}$

Scenario C

Scenario Summary	Applicable Total Dose Limit (mrem)	Downwind distance (m)	Thyroid Dose		TEDE Dose	
			7.54h	20.7h	7.54h	20.7h
Estimated dose to the member of the public: Entire source inventory was leaked through the cracks in walls.	100	10	5.4	4.8	5.9	5.2
		50	2.4	2.2	2.7	2.4
		100	0.9	0.7	0.9	0.8
		150	0.4	0.3	0.4	0.4
		200	0.2	0.2	0.2	0.2
		250	0.1	0.1	0.2	0.1

Scenario D

Scenario Summary	Exposure time	Applicable Total Dose Limit (mrem)	Thyroid Dose	TEDE Dose
Estimated dose to the worker inside the reactor room: no leaks of air from inside to outside; workers are exposed for 2 and 5 minutes (ventilation system assumed to be completely shutdown)	2 min	5,000	28	32
	5 min	5,000	69	80

*TEDE is the total effective dose equivalent

Because the UUTR is located inside of the Merrill Engineering Building (**UUTR SAR 2**), four locations were selected to estimate what will be the maximum TEDE (Total Effective Dose Equivalent) and thyroid dose during MHA assuming the ventilation system is "off" or in limited intake mode (ventilation system is "on"):

- **Location 1** (Mechanical Engineering Heat Lab, MEHL),
- **Location 2** (hallway-east side of the reactor room),
- **Location 3** (classroom next to the UNEF office area), and
- **Location 4** (2nd floor office area directly above the reactor core).

In estimating the dose values for all four locations and defined scenarios the buildup factors were not considered and it was assumed that the gamma emitted in MHA accident can be approximated with an average energy of 3 MeV that will transport through the concrete walls and air with attenuation reaching designated locations.

Location 1: Mechanical Engineering Heat Laboratory (MEHL), Fig.13.2-3

It is assumed that all isotopes homogeneously mixed with the air in the reactor room will reach the MEHL area and stay there without any leaks; this is the most conservative assumption. The following two scenarios are analyzed:

Scenario E.1: *Complete shutdown of the ventilation system, i.e. the ventilation system is "off":*
The nuclide inventory is completely released and instantaneously homogenously mixed with the existing air in the reactor room. It is assumed that the fraction of the initial source inventory is leaked through the door gaps and gap between the wall and ceiling to the MEHL. The leak rate is assumed to be $1.69 \times 10^5 cm^3/sec$ (0.037% of reactor room volume/sec) and one room-air change is expected to take 0.754 hours (~45 minutes). The assumption of the leak rate of 10 times greater than the leak rate from the reactor room to the outside of the MEB is conservative one because of the small gaps in the ceiling between the reactor room and MEHL. It is assumed that the radioactive nuclides leaked from the reactor room would instantaneously homogenously mix with the existing air in MEHL and will not leak out from MEHL. The concentration of the radioactive isotopes in MEHL will therefore increase constantly. In the dose calculation, this concentration increment was represented by 0.00037xt where t is the time in second.

Internal dose is calculated from the following equation

$$(CDE\ or\ CEDE)_{ST} = \sum_i \frac{DCF_{\text{int},i} \cdot A_i \cdot BR(1 - e^{-\lambda_{eff} t_{ST}})}{\lambda_{eff} V} \qquad (1)$$

where
CDE: Committed Dose Equivalent

$CEDE$: Committed Effective Dose Equivalent
$DCF_{int,i}$: internal dose conversion factor for isotope i, [mrem/μCi]
A_i: activity of isotope i released into MEHL, [μCi]
BR: breathing rate, [0.00033 m^3/sec]
t_{ST}: stay-time of personnel in the Heat Lab exposed [sec]
$\lambda_{eff} = \lambda_i + \lambda_v$
λ_i: decay constant for isotope i, [1/sec]
λ_v: ventilation constant, [1/sec] =2.753×10^{-4} /sec
V: MEHL volume, [613.9 m^3]

External dose is obtained as a summation of: the dose from the isotopes in MEHL and the dose from the isotopes in the reactor room. For these calculations, it is required to include the distance from the center of the reactor to the people in MEHL. Also, the thickness of any concrete wall is taken into account.

The external dose from the radionuclides in the reactor room is obtained from

$$(CDE_{Thyroid} \text{ or } DDE_{whole\_body})_{ST} = \sum_i \frac{DCF_{ext,i} \cdot A_i \cdot (1 - e^{-\lambda_{eff} t_{ST}})}{\lambda_{eff} V} \cdot e^{-\mu_{air} X_{air}} \cdot e^{-\mu_{conc} X_{conc}} \quad (2)$$

where
$CDE_{Thyroid}$: Committed Dose Equivalent for Thyroid
DDE_{whole_body}: Deep Dose Equivalent
$DCF_{ext,i}$: external dose conversion factor for isotope i, [mrem/μCi]
A_i: initial activity of isotope i released into the reactor room, [μCi]
μ_{air}: attenuation coefficient for air (1/cm)
μ_{conc}: attenuation coefficient for concrete (1/cm)
X_{air}: distance from the reactor room wall to the people in the MEHL (426.72 cm)
X_{conc}: thickness of concrete wall between reactor room and MEHL (45.72 cm)
t_{ST}: stay-time (exposure time) of the personnel in the MEHL [sec]
V: reactor room volume, [459 m^3]
λ_v: ventilation constant, [1/sec]=0

Therefore, the external dose from the radionuclides in MEHL is obtained from

$$(CDE_{Thyroid} \text{ or } DDE_{whole\_body})_{ST} = \sum_i \frac{DCF_{ext,i} \cdot A_i \cdot (1 - e^{-\lambda_{eff} t_{ST}})}{\lambda_{eff} V} \quad (3)$$

where
A_i: activity of isotope i released into the MEHL, [μCi]

Scenario F.1: *Limited intake mode, i.e. ventilation system "on":*
The assumption is that there will be the leak rate from the reactor room to MEHL. If the ventilation system is on, the reactor room pressure will drop significantly. It is therefore assumed that only small amount of radioisotopes i.e., $0.1 x 5.07 x 10^4 x 0.3\ cm^3/sec = 5.07 x 10^3 x 0.3\ cm^3/sec$ will leak into the MEHL. The factor of 0.1 is introduced to take into account the two fume hoods existing in between these two areas, MEHL and the reactor room. In other words, if the ventilation is on, then almost all of the radioisotopes will be captured by these two fume hoods before they enter into the MEHL area. It was therefore assumed that the presence of the two fume hoods would lower the leakage by a factor of 0.1. Thus, the only difference between this case and the previous one where ventilation system is "off", is that λ_v has to be added into the dose equations.

Internal dose is calculated from equation (1) with the volume of MEHL and ventilation constant of $8.26x10^{-5}$ /sec.

The external dose from the radionuclides in the reactor room is obtained from equation (2) with the ventilation constant λ_v = 0.00133 /sec. External dose from the radionuclides in MEHL is given by equation (3) with ventilation constant of $8.26x10^{-5}$ /sec, and MEHL volume of 613.9 m^3.

Location 2: Hallway (East side of the reactor room), Fig.13.2-3

It is assumed that the MHA nuclide inventory is completely released and instantaneously homogenously mixed with the existing air in the reactor room. Because there are two 1-foot-thick concrete walls between the hallway and the reactor room, it is assumed that no radioactive leak would take place from the reactor room toward the hallway. There will be no internal dose therefore for a person standing in the hallway. The external dose will come from the gamma rays from the reactor room directly. All beta and alpha particles will be blocked before they would reach the hallway area. The distance from the wall of the reactor room and hallway need to be taken into account, as well as the thickness of the two concrete walls (~1-foot-thick).

Scenario E.2: *Complete shutdown of the ventilation system, i.e. the ventilation system is "off":*
The external dose is obtained from equation (2) including the thickness of the concrete wall and air gap between the reactor room and hallway of:
X_{air}: distance from the center of the reactor to the people in the hallway (835.66cm)
X_{conc}: Thickness of concrete wall between reactor room and hallway (60.96 cm)

Scenario F.2: *Limited intake mode, i.e. ventilation system "on":*
The external dose is obtained from equation (2) with the ventilation constant of 0.00133/sec.

Location 3 (Classroom next to the UNEF office area), Fig.13.2-3

It is assumed that the MHA nuclide inventory is completely released and instantly mixed homogenously with the existing air in the reactor room. The leak rate is assumed to be $1.69 \times 10^{4} cm^{3}/sec$ from the reactor room toward the classroom in which case the isotopes will instantly and homogenously mix with the existing air in the classroom. Also it is assumed that the isotopes would stay in the classroom without leaking out (as a most conservative assumption). Therefore, the nuclide concentration will increase in time.

Scenario E.3: *Complete shutdown of the ventilation system, i.e. the ventilation system is "off":*
Internal dose is obtained from equation (1) with ventilation constant of $8.775 \times 10^{-5}/sec$ and classroom air volume of 192.6 m^3. External dose is calculated using equation (2) with the following variables:
X_{air}: distance from the reactor room wall to the people in the classroom (670.56 cm)
X_{conc}: thickness of concrete or dry wall between reactor room and classroom (30.48 cm)
λ_v: ventilation constant, [1/sec] =0

External dose from the radionuclides in the classroom is obtained from equation (3) with the ventilation constant of $8.775 \times 10^{-5}/sec$ and classroom volume of 192.6 m^3. Total external dose in the classroom is a sum of the external dose from the reactor room and the classroom.

Scenario F.3: *Limited intake mode, i.e. ventilation system "on":*
In this case, it is assumed that the leak rate from the reactor room to the classroom is 30% of the normal leak ($0.3 \times 1.69 \times 10^{4}$ $cm^3/sec = 5.07 \times 10^{3}$ cm^3/sec). The only difference between this case and the ventilation "off" case is that a different λ_v need to be included into the equations as described already for the MEHL dose estimates.

Internal dose is calculated from equation (1) with the classroom volume V=192.6 m^3 and the ventilation constant of $2.632 \times 10^{-5}/sec$.
The external dose from the radionuclides in the reactor room is calculated from equation (2) with the ventilation constant of 0.00133/sec and the reactor room volume of 459 m^3.

External dose from the radionuclides in the classroom is obtained from equation (3) with ventilation constant (2.632×10^{-5} /sec) and classroom volume V. Total external dose in the classroom is the sum of the doses from the reactor room and the classroom.

Location 4: (2nd floor office area directly above the reactor core), Fig. 13.2-3

The nuclide inventory is assumed to be completely released and instantly and homogenously mixed with the existing air in the reactor room. The dose is from gamma rays only because of the realistic assumption of no leaking from the reactor room due to 30.48-cm-thick concrete ceiling and 0.32-cm-thick stainless steel floor between the 2nd floor and the reactor room. There will be therefore no internal dose.

Scenario E.4: *Complete shutdown of the ventilation system, i.e. the ventilation system is "off":*
The external dose is obtained using equation (2) with following variables:
X_{steel}: Thickness of stainless steel floor (0.32 cm)
X_{conc}: Thickness of concrete floor (30.48 cm)
λ_v =0
All other variables and constants are same as used in the calculation for the hallway.

Scenario F.4: *Limited intake mode, i.e. ventilation system "on":*
External dose is obtained from equation (2) with the ventilation constant of 0.00133/sec.

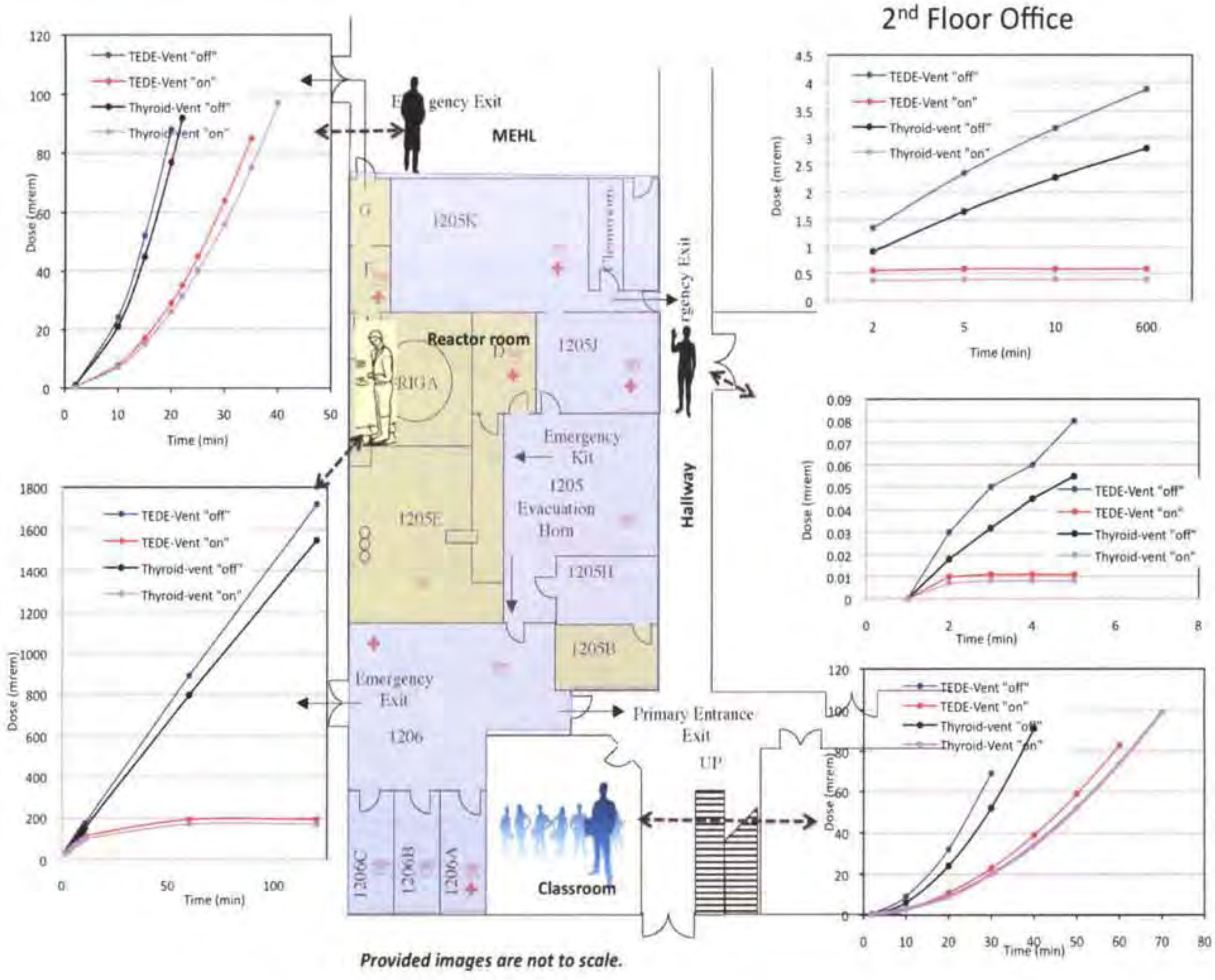


Figure 13.2-3 TEDE and thyroid MHA dose vs times in Merrill Engineering Building

Table 13.2-9 Summary of MHA doses in Merrill Engineering Building

Scenario summary	Stay time (min)	TEDE dose (mrem)	
		Complete shutdown of the ventilation system, i.e. ventilation "off")	Limited intake mode, i.e. ventilation "on"
Reactor room (Scenario D)			
Estimated dose to worker inside the reactor room: no leaks of air from inside to outside; workers are exposed for 2 ~ 120 minutes	2	32	30
	3	48	43
	4	64	55
	5	80	66
	6	95	76
	6.3	100	79
	6.5	103	81
	7	111	85
	7.5	119	90
	8	126	94
	9	142	102
	60	889	192
	120	1717	193
LOCATION 1 (MEHL)		Scenario E1	Scenario F1
Estimated dose to people in the Mechanical Engineering Heat Laboratory (MEHL)	2	1	0.3
	10	24	8
	15	52	17
	20	88	29
	22	>100	35
	25	>100	45
	30	>100	64
	35	>100	85
	40	>100	>100
	50	>100	>100
	60	>100	>100
	70	>100	>100
LOCATION 2 (Hallway Area)		Scenario E2	Scenario F2
Estimated dose to people in a hallway	2	0.03	0.01
	5	0.05	0.01
	10	0.06	0.01
	600	0.08	0.01
LOCATION 3 (Classroom)		Scenario E3	Scenario F3
Estimated dose to personnel in classroom next to the UNEF office area	10	9	3
	20	32	11
	30	69	23
	40	>100	39
	50	>100	59
	60	>100	83
	70	>100	>100

	80	>100	>100
	90	>100	>100
	100	>100	>100
	110	>100	>100
	120	>100	>100
LOCATION 4 (Office at the 2nd Floor)		Scenario E4	Scenario F4
Estimated dose to personnel in 2nd floor office area directly above the reactor core	2	1	0.56
	5	2	0.59
	10	3	0.59
	600	4	0.59

Table 13.2-10 Summary of MHA thyroid doses in Merrill Engineering Building

Scenario summary	Stay time (min)	Thyroid dose (mrem)	
		Complete shut down of the ventilation system, i.e. ventilation "off")	Limited intake mode, i.e. ventilation "on"
Reactor room (Scenario D)			
Estimated dose to the worker inside the reactor room: no leaks of air from inside to outside; workers are exposed for 2 ~ 120 minutes	2	28	26
	3	42	37
	4	55	47
	5	69	57
	6	83	66
	7	96	74
	7.5	103	78
	8	110	81
	9	124	88
	10	137	95
	11	151	101
	60	796	169
	120	1550	170
LOCATION 1 (MEHL)		Scenario E1	Scenario F1
Estimated dose to the Mechanical Engineering Heat Laboratory	2	0.92	0.29
	10	21	7
	15	45	15
	20	77	26
	22	92	31
	25	>100	40
	30	>100	56
	35	>100	75
	40	>100	97
	50	>100	>100
	60	>100	>100
	70	>100	>100
LOCATION 2 (Hallway Area)		Scenario E2	Scenario F2

Estimated dose to the people in the hallway	2	0.02	0.01
	5	0.03	0.01
	10	0.05	0.01
	600	0.06	0.01
LOCATION 3 (Classroom)		Scenario E3	Scenario F3
Estimated dose to the classroom next to the UNEF office area	2	0.33	0.17
	10	6	3
	15	14	5
	20	24	9
	25	37	14
	30	52	20
	35	70	27
	40	91	34
	50	>100	52
	60	>100	74
	70	>100	99
	80	>100	>100
	90	>100	>100
	100	>100	>100
	110	>100	>100
	120	>100	>100
LOCATION 4 (Office at the 2nd Floor)		Scenario E4	Scenario F4
2nd floor office area directly above the reactor core	2	0.91	0.37
	5	2	0.39
	10	2	0.39
	600	3	0.39

Figure 13.2-3 shows the TEDE and thyroid doses for these four locations and **scenarios E*i*** and **F*i*** (where *i* indentifies the location number) in the Merrill Engineering building that could potentially be affected by the MHA. **Table 13.2-9** shows TEDE dose values for each of the locations, while the thyroid doses are shown in **Table 13.2-10**. In summary:

- Location 1 (MEHL): When the ventilation system is "off" (i.e., *complete shut down of the ventilation system*) the doses are higher in comparison to the ventilation system "on" (limited intake mode) scenario. After around 20 min it is expected that the total as well as thyroid dose would potentially reach the limit of 100 mrem. This is a plenty of time for reactor personnel to initiate the fire alarm and evacuate the area. According to the annual fire alarm drill with Salt Lake City Fire Department, the average evacuation time in the Merill Engineering Building is approximately 5 minutes. The 10CFR20 limitations – provide more specific information about how we ensure that the evacuation will be completed in 20min. The UUTR Emergency plan addresses the evacuation procedure.
- Location 2 (Hallway area): Maximum TEDE is estimated to be 0.08 mrem after 10 hours when the ventilation system is "off". The total thyroid dose during the same time is around 0.05 mrem. Both values are obviously well below the limit.

- Location 3 (classroom): In approximately 30 min the total TEDE dose and in around 45 min the total thyroid dose will reach the limit value of 100 mrem when the ventilation system is "off". Half an hour is a long enough time to evacuate the area.
- Location 4 (office at the 2nd floor): Ten hours will be required for TEDE dose to reach 4 mrem and thyroid dose to be 3 mrem, with the ventilation system "off " (i.e., *complete shut down of the ventilation system*).
- Reactor room (Scenario D): In 120 minutes the total TEDE dose in the reactor room will reach the value of approximately 1,600 mrem when the ventilation system is "off" (*i.e., complete shut down of the ventilation system*), which is well below the dose limit for a worker in the reactor room. The UUTR internal policy requires the area to be evacuated in less than 5 min should an accident occur.

