# U.S. GEOLOGICAL SURVEY
# RESEARCH REACTOR
# LICENSE NO. R-113
# DOCKET NO. 50-274

# REVISED SAFETY ANALYSIS REPORT, TECHNICAL SPECIFICATIONS, AND ENVIRONMENTAL REPORT

## REDACTED VERSION*

## SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

# United States Department of the Interior

U.S. GEOLOGICAL SURVEY
Office of the Director
Reston, Virginia 20192

In Reply Refer To:
Mail Stop 911
2009134-DO

JAN 0 5 2009

U.S. Nuclear Regulatory Commission
ATTN: Mr. Alexander Adams
Office of Nuclear Reactor Regulation
Mail Stop 12 G13
Washington, D.C. 20555

Reference: U.S. Geological Survey TRIGA Reactor (GSTR)
Docket No. 50-274, License No. R-113

Dear Mr. Adams:

The U.S. Geological Survey (USGS) hereby requests a renewal of Operating License No. R-113 for the GSTR, a Class 104 license utilized for research and education. Our existing license expires on February 24, 2009. The requested renewal term for License R-113 is for 20 years.

Enclosed you will find copies of the updated safety analysis report (SAR), technical specifications, and the environmental report. Both the SAR and the technical specifications conform to the format given in NUREG-1537. Financial qualifications are addressed in chapter 15 of the SAR. We do not wish to make any changes to the GSTR Emergency Plan, Physical Security Plan, or the Operator Requalification Program as they are current and up-to-date. If you have any questions pertaining to this matter, please direct them to Mr. Tim DeBey, U.S. Geological Survey, Box 25046 MS-974, Denver, Colorado 80225, (303) 236-4726, or via e-mail at tdebey@usgs.gov.

In addition to the license renewal documents, I have reaffirmed our Nuclear Regulatory Commission signature authority memorandum and our commitment to keeping radiation exposures as low as practical and to fostering a safety-conscious work environment at our reactor facility.

A020

Mr. Alexander Adams

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

Executed on: 5th January 2009.

County/City of Fairfax VA
Commonwealth/State of Virginia
The foregoing instrument was subscribed and sworn before me this 5th day of January 2010, by
Mark Myers
(name of person seeking acknowledgement)
Notary Public
My commission expires: 30 June 2010
Notary Registration Number 106857

Sincerely,

Mark D. Myers
Director

Enclosures

Copy to: Document Control Desk, USNRC
(w/o attachments)
Craig Bassett, USNRC
Mark D. Myers, USGS
Tim DeBey, USGS

# Relicensing Application for the U.S. Geological Survey TRIGA Reactor – R113

December 2008

# Safety Analysis Report

USGS TRIGA Reactor

November 25, 2008

# Contents

# List of Figures

# List of Tables

# Chapter 1

# THE FACILITY

## 1.1 Introduction

This safety analysis report supports an application to the U.S. Nuclear Regulatory Commission (NRC) by the United States Geological Survey (USGS) for the utilization of a 1.0 MW TRIGA®-fueled Mark I research reactor. The reactor is owned and operated by the USGS for the purpose of performing neutron irradiation services for a wide variety of scientific applications. The reactor is located within the Denver Federal Center in Lakewood, Colorado. The reactor is known as the Geological Survey TRIGA® Reactor (GSTR). Being a TRIGA® reactor, the GSTR has an inherent safety feature of a prompt negative temperature coefficient built into the nuclear fuel; more information regarding the fuel design can be found in Chapters 3 and 6.

This document addresses only the safety issues associated with the operation of the GSTR. This document reflects experience with the operation and performance of the reactor systems, radiation surveys, and personnel exposure histories related to the operation of the GSTR.

## 1.2 Summary and Conclusions on Principal Safety Considerations

The GSTR reactor is located within the Nuclear Science Building, Building 15, located on the Denver Federal Center in Lakewood, Colorado . Building 15 was chosen to house the reactor because the personnel working in Building 15 have training and experience working with radioactive materials, and radiation. [redacted]

The GSTR facility has several design features implemented for safe operation and shutdown of the facility. The GSTR cannot be operated without the console key and a password to log into the computer. Also, interlocks prohibit the manual withdraw of more than one control rod in any operating mode. A source interlock prohibits the withdrawal of any control rod when the log scale reads less than $10^{-7}$% power. Another interlock prevents the initiation of square wave mode or pulse mode above a power level of 1 kW.

Similarly when in pulse mode or square wave mode, withdrawal of any control rod besides the transient cylinder is not allowed.

Consequences from the operation and use of non-power reactors include radiation exposure and radioactive material release to the environment. Analyses of possible accident scenarios are included in Chapter 13. The main safety feature of the TRIGA®-Mark I nuclear reactor is the strong, prompt-negative temperature coefficient built into the fuel. The negative temperature coefficient limits the steady-state and peak power achievable, thus preventing fuel damage from credible reactivity accidents. Therefore an inadvertent pneumatic withdrawal of the transient rod, from the core when the core is operating at the power level scram point will not result in fuel damage. Individual experiments are limited to the same reactivity worth limit as the transient rod, to ensure that experiment failure cannot result in more severe transients. The GSTR is designed for a 1.0 MW steady state operation by General Atomics. The licensed operating power level of 1.0 MW results in a decay heat potential in the fuel that is small enough that loss of reactor water does not result in fuel damage. Another safety feature of the GSTR is the regulated pressure of the reactor bay. The reactor bay is at a lower pressure than the control room and Reactor Operator offices. This safety feature ensures any accidental airborne radioactive release is channeled through the reactor bay, preventing contamination of the other spaces in the building.

For the maximum hypothetical accident where the fuel cladding is removed and reactor water is not present, the resulting estimated doses to occupational workers and the general public are well within the annual limits given in 10 CFR 20.

## 1.3 General Description of the Facility

The GSTR is located within Building 15 located on the Denver Federal Center in Lakewood, Colorado. Building 15 is the only building on a square area of land bordered on the north by North Center Avenue, on the east by First Street, on the south by South Center Avenue, and on the west by Second Street. Building 15 is approximately 27000 square feet, and is built in the shape of a "T." [redacted]

Building 15 provides space for about 40 USGS professional and technical personnel engaged in radioactivity measurements and the use of materials ranging in intensity from natural levels to several curies. Building 15 houses a wide variety of radioisotope laboratories, radioisotopes counting labs, and a large inventory of nuclear instrumentation useful for research applications as well as for protection.

The GSTR is owned and operated by the USGS under the NRC License Number R-113 (Docket Number 50-274). It is used for teaching, research, training, and radionuclide production. Nuclear Engineering students perform a number of tests and experiments using the reactor in order to reinforce their class work on reactor theory. Researchers use the rotating specimen rack, the pneumatic transfer system, the cadmium-lined irradiation tube, the out-of-core irradiation tubes, the beam tube, in core irradiation facilities, and the central thimble for experiments involving neutron irradiation. Radionuclides are produced for research, class

applications and commercial use. Research associated with the reactor typically involves isotope production, neutron activation analysis, geochronology, and fission track radiography.

The GSTR is a light-water-cooled, graphite-reflected reactor using less than 20%-enriched uranium-zirconium hydride TRIGA® fuel elements. These fuel elements are placed in a circular grid with more than 20 feet of water over the top of the core. The reactor has a maximum steady-state thermal power of 1.0 MW and has been pulsed to a peak power of 2,500 MW. The general arrangement of the reactor equipment and major structures is seen in Figure 1.1. TRIGA® fuel has a high, prompt negative temperature coefficient built into it with the uranium-zirconium hydride mixture. The high, prompt negative temperature coefficient is a safety feature that limits the steady-state and peak power achievable, thus preventing fuel damage from credible reactivity accidents. The coolant system for the GSTR is composed of three systems that use light-water. There is a primary water system which comes into direct contact with the reactor core and cools it through natural convection. The second system is a secondary system that cools the primary system through conduction in a heat exchanger. The third system is a demineralizer system that removes impurities from the primary water system.

Power level measurements are made with three independent power monitoring channels (NM1000, NP1000, and NPP1000) during steady state and square wave operations. The NPP1000 measures the power level during pulse operations. The nuclear reactor is controlled from the control system console built by General Atomics.

Radioactive material release is measured at the GSTR with several systems. Airborne release from the reactor bay is monitored with the continuous air monitor (CAM) and gaseous stack monitor. There is also a neutron monitor and several gamma-sensitive area monitors located within the reactor bay, as well as two exhaust systems. The primary exhaust system is used for normal operations. The other exhaust system is the emergency exhaust system used for emergency operations. The emergency exhaust filters all exiting air through a prefilter and HEPA filter. Building 15, beyond the reactor bay, is equipped with several gamma monitors and contains several other radiation counters used for experiments and protection. USGS personnel assigned to Building 15 have been trained in the safe use of radiation, in order to maintain exposures within limits as low as reasonably achievable (ALARA).

Significant features of the reactor include:

- three standard control rod and drive systems;
- one transient control rod and drive system; and
- numerous irradiation facilities including a central thimble, a cadmium lined irradiation tube, a pneumatic transfer system, a rotating rack, a pump tube, a beam tube, and other movable dry tubes.

## 1.4 Shared Facilities and Equipment

The GSTR is an integral part of Building 15, and thus shares walls, water supplies, sewage, heating, water distribution, ventilation supply, and main electrical supply with Building 15. The electrical distribution within the GSTR is controlled by the GSTR staff, however, the electrical supply is shared with Building 15.

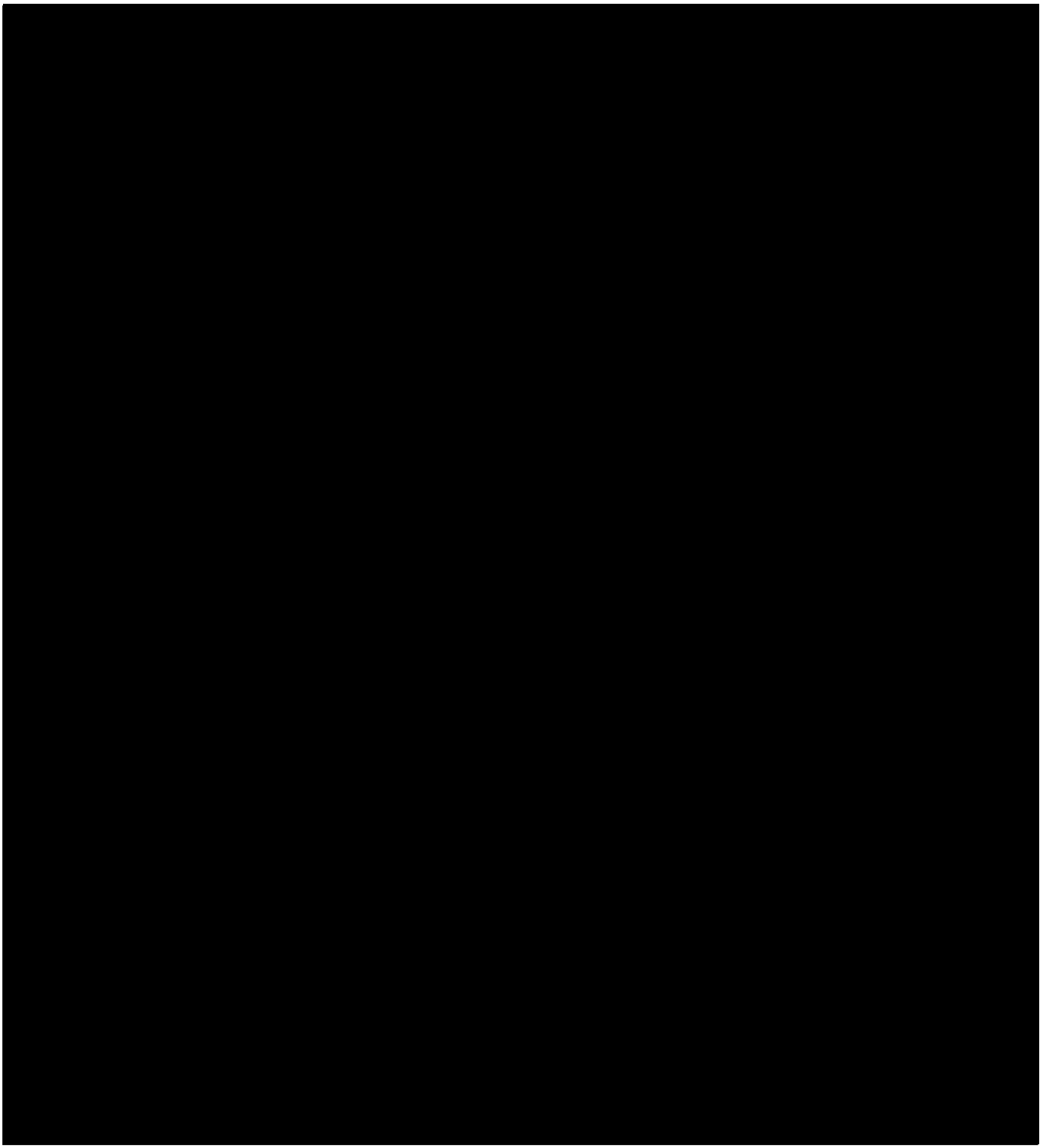

Figure 1.1: Major structures of the GSTR.

The exhaust system for the reactor bay is not shared with Building 15. The ventilation supply dampers are closed during a radioactive release into the reactor bay greater than 10 kcpm on the CAM. Access into the GSTR facility is controlled through locked doors within Building 15.

Within the GSTR facility there is a storage tank located in Room 151. The storage tank holds activated materials that belong to the reactor facility.

## 1.5 Comparison with Similar Facilities

The design of the GSTR is similar to those of approximately 30 TRIGA®-type reactors currently operating world-wide. Three facilities that are nearly identical to the GSTR include the Oregon State University - Corvallis, University of Texas - Austin, and the DOW reactor at the DOW Chemical Company. The GSTR and the three similar facilities all use similar TRIGA® fuel elements, were designed by General Atomics, and use similar control systems from General Atomics. Since most of these reactors have been in operation for more than twenty years, considerable operational information is available and their characteristics are well documented.

## 1.6 Summary of Operations

The GSTR is a unique and valuable tool for a wide variety of research applications and serves as an excellent source of neutrons and/or gamma rays. The GSTR has a number of irradiation facilities providing a wide range of neutron flux levels and neutron flux qualities, which are sufficient to meet the needs of most researchers. The typical operating power level for the GSTR is 1 MW. The average energy output per year is approximately 27 MWd. As an indication of operating tempo, operational statistics for reporting year 2007 are given in Table 1.1. Based on the analysis presented in this SAR, there are no limitations on the operating schedule.

## 1.7 Compliance With the Nuclear Waste Policy Act of 1982

All low level waste generated is disposed at a suitable licensed site. Any high level waste that the GSTR needs to dispose of is disposed of separately at a high level waste disposal sites. The GSTR meets the requirements of the Nuclear Waste Policy Act of 1982 Section 302(b)(1)(B) because it is a government facility operated by the United States Geological Survey under the Department of the Interior.

## 1.8 Major Facility Modifications and History

The initial construction of Building 15 was completed in 1966. Building 15 is approximately 27000 square feet.

Table 1.1: Operating Statistics for Reporting Year 2007.

| Operation Data | Annual Value |
|---|---|
| MWh | 645.3 |
| MWd | 26.9 |
| Number of pulses | 5 |
| Hours reactor critical | 697.6 |
| Hours reactor at full power | 645.3 |
| Number of irradiation requests | 194 |
| Number of samples irradiated | 3,654 |

A brief chronology of the key dates and events in the history of the Building 15 and the GSTR TRIGA® reactor is given below.

**1966** Completion of Building 15 consisting of 27,000 square feet of office and laboratory space.

**January 1967** Initial request for construction permit submitted

**October 1967** Issuance of the construction permit (CPRR-102) for the GSTR

**December 1968** Changes under 10 CFR 50.59 - Several changes during the Construction Permit period. Initial core loading responsibility placed upon reactor staff. Reactor administration changed from a committee to the individual holding the position of Regional Hydrologist. Amplifications of position descriptions concerning the reactor staff. Reactor Operations Committee meets semiannually instead of quarterly. Administrative controls and procedures changed and expanded. Access limitations have been expanded. Formats of routine forms have been altered by the addition of more information. Deletion of obscure statement in RHA. Amplification and change in order of presentation of health physics information presented in ROM. Changes in emergency plans and procedures.

**December 1969** Change under 10 CFR 50.59 - A report is filed with the AEC annually, relating 50.59 changes and additional parameters concerning reactor operations.

**February 1969** Issuance of the operating license (R-113) for the GSTR.

**February 1969** Initial criticality of the GSTR. The reactor was licensed to operate at a maximum steady-state power level of 1.0 MW thermal and was fueled with 8.5% enriched stainless steel fuel rods.

**May 1970** Technical Specification Amendment - Transient rod maintenance requirements. A functional performance of the transient rod shall be performed each day of pulsing. Semiannually, the transient cylinder and associated air supply system will be inspected, cleaned, and lubricated.

| | |
|---|---|
| **May 1970** | License change No.1 - Increase in $^{41}Ar$ release limits. The $^{41}Ar$ release limit from the stack is $4.8\times10^{-6}\mu Ci/mL$ averaged over a year. Efforts will be made to minimize the release of $^{41}Ar$. Periodic monitoring of the stack effluent will be performed to ensure that $^{41}Ar$ is the only routinely exhausted radionuclide. |
| **June 1970** | Change under 10 CFR 50.59 - Minor changes to the Reactor Operations Manual. Changes in titles of positions and additions to include different operating procedures. |
| **December 1970** | Change under 10 CFR 50.59 - Addition of a light on the control systems console. |
| **June 1971** | Technical Specification Amendment (Change No.2) - Received instructions regarding the use of explosives in irradiation facilities. The use of explosives is considered an un-reviewed safety question. |
| **June 1971** | Change under 10 CFR 50.59 - Replacement of the truck door with standard double-doors with sealing provision. |
| **April 1972** | Change under 10 CFR 50.59 - Installation of a fast pneumatic transfer system. |
| **January 1974** | Technical Specification Amendment (Change No.3) - Reactor area monitor alarms audibly vs. sounding evacuation alarm. The signal from reactor area radiation monitors would actuate an audible alarm rather than an evacuation alarm. |
| **November 1975** | Technical Specification Amendment No.1 (Change No.4) - Fuel element measurements required. Fuel elements are to be checked for transverse bend and longitudinal elongation after the first 100 pulses and after every 500 pulses or every 60 months, which ever comes first. |
| **January 1978** | Change under 10 CFR 50.59 - Proposed change to the reactor facility. Addition of a apparatus fitted to the outside of the graphite reflector to accommodate pneumatic transfer termini. |
| **August 1978** | Technical Specification Amendment No.2 - Surveillance pulsing requirements added. Operating in pulse mode during intervals of less than six months the reactor will be semiannually pulsed with a reactivity of insertion of 1.5% $\Delta k/k$ to compare fuel temperature and peak power levels with previous pulses of same reactivity. The same comparison will be done for the first pulse when the reactor has not pulsed within six months. |
| **July 1979** | Technical Specification Amendment No.3 - Increase fuel possession limit to 5 kg of $^{235}U$. The GSTR is licensed to possess 5.0 kg of less than 20% enriched $^{235}U$. |
| **1981** | Change under 10 CFR 50.59 - Changed requirements on personnel present during reactor operations. At least two persons must be in the reactor facility protected area during reactor operations. One of those persons must be an NRC licensed operator. |
| **February 1981** | Change under 10 CFR 50.59 - The operation of the reactor with a metal scribe dropped into the reactor tank does not constitute an un-reviewed safety question. |

**February 1984** Change under 10 CFR 50.59 - Replacement of old rotating specimen rack with new rotating pneumatic specimen rack.

**September 1987** New control system console installation process started with a contract issued from General Atomics.

**January 1988** Technical Specification Amendment No.4 - Surveillance requirement changes, high power trip testing change, square wave mode definition. GSTR can be operated at steady state power levels not in excess of 1000 kW and in the pulse mode with reactivity insertions not to exceed 2.1%$\Delta$k/k. Testing of safety circuitry will be done with the introduction of an electrical signal. During the replacement of the tank liner certain technical specifications measurements are exempt, such as measuring the pool water conductivity because the pool water will be below the intake.

**August 1988** Change under 10 CFR 50.59 - Installation of new reactor liner due to corrosion of old liner.

**December 1988** Technical Specification Amendment No.5 - Increase fuel possession limit to 9 kg of $^{235}$U. The GSTR is licensed to possess 9.0 kg of less than 20% enriched $^{235}$U.

**1989** Change under 10 CFR 50.59 - Changed requirements on personnel present during reactor operations. At least two reactor staff members must be present at reactor site during reactor operations. The operator at the console shall be an NRC licensed operator and must know the location of the second person. The Senior Reactor Operator in charge must be one of the two staff members on site.

**March 1991** Technical Specification Amendment No.6 - Installation of microprocessor based instrumentation control system. Includes safety channels indicating the actual power level as determined by a thermal power measurement. Fuel elements and fueled devices shall be stored in an array for natural convection cooling such that the temperature reached will not exceed design values. Addition to technical specifications of safety channel 1 and 2, preset time, control systems console (CSC) and data acquisition control (DAC) watchdog, and neutron source interlock.

**October 1991** Change under 10 CFR 50.59 - Evaluation of 8" vertical beam tube at USGS TRIGA® MK I Facility. Use of an 8" vertical beam tube for an additional irradiation facility was authorized.

**November 1991** Change under 10 CFR 50.59 - Evaluation of in-core irradiation tubes at USGS TRIGA® facility. Installation and use of two in core irradiation tubes.

**December 1991** Change under 10 CFR 50.59 - Evaluation of low carbon steel irradiation at the GSTR facility. Evaluation of experiment irradiating low carbon steel specimens.

**January 1992** Technical Specification Amendment No.7 - Require biennial audit of Emergency Plan, Emergency procedures, and Physical Security Plan. The reactor facility emergency

plan, emergency plan, emergency procedures, and physical security plan shall be audited by the Reactor Operations Committee at least biennially, with the interval not to exceed 30 months.

**November 1992** Change under 10 CFR 50.59 - Changed mounting for neutron detectors. The new mounts are of aluminum and stainless steel, and allows for easy and quick vertical adjustments without working under the pool water.

**October 1995** Change under 10 CFR 50.59 - Changed requirements on personnel present during reactor operations. At least two staff members on duty during reactor operations. The operator must be an NRC licensed operator. The second person must be able to reach facility within 15 minutes after being contacted by console operator. A method of communication must be available at the console in order to contact the second person within a 5 minute time period. The Senior Reactor Operator in charge must be one of the two staff members and be at the facility during startup, approach to power, fuel movement, and recovery from unplanned shutdown.

**March 1998** Technical Specification Amendment No.8 - Allow use of 12 wt% fuel in core. The use of 12 wt% fuel elements would be in addition to the use of 8.5 wt% fuel elements. A limit would be set of 22 kW max power in one fuel element during 1 MW thermal power operation. Therefore, power calculations are required for steady state power above 100 kW with a core size of less than 100 elements. Also, during pulse mode the temperature of a 12 wt% fuel element would be measured if any 12 wt% fuel elements are in the core.

**April 1998** License Amendment No.9 - Change reporting requirements to delete reference to NRC Region IV office.

**January 1999** Change under 10 CFR 50.59 - Electronics in the CAM were replaced with similar newer digital display electronics.

**November 1999** Change under 10 CFR 50.59 - Upgraded the control system console and DAC computers. The control system console software was upgraded from IC-DOS to QNX operating system. Network cards and computer chassis in the CSC and DAC were replaced.

**October 1999** Change under 10 CFR 50.59 - Secondary cooling pump and motor assembly were replaced.

**October 2001** Change under 10 CFR 50.59 - New console upgrade. Console upgrade required for Y2K compliance.

**October 2001** Change under 10 CFR 50.59 - Replaced the chart recorder located on the control systems console with a digital display, four channel strip recorder.

**June 2005** License Amendment No.10 - Extend license period to end February 24, 2009. License expiration date moved from October 10, 2007 to February 24, 2009 to recapture time of construction of facility.

**January 2006** License Amendment No.11 - Allow use of aluminum clad 8 wt% fuel elements. Aluminum clad fuel elements will only be allowed in the F and G rings. The reactor will not be operated in any manner that would cause any stainless-steel clad fuel element to produce a calculated steady state power level in excess of 22 kW. A stainless steel element in the B ring will be limited to 735 °C and in the C ring to 652 °C. During the first 5 years measurements will be made on 20% of the aluminum clad fuel elements. After the first five years the measurement cycle will revert back to the standard 60 month schedule. Fuel elements removed from service do not need to be measured.

**June 2007** Started classes for Colorado School of Mines Nuclear Engineering program. Contributed to the establishment of a new Nuclear Engineering program in Colorado.

# Chapter 2

# SITE CHARACTERISTICS

This chapter describes the site characteristics for the vicinity of the GSTR on the Denver Federal Center, Lakewood, Colorado, and their relationship to the safety and operation of the US Geological Survey TRIGA® reactor. The conclusion reached in this chapter, and throughout this document, is that the selected site is well-suited for the GSTR when considering the relatively benign operating characteristics of the reactor, including the Maximum Hypothetical Accident (MHA). This is consistent with the conclusions reached for the other ~30 TRIGA® reactors operating throughout the world. Many of them are located on university campuses, in hospitals, and other highly populated areas.

## 2.1 Geography and Demography

### 2.1.1 Site Location and Description

The GSTR is located at the Denver Federal Center which is in Lakewood, Colorado. Lakewood is located approximately 4 miles south of Interstate 70 and 10 miles west of downtown Denver.

#### 2.1.1.1 Specification and Location

The GSTR is located near the northeast corner of the Denver Federal Center which is northwest of downtown Lakewood. [redacted]. The Denver Federal Center is in the city of Lakewood, which is in Jefferson County. The city has a total area of 110.0 $km^2$ (42.5 $mi^2$), with 107.7 $km^2$ (41.6 $mi^2$) of land and 2.3 $km^2$ (0.9 $mi^2$) of water.

#### 2.1.1.2 Boundary and Zone Area Maps

Figures 2.1 and 2.2 illustrate the location of the GSTR with respect to the State of Colorado and the Denver Federal Center. An enlarged photograph of the facility can be seen in Figure 2.3.

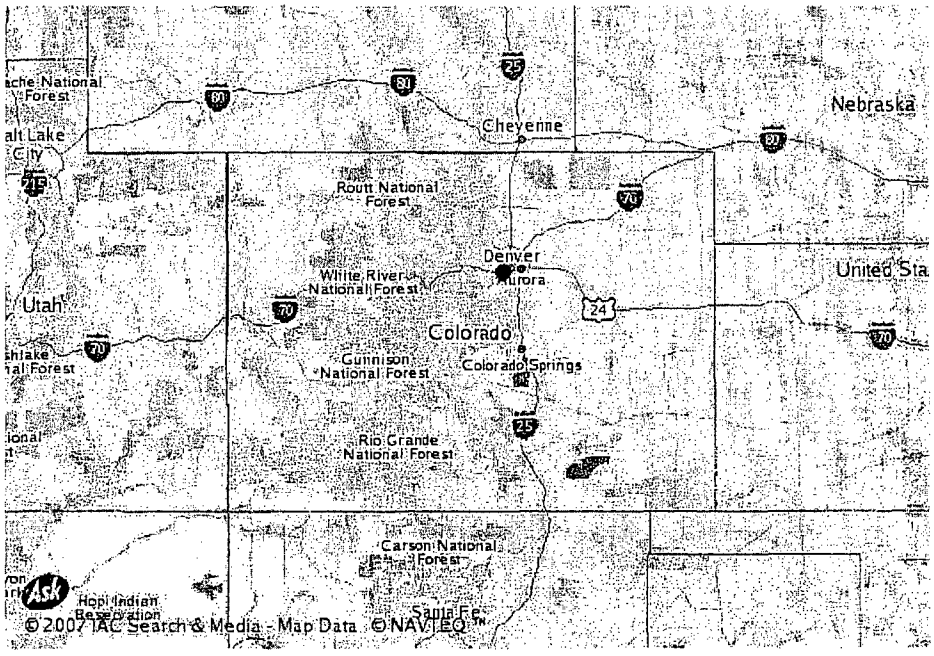


Figure 2.1: Map of Colorado showing the location of the DFC as represented by a black dot.

#### 2.1.1.3 Population Distribution

The GSTR is located at the Denver Federal Center in the city of Lakewood, Jefferson County, Colorado. Metropolitan Lakewood had a population of 144,126 in 2000, which has increased by 9% over the ten preceding years. The population of Lakewood resides primarily to the south and east of the GSTR. The total population in 2000 of all of Jefferson County was 527,056. The Denver Federal Center has a daytime population of approximately 6,200 people.

#### 2.1.1.4 Population Adjacent to the Denver Federal Center

| Census Tract | Population |
|---|---|
| 109.01 | 3195 |
| 109.02 | 2611 |
| 111.00 | 6454 |
| 112.02 | 4657 |
| 117.01 | 4515 |
| 117.08 | 4450 |
| 117.23 | 3165 |
| 117.22 | 6229 |

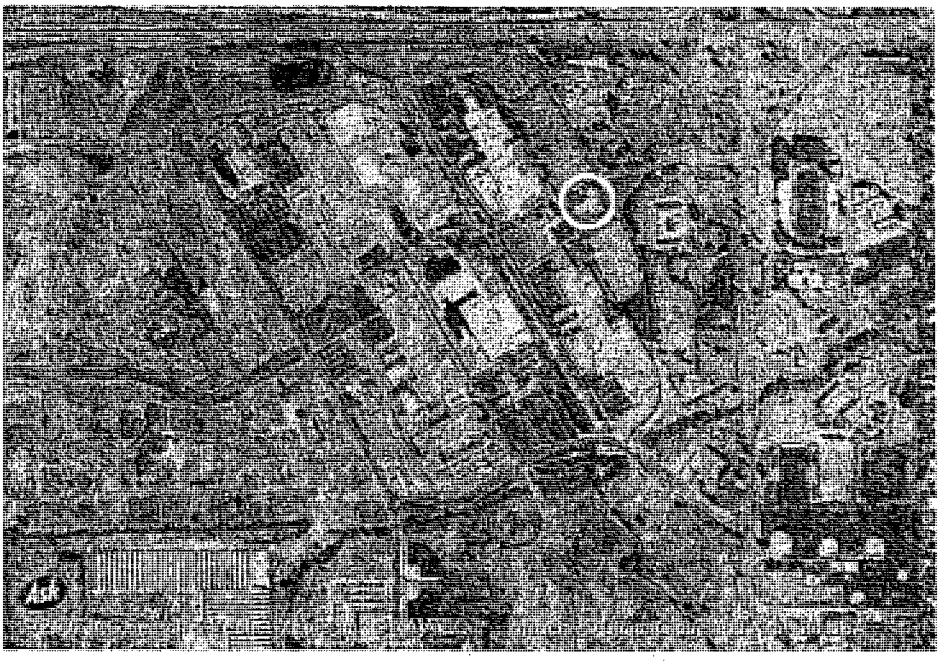

Figure 2.2: Map of Denver Federal Center showing the location of Building 15 circled in white.

## 2.2 Nearby Industrial, Transportation, and Military Facilities

### 2.2.1 Locations and Routes

There are no refineries, railways, chemical plants, mining facilities, manufacturing facilities, water transportation routes, fuel storage facilities, or rail yards located within 3 km of the GSTR. There is a National Guard motor pool north of GSTR on the Denver Federal Center. A regional transportation hub is being built in the northwest corner of the Federal Center and it will have connections for local public transportation buses as well as a light rail connection.

### 2.2.2 Air Traffic

There are no airports within sixteen kilometers of the GSTR.

### 2.2.3 Analysis of Potential Accidents at Facilities

Local buildings around the GSTR are Buildings 16, 20, 21, and 25. Building 16 contains offices of the Defense Contract Management Agency and the other buildings contain USGS offices and laboratories.

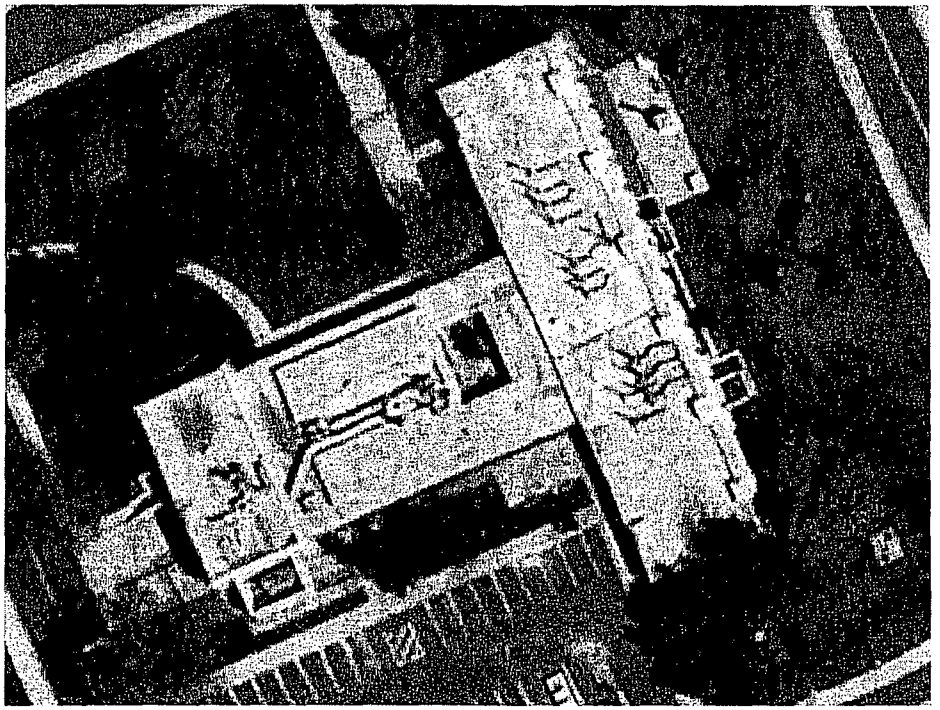

Figure 2.3: Building 15 GSTR

#### 2.2.3.1 USGS Building 20

Building 20 is located southwest of GSTR. Included in this facility are a Library, research laboratories, and administration offices. This building's activities pose no threat to the GSTR.

#### 2.2.3.2 USGS Building 21

Building 21 is located west of GSTR. Included in this facility are research laboratories, and office spaces. This facility's activities pose no threat to the GSTR.

#### 2.2.3.3 USGS Building 25

Building 25 is located northwest of GSTR. Included in this facility are research laboratories and office spaces. This facility's activities pose no threat to the GSTR.

#### 2.2.3.4 GSA Building 16

Building 16 is located southeast of GSTR. Included in this facility are office spaces. This building's activities pose no threat to the GSTR.

## 2.3 Meteorology

### 2.3.1 General and Local Climate

The city of Lakewood is located in Jefferson County, on the eastern plain of Colorado, adjacent to the foothills of the Rocky Mountains. The climate is considered mild, characterized by cool, wet winters and warm, dry summers.

#### 2.3.1.1 Humidity

Relative humidity is highest during early morning and late evening hours and is generally 40-68% throughout the year. During the afternoon, humidity is generally lowest, ranging from 45-49% during winter to 30-40% during summer.

#### 2.3.1.2 Wind Stability

Wind data are readily available from first order stations (operated by the National Weather Service and fully-instrumented), the closest being at the Denver International Airport. Annual average data from this station are available from 1995 to present.

#### 2.3.1.3 Temperatures

Temperature values for the Denver area are shown in Table 2.1[†]. Normal and mean temperatures are provided from 1971 to 2000. Extreme temperatures are provided from 1872 to present. Monthly averages and daily extremes are given for each month. The lowest monthly normal minimum daily temperature is 15.2 °F in January and the highest monthly normal maximum daily temperature is 88.0 °F in August. Extreme temperatures have ranged from -29.0 °F to 105.0 °F.

#### 2.3.1.4 Precipitation

Precipitation values, also taken from 1971 to 2000, are shown in Table 2.2. The normal precipitation for the Denver area is 15.81 inches per year, with a maximum of 23.31 inches in 1967. The maximum monthly rainfall was 8.75 inches in May of 1876. The maximum monthly snowfall was 57.4 inches in December, 1913.The annual average snow fall is 61.6 inches.