13.2.2 Insertion of Excess Reactivity

The excess reactivity insertion is modeled based on the UUTR existing guidelines for each experiment. The reactivity insertion of $1.2 is assumed (for example, the $1.2 reactivity insertion is considered to be an extreme scenario of extremely low probability to occur at the UUTR). This value is selected based on Technical Specification of UUTR according to which the maximum value for reactivity insertion for a single experiment is limited to $1.00. In order to stay on the conservative side, 20 cents is added to this value. The modeling is based on synergistic use of PARET-ANL and MCNP5codes. In these models an automatic SCRAM is introduced when the UUTR core power exceeds 100% of the licensed operational power. Due to limitations of the time step size in the model and the rapid prompt jump in the power, the SCRAM is observed at peak power of 1.1 MW rather than at 100kW [the model is thus conservative in estimating the consequences of the accident]. The SCRAM at 1.1 MW rather than at 100kW introduces more power (while very small time step) for temperatures to rise.

The reactor's initial power is set to 100kW because that is a maximum licensed power of the UUTR. The fuel temperature, cladding temperature, and the moderator temperature are analyzed to find if the fuel and cladding temperatures exceed the limits. In order to calculate the fuel temperature feedback coefficients suitable to be used for PARET-ANL [1], the neutronics calculations were performed for several temperatures and the following correlation [1] was used to fit the data:

$$\rho(\$)=D\left[\gamma_0+\gamma_1(T+\gamma_4)+\gamma_2(T+\gamma_4)^2+\gamma_3(T+\gamma_4)^n\right]$$

where

ρ: reactivity in units of dollar
γ_i: fitting coefficients
D: Doppler coefficient

T: average fuel temperature

The multiplication factor as a function of fuel temperature is given in **Table 13.2-11**. As expected, the multiplication factor decreases as the fuel temperature increases due to the fuel temperature feedback. The quadratic fitting (as used in PARET-ANL) of the reactivity versus fuel temperature is presented in **Fig. 13.2-4** giving the coefficients used in PARET-ANL for fuel temperature feedback coefficients as given in **Table 13.2-12**.

Table 13.2-11 UUTR multiplication factor as a function of fuel temperature obtained using MCNP5

T °K(°C)	k-MCNP5	ρ ($)
293.15 (20)	1.00649	0.9
600.15 (327)	0.97868	-2.8
800.15 (527)	0.95322	-6.4
1,200.15 (927)	0.91442	-12.2

Table 13.2-12 UUTR reactivity as a function of fuel temperature fitting data as used in PARET-ANL

Fitting	D	γ_0	γ_1	γ_2	γ_3	γ_4	n
Linear	1.0	5.3876	-1.4561×10^{-2}	0.0000	0.0000	0.0000	1.0
Quadratic	1.0	3.6502	-9.6773×10^{-3}	-2.9459×10^{-6}	0.0000	0.0000	1.0

The reactivity insertion accident is modeled as follows: reactivity is inserted at 0.25 seconds and the UUTR SCRAM automatically acts when the power passes the set point. The temperature coefficient is plotted in **Fig. 13.2-4**, and the power profile in **Fig. 13.2-5.** The temperature of the fuel and the moderator and the departure from nucleate boiling are analyzed to assure that fuel will not melt or the temperature would not increase over the limits. The moderator and fuel temperatures are shown in **Fig. 13.2-6**: the fuel and the moderator temperatures do not exceed the limits verifying that during an excess activity insertion accident, the fuel integrity will not be jeopardized; the fuel temperature change is more rapid than the water temperature change; the moderator temperature after 12 seconds from shutdown (SCRAM) will reach the temperature very close to inlet temperature which is 20.72 °C (293.87 °K).

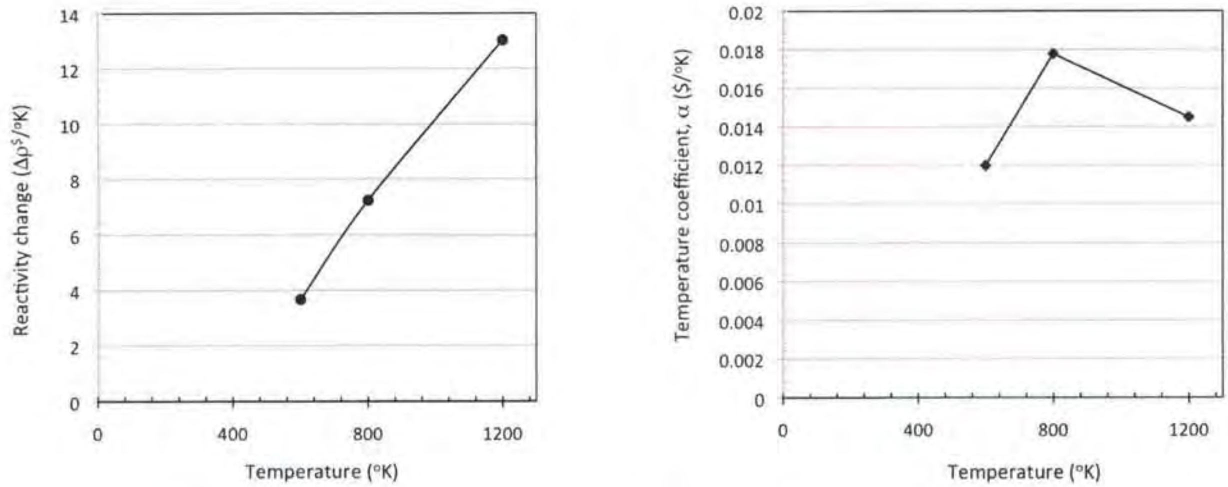


Figure 13.2-4 MCNP5 fuel temperature feedback coefficient for the 100 kW UUTR

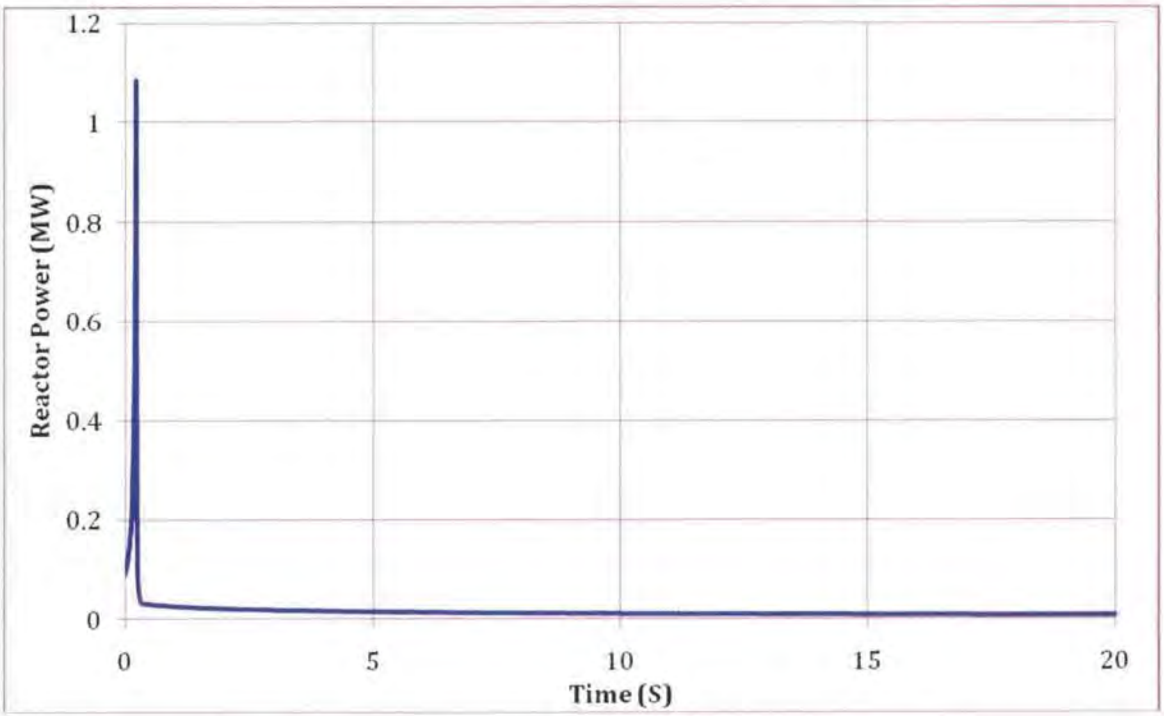


Figure 13.2-5 UUTR reactor power change versus reactor time for $1.2 reactivity insertion accident

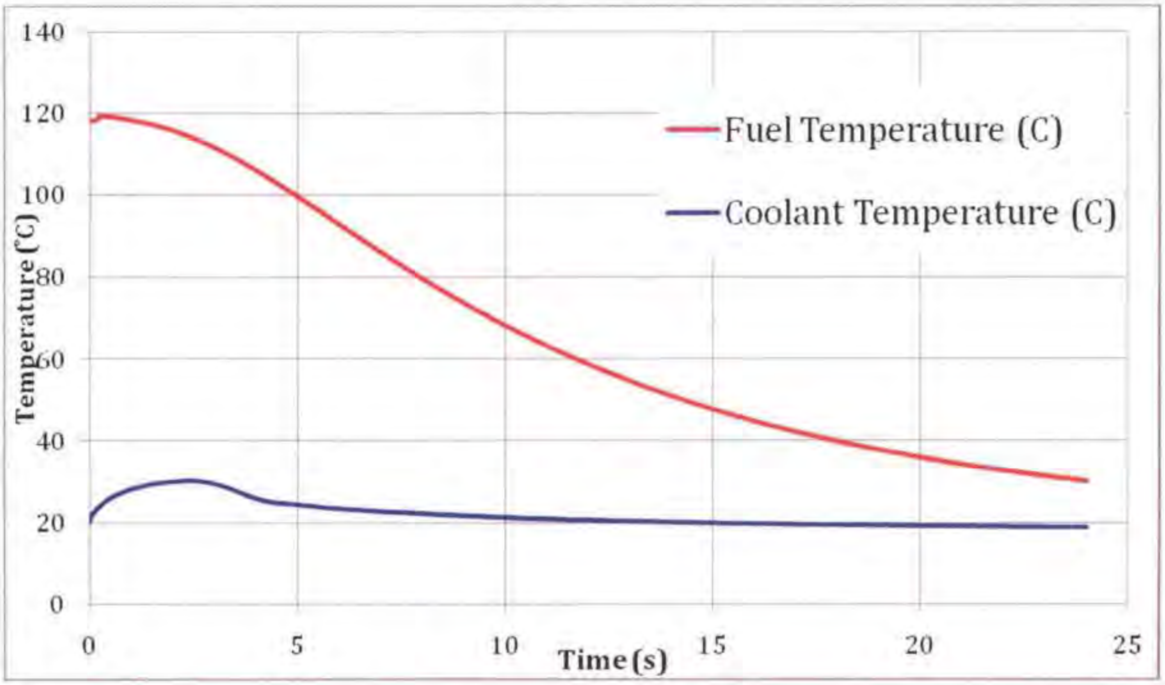


Figure 13.2-6 UUTR maximum centerline fuel temperature and coolant temperature as a function of time for $1.2 reactivity insertion

The DNBR estimates described in **Section 4.6.2** showed that the CHF is to be approximately 520kW/m^2 (obtained from **Fig. 4.6-9**). The hottest rod during the power prompt jump to 1.1 MW (as shown in **Fig. 13.2-5**) will be approximately 22kW, which gives the maximum heat flux of 491.45kW/m^2. This predicts the DNBR of 1.06, which is a safe value for the very short amount of time before the SCRAM. The fuel temperature rises to approximately 393.15 °K (120 °C) which is within the safe operating limits of TRIGA fuel. Therefore, during the unlike event of excess reactivity accident in the order of $1.2, the integrity of the fuel and cladding are not jeopardized.

13.2.3 Loss of Coolant Accident (LOCA)

__Accident initiating events and scenarios__. Although total loss of reactor pool water is considered to be an extremely improbable event, such a failure is analyzed. Limiting design basis parameters and values are addressed in "The U-Zr-Hx Alloy: Its Properties and Use in TRIGA Fuel," *Report E-117-833, Simnad, M T, General Atomics Corp., 1980* as follows:

Fuel-moderator temperature is the basic limit of TRIGA reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH, and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set *the upper* limit *on the fuel temperature. A fuel temperature safety limit of 1,423.15 °K for pulsing, stainless steel $UZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 773.15 °K. When clad temperatures can equal the, fuel temperature, the fuel temperature limit is 1,223.15 °K. There is also a steady-state operational fuel temperature design limit of 1,023.15 °K based on consideration of irradiation-and fission-product-induced fuel growth and deformation.*

Loss of pool water in UUTR. UUTR facility has no beam ports and the reactor cooling systems and return lines contain ant-siphon holes approximately one foot below the normal water level to prevent the tank from drying out. The only possible scenario for losing the water inside the tank is an intense shock from earthquake. However, this unlikely event has been analyzed for amount of radiation released from the open core, and the change of fuel temperature.

A float switch alarm, operating on a 24 hour basis is provided [redacted] if the water level falls below 6 inches of its normal level. In such an event, the reactor if under operation is automatically shut down by SCRAM signal and [redacted], the UNEF Director, the Reactor Supervisor, and the Radiation Safety Officer are alerted immediately.

The following is the calculation of the effect of loss of pool water and it follows the methodology used in the 1985 SAR:

The time it takes for the water in the tank to drain is calculated by Darcy's law as follows:

$$\frac{dx}{dt} = Kp$$

where,

K: constant depending on the nature of the sand, maximum value of 3,000 $\left[\frac{m^2}{year \cdot m(H_2O)}\right]$

[Handbook of Applied Hydraulics, McGraw Hill, New York (1952) p. 165]

Therefore, the time it would take for the tank to drain is t=1/K =19.3 hours (The height of the tank is 6.7 meters). This time is long enough for the personnel to use continual flooding of the core tank from the tap water line in the laboratory until radiation levels are acceptable, or the fuel can be removed to a storage facility.

Integrity of Fuel Element Cladding. The following analyses are the same as reported in 1985 SAR. The 1985 SAR calculations were based on a two-dimensional, transient heat transfer code, developed by Gulf Energy and Environmental Systems, Inc., for obtaining maximum temperature in the core after a water loss; the maximum fuel temperature for 100kW operation was obtained to be 334.15 °K (61 °C). Thus, as stated in the 1985 UUTR SAR, this temperature is low enough that the pressure exerted by trapped air and fission product gases in the fuel is less than 30 psi that produces a stress of 264 psi; the yield stress for aluminum at

150 °C (423.15 °K) is 8,000 psi. These values are obtained under the assumption that the heat is removed by natural convection of air; the conduction to grid and radiation heat losses are neglected.

The after-shutdown power density in the B-ring (**Fig. 4.6-1**) fuel element (hottest channel) is given by a modified version of Unterneyer-Weill formula as follows:

$$\frac{q}{V} = 0.1p\frac{P}{V_f}\cos\left[0.78\frac{\pi}{L}\left(x-\frac{L}{2}\right)\right]\left[\left(t+t_0+10\right)^{-0.2}-0.87\left(t+t_0+2\times10^7\right)^{-0.2}-0.05\right]$$

p: peak-to-average power density in the core = 1.282 as estimated using the MCNP5 model for 100kW UUTR (Section **4.5.2**)
P: reactor power = 3.413 x 10^5 Btu/hr
V_f: volume of the fuel in the core = 0.78 ft^3
L: length of the fuel = 1.25 ft
x: distance measured from the bottom of the fuel element, ft
t: time after the core exposed to the air, sec
t_0: time from shutdown to the time the core is exposed, sec

Calculated power density values for 100kW in the B-ring as a function of time exposed to air at the center are presented in **Table 13.2-13**. The assumption for this evaluation is that the reactor has been operating for 1,000 hours before the accident occurs. It is conservatively assumed that all the energy produced by fission products decay in the element is deposited in that element. Axial power density profile for 100kW is shown in **Fig. 13.2-7** at time=0 of exposure to air; as expected, the profile follows cosine shape distribution.

Table 13.2-13 Power density for fuel element in ring B as a function of time exposed to air at the center (axially)

Time exposed to air (s)	**Power Density** $\left[\frac{\text{Btu}}{\text{hr}\cdot\text{ft}^3}\right]$
	100 kW
0	1555.86
60	1554.80
120	1553.75
3600 (1 hr)	1494.21
7200 (2 hrs)	1436.16
14400 (4 hrs)	1329.42
86400 (1 day)	646.01
172800 (2 days)	211.53

The heat is removed from the by the natural convection of air. The air velocity through the channel can be determined by setting the frictional pressure loss equal to the buoyancy. Entrance and exit losses are about 2 to 5% which will be ignored:

$$\delta P(buoyancy) = \delta P(friction)$$

$$\delta P(buoyancy) = (\rho_0 - \rho_1)\frac{L}{2}$$

where L is the length of the channel and densities at the entrance and exit are ρ_0 and ρ_1 respectively. The frictional pressure drop for TRIGA is modeled so the free-flow area is converted into an annulus around the fuel element. With an annular space of inner diameter D_1 and outer diameter D_2, the frictional pressure drop becomes:

$$\delta P(friction) = \frac{32\mu v L}{g\left[D_2^2 + D_1^2 \frac{D_2^2 - D_1^2}{\ln\left(\frac{D_2}{D_1}\right)}\right]}$$

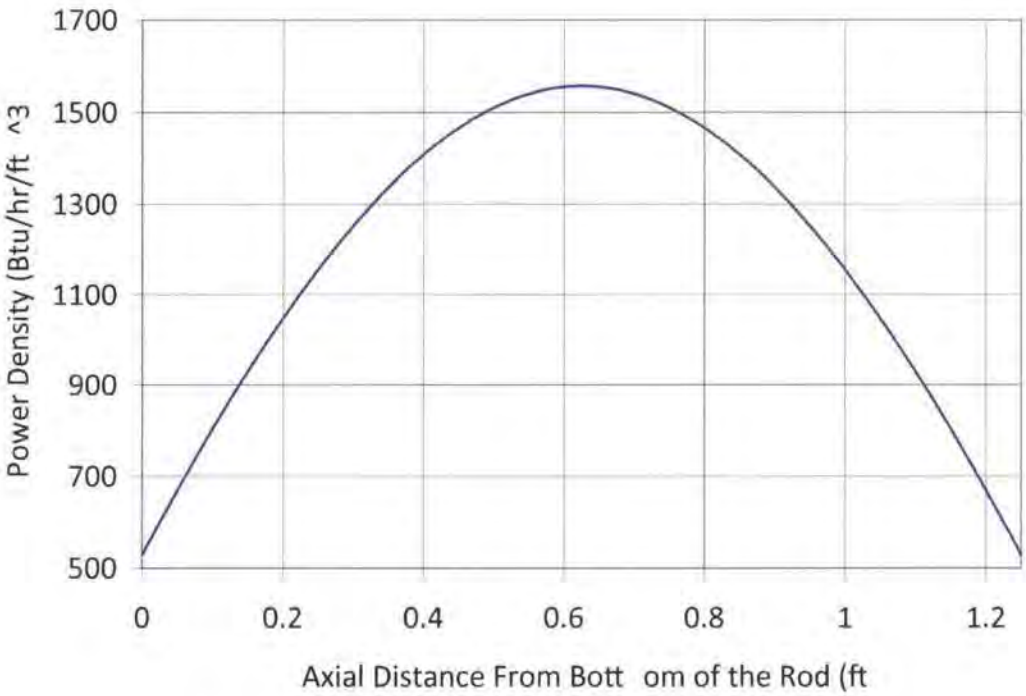


Figure 13.2-7 Axial power density profile for a fuel element in the ring B

μ: viscosity of air, lb/hr-ft
v: velocity of the air, ft/hr
L: length of the fuel element, ft
D_1: fuel element diameter, ft
D_2: $D_1 + 2b$, ft (b is the effective separation distance between B ring and C ring)

$$b = \frac{1}{2}(0.0113 + 0.04930) = 0.0303 f\, i$$

For the channel between the B ring and the C ring, the pressure balance equation showed earlier becomes:

$$(\rho_0 - \rho_1) = 6.88 \times 10^{-5}\, \overline{\mu v}$$

where the $\bar{\mu}$ is the average viscosity of the air in the channel and is a function of the entrance and exit temperatures, $\bar{v}$ and is the average air velocity in the channel. The mass flow rate of air in the channel is calculated using:

$$w = v\bar{\rho}A_c$$

where $\bar{\rho}$ is the average air density in the channel and A_c is the flow area associated with the channel. Combining the equation for mass flow rate and the pressure balance, it follows:

$$w = 405(\rho_0 - \rho_1)\frac{\bar{\rho}}{\bar{\mu}}$$

Assuming that the average quantities in above equation are averaged over properties of inlet and outlet, the equation for mass flow rate becomes:

$$w = 405\frac{(\rho_0^2 - \rho_1^2)}{(\mu_0 - \mu_1)}$$

The properties of air can be approximated by linear equation as follows:

$$\rho = \frac{1}{2.5 \times 10^{-2} T}, \qquad \mu = \left(0.01135 + 0.6017 \times 10^{-4} T\right)$$

where T is in °Rankine.

Replacing the approximations for the density and viscosity into equation for mass flow rate, it follows:

$$w = \frac{3.24 \times 10^{5}\left(T_1^2 - T_0^2\right)}{\left(T_1^2 T_0^2\right)\left[0.01135 + 0.30085 \times 10^{-4}\left(T_1 + T_0\right)\right]}\left[\frac{lb}{hr}\right]$$

Once the mass flow rate throughout the channel is obtained, the heat transfer coefficient needs to be determined and the amount of heat removed by natural circulation of air through the core. The heat transfer coefficient is conservatively given with:

$$h = 0.532 \frac{k_f}{L}\left(Gr \cdot \Pr\right)^{0.25}$$

where k_f is the thermal conductivity of film air at temperature T_f, Gr is Grashof number, and Pr is the Prandtl number. The thermal conductivity coefficient and specific heat of the air is:

$$k = \left(0.0009 + 0.26 \times 10^{-4} T\right)$$

and

$$C_p = 0.240 \frac{Btu}{lb - F}$$

Using the above expressions and the heat transfer coefficients, along with expressions for extrapolated density and viscosity presented earlier:

$$h = 50.16 \left[\begin{matrix} \left\{0.0009 + 0.13 \times 10^{-4}\left(T_w + T_a\right)\right\}^3 \times \\ \left\{ \frac{\left[1.25 \times 10^{-2}\left(T_W + T_a\right)\right]^{-2}\left[0.5\left(T_W + T_a\right)\right]}{\left[0.01135 + 0.30085 \times 10^{-4}\left(T_w + T\right)\right]} \right\}^{-1} \times \left(T_w - T\right) \end{matrix} \right]^{0.25}$$

where T_w is the wall temperature, T_a is the bulk air temperature, and T_f is the average of the two. Since the yield stress for SS is relatively much higher than aluminum, the calculations are done for aluminum cladding only. The pressure produced is negligible compared to the yield stress limit for aluminum. The specific heat and the thermal conductivity of the $UZrH_{1.0}$ is given by the following expressions:

$$C_p = 26.3 + 0.0245T$$
$$k_f = 10.7 - 6.42 \times 10^{-4} T$$

The pressure exerted on the cladding is determined as follows: the pressure from fission product accumulation, hydrogen release, and air trapped inside the clad is determined. The total number of fission product nuclei released to the gap between the fuel and the cladding is determined based on: J. O. Blomeke and Mary F. Todd (1958), "U-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time", ORNL-2127 . The following calculation is for the 100kW UUTR:

$$N_i = 0.028 \times 2.748 \times 10^{20} = 7.72 \times 10^{18} \text{ atoms}$$

The number of gram-atoms in the gap is:

$$n_{fp} = \frac{7.72 \times 10^{18}}{6.02 \times 10^{23}} = 1.282 \times 10^{-5} \text{ gram-atoms}$$

The partial pressure exerted by the fission products gases is:

$$P_{fp} = n_{fp} \frac{RT}{V}$$
$$V = 3.23 cm^3$$

Therefore, the initial pressure exerted by all the fission product gases is:

$$P_{fp} = \frac{1.282 \times 10^{-5}}{3.23} RT = 0.397 \times 10^{-5} RT$$

The partial pressure of the air in the fuel element is:

$$P_{air} = \frac{RT}{22.4 \times 10^3} = 4.46 \times 10^{-5} RT$$

The total pressure exerted by the air and fission products is:

$$P_r = \left(1 + \frac{P_{fp}}{P_{air}}\right) P_{air} = 1.09 P_{air}$$

The pressure exerted by Hydrogen is negligible for the temperature range that is considered (**Fig. 4.2-3**). Therefore, the total gas pressure at the maximum fuel temperature of 61 °C (334.15 °K) is:

$$P = P_H + P_r = 0 + 1.09(14.7)\left(\frac{61+273}{273}\right)\frac{3.23}{V} = \frac{63.3}{V}psi$$

The volume increase due to expansion of clad is ignored and the total pressure is then:

$$P = \frac{63.3}{V} = \frac{63.3}{3.23} = 19.59psi$$

The tangential stress in the fuel element cladding is then calculated as follows (more details can be found in **Section 4.2.1.2**):

$$S = \frac{P \cdot r}{t}$$

where t is the wall thickness (0.076 cm) and r is the fuel element can radius (1.8 cm), therefore S = 464.2 psi for 100kW power.

According to the Aluminum Company of America handbook [Aluminum Company of America, "Alcoa Aluminum Handbook," (1962)], the yield stress for type 6061 aluminum at 152 °C (425.15 °K) is greater than 8,000 psi. Therefore, the above calculations showed that following the loss of coolant in the tank, the rupture of cladding will not occur. The parameters of interest are summarized in **Table 13.2-14**.

Table 13.2-14 Parameters of interest for pressure inside the fuel cladding in the UUTR

Parameter	**Pressure**
Maximum fuel temperature, °C (°K)	61 (334.15)
Number of fission product atoms	7.720×10^{18}
Fission product gram-atoms	1.282×10^{-5}
Partial pressure exerted by the fission products (psi)	3.970×10^{-6}
Total pressure (psi)	19.6
Tangential stress (psi)	464.2

13.2.4 Radiation Levels After Loss of Pool Water

The dose rate on top of the UUTR core is calculated assuming that the core has been operating at 100kW for a long time (1,000 hours) before the accident. As mentioned in previous section, the water inside the tank will not drain out at once; therefore, the scenario that is analyzed assumes the pool water leaks at the rate of 0.4 m per hour. In summary the scenario used used in the model for the dose estimate is:

- reactor has been operating for 1,000 hours before the accident
- reactor is shutdown after the leak of pool water starts
- reactor core is approximated as a point source (emitting photons isotropically)
- average photon energy is 1.0 MeV
- attenuation by core components other than fuel elements is neglected

Each fuel element has 400 cm^3 volume; for the core with 78 fuel elements the whole core volume is 31,200 cm^3 which is equivalent to a sphere of a radius 19.53 cm. The fraction of photons escaping from the core is given as a function of sphere diameter (r_c) and the attenuation coefficient (μ_c) of the core material for 1 MeV photons (= 0.207 cm^{-1}) which is 0.197 [K. K. Aglintsev, Applied Dosimetry, (Eng. Ed.), Iliffe, London (1965)]. The calculated source strengths are from SAR 1985 for 100kW UUTR, **Table 13.2-15**.

The dose rate is calculated using the following correlation with assumption that the core is a point source:

$$D = \frac{S_{eff}}{4\pi R^2 K} B(R\mu) e^{-\mu x}$$

where

D: the dose rate in rads/hr
S_{eff}: effective source strength (γ-rays/sec)
R: distance between core and location considered (cm)
K: conversion factor for MeV/cm^2 – sec to rad/hr (= 5.77 x 10^5 for 1 MeV photons)
B: Build-up factors [N. Tsoulfanidis, (1995). "Measurement and Detection of Radiation". 2^{nd} edition, Taylor & Francis]
μ: linear attenuation coefficient (water = 0.07072)
x: thickness of material, cm.

The build-up factors are calculated as follows [a, b, and μ from [N. Tsoulfanidis, (1995). "Measurement and Detection of Radiation". 2^{nd} edition, Taylor & Francis]:

$$B = 1 + a\mu r e^{b\mu r}$$

The dose rate at the top of the core is shown in **Table 13.2-16**. The dose rate becomes extremely high when the water inside the tank drains out (completely). However, within first 5 hours, the dose rate is low enough (10 CFR 20.1201) for reactor supervisor and personnel to perform safety procedures such as to secure the reactor, provide emergency water to the reactor water tank and secure the reactor area.

Table 13.2-15 Source strength for 100kW UUTR [1985 UUTR SAR]

Time (hr)	Source Strength (γ-rays/sec)	Effective Source Strength (γ-rays/sec)
0.1	6.49×10^{15}	1.28×10^{15}
0.5	4.52×10^{15}	8.90×10^{14}
1	3.60×10^{15}	7.09×10^{14}
5	2.20×10^{15}	4.33×10^{14}
10	1.70×10^{15}	3.35×10^{14}
24	1.20×10^{15}	2.36×10^{14}

The dose rates shown in **Table 13.2-16** relates to a top of the UUTR tank. In the 1985 Safety Analysis Report for UUTR the dose rates are provided for the laboratory floor. After 15.2 hr from the accident, the water level is at the top of the core but the core is still surrounded with water. In other words, the core is submerged in water; however, the height of the water in the tank is just up to the top of the reactor core. The tank with two layers of aluminum and stainless steel, with 0.61 meters of sand around the tank will reduce the dose rate at the laboratory floor. In order to have the better conservative time frame for reactor personnel to work in the area after the accident happens the dose rate on top of the tank is also calculated; these dose rates will affect the personnel in the control room.

Table 13.2-16 Dose rate calculated on a top of the core for various water levels

Time (hr)	Water Level (m)	Dose (mrem/hr)	
		Top of the Core	Laboratory Floor[2]
0.1	6.1	2.68×10^{-11}	1.00×10^{-10}
1	5.6	6.14×10^{-10}	1.00×10^{-10}
5	4.1	1.85×10^{-5}	2.10×10^{-10}
10	2.1	1.88×10^{1}	5.10×10^{-2}
15.2	0.0	4.76×10^{8}	8.80×10^{2}

[2] These doses refer to the radiation levels on the reactor room floor obtained from Safety Analysis Report of UUTR (1985).

13.2.5 Loss of Coolant Flow

UUTR utilizes natural convection cooling. Therefore, there is no primary cooling system that could fail to cause loss of coolant flow accident. On the other hand, the loss of secondary system coolant flow which keeps the water inside of the tank cool, will not introduce an accident. The reactor can easily operate without pool cooling, but with reduced number of hours. A siphon break is designed to prevent the drainage of water from the tank through primary coolant line.

13.2.6 Mishandling or Malfunction of Fuel

Accident initiating events and scenarios. Events which could cause accidents at the UUTR in this category include:

- Fuel handling accidents where an irradiated or new element is dropped underwater and damaged severely enough to breach the cladding
- Simple failure of the fuel cladding due to a manufacturing defect or corrosion
- Overheating of the fuel with subsequent cladding failure
- Fuel handling and movement will follow the procedure that is described in the **UUTR SAR 9.2.1 and 9.2.3.**
- If there is any possibility of leak from the dropped fuel element, the leak test shall be performed according to the procedure described in the **UUTR SAR 9.2.6.1.**

Accident analysis and determination of consequences. All these three scenarios result in a single fuel element failure in water. In the unlikely event that this failure occurred in air, this is the MHA analyzed in **UUTR SAR 13.2.1.** In past the UUTR fuel elements are moved to new positions or removed from the core only during periods when the reactor is subcritical.

Assumptions for this accident are almost exactly the same as those used for the MHA, except that the presence of the pool water contains most of the halogens and, thereby, reduces the halogen dose contribution. This accident analyzed for similar TRIGA facilities show that the general public are well below the annual limits in 10 CFR Part 20, with the maximum dose being less than 33 mrem TEDE for 500kW TRIGA. The occupational radiation doses to workers in the reactor are are also well below the occupational annual limits as defined in 10 CFR Part 20, with the maximum dose below 100 mrem TEDE for a 5-minute exposure. Five minutes is very ample time for workers to evacuate the reactor area if such an accident were to occur.

13.2.7 Experiment Malfunction

The **UUTR TS** section 3.1 and 3.2 specify the reactivity limits to prevent a serious reactor accident. The main limits are:

1. The shutdown margin referred to the cold-critical xenon-free condition, with the highest worth rod (safety rod) fully withdrawn, is greater than $0.50.

2. The rate of reactivity insertion by control rod motion shall not excess $0.30.

3. Any experiment with the reactivity worth greater than $1.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor.

4. The excess reactivity for the cold critical, xenon free condition is less than $1.20.

5. The reactivity worth of an individual experimental is not more than $1.20.
6. Fueled experiments are limited such that the total inventories of I-131 through I-135 in the experiments are not greater than 10 mCi.

7. The quantity of known explosive materials to be irradiated is less than 25 milligrams.

8. Experiments containing materials corrosive to reactor components, compound highly reactive with water, potentially explosive materials, or liquid fissionable materials are doubly encapsulated and able to withstand any overpressure condition deemed likely to occur.

9. The concentration of Argon-41 released from the facility to the environment shall not exceed 10^{-8} $\mu Ci/cm^3$ averaged over 1 year (**UUTR TS 3.7.2**).

These limiting conditions placed on experiments and irradiations are based on following considerations:

1. The first specification is based on that the shutdown margin required by specification is necessary so that the reactor can be shutdown from any operating condition and remain shutdown after cool down and xenon decay even if one control rod should remain in the fully withdrawn position.

2. The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit (Fuel temperature).

3. Specification 8 is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

4. Specification 9 is intended to ensure that the health and safety of the public are not endangered by discharge of Argon-41 from the UUTR facility.

13.2.8 Loss of Normal Electrical Power

Loss of electrical power to the UUTR could occur due to many events and scenarios that routinely affect commercial power. Since the UUTR does not require emergency backup systems to safely maintain core cooling, there are no credible UUTR accidents associated with the loss of electrical power. A backup power system present at the UUTR mainly provides conditioned power to the instrumentation. The system provides emergency power immediately after the loss of regular electrical power and continues to supply power for a period of several hours. Battery-powered emergency lights are also located throughout the facility to allow for inspection of the reactor and for an orderly evacuation of the facility. Loss of normal electrical power to the UUTR facility during reactor operations will initiate a reactor scram. Loss of power is addressed in the UUTR Procedures, which require that, upon loss of normal power, the operator on duty should verify the reactor is shutdown. This can be done without any electrically powered indications. The backup power supply would allow enhanced monitoring.

13.2.9 External Events

Hurricanes, tornadoes, and floods are virtually nonexistent in the area around the UUTR. Therefore, these events are not considered to be viable causes of accidents for the reactor facility.

Recent history indicates that the region encompassing the state of Utah and adjacent areas are zones of moderate seismic activity and minor consequence. In recent history, seismic occurrences have been low intensity events with little or no resulting damage. Should an earthquake occur of significant severity, the consequences to the UUTR facility should not cause events more severe than the MHA. Since the reactor pool was constructed of a large mass of reinforced concrete, it will tend to vibrate as a single unit with a frequency of oscillation that may be different from that of the surrounding building. In addition, the TRIGA tank was built in an earthquake proof fashion with double wall and sand construction. Dislocation between the reactor and surrounding structures could occur. This could lead to a break in beam port tubes and other penetrations, which can cause the pool water to drain.

This non-instantaneous loss of coolant/shielding event was analyzed previously. An earthquake of sufficient severity to cause dislocation of the reactor pool or surrounding structures would undoubtedly be recognized and the reactor, if operating at the time, could be manually shutdown and not restarted until the integrity of the reactor pool had been established. The most severe consequence of a major earthquake would be the failure of the reactor's aluminum tank. Because of the containment design employing a double wall, sand, and concrete construction, the only credible damage scenario would result in a slow (**UUTR SAR 13.2.2**) draining of the pool water. This condition would be easily recognizable allowing sufficient time to shut down and secure the reactor, evacuate personnel, and consider possible mitigation actions for minimizing radiation release and exposures. Major seismic events would probably cause other related events, which would cause a reactor shutdown without operator intervention. For example, loss of building power would initiate a shim-safety rod insertion, since the control system is fail-safe with respect to power loss. Also, loss of significant quantities of pool water will cause a reactor trip from low water level in the pool.

Therefore, there are no accidents in this category that would have more on-site or off-site consequences than the MHA analyzed in **UUTR SAR 13.2.2**, and, therefore, no additional specific accidents are analyzed in this section.

13.2.10 Mishandling or Malfunction of Equipment

No credible accident initiating events were identified for this accident class. Situations involving an operator error at the reactor controls, a malfunction or loss of safety-related instruments or controls, and an electrical fault in the control rod system were anticipated at the reactor design stage. As a result, many safety features, such as control system interlocks and automatic reactor shutdown circuits, were designed into the overall TRIGA Control System. TRIGA fuel also incorporates a number of safety features (**UUTR SAR 4**) which, together with the features designed into the control system, assure safe reactor response, including in some cases reactor shutdown. No safety considerations at the UUTR depend on confinement or containment systems. Rapid leaks of the coolant have been addressed in **UUTR SAR 13.2.3** showing that no damage to the reactor occurs as a result of these leaks. Since there were no credible initiating events identified, no accident analysis was performed for this section and no consequences were identified.

14. TECHNICAL SPECIFICATIONS

FACILITY LICENSE R-126

TECHNICAL SPECIFICATIONS
AND BASES
FOR THE UNIVERSITY OF UTAH
TRIGA REACTOR

DOCKET 50-407

TECHNICAL SPECIFICATIONS AND BASES FOR THE UNIVERSITY OF UTAH TRIGA NUCLEAR REACTOR

This document constitutes the Technical Specifications for the Facility License No. R-126 and supersedes all prior Technical Specifications. This document includes the "Basis" to support the selection and significance of the specifications. The bases are included for information purposes only. They are not part of the technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1. Definitions

1.1 Audit: An audit is a quantitative examination of records, procedures or other documents after implementation from which appropriate recommendations are made.

1.2 Channel: A channel is the combination of sensor, line, amplifier, and output devices, which are connected for the purpose of measuring the value of a parameter.

1.3 Channel Calibration: A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter, which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a Channel Test.

1.4 Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5 Channel Test: A channel test is the introduction of a signal into the channel for verification that it is operable.

1.6 Confinement: Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways. These are rooms MEB 1205 (A through K) and 1206 in Merrill Engineering Building.

1.7 Control Rod: A control rod is a device fabricated from neutron absorbing material which is used to establish neutron flux changes and to compensate for routine reactivity changes. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. Types of control rods shall include:

1. **Regulating Rod (Reg Rod):** The regulating rod is a control rod having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.
2. **Shim/Safety Rod:** A shim safety rod is a control rod having an electric motor drive and scram capabilities.

1.8 Core Lattice Position: The core lattice position is defined by a particular hole in the top grid plate of the core. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

1.9 Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions.

1.10 Experiment: Any operation, hardware, or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

1. **Secured Experiment:** A secured experiment is any experiment or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces, which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.
2. **Unsecured Experiment**: An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.
3. **Movable Experiment:** A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the core while the reactor is operating.

1.11 Experimental Facilities: Experimental facilities shall mean vertical in-pool irradiation facilities, vertical tubes, in-core irradiation ports such as the A fuel ring (central ring) or other empty fuel element positions, rotating specimen rack, pneumatic transfer system, sample holding dummy fuel elements and any other in-tank irradiation facilities.

1.12 Fuel Element: A fuel element is a single TRIGA® fuel element.

1.13 Instrumented Element: An instrumented element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

1.14 Irradiation: Irradiation shall mean the insertion of any device or material that is not a part of the existing core or experimental facilities into an experimental facility so that the device or material is exposed to radiation available in that experimental facility.

1.15 Measured Value: The measured value is the value of a parameter as it appears on the output of a channel.

1.16 Operable: A system or component shall be considered operable when it is capable of performing its intended function.

1.17 Operating: Operating means a component or system is performing its intended function.

1.18 Operational Core: An operational core shall be a fuel element core which operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

1.19 Pulse Mode: Not applicable for the UUTR.

1.20 Reactivity Worth of an Experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment, being inserted into or removed from its intended position.

1.21 Reactor Operating: The reactor is operating whenever it is not secured or shut down.

1.22 Reactor Operator (RO): An individual who is licensed to manipulate the controls of a reactor.

1.23 Reactor Safety Systems: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

1.24 Reactor Secured: The reactor is secured when:

1. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material in the reactor to attain criticality under optimum available conditions of moderation and reflection; or,
2. All of the following exist:
   2.1 The three (3) neutron absorbing control rods are fully inserted as required by technical specifications;
   2.2 The console key switch is in the "off" position and the key is removed from the console.
   2.3 No experiments are being moved or serviced that have, on movement reactivity worth exceeding the maximum value allowed for a single experiment, or of one dollar, whichever is smaller.
   2.4 No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

1.25 Reactor Shutdown: The reactor is shut down when it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions, with the reactivity worth of all installed experiments and irradiation facilities included.