[†]0.* OR * means the value is between 0.0 and 0.4

Table 2.1: Temperature data for the GSTR

NORMALS AND MEANS 1971–2000 & EXTREMES 1872–PRESENT
(UPDATED 6/30/2005)
DENVER INTERNATIONAL AIRPORT, CO, LAT. 39.52N, LON. 104.40W, ELEV. 5431 FT.

| NORMALS | JAN | FEB | MAR | APR | MAY | JUN | JUN | AUG | SEP | OCT | NOV | DEC | AVG |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| TEMPS | | | | | | | | | | | | | |
| DAY MAX | 43.2 | 47.2 | 53.7 | 60.9 | 70.5 | 82.1 | 88.0 | 86.0 | 77.4 | 66.0 | 51.5 | 44.1 | 64.2 |
| DAY MIN | 15.2 | 19.1 | 25.4 | 34.2 | 43.8 | 53.0 | 58.7 | 57.4 | 47.3 | 35.9 | 23.5 | 16.4 | 35.8 |
| MONTHLY | 29.2 | 33.2 | 39.6 | 47.6 | 57.2 | 67.6 | 73.4 | 71.7 | 62.4 | 51.0 | 37.5 | 30.3 | 50.1 |
| EXTREMES | | | | | | | TIED | | TIED | | TIED | | MAX |
| REC HIGH | 76 | 77 | 84 | 90 | 95 | 104 | 103 | 105 | 97 | 90 | 79 | 79 | 105 |
| DAY | 27TH | 4TH | 26TH | 30TH | 26TH | 8TH | 8TH | 4TH | 1ST | 14TH | 5TH | AUG | |
| YEAR | 1888 | 1890 | 1971 | 1992 | 1942 | 1994 | 1989 | 1878 | 1ST | 1892 | 1990 | 1939 | 1878 |
| | | | | | | | 6TH | *MAX | 1995 | | 19TH | | |
| | | | | | | | 1973 | | MORE | | 1989 | | |
| EXTREMES | | TIED | | | | | TIED | TIED | | | | TIED | MIN |
| REC LOW | -29 | -25 | -11 | -2 | 19 | 30 | 42 | 40 | 17 | -2 | -18 | -25 | -29 |
| DAY | 9TH | 1ST | 28TH | 2ND | 3RD | 2ND | 4TH | 26TH | 29TH | 29TH | 22ND | JAN | |
| YEAR | 1875 | 1951 | 1886 | 1975 | 1907 | 1951 | 1903 | 25TH | 1985 | 1917 | 1877 | 1990 | 1875 |
| | *MIN | 8TH | | | | | 31ST | 1910 | | | | 24TH | |
| | | 1936 | | | | | 1873 | MORE | | | | 1876 | |
| MAX TEMPS | | | | | | | | | | | | | TOTAL |
| 90 & UP | 0.0 | 0.0 | 0.0 | 0.* | 0.* | 6.3 | 15.0 | 9.3 | 2.2 | 0.0 | 0.0 | 0.0 | 33.1 |
| 32 & LESS | 6.5 | 4.3 | 2.8 | 0.* | 0.0 | 0.0 | 0.0 | 0.0 | 0.* | 0.* | 2.5 | 5.3 | 22.2 |
| MINIMUM | | | | | | | | | | | | | |
| 32 & LESS | 29.8 | 26.0 | 24.3 | 11.4 | 1.5 | 0.0 | 0.0 | 0.0 | 0.8 | 8.6 | 24.4 | 29.2 | 156.1 |
| 0 & LESS | 3.8 | 1.7 | 0.5 | 0.* | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.* | 2.8 | 9.0 |

Table 2.2: Precipitation data for the GSTR

NORMALS AND MEANS 1971–2000 & EXTREMES 1872–PRESENT
(UPDATED 6/30/2005)
DENVER INTERNATIONAL AIRPORT, CO, LAT. 39.52N, LON. 104.40W, ELEV. 5431 FT.

| NORMALS | JAN | FEB | MAR | APR | MAY | JUN | JUN | AUG | SEP | OCT | NOV | DEC | AVG |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| PRECIP | | | | | | | | | | | | | TOTAL |
| .01 - UP | 5.8 | 5.9 | 8.6 | 8.5 | 10.8 | 8.7 | 9.3 | 8.8 | 6.3 | 5.3 | 5.6 | 5.3 | 89.1 |
| AVG NO. DAYS | | | | | | | | | | | | | |
| SNO/HAIL ICE | | | | | | | | | | | | | TOTAL |
| PELL 1.0 - UP | 2.4 | 2.4 | 3.6 | 2.5 | 0.4 | 0.0 | 0.0 | 0.0 | 0.3 | 1.2 | 2.6 | 2.5 | 17.9 |
| RAIN MONTHLY | | | | | | | | | | | | | |
| NORMAL | 0.51 | 0.49 | 1.28 | 1.93 | 2.32 | 1.56 | 2.16 | 1.82 | 1.14 | 0.99 | 0.98 | 0.63 | |
| | | | | | | | | | | | | | ***ANNUAL NORMAL 15.81*** |
| MAXIMUM MONTH | 1883 | 1934 | 1983 | 1900 | 1876 | 1882 | 1965 | 1979 | 1961 | 1969 | 1946 | 1913 | |
| PRECIPITATION | 2.35 | 2.01 | 4.56 | 8.24 | 8.57 | 4.96 | 6.41 | 5.85 | 4.67 | 4.17 | 3.21 | 5.21 | |
| | | | | | | | | | | | | | ***ANNUAL RECORD 23.31 IN 1967*** |
| MINIMUM MONTHLY | .01 | .01 | 0.11 | 0.03 | 0.06 | T | .01 | .02 | T | T | T | 0.00 | |
| YEAR | 1952 | 1970 | 1908 | 1963 | 1974 | 1890 | 1901 | 1924 | 1944 | 1934 | 1949 | 1881 | 1934 |
| MAXIMUM | | | | | | | | | | | | | |
| 24 HRS | 1.02 | 1.01 | 2.79 | 3.25 | 6.53 | 3.16 | 3.06 | 3.43 | 2.44 | 1.71 | 1.29 | 2.00 | |
| | 1962 | 1953 | 1983 | 1967 | 1876 | 1970 | 1997 | 1951 | 1936 | 1947 | 1975 | 1982 | |
| MONTHLY | | | | | | | | | | | | | |
| SNOW NORMAL | 7.7 | 6.3 | 11.7 | 9.1 | 1.3 | T | 0.0 | 0.0 | 2.1 | 4.1 | 10.7 | 8.7 | |
| | | | | | | | | | | | | | ***ANNUAL NORMAL 61.6*** |
| MAXIMUM MONTHLY | 24.3 | 22.1 | 32.5 | 33.8 | 15.5 | 0.4 | 0.0 | 0.0 | 17.2 | 31.2 | 42.6 | 57.4 | |
| SNOW/ICE PELLETS | 1992 | 1912 | 1944 | 1933 | 1898 | 1919 | – | – | 1936 | 1969 | 1946 | 1913 | |
| | | | | | | | | | | | | | ***ANNUAL RECORD 118.7 IN 1908/09*** |
| MAXIMUM | 14.5 | 9.5 | 18.0 | 17.3 | 10.7 | 0.3 | T | T | 16.3 | 19.1 | 15.9 | 23.6 | |
| 24 HRS SNOW | 1992 | 1953 | 1983 | 1957 | 1950 | 1951 | 1992 | 1991 | 1936 | 1997 | 1983 | 1982 | |

Table 2.3: Wind data for the GSTR

NORMALS AND MEANS 1971–2000 & EXTREMES 1872–PRESENT
(UPDATED 6/30/2005)
DENVER INTERNATIONAL AIRPORT, CO, LAT. 39.52N, LON. 104.40W, ELEV. 5431 FT.

| NORMALS | JAN | FEB | MAR | APR | MAY | JUN | JUN | AUG | SEP | OCT | NOV | DEC | AVG |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| WIND (MPH) | | | | | | | | | | | | | AVG |
| AVG SPEED | 8.6 | 8.8 | 9.7 | 10.0 | 9.3 | 8.9 | 8.3 | 8.0 | 7.9 | 7.8 | 8.2 | 8.4 | 8.6 |
| DIRECTION | S | S | S | S | S | S | S | S | S | S | S | S | S |
| FAST 1-MIN | | | | | | | | | | | | | |
| DIRECTION | 32 | 30 | 30 | 33 | 36 | 21 | 29 | 33 | 29 | 01 | 36 | 32 | FAST |
| SPEED | 44 | 36 | 41 | 46 | 43 | 38 | 46 | 33 | 36 | 36 | 36 | 38 | 46 |
| YEAR | 1982 | 1989 | 1991 | 1993 | 1983 | 1987 | 1993 | 1989 | 1988 | 1992 | 1987 | 1981 | 1993 |
| PEAK GUST | | | | | | | | | | | | | |
| DIRECTION | W | NW | W | NW | SE | NW | W | NW | NW | NW | W | W | PEAK |
| SPEED | 55 | 52 | 59 | 62 | 60 | 60 | 55 | 75 | 85 | 48 | 49 | 51 | 85 |
| YEAR | 1993 | 1990 | 1989 | 1986 | 1991 | 1988 | 1993 | 1877 | 1995 | 1990 | 1990 | 1990 | 1995 |

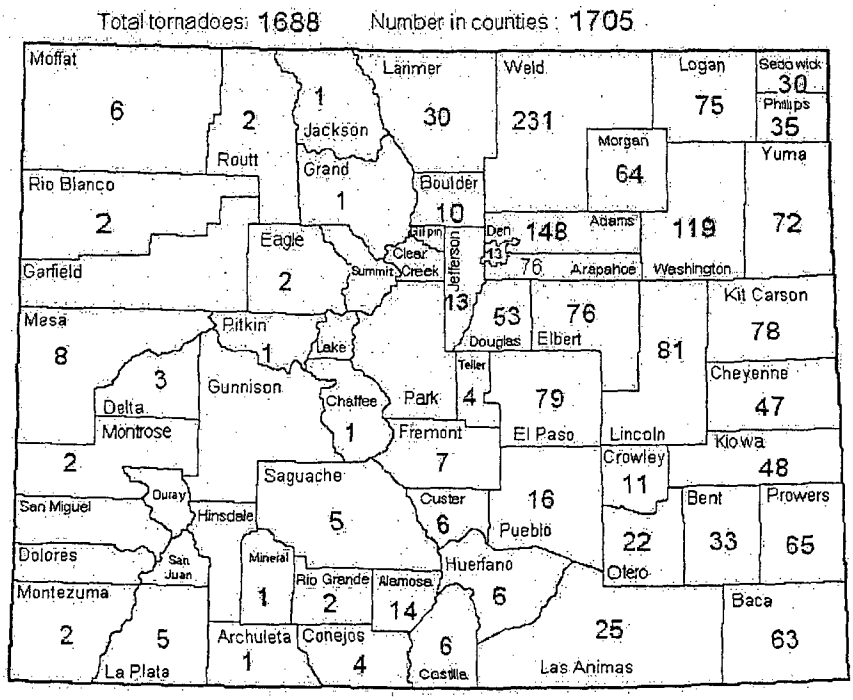


Figure 2.4: Tornadoes reported in Colorado counted from 1950–2006.

#### 2.3.1.5 Severe Weather Phenomena

Tornadoes in Jefferson County are quite infrequent due to the rugged terrain found within Colorado. Figure 2.4 shows all the tornadoes counted in Colorado per county from 1950–2006. Between the years 1887 and 1996 there has been only one tornado that measured as a 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 blow about light objects. This tornado occurred in 1981. There have been no fatalities from tornadoes in Jefferson County; Table 2.4 shows all deaths and the rating of the tornado on the Fujita scale since 1965. Wind data is shown in Table 2.3.

## 2.4 Hydrology

### 2.4.1 River Flooding

The GSTR is 5,583 feet above mean sea level. It is about 2,000 feet from any definable drainage channel and about 5 miles from and 350 feet above the nearest major river (South Platte) which flows through the Denver area. The natural drainage from the site is to the northeast. Storm sewers with inlets near the site drain to the east and northeast, and discharge into Lakewood Gulch which discharges into the South Platte River. An irrigation canal lateral crosses the Federal Center from north to southeast and passes about 750 feet west and ten feet higher than the reactor site. Possibility of overflow is remote, but if it does occur the

Table 2:4: JEFFERSON COUNTY TORNADOES

| DATE | TIME | DEATHS | FORCE |
|---|---|---|---|
| JUN 10, 1965 | 1453 | 0 | F0 059 |
| JUN 30, 1965 | 1405 | 0 | F0 059 |
| JUN 30, 1965 | 1515 | 0 | F0 059 |
| JLY 03, 1967 | 2040 | 0 | F0 059 |
| JUN 03, 1981 | 1420 | 0 | F2 059 |
| JUN 30, 1982 | 1450 | 0 | F1 059 |
| JLY 09, 1982 | 1210 | 0 | F1 059 |
| JLY 09, 1982 | 1210 | 0 | F1 059 |
| JUN 09, 1987 | 1539 | 0 | F1 059 |
| JUN 20, 1987 | 1415 | 0 | F1 059 |
| JLY 11, 1990 | 1440 | 0 | F0 059 |
| JUN 06, 1995 | 1738 | 0 | F0 059 |
| JUL 28, 1996 | 0545 | 0 | F0 059 |

entire flow would be confined to streets and gutters and would not reach the reactor bay.

### 2.4.2 Seismically-Induced Flooding

There are no lakes or dams near the GSTR and, therefore, flooding due to seismically-induced dam failure is not a risk to the GSTR. Tsunamis are of no concern as well, as the GSTR is located 1250 miles from the coastline. Liquefaction of any sandy areas is not a concern either because of shallow ground water.

## 2.5 Geology, Seismology, and Geotechnical Engineering

### 2.5.1 Earthquake History

Colorado is comprised of areas with a low to moderate potential for damaging earthquakes. There are approximately ninety (90) potentially active faults that have recorded movement in Colorado within the last 1.6 to 2.0 million years. There are several thousand other mapped faults that have not been sufficiently studied to know whether they are capable of generating earthquakes or not. However, historically speaking, Colorado is a state of little earthquake activity, as seismographs have only recorded data since 1909. The largest earthquake on record occurred in 1882 near Rocky Mountain National Park and estimates are that it had a magnitude of approximately 6.2 on the Richter scale.

In 1961 a series of wells, culminating in a 12,000 foot well, were drilled at the Rocky Mountain Arsenal, approximately 15 miles northeast of the GSTR, and a series of small, abnormal earthquakes erupted shortly after. It is postulated that the fluid flow through the wells lubricated some small fault plane and allowed detectable motion to be generated. These earthquakes have all been substantially smaller than the quake of

Table 2.5: Historic Earthquakes Within 50 miles of the Denver Federal Center

| Year | Month | Day | Epicenter | Magnitude |
|---|---|---|---|---|
| 1870 | December | 7 | 20 miles E. of Pueblo | Unknown |
| 1882 | November | 7 | Rocky Mountain National Park | 6.2 (est.) |
| 1962 | April | 24 | Rocky Mountain Arsenal (RMA) | 1.5 |
| 1962 | December | 4 | RMA | 4.5 (est.) |
| 1965 | August | 9 | Bergen Park (3 earthquakes) | 3.5 (est.) |
| 1966 | November | 14 | Denver Metro | 2.0-2.5 (est.) |
| 1966 | April | 10 | RMA | 5.0 |
| 1967 | April | 9 | RMA | 5.3 |
| 1967 | November | - | RMA (5 earthquakes) | 2.1-4.1 |
| 1967 | November | 26 | NE Denver/RMA | 5.2 |
| 1968 | July | 15 | RMA | 2.5 (est.) |
| 1971 | March | 11 | Commerce City/RMA | 3.0 |
| 1971 | August | 7 | Denver (2 earthquakes) | 3.0-4.4 |
| 1978 | June | 10 | 10 km NE of Denver | 2.9 |
| 1981 | April | 2 | 20 km N of Denver | 4.1 |
| 1981 | November | 1 | Evergreen | 3.1 |
| 1982 | March | 11 | 12 miles N. of Denver | 2.8 |
| 1982 | September | 18 | 12 miles N. of Denver | 2.8 |
| 1984 | February | 25 | 13 miles N. of Denver | 2.5 |
| 1986 | September | 21 | 35 km. W. of Denver (Conifer) | 2.5 |
| 1989 | November | 7 | 10 km. NE of Denver | 2.5 |

1882 and activity still remains minimal. Table 2.5[‡] lists known historical earthquakes.

### 2.5.2 Site Geology

Colorado's current set of the Rocky Mountains rose early in the Cenozoic Era, starting about 65 million years ago during the Laramide Orogeny. During and after this uplift there were periods of erosion, equally important to shaping today's scenery along the "front range" as is recorded by various Quaternary alluvial depositional units that date increased erosion activity from about 10,000 years ago to 1.6 million years ago. There is one principal fault in the Lakewood/Golden area. This is the Golden Fault (and it's associated Graben feature). Both are located in Jefferson County and are considered to be part of the Southern Rocky Mountains complex.

Bedrock geology in the area is composed of the Denver Formation (Tertiary/Cretaceous), the Monzanite and Latite extrusives of North and South Table Mountain (Tertiary) and conglomerates of Green Mountain (Tertiary), the Arapahoe Formation and the Laramie Formation (Cretaceous). The most deeply buried (and oldest) units are the Fox Hill Sandstone and the Pierre Shale (Cretaceous), of which the Pierre Shale is the thickest and most wide-spread. The units floor the "Denver Basin" structure, which the Denver Federal Center sits at the margin of. The USGS TRIGA reactor sits on a mantle of unconsolidated deposits of Quaternary age and interspersed outcroppings of Cretaceous or Tertiary-Cretaceous sedimentary bedrock.

[‡]Numerous other smaller earthquakes can be found in USGS Earthquake Search

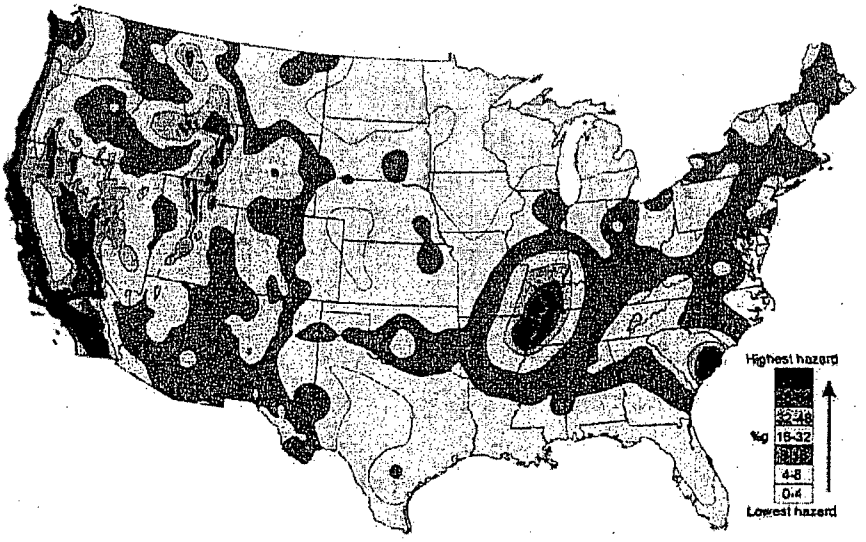


Figure 2.5: USGS 2002 Continental U.S. Earthquake Hazards Map

The Quaternary mantle is comprised of eolian deposits (quartz sand and silt), alluvium (gravel, sand, and silt of variable composition), colluvium, and few landslides (near Green Mountain only).

#### 2.5.2.1 The Golden Fault

This reverse fault, which dips west at angle of 45° or more, is the longest trending fault in the area. An exploration trench across the Golden fault was mapped by Van Horn, 1972. This trench exposed a graben that offset Verdos Alluviam (tentively dated as pre-Kansan or about 600,000 years in age) and the overlying Yarmouth soil (about 300,00 years in age). The rupture also displaced an ash bed correlated with Lava Creek B (680,000 years old). The Golden Fault was studied in detail by Dames and Moore and they concluded there was no evidence that the fault moved since deposition of the Verdos Alluvium. It is classified as a "Class B" fault with a slip rate of <0.2mm/yr, no historical earthquakes and most recent deformation as having occurred about 1.6 million years ago.

#### 2.5.2.2 Faults and Joints within 10 miles

There are numerous faults edging the mountain block; many are thrust faults along which up-lifted Precambrian rocks spread outward over the sedimentary rocks. There are several short northwest-trending faults, probably contemporaneous with Golden Fault, many of which cut the Fort Hays Limestone member of the Niobrara Formation. These faults dip about 50° SW. They evidently die out northwestward in the Benton Shale, as they are not recognized in the Hogback of the Dakota Group; they also die out southeastward in the shale of the smoky hill shale member.

A joint pattern typical of sedimentary terrain is well displayed on the large roadcut where Alameda Parkway

crosses the Dakota Sandstone hogback. There, the dominant joint set strikes N65E and dips vertically. There are no hazards associated with this joint system.

### 2.5.3 Seismicity Maximum Earthquake Potential

All recorded earthquakes within 30 miles in the past 100 years have been very minor earthquakes (none greater then magnitude 5.3, most 3 or less), mostly associated with areas west of the Front Range. Figure 2.5 shows the 2002 USGS Seismic Hazard Map for the Conterminous United States, where 60 g is determined to be "severe shaking." Colorado, specifically the Front Range, where the GSTR is located, shows a low seismic hazard of 8-16 g[§].

### 2.5.4 Surface Faulting

Subsurface soil conditions near the USGS TRIGA reactor include consolidated and unconsolidated alluvium and intermittent amounts of loess (wind-blown silt) for a depth of approximately 10 to 15 m. Modern fill dominated the first meter or so of sediment. There is no mapped surface rupture near the USGS TRIGA Reactor and there would be a very low potential for shaking due to the lack of known faulting around the site (see discussion above).

### 2.5.5 Other geologic hazards

The USGS TRIGA reactor sits on alluvial fan sediment, so there is no significant potential for liquefaction. Liquefaction would require large areas of unconsolidated sand or floodplain alluvium, or unconsolidated sediments which contain a large amount of water. Neither of these conditions exists on the Denver Federal Center. Landslide deposits are minor in nature and confined to the slopes of Green Mountain, to the west of the GSTR. Bentonitic swelling from clay soils is also a very minor problem on the Denver Federal Center. There are no significant debris flows, as the slope of the land surface is gentle (5% or less).

The most significant threat could be from sudden floods of the Bear Creek area or the South Platte River. Historic flash floods during the 1960's (specifically 1965), led to the creation of dams within the Denver Basin (Bear Creek Lake and Chatfield Lake) to control these sudden fluxes of water. Since this time there have been no floods. Flood waters from these steams have never reached the Denver Federal Center site.

[§]When acceleration acts on a physical body, the body experiences the acceleration as a force. The force we are most experienced with is the force of gravity, which caused us to have weight. The units of acceleration of the map are measured in terms of g, the acceleration due to gravity. An acceleration of 11 feet per second per second is $11 \cdot 12 \cdot 2.54 = 335$ cm/sec/sec. The acceleration due to gravity is 980 cm/sec/sec, so an acceleration of 11 feet/sec/sec is about $335/980 = 0.34$ g. Expressed as a percent, 0.34 g is 34 % g.

# References

[1] USGS website, http://earthquakes.usgs.gov/regional/qfaults

[2] Earthquake Information Bulletin, Nov-Dec. 1970

[3] Surficial and Bedrock Geologic Map of the Golden Quadrangle, Jefferson County, Colorado, Map I-161-A, Richard Van Horn, 1972

[4] 1981, Report #2684

[5] Geology of The Sedimentary Rocks of The Morrison Quadrangle, Colorado, Map I-428, J. Hiram Smith, 1964

[6] Geologic Map of the greater Denver Area, Front Range Urban Corridor, Map I-856-H, Donald Trimble and Michael Machette, 2003

[7] H.F. Matthai, USGS eye-witness account in Roadside Geology of Colorado, Halka Chronic, 1980

# Chapter 3

# DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

## 3.1 Design Criteria

The GSTR was built in 1968-69. The TRIGA® Mark I reactor, developed by the General Atomic Division of General Dynamics Corporation, is an inherently safe reactor designed for advanced research in the various fields of nuclear research and education. The TRIGA® Mark I reactor operates routinely at steady-state thermal power levels up to 1.0 MW and has the capability of being pulsed repetitively to a peak power of approximately 1600 MW.

The principal design criterion was to ensure the facility could withstand loss of pool water and any other credible accident with no hazard to the public, without reliance on engineered safety systems. The design criteria creating this negligible safety risk are the result of the fuel composition and cladding, not of specific features provided in the equipment and building that surrounds the reactor. The potential accidents described in Chapter 13 conservatively demonstrate that instrumented shutdown actions and building confinement are not necessary to ensure that radiological doses will not exceed allowable limits. All building modifications and equipment additions were in conformance with the building codes in existence at the time.

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

The basic parameter that allows the GSTR to operate safely during either steady-state or transient conditions is the prompt negative temperature coefficient of reactivity ($-\alpha$T). The fuel-moderator elements used in the 1 megawatt TRIGA® Mark I reactor, which were developed by General Atomic, consist of a homogeneous mixture of uranium-zirconium hydride in which the H-to-Zr atom ratio ranges from approximately 1:1 to

1.7:1, and the uranium is enriched to slightly less than 20% $^{235}U$. [redacted] The elements are clad with either stainless steel or aluminum. The fuel-moderator material (U-ZrH) was created such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature would automatically cause the power excursion to terminate before any core damage resulted. 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. 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, where the temperature remains relatively constant, the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy when the neutrons enter the water. The heating of the moderator mixed with the fuel thus causes the spectrum to harden more in the fuel than the water. As a result, there is a temperature-dependent disadvantage factor for the unit cell in the core that decreases the ratio of absorptions in the fuel to total-cell absorptions as the fuel element temperature is increased. This change in disadvantage factor brings about a shift in the core neutron balance, gives a loss of reactivity, and is termed the cell effect.

The reactor core is operated near the bottom of a large open pool. Approximately 20 feet of water above the core provides vertical shielding. The resulting design dose rate is approximately 10 mrem/hr at the water surface from gamma radiation directly from the core, neglecting $^{16}N$. [redacted]

The reactor core consists of a cylindrical lattice of fuel moderator elements, control rods, and an americium-beryllium neutron source. The water that surrounds these components occupies about one-third of the core volume. The core excess reactivity will be limited to $6.80. Graphite end sections of the fuel-moderator elements serve as the top and bottom reflectors. Neutron reflection in the radial direction is provided by approximately 1 foot of graphite enclosed in an aluminum housing.

The core components are contained between top and bottom aluminum grid plates which are supported by the reflector housing. The grid plate has 126 holes for fuel elements and control rods arranged in six concentric rings around a center hole.

The control rods pierce both grid plates and are guided by the grid plates. A retraining safety plate attached to the reflector housing beneath the bottom grid plate prevents the possibility of a control rod dropping out of the core. Three poison rods with fueled followers are connected to the electrically driven rack-and-pinion control rod drives located directly above the core on the bridge. The fourth poison rod (transient rod), located in an aluminum guide tube, has an air-filled follower. This transient poison rod is connected to a pneumatic-electromechanical drive, also mounted on the bridge. Physical access and observation of the core are possible at all times through the vertical water shield.

Used in conjunction with the TRIGA® Mark I reactor are facilities for high-energy neutron and gamma radiation studies, radiation effects testing, and sample activation. These facilities include a central thimble,

an isotope production facility surrounding the core (rotary specimen rack), one or more pneumatic transfer systems, facilities for "in-core" irradiations, and a large reactor pool for in-pool irradiations.

The rotary specimen rack is a doughnut-shaped watertight "lazy susan" which rests in a well in the top of the graphite reflector. The lazy susan rotates around the core and is used for isotope production. Forty test-tube-like containers in this facility may be loaded with samples through a watertight tube from the bridge above. Samples may be inserted and removed while the reactor is operating at full power.

A high-speed pneumatic transfer system permits research with extremely short-lived radioisotopes. The in-pile terminii of this system may be located in in-core fuel element positions.

A 1.33 inch ID water-filled thimble for specimen irradiation is located in the central grid position. A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 inches in diameter into the region of highest flux (the central thimble must be removed from the core, and six fuel elements comprising the B-ring must be relocated in the outer ring of the core). Two triangular-shaped holes have been provided in the upper grid plate so that specimens up to 2.4 inches in diameter may be inserted. If these facilities are not being used, three fuel elements (one D- and two E-ring) may be placed in each cutout. Both the top and bottom grid plates contain foil insertion holes for making flux measurements. Additional experimental tubes can be easily installed in the core to provide additional facilities for high-level irradiations or in-core experiments. The irradiation facilities will be discussed further in Chapter 10.

Instrumentation is provided to monitor, indicate, and record the neutron flux and power level and their rates of changes. Three modes of operation are possible: Mode 1, steady-state operation, with manual or servo control to 1000 kW; Mode 2, square-wave operation (transient reactivity insertions to reach a desired steady-state power essentially instantaneously) at power levels between 300 and 1000 kW; and Mode 3, transient (pulse) operation in the GW range. Routine steady-state power operation is performed with the transient, shim, and regulating rods partially withdrawn. As shown in Chapters 4 and 13, the most rapid possible reactivity insertion rates are adequately compensated for by the negative temperature reactivity coefficient of -1¢/°C ($7\times10^{-4}$)($\Delta$k/k)/°C). The transient-rod system is specially designed for rapid reactivity insertion. Accidental actuation of the transient rod system may cause a reactivity accident in the sense that it was not planned. However, rapid reactivity additions constitute the normal pulse mode and the maximum reactivity change and the rate of addition are limited by the design of the system (i.e., upper limit switch and bracket).

Three separate detectors-two uncompensated ion chambers and one fission chamber-are utilized to provide reactor power measurements. The fission chamber provides the linear signal for the full power range on the NM1000 channel, for the period channel and for the log display. Both uncompensated ion chambers are connected to safety channels, and one uncompensated ion chamber is used by the pulsing channel. More detailed descriptions of the power detectors are located in Chapter 7.

Bulk water temperature, fuel element temperature, and water cooling system inlet and outlet temperatures are displayed on the console. In addition to the reactor control instrumentation, an interlock system is provided to prevent reactor operation unless prescribed safety conditions have been met.

The reactor core is cooled by natural convection of the demineralized water in the reactor pool. 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. The water system provided with the TRIGA® Mark I reactor takes water from the pool and pumps it through filtering, demineralizing, and cooling units and then back into the pool. The system consists of a 350-gpm pump that moves water through a 1000 kW shell-and-tube heat exchanger, a 10-gpm pump that moves water through a filter and demineralizer, and associated piping and valving. Instruments are provided for measuring water conductivity and pressure drop across the filter.

A portion of the 350 gpm pump discharge is diverted through a high-velocity jet above the core. The resulting circulation pattern reduces the dose rate at the pool surface that is partially created from the $^{16}N$ formed in the coolant water as it passes through the core.

## 3.2 Meteorological Damage

The GSTR reactor core is protected from damage by high winds or tornadoes by virtue of the thick reinforced concrete structure surrounding the reactor tank. The superstructure of the GSTR has been designed for area wind, rain, snow, and ice loads. The GSTR has endured approximately 40 years of local weather conditions with no meteorological damage. Hurricanes, tsunamis, and seiches do not occur in the Denver area.

Only a small number of tornadoes, one every few years, have been reported in the Denver, Colorado area. 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 GSTR structure.

## 3.3 Water Damage

The site is 5,583 ft. above mean sea level. It is about 2,000 ft. from any definable drainage channel and about 5 miles laterally and 350 feet above the major river (the South Platte) which flows through the Denver area. Natural drainage from the site is to the northeast. Storm sewers with inlets near the site drain to the east and northeast and discharge into Lakewood Gulch which discharges into the South Platte River. An irrigation canal laterally crosses the Federal Center from northwest to southeast and passes about 750 feet west and 10 feet higher than the reactor site. Possibility of overflow is remote, but if it does occur the entire flow will be confined to streets and gutters and will not reach the reactor bay. The sewage system from Building 15 drains into the Denver sewage system.

The alluvium underlying the general area has low permeability and will locally yield small amounts of water to wells, possibly up to 20 gallons per minute. No pumped wells are within the Federal Center and any in the general vicinity outside of the Federal Center are used only for lawn watering. The ground water level at the site is about 20 feet below the reactor bay floor and slopes gently to the east.

## 3.4 Geology

The Federal Center is situated on alluvium which extends from the foothills of the Rocky Mountains on the west to several miles east of Denver. Test borings near the reactor site show that the underlying near-surface material is composed of gravel, sand, and shale.

No geological faults are indicated by surficial evidence within the area.

## 3.5 Seismic Damage

The site is in a seismically "quiet" area. Occasional tremors centered near Derby, Colorado, about 15 miles northeast of the site have been recorded since 1962; these are measurable, but are not otherwise discernable at the Federal Center.

## 3.6 Systems and Components

### 3.6.1 Control Rod Drives

The drive assemblies for the control rods are fastened to a mounting plate located on the center channel. The standard safety rods have electrically-driven rack-and-pinion drives, the regulating rod has an electric stepping-motor-actuated linear drive, and the transient rod has a pneumatic-electromechanical drive.

The two shim rods, specifically the shim and safety rods, have electrically-driven rack-and-pinion drives consisting of a two-phase reversible motor, a magnet rod-coupler, a rack-and-pinion gear system, an electromagnet and armature, a dashpot assembly, control rod extension shaft, and a potentiometer used to provide rod position indication. These drives are standard TRIGA® drive mechanisms manufactured by GA. Limit switches are provided to indicate the up and down positions of the magnet and the down position of the rod. The nominal drive speed for these two rods is 19 inches/min. During a scram, the control rod, rod extension, and magnet armature are detached from the electromagnet and, thus, drop by gravity. The dashpot assembly slows the rate of insertion near the bottom to limit deceleration forces.

The regulating rod is similar to the shim rods except a stepping motor and reduction gear are used in place of the reversible motor. The nominal drive speed for the regulating rod is 27 inches/min. The stepping motor speed is adjustable with a maximum rod speed of 42 inches/min. The ability to change the rod drive speed is administratively controlled and access to the reactor area is limited to authorized personnel only.

The transient rod drive mechanism is a single-acting pneumatic cylinder manufactured by GA. Compressed air drives a piston (attached to the transient rod though a connecting rod assembly) against the top of a cylinder. The cylinder holding the piston can move up and down by virtue of a motor-driven worm gear engaged with a ball nut assembly. A potentiometer attached to the worm drive provides rod position indication. Limit switches provide indication when the cylinder is fully-up or fully-down.

### 3.6.2 Ventilation System

Although there are no required engineered safety features for this reactor due to its low operating power and good fission product retention in the fuel, a controlled ventilation system acts to reduce the consequences of fission products released from the fuel or other experimental facilities. Automatic switching of the ventilation system during high airborne radioactivity causes high efficiency filtering of all air leaving the reactor building during emergency conditions. The ventilation system is specifically described in Chapter 9.

### 3.6.3 Confinement

Based on radioactivity release calculations given in Chapter 13, the reactor building is not required to provide a containment function. No special seals for doors or lines which penetrate the walls are provided. [redacted]

# References

[1] Eppley, R.A., 1965, Earthquake history of the United States: U.S. Dept. of Commerce, No. 41-1

[2] The U-ZrHx Alloy: Its Properties and Use in TRIGA® Fuel, GA-E-117-883, 1980.

[3] Simnad, M.T., Fabian, C.F. and West, G.B., Fuel Elements for Pulsed TRIGA® Research Reactors; Nuclear Technology 28:31-56; 1976.

# Chapter 4

# TRIGA® REACTOR

## 4.1 Summary Description

The GSTR reactor is a standard design, nominal 1-MW, natural-convection-cooled TRIGA® pool reactor with a graphite reflector. [redacted] The approximately 20 feet of water above the core provides adequate shielding at the top of the 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 either manual or automatic control. The reactor can also be operated in square-wave and pulse mode.

The GSTR reactor, although originally taken critical with TRIGA® Standard 8.5 wt% stainless stainless steel clad fuel, is currently fueled with a mixture of 8.5 wt% SS clad, 12 w% SS clad, and 8 wt% aluminum clad TRIGA® fuel rods. Detailed data on TRIGA® fuel rods is given in Table 4.1.

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 up to 1150 °C for SS clad elements and 560 °C for Al clad elements. The inherent safety of this TRIGA® 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 used in TRIGA® systems. 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 GSTR.

## 4.2 Reactor Core

The GSTR 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

Table 4.1: LEU Fuel Parameters

| Fuel Element Type | 8.5 wt% | 12 wt% | Al Clad |
|---|---|---|---|
| Fuel-moderator material | U-$ZrH_{1.7}$ | U-$ZrH_{1.7}$ | U-$ZrH_{1.0}$ |
| Uranium content | 8.5 wt% | 12 wt% | 8.0 wt% |
| $^{235}U$ enrichment | ~20% | ~20% | ~20% |
| [redacted] | | | |
| Burnable poison | - | - | Samarium wafers |
| Shape | cylindrical | cylindrical | cylindrical |
| Length of fuel meat | 15 inches | 15 inches | 14 inches |
| [redacted] | | | |
| Cladding material | Type 304 SS | Type 304 SS | Type 1100 Al |
| Cladding thickness | 0.020 inches | 0.020 inches | 0.03 inches |

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-clad U-$ZrH_{1.7}$ fuel-moderator elements and aluminum-clad U-$ZrH_{1.0}$ fuel-moderator elements. The fuel-moderator elements have ~3.5-inch-long graphite end sections that form the top and bottom reflector. Water occupies about one-third of the core volume. Neutron reflection in the radial direction is provided by 10.2 inches (radial thickness) of graphite in an aluminum container. The height of the graphite in the reflector is about 22 inches. 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 six concentric rings around a central thimble (used for high flux irradiations). The power level of the TRIGA® reactor is accurately controlled with four control rods: a regulating rod, two shim rods, and a transient rod.

Three 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.

The reactor core is cooled by natural convection of the demineralized water in the reactor pool. A diffuser nozzle on the reactor tank inlet provides water discharge at a high velocity above the core. This water circulation pattern reduces the dose rate at the pool surface resulting from the N–16 formed in the coolant water as it passes through the core.

### 4.2.1 Reactor Fuel

The active part of each fuel-moderator element, shown in Figure 4.1, is [redacted] TRIGA® fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing from about 8 wt% to 12 wt% of uranium enriched to ~20% in $^{235}U$. The hydrogen-to-zirconium atom ratio is approximately 1.0 to 1.7. 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 complete.

Each element is clad in 0.020 inches of stainless steel or 0.030 inches of aluminum, and all closures are made by heliarc welding. Two 3.5-inch sections of graphite are inserted in this can, one above and one below the fuel, to serve as top and bottom reflectors for the core. Stainless steel or aluminum end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element approximately 28.4 inches.

The lower end fitting fits into the countersunk holes of the bottom grid plate that supports the fuel-moderator elements. The upper end fitting is grooved and specially shaped to fit and lock into the fuel handling tool. The top end fitting also incorporates a triangular spacer block that positions the top of the elements in the top grid plate and provides passages for cooling water through the upper grid plate. The total weight of a fully loaded fuel element is about 7.5 pounds.

Figure 4.1: Typical TRIGA® Fuel Element Assembly

An instrumented fuel-moderator element has three thermocouples embedded in the fuel. The sensing tips of the fuel element thermocouples are radially located halfway between the outer radius and the vertical centerline, and vertically located at the center of the fuel section and 1 inch above and below the vertical center. The thermocouple leadout wires pass through a seal contained in a stainless steel tube welded to the upper end fixture. This tube projects about 3 inches above the upper end of the element and is extended by lengths of tubing connected by unions to provide a watertight conduit carrying the leadout wires above the water surface in the reactor pool. In other respects, the instrumented fuel-moderator element is identical to the standard element.

The specific characteristics that make TRIGA®-type fuels uniquely suited for use in extremely safe research-type reactors are covered in detail in the following portions of this section. A summary of the characteristics is given below:

- 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.

#### 4.2.1.1 TRIGA® Fuel Development

The development and use of U-ZrH fuels for the TRIGA® reactor began at GA in 1957 and continues today. Over 6,000 fuel elements of 7 distinct types have been fabricated for the 60 TRIGA® research reactors in various countries around the world. The earliest of these has now almost 50 years of operation. U-ZrH fuel has exhibited unique safety features including a large negative prompt temperature coefficient of reactivity, high fission product retentivity, chemical stability when quenched from high temperatures in water, and dimensional stability over large swings of temperature. The first TRIGA® reactor to be exported was for the U.S. exhibit at the Second Geneva Conference on the Peaceful Uses of Atomic Energy in 1958.

The standard TRIGA fuel contains 8.5 wt% uranium (20% enriched) as a fine metallic dispersion in a zirconium hydride matrix. The H:Zr ratio is nominally 1.6; measured values indicate a ratio of 1.65. (in the face-centered cubic delta phase). The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For U-$ZrH_{1.7}$, the equilibrium hydrogen pressure is 1 atm at about 1400 °F (760 °C). The single-phase, high-hydride composition eliminates the problems of density changes associated

Table 4.2: Calculated Beginning of Life Prompt Temperature Coefficient ($\alpha$T) and Core Lifetime

| TRIGA® Fuel Type | Uranium (wt%) | Samarium (grams) | Enrichment (%) | ($\alpha$T) $\Delta$k/k·°C Average (23–700 °C) | Lifetime (MWd) |
|---|---|---|---|---|---|
| Standard | 8.5 | 0.00 | 20 | $\sim$-9.5×10$^{-5}$ | $\sim$100 |
| 12 Wt% | 12 | 0.00 | 20 | - | - |
| Al Clad | 8.0 | 3.5 | 20 | - | $\sim$-9.5×10$^{-5}$ |

with phase changes and with thermal diffusion of the hydrogen. Over 25,000 pulses have been performed with the TRIGA® fuel elements at GA, with fuel temperatures reaching peaks of about 2102 °F (1150 °C).

TRIGA® fuel was developed around the concept of inherent safety. A core composition was sought which had a large negative prompt temperature coefficient of reactivity such that if all the available excess reactivity were suddenly inserted into the core, the resulting fuel temperature increase would automatically cause the power excursion to terminate before any core damage resulted. Experiments performed in the late 1950s demonstrated that zirconium hydride possessed the basic mechanism needed to produce this desired characteristic. Additional advantages were that ZrH has a good heat capacity, allowing construction of a reactor with a relatively small core size and high flux values due to the high-hydrogen content of the fuel rods, and could be used effectively to fabricate rugged fuel rods.

In early 1976, GA undertook the development of fuels containing up to 45 wt% uranium (3.7 g U/cc) in order to allow the use of low-enriched uranium (LEU) (under 20% enrichment) to replace the highly-enriched fuels while maintaining long core life. The 45 wt% fuel contains a relatively modest $\sim$20 volume percent of uranium. These fuels were fabricated successfully, with the required hydrogen content and erbium loading. The structural features of the hydrated LEU fuel were similar to those of the well-proven 8.5- and 12-wt% fuels, as shown by metallographic, electron microprobe analysis, and x-ray diffraction examination. Detailed evaluations of the new LEU fuel have shown that it performs essentially identically to the older standard TRIGA® fuel in all critical cores. Additional evaluations included analytical assessments of the prompt temperature coefficient of reactivity and the core lifetime (Table 4.2). Nuclear design and analytical studies have shown that the prompt temperature coefficient for the 20-wt% uranium fuel is essentially the same as that for standard fuel over the temperature range of interest [68 °F to 1,292 °F (20 °C to 700 °C)] and greater than that for the FLIP fuel which it replaces. The prompt temperature coefficient for the more highly-loaded LEU fuel shows a small temperature dependence, whereas the coefficient is relatively constant for standard fuel. The value of the prompt temperature coefficient of reactivity is slightly lower for the 30 wt% uranium fuel compared to the highly-enriched fuel it replaces; however, it is still large and significantly higher than the prompt temperature coefficients for any other type of reactor fuel.

Inclusion of erbium burnable poison in the TRIGA® LEU fuel has enabled core lifetimes of up to 3,000 MWd to be predicted for the 30-wt% fuel. It is emphasized that this is the core life from the time of initial refueling to end of useful life.

#### 4.2.1.2 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 (NH). 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 relation:

$$\log P = K_1 + \frac{K_2 \times 10^3}{T} \tag{4.1}$$

where

$$K_1 = -3.8415 + 38.6433X - 34.2639X^2 + 9.2821X^3, \tag{4.2}$$

$$K_2 = -31.2982 + 23.5741X - 6.0280X^2, \tag{4.3}$$

and

$P$ = pressure (atm),

$T$ = temperature (K), and

$X$ = 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 $ZrH_{1.7}$, the equilibrium hydrogen dissociation pressure is 1 atm at about 1,400 °F (760 °C). The absence of a second phase in the higher hydrides eliminates the problem of large volume changes associated with a phase transformation at approximately 995 °F (535 °C) 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 570 °F (300 °C).