1.26 Reference Core Condition: The reference core condition is the condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$ 0.30).

1.27 Review: A review is a qualitative examination of records, procedures or other documents prior to implementation from which appropriate recommendations are made.

1.28 Safety Channel: A safety channel is a measuring channel in the reactor safety system.

1.29 Scram time: Scram time is the elapsed time from the initiation of a scram signal to the time the slowest scrammable control rod is fully inserted.

1.30 Senior Reactor Operator (SRO): An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.31 Should, Shall, and May: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

1.32 Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems and will remain subcritical without further operator action, starting from any permissible operating condition with the most reactive rod is in its most reactive position.

1.33 Surveillance Intervals: Allowable surveillance intervals shall not exceed the following:

1. Biennial - interval not to exceed 30 months
2. Annual - interval not to exceed 15 months
3. Semi-annual - interval not to exceed 7.5 months
4. Quarterly - interval not to exceed 4 months
5. Monthly - interval not to exceed 6 weeks
6. Weekly - interval not to exceed 10 days

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit – Fuel Element Temperature

Applicability

This specification applies to the maximum temperature of the reactor fuel.

Objective:

The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications

1. The temperature in a stainless-steel clad, high hydride fuel element shall not exceed 1,000 °C (1,273.15 °K) under any conditions of operation, and
2. The temperature in an aluminum clad low hydride fuel element shall not exceed 500 °C (773.15 °K) under any conditions of operation (NUREG 1537 Part 1, Appendix 14.1)

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding, if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium ratio in the alloy.

The safety limit for the high hydride TRIGA fuel is based on data, including the experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1,150 °C (1,423.15 °K) and the fuel cladding is water cooled.

The safety limit for the low hydride fuel elements is based on avoiding the phase change in the zirconium hydride, which might cause excessive distortion of a fuel element. This phase change takes place at 530 °C (803.15 °K). Additional information is given in Technical Foundations of the TRIGA Report GA-471, pages 63-72, August 1958. It has been shown by experience that operation of TRIGA reactors at power level of 1,500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for years at power level of up to 1,500 kW. Analysis and measurements have shown that a power level of 1,500 kW corresponds to a peak fuel temperature of 600 °C (~873.15 °K). Therefore, establishing the Safety Limit at 500 °C ensures that the fuel integrity is maintained.

More details are provided in **UUTR SAR 4.2.1** and **4.5.3.1.**

2.2 Limiting Safety System Settings

Applicability

This specification applies to the settings that prevent the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being exceeded.

Specifications

1. i) For a core composed entirely of stainless steel cladding fuel, and
   ii) high hydride fuel elements with low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in the following table:

Location of Instrumented Fuel Element	Limiting Safety System Setting for SS Cladding
B-hexagonal ring	800 °C (1,073.15 °K)
C-hexagonal ring	755 °C (1,028.15 °K)
D-hexagonal ring	680 °C (953.15 °K)
E-hexagonal ring	580 °C (853.15 °K)

2. For a core with low hydride fuel elements installed in other than the F or G hexagonal ring, limiting safety system settings apply according to the location of the instrumented fuel element, as indicated in the following table:

Location of Instrumented Fuel Element	Limiting Safety System Setting for Al Cladding
B-hexagonal ring	460 °C (733.15 °K)
C-hexagonal ring	435 °C (700.15 °K)
D-hexagonal ring	390 °C (663.15 °K)
E-hexagonal ring	340 °C (613.15 °K)

Basis

The UUTR is equipped with two independent instrumented fuel elements that monitor the fuel temperature in the core. The fuel temperature is displayed on the reactor console. Exceeding the set point will initiate a SCRAM. For a core composed entirely of stainless steel cladding, high hydride fuel elements or a core composed of aluminum cladding, low hydride fuel elements in the F or G hexagonal ring only, limiting safety system settings apply according to the location of the instrumented fuel as indicated in **UUTR SAR Table 7.2-1** or this TS requirement. The fuel temperature monitoring channels consists of a K type thermocouple and an Omega CN9000A temperature controller. The useful range is 0 °C (273.15 °K) to 800 °C (1,073.15 °K) with a ±1 °K accuracy. Fuel temperature set-points for the SCRAM function are set at 200 °C (473.15 °K) for 100 kW operation.

From the experience of running the UUTR at 90kW power, the instrumented fuel elements indicate that the fuel element temperature for the C-ring and D-ring are 110 °C (~383.15 °K) and 95 °C (368.15 °K) respectively (measured temperature). The fuel temperature decreases toward the outer fuel rings. Therefore, the fuel temperature in the E-ring will not exceed the safety limit of 460 °C. According to the PARET-ANL calculation, the maximum centerline fuel temperature for the UUTR at 100kW power is 129.67 °C (402.82 °K) (**UUTR SAR 4.6, Table 4.6-1**). The **UUTR TS** does not limit the presence of the aluminum elements in the B-ring, but for mixed cores with the aluminum cladding elements in one of the inner rings (B through E-ring), the maximum (B-ring) limiting safety system temperature setting is 460 °C.

During the steady state operation at 100 kW, temperatures were calculated for the beginning- of-life reference UUTR core. Linear extrapolation of temperature and power indicates that an instrumented fuel element power of 1.895 kW will produce 121.7 °C (394.85 °K) in the instrumented fuel element at the midplane thermocouple location **(UUTR SAR 4.2.1.4, Table 4.2-3)**. The highest ratio of maximum to minimum power for elements in the C-ring was calculated to be 1.266 **(UUTR SAR 4.5.2.3, Table 4.5-6)**, so if the instrumented fuel element is generating 1.895 kW, the maximum power in any C-ring element would be limited to 1.895 x 1.266 = 2.399 kW. For a power of 2.399 kW, the maximum temperature anywhere in the C-ring fuel element will be 154.1 °C (427.25 °K). These values are well below the safety limits.

3. LIMITING CONDITIONS FOR OPERATION (LCO)

3.1 Reactor Core Parameters

3.1.1 Steady-State Operation

Applicability

This specification applies to the energy generated in the reactor during steady-state operation.

Objective

The objective is to assure that the fuel temperature safety limit shall not be exceeded during steady-state operation.

Specifications

The reactor power level shall not exceed 100 kW.

Basis

Thermal-hydraulics calculations and design analysis are described in **UUTR SAR Chapter 4.6** addressing the fuel temperature limits during steady-state operation of the UUTR. More specifically, these calculations are described in **UUTR SAR 4.6.1 and 4.6.3**.

3.1.2 Shutdown Margin

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. They apply for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit shall not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist: The shutdown margin provided by control rods shall be greater than $0.50 with:

1. The irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state,
2. The most reactive control rod fully-withdrawn, and
3. The reactor in the reference core condition

Basis

The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the most reactive control rod (which is a safety control rod) should remain in the fully-withdrawn position **(UUTR SAR 4.2, 4.5.3.9)**.

3.1.3 Core Excess Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. It applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit shall not be exceeded.

Specifications

The maximum available excess reactivity based on the reference core configuration shall not exceed $1.20.

Basis

If operating the UUTR with the minimum shutdown margin of $0.50 (**UUTR TS 3.1.2**) and the calculated control rod worths of $1.924 (Safety), $1.468 (Shim), and $0.294 (Regulating) from MCNP5, (**UUTR SAR Section 4.5.2.3, Table 4.5-5**), the calculated core excess reactivity is $1.468+$0.294-$0.50 = $1.262. This assumes:

a) irradiation facilities and experiments are in place, and
b) the most reactive control rod (safety control rod) is fully-withdrawn.

Changing the core configuration, or adding negative worth experiments will make core excess reactivity more negative and shutdown margin less positive. The only activity which could result in requiring fuel movement to meet shutdown margin and core excess limit would be the unusual activity of adding an experiment with large positive reactivity worth.

3.1.4 Core Configuration

Applicability

This specification applies to the configuration of fuel elements and in-core experiments.

Objective

The objective is to assure the provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

1. The reactor core shall be an arrangement of TRIGA LEU cylindrical stainless-steel-cladding U-$ZrH_{1.6}$ fuel-moderator elements (SS) and aluminum-cladding U-$ZrH_{1.0}$ fuel-moderator elements with neutron reflectors provided by up to 12 graphite and 12 heavy water elements in aluminum cladding,
2. The reflector, excluding experiments and experimental facilities, shall be a combination of water, graphite and heavy water,
3. Fuel shall not be removed from or inserted into the core unless the reactor is subcritical by more than calculated worth of the most reactive fuel element, and
4. Control rods shall not be removed manually from the core unless the core has been shown to be subcritical with all control rods fully withdrawn from the core.

Basis

1. The UUTR utilizes solid fuel elements, developed by General Atomics (GA), in which the zirconium-hydride moderator is homogeneously combined with enriched uranium. The unique feature of these fuel-moderator elements is the prompt temperature coefficient of reactivity, which gives the TRIGA reactor its built-in safety by automatically limiting the reactor power to a safe level in the event of a power excursion. The UUTR reactor core consists of a lattice of cylindrical stainless-steel-cladding U-$ZrH_{1.6}$ fuel-moderator elements (SS), and aluminum-cladding U-$ZrH_{1.0}$ fuel-moderator elements. Neutron reflection in the radial direction is provided by 12 graphite and 12 heavy water elements in an aluminum cladding. Also the core is emerged in water tank, which acts as a thermal shield and a moderator. The core components are contained between top and bottom aluminum grid plates. The top grid plate has 126 positions for fuel elements and control rods arranged in 6 concentric rings around a central port (used for high flux irradiations). More details are provided in **UUTR SAR 4.2** and **UUTR SAR 4.5**.
2. The core will be assembled in the reactor grid plate located at the bottom of

tank filled with light water. Light water of the tank, in combination with graphite and heavy water reflector elements can be used for neutron economy and to enhance requirements for experimental facilities.

3. Manual manipulation of fuel elements will be allowed only when single fuel element manipulation cannot result in an inadvertent criticality.

4. Manual movement of control rods will be allowed only when single manipulation cannot result in an inadvertent criticality.

3.1.5 Reactivity Coefficients

Does not apply to UUTR.

3.1.6 Fuel Parameters

Applicability

This specification applies to all fuel elements.

Objective

The objective is to maintain integrity of the fuel element cladding.

Specifications

The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:

1. The transverse bend exceeds 0.0625 inches over the length of the cladding,
2. Its length exceeds its original length by 0.125 inches,
3. A cladding defect exists as indicated by release of fission products, or
4. Visual inspection identifies bulges, gross pitting, or corrosion, and
5. Fuel burnup of Uranium-235 in the UZrH fuel matrix exceeds 50% of the initial content.

Basis

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

3.2 Reactor Control and Safety System

3.2.1 Control Rods

Applicability

This specification applies to the function of the control rods.

Objective

The objective is to determine that the control rods are operable.

Specification

The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if:

1. Damage is apparent to the rod or rod drive assemblies; or
2. The scram time exceeds 2 seconds; or
3. The rate of reactivity insertion by control rod motion shall exceed $0.30 per second.

Basis

This specification assures that the reactor shall be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor. The speed of the rods is adjustable and rods are normally set to insert or withdraw at a nominal rate of 0.494 cm/sec **(UUTR SAR 4.2)**. The control rods are designed to safely change the reactor power and/or shut the reactor down. According to the transient analysis of an uncontrolled rods withdrawal (**UUTR SAR 4.5.3.10**), the peak reactivity insertion is $0.72 that is significantly lower than **UUTR TS** requirement.

All three-control rods have the same rise time. Therefore, the obtained $0.046/second is much lower than this TS requirement of $0.30/sec. The shutdown margin must be greater than $0.50 [**UUTR TS 3.1.2**] and the excess reactivity must be less than $1.20 [**UUTR TS 3.1.3**].

The scram time shall not exceed 2 seconds measured from the time when one of the scram set points is exceeded (power level or fuel temperature) to time when the slowest scrammable control rod (which is a safety control rod) is fully inserted into the core. The scram time specification is satisfied when the sum of the response times of the slowest responding safety channel (that could be either power level or fuel temperature exceeding the corresponding limits), plus the fall time of the slowest scrammable control rod (which is a safety control rod), is less than or equal to 2 seconds.

3.2.2 Reactor Measuring Channels

Applicability

This specification applies to the information, which shall be available to the reactor operator during reactor operation.

Objective

The objective is to specify the minimum number of measuring channels that shall be available to the operator to assure safe operation of the reactor.

Specifications

The reactor shall not be operated in the specified mode unless the minimum number of measuring channels listed in this table are operable:

Measuring Channel	Minimum Number Operable
Start-up Count Rate	1
Fuel element temperature	1
Linear power level	1
Percent power level	1

Basis

Start-up Count Rate: The neutron count rate in the UUTR core must be greater than 2 cps for the reactor to be operable [**UUTR SAR 4.2.4**].

Fuel element temperature: Fuel element temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit.

Linear and percent power level: The linear and percent power level monitors assure that the reactor power level is adequately monitored during the reactor operation.

3.2.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to specify the minimum number of reactor safety system channels that shall be available to the operator to assure safe operation of the reactor.

Specifications

The reactor shall not be operated unless the minimum number of safety channels described in this table are operable[1].

Table 1. Minimum reactor safety channels

Safety channel	Minimum Number Operable	Function
Fuel element temperature	1	Scram at 200 °C (473.15 °K)
Linear power level	1	Scram at 100 kW
Percent power level[2]	1	Scram at 110% of full licensed power
Manual Console scram	1	Manual scram
Magnet current key switch	1	Manual scram
Console power supply	1	Scram on loss of electrical power
Reactor tank water level	1	Scram at 15.5 inches below the top of the UUTR tank

Table 2. Minimum interlocks

Safety System Interlock	Minimum Number Operable	Function
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second
Control rod withdrawal interlocks	All control rods	Prevent manual withdrawal of more than one control rod simultaneously

[1] If any required safety channel or interlock becomes inoperable while the reactor is operating for reasons other than that identified below, the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

[2] Any single linear or percent power level channel or interlock may be inoperable while the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration.

Basis

Safety System Measuring Channel

Fuel element temperature scram:

The fuel element temperature scram is set to one fifth of the LSSS for the stainless steel cladding high hydride fuel element located in the B-hexagonal ring, which is 200 °C (473.15 °K) (**UUTR SAR 7.2.3.2**). This is more than adequate to account for uncertainties in instrument response and core position of the instrumented fuel element. Additional information regarding the safe fuel temperature limits are provided in **UUTR SAR 4.6**.

Power level scram:

Linear power channel scram is at 100kW, and percent power channel scram is at 110% of full licensed power. Therefore, the UUTR is operated at around 90kW without scramming the reactor unnecessarily. The difference in around 10kW allows for expected and observed instrument fluctuations at the normal full operating power at 90kW. Conversely, **UUTR SAR 13.2.2.1** shows that this set point is more than sufficient to prevent exceeding the reactivity insertion limit during normal operation and prevent the operator from inadvertently exceeding the licensed power.

Manual console scram:

The manual scram must be functional at all times the reactor is in operation. It has no specified value for a scram set point. It is initiated by the reactor operator manually.

Magnet current key switch:

The reactor key must be in the key hole and "on" position during the reactor operation. If the reactor is key removed from the key hole and moved to "off" position, the magnet current in the control system will be interrupted and all three control rods will be dropped in to the core. The reactor will be scrammed.

Console power supply:

If the reactor console loses the electrical power, the reactor will be scrammed even if the UPS (Uninterrupted Power System) is installed.

Reactor tank water level:

The UUTR pool water must have a specific water level to avoid the reactor scram. The distance from the top of the reactor tank to the surface of the pool water must be less than 15.5 inches. If the pool water level decreases, the distance will be increased inducing the scram.

Safety System Interlock

Startup count rate interlock:

The control rod withdrawal interlock prevents the operator from adding reactivity when the startup count rate falls below 2 cps. When this happens, the count rate is insufficient to produce meaningful instrumentation response. If the operator were to insert reactivity under this condition, the period could quickly become very short resulting in an inadvertent power excursion. Then, a neutron source is added to the core to create sufficient instrument response that the operator can recognize and respond to changing conditions.

Control rod withdrawal interlocks:

The single rod withdrawal interlock prevents the operator from removing multiple control rods simultaneously assuring that reactivity insertions from control rod manipulation is controlled. The analysis presented in **UUTR SAR 13.2.2 and UUTR SAR 4.2.2** show that the reactivity insertion due to the removal rate of the most reactive control rod (which is the safety control rod), or all the control rods simultaneously, is still well below the reactivity insertion design limit of $0.30/sec.

3.3 Coolant System

Applicability

This specification applies to the primary water of the reactor tank.

Objective

The objective is to assure that there is an adequate amount of water in the reactor tank for fuel cooling and shielding purposes, and that the bulk temperature of the reactor tank water remains sufficiently low to guarantee reactor tank integrity.

Specifications

The reactor primary water shall exhibit the following parameters:

1. The reactor tank water level alarm shall indicate loss of coolant if the tank water level drops 15.5 inches from the top of the UUTR water tank,
2. The reactor tank water temperature shall be less than 35 ^{o}C (308.15 ^{o}K),
3. The conductivity of the reactor tank water shall be less than 5 μmhos/cm,
4. The pH shall be between 5.5 and 7.5, and
5. The reactor shall not be operated if the radioactivity of reactor pool water exceeds the limits of 10 CFR 20 Appendix B Table 3 for radioisotopes with half-lives > 24 hours.

Basis

1. The distance from the top of the reactor tank to the surface of the pool water must be less than 15.5 inches for reactor to be operable without setting the alarm (**UUTR SAR 5.2**). The alarm will be sound if the tank water level drops below 15.5 inches measured from the top of the tank. This alarm sound is observed in the control room and reactor room as well as the campus dispatcher is notified automatically. A corrective action is specified for the low water level alarm.
2. The *bulk water temperature* limit is necessary, according to the reactor manufacturer, to ensure that the aluminum reactor tank maintains its integrity and is not degraded (**UUTR SAR 4.3**). This was input for the reactor pool water temperature analysis (**SAR Ch. 4.6.3**).
3. Experience at many research reactor facilities has shown that maintaining the *conductivity* within the specified limit provides acceptable control of corrosion (NUREG-1537). More information regarding the UUTR is provided in **UUTR SAR 5.2.3**.
4. The pH of reactor tank water is kept between 5.5 and 7.5 (**UUTR SAR 5.3**) assuring the water is kept chemically neutral.
5. A monthly checkout of the reactor tank water is performed using a high purity gamma spectroscopy system. Typical survey shows that the total activity of the reactor tank water ranges between 0.3 to 0.7 nCi/l (=7.0x10^{-7}

μCi/ml). This amount of activity is substantially lower than 10 CFR 20 Appendix B (Table 3) requirement. Analyses using a high purity gamma spectroscopy system show that limiting the activity to this level will not result in any person being exposed to concentrations greater than those permitted by 10 CFR Part 20.

3.4 Confinement

Applicability

These specifications apply to the area housing the reactor and the ventilation system controlling that area.

Objective

The objective is to provide restrictions on radioactive airborne materials releases into environment.

Specifications

1. Confinement is required for reactor operation and/or any movement of irradiated fuel, and
2. To achieve confinement, the ventilation system shall be operating in accordance with **UUTR TS 3.5**.

Basis

1. During reactor operation and/or any movement of irradiated fuel there is the potential for release of radioactivity from fuel elements. Confinement will limit the consequences to the public from such a release.
2. During reactor operation and/or any movement of irradiated fuel and/or fueled experiment the potential for release of radioactivity from fuel elements will be controlled by operating ventilation system in limiting the consequences to the public from such a release.

3.5 Ventilation System

Applicability

This specification applies to the operation of the reactor area ventilation system.

Objective

The objective is to assure that the ventilation system shall be in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

Specifications

The reactor shall not be operated unless the ventilation system is in fully operable mode. The ventilation system is considered to be fully operable when the following conditions exist:

1. The pressure difference between the reactor room and outside of the building (Merrill Engineering Building) should be larger than 0.1 inches-of-water.
2. In the event of a substantial release of airborne radioactivity within the reactor area, the ventilation system will be secured or operated in the limited intake mode to prevent the release of a significant quantity of airborne radioactivity from reactor area.

Basis

1. In the operational mode of the ventilation system, the air in the controlled access area (reactor room area) is constantly being exchanged. The air leaving the facility has a volumetric flow rate of more than 100 CFM per each of the two fume hoods. The result of this is a negative pressure of greater than 0.01 inches of water in the reactor room.
2. The worst-case maximum total effective dose equivalent is well below the applicable annual limit for individual members of the public and building residents during the maximum hypothetical accident (MHA). More details are provided in the **UUTR SAR 13.2.1**.

3.6 Emergency Power

Does not apply to UUTR.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

<u>Applicability</u>

This specification applies to the radiation monitoring information, which must be available to the reactor operator during reactor operation.

<u>Objective</u>

The objective is to specify the minimum radiation monitoring channels that shall be available to the operator to assure safe operation of the reactor.

<u>Specifications</u>

The reactor shall not be operated unless the <u>minimum number of radiation monitoring channels</u> are operating as in the accompanying table:

Radiation Monitoring Channels	**Number**
Reactor Room Area Radiation Monitor (ARM)	1
Continuous Air Monitor CAM (particulate, noble gas, and iodine)	1

<u>Basis</u>

The radiation monitors provide information to operating personnel regarding routine releases of radioactivity and any impending or existing danger from radiation. Their operation will provide sufficient time to evacuate the facility or take the necessary steps to prevent the spread of radioactivity to the surroundings. The calculations show that for routine operations and under the accident scenarios identified in **UUTR SAR 13.2.1.1**, predicted occupational and general public doses are below the applicable annual limits specified in 10 CFR 20 (**UUTR SAR 11.1.1.1** and **UUTR SAR 13.2.2**).

Area Radiation Monitor (ARM): The reactor will be scrammed if radiation level in reactor room is higher than 10 mrem/hr. Additionally, the radiation monitors provide information to operating personnel of an emergency or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings (**UUTR SAR 11.1.1.3**).

Continuous Air Monitor (CAM): The reactor can be operable for 48 hours without the CAM system (**UUTR SAR 5.6**).

3.7.2 Effluents

Applicability

This specification applies to the release rate of ^{41}Ar.

Objective

The objective is to ensure that the concentration of the ^{41}Ar in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specifications

The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 1 x10^{-8} μCi/ml at the point of discharge averaged over one year.

Basis

Based on the calculation as shown in the **UUTR SAR 11.1.1.1.5**, the equilibrium Argon-41 concentration during full power steady state at 100kW in the reactor room area would be 0.024 Bq/cm^3 (=6.4 x 10^{-7} μCi /cm^3). The 10CFR20 Appendix B lists that the Effluent Concentration (EC) for Argon-41 is 1x10^{-8} μCi/cm^3 for 50 mrem to the public exposed for a full year of 8,760 hours. This number corresponds to 5.7x10^{-3} mrem/hour per μCi/cm^3. The average operating time of the UUTR for last 10 years is ~50 hours per year, which is less than 1% of 8,760 hours/year.

In the **UUTR SAR 11.1.1.1.7** it is also shown that the peak downwind concentration is substantially below the DAC of 3x10^{-6} μCi/cm^3 established in 10CFR20 Appendix B and less than the permissible effluent concentration of 1x10^{-8} μCi/cm^3 for all meteorological conditions. (**UUTR SAR 11.1.1.1.5** and **UUTR SAR 11.1.1.1.7**).

3.8 Experiments

3.8.1 Reactivity Limits

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worth of control elements and experiments.

Objective

The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist:

1. The absolute value of the reactivity worth of any single secured or unsecured experiment shall be less than $1.00;
2. The sum of the absolute values of the reactivity worths of all experiments shall be less than $1.20.

Basis

The UUTR limits the worth of a single experiment to assure that sudden removal of the experiment will not cause the fuel temperature to rise above the critical temperature level of 500 °C (773.15 °K).

By limiting the absolute values of the reactivity worths of all experiments in the reactor at one time to $1.20 it assures that the removal of the total worth for all experiments not to exceed the fuel element temperature limit of 500 °C (773.15 °K) for an aluminum element and 1,000 °C (1,273.15 °K) for a stainless steel element.

Regardless of any other administrative or physical requirements, this limit has been shown in **UUTR SAR 13.2.2** to protect the reactor during the fuel's entire lifetime.

3.8.2 Materials

Applicability

This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist:

1. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams TNT equivalent shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than 25 milligrams TNT equivalent may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container; and
2. Experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components.

Basis

This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials. Operation of the reactor with the reactor fuel or structure potential damages is prohibited to avoid potential release of fission products.

3.8.3 Failures and Malfunctions

Applicability

This specification applies to experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

Where the possibility exists that the failure of an experiment under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor room or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR 20, assuming that:

1. 100% of the gases or aerosols escape from the experiment;
2. If the effluent from an irradiation facility exhausts through a holdup tank, which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;
3. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and
4. For materials whose boiling point is above 54.4 °C (130 °F or 327.6 °K) and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, 10% of these vapors can escape.

Basis

This specification is intended to meet the purpose of 10 CFR 20 by reducing the likelihood that released airborne radioactivity to the reactor bay or unrestricted area surrounding the UUTR will result in exceeding the total dose limits to an individual as specified in 10 CFR 20.

3.9 Facility Specifics LCOs

There are no facility specifics LCOs at the UUTR.

4 SURVEILLANCE REQUIREMENTS

4.0 General

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specifications

1. Surveillance requirements may be deferred during reactor shutdown [except Technical Specifications 4.3 (1) and 4.3 (5)]; however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.
2. Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications reviewed by the Reactor Safety Committee. A system shall not be considered operable until after it is successfully tested.

Basis

This specification is related to changes in reactor systems, which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

Applicability

This specification applies to the surveillance requirements for reactor core parameters.

Objective

The objective is to verify that the reactor does not exceed the authorized limits for power, shutdown margin, core excess reactivity, specifications for fuel element condition and verification of the total reactivity worth of each control rod.

Specifications

1. The shutdown margin shall be determined prior to each day's operation, prior to each operation extending more than one day, or following any change (>$0.25) from a reference core,
2. The total reactivity worth of each control rod shall be measured semi-annually or following any change (>$0.25) from a reference core,
3. The core excess reactivity shall be determined semi-annually or following any reactivity change (>$0.25) from a reference core,
4. Each planed change in core configuration shall be determined to meet the requirements of **UUTR TS 3.1.4** of these specifications before the core is loaded,
5. Inspection for transverse bend and length exceeding for fuel elements, cladding defect, overall visual inspection shall be performed biennially, and
6. Fuel burnup of Uranium-235 in the UZrH fuel matrix shall not exceeds 50% of initial content. Fuel burnup calculation shall be performed biennially.

Basis

Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

The value of a significant change in reactivity (>$0.25) is measurable and will ensure adequate coverage of the shutdown margin after taking into account the accumulation of poisons.

For inspection, looking at fuel elements from each ring biennially will identify any developing fuel integrity issues in the core. The fuel is inspected for defects including surface anomalies (spots or scratches of reddish brown, black, or white), cladding dents and bent pins. The UUTR core has upper and lower grids that have holes for a fuel element. A fuel element would not fit in the core if the transverse bend exceeds 0.0625 inches over the length of the cladding or its length exceeds its original length by 0.125 inches. An underwater camera can be used to check if any fuel element is placed correctly in the core **(UUTR SAR 4.2.5)**.

An element is considered damaged if it meets the criteria outlined in **UUTR TS**

3.1.6. Also, if an element releases bubbles directly from the cladding region during or after being raised near the surface for visual inspection, then it may be assumed that a pin hole leak was induced through depressurization. A leaking fuel element is potentially very difficult to detect and recognize because of the low levels of activity associated with such a leak. There are two mechanisms of regular surveillance which may detect the release of active materials: the Continuous Air Monitor (CAM) and monthly spectroscopy of the tank water. The purpose of the CAM is to detect short-lived gaseous products from gross leakage. The purpose of the monthly spectroscopy is to detect and differentiate long-lived soluble products released in small quantities. More information is provided in **UUTR SAR 9.2.6**. An estimated calculated maximum fuel burnup (using MCNP5) for the UUTR is 8.91% for aluminum cladding elements and 8.77% for stainless steel cladding elements (**UUTR SAR 4.2.1.1**).

4.2 Reactor Control and Safety Systems

Applicability

This specification applies to the surveillance requirements of reactor control and safety systems.

Objective

The objective is to verify performance and operability of those systems and components, which are directly related to reactor safety.

Specifications

1. Control rod inspection: The control rods and drives shall be visually inspected for damage or deterioration biennially.
2. SCRAM time: The scram time shall be measured annually and following maintenance to the control element or their drives.
3. Control rod movement: The speed of the control rod movement shall be measured annually.
4. Fuel element temperature (channel calibration, channel test, and channel check): The fuel element temperature measuring channel shall be calibrated semi-annually or at an interval not to exceed 8 months by the substitution of a known signal in place of the instrumented fuel element thermocouple. The channel test shall be performed annually. The channel check shall be performed prior and during start-up and during every operation of the reactor.
5. Linear power level (channel check and channel calibration): Channel check shall be performed for every operation of the reactor. Channel calibration of the linear power channel shall be performed semi-annually.
6. Percent power level (channel check and channel calibration): Channel check shall be performed for every operation of the reactor and channel calibration of the percent power channel shall be performed semi-annually.
7. Manual console scram (channel test): Manual console scram function channel test shall be performed prior to every reactor operation.
8. Magnet key current switch (channel test): The magnet key current channel test switch shall be performed prior to every reactor operation.
9. Console power supply (channel test): Console power supply system shall be tested prior to every reactor operation.
10. Reactor tank water level (channel check and channel test): Reactor tank water level shall be channel checked and channel tested prior to every reactor operation.
11. Startup count rate interlock (channel test): Startup count rate interlock system shall be channel tested prior to every reactor operation.
12. Control rod withdrawal interlocks (channel check and channel test): Control

rod interlock function shall be channel checked prior to every reactor operation. Control rod interlock function shall be channel tested prior to every reactor operation and semi-annually.

Basis

1. All three control rods are required to be inspected during biennial fuel inspection. Control rod removed from the core, cleaned up, and placed in the original positions after inspection.
2. Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control elements to perform properly.
3. The control rod movement speed needs to be measured annually during control rod worth measurement. The reactivity insertion for each control rod should not exceed $0.30/sec.
4. Over 35 years of the UUTR operation showed that the fuel element temperature-measuring channel calibrated semi-annually or at an interval not to exceed 8 months, the fuel temperature channel tested annually and the channel check performed prior and during start-up and during every operation of the reactor have been sufficient to assure proper operation. Details are provided in **UUTR SAR 7.2.3.2.**
5. Experience show that the linear power level channel calibration and channel check assure that the reactor is operated at the proper power levels.
6. Semi-annual percent power level channel calibration and channel check assure that the reactor is operated at the proper power level.
7. Manual console scram function is tested prior to every reactor operation assuring the function for the manual scram works properly.
8. Magnet key current switch provides a power to the console. An operator tests its proper function prior to every reactor operation.
9. Reactor will be scrammed if the power supply works incorrectly. A reactor operator therefore tests the console power supply system prior to every reactor operation.
10. Reactor tank water level will scram the reactor if the water height is below 15.5 inches measured from the top of the reactor water tank. An operator checks and tests the water height level prior to every reactor operation.
11. Experience showed that source interlock system tested prior to every reactor operation will provide required conditions for a proper reactor operation.
12. The control rod should not be moved if the neutron count rate is less than ~40 counts/sec. Control rod withdrawal interlocks prevent the withdrawal of more than one control rod at once. An interlock circuit is used to prevent rod withdrawal unless the source count level is above the required minimum value of at least 2 counts/second. Experience showed that control rod interlock function checked prior to every reactor operation and control rod interlock function tested prior to every reactor operation and semi-

annually, are satisfactory in assuring that the reactor is operated properly.

The following table summarizes the frequency of channel test, channel check and channel calibration per specifications as listed in this TS:

Specification	**Channel Test Frequency**	**Channel Check Frequency**	**Channel Calibration Frequency**
Fuel element temperature	X annually	X prior and during start-up and during every operation of the reactor	X Semi-annually
Linear Power level	NA	X during every operation of the reactor	X Semi-annually
Percent power level	NA	X during every operation of the reactor	X Semi-annually
Manual console scram	X prior and during start-up	NA	N/A
Magnetic current key switch	X prior and during start-up	NA	NA
Console power supply	X prior and during start-up	X prior every operation of the reactor	NA
Reactor tank water level	X prior every operation of the reactor	X prior every operation of the reactor	NA
Startup count rate interlock	X prior and during start-up	NA	NA
Control rod withdrawal interloks	X prior and during start-up	X prior and during start-up and during every operation of the reactor	X Semi-annually

4.3 Coolant System

Applicability

This specification applies to the surveillance requirements for the reactor tank water.

Objective

The objective is to assure that the reactor tank water level and the bulk water temperature monitoring systems are operating, and to verify appropriate alarm settings.

Specifications

1. A channel check of the reactor tank water level monitor shall be performed monthly.
2. A channel test of the reactor tank water temperature system shall be performed prior to each day's operation or prior to each operation extending more than one day.
3. A channel calibration of the reactor tank water temperature system shall be performed semi-annually.
4. The reactor tank water conductivity and pH shall be measured monthly.
5. The reactor tank water radioactivity shall be measured monthly.

Basis

Experience has shown that the frequencies of checks on systems, which monitor reactor primary water level, temperature, and conductivity adequately, keep the tank water at the proper level and maintain water quality at such a level to minimize corrosion and maintain safety.

Reactor tank water conductivity is continuously monitored; it would be manually monitored on a monthly basis if the instruments failed. Radioactivity is indirectly monitored by an area radiation monitor placed near the reactor water tank and reactor ceiling, so gross activity increases would be detected immediately. Experience with TRIGA reactors indicates the earliest detection of fuel cladding leaks is usually from airborne activity, rather than pool water activity. The quarterly measurement can identify specific radionuclides.

Analysis has shown that as long as reactor tank water conductivity is less than 0.1 μS/cm (resistivity >10MΩ-cm), pool water pH is between 7.5 and 6.5. During periods of time when reactor tank water conductivity is greater than 0.1 μS/cm (resistivity <10MΩ-cm), reactor tank water pH must be measured to ensure compliance with **UUTR TS 3.3**.

4.4 Confinement

Applicability

This specification applies to the reactor confinement.

Objective

The objective is to assure that air is swept out of confinement and exhausted through a monitored release point (two fume hood systems located at Fuel Inspection area).

Specification

The ventilation system shall be verified operable in accordance with **UUTR TS 4.5** monthly.

Basis

Because the ventilation system is the only equipment required to achieve confinement, operability checks of the ventilation system meet the functional testing requirements for confinement. The pressure difference between the reactor room and outside of the building (Merrill Engineering Building) should be larger than 0.1 inches-of-water. To keep this pressure difference, two fume hoods should be operated with the flow rate of 90 CFM or higher. Current flow rate for two fume hoods are >100 CFM.

4.5 Ventilation System

Applicability

This specification applies to the reactor area confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the unrestricted area.

Specifications

1. A channel check of the reactor area confinement ventilation system's ability to maintain a negative pressure in the reactor room with respect to surrounding areas shall be performed prior to each day's operation or prior to each operation extending more than one day.
2. A channel test of the reactor area confinement ventilation system's ability to be secured shall be performed monthly.

Basis

Over 35 years of experience has demonstrated that tests of the ventilation system on the prescribed daily and annual basis are sufficient to assure proper operation of the system and its control over releases of radioactive material. In December of 2008 improvements are made to two fume hoods in the radiochemistry lab and all ventilation duct and pipes; the more powerful motor system was installed as well. These improvements assure higher reliability in comparison to the previous system.

4.6 Emergency Power System

Does not apply to UUTR.

4.7 Radiation Monitoring Systems

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the air monitoring systems.

Objective

The objective is to assure that the radiation monitoring equipment is operating properly and to verify the appropriate alarm settings.

Specifications

1. A channel test of the Area Radiation Monitoring (ARM) system as in **UUTR TS 3.7.1** shall be performed prior to each day's operation or prior to each operation extending more than one day.
2. A channel test of the Continuous Air Monitor (CAM) system as in **UUTR TS 3.7.1** shall be performed monthly.
3. A channel calibration of the radiation monitoring systems in **UUTR TS 3.7.1** shall be performed annually.

Basis

1. The reactor checkout procedure requires the radiation level check-out for reactor room and stack release. An operator is expected to check the responses for the radiation monitoring system using Eu-152 checkout source for every day of reactor operation following the UUTR internal procedures.
2. Continuous Air Monitor (CAM) detects particulate, iodine and noble gas from the reactor operation. Experience has shown that monthly verification (test) is adequate frequency to assure operability of the CAM system. An operator shall have 48 hours to repair the CAM system when it is broken.
3. Experience has shown that an annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.

4.8 Experiments

Applicability

This specification applies to the surveillance requirements for experiments installed in the reactor and its irradiation facilities.

Objective

The objective is to prevent the conduct of experiments, which may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specifications

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment.
2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with **UUTR TS 3.8** by the Reactor Safety Committee in full accord with **UUTR TS 6.2.3** of these Technical Specifications, and the procedures, which are established for this purpose.

Basis

Evaluating an experiment prior to inserting in the reactor and its irradiation facilities will provide assurance that no damage to the reactor fuel or structure will occur. According to the **UUTR SAR 13.2.2**, with a $1.20 reactivity insertion, the fuel temperature and tank water temperature will stay below safety limits.

4.9 Facility-Specific Surveillance

Not applicable to UUTR. There is no facility-specific surveillance.

5. DESIGN FEATURES

5.1 Site and Facility Description

Applicability

This specification applies to the University of Utah TRIGA Reactor site location and specific facility design features.

Objective

The objective is to specify the location of specific facility design features.

Specifications

1. The restricted area is that area inside the MEB 1205 A room through 1205 G room. The unrestricted area is that area outside the MEB 1205 A room through 1205 G room, and MEB 1206,
2. The Merrill Engineering Building houses the TRIGA reactor,
3. The reactor room shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor room and release them from a stack at a minimum of 40 feet from ground level,
4. Emergency shutdown controls for the ventilation systems shall be located in the reactor control room, and
5. Free volume of the reactor area shall be 5.65×10^8 cm^3.

Basis

The Utah Nuclear Engineering Facility and site description are strictly defined (**UUTR SAR 2**). The facility is designed such that the ventilation system will normally maintain a negative pressure in the reactor room with respect to the outside atmosphere so that there will be no uncontrolled leakage to the unrestricted environment (**UUTR SAR 9.1.4.1**). Controls for startup and normal operation of the ventilation system are located in the reactor control room (**UUTR 9.1.2**). Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reactor control room with a minimum of exposure to operating personnel (**UUTR SAR 9.1**, **UUTR SAR 11.1.1.1**, and **UUTR SAR 13.2.4**).

5.2 Reactor Coolant System

Applicability

This specification applies to the tank containing the reactor and to the cooling of the core by the tank water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

1. The reactor core shall be cooled by natural convection water flow,
2. The reactor tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks not less than 18 feet above the top of the core,
3. A reactor tank water level alarm shall be provided to indicate loss of coolant if the water level drops 15.5 inches from the top of the reactor tank, and
4. A reactor tank water temperature shall be kept below 35 °C .

Basis

1. This specification is based on thermal and hydraulic calculations which show that the TRIGA can operate in a safe manner at power levels up to 100 kW with natural convection flow of the coolant water (**UUTR SAR 4.6.1**).
2. In the event of accidental siphoning of tank water through inlet and outlet pipes of the heat exchanger or demineralizer system, the tank water level will not drop below 18 feet measured from the top of the UUTR core (**UUTR SAR 5.2**).
3. The scram set point and the alarm is set when the water level drops below 15.5 inches from the top of the reactor tank providing a timely warning so that corrective action can be initiated. The alarm and scram for the water level have the same set points. This alarm is located in the control room (**UUTR SAR 5.2**).
4. The UUTR water clean up system utilizes two resin beds that collect all minerals and dirt thus minimizing the neutron activation of the water in the reactor water tank. According to the manufacturer's information, the resins will melt if the temperature of the reactor water tank is over approximately 40 °C. Limiting the reactor water tank temperature to 35 °C keeps the resin beds safe from melting; a reactor operator therefore shall check the water temperature every hour during the UUTR reactor operation.

5.3 Reactor Core and Fuel

5.3.1 Reactor Core

Applicability

This specification applies to the configuration of fuel and in-core experiments.

Objective

The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities shall not be produced.

Specifications

1. The core assembly shall consist of TRIGA fuel elements,
2. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite dummies, aluminum dummies, stainless steel dummies, control rods, heavy-water elements, startup sources, and vacant positions that are filled with water, and
3. The reflector, excluding experiments and irradiation facilities, shall be water or a combination of graphite and heavy water elements and water.