#### 4.2.1.3 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.

#### 4.2.1.4 Hydrogen Retention

The rates of hydrogen loss through 250-$\mu$m-thick stainless steel cladding are low at cladding temperatures characteristic of TRIGA® fuel elements. A 1% loss of hydrogen per year occurs at about 930 °F (500 °C) clad temperature.

#### 4.2.1.5 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 1.7, the following relationships for the uranium density, $\rho u(A)$, and weight fraction, $^{W}U$, in the U-$ZrH_{1.7}$ alloy apply:

$$\rho_u(A) = \frac{^{W}U}{0.177 - 0.125\,^{W}U} \tag{4.4}$$

$$^{W}U = \frac{0.177\rho_u(A)}{1 + 0.125\rho_u(A)}. \tag{4.5}$$

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

$$\rho_u(A) = 19.07 V_f^{U(A)}, \tag{4.6}$$

where: $V_f^{U(A)}$ = volume fraction of uranium in the U-$ZrH_{1.7}$ alloy.

#### 4.2.1.6 Thermal Conductivity

Thermal conductivity measurements have been made over a range of temperatures. A problem in carrying out these measurements by conventional methods is the disrupting effect of hydrogen migration under the thermal gradients imposed on the specimens during the experiments. This has been minimized at GA by using a short-pulse heating technique to determine the thermal diffusivity and hence to permit calculation of the thermal conductivity. From measurements by GA 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 1.7 is 0.042 ± 0.002 cal/sec·cm·°C and is insensitive both to the weight fraction of uranium and to the temperature.

#### 4.2.1.7 Volumetric Specific Heat

The volumetric specific heat of zirconium hydride TRIGA® fuel is a function of temperature and composition. The volumetric specific heat (per element) of 8.5-wt% U-$ZrH_{1.6}$ is calculated to be

$$C_P = 2.04 + 4.17 \times 10^{-3} \cdot T \ (\mathrm{W \cdot sec/cm^3})\ ^{\circ}\mathrm{C} \ (\text{from } 0^{\circ}\mathrm{C}), \tag{4.7}$$

and for 20-wt% U-$ZrH_{1.7}$ is calculated to be

$$C_P = 2.17 + 4.36 \times 10^{-3} \cdot T \ (\mathrm{W \cdot sec/cm^3})\ ^{\circ}\mathrm{C} \ (\text{from } 0^{\circ}\mathrm{C}). \tag{4.8}$$

#### 4.2.1.8 Chemical Reactivity

Zirconium hydride has a relatively low reactivity in water, steam, and air at temperatures up to about 1,112 °F (600 °C). Zirconium hydride has been heated in air for extended periods of time at temperatures up to 1,112 °F (600 °C) with negligible loss of hydrogen. 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 570 °F (299 °C) and 1,230 psi during a 400-hour 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 hours 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 rods [redacted] were heated to 1,470 °F (800 °C) and end-quenched to test for thermal shock and corrosion resistance. No deleterious effects were observed. Also, a 6-mm diameter fuel rod was heated electrically to about 1,470 °F (800 °C) and a rapid stream of water was sprayed on it; no significant reaction was observed. Small and large samples were heated to 1,650 °F (900 °C) and quenched in water; the only effect observed was a slight surface discoloration. Finely-divided U-ZrH powder was heated to 570 °F (300 °C) and quenched to 175 °F (80 °C) in water; no reaction was observed. Later, these tests were extended to temperatures as high as 2,200 °F (1,200 °C), in which tapered fuel rods were dropped into tapered aluminum cans in water. Although the samples cracked and lost hydrogen, no safety problems arose in these tests.

Low-enriched TRIGA® fuels have been subjected to water-quench safety tests at GA. Quench tests were performed on 20%-enriched TRIGA® fuel samples (45 wt% uranium, 53 wt% zirconium, 1 wt% erbium, 1 wt% hydrogen) to simulate cladding rupture and water ingress into the TRIGA® reactor fuel rods during operation.

These results indicate satisfactory behavior of TRIGA® fuel for temperatures to at least 2,200 °F (1,200 °C). Under conditions where the clad 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 1,925 °F (1,050 °C). This is still about 125 to 210 °F (50 to 100 °C) 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 1,200 °F (650 °C) in long-term steady-state operations, and up to 2,100 °F (1,150 °C) in very short-time pulse tests.

#### 4.2.1.9 Irradiation Effects

Most of the irradiation experience to date has been with the uranium-zirconium hydride fuels used in the Space Nuclear Auxiliary Power (SNAP) Program (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 the results of these experiments. The uranium is present as a fine dispersal (about 1 $\mu$m-dia) in the U-ZrH fuels, and hence the recoil damage is

limited to small regions within the short (~10-$\mu$m) range of the fission recoils.

The U-ZrH 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 U-ZrH fuels at high burnup is governed by three basic mechanisms:

- The accommodation of solid fission products resulting from fission of $^{235}$U. This growth is approximately 3% $\Delta$V/V per metal atom % burnup. This mechanism is relatively temperature insensitive;
- The agglomeration of fission gases at elevated temperatures [above 1,300 °F (750 °C)]. This takes place by diffusion of the xenon and krypton to form gas bubbles; and
- 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. This was deduced from the rapid decrease in fuel-to-cladding $\Delta$T experienced during the early part of the irradiation. The saturation was reached in approximately 1,500 hours. 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 $^{235}$U.

#### 4.2.1.10 Prompt Fuel Temperature Coefficient of Reactivity

The basic parameter which provides the TRIGA® reactor system with a large safety factor in steady-state operation and under transient conditions is the rather constant prompt fuel temperature coefficient of reactivity ($\alpha$T). This temperature coefficient, which is a function of the fuel composition and core geometry, 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 temperature coefficient of reactivity for TRIGA® fuels is based on the neutron spectrum hardening characteristic that occurs in a zirconium hydride fuel. The spectrum hardening is caused by heating of the fuel-moderator elements. 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 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 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 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.

In the TRIGA® fuel, the temperature-hardened spectrum is used to decrease the fuel's reactivity through its interactions with a low-energy resonance material.

For reasons just discussed, more than 50% of ($\alpha$T) for a standard TRIGA® core comes from the temperature-dependent disadvantage factor, or cell effect, and ~20% each from Doppler broadening of the U-238 resonances and temperature-dependent leakage from the core. These effects produce a ($\alpha$T) of $\sim -1.0 \times 10^{-4} \Delta k/k \cdot °C$, which is essentially constant with temperature.

#### 4.2.1.11 Fission Product Retention

A number of experiments have been performed to determine the extent to which fission products are retained by U-ZrH (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. Results prove that only a small fraction of the fission products are released, even in completely unclad U-ZrH fuel. The release fraction varies from $1.5 \times 10^{-5}$ for an irradiation temperature of 660 °F (350 °C) to $\sim 10^{-2}$ at 1,475 °F (800 °C).

The experiments show that there are two mechanisms involved in the release of fission products from U-ZrH fuel, each of which is dominant over a different temperature range.

The first mechanism is that of fission fragment recoil into the gap between the fuels and clad. This effect predominates in fuel at temperatures up to ~750 °F (400 °C); the recoil release rate is dependent on the fuel surface-to-volume ratio but is independent of fuel temperature. Above ~750 °F (400 °C), the controlling mechanism for fission product release from U-ZrH 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 U-ZrH experiments, and measurements by others of fission product release from SNAP Program fuel, have been compared and found to be in good agreement.

The fractional release, N, of fission product gases into the gap between fuel and clad from a full-size standard U-ZrH fuel element is given by:

$$\phi = 1.5 \times 10^{-5} + 3.6 \times 10^{3} e^{-\frac{1.34 \times 10^{4}}{T+273}}, \tag{4.9}$$

where T = fuel temperature (0 - 1600 °C). The first term of this function is a constant for low-temperature release; the second term is the high-temperature portion.

The function given above applies to a fuel element which has been irradiated for a time sufficiently long so that all fission product activity is at equilibrium. Actual measured values of fractional releases fall well below that calculated by Equation 4.9. Therefore, for safety considerations, Equation 4.9 conservative values for the high-temperature release from U-ZrH fuel.

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 with the lower U loadings, the release was determined to be predominantly recoil-controlled at temperatures 750 °F (400 °C) and controlled by a migration or diffusion-like process above ~750 °F (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®) fuel as well as the 8.5-wt% U-ZrH fuel for which it was originally derived.

This correlation predicts higher fission product releases than measurements would indicate up to 2,010F (1,100C). At normal TRIGA® operating temperatures [<1,380 °F (750 °C)], there is a safety factor of approximately four between predicted and experimentally determined values.

### 4.2.2 Control Rods

Three motor-driven control rods (1 regulating, and 2 shim) and one electro-pneumatic safety transient rod control the reactor power. The transient rod, located in the C-ring, contains a solid rod of boron-carbide-impregnated graphite as a neutron poison. The transient rod assembly is about 37 inches long and is clad in a 1-1/4 inch O.D. aluminum tube. The borated graphite poison section is 15 inches long. The transient rod has an air-filled follower about 21 inches long. The transient rod is guided laterally in the core by a thin-walled aluminum guide tube that passes through the upper and lower grid plates and is screwed into, and supported by, the aluminum safety plate beneath the lower grid.

The fuel-follower shim and regulating rods, located in the D-ring and C-rings, pass through, and are guided by, 1-1/2 inch-diameter holes in the top and bottom grid plates. The rod exterior cladding is a sealed stainless steel tube approximately 43 inches long by 1-3/8 inches in diameter. A standard control rod is illustrated, both withdrawn and inserted, in Figure 4.2. The upper section of the rod is graphite and the next 15 inches is the neutron absorber (graphite impregnated with boron carbide). The follower section consists of 15 inches of U-$ZrH_{1.7}$ fuel, and the bottom section has 6-1/2 inches of graphite. An aluminum safety plate attached to the shroud beneath the lower grid plate prevents the possibility of a control rod, accidentally disconnected from its drive, from dropping out of the core.

Vertical travel of the control rods is approximately 15 inches. The nominal rod worth of the transient rod, regulating rod, shim I and shim II rods, are about \$2.36 (1.65% $\Delta$k/k), \$3.88 (2.72% $\Delta$k/k), \$2.56 (1.79% $\Delta$k/k), and \$2.47 (1.73% $\Delta$k/k), respectively. That makes the total rod worth about \$11.27 (7.89% $\Delta$k/k). The total excess reactivity in the core does not exceed \$6.80 (4.76% $\Delta$k/k). The transient control rod drive is mounted on a steel frame that bolts to the reactor bridge. From zero to a maximum of 15 inches of rod may be withdrawn from the core; however, administrative control must be exercised to restrict the travel so as not to exceed the maximum permissible step insertion of reactivity (2.1% $\Delta$k/k or \$3.00) since the maximum allowable worth of the transient rod is 2.9% $\Delta$k/k (approximately \$4.14). The $\beta$ value used for reactivity conversions was 0.0070.

A system of limit switches similar to that used with the standard control rod drives is used to indicate the position of the air cylinder and the transient rod. Two of these switches, the DRIVE UP and DRIVE DOWN switches, are actuated by a small bar attached to the bottom of the air cylinder. A third limit switch, the ROD DOWN switch, is actuated when the piston reaches its lower limit of travel. A 10-turn potentiometer driven by the motor shaft controls the position indicator on the console.

Figure 4.2: Typical Fuel-Follower-Type Control Rod Shown Withdrawn and Inserted

### 4.2.3 Neutron Moderator and Reflector

The reflector is a ring-shaped block of graphite that surrounds the core radially. The graphite is 10.2 inches thick radially, with an inside diameter of 21-5/8 inches (below the Lazy Susan) and a height of approximately 22 inches. The graphite is protected from water penetration by a leak-tight welded aluminum cover.

A "well" in the top of the graphite reflector is provided for the rotary specimen rack. This well is also aluminum-lined, the lining being an integral part of the aluminum reflector can. The rotary specimen rack is a self-contained unit and does not penetrate the sealed reflector at any point.

### 4.2.4 Neutron Startup Source

A [redacted] Curie americium beryllium ($^{241}$AmBe) neutron source is used in the GSTR core. The neutron source holder is made of aluminum, is cylindrical in shape, and has a cavity to hold the source. The source holder is approximately [redacted]. Two locations, located 180° in the upper grid plate between the F and G Rings, are provided for positioning and supporting the neutron source holder. A shoulder at the upper end of the holder supports the assembly on the upper grid plate, with the holder itself, which contains the source, extending down into the core region. The neutron source is contained in the cavity in the lower portion of the holder assembly at the vertical centerline of the core. The upper and lower portions of the holder are screwed together. A soft aluminum ring provides sealing against water leakage into the cavity. [redacted].

### 4.2.5 Core Support Structure

The reflector assembly rests on an aluminum platform at the bottom of the tank, and provides the support for the two grid plates and the safety plate. Four lugs are provided for lifting the assembly.

The top grid plate is an aluminum plate 5/8 inches thick (3/8 inches thick in the central region) that provides accurate lateral positioning for the core components. The plate is supported by a ring welded to the top inside surface of the reflector container and is anodized to resist wear and corrosion.

One hundred twenty six (126) holes, 1.505 inches in diameter, are drilled into the top grid plate in six concentric rings around a central hole to locate the fuel-moderator, the control rods and guide tubes, and the pneumatic system terminus (See Figure 4.3). A 1.505-inch-diameter center hole accommodates the central thimble. Small holes at various positions in the top grid plate permit insertion of foils into the core to obtain flux data.

A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 inches in diameter into the region of highest flux; this requires prior relocation of the six fuel elements from the B ring to the outer portion of the core and removal of the central thimble.

Two triangular-shaped sections are cut out of the upper grid plate. Each encompasses two E and one D ring positions. When fuel elements are placed in these locations, their lateral support is provided by a stainless steel spacer fitted over the top end fitting of the three elements involved. When the fuel elements and support are removed, there is room for inserting specimens up to 2.4 inches in diameter.

Two 5/8-inch diameter holes between the F and G rings of the grid plate locate and provide support for the source holder. The source can be stored in either one of these locations. The differential area between the triangular-shaped spacer blocks at the top of the fuel element and the round holes in the top grid plate permit passage of cooling water through the plate. The lower grid plate is an aluminum plate 3/4 inch thick which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. It is supported by six pads, welded to a support ring, which in turn, is welded to the lower portion

of the reflector assembly. Holes are countersunk in the lower grid plate to receive the lower end fixture of the fuel-moderator elements and the pneumatic system terminii. The holes on the lower grid plate are aligned with the fuel element holes from the top grid plate.

Figure 4.3: Top Grid Plate and Typical Core Configuration

A central hole 1.505 inches in diameter in the lower grid serves as a clearance hole for the central thimble. Eight additional 1.505-inch-diameter holes are aligned with upper grid plate holes to provide passage of fuel-follower control rods. The holes in the bottom grid plate not occupied by a control rod are plugged with removable bottom grid plate adapters that rest on the safety plate and make the position available for a fuel element. These bottom grid plate adapters are solid aluminum cylinders 1.5 inches in diameter by 17 inches long. At the lower end is a fitting that is accommodated by a hole in the safety plate. The upper end of the cylinder is flush with the upper surface of the bottom grid plate when the adapter is in place. This

end of the adapter has a countersunk hole similar to that in the bottom grid plate for accepting the fuel element lower end fitting. With the adapter in place, a position formerly occupied by a control rod with a fuel follower will now accept a standard fuel element. The adapter can be removed with a special handling tool.

The safety plate is provided to preclude the possibility of control rods falling out of the core. It is a 1/2-inch-thick plate of aluminum welded to the extension of the inner reflector liner and placed about 16 inches below the bottom grid plate.

## 4.3 Reactor Tank or Pool

The reactor core is located at the bottom of an aluminum liner inside an aluminum tank, which is in the center of the concrete shield structure described in Section 4.4. The tank has an outside diameter of 7 feet 7 inches, a depth of approximately 25 feet, and a minimum thickness of 1/4 inch.

The inner sections for the facilities have continuous welded joints, and the integrity of the joints was verified by X-ray testing, pressure testing, dye penetrant checking, and soap-bubble leak testing.

The reactor tank is filled with demineralized water to provide approximately 20 feet of shielding water above the top of the reflector. The tank holds about 8,000 gallons of demineralized water.

The top of the reactor tank is closed by hinged aluminum grating covers. A sheet of Lucite plastic is inserted into the bottom of selected grating sections to prevent the entry of foreign matter into the tank while still permitting visual observation of the reactor. (The plastic sheets are easily removed for cleaning.) A gap around the perimeter of the plastic and some holes in the plastic permit adequate venting of the small quantities of hydrogen gas which may be released during reactor operations. The reactor tank covers are designed to support only the weight of people who are walking and working over the reactor. Therefore, heavy equipment or isotope casks will not be placed on these tank covers.

Support for the central thimble, rotary specimen drive-and-indicator assembly facilities, control-rod drive mechanisms, and the tank covers is provided by the center channel assembly (reactor bridge). This assembly is placed over the top of the reactor tank and consists of two structural steel channels with steel cover plates. The center channel assembly is designed to support a cask weighing 3-1/2 tons.

## 4.4 Biological Shield

The reactor is placed inside an aluminum liner that sits inside an aluminum tank. [redacted] The dimensions of the liner and tank are given in Table 4.3.†

†The liner sits inside the tank and extends approximately 5 inches above the tank.

Table 4.3: Dimensions of the liner and tank for the reactor

[redacted]

The core is shielded vertically by approximately 20 feet of demineralized water and radially by ~10 inches of graphite reflector, ~2 feet of water, and ~4 feet of concrete.

## 4.5 Nuclear Design

The reactor design bases are established by the maximum operational capability for the fuel elements and configuration described in this report. The TRIGA® reactor system has three major areas which are used to define the reactor design bases:

- fuel temperature,
- prompt temperature coefficient, and
- reactor power.

The ultimate safety limit is based on fuel temperature, while the large prompt negative temperature coefficient of reactivity of the fuel contributes to the inherent safety of the TRIGA® reactor. A limit on steady state reactor power of 1000 kW (1 MW) thermal is set to ensure operation below the fuel temperature safety limit.

Criticality was attained in the GA TRIGA® Mark I reactor with 54 standard (8.5 wt% 20% enriched) TRIGA® fuel elements [redacted] $^{235}$U) and four water-filled control rod positions. The critical mass for the Advanced TRIGA® Prototype was 75 standard TRIGA® fuel elements [redacted]g $^{235}$U) with three fuel followers and one transient rod without a follower. (Fuel followers on control rods are not counted as fuel elements.)

For the GA TRIGA® Mark III reactor, the critical mass was 56 standard TRIGA® fuel elements [redacted] $^{235}$U) using an "A" ring element and four fuel followers. Changes in critical mass occurred with varying control rod configurations and follower materials. Tolerances in the manufacture of the fuel elements (hydrogen and uranium content) also contributed a variance of about ±3% in the critical mass of any given system.

The reflector materials around the core and the absence or presence of graphite reflector elements in empty fuel positions also affected the critical mass rather strongly. Replacing the graphite with a water reflector on the TRIGA® Mark I raised the critical mass about 25%; while placing a row of graphite reflector elements around the water-reflected core cut this difference approximately in half.

### 4.5.1 Critical Mass of the GSTR Using Standard 20% Enriched Fuel

Criticality of the Geological Survey TRIGA® was achieved on 26 February 1969 with 58 standard TRIGA® fuel elements, one instrumented fuel element, and three fuel follower control rods (approximately [redacted] $^{235}U$). Also in the core was one transient rod with an air (void) follower and 40 graphite reflector elements arranged in the F and G rings. The critical configuration resulted in a core excess of $0.0004.

#### 4.5.1.1 Normal Operating Conditions

The normal core is an assembly of TRIGA® aluminum and/or stainless steel clad fuel-moderator elements, nominally 8.0 to 12.0 wt% uranium, arranged in a close packed array except for (1) replacement of single individual elements with incore irradiation facilities or control rods; (2) two separated experiments positions in the D through E rings, each occupying a maximum of three fuel element positions. The reflector (excluding experiments and experimental facilities) shall be graphite or a combination of graphite and water. The reactor shall not be operated in any manner that could cause the temperature in an aluminum-clad TRIGA® fuel element to exceed 995 °F (535 °C) or a stainless-steel-clad TRIGA® fuel element to exceed 1830 °F (1000 °C). Aluminum clad fuel-moderator elements will only be allowed in the F and G rings or the core assembly. The grid pattern is described in Section 4.2.5.

#### 4.5.1.2 Limiting Core Configuration

The GSTR technical specifications limit the GSTR thermal power to 100 kW with less than 100 fuel elements in the core. With 100 or more fuel elements the GSTR thermal power is limited to 1000 kW steady state. It has been shown that with at least 100 fuel elements in the core, power will be less than 22 kW in any single fuel element. The analysis was conducted on this 100 element core by a Monte Carlo program. Monte Carlo N-Particle (MCNP) that was developed and tested at a Department of Energy Laboratory and has been benchmarked against TRIGA®facilities. This analysis indicates that the maximum power in one element does not exceed 22 kW at 1MW reactor power. General Atomics (GA-9064) has calculated that departure from nuclear boiling did not occur if thermal power developed in a fuel element does not exceed 43 kW. The GA analysis also indicated that the axial power peaking is primarily dependent upon fuel rod lengths and, therefore, should be the same for all standard length fuel elements. By limiting the maximum power in one element to 22 kw, the nucleate boiling ratio does not fall below two at the hottest point in the reactor fuel. Since the aluminum clad fuel has a lower temperature safety limit, the aluminum clad fuel is limited to the 2 outermost rings (F and G). The MCNP program indicates that the ratio of peak to average power is approximately two. The calculated power per element is shown in Table 4.4 and Table 4.5.

An important result can be seen from Tables 4.4 and 4.5:

- The maximum power per element that occurs for the 100-element core is less than 22 kW per element.

Table 4.4: Power in each element averaged per ring at 1 MW (100-element core)

| B Ring (kW/F.E[a]) | C Ring (kW/F.E) | D Ring (kW/F.E) | E Ring (kW/F.E) | F Ring (kW/F.E) | G Ring (kW/F.E) |
|---|---|---|---|---|---|
| 21.1 | 15.7 | 12.0 | 9.8 | 6.9 | 4.8 |

[a]F.E. – fuel element

Table 4.5: Calculated Maximum, Minimum and Average Power per Element (100-element core)

| Maximum Power per Element $P_{max}$ (kW/F.E.) | Minimum Power per Element $P_{min}$ (kW/F.E.) | Average Power per Element $P_{avg}$ (kW/F.E.) | $P_{max}$, $P_{avg}$ |
|---|---|---|---|
| 21.7 | 4.38 | 10.12 | 2.15 |

### 4.5.2 Reactor Core Physics Parameters

#### 4.5.2.1 Core Description

The GSTR has a circular grid pattern, thus a simple way to identify core loadings is by reference to specific rings (A through G) in the core and the loadings within each ring. Several core loadings have actually been used and three are given in Table 4.6. Core #1 was the initial just-critical core in February, 1969, core #2 was the first operational core in March, 1969, and the last core loading is the current core loading.

Table 4.8 has values typical values for $\beta_{eff}$ and neutron lifetimes as calculated by General Atomics.

### 4.5.3 Operating Limits

The upper limit on excess reactivity is \$6.80 (4.76 %$\Delta$k/k) and a minimum shutdown margin of \$0.55 (0.385 %$\Delta$k/k).

### 4.5.4 Reactor Fuel Temperature

The basic safety limit for the TRIGA® reactor system is the fuel temperature; this applies for both the steady-state and pulse mode of operation. The TRIGA® fuel which is considered low hydride, with a H/Zr ratio of less than 1.5, has a lower temperature limit than fuel with a higher H/Zr ratio. The GSTR utilizes fuel with H/Zr ratios of 1.0 and 1.7, (below and above 1.5) the difference between the different hydride ratios for the prompt negative temperature coefficient is illustrated in Figure 4.4.

Table 4.6: GSTR Core Loadings

| Core Loading Information | Core #2 | Critical Core Core #2 | Present typical Core[a] |
|---|---|---|---|
| Date | 2/26/1969 | March, 1969 | January, 2007 |
| Core Excess Reactivity | $0.0004 | $6.91 | $6.10 |
| $k_{eff}$ | 1 | 1.0484 | 1.0427 |
| A Ring | Water | Water | Water |
| B Ring | 5 Std<br>1 Std IFE[b] | 5 Std<br>1 Std IFE | 5 Std<br>1 Std IFE |
| C Ring | 10 Std<br>1 VFCR[c]<br>1 FFCR[d] | 10 Std<br>1 VFCR<br>1 FFCR | 5 12wt%<br>1 12wt% IFE<br>3 Std<br>1 Std IFE<br>1 VFCR<br>1 FFCR |
| D Ring | 16 Std<br>2 FFCR | 16 Std<br>2 FFCR | 1 12wt%<br>1 12wt% IFE<br>14 Std<br>2 FFCR |
| E Ring | 24 Std | 24 Std | 24 Std |
| F Ring | 3 Std<br>21 water<br>6 graphite | 15 Std<br>15 graphite | 30 Al (8 wt%) |
| G Ring | 34 graphite<br>2 termini | 7 Std<br>4 Water<br>23 graphite<br>2 termini | 35 Std<br>1 Std IFE |

[a]Core loading occasionally has minor changes
[b]IFE - Instrumented Fuel element
[c]VFCR - Void-Followed Control Rod
[d]FFCR - Fuel-Followed (Std) Control Rod

The low H/Zr ratio (aluminum clad fuel) is restricted to the outer rings (F and G only). Figure 4.5 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 995 °F (535 °C) in the lower hydrides. It has been noted in that the higher hydrides lack any significant thermal diffusion of hydrogen. These two facts preclude concomitant volume changes.

The important properties of delta phase U-ZrH are given in Table 4.7.

Among the chemical properties of U-ZrH and ZrH, the reaction rate of the hydride with water is of particular interest. Since the hydriding reaction is exothermic, water will react more readily with zirconium than with zirconium hydride systems. Zirconium is frequently used in contact with water in reactors, and the zirconium-water reaction is not a safety hazard. Experiments carried out at GA show that the zirconium hydride systems have a relatively low chemical reactivity with respect to water and air. These tests have involved the quenching with water of both powders and solid specimens of U-ZrH after heating to temperatures as

Table 4.7: Physical Properties of Delta Phase U–ZrH

| | |
|---|---|
| **Thermal conductivity**<br>**200 °F–1,200 °F (93 °C - 650 °C)** | 13 Btu/h· $ft^2$·°F |
| **Elastic modulus:**<br>**70 °F (20 °C)**<br>**1,200 °F (650 °C)** | $9.1 \times 10^6$ psi<br>$6.0 \times 10^6$ psi |
| **Ultimate tensile strength to 1,200 °F (to 650 °C)** | 24,000 psi |
| **Compressive strength 70 °F (20 °C)** | 60,000 psi |
| **Compressive yield 70 °F (20 °C)** | 35,000 psi |
| **Heat of formation [*HOF 570 °F (298 °C)]** | 37.72 kcal/g·mole |

high as 1,560 °F (850 °C), and of solid U-Zr alloy after heating to temperatures as high as 2,190 °F (1,200 °C). Tests have also been made to determine the extent to which fission products are removed from the surfaces of the fuel elements at room temperature. Results prove that, because of the high resistance to leaching, a large fraction of the fission products are retained in even completely unclad U-ZrH fuel.

Table 4.8: Reactor Core Physics Parameters

| Fuel element property | 8.5 wt% U, stainless steel clad | 12 wt% U, stainless steel clad | 8.0 wt% U, aluminum clad |
|---|---|---|---|
| [redacted] | | | |
| **Enrichment %** | <20 | <20 | <20 |
| **wt% uranium** | 8.5 | 12 | 8 |
| **H/Zr atom ratio** | 1.7:1 | 1.7:1 | 1:1 |
| **Cladding material** | stainless steel | stainless steel | aluminum |
| **Cladding thickness (in)** | 0.02 | 0.02 | 0.03 |
| **Graphite plugs?** | yes | yes | yes |
| **Burnable poison** | no | no | yes - 0.05" Sm |
| **Max. recommended (°C) operating temp** | 800 | 800 | 535 |
| $\beta_{eff}$ **w/graphite reflector** | 0.007 | 0.007 | 0.0073 |
| **Prompt neutron lifetime** | 43 $\mu$sec | 43 $\mu$sec | 60 $\mu$sec |

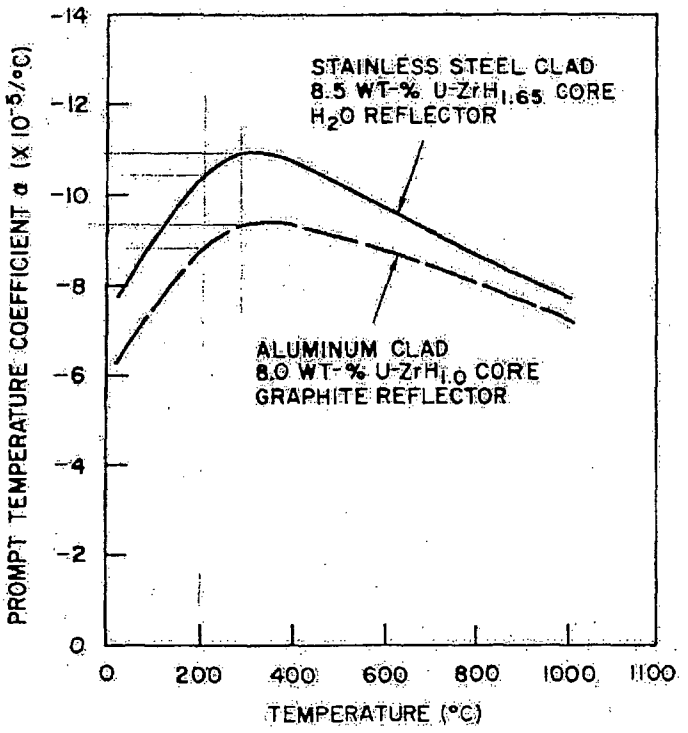


Figure 4.4: Prompt Temperature Coefficient of Reactivity vs. Temperature

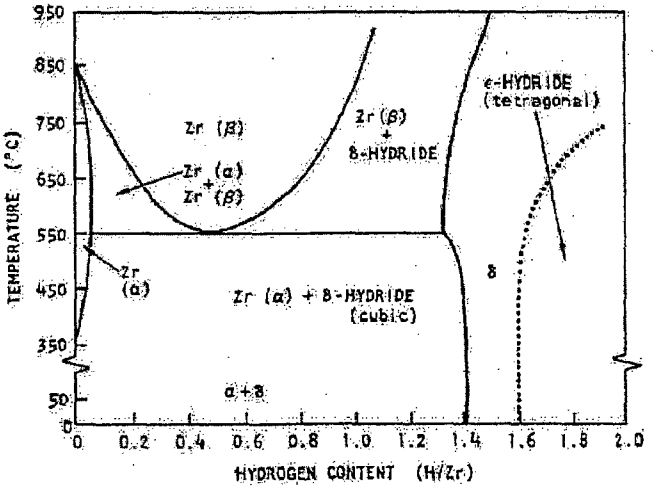


Figure 4.5: Phase Diagram of the Zirconium-Hydrogen System

At room temperature, the hydride is like a ceramic and shows little ductility. However, at the elevated temperatures of interest for pulsing, the material is found to be more ductile. The effect of very large thermal stresses on hydride fuel bodies has been observed in hot cell observations to cause relatively widely spaced cracks which tend to be either radial or normal to the central axis and do not interfere with radial heat flow. Since the segments tend to be orthogonal, their relative positions appear to be quite stable.

The limiting effect of fuel temperature is the hydrogen gas pressure causing cladding stress. Figure 4.6 relates equilibrium hydrogen pressure in ZrH with varying hydrogen content as a function of temperature for three different H/Zr ratios. The main concern regarding hydrogen pressure is to ensure that the cladding ultimate strength is not exceeded by the stress caused by the pressure. The mechanisms in obtaining temperatures and pressures of concern are different in the pulsing and steady-state mode of operation, and each mechanism will be discussed separately.

The GSTR fuel consists of U-ZrH with a H-Zr ratio of either 1.0 or 1.7 and with the uranium being 8.0 wt% to 12 wt% enriched in $^{235}$U to approximately 20%. The cladding is 0.020 inch thick stainless steel and has an inside diameter of 1.43 inches. The rest of the discussion on fuel temperatures will be concerned with fuel having H/Zr ratios greater than 1.5 [i.e., single-phase and not subject to the large volume changes associated with phase transformation at approximately 995 °F (535 °C) in the lower hydrides]. Further, it will specifically address fuel with an H/Zr ratio of 1.7 since this is the highest ratio fuel to be used in the GSTR and will produce the highest clad pressure and stress for a given temperature. Figure 4.7 shows the characteristic of 304 stainless steel with regard to yield and ultimate strengths as a function of temperature.

The stress applied to the cladding from the internal hydrogen gas pressure is given by:

$$S = \frac{\mathrm{Pr}}{t}, \tag{4.10}$$

where:

$S$ = stress in psi,

$P$ = internal pressure in psi,

$r$ = radius of the stainless steel cylinder, and

$t$ = wall thickness of the stainless steel clad.

Using the parameters given above:

$$S = \frac{P(1.43\mathrm{in.}/2)}{0.020\mathrm{in.}} = 35.75P. \tag{4.11}$$

For safety considerations, it is necessary to relate the strength of the cladding material at its operating temperature to the stress applied to the cladding due to the internal gas pressure associated with the fuel temperature. Figure 4.8 gives the ultimate cladding strength and the stress applied to the cladding as a result of hydrogen dissociation for fuel having H/Zr ratios of 1.65 and 1.7, both as a function of temperature. This curve shows that the cladding will not fail for fuel with $ZrH_{1.7}$ if both the clad and fuel temperatures

are equal and below about 1,700 °F (930 °C). This is conservative since the cladding temperature will be below the fuel temperature. This establishes the safety limit on fuel temperature for steady-state operations. The actual steady-state peak fuel temperature at 1 MW will be below 932 °F (500 °C). The remainder of this section deals with the safety limit for transient operation.

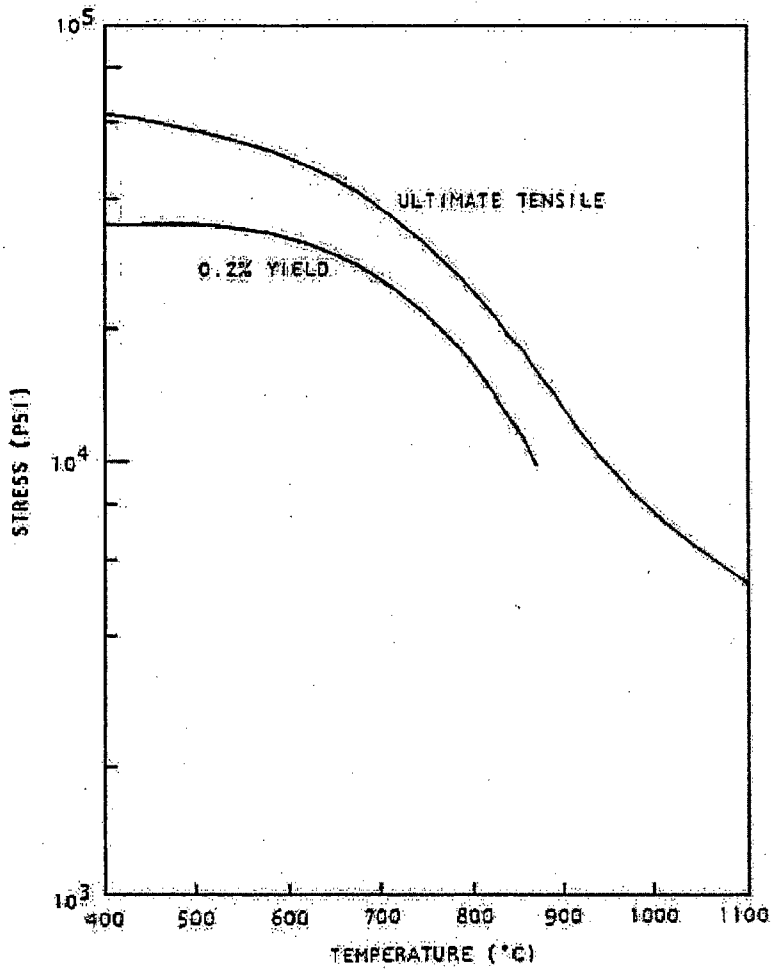


Figure 4.7: Strength of Type 304 Stainless Steel as a Function of Energy

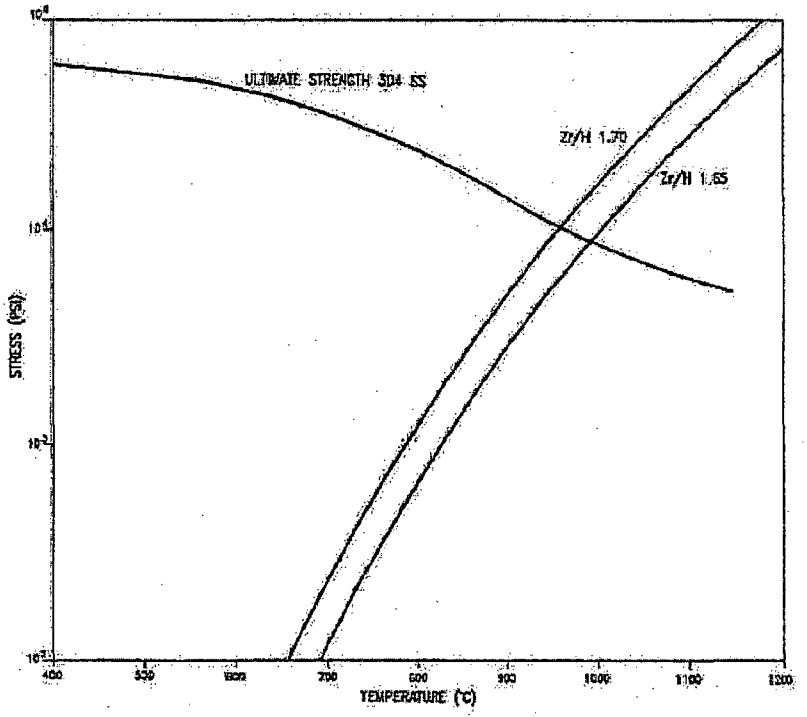


Figure 4.8: Strength and Applied Stress as a Function of Temperature, Equilibrium at Equilibrium Hydrogen Dissociation Pressure

During transient operation, it is necessary to account for the difference in fuel and cladding temperatures to establish a safety limit based on fuel temperature. Additionally, the diffusion of hydrogen reduces peak pressures from those predicted at equilibrium at the peak temperatures. The net result of these two points is that a higher safety limit exists for transient operation. An analysis of the two points is given in the following two subsections.

#### 4.5.4.1 Fuel and Clad Temperature

For the steady-state safety limit, it was assumed that the cladding and fuel temperatures were the same. The following discussion shows that the cladding temperature is well below the maximum fuel temperature after a pulse. This allows a higher safety limit on fuel temperature.

The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4.9. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may

occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to approach the fuel temperature.

Thermal transient calculations were made using the RAT computer code. RAT is a 2D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that, if film boiling occurs after a pulse, it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a reduced threshold for film boiling. Data obtained for transient heating of ribbons in 100 °F (38 °C) water, showed burnout fluxes of 0.9 to 2.0 MBtu/ft$^2$·h for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channel between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 MBtu/ft$^2$·h. It is shown, on the basis of the following analysis, that the second mode is the most likely; i.e., when film boiling occurs, it takes place under essentially steady-state conditions at local water temperatures near saturation.

A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figure 4.9 and 4.10, respectively. The thermal resistance at the fuel-clad interface was assumed to be zero. A boiling heat transfer model, as shown in Figure 4.11, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams for the subcooled boiling and the work of Sparrow and Cess for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler et al., Zuber, and Rohsenow and Choi to find the minimum temperature point at which film boiling could occur.

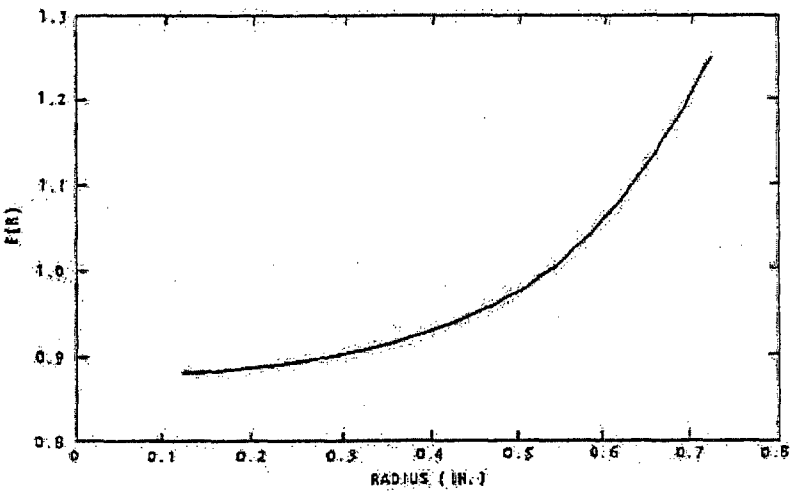


Figure 4.9: Radial Power Distribution in the U-ZrH Fuel Element

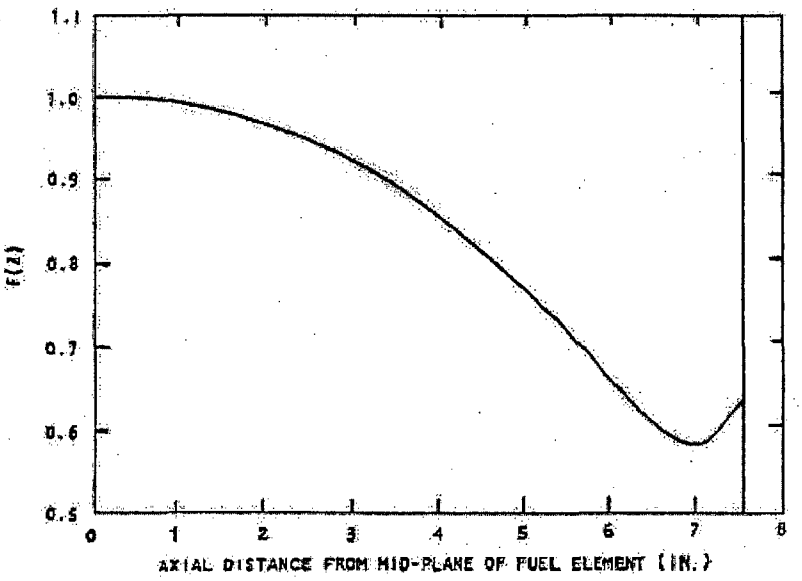


Figure 4.10: Axial Power Distribution in the U-ZrH Fuel Element

This calculation gave an upper limit of 1,400 °F (760 °C) clad temperature for a peak initial fuel temperature of 1,830 °F (1,000 °C), as shown in Figure 4.12. Fuel temperature distributions for this case are shown in Figure 4.13 and the heat flux into the water from the clad is shown in Figure 4.14. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion for water conditions similar to those for the TRIGA® system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 880 °F (470 °C) as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 750 °F to 930 °F (400 °C to 500 °C) for TRIGA® Mark F fuel elements which have been operated under film boiling conditions. Based on this analysis, the peak cladding temperature will be 880 °F (470 °C) for a transient fuel temperature of 1,830 °F (1,000 °C). Further analysis will show that this peak clad temperature is valid for a higher peak fuel temperature. The preceding analysis assessing the maximum clad temperatures associated with film boiling assumed no thermal resistance at the fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad. This gap exists at all temperatures below the maximum operating temperature. The gap thickness varies with fuel temperature and clad temperature: cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction because the basic data required for input are presently not fully known.

Instead, several transient thermal computations were made using the RAT code, varying the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1,830 °F (1,000 °C) peak fuel temperature case.

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculation. The calculations assumed a coolant flow velocity of 1 foot per second which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a complete boiling curve heat transfer model, but instead, included a convection cooled region (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath which has been reported by Spano to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate velocity of 1 foot per second which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors.

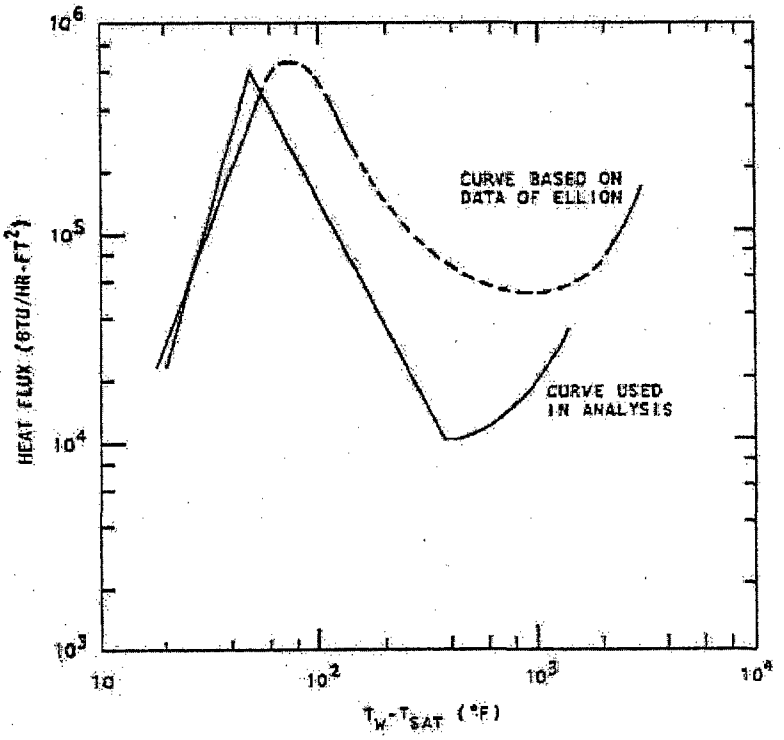


Figure 4.11: Subcooled Boiling Heat Transfer For Water

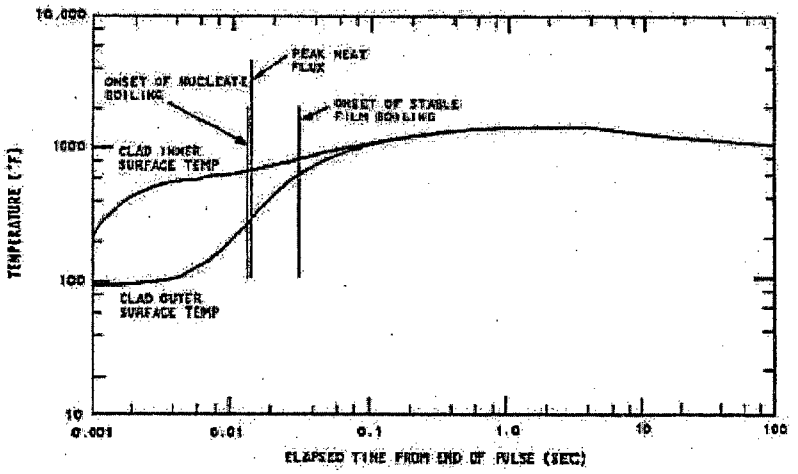


Figure 4.12: Clad Temperature at Midpoint of Well-Bonded Fuel Element

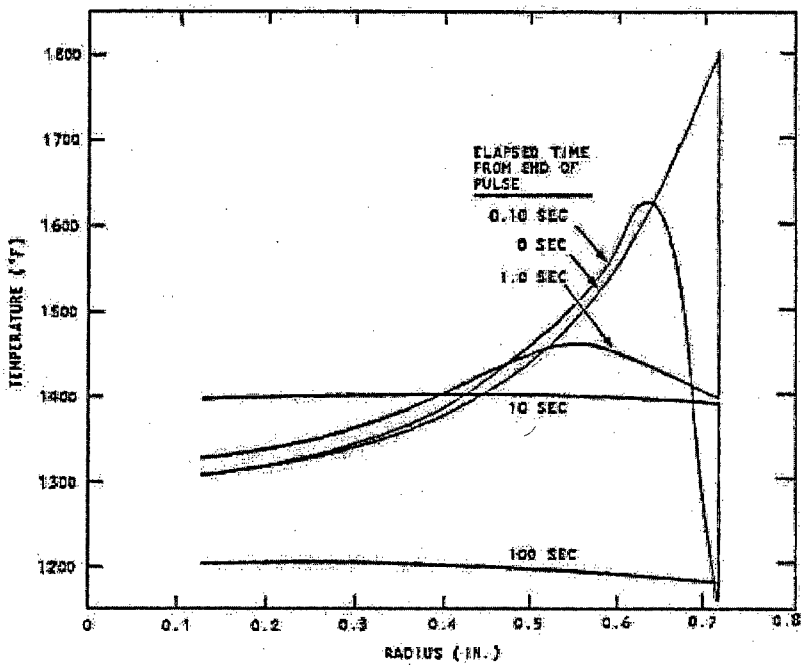


Figure 4.13: Fuel Body Temperatures at Midplane of Well-Bonded Fuel Element After a Pulse

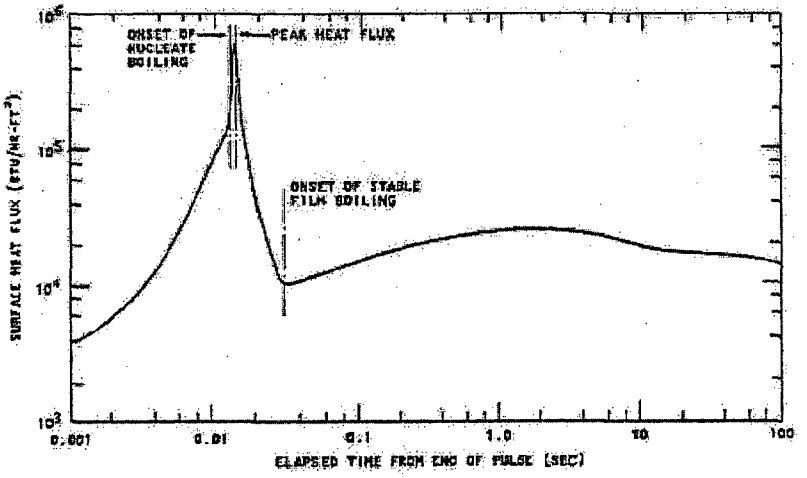


Figure 4.14: Surface Heat Flux at Midplane of Well-Bonded Fuel Element After a Pulse

The results were analyzed by inspection using the extended steady-state correlation of Bernath which has been reported by Spano to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu/h·ft$^2$· °F. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach are shown in Figures 4.15 through 4.17. If the minimum approach to DNB is corrected to TRIGA® Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu/h·ft$^2$· °F is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA® fuel.

The surface heat flux at the midplane of the element is shown in Figure 4.18 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4.15 through 4.17. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu/h·ft$^2$· °F according to the more conservative Bernath correlations at a local water temperature approaching saturation.

From this analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1,830 °F (1,000 °C) is estimated to be 880 °F (470 °C). This is conservative since it was obtained by assuming no thermal resistance between the fuel and the clad. As was shown above, a value of 500 Btu/h·ft$^2$· °F for the gap conduction is more realistic.

As can be seen from Figure 4.7, the ultimate strength of the cladding at a temperature of 880 °F (470

°C) is 59,000 psi. If the stress produced by the hydrogen overpressure on the clad is less than 59,000 psi, the cladding will not be ruptured. Referring to Figure 4.8, and considering U-$ZrH_{1.7}$ fuel with a peak temperature of 1,830 °F (1,000 °C), one finds the stress on the clad to be 24,000 psi. Analysis in the next section which considers diffusion will show that the actual hydrogen pressure produced in a pulse is less than the equilibrium pressure for the peak temperature. This allows a safe limit on fuel temperature to be 2,012 °F (1,100 °C). TRIGA® fuel with a hydrogen to zirconium ratio of at least 1.6 has been pulsed to temperatures approaching 2,100 °F (1,150 °C) without damage to the clad.

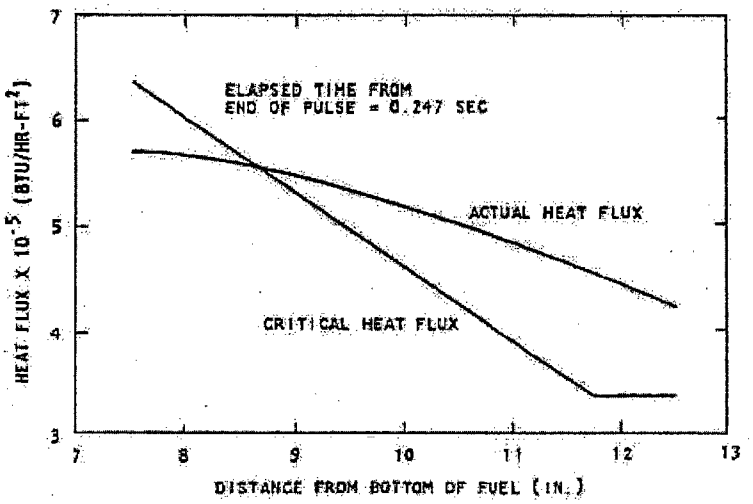


Figure 4.15: Surface Heat Flux for Standard Nongapped (hgap=500) Fuel Element After a Pulse

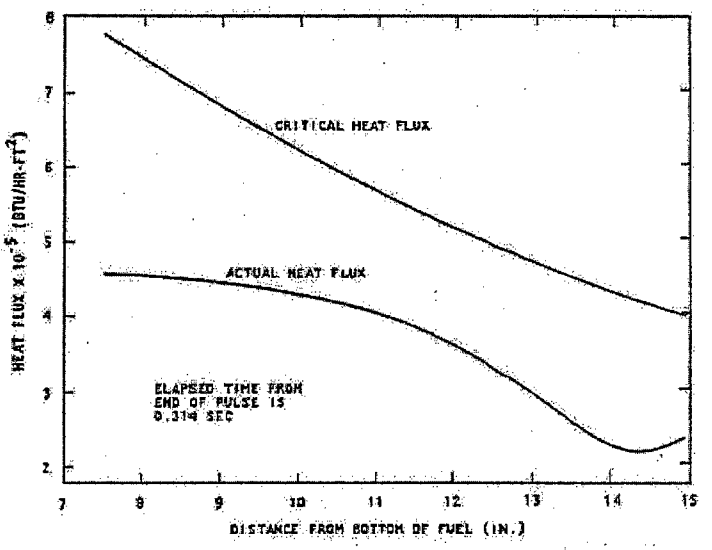


Figure 4.16: Surface Heat Flux for Standard Nongapped (hgap=375) Fuel Element After a Pulse

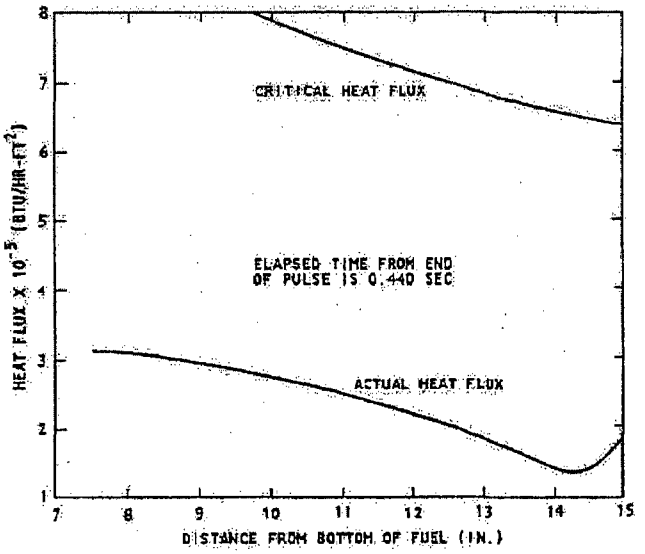


Figure 4.17: Surface Heat Flux for Standard Nongapped (hgap=250) Fuel Element After a Pulse

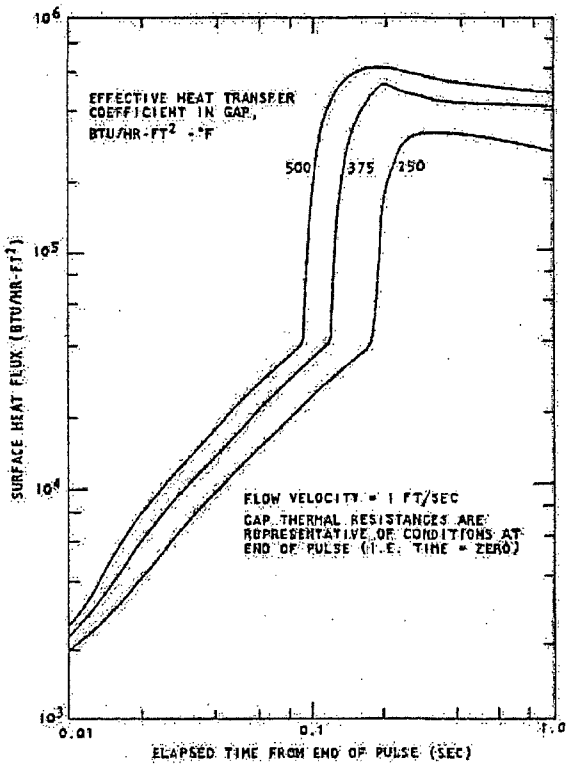


Figure 4.18: Surface Heat Flux at Midpoint vs. Time For Standard Nongapped Fuel Elements After a Pulse

#### 4.5.4.2 Finite Diffusion Rate

To assess the effect of the finite diffusion rate and the rehydriding at the cooler surfaces, the following analysis is presented.

As hydrogen is released from the hot fuel regions, it migrates to the cooler regions and the equilibrium pressure that is obtained is characteristic of some temperature lower than the maximum. To evaluate this reduced pressure, diffusion theory is used to calculate the rate at which hydrogen is evolved and reabsorbed at the fuel surface.

Ordinary diffusion theory provides an expression describing the time dependent loss of gas from a cylinder:

$$\frac{\bar{c} - c_f}{c_i - c_f} = \sum_{n=1}^{\infty} \frac{4}{Z_n^2} e^{-\frac{Z_n^2 Dt}{r_0^2}}, \tag{4.12}$$

where:

$\bar{c}$, $c_i$, $c_f$ = the average, the initial, and the final gas concentration in the cylinder, respectively;

$Z_n$ = the roots of the Bessel function of the first kind where $J_0(x) = 0$;

$D$ = the diffusion coefficient for the gas in the cylinder;

$r_0$ = the radius of the cylinder; and

$t$ = time.

Setting the term on the right-hand side of Equation 4.12 equal to $\kappa$, one can rewrite Equation 3 as:

$$\frac{\bar{c}}{c_i} = \frac{c_f}{c_i} + \left(1 - \frac{c_f}{c_i}\right)\kappa, \tag{4.13}$$

and the derivative in time is given by

$$\frac{\mathrm{d}\frac{\bar{c}}{c_i}}{\mathrm{d}t} = \left(1 - \frac{c_f}{c_i}\right)\frac{\mathrm{d}\kappa}{\mathrm{d}t}. \tag{4.14}$$

This represents the fractional release rate of hydrogen from the cylinder, f(t). The derivative of the series in the right-hand side of Equation 4.14 was approximated by

$$\frac{\mathrm{d}\kappa}{\mathrm{d}t} = -\left(7.339e^{-8.34\epsilon} + 29.88e^{-249\epsilon}\right)\frac{\mathrm{d}\epsilon}{\mathrm{d}t}, \tag{4.15}$$

where

$$\epsilon = \frac{Dt}{r_0^2}. \tag{4.16}$$

The diffusion coefficient for hydrogen in zirconium hydride in which the H/Zr ratio is between 1.56 and 1.86 is given by

$$D = 0.25e^{-17800/[R(T+273)]}, \tag{4.17}$$

where:

$R$ = the gas constant; and

$T$ = the zirconium hydride temperature in °C.

Equation 4.12 describes the escape of gas from a cylinder through diffusion until some final concentration is achieved. Actually, in the closed system considered here, not only does the hydrogen diffuse into the fuel-clad gap, but also diffuses back into the fuel in the regions of lower fuel temperature. The gas diffuses through the clad at a rate dependent on the clad temperature. Although this tends to reduce the hydrogen pressure, it is not considered in this analysis. When the diffusion rates are equal, an equilibrium condition will exist. To account for this, Equation 4.14 was modified by replacing the concentration ratios by the ratio of the hydrogen pressure in the gap to the equilibrium hydrogen pressure, $P_h/P_e$. Thus,

$$\frac{\mathrm{d}\frac{c}{c_i}}{\mathrm{d}t} = \left(1 - \frac{P_h(t)}{P_e}\right)\frac{\mathrm{d}\kappa}{\mathrm{d}t}, \tag{4.18}$$

where:

$P_h(t)$ = the hydrogen pressure, a function of time; and

$P_e$ = the equilibrium hydrogen pressure over the zirconium hydride which is a function of the fuel temperature and H/Zr ratios.

The rate of change of the internal hydrogen pressure, in psi, inside the fuel element cladding is

$$\frac{\mathrm{d}P_h}{\mathrm{d}t} = \frac{14.7f(t)N_h}{6.02 \times 10^{23}}\frac{22.4}{V_g}\frac{T+273}{273}, \tag{4.19}$$

where:

$N_h$ = the number of molecules of $H_2$ in the fuel;

$T$ = the gas temperature (°C);

$f(t)$ = the fractional loss rate from Equation 8; and

$V_g$ = the free volume inside the fuel clad (liters).

For a fuel volume of 400 $cm^3$, the moles of $H_2$ available from fuel with $ZrH_{1.65}$ and $ZrH_{1.7}$ is 19.9 and 20.6 moles, respectively. The free volume is assumed to consist of a cylindrical volume, at the top of the element,

1/8 inch high with a diameter of 1.43 inches for a total of 3.3 $cm^3$. The temperature of the hydrogen in the gap was assumed to be the temperature of the clad. The effect of changing these two assumptions was tested by calculations in which the gap volume was decreased by 90% and the temperature of the hydrogen in the gap was set equal to the maximum fuel temperature. Neither of these changes resulted in maximum pressures different from those based on the original assumptions although the initial rate of pressure increase was greater. For these conditions

$$P_h = A \times 10^3 (T + 273) \infty(t) dt, \tag{4.20}$$

where A = 7.29 for $ZrH_1.65$ and 7.53 for $ZrH_1.7$.

The fuel temperature used in Equation 4.17 to evaluate the diffusion coefficient is expressed as:

$$\begin{aligned} T(z) &= T_0; t < 0, \\ &= T_0 + (T_m + T_0) \cos\left[2.45(z - 0.5t)\right]; t \geq 0 \end{aligned} \tag{4.21}$$

where:

$T_m$ = the peak fuel temperature (°C),

$T_0$ = the clad temperature (°C),

$Z$ = the axial distance expressed as a fraction of the fuel length, and

$T$ = the time after step increase in power.

It was assumed that the fuel temperature was invariant with radius. The hydrogen pressure over the zirconium hydride surface when equilibrium prevails is strongly temperature dependent as shown in Figure 4.6, and for ZrH, can be expressed by:

$$P_e = 2.07 \times 10^9 e^{-1.974 \times 10^4/(T+273)}. \tag{4.22}$$

The coefficients have been derived from data developed by Johnson. The rate at which hydrogen is released or reabsorbed takes the form:

$$g(t, z) = \frac{[P_e(z) - P_h(t)]}{P_e(z)} f(t, z), \tag{4.23}$$

where:

$f(t, z)$ = the derivative given in Equation 8 with respect to time evaluated at the axial position z,

$P_h(t)$ = the hydrogen pressure in the gap at time t, and

$P_e(z)$ = the equilibrium hydrogen pressure at the ZrH temperature at position z.

The internal hydrogen pressure is then

$$P_h(t) + A \times 10^3 (T_0 + 273) \int_0^t \int_0^1 g(t,z) \mathrm{d}z. \tag{4.24}$$

This equation was approximated by:

$$P_h(t_i) = A \times 1063(T_0 + 273) \cdot \sum_{i=1}^{n} \sum_{j=1}^{m} \left[1 - \frac{P_h(t_{i-1})}{P_e(z_j)}\right] l \cdot f(t_i, z_j) \partial z \partial t, \tag{4.25}$$

where the inner summation is over the fuel element's length increments and the outer summation is over time.

For the cases where the maximum fuel temperature is 2,100 °F (1,150 °C) for $ZrH_{1.65}$ and 2,020 °F (1,100 °C) for $ZrH_{1.7}$, the equilibrium hydrogen pressure in ZrH is 2,000 psi, which leads to an internal stress of 72,000 psi. Using Equation 4.25, it is found that the internal pressure for both $ZrH_{1.65}$ and $ZrH_{1.7}$ increases to a peak at about 0.3 sec, at which time the pressure is about one-fifth of the equilibrium value or about 400 psi (a stress of 14,700 psi). After this time, the pressure slowly decreases as the hydrogen continues to be redistributed along the length of the element from the hot regions to the cooler regions.

Calculations have also been made for step increases in power to peak $ZrH_{1.65}$ fuel temperatures greater than 2,100 °F (1,150 °C). Over a 390 °F (200 °C) range, the time to the peak pressure and the fraction of the equilibrium pressure value achieved were approximately the same as for the 2,100 °F (1,150 °C) case. Similar results were found for fuel with $ZrH_{1.7}$. Thus, if the clad remains below about 930°F (500°C), the internal pressure that would produce the yield stress in the clad (35,000 psi) is about 1,000 psi and the corresponding equilibrium hydrogen pressure is 5,000 psi. This corresponds to a maximum fuel temperature of about 2,280 °F (1,250 °C) in $ZrH_{1.65}$ and 2,160 °F (1,180 °C) in $ZrH_{1.7}$. Similarly, an internal pressure of 1,600 psi would produce a stress equal to the ultimate clad strength (over 59,000 psi). This corresponds to an equilibrium hydrogen pressure of 5 × 1,600 or 8,000 psi and a fuel temperature of about 2,370 °F (1,300 °C) in $ZrH_{1.65}$ and 2,260 °F (1,240 °C) in $ZrH_{1.7}$.

Measurements of hydrogen pressure in TRIGA® fuel elements during steady-state operation have not been made. However, measurements have been made during transient operations and compared with the results of an analysis similar to that described here. These measurements indicated that, in a pulse in which the maximum temperature in the fuel was greater than 1,830 °F (1,000 °C), the maximum pressure ($ZrH_{1.65}$) was only about 6% of the equilibrium value evaluated at the peak temperature. Calculations of the pressure resulting from such a pulse using the methods described above gave calculated pressure values about three times greater than the measured values.

An instantaneous increase in fuel temperature will produce the most severe pressure conditions. When a peak fuel temperature is reached by increasing the power over a finite period of time, the resulting pressure will be no greater than that for the step change in power analyzed above. As the temperature rise time becomes long compared with the diffusion time of hydrogen, the pressure will become increasingly less than

for the case of a step change in power. The reason for this is that the pressure in the clad element results from the hot fuel dehydriding faster than the cooler fuel rehydrides (takes up the excess hydrogen to reach an equilibrium with the hydrogen overpressure in the can). The slower the rise to peak temperature, the lower the pressure because of the additional time available for rehydriding.

#### 4.5.4.3 Summary

The foregoing analysis gives a strong indication that the cladding will not be ruptured if fuel temperatures are never greater than in the range of 2,190 °F to 2,280 °F (1,200 °C to 1,250 °C), providing that the cladding temperature is less than about 930 °F (500 °C).

However, for fuel with a $ZrH_{1.7}$, a conservative safety limit of 2,012 °F (1,100 °C) has been chosen for this condition. As a result, at this safety limit temperature the pressure is about a factor of 4 lower than would be necessary for cladding failure. This factor of 4 is more than adequate to account for uncertainties in cladding strength and manufacturing tolerances. As a safety limit, the peak adiabatic fuel temperature to be allowed during transient conditions is considered to be 2,012 °F (1,100 °C) for U-ZrH fuel with ratios up to 1.70.

Under any condition in which the cladding temperature increases above 930 °F (500 °C), the temperature safety limit must be decreased as the cladding material loses strength at elevated temperatures. To establish this limit, it is assumed that the fuel and the cladding are at the same temperature. There are no conceivable circumstances that could give rise to a situation in which the cladding temperatures was higher than the fuel temperature.

In Figure 4.8, the stress imposed on the clad by the equilibrium hydrogen pressure as a function of the fuel temperature is plotted. Also shown is the ultimate strength of 304 stainless steel at the same temperatures. The use of these data for establishing the safety limit for conditions in which the cladding temperature is greater than 930 °F (500 °C) is justified as:

- the method used to measure ultimate strength requires the imposition of the stress over a longer time than would be imposed for accident conditions, and

- the stress is not applied biaxially in the ultimate strength measurements as it is in the fuel clad.

The point at which the two curves in Figure 4.8 intersect (for $ZrH_{1.7}$) is the safety limit, that is, 1,710 °F (930 °C) for conditions in which the cladding temperature is above 930 °F (500 °C). At that temperature, the equilibrium hydrogen pressure would impose a stress on the cladding equal to the ultimate strength of the clad.

The same argument about the redistribution of the hydrogen within the fuel presented earlier is valid for this case. In addition, at elevated temperatures the cladding becomes quite permeable to hydrogen. Thus, not only will hydrogen redistribute itself within the fuel to reduce the pressure, but some hydrogen will escape from the system entirely.

The use of the ultimate strength of the cladding material in the establishment of the safety limit under these conditions is justified because of the transient nature of accidents. Although the high cladding temperatures imply sharply reduced heat transfer rates to the surroundings (and consequently longer cooling times), only slight reductions in the fuel temperature are necessary to reduce the stress sharply. For a fuel with $ZrH_{1.7}$, a 70 °F (40 °C) decrease in temperature from 1,700 °F to 1,630 °F (930 °C to 890 °C) will reduce the stress by a factor of 2.

#### 4.5.4.4 Prompt Temperature Coefficient

The basic parameter which allows the TRIGA® reactor system to operate safely with large step insertions of reactivity is the strongly negative ($\alpha$T) 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.

GA, the designer of the reactor, has developed techniques to calculate ($\alpha$T) accurately and, therefore, predict the transient behavior of the reactor. 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 of ($\alpha$T) requires a 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 can be described as follows: 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. This is shown qualitatively in Figure 4.19. 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. A quantitative description of these processes requires a knowledge of the differential slow neutron energy transfer cross section in water and zirconium hydride, the energy dependence of the transport cross section of hydrogen as bound in water and zirconium hydride, the energy dependence of the capture and fission cross sections of all relevant materials, and a multigroup transport theory reactor description which allows for the coupling of groups by speeding up as well as by slowing down.

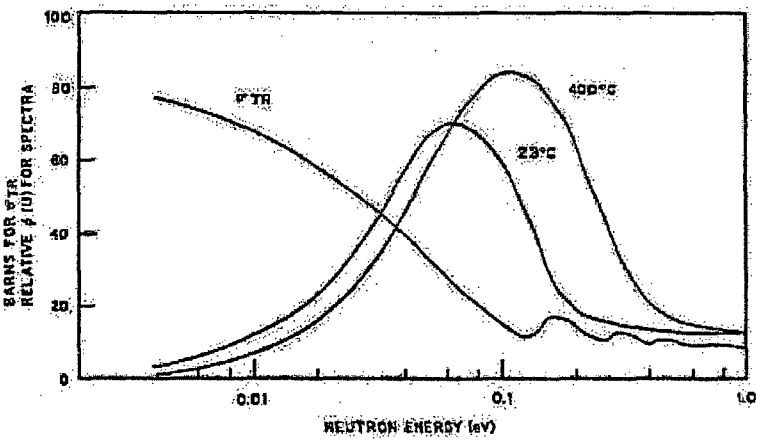


Figure 4.19: Transport Cross Section for Hydrogen in ArH and Average Neutron Spectra in Fuel Element

#### 4.5.4.4.1 Codes used for Calculations

Calculational work on ($\alpha$T) made use of a group of codes developed by GA: GGC-3; GAZE-2; and GAMBLE-5, as well as DTF-IV, an Sn multigroup transport code written at Los Alamos. Neutron cross sections for energies above thermal (>1 eV) were generated by the GGC-3 code. In this code, fine group cross sections (~100 groups), stored for all commonly used isotopes, are averaged over a space independent flux derived by solution of the B1 equations for each discrete reactor region composition. This code and its related cross-section library predict the age of each of the common moderating materials to within a few percent of the experimentally determined values and use the resonance integral work to generate cross sections for resonance materials which are properly averaged over the entire region.

Thermal cross sections were obtained in essentially the same manner using the GGC-3 code. However, scattering kernels were used to describe properly the interactions of the neutrons with the chemically bound moderator atoms (ZrH). The bound hydrogen kernels used for hydrogen in the water were generated by the THERMIDOR code using the thermalization work. Early thermalization work on zirconium hydride has been greatly extended at GA and work by Parks resulted in the SUMMIT code which was used to generate the kernels for hydrogen as bound in ZrH. These scattering models have been used to predict adequately the water and hydride (temperature dependent) spectra as measured at the GA linear accelerator.

#### 4.5.4.4.2 ZrH Model

Qualitatively, the scattering of slow neutrons by zirconium hydride can be described by a model in which the hydrogen atom motion is treated as an isotropic harmonic oscillator with energy transfer quantized in multiples of ~0.14 eV. More precisely, the SUMMIT model uses a frequency spectrum with two branches, one for the optical modes for energy transfer with the bound proton, and the other for the acoustical modes for energy transfer with the lattice as a whole. The optical modes are represented as a broad frequency band centered at 0.14 eV, and whose width is adjusted to fit the cross section data of Woods et. al. The low

frequency acoustical modes are assumed to have a Debye spectrum with a cutoff of 0.02 eV and a weight determined by an effective mass of 360.

This structure then allows a neutron to slow down by the transition in energy units of -0.14 eV as long as its energy is above 0.14 eV. Below 0.14 eV, the neutron can still lose energy by the inefficient process of exciting acoustic Debye type modes in which the hydrogen atoms move in phase with the zirconium atoms, which, in turn, move in phase with one another. These modes, therefore, correspond to the motion of a group of atoms whose mass is much greater than that of hydrogen, and indeed, even greater than the mass of zirconium. Because of the large effective mass, these modes are very inefficient for thermalizing neutrons, but for neutron energies below 0.14 eV, they provide the only mechanism for neutrons slowing down within the ZrH. (In a TRIGA® core, the water also provides for neutron thermalization below 0.14 eV.) In addition, in the ZrH, it is possible for a neutron to gain one or more energy units of ~0.14 eV in one or several scatterings from excited Einstein oscillators. Since the number of excited oscillators present in a ZrH lattice increases with temperature, this process of neutron speeding up is strongly temperature dependent and plays an important role in the behavior of ZrH-moderated reactors.

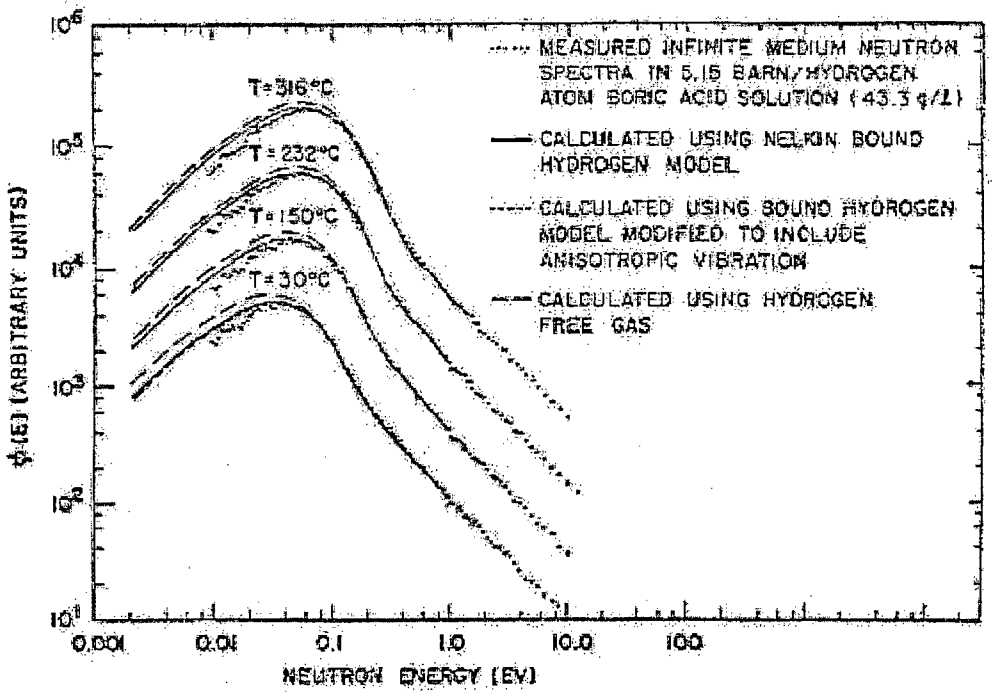


Figure 4.20: A Comparison for Neutron Spectra Between Experiments and Several Hydrogen Models

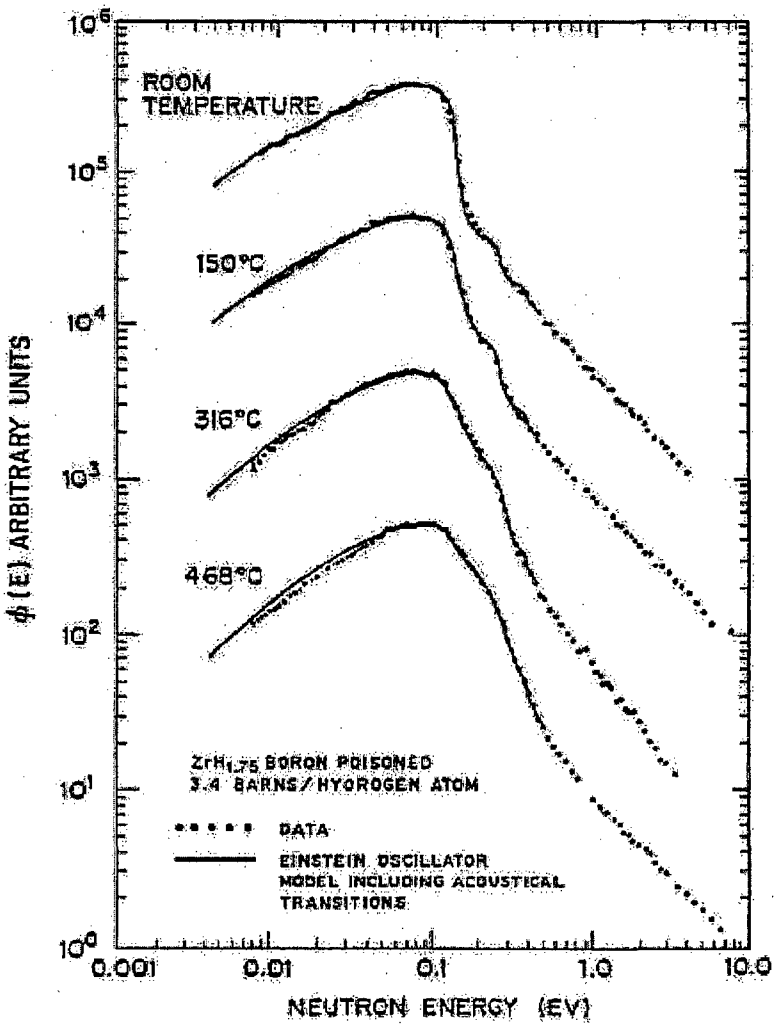


Figure 4.21: Effect of Temperature Variation on ZrH Neutron Spectra

#### 4.5.4.4.3 Calculations

Calculations of ($\alpha$T) were done in the following steps:

- Multigroup cross sections were generated by the GGC-3 code for a homogenized unit cell. Separate cross-section sets were generated for each fuel element temperature by use of the temperature dependent hydride kernels and Doppler broadening of the $^{238}$U resonance integral to reflect the proper temperature. Water at room temperature was used for all prompt coefficient calculations;
- A value for $k_\infty$ was computed for each fuel element temperature by transport cell calculations. Group dependent disadvantage factors were calculated for each cell region (fuel, clad, and water) where the

Table 4.9: Hydraulic Flow Parameters

| | |
|---|---|
| Flow area ($ft^2$/elem) | 0.0058 |
| Wetted perimeter (ft/elem) | 0.3862 |
| Hydraulic diameter (ft) | 0.0601 |
| Fuel element diameter (ft) | 0.1229 |
| Fuel surface area ($ft^2$) | 0.4827 |

disadvantage factor is defined as the ratio: $\phi_g^r/\phi_g^c$ (region/cell);

- The thermal group disadvantage factors were used as input for a second GGC-3 calculation where cross sections for a homogenized core were generated which gave the same neutron balance as the thermal group portion of the discrete cell calculation; and

- The cross sections for an equivalent homogenized core were used in a full reactor calculation to determine the contribution to ($\alpha$T) due to the increased leakage of thermal neutrons into the reflector with increasing hydride temperature. This calculation still requires several thermal groups, but transport effects are no longer of major concern. Thus, reactivity calculations as a function of fuel element temperature have been done on the entire reactor with the use of diffusion theory codes. Results from the above calculations indicate that more than 50% of ($\alpha$T) for a standard TRIGA® core comes from the temperature-dependent disadvantage factor or "cell effect" and approximately 20% each from Doppler broadening of the $^{238}U$ resonances and temperature dependent leakage from the core. These effects make ($\alpha$T) approximately 0.01% $\Delta k/k \cdot °C$, which is rather constant with temperature. The temperature coefficient is shown in Figure 4.22 for a typical high-hydride TRIGA® core.