Basis

1. Only TRIGA *fuel elements* are authorized ever to be used (**UUTR SAR 4.2**).
2. *In-core water-filled experiment positions* have been demonstrated to be safe in the Gulf Mark III reactor. The largest values of flux peaking will be experienced in hydrogenous in-core irradiation positions. Various non-hydrogenous experiments positioned in element positions have been demonstrated to be safe in TRIGA fuel element cores of up to 2-MW operation (**UUTR SAR 4.2, 4.5**).
3. The core will be assembled in the reactor grid plate, which is located in a reactor tank of light water. *Water in combination with graphite and heavy water reflectors* can be used for neutron economy and the enhancement of irradiation facility radiation requirements (**UUTR SAR 4.2**).

5.3.2 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

The shim, safety, and regulating control rods shall have scram capability and contain borated graphite, B_4C powder or boron, with its compounds in solid form as a poison, in aluminum or stainless steel cladding.

Basis

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder or boron as its compounds. These materials must be contained in a suitable clad material such as aluminum or stainless steel to ensure mechanical stability during movement and to isolate the poison from the tank water environment. Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor. More information is provided in **UUTR SAR 4.2.2**.

5.3.3 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

The individual TRIGA fuel elements shall have the following characteristics:

1. Uranium content: maximum of 8.5 wt% enriched to less than 20% ^{235}U,
2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.0 and 1.60,
3. Cladding: 304 stainless steel or aluminum, nominal 0.02 or 0.03 inches thick respectively,
4. Identification: top pieces of fuel elements will have characteristic markings to allow visual identification of fuel elements, and
5. Burnable poisons: the fuel elements shall not include burnable poisons.

Basis

1., 2., and 3.

Each high hydride fuel element shall contain uranium-zirconium hydride and be cladded with 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 20 weight percent uranium, which has a maximum enrichment of less than 20 percent and 1.6 hydrogen atoms to 1.0 zirconium atoms. Each low hydride fuel element shall contain uranium-zirconium hydride and be cladded with 0.030 inch of aluminum or 0.020 inch of 304 stainless steel. Each element shall contain a maximum of 8.5 weight percent uranium, which has a maximum enrichment of less than 20 % and 0.9 to 1.6 hydrogen atoms to 1.0 zirconium atoms. These types of fuel elements have a long history of successful use in TRIGA reactors. More information is provided in **UUTR SAR 4.2.1**.

4. The UUTR has three different types of top pieces of the fuel elements: the old stainless steel, new stainless steel and the aluminum cladded fuel elements. The stainless steel cladded fuel elements has torpedo or triangular shape seen from the top view. Aluminum cladded fuel element has also triangular shape from the top view but it has different color from the stainless steel cladded fuel element. More detailed identification is achievable when using an underwater camera or binocular, both available at the facility (**UUTR SAR 4.2.4, Figure 4.2-7**).
5. The UUTR core does not use burnable poison fuel elements.

5.4 Fuel Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel, which is being stored shall not become critical and shall not reach an unsafe temperature.

Specifications

1. All fuel elements shall be stored in a geometrical array where the k- effective is less than 0.9 for all conditions of moderation, and
2. Irradiated fuel elements and fuel devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the temperature of the fuel element or fueled device will not exceed the safety limit.

Basis

The limits imposed are conservative and assure safe storage (NUREG-1537). Detailed calculations confirming that the stored fuel shall not become critical are presented in **UUTR SAR 9.2.4** and **9.2.5**.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiological exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations.

6.1.1 Structure

The reactor administration shall be related to the University as shown in **Fig. 6-1.**

6.1.2 Responsibilities

The following specific organizational levels, and responsibilities shall exist:

1. The UUTR is an integral part of the Utah Nuclear Engineering Facilities (UNEF) of the University of Utah. The organization of the facility management and operation is illustrated in **Fig. 6-1**. The responsibilities and authority of each member of the operating staff shall be defined in writing, and

2. As indicated in **Fig. 6.1**, the Reactor Safety Committee shall report to Level 1. Radiation safety personnel shall report to Level 2. Additional description of levels follows:

   1.1 Level 1: Individual responsible for the reactor facility's licenses, i.e., the Associate Vice President for Research in the Office of Vice President for Research; The Vice President for Research will assign which of the Associate Vice Presidents for Research will be the responsible Level 1 individual.

   1.2 Level 2: Individual responsible for reactor facility operation, i.e., the Utah Nuclear Engineering Facility (UNEF) Manager shall be the Director of the Utah Nuclear Engineering Program (UNEP).

   1.3 Level 3: Individual responsible for day-to-day operation or shift shall be the reactor supervisor (RS). This person shall be a senior reactor operator (SRO).

   1.4 Level 4: Operating staff shall be senior reactor operators, reactor operators, and trainees.

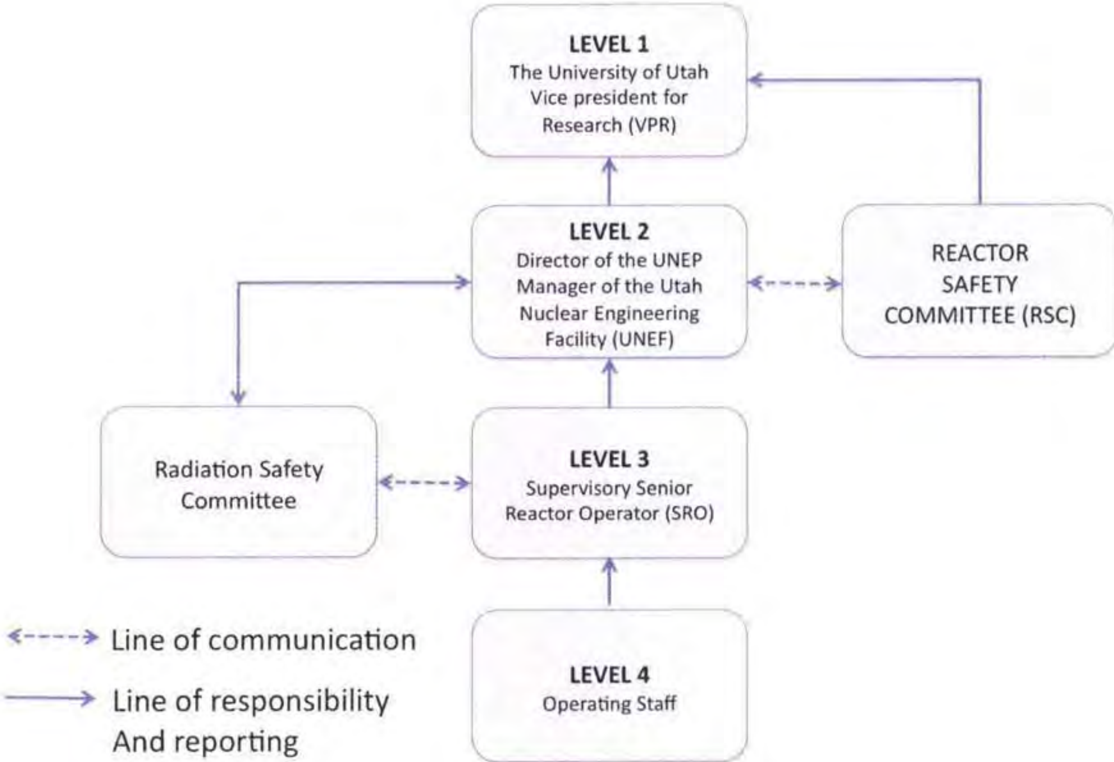


Figure 6-1 University of Utah Administrative Organization for Nuclear Reactor Operations

6.1.3 Staffing

1. The *minimum* staffing when the reactor is operating shall be:
   - 1.1 A reactor operator or the Reactor Supervisor in the control room;
   - 1.2 A second person present in the UNEF able to carry out prescribed instructions; and
   - 1.3 If neither of these two individuals is the Reactor Supervisor, the Reactor Supervisor shall be readily available on call. Readily available on call means an individual who:
     - i. Has been specifically designated and the designation is known to the operator on duty;
     - ii. Can be rapidly contacted by phone by the operator on duty; and
     - iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).
2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:
   - 2.1 UNEP Director and/or UNEF Manager
   - 2.2 Reactor Supervisor
   - 2.3 Radiation Safety Officer
   - 2.4 Any Licensed Reactor or Senior Reactor Operator
3. Events requiring the direction of the Reactor Supervisor:
   - 3.1 Initial startup and approach to power of the day;
   - 3.2 All fuel or control-rod relocations within the reactor core region;
   - 3.3 Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar; and
   - 3.4 Recovery from unplanned or unscheduled shutdown or significant power reduction.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall be in accordance with ANSI/ANS 15.4 – 1988; R1999, "Standard for the Selection and Training of Personnel for Research Reactors."

6.2 Review and Audit

The Reactor Safety Committee (RSC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. Minutes, findings or reports of the RSC shall be presented to Level 1 and Level 2 management within ninety (90) days of completion.

6.2.1 RSC Composition and Qualifications

An RSC of at least five (5) members knowledgeable in fields, which relate to reactor engineering and nuclear safety shall review and evaluate the safety aspects associated with the operation and use of the facility. Level 1 management shall appoint the RSC members and RSC chair. Individuals may be either from within or outside the University of Utah. Qualified and approved alternates may serve in the absence of regular members. The Level 2 and Level 3 should be the members of the RSC but they would not comprise a majority of voting RSC members.

6.2.2 RSC Rules

The operations of the RSC shall be in accordance with written procedures including provisions for:

1. Meeting frequency (at least annually),
2. Voting rules,
3. Quorums (5 members, no more than two voting members may be of the operating staff at any time),
4. Method of submission and content of presentation to the committee;
5. Use of subcommittees, and
6. Review, approval, and dissemination of minutes.

6.2.3 RSC Review Function

The responsibilities of the RSC, or designated Subcommittee thereof, include, but are not limited to, the following:

1. Review all changes made under 10 CFR 50.59,
2. Review of all new procedures and substantive changes to existing procedures,
3. Review of proposed changes to the technical specifications, license or charter,
4. Review of violations of technical specifications, license, or violations of internal procedures or instructions having safety significance,
5. Review of operating abnormalities having safety significance,
6. Review of all events from reports required in Sections 6.6.1 and 6.7.2 of these Technical Specifications,
7. Review of audit reports, and
8. Review of the experiments and classes of the experiments.

6.2.4 RSC Audit Function

The RSC or a Subcommittee thereof shall audit reactor operations at least annually. The annual audit shall include at least the following:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions,
2. The retraining and requalification program for the operating staff,
3. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety, and
4. The Emergency Response Plan and implementing procedures.

6.3 Radiation Safety

The Radiation Health Physicist from the Radiological Health Department shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program shall use the guidelines of the ANSI/ANS 15.11 – 1993; R2004, "Radiation Protection at Research Reactor Facilities" .

6.4 Procedures

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

1. Startup, operation and shutdown of the reactor,
2. Fuel loading, unloading, and movement within the reactor,
3. Maintenance of major components of systems that could have an effect on reactor safety,
4. Surveillance checks, calibrations, and inspections required by the technical specifications or those that have an effect on reactor safety,
5. Radiation protection,
6. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity,
7. Shipping of radioactive materials, and
8. Implementation of the Emergency Response Plan.

Substantive changes to the above procedures shall be made only after review by the RSC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the UNEP director and documented by the UNEP director within 120 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved prior to implementation by the RSO and documented by the RSO (Radiation Safety Officer) within 120 days of implementation. Temporary deviations from the procedures may be made by the responsible senior reactor operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the UNEP director.

6.5 Experiments Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include:

1. All new experiments or class of experiments shall be reviewed by the RSC and approved in writing by the Level 2 or designated alternates prior to initiation, and
2. Substantive changes to previously approved experiments shall be made only after review by the RSC and approved in writing by the Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

6.6 Required Actions

6.6.1 Actions to Be Taken in Case of Safety Limit Violation

In the event a safety limit (fuel temperature) is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC,
2. An immediate notification of the occurrence shall be made to the UNEP director, and Chairperson, RSC, and
3. A report, and any applicable follow-up report, shall be prepared and reviewed by the RSC. The report shall describe the following:
   - 3.1 Applicable circumstances leading to the violation including, when known, the cause and contributing factors,
   - 3.2 Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public, and
   - 3.3 Corrective action to be taken to prevent recurrence.

6.6.2 Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 Other than a Safety Limit Violation

For all events, which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 6.7.2, except a safety limit violation, the following actions shall be taken:

1. The reactor shall be secured and Director notified,
2. Operations shall not resume unless authorized by the Director,
3. The Reactor Safety Committee shall review the occurrence at their next scheduled meeting, and
4. A report shall be submitted to the NRC in accordance with Section 6.7.2 of these Technical Specifications.

6.7 Reports

6.7.1 Annual Operating Report

An annual report shall be created and submitted by the UNEP Director to the USNRC by the end of July of each year consisting of:

1. A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical,
2. The number of unplanned SCRAMs, including reasons therefore,
3. A tabulation of major preventative and corrective maintenance operations having safety significance,
4. A brief description, including a summary of the safety evaluations, of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59,
5. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 % of the concentration allowed or recommended, a statement to this effect is sufficient,
6. A summarized result of environmental surveys performed outside the facility, and
7. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 % of that allowed.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the UNEP Director to the NRC as follows:

1. A report not later than the following working day by telephone and confirmed in writing by facsimile to the NRC Operations Center, to be followed by a written report that describes the circumstances of the event within 14 days to the NRC Document Control Desk of any of the following:

- 1.1 Violation of the safety limit,
- 1.2 Release of radioactivity from the site above allowed limits,
- 1.3 Operation with actual safety system settings from required systems less conservative than the limiting safety system setting,
- 1.4 Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in **UUTR TS 1.3**,
- 1.5 A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required,
- 1.6 An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;
- 1.7 Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable, or
- 1.8 An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations, and

2. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:

- 2.1 Permanent changes in the facility organization involving Level 1- 2 personnel, and
- 2.2 Significant changes in the transient or accident analyses as described in the Safety Analysis Report.

6.8 Records

6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years

1. Normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year),
2. Principal maintenance activities,
3. Reportable occurrences,
4. Surveillance activities required by the Technical Specifications,
5. Reactor facility radiation and contamination surveys,
6. Experiments performed with the reactor,
7. Fuel inventories, receipts, and shipments,
8. Approved changes to the operating procedures, and
9. RSC meetings and audit reports.

6.8.2. Records to be Retained for at Least One Certification Cycle

Records of retraining and requalification of licensed reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the certification is renewed.

6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

1. Gaseous and liquid radioactive effluents released to the environs;
2. Offsite environmental monitoring surveys,
3. Radiation exposures for all personnel monitored,
4. Drawings of the reactor facility, and
5. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

This is not applicable for a renewal application.

15.2 Financial Ability to Operate a Non-Power Reactor

Pursuant to 10 CFR 50.33(f)2 we are submitting the operating costs for the UUTR (University of Utah TRIGA Reactor) for FY2012 to FY2016. The current cost to operate the reactor is approximately $140,436.00 for the year 2012 and increases to $158,062.00 for the year 2016. The cost includes salary and benefits for the Reactor supervisor. The University of Utah covers this salary and associated fringe benefit. All activates other than those required for regulatory compliance are covered by the research or service contract for which the work is preformed. Additional expenses in the next four years are shown in **Table 15.2-1**. In this table we project a conservative increase in the cost of the same expenditures at a rate of 3% per year.

The University of Utah covers the cost of insurance for the UUTR. Coverage is provided by "American Nuclear Insurer's" for an annual premium of approximately $9,500.00. Overhead costs such as utilities, confinement building maintenance and health physics monitoring are provided by the University of Utah and are excluded from this analysis of operating cost.

Table15.2- 1 The UUTR operating budget from 2012 to 2016. Equipment upgrades and lab supplies cost will be directly provided from the Utah Nuclear Engineering program (UNEP).

Year/Item	2012	2013	2014	2015	2016
Salary for Reactor Supervisor	$92,600.00	$95,378.00	$98,239.00	$101,186.00	$104,222.00
Benefits for RS	$33,336.00	$34,336.00	$35,366.00	$36,427.00	$37,520.00
Electricity, Water	$5,000.00	$5,150.00	$5,305.00	$5,464.00	$5,628.00
Insurance	$9,500.00	$9,785	$10,079.00	$10,381.00	$10,692.00
Total	**$140,436.00**	**$144,649.00**	**$148,989.00**	**$153,458.00**	**$158,062.00**

15.3 Financial Ability to Decommission the Facility

The following analysis for decommissioning of the TRIGA reactor at the University of Utah is based on the analysis done by the Department of Defense (DOD), [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145], for the AFRRI TRIGA reactor facility. The cost of decommissioning is divided into three major categories:

- Waste disposal costs
- Labor costs
- Energy costs

For each of the major categories of costs, a detailed data is provided based on the report by DOD [1] and the differences in design have been taken into account. The dollars are adjusted to 2005 based on Consumer Price Index (CPI)1.

15.3.1 Waste Disposal Costs

The amount of structural material that has been exposed to radiation in the reactor building and the cost for transportation are provided in **Table 15.3-2.** The cost of crates and transportations are obtained from [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145] which is developed based on data provided in NUREG/CR-1756. For the purposes of this report, the worst case scenario of shipment to a destination in Washington DC has been considered. The cost per volume for disposing radioactive waste depository was obtained from [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145] which is based on Barnwell charges to be $\$2825/m^3$ for 1989 dollars (which is equivalent to $\$4450/m^3$ for 2005 dollars). Plywood 3.5 m^3 crates are used for removing the waste which costs \$400 for 1981 dollars (which is equivalent to \$860 for 2005 dollars).

[1] The CPI used in our calculations is obtained from Bureau of Labor Statistics (BLS) (www.bls.gov/cpi). On their website, the CPI is defined as: "The Consumer Price Index (CPI) is a measure of the average change over time in the prices paid by urban consumers for a market basket of consumer goods and services." This index value has been calculated since 1913 and for the current year, the index for the last month is used.

The volumes are rounded up to stay on the conservative side for the estimation of the costs. The shipping costs are adjusted based on [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145] for 2005 dollars and the highest value (which is for Stainless Steel) is used for all the materials to again stay on the safe side for estimation of the costs.

Table 15.3-2 Waste disposal costs for 2005 dollars

Material	**Volume (m^3)**	**Crates (no.)**	**Shipping**	**Costs**
Contaminated concrete	10	3	$53,350	$100,750
Contaminated sand	60	18	$320,100	$591,600
Contaminated aluminum	5	2	$26,675	$51,665
Contaminated Stainless Steel	5	2	$26,675	$51,665
Total				$795,680

15.3.2 Labor Costs

The labor costs (**Table 15.3-3**) are obtained from [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145] which is based on NUREG/CR-1756. The TRIGA reactor at the University of Utah is smaller than the AFRRI TRIGA facility; however, the numbers are unchanged to have a conservative estimation of the costs of labor. The numbers are adjusted based on CPI from 1981 dollars to 2005 dollars.

15.3.3 Energy Costs

The energy costs (**Table 15.3-4**) are obtained also from [M. Forsbacka, M. Moore. An Analysis of Decommissioning Costs for the AFRRI TRIGA Reactor Facility. Defense Nuclear Agency, Armed Forces Radiobiology Research Institute. Bethesda, Maryland 20814-5145] which his based on NUREG/CR-1756 and the energy cost per kWh is obtained from Department of Energy report for Electric Power Monthly averages for Nov-09 report to be 6.18 cents per kWh for the state of Utah for all sectors.

15.3.4 Total Decommissioning Cost and Inflation Adjusting Methodology

The total cost for the reactor decommissioning based on cost break down shown above is provided in Table 15.3-5. The cost of spent fuel removal, shipment, and building demolition costs are provided in Table 15.3-5 as well.