#### 4.5.4.5 Steady-State Reactor Power

The following evaluation has been made for a TRIGA® system operating with cooling from natural convection flow around the fuel elements. This analysis investigates the limits to which such a system may be operated.

The analysis was conducted by considering the hydraulic characteristics of the flow channel from which the heat rejection rate is a maximum. The geometrical data from this channel is given in Table 4.9. All symbols in Equations 4.11 through 4.59 are defined in the list of nomenclature in Section 4.5.4.5.10.

The heat generation rate in the fuel element is distributed axially in a cosine distribution shaped at the end such that the peak-to-average ratio is 1.25. The number of fuel elements in the core is assumed to be 100 elements for 1 MW operation, but the departure from nucleate boiling (DNB) ratio is conservatively evaluated on the basis of 85 elements.

The driving force is supplied by the buoyancy of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel, the acceleration and potential energy and friction losses in the cooling channel itself.

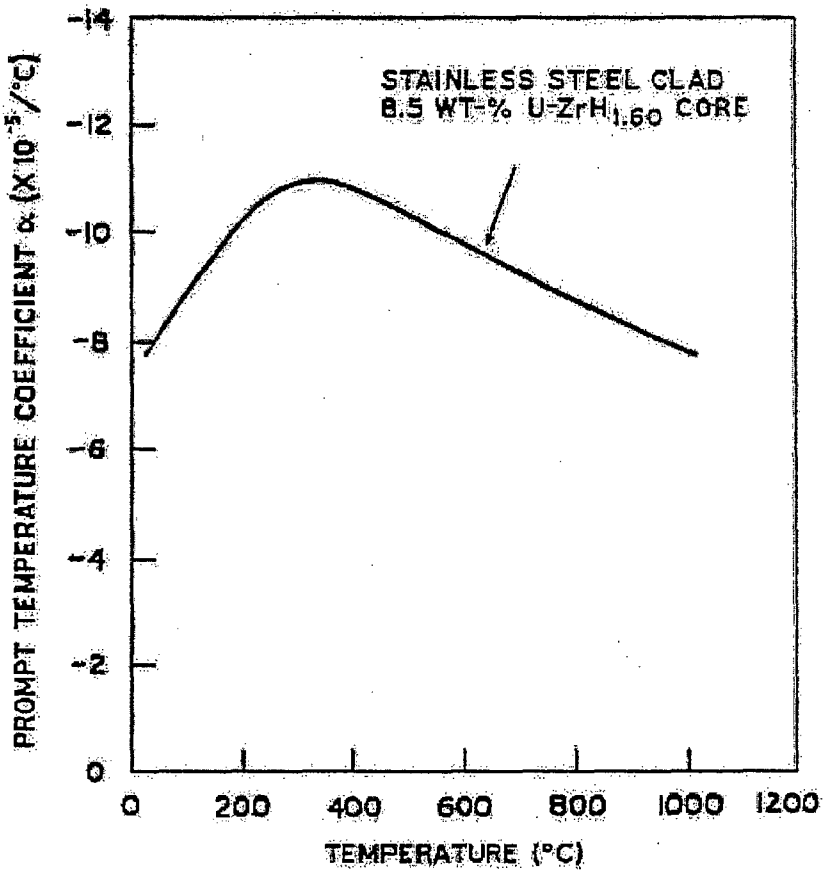


Figure 4.22: Prompt Temperature Coefficient of Reactivity Vs. Average Fuel Temperature for TRIGA® Standard Fuel

Figure 4.23 illustrates schematically the natural convection system established by the fuel elements bounding one flow channel in the core. The system shown is general and does not represent any specific configuration. Steady-state flow is governed by the equation

$$\partial p_i + \partial p_e + \partial p_1 + p_u + \sum_{j=1}^{n} w p_j = z_t / v_0, \tag{4.26}$$

where the left-hand members represent the pressure drops through the flow channel due to entrance, exit, friction, acceleration, and gravity losses and the right-hand member represents the driving pressure due to the static head in the pool. The pressure drops through the flow channel are dependent on the flow rate while the available static driving pressure is fixed for a known core height and pool temperature. The analysis, therefore, becomes an iterative one in which the left-hand side of Equation 4.26 is evaluated on the basis of an assumed flow rate and compared with the known right-hand side until equality is achieved. The method has been programmed for digital computer solution. The methods of evaluating each of the *p terms in

Equation 4.26 for known power distribution and flow geometry and assumed flow rates are discussed below.

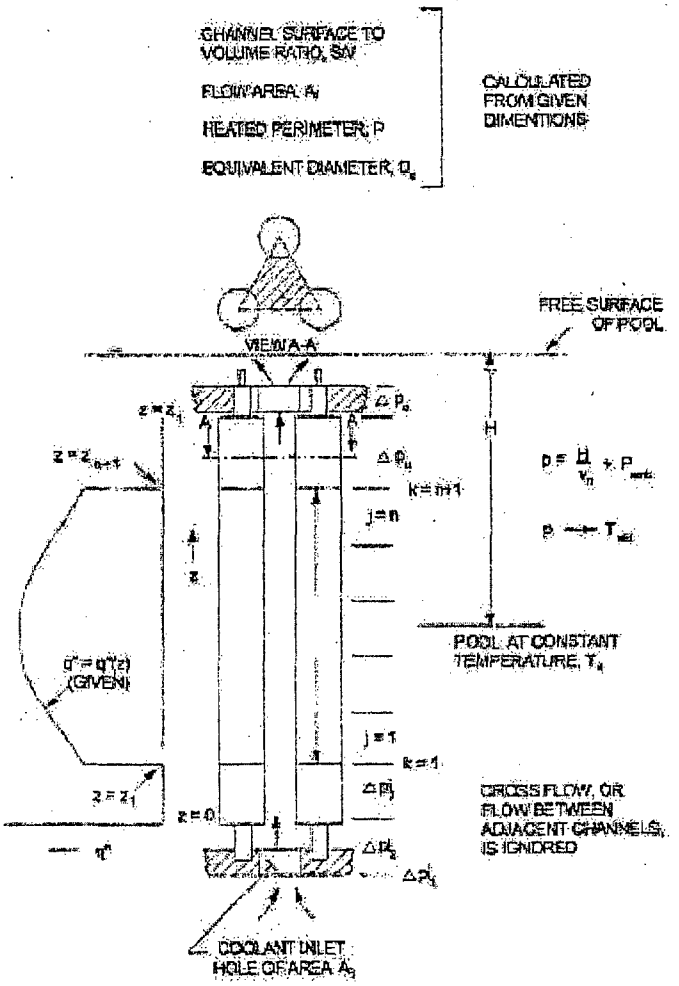


Figure 4.23: General Fuel Element Configuration for Single Coolant Channel in TRIGA® Fuel

#### 4.5.4.5.1 Entrance Loss $\partial p_i$

The entrance loss, $\partial p_i$, may be evaluated in the usual way as a fraction of the velocity head in the lower grid plate hole:

$$\partial p_i = (k_{i1} + k_{i2}) v_0 (NW)^2 / 2g(A_i)^2, \tag{4.27}$$

where:

$N$ = the number of channels which receive their flow from a single hole in the lower grid plate,

$k_{i1}$ = the loss factor for the entrance to the hole in the lower grid plate. For even slight rounding of the entrance, $k_{i1}$ will be no greater than 0.30, and

$k_{i2}$ = the loss factor covering transfer of the flow from the hole in the lower grid plate to the coolant channels. In most cases this can be satisfactorily approximated as a sudden expansion using $k_{i2} = 1.0$.

#### 4.5.4.5.2 Exit Loss $\partial p_e$

The exit loss is expressed in terms of a coefficient $K_e$ which is the fraction of the velocity head in the flow channel which is not recovered:

$$\partial p_e = \frac{K_e v_{n+1}}{2gA_f^2} W^2. \tag{4.28}$$

The term $V_{n+1}$ is the specific volume at the highest axial station along the heated length of the core. It is evaluated from the temperature $T_{n+1}$ which is obtained from an overall heat balance:

$$T_{n+1} = \frac{q_t}{WC} + , \tag{4.29}$$

where:

$$q_t = P \int_{z_q}^{z_{n+1}} q''(z) \mathrm{d}z. \tag{4.30}$$

#### 4.5.4.5.3 Loss Through Portion of Channel Adjacent to Lower Reflector $\partial p_1$

The flow is isothermal at the bulk pool temperature so that

$$\partial p_1 = \frac{f_m v_0 \partial z_1}{2gD_e A_f^2} W^2 + \frac{\partial z_1}{v_0}. \tag{4.31}$$

The term $f_m$ is evaluated from the Moody chart (assuming smooth surface) on the basis of a Reynolds number which is

$$R_e = \frac{D_e v_0}{A_f \nu_0} W. \tag{4.32}$$

#### 4.5.4.5.4 Loss Through Portion of Channel Adjacent to Upper Reflector, $\partial p_u$

The flow is isothermal at $T_{n+1}$ where $T_{n+1}$ is determined by Equation 4.29. Thus,

$$\partial p_u = \frac{f_m v_n \partial z_u}{2gD_e A_f^2} W^2 + \frac{\partial z_u}{vn}. \tag{4.33}$$

The term $f_m$ is again evaluated from the Moody chart, assuming smooth surface, on the basis of a Reynolds number which is

$$R_e = \frac{D_e v_n}{A_f \nu_n} W. \tag{4.34}$$

#### 4.5.4.5.5 Loss Through Each Increment of the Channel Adjacent to the Fueled Portion of the Elements, $*p_i$

It is initially assumed that the entire heated portion of the channel is in subcooled boiling. This implies that the wall temperatures calculated from subcooled boiling correlations are lower than those calculated for convection alone and that the liquid is below its saturation temperature at all locations. The pressure drop through an increment is given by

$$\partial p_{n-(n+1)} = \underbrace{\frac{v_{m_{k+1}} - v_{m_k}}{g A_f^2} W^2}_{\text{(acceleration)}} + \underbrace{\frac{f_{b_j} v_{m_j} \partial z}{2 g A_f^2 D} W^2}_{\text{(friction)}} + \underbrace{\frac{\partial z}{v_{m_j}}}_{\text{(gravity)}}. \tag{4.35}$$

#### 4.5.4.5.6 Acceleration Term

The term $v_m$ denotes the mean specific volume and is larger than the liquid specific volume, because of the vapor voidage:

$$v_m = v/(1 - \alpha), \tag{4.36}$$

where $\alpha$ is the void fraction or the fraction of a channel cross section which is occupied by vapor. The term $\alpha$ may be calculated from the vapor volume (cubic inches vapor/square inches heating surface) and the flow channel geometry. Denoting the vapor volume as $\xi$,

$$\alpha = \xi(S/V), \tag{4.37}$$

where $S/V$ is the surface to volume ratio of the coolant channel. The parameter $\xi$ is dependent on the surface heat flux, the subcooling of the liquid and the velocity of the liquid. It can be evaluated only by experiment. Data were used to estimate $\xi$; these data are plotted in Figures 4.24 and 4.25. Most of this represents a flow velocity of 4 ft/sec and appears to be the only available data applicable under the thermal conditions encountered in TRIGA®-type reactors. Extrapolations from these data are made for flow velocities different from 4 ft/sec. The extrapolations were based on a small amount of data given for flow velocities other than 4 ft/sec. The liquid temperature at a station $k$, $T_k$, may be calculated from:

$$T_k = \frac{P \int_{z_1}^{z_k} q''(z) \mathrm{d}z}{WC} + T_0. \tag{4.38}$$

$T_{sat}$ and $q''_k$ are known. Therefore, from $T_{sat} - T_k$ and $q''_k$ one finds $\xi_k$ (Figure 4.25)

Since

$$\alpha_k = \xi_k(S/V), \tag{4.39}$$

and $v_k$ is a function of $T_k$, $v_{mk}$ may be evaluated from Equation 4.36.

#### 4.5.4.5.7 Friction Term

The term $v_{mj}$ denotes a linear average of the mean specific volumes at the upper and lower boundaries of an increment. The approximate mean value is assumed to apply over the entire increment so that:

$$v_{m_j} = \frac{v_{m_k} + v_{m_{k+1}}}{2}. \tag{4.40}$$

A friction factor, $f_{bj}$, is applied locally to calculate the friction pressure drop over the increment in subcooled boiling. Jordan and Leppert developed the correlation

$$f_b = 8S_t = \frac{8h_b}{\rho CV} = \frac{8q''}{\rho CV(T_W - T)}, \tag{4.41}$$

and provide experimental verification near atmospheric pressure in the range $0.0015 < S_t < 0.0050$. This is simply an extension of Reynold's analogy to the case of subcooled boiling. The equation of continuity is used to write Equation 4.41 as:

$$f_b = \frac{8q''A_f}{WC(T_W - T)}, \tag{4.42}$$

which may be evaluated if $T_W$ is known. For subcooled boiling, the heat transfer is usually defined by an experimentally determined correlation of $q''$ vs. $(T_W - T_{sat})$, which has been obtained over a given range of flow velocity and pressure. such a correlation is given for pressures between 2 and 6 atmospheres and flow velocities between 1 and 12 ft/sec. This correlation will be used to determine $T_W$ for use in Equation 4.42.

Approximate mean values are assumed to apply over the entire increment so that:

$$f_{b_j} = (1/2)\,\frac{8A_f}{WC}\left[\frac{q''_k}{T_{wk} - T} + \frac{q''_{k+q}}{T_{wk+1} - T_{k+1}}\right], \tag{4.43}$$

and

$$T_W - T_{\mathrm{sat},j} = \frac{\Phi(q''_k) + \Phi(q''_{k+1})}{2}, \tag{4.44}$$

where $\Phi(q''_k)$ is the correlation previously cited.

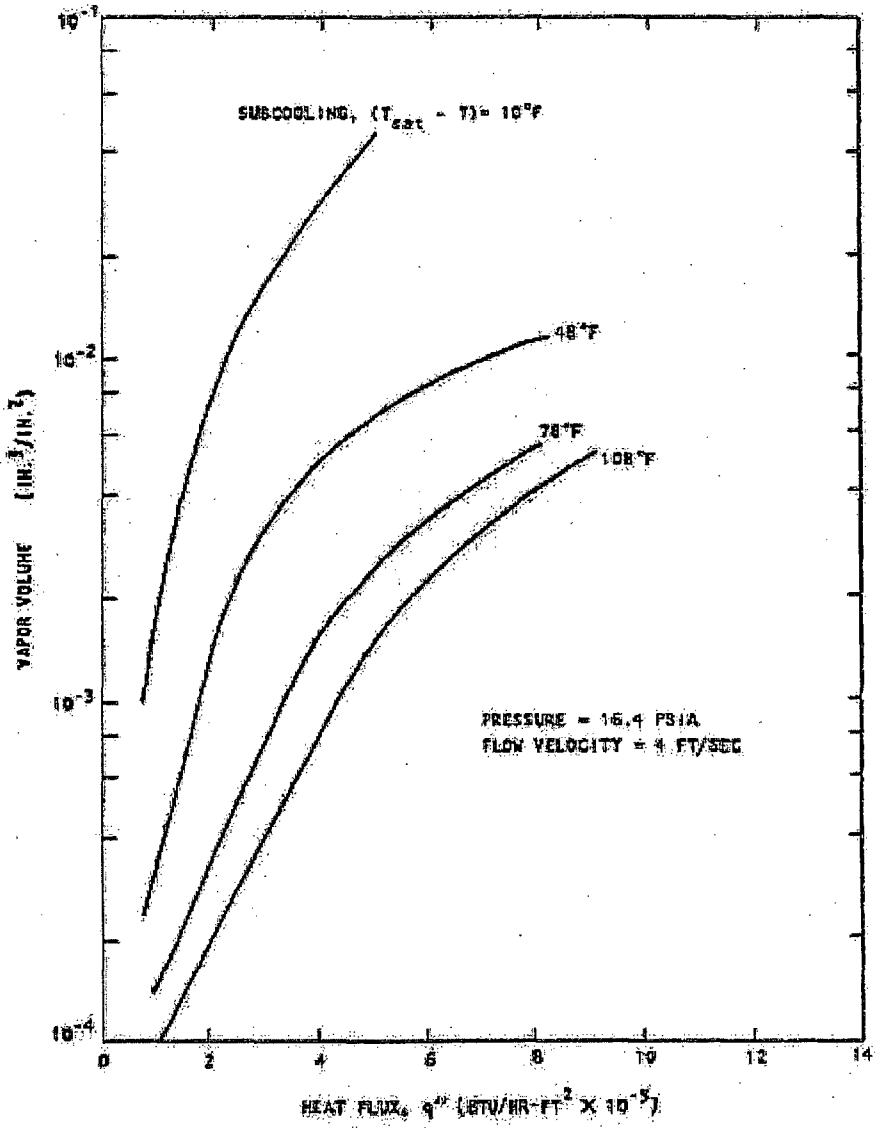


Figure 4.24: Experimentally Determined Vapor Volumes for Subcooled Boiling in a Narrow Vertical Annulus

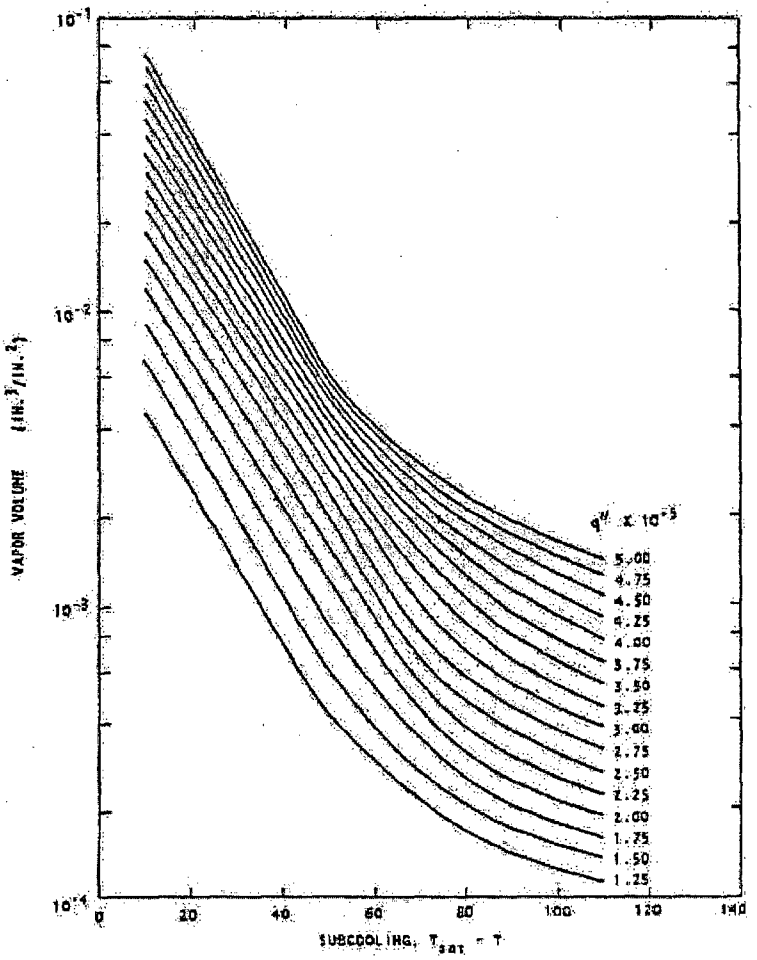


Figure 4.25: Cross Plot of Figure 4.23 Used in Calculations

#### 4.5.4.5.8 Gravity Term

The gravity term is evaluated using $v_{m_j}$ calculated from Equation 4.40. As implied in Section 4.5.4.5.5, each increment must be checked to determine whether heat is being transferred by subcooled boiling or by convection. The term $T_W$ is evaluated at the lower boundary of the increment on the basis of both the correlation for subcooled boiling and a standard correlation for convection. If the $T_W$ calculated from convection correlations is less than that obtained for subcooled boiling, boiling is assumed not to be present in the increment. Equation 4.35 still applies, but since there is no boiling, and hence no vapor void, $v_m$ becomes $v$ and $f_b$ becomes $f_m$.

In the foregoing analysis, an assumption was made that all of the vapor formed on the surface of the fuel element detaches and adds to the fluid buoyance. This is not a conservative assumption. The position where vapor bubbles first leave the heated surface is obtained from two considerations; first, the balance of

the forces exerted on the vapor bubble while it is in contact with the wall (buoyancy, surface tension, and friction), and second, the temperature distribution in the single phase liquid away from the walls.

Determination of the buoyancy forces resulting from the formation and subsequent detachment of vapor bubbles is complicated by the difficulty in predicting the point at which the vapor detaches, and the fraction of that vapor which subsequently condenses. The problem was simplified by making use of an analysis performed to determine the position at which the vapor detaches from the wall, assuming that at that point all of the vapor detaches and, finally, that there is no recombination of the vapor with subcooled fluid.

According to Levy, the position at which the vapor leaves the surface is obtained from considering the balance of forces exerted on the vapor bubble while it is in contact with the wall, and the temperature distribution in the single phase liquid away from the wall.

The forces acting on the bubble in the vertical direction consist of a buoyant force, $F_B$, a frictional force, $F_F$, exerted by the liquid on the bubble, and a vertical component of the surface tension force, $F_S$.

The buoyant force, $F_B$, is given by:

$$F_B = \frac{C_B r_B^3 (\rho_L - \rho_v) g}{g_c}, \tag{4.45}$$

where $r_B$ is the bubble radius, $C_B$ is a proportionality constant, $\rho_L$ and $\rho_v$ are the liquid and vapor density, $g$ is the acceleration due to gravity and $g_c$ is the conversion ratio from lb-force to lb-mass. The frictional force, $F_F$, is related to the liquid frictional pressure drop per unit length, $(-d_p/d_z)F$. The pressure differential across the bubble is proportional to the pressure differential times the bubble radius and acts across an area proportional to the square of the bubble radius. Relating the pressure differential to the wall shear stress, $\tau_W$,

$$-(\mathrm{d}p/\mathrm{d}z)_F = 4\tau_W/D_H, \tag{4.46}$$

the following result for $F_F$ is then obtained:

$$F_F = C_F \frac{\tau_W}{D_H} r_B^3, \tag{4.47}$$

where $C_F$ is a constant of proportionality and $D_H$ is the hydraulic diameter (four times the cross-sectional area divided by the wetted perimeter). The surface tension force, $F_S$, is given by:

$$F_S = C_S r_B \sigma, \tag{4.48}$$

where $C_S$ is a proportionality constant and $F_S$ is the surface tension. Assuming upward flow, the balance of these forces results in the following solutions for the bubble radius:

$$r_B = \left[ \frac{C_S \sigma}{C_B \frac{g}{g_c} (\rho_L - \rho_v) + C_F \frac{\tau_W}{D_H}} \right] (1/2). \tag{4.49}$$

Assuming that the distance from the wall to the tip of the bubble is proportional to the bubble radius, a non-dimensional distance corresponding to this real distance can be given by:

$$Y_B = C\frac{(\sigma g_C D_H - \rho_L)^{1/2}}{\mu_L}\left[1 + c'\frac{g}{g_c}\frac{(\rho_L - \rho_v)}{\tau_W}\right]^{-1/2}, \tag{4.50}$$

where $C$ and $C'$ are appropriate constants. For those cases where the fluid forces are considerably greater than the buoyant forces, this expression reduces to:

$$Y_B = C(\sigma g_c D_H - \rho_L)^{1/2}(1/\mu_L). \tag{4.51}$$

For the bubble to detach, the fluid temperature at the tip of the bubble must exceed the saturation temperature by an amount such that the pressure differential acting across the interface at the tip of the bubble balances the surface tension forces at the same position. By using the Clausius-Clapeyron solution of this pressure differential, one finds that the fluid temperature-saturation temperature difference can be assumed to be zero.

The temperature at the tip of the bubble can be specified from existing solutions for the fluid temperature distribution. Thus, if the flow is assumed to be turbulent, and using a possible solution, we have:

$$\begin{aligned} T_W - T_B &= \theta P_t Y_B && ; \quad 0 \le Y_B \le 5 \\ &= 5\theta\left\{P_r + \ln\left[1 + P_r\left(Y_b/5 - 1\right)\right]\right\} && ; \quad 5 \le Y_B \le 30 \\ &= 5\theta\left\{P_r + \ln\left[1 + 5P_t\right] + 0.5\ln\left[Y_B/30\right]\right\} && ; \quad Y_B > 30 \end{aligned}$$

The parameter $\theta$ is a non-dimensional term defined through the heat flux and liquid specific heat, that is,

$$\theta = \frac{q/A}{\rho_L C_{pL}\left(\tau_W g_c/\rho_L\right)^{1/2}}. \tag{4.52}$$

Levy obtained values for the constants $C$ and $C_N$ by correlation with available experimental data. Using the accepted heat-transfer relation from Dittus-Boelter, one obtains:

$$hD_H/k_L = 0.023\left(WD_H/\mu_L\right)^{0.8}\left(\mathrm{Pr}\right)^{0.4}. \tag{4.53}$$

Calculating the friction factor from:

$$f = 0.0055\left\{1 + \left[20{,}000\left(\epsilon D_H\right) + 10^6/\left(WD_H/\mu_L\right)\right]^{1/3}\right\}, \tag{4.54}$$

we are able to find the wall shear stress from:

$$\tau_W = (f/8)\left(W^2/\rho_L g_c\right) \tag{4.55}$$

The correlation with experimental results yielded values for the constants of:

$$\begin{aligned} c &= 0.015 \text{ and} \\ c' &= 0. \end{aligned} \tag{4.56}$$

Finally, from the definition of the heat transfer coefficient, one obtains:

$$T_W - T = (q/A)/h, \tag{4.57}$$

and setting the bubble tip temperature, $T_B$, equal to the saturation temperature, $T_{\text{sat}}$, we can express the relationship between the saturation temperature, the wall temperature, and the fluid temperature at which the bubble would detach from the wall by:

$$(T_W - T_{\text{sat}})/(T_W - T) = 0.023\,(WD_H/\mu_L)^{-0.2}\,(\text{Pr})^{-0.6}\,(f/8)^{-0.5}\,\Omega, \tag{4.58}$$

where:

$$\begin{aligned} \Omega &= \text{Pr}Y_B && ; \quad 0 < Y_B < 5 \\ &= 5\{\text{Pr} + \ln[+\text{Pr}\,(0.2Y_B - 1)]\} && ; \quad 5 \le Y_B < 30 \\ &= 5\{\text{Pr} + \ln[1 + 5\text{Pr}] + 0.5\ln[Y_B/30]\}. \end{aligned} \tag{4.59}$$

#### 4.5.4.5.9 Results of Calculations

The solution of the force balance equation with void detachment was accomplished by iterating on the void detachment point to find where the right and left sides of Equation 4.58 were equal. The point at which the void was assumed to separate from the surface was taken as the point at which equality was obtained.

The peak heat flux, that is, the heat flux at which there is a departure from nucleate boiling and the transition to film boiling begins, was determined by two correlations. The first, indicates that the peak heat flux is a function of the fluid velocity and the fluid only. The second encompasses a wider range of variables over which the correlation was made and takes into account the effect of differential flow geometries. It generally gives a lower value for the peak heat flux and is the value used for determining the minimum DNB ratio, that is, the minimum ratio of the local allowable heat flux to the actual heat flux. In general, the first correlation gives a DNB ratio 50% to 80% higher than the second correlation.

Figure 4.26 shows the results of this analysis. This figures shows the maximum channel heat flux for which the DNB ratio is 1, with bulk pool water temperature as a parameter. It is assumed that all the vapor above the detachment point separates from the heated surface. From this figure, it can be seen that with the design cooling water temperature at the core inlet (120 °F) the maximum heat flux is 325 kBTU/h·ft$^2$. For a 75 or 85 element core with an overall peak-to-average power density ratio of 2.0, this heat flux corresponds to a maximum reactor power of 1,675 and 1,900 kW, respectively.

#### 4.5.4.5.10 Nomenclature

$A$ cross-sectional area, $ft^2$

$A_f$ channel free flow area, $ft^2$

$C$ coolant specific heat, Btu/lb·°F

$d$ diameter, inches

$D_e$ channel equivalent diameter, ft

$D_H$ hydraulic diameter, ft

$f_b$ friction factor with subcooled boiling, dimensionless

$f_m$ friction factor without boiling, dimensionless

$F$ forces acting on vapor bubble

$g$ constant, $4.18 \times 10^8$ $ft/h^2$

$h_b$ heat transfer coefficient with subcooled boiling, Btu/h·$ft^2$·°F

$H$ distance from midplane of heated channel to free surface of pool, ft

$K$ pressure loss factor at channel inlet or exit, dimensionless

$n$ number of equal axial increments into which heated length of core is subdivided

$N$ number of channels which receive their flow from a single opening in the lower grid plate

$p$ absolute pressure, lb/ $ft^2$

$P$ heated perimeter of channel, ft

$Pr$ Prandtl number

$\partial p$ pressure loss, lb/ $ft^2$

$q$ heat load, Btu/h

$q_t$ total heat load to channel, Btu/h

$q''$ heat flux, Btu/h·$ft^2$

$q''_p$ peat or "burnout" heat flux, Btu/h·$ft^2$

$r_B$ bubble radius

$R_e$ Reynolds number, dimensionless

$S/V$ channel surface to volume ratio, in-.1

$T$ coolant temperature, °F

$T_{sat}$ coolant saturation temperature, °F

$v$ specific volume, $ft^3/lb$

$V$ flow velocity, ft/h

$W$ mass flow rate, lb/h

$Y$ non-dimensional radius

$z$ axial coordinate in channel, ft

$z_t$ total length of channel, ft

$\Delta z$ length of a calculation increment in the channel, ft

$\alpha$ void fraction or fraction of a channel cross section which is occupied by vapor, dimensionless

$F_s$ surface tension, lb/ft

$\xi$ vapor volume, or volume of vapor produced per unit area of heated surface, cubic inch/square inch

$\eta$ kinematic viscosity, $ft^2/h$

$\tau_W$ shear stress, $lb/ ft^2$

$\rho$ density, $lb/ft^3$

Subscripts

$e$ conditions at channel exit

$i$ conditions at channel entrance or inlet

$l$ conditions in portion of channel adjacent to lower reflector

$m$ conditions averaged over the liquid and vapor phases

$o$ bulk pool conditions

$u$ conditions in portions of channel adjacent to upper reflector

$j$ axial increment index

$k$ axial station index

$w$ conditions at cladding outer surface

$v$ vapor

$L$ liquid

## 4.6 Thermal-Hydraulic Design

Very extensive thermal-hydraulic design studies and extensive actual performance tests have been done by GA over the years on reactor cores utilizing TRIGA® fuel. This well known volume of analysis and testing will not be repeated here in this SAR and only the relevant results as they apply to the GSTR will be presented.

The calculated and measured heat transfer characteristics of four TRIGA® reactors, including the GSTR, are given in Table 4.10. The McClellan reactor (1 MW) operated throughout its early lifetime with only LEU fuel with 85 elements compared to the GSTR with 125 elements. These additional fuel elements in the GSTR reduces the maximum heat flux and the maximum fuel temperature, and all calculations and tests were done under conditions of natural convection circulation of water through the various cores. Design-basis conditions evaluated for TRIGA® reactors using stainless steel clad U–$ZrH_{1.7}$ fuel elements provide a generous safety margin for the GSTR. These general evaluations are supported by extensive experience in operation of TRIGA® cores at equivalent fuel temperatures and power levels. No adverse results are reported from operation of TRIGA® cores at fuel temperatures and power levels greater than this design. Table 4.11 lists the pertinent heat transfer and hydraulic parameters for the typical TRIGA® operating at 1 MW.

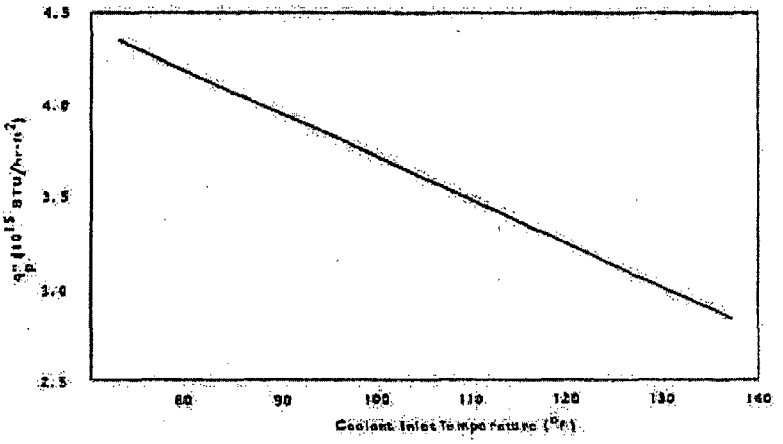


Figure 4.26: Plot for Which DNB Ratio is 1.0 of Maximum Heat Flux Vs. Coolant Temperature

Table 4.10: Comparison of Heat Removal from a Standard TRIGA® Reactors

| | Standard TRIGA[a] | SNRS TRIGA[a] | OSTR | GSTR |
|---|---|---|---|---|
| Reactor Power (MW) | 1.5 | 1.0 | 1.0[b] | 1.0 |
| [redacted] | | | | |
| Hydraulic Diameter (ft) | 0.0601 | 0.0601 | 0.0601 | 0.0601 |
| Max. Heat Flux (Btu/h/ft$^2$) | 284,500 | 157,100 | 166,500 | 113,200 |
| Fuel Surface Area (ft$^2$/rod) | 0.4826 | 0.4826 | 0.4826 | 0.4826 |
| Heat Transfer Surface (ft$^2$) | 35.71 | 43.44 | 41.03 | 60.32 |
| Saturation Temp. (°F) | 239 | - | - | - |
| Inlet Temp. (°F) | 60 | 120 | - | - |
| Exit Subcooling (°F) | 0 | - | - | - |
| Mass Flow Rate (#/h·in$^2$) | 1160[a] | - | - | - |
| Min. DNB Ratio | 1.15[a] | 2.0 | - | - |

[a]Extrapolated
[b]Licensed 1.1 MW

Table 4.11: 1 MW SNRS TRIGA® Heat Transfer And Hydraulic Parameters

| [redacted] | |
|---|---|
| Length (heated) | 15.0 inches |
| Flow area | 0.522 ft$^2$ |
| Wetted perimeter | 34.75 ft |
| Hydraulic diameter | 0.0601 ft |
| Heat transfer surface | 43.44 ft$^2$ |
| Inlet coolant temperature | 120 °F (48.9 °C) |
| Exit coolant temperature (average) | 174 °F (78.9 °C) |
| Coolant mass flow | 63,700 lb/h |
| Average flow velocity | 0.55 ft/sec |
| Average fuel temperature | 500 °F (260 °C) |
| Maximum wall temperature | 280 °F (138 °C) |
| Maximum fuel temperature | 842 °F (450 °C) |
| Average heat flux | 78,600 Btu/h·ft$^2$ |
| Maximum heat flux | 157,100 Btu/h·ft$^2$ |
| Minimum DNB ratio | 2.0 |

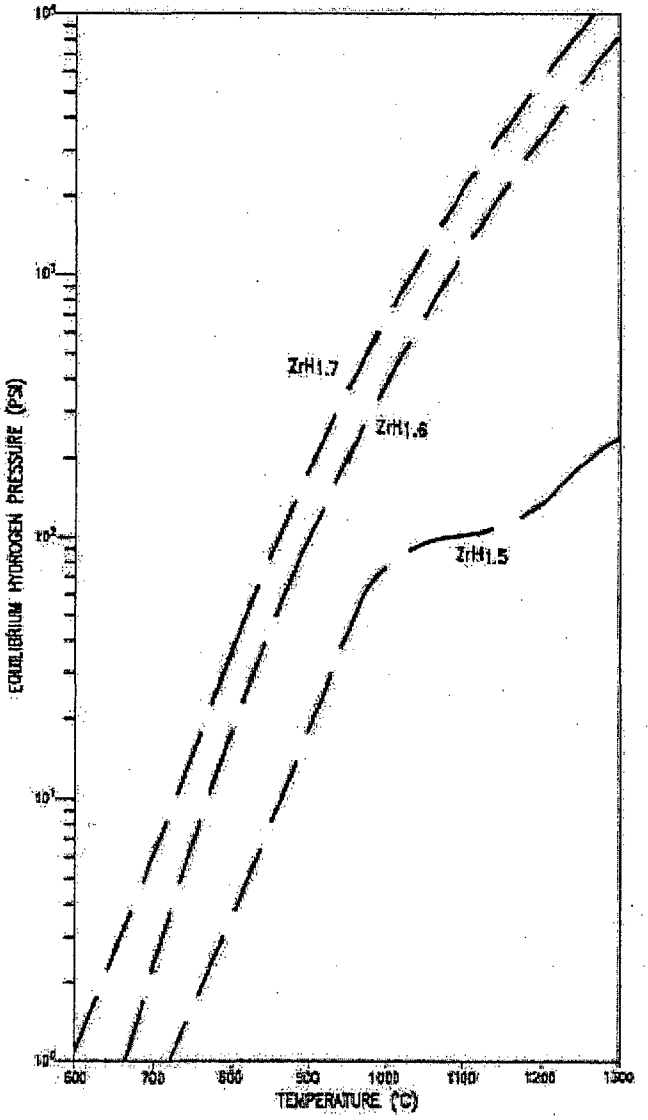


Figure 4.6: Equilibrium Hydrogen Pressure vs. Temperature from Zirconium-Hydrogen

# References

[1] "The U-ZrHx Alloy: Its Properties and Use in TRIGA® Fuel," GA Report 414, February 1980, Mt. Simnad.

[2] "Fission Product Releases from TRIGA®-LEU Reactor Fuels," GA-A16287, November 1980, Baldwin, Foushee, and Greenwood.

[3] "Summary of TRIGA® Fuel Fission Product Release Experiments," Gulf-EES-A10801, September 1971.

[4] Merten, U., et. al., "Thermal Migration of Hydrogen in Uranium-Zirconium Alloys," General Dynamics, General Atomic Division Report GA-3618, 1962.

[5] G. A. Report, "Low Enriched TRIGA Fuel-Water Quench Safety Test," GA-15413, June 1979.

[6] Johnson, H. A., et. al., "Temperature Variation, Heat Transfer, and Void Volume Development in the Transient Atmosphere Boiling of Water," SAN-1001, University of California, Berkeley, 1961.

[7] McAdams, W. H., Heat Transmission, 3rd Ed, McGraw-Hill Book Co., New York, 1954.

[8] Sparrow, E. M. and R. D. Cess, "The Effect of Subcooled Liquid on Film Boiling," Heat Transfer, 84, 149-156 (1962).

[9] Speigler, P., et. al., "Onset of Stable Film Boiling and the Foam Limit," Int. J. Heat and Mass Transfer, 6, 987-989 (1963).

[10] Zuber, W., "Hydrodynamic Aspects of Boiling Heat Transfer," AEC Report AECV-4439, TIS, ORNL, 1959.

[11] Rohsenow, W., and H. Choi, Heat, Mass and Momentum Transfer, Prentice-Hall, 1961, pp. 231-232.

[12] Ellion, M. E., "A Study of the Mechanism of Boiling Heat Transfer," Jet Propulsion Laboratory Memo, No. 20-88, 1954.

[13] Coffer, C. O., et. al., "Characteristics of Large Reactivity Insertions in a High Performance TRIGA® U-ZrH Core," General Dynamics, General Atomic Division Report GA-6216, 1965.

[14] Fenech, H., and W. Rohsenow, "Thermal Conductance of Metallic Surfaces in Contact," USAEC NYO-2130, 1959.

[15] Graff, W. J., "Thermal Conductance Across Metal Joints," Machine Design, Sept. 15, 1960, pp. 166-172.

[16] Fenech, H., and J. J. Henry, "An Analysis of a Thermal Contact Resistance," Trans. Am, Nucl. Soc. 5, 476 (1962).

[17] Bernath, L., "A Theory of Local Boiling Burnout and Its Application to Existing Data," Heat Transfer - Chemical Engineering Progress Symposium Series, Storrs.

[18] Spano, A. H., "Quarterly Technical Report, SPERT Project," April, May, and June 1964, 150-170.

[19] Dee, J. B., et. al., "Annular Core Pulse Reactor," General Dynamics, General Atomics Division Report GACD-6977, Supplement 2, 1966.