Table 15.3-3 Decommissioning labor costs (for DECON) for 2005 dollars

	Workyears (no.)	Rate ($1000/hr)	Cost (1000$)
Management and support staff			
Decomm superintendent	2	$192	$383
Decomm engineer	2	$163	$327
Secretary	2	$52	$104
Clerk	0.5	$52	$26
Health physicist	2	$101	$202
Radioactive shipment specialist	0.5	$84	$42
Procurement specialist	0.5	$84	$42
Contract and accounting specialist	0.8	$101	$81
Security supervisor	0.625	$120	$75
Security patrol officer	3.6	$55	$197
QA engineer	0.7	$101	$71
Control room operator	1	$74	$74
Consultant	1	$215	$215
Decomm workers			
Shift engineer	1	$112	$112
Craftsman	2	$69	$138
Crew leader	0.5	$95	$48
Utility operator	0.342	$69	$24
Laborer	6	$66	$399
Health physics technician	3	$65	$194
Total			$2,752

Table 15.3-4 Energy costs

Equipment	Energy use (kWh)	Cost
General system	9000	$556.20
HV AV	20000	$1,236.00
Lighting	23000	$1,421.40
Control room	5200	$321.36
Fire protection	600	$37.08
Security	5600	$346.08
Communications	900	$55.62
Domestic water	36300	$2,243.34
Reactor water	23400	$1,446.12
Compressed air	15000	$927.00
Building heating	302600	$18,700.68
Decommissioning equipment	20000	$1,236.00
Total		$28,526.88

Table 15.3-5 Total cost of decommissioning of TRIGA reactor the University of Utah using DECON for 2005 dollars

Category	Costs
DECON	
Waste Disposal	$795,680
Labor	$2,752,000
Energy	$28,527
Contigency fund	$894,052
Ancillary	
Spent fuel removal + Shipment	$337,853
Site demolition	$563,038
Total	$5,371,150

The estimated cost of decommissioning the TRIGA reactor at the University of Utah is based on 2005 dollars and it is intended to use CPI for adjusting the cost for future dollar values.

15.4 Numerical Example

The dollar values are updated based on CPI from Bureau of Labor Statistics (BLS) are used to take into account the inflation over the years of life of the reactor. These inflation adjusted factors are provided in **Table 15.4-1**.

Table 15.4-1 Inflation adjusted factors from 1981 to 2010 obtained from Bureau of Labor Statistics website

Year Range	Adjusted Factor (Based on CPI)
1981-1985	1.18
1985-1990	1.21
1990-1995	1.17
1995-2000	1.13
2000-2005	1.13
2005-2010	1.11

As an example, equivalent of 25,000$ of year 1981 in year 2000 can be calculated as follows:

1$ (1981) × 1.18 × 1.21 × 1.17 × 1.13 = 1.89$ (2000)

25,000$ × 1.89 = 47,250$

Note: all the inflation adjusted factors are rounded to two significant digits.

16. OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

16.1.1 Reactor Tank

The UUTR has been operated for over 35 years with existing components including ion chambers, control rods, fuel elements, thermal irradiator, and inner reactor pool tank. The upgraded reactor instruments include the CAM, flow meter, and some other electronic parts in the control console. Recently, the UUTR staff installed a new reactor monitoring system using the Lab View™. Some of the fuel elements were replaced with new elements. Some of the fuel elements are used for many years. However, the total reactor operation time of 50 hours per year is quite small compared to other TRIGA facilities. For example, the UUTR usage in last 10 years (from 2000 to 2010 approximately) is between 0.0386 MW-day and 0.0835 MW-day. The UUTR facility has never experienced a fission product leak problem with any of the fuel elements. The existing TRIGA fuel in the UUTR is estimated to be adequate for operation in safe and reliable manner for the next period of the license renewal.

The UUTR reactor tank is made of double wall. The inner aluminum tank wall is 8 ft in diameter. The distance between the inner aluminum tank and the side of the core is ~3 ft. The aluminum tank is at a sufficient distance from the UUTR core so that the radiation damage is negligible. The neutron flux was determined based on the MCNP5 model of the 100kW UUTR core [100kW was used as conservative estimate because the UUTR operates only at 90kW]. At 100 kW UUTR reactor power, the total core flux is estimated using the MCNP5 core model to be 7.53×10^{15} neutrons/sec. Based on the MCNP5 model, the average neutron flux (over the core) is 3.54×10^{12} neutrons/cm^2-sec at 100 kW. Therefore, total neutron flux that passes through the entire core surface area is 7.53×10^{15} neutrons/sec at 100 kW. The distance between the bottom of the core grid and the reactor pool floor is approximately 15 cm. At the floor of the tank, the total neutron flux is estimated to be 2.73×10^{9} neutrons/sec at 100 kW. **Table 16.1-1** shows the total neutron flux at the aluminum tank wall and the bottom of the reactor tank. The neutron and gamma dose distribution at the core upper plane level across the pool tank is shown in **Fig. 16.1-1**. The MCNP5 input file indicating details of these simulation s listed in **Appendix 16.A**.

An 6061 aluminum alloy examined at the Oak Ridge National Laboratory for neutron radiation damage after exposure to a maximum fast neutron (E>0.1 MeV) fluence of 9.2×10^{22} neutrons/cm^2 and a thermal neutron fluence of 1.38×10^{23} neutrons/cm^2 at 60° C showing that the voids and transmutation-produced silicon precipitate were found to cause about 1.1% internal swelling [Kng, RT, Jostsons, A, Farrell, K, "Neutron Irradiation Damage in a Precipitation-Hardened Aluminum Alloy," Oak Ridge National Laboratory Australian Atomic Energy Commission Research Establishment, Tenn. Jan. 1973]. Almost all of the neutrons that reach the aluminum tank are thermal neutrons because the distance between the core surface and aluminum tank is approximately 60 cm. The neutron flux 70 cm above the top of the reactor core is negligible. Therefore, it is assumed that the neutron flux from the core diminishes above the height of 146 cm (measured from the floor of the tank). As shown in **Figure 16.1-1**, the gamma and neutron doses are negligible near the aluminum tank. Maximum neutron and gamma doses on the surface of the core at the A-ring are 9.70×10^5 rem/hr and 2.68×10^5 rem/hr respectively. If the UUTR operates at 100 kW with the calculated average neutron flux, it will take approximately 1.4×10^{13} hours of operation to reach the neutron fluence of 1.38×10^{23}neutrons/cm^2. This estimate is highly conservative and is showing that it is not possible to expect any damage of the inner aluminum tank wall or the bottom surface of the reactor tank.

Table 16.1-1 Total neutron fluence near aluminum tank for 100kW UUTR (obtained using the MCNP5)

-"Location" indicates the vertical height from the bottom of the reactor tank. When the height increases more than 100 cm, there is no neutron flux. The center of the core was selected as a reference point (0 cm) -

		100 kW
Surface	**Location**	**Ave ϕ (n/sec)**
Aluminum Tank	-43 to -35 cm	9.79E+04
	-35 to -25 cm	1.92E+05
	-25 to -15 cm	1.72E+05
	-15 to -5 cm	4.42E+05
	-5 to 5 cm	6.28E+05
	5 to 15 cm	3.38E+05
	15 to 25 cm	5.27E+05
	25 to 35 cm	9.11E+04
	35 to 50 cm	3.28E+05
	50 to 70 cm	4.41E+03
	70 to 100	0.00E+00
Al Tank Total	-43 to 70 cm	**2.82E+06**
Tank Bottom	-43 cm	**2.73E+09**
TI	-	**3.99E+11**

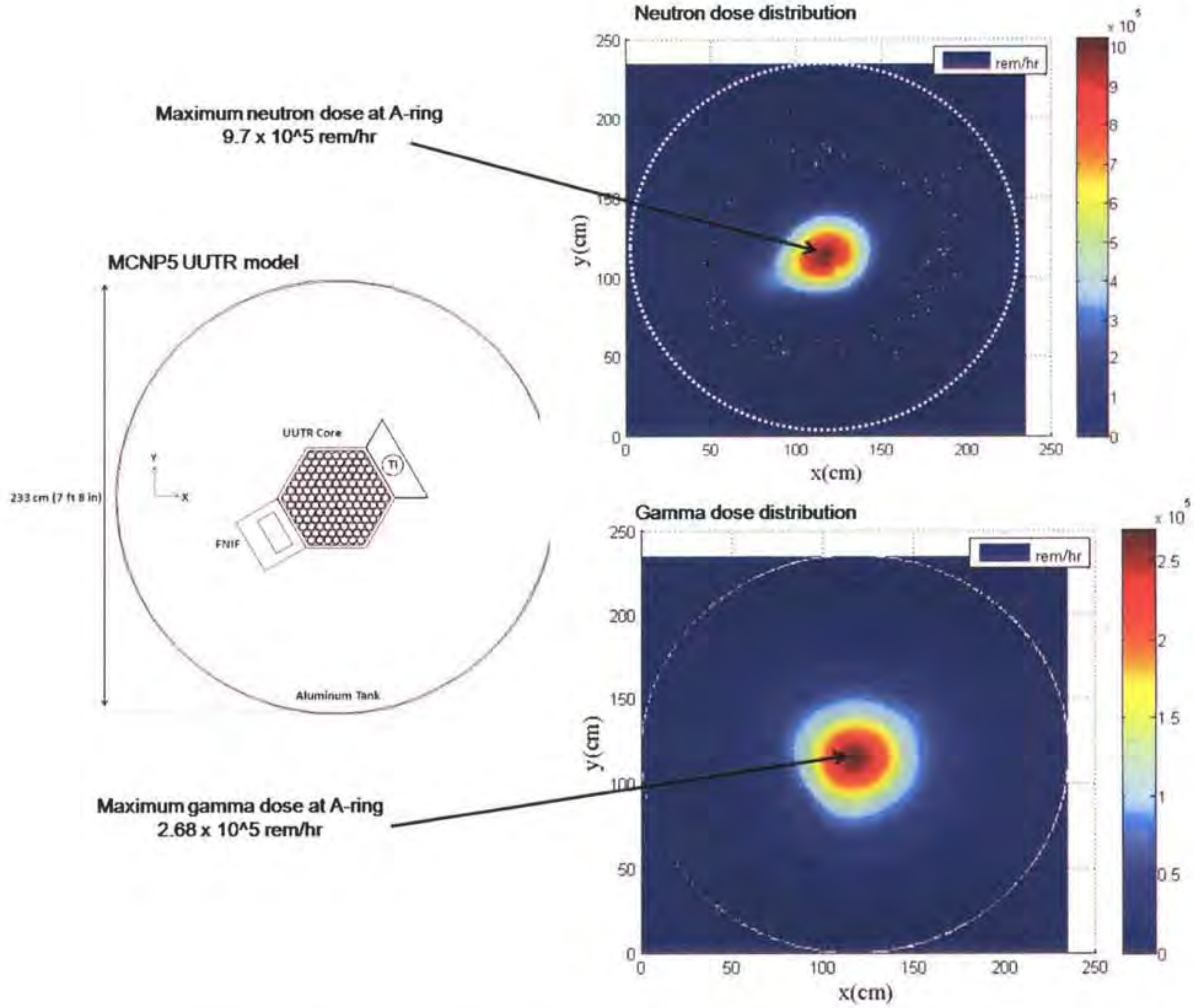


Figure 16.1-1 MCNP5 Neutron and Gamma Dose Distribution at the UUTR Core Surface across the Pool Water Tank

16.1.2 Ion Chambers

The UUTR has one fission chamber, one compensated ionization chamber, and two uncompensated ionization chambers. The location of these ionization chambers and fission chamber are shown in **Fig. 16.1-2**. These fission chamber and ionization chambers are connected to the neutron monitoring channels in the reactor control console. The neutron monitoring channels contain startup channel, linear power channel, percent power channel, and log power and reactor period channel. All neutron–sensing chambers are sealed in aluminum can and mounted in the water reflector on the outside of the core so that their positions are adjustable vertically to change sensitivity and for calibration.

The startup channel includes a fission chamber, power supply, preamplifier, linear amplifier and linear and log-count-rate circuits. The channel provides power indication from below the source level to approximately 10 W. In addition, a minimum source-count interlock prevent control rod withdrawal unless the measured source level exceeds a predetermined value (currently it is approximately 28 counts/sec). This number will not be affected by the core power uprate.

The log channel displays the logarithm (base 10) of the reactor power. The log channel and percent power channel are connected to the uncompensated ionization chambers.

[redacted]

Figure 16.1-2 Fission chamber, compensated ionization chamber, and two uncompensated ionization chambers

Table 16.1-2 shows the useful range of various channels on the reactor control console. As shown in **Table 16.1-2**, the second column "Useful range" represents the setting point on the control console. These setting points will be changed and calibrated accordingly when the power uprate is admitted. Operating range of the 100kW UUTR's ion chambers are shown in **Fig. 16.1-3**. The compensated and uncompensated ion chamber can be used for the reactor power of up to 1 MW. Fission chamber can be used for several hundreds of counts/second. This fission counter will not be affected by the UUTR reactor power uprate.

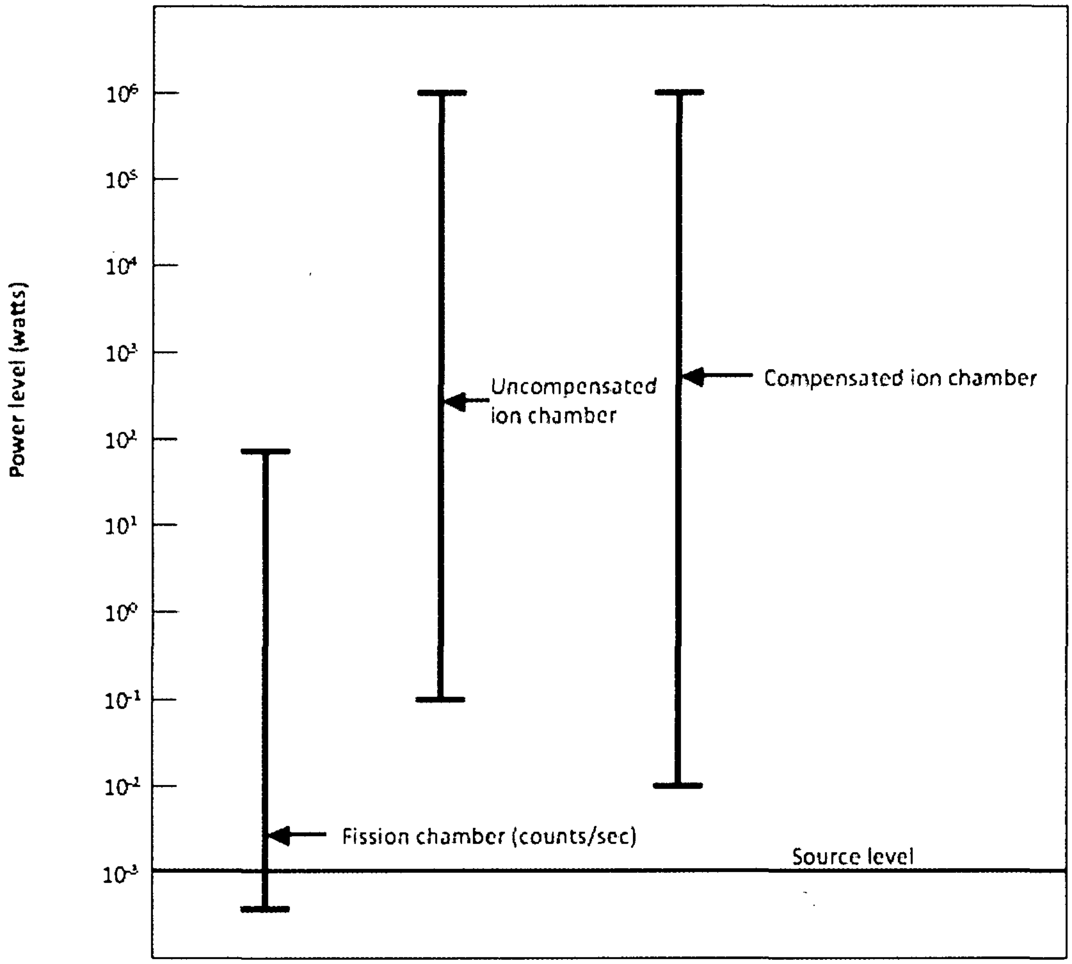


Figure 16.1-3 The operating range of fission chamber and ion chambers

16.2 Core Assembly

The core assembly consists of the upper, and lower grid plates, support assembly, and the side plates. These plates have been subjected to a high radiation environment and there has been no evidence of deformation or discoloration due to this exposure, and no expectation of damage at proposed power levels. Thermal cycling of these components is on the order of 15 to 100 °C with a typical period of one hour.

The rise time to these sorts of temperatures is typically much faster than the cycling period. Nevertheless these temperature effects are incapable of causing any damaging or permanent deformations.

Table 16.1-2 Various channels for the UUTR. Useful range is the setting value on the control console, not the detector or ionization chamber's range

Channel	Useful range	Ion chamber's response range	Sensor type	Accuracy	Calibration interval
Water temperature	0° C to 100° C		K-type thermal couple, except for Refrig. Water that is E-type	$\pm 1^\circ$C	semiannual
Fuel temperature channels	0° C to 800° C		K-type thermal couple	$\pm 1^\circ$C	semiannual
Linear power channel	100 mW to 100 kW	100 mW to 1 MW	Compensated ion chamber 3dB Bandwidth: 20Hz	$\pm$ 1% of reading $\pm$ 1% of range switch setting	semiannual
Integrated power channel	1 W-hr to 100 MW-hr	1 W-hr to 100 MW-hr	Compensated ion chamber (from linear power channel)	$\pm$ 1% of reading $\pm$ 1% of range switch setting	semiannual
% power channel	100 W to 100 kW	100 W to 1 MW	Uncompensated ion chamber 3dB Bandwidth: 20Hz	$\pm$ 1% of reading	semiannual
Log power channel	100 mW to 100 kW	100 mW to 1 MW	Uncompensated ion chamber 3dB Bandwidth: 20Hz	+3 % of reading	semiannual
Reactor period	-32 seconds to +4 seconds		Uncompensated ion chamber (from log power channel)	$\pm$ 20%	unnecessary
Rod position indicators	0 to 100 % of rod withdrawal		Multiturn potentiometer	$\pm$ 0.2%	unnecessary

16.3 Fuel and Instrumented Fuel

Because the surveillance procedures and design criteria for fuel and instrumented fuel are identical, they are evaluated together. TRIGA fuel used in the UUTR is designed for a core lifetime of 7,000 MW days. Since the reactor's operation began in 1975 the core has seen 17,000 kW hours (0.71 MW days) of operation or 0.01% of the fuel's operational lifetime. However, the majority of the fuel in this core comes from the University of Arizona and General Atomics (GA) and had an operational use of approximately 210 MW days. As can be seen from these values, the UUTR current fuel life is well below the limits established by General Atomics. Additionally, the fuel's temperature limits (design basis limits of 530 °C for aluminum fuel, 1150 °C for stainless steel fuel) have never been approached (maximum operating fuel temperature of the UUTR reactor, at 100 kW, is less than 130 °C). To insure that corrosion, erosion, or mechanical damage to the fuel is well within.acceptable limits, the surveillance of the fuel is performed monthly through water analysis, fuel leaks, and every two years visual inspections of the fuel are performed to look for evidence of corrosion, erosion or mechanical damage.

16.4 Neutron Source

The UUTR neutron source is a double encapsulated Pu-Be neutron source. This source is removed from its position in the reactor core at power levels greater than 1.0 W, and therefore is not exposed to significant neutron flux at power greater than 1.0 W or the heated cooling water thermal environment. Also, by visual inspection no evidence of erosion, corrosion, or mechanical damage has been observed on the neutron source.

16.5 Control Rods and Control Rod Drives

Because of the routine surveillance procedures for the control rods and the integrally connected control rod drives these systems are evaluated together. The control rods and control rod drives are physically inspected for erosion, corrosion, and mechanical damage every two years. Every six months the rods and rod drives are calibrated and the rod reactivity worths are determined experimentally.

16.6 Fission Chamber

The fission chamber is used to verify a continuous neutron population in the UUTR core. If a problem develops with the fission chamber, it will be removed from service, and repaired or replaced as a maintenance procedure for the reactor.

16.7 Medical Use of Non-Power Reactors

No human-use irradiation studies will be performed at the UUTR. The University of Utah's Radiation Safety Committee's Human Use Subcommittee must review and approve through appropriate designated medical personnel all such applications. Furthermore, while the UUTR may conduct research on the production of medical isotopes there are no plans to produce any medical isotopes intended for human use. Approval for such use must be obtained from the above medical channels as well as both Utah and NRC regulatory agencies. If either of these circumstances change the appropriate analysis will be performed, and the application for facility amendment will be sent to the NRC.

Appendix 16.A

```
TRIGGA 3D Model 1.72 in pitch, n-dose, p-dose, ssw, core top, 90kW
c                              KCODE 500E6
c
c New SS Fuel
100   1   -5.636 -2 11 -12                u=1   imp:n=1 $Fuel Meat
101   2   -1.70  -2 12 -14                u=1   imp:n=1 $Up Graphite
102   2   -1.70  -2 13 -11                u=1   imp:n=1 $Down Graphite
103   3   -7.92  (-1 15 -16) (2:-13:14)   u=1   imp:n=1 $Cladding
104   4   -1.0   1:-15:16 92 -93          u=1   imp:n=1 $H2O
c Old SS Fuel
110 like 100 but mat=12 rho=-5.636        u=2   imp:n=1 $Fuel Meat
111 like 101 but                          u=2   imp:n=1 $Up Graphite
112 like 102 but                          u=2   imp:n=1 $Down Graphite
113 like 103 but                          u=2   imp:n=1 $Cladding
114 like 104 but                          u=2   imp:n=1 $H2O
c Al Fuel
120   5   -6.143 -3 21 -22                u=3   imp:n=1 $Fuel Meat
121   2   -1.70  -3 22 -24                u=3   imp:n=1 $Up Graphite
122   2   -1.70  -3 23 -21                u=3   imp:n=1 $Down Graphite
123   6   -2.70  (-1 25 -26) (3:-23:24)   u=3   imp:n=1 $Cladding
124   4   -1.0   1:-25:26 92 -93                u=3   imp:n=1 $H2O
c Instrumental Fuel
130 like 110 but                          u=4   imp:n=1 $Fuel Meat
131 like 111 but                          u=4   imp:n=1 $Up Graphite
132 like 112 but                          u=4   imp:n=1 $Down Graphite
133 like 113 but                          u=4   imp:n=1 $Cladding
134 like 114 but                          u=4   imp:n=1 $H2O
c Graphite
140   2   -1.70   -3 23 -24               u=6   imp:n=1 $Graphite
143   like 123 but                        u=6   imp:n=1 $Cladding
144   like 124 but                        u=6   imp:n=1 $H2O
c Heavy Water
150   7   -1.056  -3 23 -24               u=7   imp:n=1 $D2O
153   like 123 but                        u=7   imp:n=1 $Cladding
154   like 124 but                        u=7   imp:n=1 $H2O
c Water
160   4   -1.0    -1 92 -93               u=8   imp:n=1 $H2O
161   4   -1.0     1 92 -93               u=8   imp:n=1 $H2O
c Safety Control Rod
170   9   -2.52   -46 11 -93              u=10  imp:n=1 $B4C
171   6   -2.7     46 -47 11 -93          u=10  imp:n=1 $Al Cladding
172   4   -1.0    (47 -50 11 -93):(-50 -11 92)  u=10  imp:n=1 $H2O
173   6   -2.7     50 -1 92 -93           u=10  imp:n=1 $Al Tube
174   4   -1.0     1 92 -93               u=10  imp:n=1 $H2O
c Shim Control Rod
180   9   -2.52   -46 11 -93              u=11  imp:n=1 $B4C
181   6   -2.7     46 -47 11 -93          u=11  imp:n=1 $Al Cladding
182   4   -1.0    (47 -50 11 -93):(-50 -11 92)  u=11  imp:n=1 $H2O
183   6   -2.7     50 -1 92 -93           u=11  imp:n=1 $Al Tube
184   4   -1.0     1 92 -93               u=11  imp:n=1 $H2O
c Reg Control Rod
190   9   -2.52   -48 11 -93              u=12  imp:n=1 $B4C
191   6   -2.7     48 -49 11 -93          u=12  imp:n=1 $Al Cladding
```

```
192   4   -1.0    (49 -50 11 -93):(-50 -11 92)  u=12  imp:n=1 $H2O
193   6   -2.7     50 -1 92 -93             u=12  imp:n=1 $Al Tube
194   4   -1.0     1 92 -93                 u=12  imp:n=1 $H2O
c Empty Control Rod
196   4   -1.0    -50 92 -93                u=5   imp:n=1 $H2O
197   6   -2.7     50 -1 92 -93             u=5   imp:n=1 $Al Tube
198   4   -1.0     1 92 -93                 u=5   imp:n=1 $H2O
c Brand New SS Fuel, more U235
c 310 like 100 but mat=5  rho=-5.781        u=15   imp:n=1 $Fuel Meat
c 311 like 101 but                          u=15   imp:n=1 $Up Graphite
c 312 like 102 but                          u=15   imp:n=1 $Down
Graphite
c 313 like 103 but                          u=15   imp:n=1 $Cladding
c 314 like 104 but                          u=15   imp:n=1 $H2O
c Lattice
200   4   -1.0   -101 102 -103 104 -105 106 92 -93    lat=2    u=9
                  fill=-7:7 -7:7 0:0
                  0 0 0 0 0 0 0 9 9 9 9 9 9 9 9
                   0 0 0 0 0 0 9 8 7 8 7 8 7 8 9
                    0 0 0 0 0 9 3 2 3 1 1 3 2 7 9
                     0 0 0 0 9 1 3 1 1 1 1 3 3 8 9
                      0 0 0 9 1 1 3 1 3 1 5 3 3 7 9
                       0 0 9 3 1 1 4 1 2 2 1 2 3 7 9
                        0 9 3 3 1 3 2 2 2 4 2 3 3 7 9
                         9 8 7 2 5 1 1 5 1 1 8 3 1 6 9
                          9 8 7 1 2 2 1 1 1 2 1 3 6 9 0
                           9 8 7 1 3 8 2 2 1 3 1 6 9 0 0
                            9 8 7 1 1 2 2 5 3 3 6 9 0 0 0
                             9 8 7 8 1 1 1 1 1 6 9 0 0 0 0
                              9 8 6 6 6 6 6 6 6 9 0 0 0 0 0
                               9 8 8 8 8 8 8 8 9 0 0 0 0 0 0
                                9 9 9 9 9 9 9 9 0 0 0 0 0 0 0 imp:n=1
201   4   -1.0   -111 112 -113 114 -115 116 92 -93  fill=9 imp:n=1
$Lattices
202   6   -2.7   (-121 122 -123 124 -125 126) 91 -94
                 (111:-112:113:-114:115:-116)         imp:n=1      $Al
Wall
203   6   -2.7   -111 112 -113 114 -115 116 91 -92 imp:n=1   $Lower Al
Plate
204   6   -2.7   -111 112 -113 114 -115 116 93 -94 41 43 45
                                                   imp:n=1 $Upper Al
Plate
206   4   -1.0   -131 94 -97 41 43 45 imp:n=1              $Top Water
207   4   -1.0   -131 96 -91                 imp:n=1       $Bottom
Water
208   10  -2.30  -131 -96 95                 imp:n=1       $Bottom
Concrete
301   9   -2.52  -40 93 -97      imp:n=1 $Safety Rod above core region
302   6   -2.7    40 -41 93 -97  imp:n=1
303   9   -2.52  -42 93 -97      imp:n=1 $Shim Rod above core region
304   6   -2.7    42 -43 93 -97  imp:n=1
305   9   -2.52  -44 93 -97      imp:n=1 $Reg Rod above core region
306   6   -2.7    44 -45 93 -97  imp:n=1
c FNIF
400   11  -0.00115   -141        imp:n=1 $ FNIF Air
401   8  -11.34  -140 141        imp:n=1 $ FNIF Pb
c Heavy water block
```

```
500   11   -0.00115   -159 160 -161 imp:n=1 $ Heavy water Air
501   6    -2.7    159 -158 160 -161 imp:n=1
502   7    -1.056 158 154 -155 156 157 160 -161  imp:n=1
503   6    -2.7    (-154:155:-156:-157)
                    150 -151 152 153 160 -161    imp:n=1
c Al Tank
600   6    -2.7  131 -132 -97 95                imp:n=1  $ Al tank
c
900   4    -1.0    -131 91 -94 140
                   (-150:151:-152:-153:-160:161)
                   (121:-122:123:-124:125:-126)    imp:n=1  $Water
Arround Core
999   0             132:-95:97   imp:n=0

C Surface Cards
1      cz      1.873  $Outer Radius
2      cz      1.82   $Inner Radius
3      cz      1.79   $Inner Radius for Aluminum Container
11     pz    -19.05   $SS Fuel Meat Bottom (7.5 inch * 2)
12     pz     19.05   $SS Fuel Meat Top
13     pz    -29.21   $SS Fuel Graphite Bottom (4 inch)
14     pz     29.21   $SS Fuel Graphite Top
15     pz    -30.39   $SS Cladding Bottom (1.18 cm)
16     pz     30.39   $SS Cladding Top (1.18 cm)
21     pz    -17.78   $Al Fuel Meat Bottom (7 inch * 2)
22     pz     17.78   $Al Fuel Meat Top
23     pz    -27.94   $Al Fuel Graphite Bottom (4 inch)
24     pz     27.94   $Al Fuel Graphite Top
25     pz    -29.12   $Al Cladding Bottom (1.18 cm)
26     pz     29.12   $Al Cladding Top (1.18 cm)
40     c/z     6.555 -11.354 1.00  $Safety Control Rod
41     c/z     6.555 -11.354 1.11  $Safety Control Rod Cladding
42     c/z   -13.11     0.0   1.00  $Shim Control Rod
43     c/z   -13.11     0.0   1.11  $Shim Control Rod Cladding
44     c/z     6.555  11.354 0.200 $Reg Control Rod
45     c/z     6.555  11.354 0.318 $Reg Control Rod Cladding
46     cz      1.00   $ Safety and Shim Rod in Unit
47     cz      1.11   $ Safety and Shim Rod in Unit Cladding
48     cz      0.200  $ Reg Rod in Unit
49     cz      0.318  $ Reg Rod in Unit Cladding
50     cz      1.750  $ Inner radius of Al tube for control rod
91     pz    -33.43   $Lower Plate Bottom
92     pz    -30.89   $Lower Plate Top (1 inch)
93     pz     30.89   $Upper Plate Bottom
94     pz     32.79   $Upper Plate Top (0.75 inch)
95     pz    -55.0    $Concrete Bottom
96     pz    -51.12   $Water Bottom (2 inch)
97     pz     45.0    $Water Top
C Lattice Cells
101    px     2.185
102    px    -2.185
103    p      0.5    0.8660254  0    2.185
104    p      0.5    0.8660254  0   -2.185
105    p     -0.5    0.8660254  0    2.185
106    p     -0.5    0.8660254  0   -2.185
c Frame Boundary
111    p      1.732038    1    0    50.460
```

```
112   p      1.732038    1   0   -50.460
113   p      1.732038   -1   0    50.460
114   p      1.732038   -1   0   -50.460
115   py   25.230
116   py  -25.230
c Al Wall
121   p      1.732038    1   0    54.270
122   p      1.732038    1   0   -54.270
123   p      1.732038   -1   0    54.270
124   p      1.732038   -1   0   -54.270
125   py   27.135
126   py  -27.135
C Reflector Surfaces
131   cz   116.046       $ Water reflector
132   cz   116.84        $ Al Tank
c 131   p      1.732038    1     0  83.259682
c 132   p      1.732038    1     0 -83.259682
c 133   p      1.732038   -1     0  83.259682
c 134   p      1.732038   -1     0 -83.259682
c 135   py   41.629841
c 136   py  -41.629841
c FNIF
140   BOX -15.88 -26.77 -30.48   -15.24 26.40 0   -22.00 -12.7 0   0 0
60.96
141   BOX -22.82 -24.91 -30.48   -10.16 17.60 0    -8.80  -5.1 0   0 0
60.96
c Heavy water beside core
150   p      1.732038    1   0    54.270  $ Al outer
151   p      1.732038    1   0    84.670
152   py     0.0
153   p      1.732038   -1   0     0.0
154   p      1.732038    1   0    54.670  $ Al outer
155   p      1.732038    1   0    84.270
156   py     0.2
157   p      1.732038   -1   0     0.4
158   c/z    30.08 17.37 5.7  $ Air tub Al wall
159   c/z    30.08 17.37 5.5  $ Air tub
160   pz    -30.0            $ Heavy water top
161   pz     30.0            $ Heavy water bottom
201 pz -18.5
202 pz -17.5
203 pz -16.5
204 pz -15.5
205 pz -14.5
206 pz -13.5
207 pz -12.5
208 pz -11.5
209 pz -10.5
210 pz -9.5
211 pz -8.5
212 pz -7.5
213 pz -6.5
214 pz -5.5
215 pz -4.5
216 pz -3.5
217 pz -2.5
218 pz -1.5
```

```
219 pz -0.5
220 pz 0.5
221 pz 1.5
222 pz 2.5
223 pz 3.5
224 pz 4.5
225 pz 5.5
226 pz 6.5
227 pz 7.5
228 pz 8.5
229 pz 9.5
230 pz 10.5
231 pz 11.5
232 pz 12.5
233 pz 13.5
234 pz 14.5
235 pz 15.5
236 pz 16.5
237 pz 17.5
238 pz 18.5
239 pz 40.1                  $ SSW surface
====================================

mode n p
kcode 1000000 1.0 10 500
ksrc   -15.2950   -18.9227    0.0000
      -10.9250   -18.9227    0.0000
       -6.5550   -18.9227    0.0000
       -2.1850   -18.9227    0.0000
        2.1850   -18.9227    0.0000
        6.5550   -18.9227    0.0000
       10.9250   -18.9227    0.0000
      -17.4800   -15.1381    0.0000
      -13.1100   -15.1381    0.0000
       -8.7400   -15.1381    0.0000
       -4.3700   -15.1381    0.0000
        0.0000   -15.1381    0.0000
        4.3700   -15.1381    0.0000
        8.7400   -15.1381    0.0000
       13.1100   -15.1381    0.0000
      -19.6650   -11.3536    0.0000
      -15.2950   -11.3536    0.0000
      -10.9250   -11.3536    0.0000
       -6.5550   -11.3536    0.0000
       -2.1850   -11.3536    0.0000
        2.1850   -11.3536    0.0000
       10.9250   -11.3536    0.0000
       15.2950   -11.3536    0.0000
      -21.8500    -7.5691    0.0000
      -17.4800    -7.5691    0.0000
      -13.1100    -7.5691    0.0000
       -8.7400    -7.5691    0.0000
       -4.3700    -7.5691    0.0000
        0.0000    -7.5691    0.0000
        4.3700    -7.5691    0.0000
        8.7400    -7.5691    0.0000
       13.1100    -7.5691    0.0000
```

```
          17.4800    -7.5691     0.0000
         -24.0350    -3.7845     0.0000
         -19.6650    -3.7845     0.0000
         -15.2950    -3.7845     0.0000
         -10.9250    -3.7845     0.0000
          -6.5550    -3.7845     0.0000
          -2.1850    -3.7845     0.0000
           2.1850    -3.7845     0.0000
           6.5550    -3.7845     0.0000
          10.9250    -3.7845     0.0000
          15.2950    -3.7845     0.0000
          19.6650    -3.7845     0.0000
         -17.4800     0.0000     0.0000
          -8.7400     0.0000     0.0000
          -4.3700     0.0000     0.0000
           4.3700     0.0000     0.0000
           8.7400     0.0000     0.0000
          17.4800     0.0000     0.0000
          21.8500     0.0000     0.0000
         -15.2950     3.7845     0.0000
         -10.9250     3.7845     0.0000
          -6.5550     3.7845     0.0000
          -2.1850     3.7845     0.0000
           2.1850     3.7845     0.0000
           6.5550     3.7845     0.0000
          10.9250     3.7845     0.0000
          15.2950     3.7845     0.0000
          19.6650     3.7845     0.0000
         -13.1100     7.5691     0.0000
          -8.7400     7.5691     0.0000
           0.0000     7.5691     0.0000
           4.3700     7.5691     0.0000
           8.7400     7.5691     0.0000
          13.1100     7.5691     0.0000
          17.4800     7.5691     0.0000
         -10.9250    11.3536     0.0000
          -6.5550    11.3536     0.0000
          -2.1850    11.3536     0.0000
           2.1850    11.3536     0.0000
          10.9250    11.3536     0.0000
          15.2950    11.3536     0.0000
          -4.3700    15.1381     0.0000
           0.0000    15.1381     0.0000
           4.3700    15.1381     0.0000
           8.7400    15.1381     0.0000
          13.1100    15.1381     0.0000
m1        1001.66c    -0.015896   $ new SS meat, H/Zr=1.6. 0.59% burn-up
         40000.66c    -0.899104
         92235.66c    -0.016728
         92238.66c    -0.068272
mt1       h/zr.60t
          zr/h.60t
m2        6000.66c     1.0        $ graphite
mt2       grph.60t
m3        6000.66c    -0.0004     $ ss cladding
         14000.60c    -0.0046
         24000.50c    -0.190
```

```
        25055.66c    -0.009
        26000.50c    -0.699
        28000.50c    -0.097
m4       1001.66c     2.0        $ H2O
         8016.66c     1.0
mt4      lwtr.60t
m5       1001.66c    -0.010   $ Al meat, H/Zr=1.0, 8.91% burnup
        40000.66c    -0.905
        92235.66c    -0.01533
        92238.66c    -0.06967
mt5      h/zr.60t
         zr/h.60t
m6      13027.66c     1.0        $ Al
m7       1001.66c     0.64       $ D20 (68% atom)
         1002.66c     1.36
         8016.66c     1.00
mt7      lwtr.60t
         hwtr.60t
m8      82000.50c     1.0        $ Pb
m9       5010.66c    -0.1566     $ b4c
         5011.66c    -0.6264
         6000.66c    -0.217
m10      1001.66c    -0.00619    $ Concrete
         6000.66c    -0.17520
         8016.66c    -0.41020
        11023.66c    -0.00027
        12000.66c    -0.03265
        13027.66c    -0.01083
        14000.60c    -0.03448
        19000.66c    -0.00114
        20000.66c    -0.32130
        26000.50c    -0.00778
m11      7014.66c     0.0000381259 $Air
         8016.66c     0.0000095012
        18000.59c     0.0000001664
m12      1001.66c    -0.015896   $ Old SS meat, H/Zr=1.6, 8.77% burnup
        40000.66c    -0.899104
        92235.66c    -0.015354
        92238.66c    -0.069646
mt12     h/zr.60t
         zr/h.60t
phys:n 20
phys:p 30
pwt                                  $ 1 photon per n collision
(defualt)
FMESH4:n GEOM=rec ORIGIN=-117 -117 38    $ bottom, left, behind
           IMESH=117 IINTS=234         $ 234 bins from x = -117 to 117
           JMESH=117 JINTS=234         $ 234 bins from y = -117 to 117
           KMESH=39                   $ 1 bin from z = 38 to 39
           OUT=ij                     $ output = array 54,756
#    de4       df4        $ n dose function table H-1
     log       log        $ biological dose equivalent dose rate
factors
     2.5E-8    3.67E-6
     1.0E-7    3.67E-6
     1.0E-6    4.46E-6
     1.0E-5    4.54E-6
```

```
     1.0E-4     4.18E-6
     1.0E-3     3.76E-6
     1.0E-2     3.56E-6
     1.0E-1     2.17E-5
     5.0E-1     9.26E-5
     1.0        1.32E-4
     2.5        1.25E-4
     5.0        1.56E-4
     7.0        1.47E-4
     10.0       1.47E-4
     14.0       2.08E-4
     20.0       2.27E-4
fm4 3.061E15    $ =P(90,000W)/(1.602E-19*Fission Rate*Fission E dep in
core)
FMESH24:p GEOM=rec ORIGIN=-117 -117 39  $ bottom, left, behind
           IMESH=117 IINTS=234         $ 234 bins from x = -117 to 117
           JMESH=117 JINTS=234         $ 234 bins from y = -117 to 117
           KMESH=40                   $ 1 bin from z = 39 to 40
           OUT=ij
#      de24       df24                $ p dose functions table H-2
       log        log
       0.01       3.96E-6
       0.03       5.82E-7
       0.05       2.90E-7
       0.07       2.58E-7
       0.1        2.83E-7
       0.15       3.79E-7
       0.2        5.01E-7
       0.25       6.31E-7
       0.3        7.59E-7
       0.35       8.78E-7
       0.4        9.85E-7
       0.45       1.08E-6
       0.5        1.17E-6
       0.55       1.27E-6
       0.6        1.36E-6
       0.65       1.44E-6
       0.7        1.52E-6
       0.8        1.68E-6
       1.0        1.98E-6
       1.4        2.51E-6
       1.8        2.99E-6
       2.2        3.42E-6
       2.6        3.82E-6
       2.8        4.01E-6
       3.25       4.41E-6
       3.75       4.83E-6
       4.25       5.23E-6
       4.75       5.60E-6
       5.0        5.80E-6
       5.25       6.01E-6
       5.75       6.37E-6
       6.25       6.74E-6
       6.75       7.11E-6
       7.5        7.66E-6
       9.0        8.77E-6
       11.0       1.03E-5
```

```
      13.0       1.18E-5
      15.0       1.33E-5
fm24 3.061E15
c
c
c ============== Surface Source Write Card SSW
===============================
ssw 239             $ Surface Source Write card for surface 239 at z =
40.5 cm
c
```