[20] Johnson, H. E., "Hydrogen Disassociation Pressures of Modified SNAP Fuel," Report NAA-SR-9295, Atomics International, 1964.

[21] Adir, J., et. al., "Users and Programmers Manual for the GGC-3 Multigroup Cross Section Code," General Dynamics, General Atomics Division Report GA-7157, 1967.

[22] Lenihan, S. R., "GAZE-2: A One-Dimensional, Multigroup, Neutron Diffusion Theory Code for the IBM-7090," General Dynamics, General Atomic Division Report GA-3152, 1962.

[23] Dorsey, J. P., and R. Froehlich, "GAMBLE-5 - A Program for the Solution of the Multigroup-Neutron-Diffusion Equations in Two Dimensions, with Arbitrary Group Scattering, for the UNIVAC-1108 Computer," Gulf General Atomic Report GA-8188, 1967.

[24] Lathrop, D. K., "DTF-IV, A GORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scatterings," USAEC Report LA-3373, Los Alamos Scientific Laboratory, New Mexico, 1965.

[25] Adler, F. T., G. W. Hinman, and L. W. Nordheim, "The Quantitative Evaluation of Resonance Integrals," in Proc. 2nd Intern. Conf. Peaceful Uses At. Energy (A/CONF. 15/P/1983), Geneva, International Atomic Energy Agency, 1958.

[26] Brown, H. D., Jr., Gulf General Atomic Inc., "THERMIDOR - A FORTRAN-II Code for Calculating the Nelkin Scattering Kernel for Bound Hydrogen (A Modification of Robespierre)," unpublished date.

[27] Nelkin, M. S., "Scattering of Slow Neutrons by Water," Phys. Rev. 119, 741-746 (1960).

[28] McReynolds, A. W., et. al., "Neutron Thermalization by Chemically-Bound Hydrogen and Carbon," in Proc. 2nd Intern. Conf. Peaceful Uses At. Energy (A/CONF. 15/P/1640), Geneva, International Atomic Energy Agency, 1958.

[29] Whittemore, W. L., "Neutron Interactions in Zirconium Hydride," USAEC Report GA-4490 (Rev.), General Dynamics, General Atomic Division, 1964.

[30] Bell, J., "SUMMIT: An IBM-7090 Program for the Computation of Crystalline Scattering Kernels," USAEC Report, General Dynamics, General Atomic Division Report GA-2492, 1962.

[31] Beyster, J. R., et. al., "Neutron Thermalization in Zirconium Hydride," USAEC Report, General Dynamics, General Atomic Division Report GA-4581, 1963.

[32] Woods, A. D. B., et. al., "Energy Distribution of Neutrons Scattered from Graphite, Light and Heavy Water, Ice, Zirconium Hydride, Lithium Hydride, Sodium Hydride, and Ammonium Chloride, by the Beryllium Detector Method," in Proc. Symp. Inelastic Scattering of Neutrons in Solids and Liquids, Vienna, Austria, Oct. 11-14, International Atomic Energy Agency, 1960.

[33] Jordon, D. P., and G. Leppert, "Pressure Drop and Vapor Volume with Subcooled Nucleate Boiling," Int. J. Heat Mass Trans. 5, 751-761 (1962).

[34] McAdams, op. cit., pp. 390-392.

[35] Levy, S., "Forced Convection Subcooled Boiling-Prediction of Vapor Volumetric Fraction," Int. J. Heat Mass Trans. 10, 861-965 (1967).

[36] McAdams, op. cit., p. 392.

[37] Bernath, op. cit., pp. 95-116.

[38] "Safety Analysis Report for the Torrey Pines TRIGA® Mark IV Reactor," GA-9064, Gulf General Atomic, January 5, 1970.

[39] "Annular Core Pulse Reactor," General Dynamics, General Atomic Division Report GACD 6977, Supplement 2, September 30, 1966.

# Chapter 5

# REACTOR COOLANT SYSTEMS

## 5.1 Summary Description

Water is used in the GSTR to cool the reactor, to provide a shield for the reactor while still maintaining visibility, and to moderate (thermalize) the fast neutrons to enable the fission reactions to take place. [redacted] Water level over the core is normally maintained at 20 feet to provide radiation shielding over the core. The reactor core is cooled by natural convection water flow through the core area combined with a forced-flow of water into and out of the reactor tank. A remote secondary heat exchanger transfers core heat from the reactor tank primary water system through a secondary water system to a fan-assisted, water-to-air heat exchanger to assist in maintaining low coolant temperatures during extended 1-MW full-power operations.

## 5.2 Primary Coolant System

The primary coolant system, shown in Figure 5.1, consists principally of a pump, heat exchanger, temperature probes, two valves to control flow rates, and associated valves and piping. The primary cooling system pump is a centrifugal type with all "wet" parts, including the case and impeller, made of type 316 stainless steel. It has a 4-inch inlet, a 3-inch outlet, and is driven by a 7.5-hp motor. The primary pump delivers approximately 350 gpm of water through the primary side of the heat exchanger.

A tubular type heat exchanger, manufactured to ASME and TEMA-C specifications, is provided for removal of heat from the reactor. The steel shell with protruding primary connection is 169 inches long by 22 inches outside diameter. The empty weight of the unit is approximately 4000 pounds. It contains 3/4 inch outside diameter U-shaped type 304 stainless steel tubes that are welded into a removable stainless steel tube bundle. All parts of the heat exchanger in contact with the demineralized primary water are made of type 304 stainless steel. The water in the secondary cooling circuit flows on the shell side of the heat

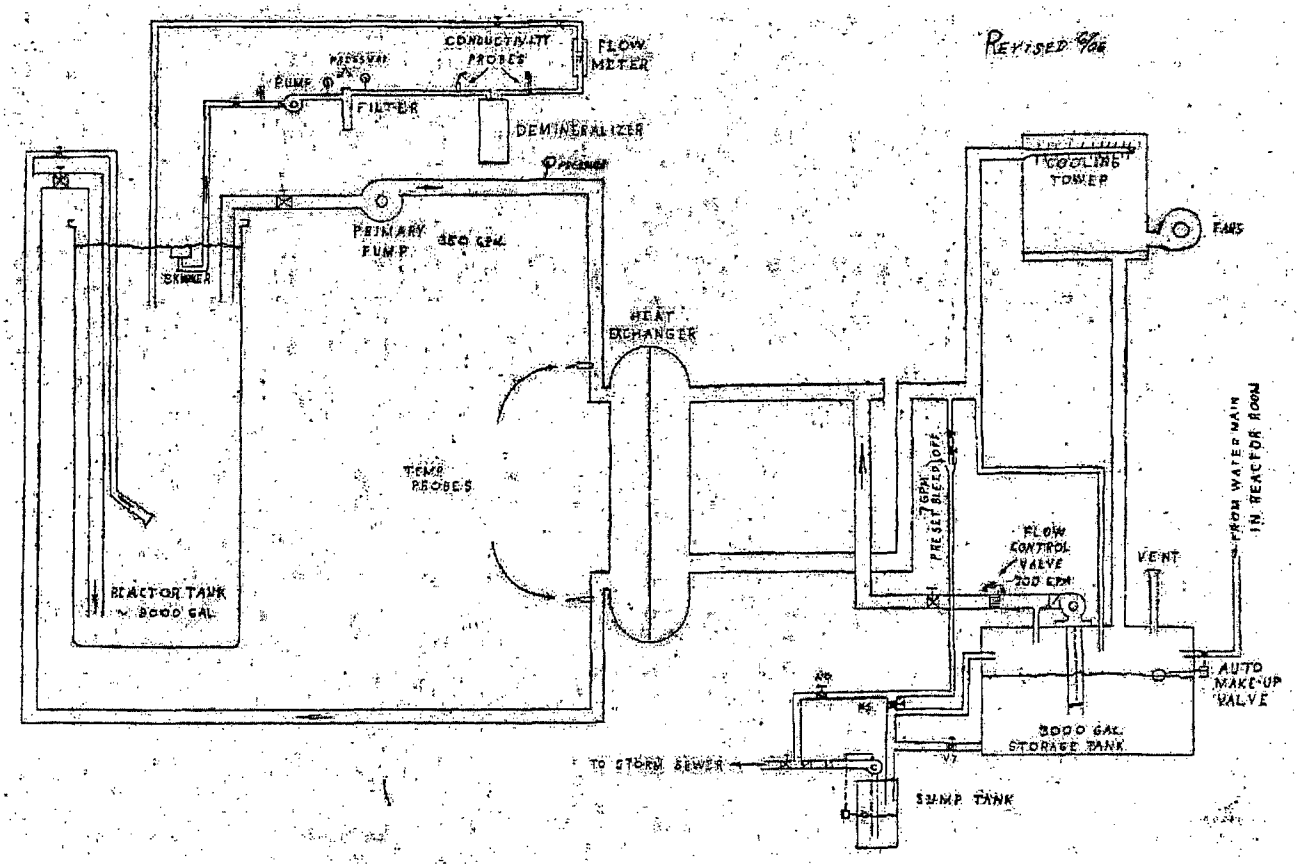


Figure 5.1: Reactor Coolant and Purification Systems.

exchanger. The heat exchanger is designed for a capacity of 3.5 MBtu per hour. The primary cooling system connections to the heat exchanger are made through sleeve joints. The secondary, or shell side, water system connections are made through 10-inch 150-pound ASA flat-faced flanges; special reducing flanges mate with these flanges, permitting attachment to the secondary cooling system piping.

The transport time of the primary water convecting through the core, from the reactor core to the surface of the tank, a distance of approximately 20 feet, permits much of the $^{16}N$ with its 7 second half-life to decay before it reaches the surface. This decay time is further increased by means of a diffuser nozzle located approximately 2 feet above the top grid plate which directs a portion of the cooling water downward over the core.

Temperature probes are used to measure the temperature of the water before and after passing through the heat exchanger. Each probe is a resistance thermometer of the tip-sensitive type. A small coil of nickel alloy wire with a large temperature coefficient of resistivity is used in the probe as a temperature-sensitive element. The temperature probes assist the operator in monitoring water temperatures to assess system operation and ensure that the maximum tank water temperature is not exceeded. An in-tank low-level alarm provides operator notification of decreasing tank level. Minimum water level in the tank is maintained by siphon

break holes in the in-tank system piping at approximately 22 inches below normal tank level.

## 5.3 Secondary Coolant System

The secondary coolant system, shown in Figure 5.1, consists principally of a pump, heat exchanger, cooling tower, connecting piping, valves, and associated instrumentation. The flow rate in this water system is set at approximately 700 gpm.

The water in the secondary coolant system flows into the carbon steel shell side of the heat exchanger described in Section 5.2. The secondary system cooling tower is located outside and holds approximately 400 gallons of water when operating. The cooling tower implements a crossflow design in which the air flow is directed perpendicular to the water flow. Air flow enters one or more vertical faces of the cooling tower to meet the fill material. Water flows through the fill by gravity. The air continues upward through the fill and thus past the water flow into an open plenum area. Gravity distributes the water through the spray nozzles uniformly across the fill material. A set of coarse screens is mounted in the basin to catch coarse debris. The forced air for cooling the inlet water spray is provided by a 20 hp motor turning double-width double-inlet fans. Secondary system pressure exceeds primary system pressure to ensure that a heat exchanger leak results in the containment of potentially contaminated primary fluid.

Make-up water to the cooling tower comes from the city water system and is automatically added as needed by a float valve.

## 5.4 Primary Water Purification System

The primary water purification system, also shown in Figure 5.1, consists principally of a pump, fiber cartridge filter, mixed-bed type demineralizer, flow meter, and surface skimmer all connected by suitable aluminum or PVC piping and valving. Probes for measuring the conductivity of the water as it passes through the system, and pressure guages for determining the degree of filter clogging are also employed.

The surface skimmer collects foreign particles that float on the surface of the reactor-tank water. Water flows over the top of a floating cylinder, enabling large floating foreign particles to be captured in a basket-type screen. Particles small enough to pass through the screen are collected in filter cartridges located downstream in the purification loop.

There are two conductivity probes in the water system. One, located upstream from the demineralizer, measures the conductivity of the water leaving the reactor tank. The other, located downstream from the demineralizer, measures the conductivity of the water as it leaves the demineralizer. The second conductivity probe indicates whether the demineralizer is operating properly or the resin is depleted. Each conductivity probe consists of two electrodes constructed of concentric cylinders and mounted in the water system through a threaded pipe fitting. Each probe has an internal temperature sensor to allow for automatic water temperature compensation. The probes measure water conductivity from 0 to 200 $\mu$S at a temperature range of 32 to 250 °F.

The prime function of the demineralizer is to maintain the conductivity of the water at a sufficiently low level to prevent corrosion of the reactor components exposed to the water, particularly the fuel elements. A secondary function is to remove impurities that could become activated and create a water contamination problem. The demineralizer performs these functions by removing soluble impurities from the water. The demineralizer is a mixed-bed type, removing both positive and negative ions from the circulating water, using approximately 3 cubic feet of anion and cation resin in equal quantities. The positive ions are replaced by hydroxyl (OH-) ions and the negative ions by hydrogen (H+) ions. The OH- and H+ combine to form water. Consequently, any contaminants in the water are concentrated on the resin and replaced by pure water. Any radioactive ions in the water are therefore absorbed and concentrated in the resin bed. In normal use, a demineralizer will become slightly radioactive. Excessive radioactivity levels are warning signs of significantly impure water or a failure of fuel-element cladding.

## 5.5 Primary Coolant Makeup Water System

The makeup water system provides deionized water from a common facility water purification unit for the reactor building. This very pure water source is used to prolong resin life and further reduce primary water activity. Makeup water to compensate for reactor tank evaporation and sampling is added by discharging water from the deioinized water system through the demineralizer and its discharge system directly to the reactor tank.

## 5.6 $^{16}$N Control System

A portion of the reactor tank inlet water is diverted through a diffuser located above the core to create a lateral dispersion of $^{16}$N rising from the core. This dispersion significantly increases $^{16}$N hold-up time.

# Chapter 6

# ENGINEERED SAFETY FEATURES

The GSTR does not have any credited Engineered Safety Features

# Chapter 7

# INSTRUMENTATION AND CONTROL SYSTEMS

## 7.1 Summary Description

The reactor is operated from a console located in the control room. Additional ventilation system instrumentation and controls are housed in a separate cabinet near the console. Another equipment cabinet is located in the reactor bay along with two wall-mounted cabinets that house the digital nuclear power channel (NM1000).

The operating mode of the reactor is determined by four push-button mode selector switches on the console. In Automatic and Steady-State modes, the reactor can operate at power levels up to 1 MW. In Square Wave mode, a step insertion of reactivity rapidly raises reactor power to a steady-state level up to 1 MW. In the Pulse mode, a large-step insertion of reactivity results in a short duration reactor power pulse.

The reactor instrumentation is all solid-state circuitry with a mixture of analog and digital modes of operation.

## 7.2 Design of Instrumentation and Control Systems

Three independent power measuring channels provide for a continuous indication of power from the source level to peak power resulting from the maximum allowed pulse reactivity insertion. Two of the channels are analog instruments and the third is a digital instrument. Trips are provided for over power and loss of detector high voltage on the two analog channels. Fuel temperature is measured for display as well. Other parameters not used by the reactor protection system 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 systems;
- a means for manually withdrawing or inserting control rods;
- automatic control of reactor power level;
- automatic scrams in response to over power and loss of detector high voltage;
- automatic scrams in response to a loss of operability of the digital computer system; and
- monitoring of radiation and airborne radioactivity levels.

### 7.2.2 *Design-Basis Requirements*

The primary design basis for the GSTR is the safety limit on fuel temperature. To prevent exceeding the safety limit, design features, operating limitations, and automatic scrams are provided for over power conditions. Interlocks limit the magnitude of transient reactivity insertions.

### 7.2.3 System Description

#### 7.2.3.1 Reactor Power Measurements

Reactor power is measured by three separate detectors; a fission chamber and two uncompensated ion chambers. The signal from the fission chamber is used by the wide range NM1000 channel and the period channel. One uncompensated ion chamber is connected to the NP1000 safety channel. A second uncompensated ion chamber is used by the NPP1000 percent power and pulsing channel. Figure 7.1 shows the relative ranges of the channels and the detectors.

The fission chamber for the NM1000 wide range instrument is connected to an analog circuitry in a metal box mounted on the wall of the reactor bay. This box contains a high voltage power supply, low voltage power supplies, preamplifier, and other circuitry for the count rate and Campbelling modes of operation. The high voltage power supply also monitors the high voltage to the fission chamber. The analog output from the metal box described above is sent to components housed in a second metal box that is also mounted on the wall of the reactor bay. This box contains low voltage power supplies and microprocessor circuitry to convert the analog signal to a digital signal and transmit that signal to the DAC computer system. The second box also contains circuitry for trip setpoints that provide interlock functions and indications on the console.

The NM1000 log display provides a continuous indication from $10^{-8}$ to 100% of full power for the console display, analog bar graph display, and the console chart recorder.

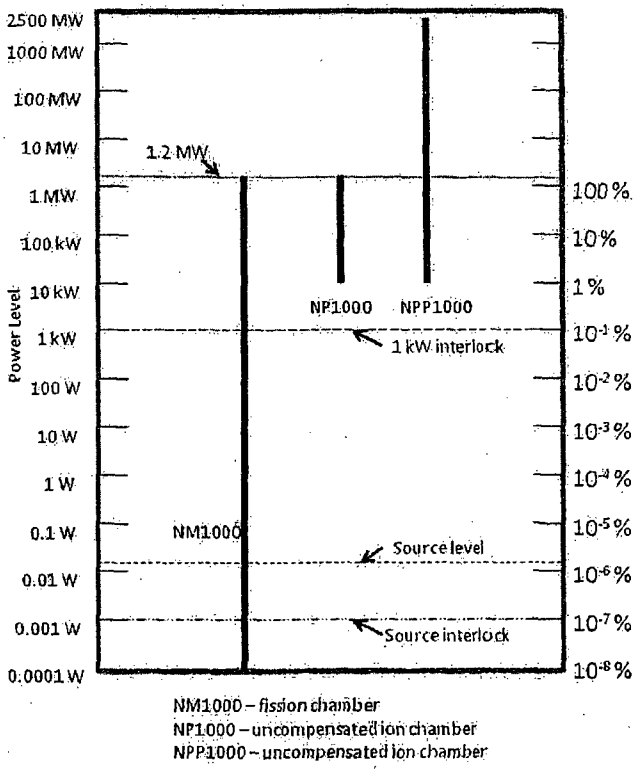


Figure 7.1: GSTR Power Instrument Ranges

The reactor period signal is generated by the microprocessor assembly of the NM1000. Reactor period is displayed on the console display and analog bar graph display. The period signal is also used by the AUTO control system.

The NM1000 linear power signal provides console display power indication on auto-ranging scales that have maximum indications from 1.2 milliWatts to 1.2 MegaWatts. Linear power is displayed on the console display, analog bar graph display, and console chart recorder. The NM1000 linear signal is also used by the AUTO controller system.

The NP1000 safety channel provides a linear power signal to the console display and analog bar graph display. These displays are scaled at 0 to 120% of full power. A bistable circuit provides scram and alarm functions if the high power setpoint is exceeded. The detector input to the NP1000 safety channel is disabled during pulse mode operations. A separate bistable circuit provides a scram signal to the reactor protection system upon a loss of detector high voltage.

The NPP1000 pulsing channel displays peak power from a pulse on the scale of 0 to 2,000 MW on the analog bar graph and a scale of 0 to 2,400 MW on the console display. An analog bar graph display of integrated energy is also provided with a scale of 0 to 30 MW·s. A graphical display of a pulse is available on the console display, along with text information on the pulse number, pulse time and date, full-width at half-maximum power, peak power, integrated power, minimum period, and peak fuel temperature. These data are recorded and may be stored and recalled at a later date. The pulsing channel is enabled when

the pulse mode switch is pressed, as long as all interlock conditions are met. The pulse data collection is performed by the DAC computer and it begins when the pulse rod "Fire" button is depressed. This also enables the peak hold circuit and starts a one-minute timer. The peak power and energy displays are reset at the end of the one-minute period. The peak power is also recorded on the console data recorder.

The NPP1000 channel contains bistable circuits that will produce a scram and alarm output for the conditions of the high power setpoint being exceeded and for loss of high voltage.

#### 7.2.3.2 Temperature Measurements

As illustrated in Figure 7.2, fuel temperature may be measured by three thermocouples embedded in an instrumented fuel element. There are two fuel temperature channels in the reactor instrumentation system, so two thermocouples may be connected at one time. The two thermocouples may be from the same instrumented fuel element or from two different instrumented fuel elements. Fuel temperature is displayed on the console display, console analog bar graphs and on the console data recorder. No scrams are provided for the fuel temperature channels.

Temperature of the bulk pool water is measured by a resistance temperature detector (RTD) and a thermometer. The thermometer is a local readout device only. The RTD is mounted to the top of the reactor tank and the probe extends about 18" below the top of the tank. It sends a signal to the console for display as the pool water temperature. A temperature alarm circuit on the pool water channel will annunciate an audible and visual alarm on the console if the water temperature exceeds a preset temperature. Two additional RTDs are located in the primary piping, one on the inlet to the heat exchanger and one on the outlet of the heat exchanger. The temperature signals from these detectors are sent to the console for display as the pool outlet water temperature and the pool inlet water temperature. These primary piping RTDs may not display accurate temperatures for the primary cooling water if the primary pump is not operating.

## 7.3 Reactor Control System

### 7.3.1 Control Rod Drives

The four control rods are positioned by control rod drives mounted on the reactor bridge.

As illustrated in Figure 7.3, the regulating rod drive and the two shim control rod drives are rack-and-pinion linear actuators. The regulating rod drive uses a stepper motor that is able to operate at variable speeds when operated by the servo system. The regulating rod drive operates at its maximum speed when controlled in the manual mode by the Reactor Operator. The shim rod drives use synchronous AC motors that operate at a single speed. An electromagnet is secured to the bottom of the draw tube to which the rack is mounted. The magnet is moved up or down in response to rotation of the pinion shaft. The control rod is attached to the armature by a long connecting rod. When the magnet is energized, the armature is magnetically coupled to the draw tube. De-energizing the magnet causes the rod to drop. A dash pot is incorporated into the armature section to decelerate the rod near the bottom following a scram. Limit switches sense when the magnet is fully withdrawn, the magnet is fully down, and the armature (and thereby the rod) is fully down.

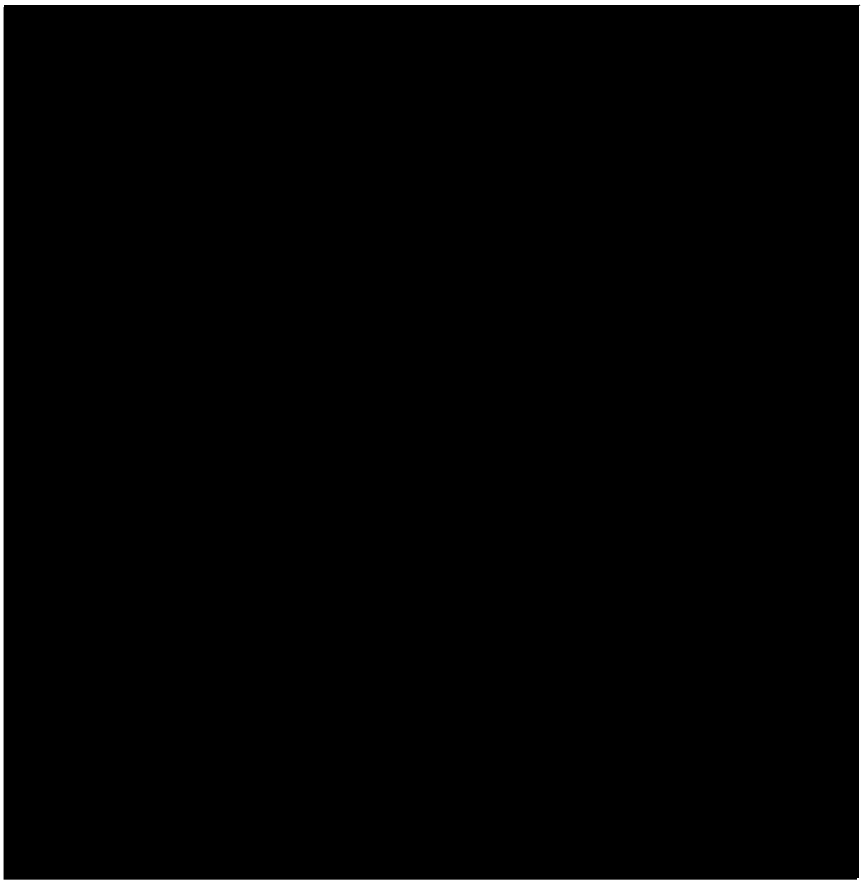

Figure 7.2: Instrumented Fuel Element

A ten-turn potentiometer is coupled to the pinion shaft to provide for rod position indication. The pinion shafts of the safety and shim control rod drives are shaft-coupled to AC gear reduction motors. The pinion shaft of the regulating control rod drive is chain-and-sprocket coupled to a DC stepper motor.

The transient rod (also called the pulse rod) is operated by a pneumatic/electric drive. A connecting rod couples the transient rod to a piston rod assembly. As illustrated in Figure 7.4, the piston resides within an externally threaded cylinder. A ball screw nut acts on these external threads to raise or lower the cylinder. Rotation of the ball screw nut is accomplished by a worm gear coupled to an AC motor. A potentiometer is gear-driven by the worm gear shaft to provide rod position indication. A hydraulic shock absorber is incorporated into the top of the cylinder. Air from a compressor is connected to a normally-closed port of a three-way air solenoid valve. The common port is connected to the transient control rod drive cylinder below the piston. The normally-open port is vented. When the air solenoid valve is energized, air pressure is placed on the bottom of the piston causing the piston to be brought in contact with the shock absorber. The resulting reactivity insertion is dependent on the position of the cylinder prior to applying air. With air applied, energizing the motor in the up or down direction will cause the cylinder, piston, and control rod to move up or down as a unit. Scram of the transient rod is accomplished by de-energizing the air solenoid

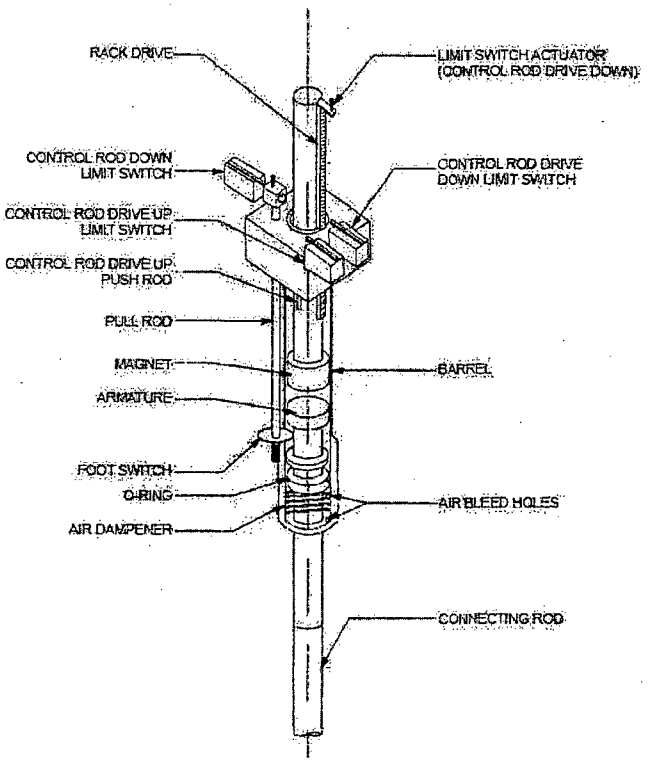


Figure 7.3: Standard Control Rod Drive and Limit Switches

valve. This vents the air pressure under the piston and results in the control rod dropping. As illustrated in Figure 7.5, limit switches provide for sensing cylinder up, cylinder down, and rod down. A bracket extends over the top of the cylinder. A switch on the bracket opens a contact in the up circuitry when the shock absorber assembly contacts it. The bracket itself is substantial enough to stall the motor should the switch contact fail to open.

### 7.3.2 Servo System

In the Automatic and Square-Wave modes of operation, the regulating rod is controlled by the servo system to control reactor power based on input signals from a power channel, the reactor period signal from the NM1000 channel, and the power demand control.

In Automatic, the reactor power is compared against the power demand setting to obtain power error. The period signal is monitored by the controller to limit the reactor period to a minimum of +8 seconds when power is being increased. To reduce hunting of the regulating rod, a deadband is incorporated in the system. The power error signal is used by the DAC computer to determine which direction (if any) the regulating rod needs to move to correct the power error. The regulating rod speed is variable and it will move slowly for small errors and it will move fast for large errors. The regulating rod speed cannot exceed the travel speed that is used in manual control. The variable speed ability of the servo system reduces power overshoot

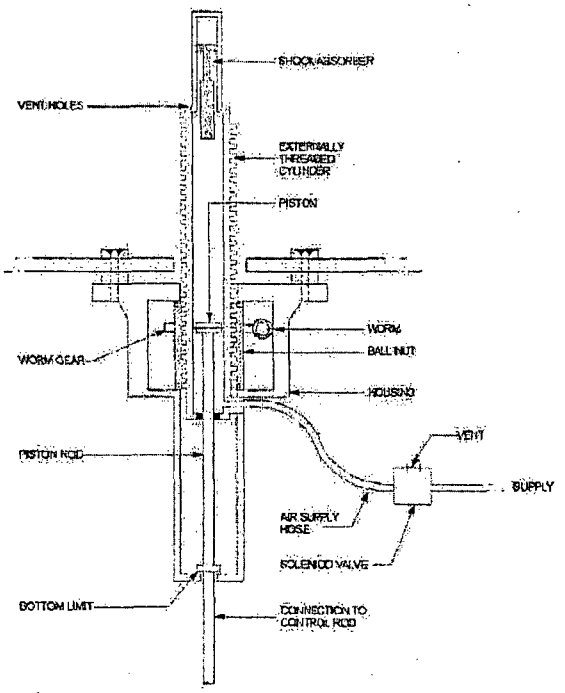


Figure 7.4: Transient Rod Drive

during transients.

To perform a Square-Wave, the reactor must be configured in Steady-State mode. First, the reactor power is raised to some nominal low power (less than 1000 W) with the air to the transient rod off. Second, the transient rod cylinder is raised to the position corresponding to the desired reactivity insertion. Finally, the square wave mode switch is depressed to change the console mode from Steady-State to Square Wave and the transient rod fire button is pressed. Reactor power will increase to the desired power level and then switch to the Automatic mode to maintain a constant power level.

### 7.3.3 Interlocks

The following are the interlocks utilized by the GSTR console:

- the 1-kW permissive interlock to prevent pulsing when wide range log power is above 1 kW;
- interlock to prevent the shim and regulating rods from being withdrawn in pulse mode;
- interlock to ensure that only one control rod can be manually withdrawn at a time in the steady state mode;
- the rod withdrawal prohibit interlock, activated by a low count rate on the NM1000 when the log power is not greater than $1 \times 10^{-7}$% power. An indication is provided on the console low resolution monitor to indicate when a source level rod withdrawal interlock is present.

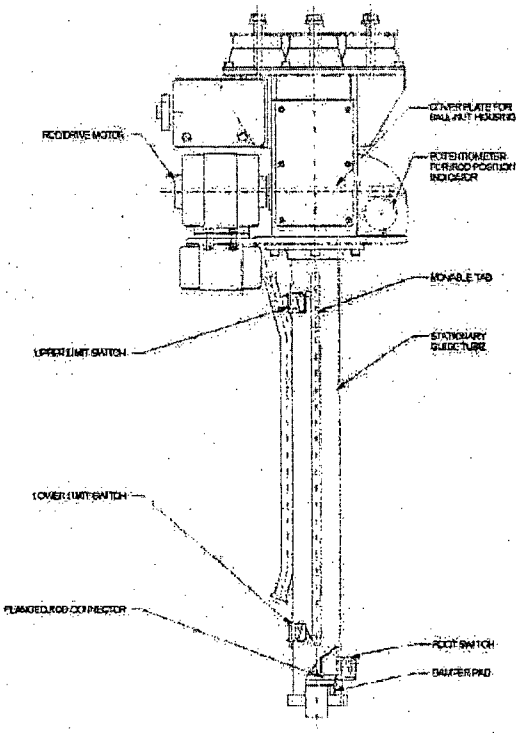


Figure 7.5: Transient Rod Drive Limit Switches

- interlock to prevent the application of air to the transient rod drive mechanism in the steady state mode unless the drive cylinder is fully inserted;
- interlock to ensure that only one control rod can be manually withdrawn at a time in the square wave mode, excluding the transient rod.

## 7.4 Reactor Protection System

### 7.4.1 Scram Circuits

The scram circuits function to shut down the reactor by dropping all four control rods to their fully inserted positions. Scram is accomplished by de-energizing the magnets for the safety, shim, regulating rods and by de-energizing the air solenoid valve for the transient rod.

A reactor scram will result under any of the following conditions:

- operator-initiated manual scram;
- safety or percent power channels measuring power in excess of the setpoint;
- loss of high voltage to the power measuring channels;

- DAC and CSC loss of communication; or
- external scram.

## 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

The console data recorder digitally records and displays wide-range log and linear power and fuel temperature. In Pulse mode, the log input is disconnected and the linear input is replaced with peak power.

### 7.6.2 Rod Position Indication

Four rod position indicators displayed on the console high resolution monitor. Rod positions are displayed as 0 to 1000 units of withdrawal with 0.1% resolution.

### 7.6.3 Annunciator Function

When an alarm or warning is received at the control console, an audible signal will sound and a message will be displayed on the monitor. When the operator presses the acknowledge button, the audible signal will be silenced and the message display will remain until the alarm or warning has cleared. If the alarm or warning condition has cleared, the message will clear.

## 7.7 Radiation Monitoring Systems

### 7.7.1 Area Radiation Monitors

Radiation levels are monitored at strategic locations throughout the reactor building. Each channel consists of a detector, a local meter with alert and alarm functions, and a remote indicator.

### 7.7.2 Airborne Radioactivity Monitors

Two airborne radioactivity monitors are in use at the GSTR. One unit samples the effluent from the reactor bay exhaust stack and determines the amount of $^{41}Ar$ gas being released. The other unit is a continuous air monitor (CAM)that samples in the reactor bay but is located in the reactor control room.

In the $^{41}$Ar stack monitor, air from the reactor bay exhaust stack is transported through a pipe into a shielded air volume that contains a gamma detector. The sampled air then passes out of the shielded volume and is transported back to the building roof and discharged. The gamma detector is connected to instrumentation that detects the $^{41}$Ar gamma peak at about 1293 keV and integrates the pulses in that peak. The integrated $^{41}$Ar that passes through the shielded air volume is then used to periodically calculate and record the total $^{41}$Ar released from the reactor bay exhaust. This is reported as an airborne radionuclide release from the facility.

The reactor bay CAM samples air from the reactor bay and discharges air back into the reactor bay. The sampled air passes through a shielded volume that contains two filters (one paper and one charcoal) that sit in front of a thin window GM detector. The detector and its instrumentation provide an output to indicate the gross beta/gamma activity that is present on the two filters. This signal is displayed locally at the CAM and also on the control console.

Alarms are provided on the CAM for a low level and high level of activity. A high level alarm will result in a signal to switch the ventilation system to the emergency mode and a building evacuation signal. More information regarding the switching to emergency mode via the CAM can be found in Chapter 9.

# Chapter 8

# ELECTRICAL POWER SYSTEMS

## 8.1 Normal Electrical Power Systems

The design of the GSTR is such that the reactor can be shut down and safely maintained in a shutdown condition under a complete loss of electrical power. Three of the control rods are raised with electromagnets, and when electrical power to the magnet is lost the control rods are pulled back into the core by gravity. The fourth control rod is raised through air pressure and when electrical power is lost a solenoid valve removes the air pressure from the control rod, resulting in the control rod being pulled back into the core by gravity.

A schematic representation of the electrical power system is provided in Figure 8.1.

Normal electrical power for the reactor and its associated equipment is supplied by a 13800-VAC, three-phase service from a substation located on the Denver Federal Center to the north-west of Building 15. The service entrance conductors are contained in underground conduits from the substation. The incoming power line passes through a lightning arrester before reaching a 750 kVA transformer in the basement of Building 15 in room B-5, the transformer room. The 750-kVA transformer provides power for the primary pump, secondary pump, sump pump, the cooling tower, the air compressor, the crane, and other systems in Building 15.

The remainder of the reactor loads, as well as the majority of Building 15, are supplied by a 225-kVA transformer and a 45-kVA isolation transformer located in room B-5. A 600-amp breaker in the main switchgear supplies the 750-kVA transformer, which supplies the secondary distribution panel, all of which are located in room B-5. From there, power is distributed to panels and loads throughout the reactor building.

Power for reactor control and instrumentation equipment is supplied by the 5-kVA uninterruptible power supply (UPS). The 5-kVA UPS supplies power to the radiation area monitors (RAMs) monitoring the reactor bay for radiation levels. Therefore a loss in normal operating power will not remove power from the radiation monitoring instruments. More discussion on loss of normal operating power is found in section 8.2.

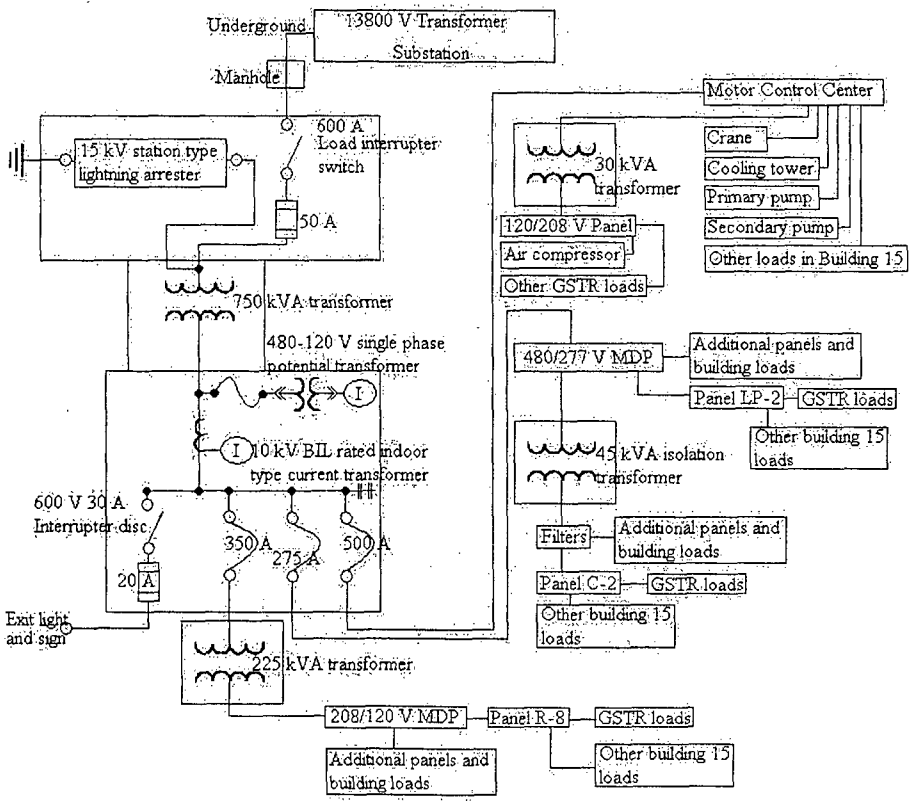


Figure 8.1: Electrical Power System

## 8.2 Emergency Electrical Power Systems

The emergency power systems consist of a 100/115-kW natural gas generator, a 5-kVA UPS, and associated panels and switchgear. These systems are provided by the GSTR staff and the General Services Administration (GSA), but are not required for safe operation of the reactor.

Safe reactor shutdown is ensured during a lost of normal operating power because gravity takes control and pulls the control rods back into the reactor core. Also, safe shutdown is ensured because the reactor bay normal exhaust, emergency exhaust, and RAMs continue to receive power when a loss of normal operating power occurs. In addition, each RAM is equipped with a self-contained battery back up, to ensure continuous monitoring of radiation in the reactor bay.

The natural gas generator supplies power to the facility in the event of a loss of normal operating power. The generator is designed to turn on automatically with a loss of normal operating power. The generator supplies power to RAM number one located in the reactor bay; select lighting; normal and emergency exhaust for the reactor facility; the building heating, cooling, and ventilation; communication systems within Building 15; as well as various other systems throughout Building 15. The fuel and maintenance for the generator are

provided by GSA.

The 5-kVA UPS supplies the control system console, RAMs four and six located in the reactor bay, the power measuring instruments, and the DAC cabinet. These loads normally receive conditioned power from the UPS and are unaffected by a loss of power to the building. The UPS is capable of carrying these loads on battery power for at least 26 minutes. If a prolonged loss of normal operating power occurs, the components the UPS powers can be plugged into sockets powered by the generator.

[redacted]

# Chapter 9

# AUXILIARY SYSTEMS

## 9.1 Heating, Ventilation, and Air Conditioning Systems

The controlled ventilation system acts to reduce the consequences of fission products released from the fuel or other experimental facilities. The objective of the structure surrounding the GSTR is to ensure that provisions are made to reduce the amount of radioactivity released into the environment by maintaining a negative pressure from the reactor bay to the reactor facility and a negative pressure from the reactor bay to the environment during operation. Automatic shutdown of the ventilation system confines the free air volume of the reactor bay during emergency conditions. Remote monitoring of the conditions within the reactor bay can be conducted.

### 9.1.1 Reactor Building Confinement

Building 15 superstructure consists of architectural cast stone panels on two inch concrete masonry units, fixed glass panels, dry wall (plasterboard), reinforced concrete walls, and eight inch concrete blocks. The building has a structural steel roof frame with a steel deck and insulating concrete fill. The floor of the first floor is composed of a lightweight aggregate concrete slab. The concrete slab is on top of well-compacted granular fill where there is no basement and on top of reinforced concrete where there is a basement. The reactor facility is surrounded by reinforced concrete walls, dry wall, and eight inch concrete blocks. The roof of the reactor bay and the radioisotope storage room is a one foot thick reinforced concrete slab. The reactor building has a dry gravel roof. The reactor bay floor is a six inch concrete slab placed on a well compacted granular fill. A bridge crane with a five ton capacity serves the reactor bay area.

The first floor of Building 15 contains the following:

1. The reactor bay, which accommodates the reactor, heat exchanger, pneumatic transfer system blower, reactor bay ventilation fans, and the fuel storage pits;

2. The reactor radioisotope lab, which houses fume hoods and radiation detection instruments;

3. The control room, which contains the reactor bay pressure measuring systems, reactor controls, and instrumentation;
4. The machine shop, which houses the stack monitoring equipment,
5. Five offices used by reactor staff, including the health physics office and reactor supervisor's office;
6. A radioisotope detection and counting lab;
7. The delayed-neutron lab, which houses the pneumatic transfer system and receiver-sender stations;
8. A conference room; and
9. Various other labs and offices not associated directly with the use of the reactor.

[redacted]

The basement contains the following:

1. The mechanical equipment room;
2. The transformer and electrical control room, which houses the backup power generator and the maintenance office;
3. General building ventilation and water controls;
4. A laboratory; and
5. [redacted]

[redacted]

The reactor secondary cooling system tower is adjacent to the southwest corner of the building. The cooling tower is housed within a walled area, access to which is controlled by a locked fence and a locked door.

### 9.1.2 Heating and Ventilation

The reactor bay ventilation system provides fresh air to the reactor bay area at a nominal rate of 1000 cfm and is independent of the GSTR and the laboratories located within the reactor building. A schematic of the ventilation system is shown in Figure 9.1. The incoming air travels through a heating coil, cooling coil, filter, supply fan, manual damper, automatic fire damper, and automatic damper to restrict release of airborne radioactive particles. The air then discharges into the reactor bay through two diffuser ducts near the ceiling. The exhaust air exits the reactor bay through one of two exhaust systems. The two exhaust systems are the main exhaust and the emergency exhaust system, used during normal operations and emergency situations, respectively. The main exhaust extends approximately 67.5 in above the roof of the building, which places the exhaust approximately 22.6 ft above the ground. The emergency exhaust system extends approximately

69.0 in above the roof of the building, which places the exhaust approximately 22.8 ft above the ground. The air is discharged at a nominal flow rate of 1000 cfm through the main exhaust system and at a nominal flow rate of 700 cfm through the emergency exhaust system. The high rate of flow from the exhaust ensures that the exhaust air carries to higher elevations and mixes rapidly with the surrounding air. The reactor bay ventilation systems can be switched manually from the control room. When the exhaust system is switched, the normal exhaust system stops, the emergency exhaust system begins, and the air supply to the reactor bay is sealed off. More detail on the operation of switching between normal and emergency exhaust can be found in Chapter 7 and Section 9.1.3.

The reactor bay is maintained at a negative pressure in relationship to the reactor facility and the outside static air pressure. The pressure differential was established by adjusting the manual damper for the airflow into the reactor bay. Once the negative pressure was established, the rate of flow of air into the reactor bay was matched with the rate of flow out of the reactor bay, in order to maintain the negative pressure. The negative pressure in the reactor bay ensures any airborne radioactive effluent release will be contained within the reactor bay and exhausted through the reactor exhaust. In order to ensure the pressure differential is always present, the reactor bay pressure is consistently monitored. When the reactor bay is no longer under negative pressure an alarm sounds on the control system console with an alert stating the reactor bay "under pressure" has been lost. If the reactor bay is not at a negative pressure of at least 0.1" of water, the reactor shall not be operated other than for exceptions as described in the Technical Specifications. In order to regain the under pressure status the operator can stop the incoming supply air, or the manual damper can be adjusted, and/or the emergency exhaust fan can be initiated with the normal exhaust.

Exhaust from the laboratories located in the reactor building is discharged above the roof of the building through individual fume hoods. The fume hood exhaust fans are controlled independently at each fume hood. Experiments with the possibility of hazardous effluents are to be conducted in a fume hood in order to limit the exposure of employees to radiation effluent and other chemicals.

### 9.1.3 Ventilation System Emergency Shutdown

The ventilation system is designed to provide for the automatic containment of airborne radioactive material. The reactor bay is always monitored by the continuous air monitor (CAM) and stack gas detector. The CAM continuously monitors the air within the reactor bay and the stack gas detector constantly monitors the outgoing reactor bay effluent. In addition, the stack effluent air is analyzed quarterly to determine the isotopic composition of radionuclides emitted. The CAM is calibrated annually and the stack monitoring instrumentation is calibrated semi-annually. Radioactive gases normally produced are H–3, N–16 and $^{41}$Ar. However, due to transport delays the amount of N–16 (formed from 0–16 in the pool water) released is insignificant. Most of the $^{41}$Ar produced during routine operation comes from the experimental facilities. The routine release of $^{3}$H from primary water evaporation is well below the limits of 10 CFR 20.

When the CAM registers a high alarm of 10 kcpm, it signals the reactor bay supply dampers to shut, applies power to the emergency exhaust, removes power from the pneumatic system exhaust, and removes power from the normal exhaust within 1 to 15 seconds after the alarm which results in the shutting of the normal exhaust damper. The reason for the closing of the supply dampers and the normal exhaust system is to

maintain a negative pressure differential between the reactor bay and the adjoining rooms. The signal path from the CAM is shown as a dashed line in Figure 9.1. More detail on the operation of the switching over between the emergency and normal exhaust systems can be found in Chapter 7. An operational check of the CAM is performed every operating day before operation of the reactor. A manual reset by the operator is required to restart the system if the ventilation system is in the emergency exhaust mode for any reason.

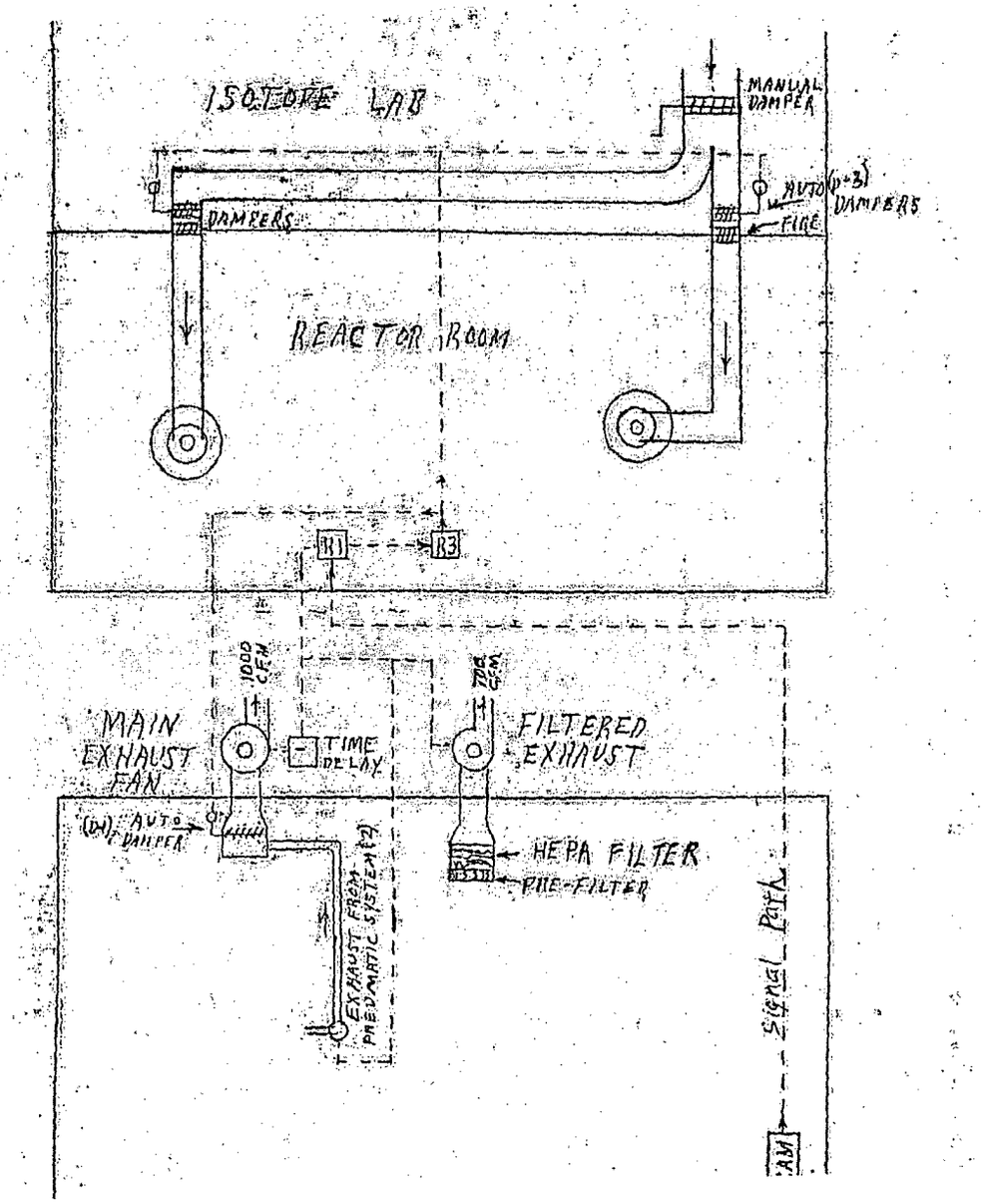


Figure 9.1: GSTR Ventilation System

### 9.1.4 Ventilation system malfunctions

When the ventilation system switches from the normal exhaust to the emergency exhaust a malfunction can occur at the CAM or at the reactor bay supply dampers. Should the CAM alarm and not actuate the ventilation transition, the process to switch from the normal exhaust mode to the emergency mode can be manually actuated by the Reactor Operator. Also, the CAM is only one of several instruments monitoring the reactor bay; there is a neutron detector and several gamma-sensitive ion detectors monitoring the reactor bay as well. If the CAM did not alarm and another instrument did, the Reactor Operator would be able to manually actuate the process and switch the exhaust systems. The ventilation system is checked every day reactor operation occurs unless operation is continuous from one day to the next and then it is checked prior to the start of the extended operation. Radiological effects of malfunctions is discussed in Chapter 11.

## 9.2 Handling and Storage of Reactor Fuel

The core loading for the GSTR consists of a transient control rod and approximately [redacted] elements, including three fuel-followed control rods and instrumented elements. [redacted].

Fuel handling consists of the movement of any fuel element or fuel follower control rod, whether it is irradiated or unirradiated.

### 9.2.1 Fuel Storage Racks

[redacted]

### 9.2.2 Fuel Handling Tool

Tools are provided for handling individual fuel elements and for manipulating other core components. Individual fuel elements are handled with tools designed by General Atomics®. The tool utilizes a locking ball-detent grapple that attaches to the top end fitting of a fuel element. [redacted]

### 9.2.3 Long Handled Socket Wrench, Hook, and Adapter Removal Tool

The long handled socket wrench, hook, and adapter removal tool are tools that may be used in the removal of foreign objects, experiments, and irradiation facilities. The three tools all use a common extension handle. The extension handle and the tool adapters have water inlet holes to allow the tools to fill with water to shield from radiation streaming. The hook and adapter removal tool are the same attachment; however, to use the hook tool versus the adapter removal tool an additional hook attachment is required. The socket wrench tool is a separate attachment. All three tools are stored in the reactor bay when not in use.

### 9.2.4 Fuel Element Transport Cask

The fuel element transport cask is used when transporting irradiatated fuel elements and for removing fuel elements from the reactor tank. The fuel element transport cask is a steel casing filled with lead, weighing 5700 lbs. Movement of the cask is performed by utilizing the crane. The interior of the cask is designed to hold one fuel element and it is designed to allow the fuel handling tool to travel through the cask. Access to the fuel element is through two lead plugs located on the top and bottom of the cask. The cask is painted to prevent corrosion. This allows it to be inserted into the reactor tank for usage without undue corrosion.

### 9.2.5 Crane

A crane with a maximum capacity of 5 tons (10,000 lbs) has been installed in the reactor bay. The crane is used to secure heavy items when maintenance, inspection, or relocation is being performed. The crane is only energized when its use is required and after it has been used it is de-energized to protect reactor staff from accidental electrocution. For safety reasons, usage of the crane is only permitted when two reactor staff members are on site.

### 9.2.6 Fuel Element Inspection Tool

The fuel element inspection tool is used to accurately inspect fuel elements for longitudinal growth and for bowing. The upper support plate of the inspection tool is secured to the top of the reactor tank. The inspection tool extends downward into the tank permitting the inspection of an irradiated fuel element while maintaining water shielding over the element. The tool is fabricated from aluminum and stainless steel. The aluminum support tube is open at the bottom and top to allow water to fill the interior of the pipe for radiation shielding.

Bowing of a fuel element is detected by a carefully machined cylinder (called a go/no-go gauge) attached to the bottom of the tool. If the fuel element will not slide completely into the machined cylinder, the fuel element exceeds the specified limits for bowing of 1/16 inch over the clad section of the element. When the element passes through the cylinder, it will come to rest on the lower plunger of a hydraulic bellows assembly. The length of the fuel element is measured by screwing an indexing rod into the top plate, until tight. While screwing in the indexing rod, the indexing rod rotates over the fuel element's top end fixture and pushes down on the fuel element. Pushing the fuel element downwards displaces the lower plunger of the hydraulic

bellows assembly, which in turn displaces the upper plunger located near the surface of the water on the fuel element inspection support plate. The displacement of the upper plunger is measured with a micrometer and the measured value is compared to the measurement of the aluminum standard dummy element. The difference between the fuel element measurement and standard element is recorded and compared to the previous measurements to determine if the fuel element is within the specified limits. The specified limit for elongation of the aluminum fuel elements is 0.5 inch and the limit for the stainless steel fuel elements is 0.1 inch.

## 9.3 Fire Protection System

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. Building 15 maintains compliance with local fire codes and passed the last fire inspection, the results of which are in Appendix 9–A. The building is equipped with an active fire protection system of sprinklers and pull-stations, except in the reactor bay. In addition, throughout the building there are several audible horns and strobe lights that activate when a fire alarm is activated. A number of fire extinguishers are positioned in the reactor bay, adjacent rooms, hallways, and laboratories. The fire extinguishers and detection system are regularly inspected by a contractor employed by GSA, the company responsible for the care and upkeep of government buildings. To maintain a level of safety within the facility, combustible chemicals are stored within a fire proof cabinet that is not in the reactor bay. If a fire does occur within the facility, the action taken is documented in the facility emergency plan.

Consequences of a fire could result in a release of radioactive materials or a malfunction of safety equipment controlling the reactor. Any radioactive material released into the reactor bay will be detected by the CAM and/or the RAMs. The CAM will switch the exhaust from the normal exhaust to the emergency exhaust when it measures 10 kcpm. Also, the CAM will alert the reactor staff when it measures 3000 cpm. The consequence of a fire affecting a reactor safety system is not a direct concern because the scramming of the control rods only requires the force of gravity. Any fire affecting the controls of the reactor cannot prevent the control rods from inserting upon the initiation of a scram. Also, if the reactor is shut down the fire cannot cause the removal of control rods. Thus the reactor will stay subcritical.

## 9.4 Communications

The reactor control room is equipped with a multi-line commercial telephone. An intercom system is available at the control console. It can be used for communication between the reactor bay and the control console or for communication between the control console and other offices and labs within Building 15. The system provides two-way communications. Also, there is an evacuation alarm within the reactor bay that can be initiated from the control systems console.

## 9.5 Possession and Use of Byproduct, Source and Special Nuclear Material

All activities using source and special nuclear material covered under the reactor license take place within the reactor facility. Material covered under the reactor license is located in the reactor bay, the control room, the storage tank in Room 151 or Room 150 of the facility. All activities using source and byproduct material covered under the radioisotope utilization permit (RUP) may take place within all facility spaces in Building 15. Material covered under the RUP may be located in various rooms within Building 15 and Building 10 for the purpose of analyzing samples with instrumentation otherwise unavailable in the reactor facility (e. g., gamma spectroscopy, liquid scintillation counting, gross alpha/beta counting), instrument calibrations, or performing experiments. Byproduct, source, and special nuclear material (SNM) use other than identified above is covered under Broad Scope License 05-1399-08.

The reactor license authorizes the facility to have special nuclear materials; a sealed americium-beryllium source; a sealed polonium-beryllium source; any byproduct material produced as a result of reactor; and the license authorizes the USGS to possess, use and operate the reactor as a utilization facility.

Building 15 is equipped with several RAMs to monitor and detect radiation. The RAMs are connected to the building evacuation alarm. Usage of radioactive materials within Building 15 is confined to authorized locations. All radioactive material is secured from unauthorized access.

## 9.6 Cover Gas Control in Closed Primary Coolant Systems

This section is not applicable because no such system exists at the GSTR.

## 9.7 Other Auxiliary Systems

This section is not applicable because there are no other auxiliary systems.

# APPENDIX 9–A

***GENERAL SERVICES ADMINISTRATION***
***FIRE & SAFETY MANAGEMENT***
***FACILITY ASSESSMENT SURVEY***

***REGION 8***

***FACILITY INSPECTED:***
**Building #15**
**Denver Federal Center**
**Lakewood, CO**
**(CO0625AA)**

**Facility Surveyed and Inspected By:**
**Alan J. Antonio, C.F.P.S.**
**Fire Protection Technical Engineer**
**U.S. General Services Administration**
**Rocky Mountain Region**

***FIRESAFETY EQUIVALENCY EVALUATION***

| *Provided - Required = Equivalency* | Provided | Required | Equivalency (Y/N) |
|---|---|---|---|
| Control Requirement | 12.5 | -1 | 13.5(Y) |
| Egress Requirement | 14 | 0 | 14(Y) |
| General Fire Safety | 21 | -1 | 22(Y) |

***SPRINKLER COVERAGE*: Complete**
***BUILDING HEIGHT*: 12 Feet**

***SURVEY DATE: January 7, 2005***

***REPORT DATE: January 28, 2005***

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

## TABLE OF CONTENTS

## EXECUTIVE SUMMARY

The purpose of this survey is to identify, through evaluation of administrative records, site inspection, testing of fire protection equipment and recognized methods of measurement, sampling, and monitoring, potential fire safety and/or environmental hazards that are or could be encountered in the facility. This report explains what is required to eliminate or reduce the risk of injury/illness and/or property damage or loss by means of engineering controls or administrative controls. This survey is not a code audit, but rather a professional assessment to determine the potential risks associated with the use of this building.

There are eighteen recommendations made as a result of this survey. The sprinkler system is being maintained and the main drain was tested.
A life safety analysis from a fire in the building was performed using NFPA 101A and is included at the end of this report. That analysis indicates that the building **passes** the evaluations for fire control, egress and general fire Safety.

## GSA FIRE SAFETY SURVEY REPORT

## INTRODUCTION

Building #15 is located adjacent to intersection of Center Avenue and Second Street in the Denver Federal Center. The building was constructed in the 1940's. The building is used for offices and laboratories by the USGS. The survey was completed on January 28, 2005. The survey was conducted by Alan J. Antonio C.F.P.S., Fire Protection Technical Engineer, U.S. General Service Administration, Rocky Mountain Region.

There was a record of a previous SEM survey being performed in 1996. At that time, there were nineteen recommendations and all have been closed in the "Safe" system by either corrective action or no action being required as indicated by the Regional Office.

## BUILDING PROFILE

### Fire Safety Features and Building Description

This one story building has one basement level. This building houses the nuclear reactor. There is approximately 29,540 SF of space in the building. The occupancy type is business as offices and labs are located throughout. The basement houses the mechanical equipment for the building. The building construction is Type II-000. The exterior walls consist of concrete masonry construction. The floor construction consists of concrete. The roof construction consists of a metal deck on steel bar joists. The interior partitions consist of plaster or gypsum board. The floor finish is vinyl tile and carpet. The ceiling finish is spline type acoustical tile.

Exiting from the first floor is provided via numerous exits that discharge directly to the outside of the building. The exits are 100% remote as one is located on each end of the buildings main two corridors. Exit signs throughout the building are the electrically illuminated type. No emergency lighting units have been provided on the first floor area but exterior windows are located throughout. There are no concerns in reference to exiting from this one story building.

The building has been equipped with an addressable fire alarm system. The Silent Knight Addressable FACP is located in the basement mechanical room. The FACP sends an alarm signal to the Denver Mega Control Center in Building #53. Devices monitored by the fire alarm system include the water flow switches for the basement and 1st floor wet pipe sprinkler systems, the pressure switch for the dry pipe sprinkler system, duct detectors and the manual pull stations adjacent to each exit door. The supervision for the wiring of the initiating device circuits is Style D and for the notification appliance circuits Style Z as indicated by the four wire circuits. The FACP is equipped with a battery and charger for secondary power. There are electric horns located throughout for notification. Flashing lights are located in the mechanical room. No audibility deficiencies were noted during testing of the FACP and its devices. The fire alarm system is being maintained by operations and maintenance contractor.

A wet pipe sprinkler system has been provided throughout the basement and first floor of the building and the alarm valves are located in basement mechanical room. The wet pipe systems are equipped with 4" inch alarm valves and associated trim. The hydraulic design indicated on the sprinkler risers is .15gpm/sf for an area of 3000 sf. The dry pipe system protects the electrical room and the elevator machine room in the basement and serves only 6 heads. During the evening testing, the static pressure was 110 psi. The sprinkler systems pressure switch is not being tested quarterly and the main drain is not being operated as required. The main drain test could not be completed as water began to flood back into the room. The sprinkler head spacing was noted light hazard in the office and ordinary in the remaining areas. The fire department connection was not obstructed and equipped with caps. A six inch water main extends to the building for the sprinkler systems supply. This six inch is connected to a six inch which connects to the 10 inch base loop in the vicinity of Second Street and South Center Avenue.

Water flow tests were conducted in January 20, 2005 in front of Building #15. Hydrant FH-15-1 was used as the gauge hydrant and the flow hydrant was FH-16-1. Static pressure = 104 psi; Residual pressure = 96 psi; Pitot pressure = 42 psi; Flow = 1087 gpm.

There are multi-purpose fire extinguishers located throughout the building. The fire extinguishers were new in 1979 or 1989 and six-year service was last performed in 1989. The extinguishers are being serviced and maintained as required by NFPA 10. A recommendation has been made at the end of this report.

Heating for the building is provided by gas fired boilers. The equipment including the air handlers is located in the basement mechanical room. Duct detectors have been provided in the supply ducts and shut down the affected air handling equipment upon activation. The fresh air intakes are located on the East side of the building. There are no concerns in reference to the heating and cooling systems for this building.

There are no significant structures exposing the subject building. West Metro Fire District provides fire fighting suppression for the Federal Center and the closest station is approximately 1/2 mile from the entrance gate of the Denver Federal Center.

A life safety analysis from a fire in the building was performed using NFPA 101A and is included at the end of this report. The analysis indicates that the building passes the evaluation for fire control, egress and general fire safety.

**Record Review**
The preventive maintenance records were reviewed for the last three years. All records are submitted on a monthly basis to GSA by the operations and maintenance contractor.

**Visual Inspection of Equipment/Systems**

There were no concerns noted with the visual inspection of equipment.

**Existing/Potential Hazards**

Several existing and potential hazards were noted during this inspection.

**Analysis of Risk**

Eighteen *Findings and Recommendations* were noted and indicated at the end of this report.

## BUILDING PROFILE

**DATE:** January 29, 2005

**SURVEYOR:** Alan J. Antonio, C.F.P.S., Fire Protection Technical Engineer

## GENERAL BUILDING INFORMATION

**BUILDING NAME:** Building #15
**ADDRESS:** Denver Federal Center
Lakewood, CO

**BUILDING NUMBER:** CO0625AA

**STORIES BELOW GRADE:** 1

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**STORIES ABOVE GRADE:** 1

**AREA/FLOOR (SQ. FT.):** 25,000 SF - 1st

**TOTAL AREA (SQ. FT.):** 29,540 SF

**FLOORS OCCUPIED BY THE GOVERNMENT:** 1

**HEIGHT (FROM FIRE DEPARTMENT ACCESS TO HIGHEST GOVERNMENT FLOOR LEVEL):** 12'

**HEIGHT (FROM FIRE DEPARTMENT ACCESS TO ROOF OF BUILDING):** 12'

**OCCUPANCY TYPE ON GOVERNMENT FLOORS:**
Labs: 1
Offices: 1
Mechanical Equipment: B

**OCCUPANCY TYPE ON NON-GOVERNMENT FLOORS:** N/A

**POPULATION OF FEDERAL OCCUPANTS:** 15

**HAZARDS (SIGNIFICANT FUEL LOAD AREAS):** None

## CONSTRUCTION

**TYPE OF CONSTRUCTION:** Type II-000

**DESCRIPTION OF CONSTRUCTION:** Exterior walls are concrete masonry

**TYPE OF FLOOR CONSTRUCTION:** Reinforced concrete

**TYPE OF ROOF CONSTRUCTION:** Metal deck on steel bar joists

**FIRE SEPARATIONS:** None

**CORRIDORS:** Non-rated corridors with 45 minute fire rated doors

**FIRE WALLS:** None

**VERTICAL OPENINGS/ENCLOSURES:** Concrete block walls

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**EXIT STAIRWAY ENCLOSURES:** Stair to basement is enclosed with concrete block walls with the doors having no indicated fire rating but consisting of substantial components

## FIRE EXPOSURE

No significant exposures affecting the subject building

## EGRESS

**STAIRS SERVING ABOVE GRADE FLOORS:** N/A

**STAIRS SERVING BELOW GRADE LEVELS:** None used for egress purposes for the occupied areas as only mechanical space is located in basement

**HANDRAILS:** N/A

**CORRIDORS:** 80"w X 96"h

**RAMPS:** None

**EXIT DISCHARGE:** Exit directly to the outside

**CALCULATED OCCUPANT LOAD:** 290

**EXIT CAPACITY:** 720

**EXIT REMOTENESS:** Exits are 100% remote from each other and located off of the two main corridors in the building

**MAXIMUM TRAVEL DISTANCE:** 50 - 100'

**COMMON PATH OF TRAVEL:** None

**DEAD END:** None

**EXIT TIMED CALCULATIONS:** The building should be able to be evacuated within one minute according to the calculations

**EXIT SIGNAGE:**
Visibility: Electrically illuminated
Secondary Power: None

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**EMERGENCY LIGHTING:** None

## INTERIOR FINISH

**DESCRIPTION:**

Ceiling Finish: Spline type acoustical tile

Wall Finish: Gypsum board; Plaster

Floor Finish: Vinyl tile; Carpet

## FIRE DETECTION AND ALARM

**FIRE ALARM SYSTEM:** Yes

**CONNECTION TO FIRE DEPARTMENT:** Yes
Name: Lakewood F.D.

**MANUFACTURER (TYPE/MODEL):** Silent Knight Addressable FACP

**NO. OF INITIATING ZONES:** 10

**ANNUNCIATOR:** No

**FACP:** Located in the basement mechanical room

**Wiring:**
Supervision of Signal Line Circuits: Style 4
Supervision of Notification Appliance Circuits: Style Z

**AUDIBILITY PROBLEMS:** None noted

**SECONDARY POWER SOURCE:** Battery and charger

**FIRE COMMAND STATION:** None

**INITIATING DEVICES:**

**MANUAL PULL STATION:** Adjacent to each exit door

**WATERFLOW SWITCHES:** Basement mech. room

**DUCT DETECTORS:** In supply air ducting

**PRESSURE SWITCHES:** Serves basement elevator and electrical room

**SMOKE DETECTORS:** A Silent Knight Remote panel is located near the Southwest exit door and monitors area smoke detectors throughout. This panel has one output that is connected to the Silent Knight Panel.

**DEVICES TESTED:**

Manual pull stations: Throughout 1st floor
Water flow switch: Basement system
Air handler shutdown: Basement

**INDICATING DEVICES:**

**HORNS:**

Manufacturer: Kidde
Location: Throughout

## FIRE SUPPRESSION

**AUTOMATIC SPRINKLERS:** Yes

**TYPES:** Wet and dry pipe systems
Location: Throughout building
Spacing: Ordinary/light
Hydraulics: .15gpm/sf for 3000 sf
Fire Department Connection: South side of building
Tamper Switches: Connected to FACP

**SPRINKLERS OMITTED:**

Location: None noted

**CITY WATER MAIN:**

**MAIN LOCATION:**

Size: 6" interconnected to the 10" base loop in the vicinity of South Center Avenue and Second Street

**SIZE INTO BUILDING FOR FIRE PROTECTION (IN.):** 1- 6"

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**STANDPIPES:** None

**FIRE EXTINGUISHER: (See Recommendation)**
Location: Corridor
Type: Multi-purpose dry chemical
Annual Service Date: None
Six-year Maintenance: 1989
Hydrostatic Test Date: New in 1979; 1988

**NEAREST FIRE HYDRANT:**
Location: 40 feet East of subject building

**WATER SUPPLY (GPM):** 1087

**STATIC PRESSURE:** 104 psi

**RESIDUAL PRESSURE:** 96 psi

**NEAREST FIRE DEPARTMENT:**
Location: West Metro Fire Department
Approximate distance from building: 1/2 mile from entrance gate

**FIRE PUMP:** None

**OTHER SUPPRESSION SYSTEMS:** None

## SMOKE CONTROL

**SMOKE CONTROL:** No

## HVAC

**AIR HANDLER UNITS:**
Location: Basement mechanical room
Zones: Entire building

**FRESH AIR INTAKES:** East side of building at ground level

**FIRE DAMPERS:** None

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**DUCT DETECTION:** Yes

**FAN SHUT DOWN:** Yes

## ELEVATORS

**NUMBER OF ELEVATORS:** One hydraulic elevator having no special recall or fire features as none are required

## EMERGENCY POWER

**EMERGENCY GENERATOR:** None

## MAINTENANCE OF EQUIPMENT/SYSTEMS

**GENERAL CONDITIONS OF EQUIPMENT/SYSTEMS:** Satisfactory- Sprinklers or fire extinguishers

**PREVENTIVE MAINTENANCE RECORDS:** Satisfactory

## EMERGENCY PLAN

**DATE OF PLAN:** Posted and located at the exist and hallways

## SPECIAL USE AREAS

**CHILD CARE CENTER:**
Location: None present
Fire protection features: N/A

**BATTERY CHARGING AREAS:**
Location: None present
Ventilation: N/A
Eye wash protection: N/A

**MOTOR POOL SERVICE:**
Location: None present

Type: N/A

**RADIATION SERVICE AREAS:**
Location: This building houses a nuclear reactor. That portion of the building was not inspected.

**FIRING RANGES:**
Location: None present.
Cleaning Procedures: N/A

**PHOTO PROCESSING:**
Location: None present
Type: N/A

**INCINERATORS:**
Location: None present

**PRINTING PLANTS:**
Location: None present
Type: N/A

**SPRAY PAINTING OPERATIONS:**
Location: None present
Type: N/A

**HAZARDOUS AREAS:**
Location: Room 120
Protective Equipment: Acids and corrosives are used in this laboratory which is not equipped with and emergency eye wash station. Access to the emergency shower was blocked.

**STORAGE AREAS:**
Location: None
Description: N/A

**COMPUTER ROOMS:**
Location: None
Features: N/A

**DETENTION AREAS:**
Location: None
Features: N/A

## CONTAMINANT EXPOSURES

**EXPOSURES:** There is the potential for exhaust from the boiler to be entrained into the building via the outside air intake which is located at ground level on the East side of the building.

**AIR DAMPERS:**
Location: NA

Description: NA

## STORAGE TANKS

**LOCATION/TYPE:**
Location: None
Type: N/A
**SUBSTANCE STORED:**
Type: N/A

**LEAK DETECTION TESTS:**
Results of tests: N/A

**PROCEDURES FOR AFTER-HOURS:**
Description of procedures: N/A

**FIRE DEPARTMENT PRE-FIRE PLAN:** N/A

## POLYCHLORINATED BIPHENYLS

**LOCATION:** There are no PCB containing transformers in the building.

**LABELED:** N/A

**SPILLED CONTROLS:** N/A

## HAZARDOUS WASTES

**LOCATION:** Small quantities in various labs; other quantities sent to Building 25.

**STORAGE COMPATIBILITY:** OK

**CONTINGENCY PLAN:** USGS Spill Response Team

**WASTE DISPOSAL:** Acids and bases are neutralized to pH 6.0-8.5 and flushed with water to the domestic sewer

**MATERIAL SAFETY DATA SHEETS:** Yes

## LABORATORIES

**LOCATION/PURPOSE:** Several labs located on East side of building

**CHEMICAL HYGIENE PLAN:** Yes

**HOODS/EXHAUST:** Yes

**MATERIAL SAFETY DATA SHEETS:** Yes

**WASTE DISPOSAL:** See above.

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

## *NFPA 101A*
## *Fire Safety Evaluation Worksheet for Business Occupancies*

### Table 7-1 Safety Parameters

| Safety Parameters | Parameter Values | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 1. Construction | Noncombustible | | | Combustible | | | | |
| NFPA 220 Bldg Constr. Types | Type I (443) or (332) Type II (222) | Type II (111) | Type II (000) | Type III (211) | (200) | Type IV (2HH) | Type V (111) | (000) |
| 1 story | 0 | →0 | 0 | 0 | -1 | 0 | 0 | -1 |
| 2 stories | 2 | 2 | -4 | 0 | -2 | 0 | 0 | -4 |
| 3 stories | 2 | 2 | -6 | 0 | -6 | 0 | 0 | -12 |
| 4-5 stories but ≤ 75' | 2 | 2 | -10 | 0 | -12 | 0 | -3 | -12 |
| > 5 stories but ≤ 75' | 2 | 2 | NV | 0 | NV | 0 | -6 | NV |
| >75' but ≤ 150' | 2 | -1 | NV | 0 | NV | 0 | NV | NV |
| ≤ 150' | 2 | NV | NV | 0 | NV | 0 | NV | NV |
| 2. Segregation of hazards | Exposed exit system | | Segregation from exit routes | | | | None or no deficiencies | |
| | Double Def. | Single Def. | Double Def. | | Single Def. | | | |
| | -7 | -4 | -4 | | 0 | | →0 | |
| 3. Vertical openings[A] | Open (or incomplete enclosure) | | | | Enclosed | | | |
| | Connects 5 or more floors | 4 flrs. | 3 flrs. | 2 flrs. | <30 min. | 30 min. to 1 hr | > 1 hr[G] | |
| | -10 | -7 | -4 | -2 | -1 | →0 | 1 | |
| 4. Sprinklers | None | Corridors only | All but corridors and lobbies | | | | Total building | |
| | 0 | 0 | 4 | | 6 | →10 | 12 | |
| 5. Fire alarm system | None | w/o fire department notification | | | w/ fire department notification | | | |
| | | w/o Voice Comm. | w/ Voice Comm. | | w/o Voice Comm. | | w/ Voice Comm. | |
| | 0(-2)[L] | 1(0)[L](-1)[P] | →2(0)[P] | | 2(1)[L](-1)[P] | | 4(2)[P] | |
| 6. Smoke detection | None | Corridor | Rooms | | Total bldg. (zone) | | | |
| | 0 | →1 | 2 | | 4 | | | |
| 7. Interior finish exit routes rooms suites | Flame spread ratings[B] | | | | | | | |
| | >75 to ≤200 | | >25 to ≤75 | | ≤25 | | | |
| | >75 to ≤200 | ≤75 | >75 to ≤200 | ≤75 | >25 to ≤200 | ≤25 | | |
| | -3 | -1 | 0 | 1 | 1 | →2 | | |
| 8. Smoke control | None | | Passive | | Active | | | |
| | 0 | | 3 | | 4(3)[K] | | | |
| 9. Exit access | Max. dead ends | | No dead end > 50 ft. & travel is: | | | | | |
| | >75' to ≤100' | >50' (20')[H] ≤75' | >200'[C] to <400' | >100'- 200'[C] | | >50' - 100' | ≤50' | |
| | -2[D] | -1 | -1 | 0 | | →1 | 3 | |
| 10. Exit system | Single | Multiple routes | | | | | | |
| | | Deficient | Not Deficient | Smokeproof enclosures | | Direct exits | | |
| | -6(0)[J] | -2 | 0 | 3 | | →5 | | |
| 11. Corridor/room separation (compartmentation) | Separation exists and level of protection is: | | | | | | | No separation |
| | Incomplete | Smoke Resistive[E] | | >20 min. | | ≥1 hr[E] | | |
| | | w/o door closer | w/ door closer | w/o door closer | w/door closer | w/ door closer | | |
| | -6 to 0[M] | →0 | 1(2)[F] | 1 | 2(3)[F] | 3(4)[F] | | 0 |
| 12. Occupant emergency program | Number of fire drills conducted per year | | | | | | | |
| | 0 | | 1 to 2 | | >2 | | | |
| | -2(-3)[N] | | →0(1)[O] | | 1(2)[O] | | | |

## Table 7-2 Individual Safety Evaluation (Revised)

| SAFETY PARAMETER | FIRE CONTROL ($S_1$) | EGRESS PROVIDED ($S_2$) | GENERAL FIRESAFETY PROVIDED ($S_3$) |
|---|---|---|---|
| 1. CONSTRUCTION | 0 | | 0 |
| 2. HAZARDOUS AREAS | 0 | 0 | 0 |
| 3. VERTICAL OPENING | 0 | 0 | 0 |
| 4. SPRINKLERS | 10 | 5 | 10 |
| 5. MANUAL FIRE ALARM SYSTEMS | 1 | 2 | 2 |
| 6. SMOKE DETECTION | .5 | 1 | 1 |
| 7. INTERIOR FINISH | 1 | | 2 |
| 8. SMOKE CONTROL | | 0 | 0 |
| 9. EXIT ACCESS | | 1 | 1 |
| 10. EXIT SYSTEM | | 5 | 5 |
| 11. CORRIDOR SEPARATION | 0 | 0 | 0 |
| 12. OCCUPANT EMERGENCY PROGRAM | | 0 | 0 |
| TOTAL | $S_1 = 12.5$ | $S_2 = 14$ | $S_3 = 21$ |

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

**Table 7-3 Mandatory Requirements (Revised)**

| Building Height | Control Requirement ($S_a$) | | Egress Requirement ($S_b$) | | General Fire Safety ($S_c$) | |
|---|---|---|---|---|---|---|
| | New | Existing | New | Existing | New | Existing |
| **1 story** | **0.5** | **-1.0** | **1.5** | **0** | **2** | **-1** |
| 2 stories | -2.5 | -4.0 | 1.5 | 0 | -1 | -4 |
| 3 stories | 1.5 | 0 | 1.5 | 0 | 3 | 0 |
| > 3 stories and ≤75 ft | 4.0 | 2.0 | 2.5 | 0 | 6 | 2 |
| >75 ft but <150 ft | 9.5 | 7.5 | 7.5 | 5 | 10 | 6 |
| ≥ 150 ft | 12.5 | 10.5 | 7.5 | 5 | 13 | 9 |

**Table 7-4 Equivalency Evaluation**

| | | | | | Yes | No |
|---|---|---|---|---|---|---|
| Control Provided ($S_1$) | minus | Required Control ($S_a$) | ≥ 0 | $S_1 - S_a$ 12.5 - (-1) = 13.5 | ✓ | |
| Egress Provided ($S_2$) | minus | Required Egress ($S_b$) | ≥ 0 | $S_2 - S_b$ 14 - 0 = 14 | ✓ | |
| General Provided ($S_3$) | minus | Required General Firesafety ($S_c$) | ≥ 0 | $S_3 - S_c$ 21 - (-1) = 22 | ✓ | |

*GSA Fire & Safety Survey, CO0625AA –January 29, 2005*

## Table 7-5 Facility Firesafety Requirements Worksheet

| Considerations | Met | Not Met | Not Applic. |
|---|---|---|---|
| A. Building utilities conform to the requirements of Section 7-1. | ✓ | | |
| B. The air conditioning, heating, and ventilating systems conform to Section 7-2, except for enclosure of vertical openings, which have been considered in Safety Parameter 3 of Table 7-1. | ✓ | | |
| C. Elevator installations are made in accordance with the requirements of Section 7-4. | | | ✓ |
| D. Rubbish chutes, incinerators, and laundry chutes are installed in accordance with Section 7-5. | | | ✓ |
| E. Portable fire extinguishers are installed and maintained in accordance with the requirements of 26-3.5/27-3.5 and 7-7.4.1. | | ✓ | |
| F. Standpipes are provided in all new high rise buildings as required by 26-4.2. | | | ✓ |

# References

[1] Memorandum, Fabian C. Foushee to Distribution, "Storage of TRIGA® Fuel Elements," General Dynamics, General Atomic Division, March 1, 1966.

# Chapter 10

# EXPERIMENTAL FACILITIES AND UTILIZATION

## 10.1 Summary Description

The GSTR provides neutron and gamma irradiation facilities for use by U.S. Geological Survey researchers and other public and institutional users. All systems are designed and operated to control the amount of radiation exposure received by the general public, as well as facility personnel. Incidental production of $^{41}Ar$ is also mitigated in order to minimize both the environmental release amounts and the exposure of personnel within the reactor facility.

Specimens may be exposed to neutron and gamma radiation by placing them in the reactor pool adjacent to the core by using suitable watertight containers. If very high level irradiations are required, specimens can be inserted directly into the highest flux region of the core in the central thimble. Very high level irradiations can also be performed by relocating the six B-ring fuel elements to outer grid positions and removing the central thimble and the 4.4 inch diameter hexagonal section in the top grid plate. Two other facilities, which can accept samples up to 2.4 inches in diameter, are available by removing three fuel elements, one from the D-ring and two from the E-ring. Or, smaller diameter (less than 1-1/2 inch) specimens can be inserted in special tubes or containers by removing a fuel element from the desired grid location.

The following experimental facilities are provided at the GSTR:

- The Central Thimble;
- Rotary Specimen Rack Assembly;
- Vertical Irradiation Tubes;
- The Pneumatic Transfer System; and
- 8-inch Beam Tube.

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 Section 10.3 prior to performance.

## 10.2 Experimental Facilities

### 10.2.1 The Central Thimble

The central thimble, located in the center of the core, provides space for the irradiation of small samples at the point of maximum flux. It also makes possible the extraction of a highly collimated beam of neutron and gamma radiation.

The thimble is an aluminum tube 1-1/2 inches in outside diameter with a wall thickness of 0.083 inches. It extends from the bridge straight down through the central hole of the removable hexagonal section in the top grid plate then through the lower grid plate, terminating with a plug in the lower end. The central thimble is supported at its lower end by the safety plate situated about 16 inches beneath the lower grid plate. The central thimble is normally water-filled, but may be air-filled for certain vertical beam applications.

The central thimble is made in three sections to facilitate its removal from the reactor. The lower section is approximately 12 feet long, the middle section is about 4-1/2 feet long, and the upper section is about 12 feet. The lower and middle sections are welded together. To join the middle and top sections and provide proper alignment, a 26-inch length of 1-7/8 inch outside diameter tubing is slipped over and welded to the top 3 inches of the middle section. The top 3 inches of the 1-7/8 inch tubing is chamfered and machined to accept the lower end of the top section of the thimble tubing. A Buna-N O-ring recessed into the 1-7/8 inch tubing provides a leakage restrictor. A small plate with a 1/2 inch hole for the attachment of a handling tool is welded to and extends radially from the top of the 1-7/8 inch tubing.

Removable rings 1-5/8 inch in diameter and 3/8 inch thick are located on the lower section of the central thimble tubing just above and below the hexagonal grid section. The rings, which are each fastened to the tubing with set screws, support the hexagonal section and ensure its proper vertical placement in the top grid plate.

The top end of the central thimble is fastened by a holding fixture, threaded into the center channel mounting plate. The holding fixture (a modified Weatherhead union) permits the tube to slide freely through it when loosened, but can be tightened to hold the tube securely in the fixture.

Central thimble samples are normally sealed inside aluminum containers because of the high neutron fluence exposure. Plastic containers should be used with caution and only for low fluence experiments. Since the central thimble is water filled, the sample containers are normally tested for leakage prior to being loaded into the reactor. The sample containers typically have positive buoyancy and require either weights or a solid rod to hold them at the core position. Attachments to the sample container should be aluminum in the core vicinity to prevent adverse effects of the radiation exposure.

Vertical placement of the sample container is controlled by measuring and adjusting the distance from the container to the top of the central thimble tube. When aluminum rods are used to suspend the sample, care

is used in tightening the rod sections to ensure that they will not come loose during the irradiation. Long term irradiations include wiring of the rod sections to prevent their disassembly.

Samples suspected to have high reactivity values are inserted and removed only when the reactor is shutdown, and the reactivity effect is measured immediately after the sample is first inserted. If the sample is worth $1.00 or more, the experiment cannot be performed unless it is an approved Class II experiment, or similar to an existing Class II experiment and the sample is firmly fixed in position. Loading of samples requires the appropriate entry in the operations logbook.

Samples irradiated in the central thimble are usually highly radioactive following irradiation and Health Physics monitoring is required. If radiation levels are excessive during sample unloading, the sample should be lowered back into the central thimble and allowed to decay longer, or the unloading should be done remotely (e.g., from the building roof). Unloading of samples requires the proper entries on the irradiation request form and in the operations logbook.

### 10.2.2 Rotary Specimen Rack Assembly (Lazy Susan)

This isotope production facility permits dry irradiation of specimens and production of isotopes immediately external to the core. The facility consists primarily of five components:

1. the watertight rotary specimen rack assembly which surrounds the core;
2. the specimen removal tube;
3. the rotating drive tube-and-shaft assembly;
4. the electric motor drive-and-indicator assembly on the reactor bridge; and
5. the pneumatic specimen-removal assembly which is used for the insertion and removal of specimen containers.

Loading and unloading of the isotope-production facility can be done while the reactor is operating.

The rotary specimen rack assembly consists of an aluminum rack that holds specimens during irradiation, and an outer ring-shaped seal-welded aluminum housing. The entire assembly rests in a well in the graphite reflector. The rack, which can be rotated inside the housing, supports 40 evenly-spaced aluminum tubes which are open at the top and closed at the bottom. The tubes serve as receptacles for 1.12 inch outside diameter specimen containers. The rack can be rotated manually or electrically, and the orientation is controlled from the drive on the bridge. Motion is transmitted through a drive shaft to a stainless steel sprocket and chain drive in the rotary specimen rack housing.

Specimen containers are inserted into and removed from the 40 specimen tubes through the specimen-removal tube, which has an internal diameter of 1.33 inches. The specimen-removal tube extends from the rotary specimen rack housing to the reactor bridge, and is offset by means of a large radii tube which bends to avoid direct-line radiation streaming from the core.

The tube-and-shaft assembly is a sealed straight tube that encloses the drive shaft and connects the rotary specimen rack housing with the drive-and-indicator assembly on the bridge. Since this tube is in a straight line from the reactor, shielding is provided by plastic enclosed within the tubing.

The drive-and-indicator assembly is located on the bridge. It includes an indicator dial with 40 divisions (one for each rack position), a crank for manual rotation of the specimen rack, and a motor and slip-clutch for electric operation. To rotate the rotary specimen rack inside its housing, the crank must be rotated or the electric motor energized. When the indicator pointer is properly positioned at the desired specimen rack position, a cam with two detents will have a roller aligned with one of the detents. This roller is attached to a lever that actuates a limit switch to stop the motor and stop the gear rotation in the drive box on the bridge for loading or unloading of specimens. After the rack is loaded, it can be continuously rotated electrically to provide uniform exposure of specimens.

Lazy Susan samples are normally placed in polyethylene containers; however, aluminum and other plastic materials may also be used. The containers vary considerably in diameter and length, thus requiring different handling techniques. The standard USGS polyethylene "rabbit" container is normally gravity dropped into the Lazy Susan and pneumatically unloaded into a small lead shielding cave. Other containers are typically gravity dropped and unloaded with a fishing-pole-type pickup device which enables the operator to keep isotopes at a safe distance and provide maximum flexibility during handling; an electric cable attached to the reel serves as a hoisting cable for the specimen container and a power conductor for actuating the specimen pickup tool. The samples can also be lowered and raised at the end of a string. Loading of samples requires the appropriate entry in the operations logbook.

Various capsules can be irradiated in the Lazy Susan. Two polyethylene containers used most frequently are the "TRIGA® tube" and the "rabbit." The capsules are designed to pass freely in a tube with a curved section radius no smaller than 2 feet. The TRIGA® tube capsule provides a space for irradiating specimens 0.56 inch in diameter and about 4 inches long. Two TRIGA® tubes may be inserted in each rotary rack position, so the total capacity is therefore 80 tubes. The "rabbit" provides a space for irradiating specimens 0.687 inch in diameter and about 2.258 inches long. Because four capsules may be inserted in each rotary rack position, the total capacity is 160 specimen containers. The total possible volume for samples for irradiation is 78.8 $in^3$ if using TRIGA® tubes or 133.8 $in^3$ if using rabbits. Any capsule is discarded when it shows a significant discoloration, as this indicates that is becoming embrittled by irradiation. A brittle capsule could shatter upon impact in the terminus, which could necessitate removal of the terminus from the core to retrieve the pieces of the capsule.

Low reactivity samples may be loaded and unloaded during reactor operation as long as care is exercised to prevent undesirable reactivity effects and excessive radiation fields. The handling personnel should always be aware of the large quantities of $^{41}Ar$ that may be produced in the Lazy Susan and the need to control the release of this material. Pneumatic unloading of the Lazy Susan should not be performed unless this argon quantity has been reduced either by sufficient decay or by pre-operational flooding of the Lazy Susan with an inert gas. If a pickup device is used to unload samples during reactor operation, the device should be moved quickly to minimize the activation caused by its presence in the core region. As always, Health Physics monitoring is required during unloading operations. Unloading of samples requires the proper entries on the irradiation request form and in the operations logbook.

### 10.2.3 Vertical Irradiation Tubes

There are two vertical irradiation tubes located in a rack mounted to the exterior of the graphite reflector. The reflector mount has four slots for insertion of tubes; these four mounting positions are labeled R1 – R4. One vertical irradiation tube, or dry tube, is located in reflector mounted position R1. This tube is aluminum for the bottom ~12 feet and polyethylene tubing for the top section. The second tube is located in position R3 and is an all-aluminum dry tube. Various spacers are used inside the bases of these tubes to raise the specimen levels to the approximate mid-plane of the core. The primary capsules used for sample irradiation are both polyethylene and aluminum "TRIGA® tubes" described in Section 10.2.2 Rotary Specimen Rack Assembly. Each dry tube is bent or curved slightly to reduce radiation streaming from the core.

The pump tube is also used as a vertical irradiation tube. This tube is used as a source of streaming neutrons and gamma particles. The tube has an outside diameter of 6" with shielding (lead and borated polymer plug) to reduce radiation streaming when not in use.

In all vertical irradiation tubes, samples are normally lowered into position by string or with the fishing-pole-type pickup device described in section 10.2.2 Rotary Specimen Rack Assembly.

### 10.2.4 The Pneumatic Transfer System

Short-lived radioisotopes are produced in a pneumatic transfer system which rapidly conveys a specimen to and from a position in a reflector mounted position or an in-core position. Four termini are available for use in this system and they may be located in-core or in a reflector mounted position (e.g. R4). Each terminus is a double-walled assembly with concentric tubes. The inner volume is where the specimen travels and the outer annulus is used for an air path.

The system includes a specimen capsule, a blower-and-filter assembly, a valve assembly, a core terminus assembly, a transfer box, up to 4 receiver/sender assemblies, timer and control assemblies, and such items as tubing, flexible hoses, and fittings.

A single large blower is used for all four pneumatic systems. Tubing from the blower extends both to the terminus in the core and to the receiver/sender unit in the counting room (Room 153). Injection and ejection of the specimen capsule are by means of a vacuum maintained by the blower. The terminus in the reactor is located in a reflector-mounted position. When the specimen capsule is injected into the terminus, it comes to rest in a vertical position approximately at the midplane of the core. The sample changer/counter are in a partitioned area of the Room 153 which is surrounded by a 2-foot thick, 9-foot high high-density concrete wall, equivalent to about 4 inches of lead shielding.

There are particular procedures that are adhered to for safe and proper operation of the pneumatic transfer system.

### 10.2.5 8-Inch Beam Tube

The 8-inch beam tube provides a large cavity for irradiating specimens which would be too large to irradiate otherwise. These specimens would otherwise not be irradiated or would have to be broken or ground into

smaller pieces which might not be possible or practical.

The tube consists of an 8" diameter schedule 40 aluminum pipe, in two sections, that totals over 25 feet long. The pipe has a welded aluminum bottom and two sealed flanges to provide for airtight closure. When the beam tube is not in use, and is located in the tank, it is flooded with water to provide radiation shielding. Prior to use, the top of the tube is sealed and air pressure is applied at $\leq$25 psig to blow out the shielding water. Lead is used to weight the bottom of the tube and provide stability. Position pins are provided at fixed vertical intervals to allow the placement of an irradiation platform at the desired position. A shielding plug is used during operation of the beam tube to reduce neutron and gamma levels in the reactor bay. Radial positioning of the tube is accomplished by moving the tube on a trolley system and pinning the trolley at the desired location. The beam tube is normally not in the reactor tank, but when in the tank and not in use, it is positioned near the tank wall and filled with water.

The pipe used is 8", schedule 40, 6061 aluminum. It has an outside diameter of 8.625" and an inside diameter of 7.981". The assembled pipe length is 25'-7.5", with approximately 24' of the pipe below water level in the reactor tank. The total weight of the pipe is approximately 260 pounds. The weight of 20 °C water displaced by the dry pipe lowered 24' into the reactor tank is 562 pounds. The approximately 302 pounds of additional weight needed to counteract the buoyancy of the dry pipe is provided by three (3) lead-filled weight cans. Each can is constructed of aluminum, with a 5.9" inside diameter and 10" height. These cans are filled with lead. The lead is canned to ensure that no lead or lead oxidation can cause blockage of the vent tube. The weight of each can is approximately 100 pounds for a three can total of about 300 pounds. This weight, in addition to other miscellaneous pipe fixtures, should be sufficient to counteract the buoyancy of the empty 8" pipe. Each weight can has an aluminum nut welded to the center of the top plate for remote handling. The pipe is also provided with internal positioning pins at fixed intervals along its vertical length. The pins are arranged in sets of three at 120° spacing around the inside diameter of the pipe. They are made of 0.25" diameter aluminum rod that protrudes 0.5" inside of the pipe. A position platform is made with a slotted edge to clear the pins as it is lowered into the pipe. When the desired position is reached, the platform is rotated and rested on the position pins. A 0.5" aluminum vent tube is provided for removal of water from the 8" pipe. The vent tube penetrates the 8" pipe at a position approximately 2" above the bottom. The tube then goes vertically through one connector to a valve that is just above the tank water level. The vent valve will normally be shut. Water is removed from the 8" pipe by pressurizing the pipe at the top and opening the vent valve. This results in water flowing from the vent tube into a suitable receptacle until the water level reaches the vent tube penetration. The vent tube penetration is slightly above the bottom of the 8" pipe to reduce the potential of blockage from objects that might fall into the pipe.

The beam plug, which is needed to reduce the neutron and gamma radiation levels at the top of the air-filled 8" pipe during reactor operations, is a structure consisting of shielding material that is supported on a 3 foot long, 0.75" diameter steel rod. Steel disks are located at each end of the rod, with one disk at 7.97" diameter so that it fits inside the 8" pipe and the other disk at 10" diameter to prevent it entering the pipe. The shielding material is layered, with boron-loaded neutron shielding toward the core and lead-loaded gamma shielding above the neutron shielding. The shielding material totals approximately 2 feet in thickness and weighs approximately 350 pounds. This weight will provide additional negative buoyancy to the 8" pipe. The neutron shielding consists of 0.5" of 25% boron-loaded sheeting and 12" of 0.9% boron-loaded castable moderating material. The gamma shielding consists of 12" of(76% by weight)lead-loaded polyethylene and 4"

of lead. Estimated attenuation factors are $1 \times 10^{10}$ for neutrons and 60,000 for gamma rays. This shield plug also contains a twisted pair of unshielded wire and one coaxial cable penetrating the shielding materials. These conductors are not located in a straight line through the shielding so that no radiation streaming should be seen.

Neutron activation of the 56 cm of aluminum pipe in the core region is of concern when the lower segment of the beam tube is removed from the reactor tank. The beam tube components are constructed of 6061 aluminum. This material contains a maximum of 0.25% copper, 0.6% silicon, 1% magnesium, and 0.25% chromium. Examining all significant activation products with half lives great than 10 hours shows that if the tube is allowed to decay for approximately 6 days, the majority of the activity will be gone. This decay would take place with the beam tube positioned against the reactor tank wall, where the neutron flux is down almost 5 orders of magnitude from the core reflector position. With a 150 hour decay period, the $^{51}Cr$ activity would create the dominant radiation hazard until the decay time was greater than about 14 months, when $^{60}Co$ would become dominant. It is preferable to allow at least a 6 month decay period prior to removing the lower beam tube segment from the reactor tank.

## 10.3 Experiment Review

Administrative requirements are in place at the GSTR to assure that all experiments are performed in a manner which will ensure the protection of the public. Experiment review meets the requirements of Regulatory Guide 2.2, and Standard ANSI N401-1974 (ANS-15.6) as modified by Regulatory Guide 2.4.

All experiments proposed for the reactor will be either Class I or Class II experiments. The classification of the proposed experiments will be the responsibility of the Reactor Supervisor.

Class I experiments include all experiments that have been run previously or that are minor modifications to a previous experiment. These are experiments which involve small changes in reactivity, no external shielding changes, and/or limited amounts of radioisotope production. The Reactor Supervisor has the authority to approve the following:

1. Experiments for which there exists adequate precedence for assurance of safety.
2. Experiments which represent less than that amount of reactivity worth necessary for prompt criticality.
3. Experiments in which any significant reactivity worth is stable and mechanically fixed, that is, securely fastened or bolted to the reactor structure.

Class II experiments include all new experiments and major modifications of previous experiments. These experiments must be reviewed and approved by the Reactor Operations Committee before being run. The Radiation Safety Committee may also be consulted. These experiments may involve larger changes in reactivity, external shielding changes, and/or larger amounts of radioisotope production. These include:

1. In-core experiments which involve, in an unstable form, reactivity worth greater than that necessary to produce a prompt critical condition in the reactor core.

# Chapter 11

# RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter deals with the radiation protection and waste management programs of the GSTR. The GSTR is housed within Building 15.

## 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 GSTR Radiation Protection Program is to keep radiation exposures at the GSTR and to the general public ALARA.

### 11.1.1 Radiation Sources

#### 11.1.1.1 Airborne Radiation Sources

The radioisotopes $^{41}$Ar and N–16 are the only airborne radioisotopes produced during normal reactor operations. In the reactor bay, $^{41}$Ar is produced primarily from irradiation of dissolved air in the primary water which eventually transfers into the air in the reactor bay. This evolution results from the reduced solubility of argon gas in water as the water temperature increases. Additionally, $^{41}$Ar can be generated from activated argon in air-filled irradiation facilities (e. g., rotary specimen rack, dry tubes, and the pneumatic transfer system), but little or none of this $^{41}$Ar gets into the reactor bay air. N–16 is predominately created by the reaction of fast neutrons with O–16 in water passing through the core. The amount of oxygen present in

air entrained in the water near the reactor core, is insignificant compared to the amount of oxygen in water. Additionally, N–16 has a short half-life of seven seconds, and the GSTR uses a nitrogen diffuser to slow the transit of N–16 from the core to the surface of the pool water to ensure the majority of N–16 has decayed before reaching the reactor bay. The N–16 diffuser is a discharge line from the primary coolant system that discharges water over the top of the reactor core. Calculations and measurements have been performed to determine production and release rates of $^{41}$Ar and N–16 due to normal reactor operations.

#### 11.1.1.1.1 Estimated Annual Dose in the Unrestricted Area from $^{41}$Ar Released During Routine Reactor Operations

In the TRIGA® Mark I reactor installation, the pneumatic transfer tubes, air-filled single irradiation tubes, and the rotary specimen rack contain air. In addition, the main reactor bay is of course also filled with air that is in contact with the reactor pool. Of the radioisotopes produced in these cavities, $^{41}$Ar is the most significant with respect to hazards and N–16 is considerably less significant.

At a prolonged 1000-kW operation with no air exchange, the $^{41}$Ar activity in the pneumatic transfer tube is 4 mCi, in the air-filled single irradiation tubes the $^{41}$Ar activity is approximately 55 mCi per tube, and in the rotary specimen rack it is 860 mCi. A measurable $^{41}$Ar release from the reactor pool water occurs during normal 1000-kw operation. The release of $^{41}$Ar from the reactor pool water is discussed in Section 11.1.1.1.2.

Release of $^{41}$Ar into the reactor bay is evaluated by representing the room as a hemisphere with a 550 cm radius. This results in a whole body dose from $^{41}$Ar that is a factor of 48.7 less than the dose received by an infinite radius hemispherical cloud, as assumed in 10 CFR Appendix B. Therefore, the values of Appendix B do not directly correspond to the radiological hazard present by the $^{41}$Ar dispersed in the reactor bay.

Operational data shows that typical $^{41}$Ar concentration in the reactor bay during 1000-kW operation is $4.35 \times 10^{-6} \mu$Ci/mL. A typical stay time in this environment of eight hours per week would give a whole body dose of about 66 mrem per calendar year. Therefore, if a person remained in the reactor bay 40 hours per week under the same conditions, a quarterly whole dose of about 82.5 mrem would result. These doses are well within the limitations of 10 CFR 20. Thus routine releases of $^{41}$Ar from reactor operation do not present a significant radiological hazard to the staff.

Accidental flooding of an air-filled experimental cavity could release larger quantities of $^{41}$Ar into the reactor bay. Flooding of the rotary specimen rack could result in the acute release of up to 860 mCi into the reactor bay instantaneously. If the room ventilation system is operating normally, this activity will remain in the room for about 12 minutes. The maximum $^{41}$Ar concentration in the room would be $2.46 \times 10^{-3} \mu$Ci/mL, assuming uniform mixing. Using the finite hemisphere cloud approximation, the discharge of 860 mCi into the reactor bay would give a whole body dose of about 17.9 mrem to a person positioned in the center of the room. The same accident, releasing 10 Ci of $^{41}$Ar would give a whole body dose of about 208 mrem. This is well below the occupational limits of 10 CFR 20.

The routine exposure of personnel in unrestricted areas from $^{41}$Ar releases is evaluated by assuming a person is standing next to the reactor building, at a distance of eight meters from the release point. An annual release of 25 Ci of $^{41}$Ar would give a whole body dose of less than 17 mrem per year to a person near the

building wall. The acute release of 3 Ci of $^{41}$Ar would give a whole body dose of about 2 mrem to a person standing outside, near the reactor building wall.

In summary, the $^{41}$Ar produced in the reactor cavities is not a significant hazard to operating personnel or the general public.

#### 11.1.1.1.2 Release of $^{41}$Ar from the Reactor Water

$^{41}$Ar activity in the reactor pool water results from irradiation of the air dissolved in the water.

The following calculations were performed to evaluate the rate of $^{41}$Ar escaping from the reactor pool into the reactor bay. The calculations show that $^{41}$Ar decays while in the water, and most of the radiation is safely absorbed in the water. The changes in $^{41}$Ar concentration in the reactor, in the pool water external to the reactor, and in the air of the reactor bay are given by:

$$V_1 \frac{\mathrm{d}N_1^{41}}{\mathrm{d}t} = V_1 \bar{\Phi} N_1^{40} \sigma^{40} - N_1^{41} \left( v_1 + V_1 \bar{\Phi} \sigma^{41} + \lambda^{41} V_1 \right) + N_2^{41} v_1, \tag{11.1}$$

$$V_2 \frac{\mathrm{d}N_2^{41}}{\mathrm{d}t} = -\lambda^{41} N_2^{41} V_2 + v_1 \left( N_1^{41} - N_2^{41} \right) - \left( f_{2 \to 3} N_2^{41} V_2 - f_{3 \to 2} N_3^{41} V_3 \right), \text{ and} \tag{11.2}$$

$$V_3 \frac{\mathrm{d}N_3^{41}}{\mathrm{d}t} = \left( f_{2 \to 3} N_2^{41} V_2 - f_{3 \to 2} N_3^{41} V_3 \right) - N_3^{41} \left( \lambda^{41} V_3 + q \right). \tag{11.3}$$

Where:

**subscript 1** = Reactor region (water in the core),

**subscript 2** = Reactor tank water region external to the reactor,

**subscript 3** = Reactor bay region,

**superscript 40** = Ar–40,

**superscript 41** = $^{41}$Ar,

**superscript** $A$ = Ar–40 plus $^{41}$Ar,

$V$ = Volume of region, cm$^3$,

$N$ = Atomic density, atoms/cm$^3$,

$\lambda$ = Decay constant, sec$^{-1}$,

$\sigma$ = Absorption cross section, cm$^2$,

$q$ = Volume flow rate from reactor bay exhaust, (cm$^3$·sec),

$v_1$ = Volume flow rate through region No.1, (cm$^3$·sec),

$\bar{\Phi}$ = Average thermal neutron flux in Region No. 1, (n/cm$^2$·sec),

$f_{i \to j}$ = Fraction of $^{41}$Ar atoms in region I that escape to region j per unit time, $\text{sec}^{-1}$.

To estimate the volume flow of water through the reactor, the following equation is used:

$$V_1 = \frac{Q}{C_p \delta T \rho}, \tag{11.4}$$

where:

$V_1$ = Volume flow rate of the water through the core,

$Q$ = Reactor power = 106 watts,

$C_p$ = Specific heat of the water $\sim 4.19$ watt·sec/g·°C,

$\delta T$ = Temperature rise across the core $\sim 75$°C ( conservatively high),

$\rho$ = Exit water density = $0.958 \text{g/cm}^3$.

Thus

$$V_1 = \frac{10^6}{(4.19)(75)(0.958)} \frac{\text{cm}^3}{\text{sec}} = 3.3 \times \text{cm}^3/\text{sec}. \tag{11.5}$$

Equation 11.1 can be reduced to:

$$V_1 \frac{\mathrm{d}N_1^{41}}{\mathrm{d}t} = V_1 \bar{\Phi} N_1^{40} \sigma^{40} - \left(N_1^{41} - N_2^{41}\right) v_1, \tag{11.6}$$

by considering the following numbers:

$v_1 = 3.3 \times 10^3$ $\text{cm}^3$/sec,

$V_1 = 1.45 \times 10^4$ $\text{cm}^3$,

$\bar{\Phi} = 1.2 \times 10^{13}$ $\text{n/cm}^2$·sec,

$\sigma^{40} = 0.60 \times 10^{-24}$ $\text{cm}^2$, and

$\lambda^{41} = 1.06 \times 10^{-4}$ $\text{sec}^{-1}$,

to show that $v_1 + V_1 \bar{\Phi} \sigma^{40} + \lambda V_1 \simeq v_1$.

During equilibrium conditions the three Equations reduce to:

$$V_1 \bar{\Phi} N_1^{40} \sigma^{40} = \left(N_1^{41} - N_2^{41}\right) v_1, \tag{11.7}$$

$$N_2^{41} = \left[\lambda^{41} V_2 f_{2 \to 3} v_2\right] = \left(N_1^{41} - N_2^{41}\right) v_1 + f_{3 \to 2} N_3^{41} V_3, \tag{11.8}$$

$$N_3^{41}\left[\lambda^{41}V_3 + q + f_{3\to 2}V_3\right] = f_{2\to 3}N_2^{41}V_2. \tag{11.9}$$

Combining Equations 11.7 and 11.8 gives

$$N_2^{41} = \frac{V_1\bar{\Phi}N_1^{40}\sigma^{40}}{\lambda^{41}V_2 + f_{2\to 3}V_2} + \frac{f_{3\to 2}N_3^{41}V_3}{\lambda^{41}V_2 + f_{2\to 3}V_2}, \tag{11.10}$$

which inserted into Equation 11.9 for $N_2^{41}$ yields

$$N_3^{41}\left[\frac{\lambda^{41}V_3 + q + f_{3\to 2}V_3}{f_{2\to 3}V_2} - \frac{f_{3\to 2}V_3}{\lambda^{41}V_2 + f_{2\to 3}V_2}\right] = \frac{V_1\bar{\Phi}N_1^{40}\sigma^{40}}{\lambda^{41}V_2 + f_{2\to 3}V_2}. \tag{11.11}$$

The values of the constants in Equation 11.11 are:

$V_2 = 4.02 \times 10^7$ cm$^3$ (8.5 ft by 25 ft),

$V_3 = 3.12 \times 10^8$ cm$^3$ (21 ft by 36.5 ft by 16 ft),[†]

$q = 1.73 \times 10^5$ cm$^3$/sec,

$\sigma^{40} = 0.53 \times 10^{-24}$ cm$^2 \cdot \frac{1}{\sqrt{\pi}} = 0.47 \times 10^{-24}$ cm$^2$.

Which leaves $N_1^{40}$, $f_{3\to 2}$, and $N_2^{41}$ to be evaluated. The saturated concentration of argon in water according to Henry's law is

$$X = \frac{P}{K}, \tag{11.12}$$

where $X$ is the mole fraction of argon in water, $P$ is the partial pressure in mm Hg of argon above the water, and $K$ is Henry's constant, which for argon is $2.9 \times 10^7$ mm Hg per mole fraction of gas in solution at 30 °C. Since the argon content of air is 0.94% by volume, the partial pressure of argon above the water is 0.0094(650 mm Hg - 23 mm Hg) = 5.89 mm Hg. This yields $6.77 \times 10^{15}$ argon atoms per cc of water.

An estimate of the parameter $f_{2\to 3}$ (that is the fraction of argon atoms in the pool water that escape each second) can be obtained by examining the motility of ions in dilute solution. Most ions have velocities of the order of 3 to $8 \times 10^{-4}$ cm/sec under a potential gradient of 1 volt per cm.[‡]

In the potential gradient, the argon's velocity should be less than $3 \times 10^{-4}$ cm/sec. Therefore, only the argon atoms within $3 \times 10^{-4}$ cm of the pool surface will be in a region in which the argon atoms can leave the water within any given second. Actually, even this source volume is still too large. Nevertheless it gives an upper limit for the fraction of the total argon atoms that can leave the water per second:

$$f_{2\to 3} = \frac{3 \times 10^{-4}}{\text{water height}} = \frac{3 \times 10^{-4}}{7.31 \times 10^2} = 4.1 \times 10^{-7}\text{sec}^{-1}. \tag{11.13}$$

[†]The room volume has been reduced by 10% to account for the presence of equipment, instuments, tools, etc.
[‡]Daniels, F. Outlines of Physical Chemistry, Wiley and Sons, New York, 1948, p414.

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

$$f_{2\rightarrow 3} N_2^A V_2 = f_{3\rightarrow 2} N_3^A V_3, \tag{11.14}$$

where

$N_3^A = 2.1 \times 10^{17}$ argon atoms/cm$^3$ of air $\simeq N_3^{40}$,

$N_2^A = 6.77 \times 10^{15}$ argon atoms/cm$^3$ of water $\simeq N_1^{40}$.

Solving for $f_{3\rightarrow 2}$ gives

$$f_{3\rightarrow 2} = f_{2\rightarrow 3} \frac{N_2^A V_2}{N_3^A V_3} = 1.70 \times 10^{-9} \text{sec}^{-1}. \tag{11.15}$$

Since $\lambda^{41} > f_{2\rightarrow 3} > f_{3\rightarrow 2}$, Equation 11.11 reduces to

$$N_3^{41} = \frac{V_1 \Phi N_1^{40} \sigma^{40}}{\lambda^{41} V_3 + q} \frac{f_{2\rightarrow 3}}{\lambda^{41}} = 1.04 \text{ atoms/cm}^3. \tag{11.16}$$

Solving for $N^{41}$ yields 0.48 atoms/cm$^3$. This corresponds to a concentration of $^{41}$Ar activity of

$$A^{41} = \frac{\lambda^{41} N^{41}}{C} = \frac{1.06 \times 10^{-4} \cdot 1.04}{3.7 \times 10^4} = 0.297 \times 10^{-8} \quad \mu \text{ Ci/cm}^3, \tag{11.17}$$

where $A^{41}$ = $^{41}$Ar concentration, $\mu$Ci/cm$^3$, and $C$ = Conversion factor from disintegration/sec to $\mu$Ci. The $^{41}$Ar concentration of $0.297 \times 10^{-8}$ $\mu$Ci/cm$^3$, is below the limit of $1 \times 10^{-8}$ $\mu$Ci/mL for uncontrolled access.

The $^{41}$Ar activity discharge rate from the reactor bay is obtained by multiplying the activity concentration by q, the value of the air discharged per unit time, that is

$$A^{41} q = 0.297 \times 10^{-8} \times 1.73 \times 10^5 = 0.513 \times 10^{-3} \quad \mu\text{Ci/sec}. \tag{11.18}$$

#### 11.1.1.1.3 N–16 Activity in the Reactor Bay

The cross section threshold for the O–16 (n,p) to N–16 reaction is 9.4 MeV; however, the minimum energy of the incident neutrons must be about 10.2 Mev because of the center of mass corrections. The high threshold limits the production of N–16 since only about 0.1% of all fission neutrons have energy in excess of 10 MeV. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the threshold. The measured effective cross section for O–16 (n,p) to N–16 reaction is 0.099 mbarn. The concentration of N–16 atoms per cm3 of water as it leaves the reactor core is given by

$$N^N = \frac{\Phi_v N^O \sigma^O}{\lambda^N} \left[1 - e^{lambda^N t}\right], \tag{11.19}$$

where

$N^N$ = N–16 atoms per $cm^3$ of water,

$\Phi_v$ = Virgin fission neutron flux $\simeq 10^{13}$ n/$cm^2$·sec at 1000 kW,

$N^O$ = Oxygen atoms per $cm^3$ of water = $3.3 \times 10^{22}$ atoms/$cm^3$,

$\sigma^O$ = Absorption cross section of oxygen = $2 \times 10^{-29}$ $cm^2$,

$\lambda^N$ = N–16 decay constant $-9.35 \times 10^{-2}$ $sec^{-1}$,

$t$ = Average time of exposure in reactor.

The average exposure time in the reactor is given by:

$$t = \frac{V_c}{v1}, \tag{11.20}$$

where $V_c$ is the core water volume exposed to flux $\Phi_v$, and $v_1$ is the volume flow rate through the core (see section 11.1.1.1.2). Thus,

$$t = \frac{1.45 \times 10^4 \text{ cm}^3}{3.3 \times 10^3 \text{ cm}^3/\text{sec}} = 4.4 \text{ sec.} \tag{11.21}$$

Solving for $N^N$ from Equation 11.19, one obtains $2.4 \times 10^7$ N–16 atoms per $cm^3$ of water leaving the core. With a flow rate of $3.3 \times 10^3$ $cm^3$/sec, the rate of nitrogen leaving the core is therefore $7.9 \times 10^{10}$ atoms/sec.

In the TRIGA® Mark I reactor the measured transport time for the water to travel the 16 feet from the reactor core to the surface of the tank is 42 seconds when the reactor is operating at 100 kW. To a first approximation, the velocity of the rising water is proportional to the density difference between the pool water and the heated water from the core,

$$v = k\left(\rho_O - \rho_{\text{exit}}\right). \tag{11.22}$$

Thus the velocity of the rising water column for the TRIGA® Mark III reactor can be estimated from

$$v_{\text{III}} = \frac{v_1\left(\rho_O - \rho_{\text{exit III}}\right)}{\rho_O - \rho_{exitI}}, \tag{11.23}$$

where the subscripts I and III refer to TRIGA® Mark I and Mark III reactors respectively.

The water leaving the TRIGA® Mark I core at 100 kW is at 45 °C with a density of 0.9902 g/$cm^3$. The water leaving the TRIGA® Mark III core at 1000 kW is conservatively assumed to be at 100 °C for this calculation, with a density of 0.9583 g/$cm^3$. Thus, from Equation 11.23

$$v_{III} = \frac{16 \times 30.5 \text{ cm}}{42.0 \text{ sec}} \frac{0.99705 - 0.95835}{0.99705 - 0.99021} = 65.6 \text{ cm/sec.} \tag{11.24}$$

The transport time for N–16 through the 20 ft of water above the TRIGA® Mark III is then:

$$t_{rise} = \frac{20 \times 30.5 \text{ cm}}{65.6} = 9.3 \text{ sec.} \tag{11.25}$$

Equation 11.25 assumes that the N–16 rises straight up toward the pool surface. In practice, however, the N–16 is slowed down by the interruption of the vertical convective currents from the discharge of water through the diffuser nozzle across the core.

Measurements at the prototype TRIGA® Mark III reactor at 1000 kW indicate that the diffuser results in a hold-up time increase by a factor of 3, so that the actual rise time is closer to 25 seconds. In 25 seconds, the N–16 decays to 0.096 of its initial value. Thus, the number of N–16 atoms that reach the water near the pool surface is about $7.6 \times 10^9$ atoms/second.

Only a small portion of the N–16 atoms present near the pool surface are transferred into the air of the reactor bay. When a N–16 atom is formed, it appears as a recoil atom with various degrees of ionization. For high-purity water($\approx 2$ $\mu$mho), practically all of the N–16 combines with the oxygen and hydrogen atoms of the water. Most of it combines in an anion form, which has a tendency to remain in the water. It is assumed that it least one-half of all ions formed are anions. Because of its 7.4 second half-life, the N–16 will not live long enough to attain a uniform concentration in the tank water. With the diffuser in operation, one can assume that the N–16 atoms will be dispersed in the one ft of water at the top of the pool directly above the core. Actually, they are more likely to be dispersed over a wider area in the pool and will decay before this lateral movement is completed. In the area directly above the core, the dominant contribution to the dose rate is the direct radiation from the core. The interest from a hazard point of view is then the number of N–16 atoms escaping into the air and diffusing away from the area above the core. The maximum fraction of N–16 atoms that can escape from the water to the air per second can be estimated similarly to the case of argon (see Section 11.1.1.1.2). Thus

$$f^N_{2\to 3} \leq \frac{1}{2}\frac{3 \times 10^{-4} \text{ cm/sec}}{30 \text{ cm}} = 0.5 \times 10^{-5} \text{ sec}^{-1}, \tag{11.26}$$

where it is assumed that one-half of the ions formed, namely the anions, remain in the water. Thus the number of N–16 atoms entering the air is given by

$$f^N_{2\to 3} N^N_V = \frac{f_{2\to 3}\left(7.6 \times 10^9\right)}{f_{2\to 3} + \lambda^N} = \frac{0.5 \times 10^{-5}\left(7.6 \times 10^9\right)}{0.5 \times 10^{-5} + 9.35 \times 10^{-2}} = 4.06 \times 10^5 \text{ atoms/sec.} \tag{11.27}$$

The N–16 concentration (assuming no mixing with the air in the rest of the room) in the volume immediately above the mechanism bridge, three meters high and having a 36 inch diameter, is

$$A = \frac{4.06 \times 10^5}{3.7 \times 10^4 \cdot 300 \text{ cm} \cdot \pi \cdot 18^2 \cdot 2.54^2} = 5.6 \times 10^{-6} \quad \mu\text{Ci/cm}^3. \tag{11.28}$$

The dose rate from the concentration in Equation 11.28, in the volume described, 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. Therefore the rate of

accumulation of N–16 in the room as a whole is given by

$$\frac{\mathrm{d}\left(v_3 N_3^{16}\right)}{\mathrm{d}t} = S - \left(\lambda^N + q/V_3\right) N_3^{16} V_3, \tag{11.29}$$

where

$S =$ Number of N–16 atoms entering the room from the pool per second,

$V_3 =$ Volume of the reactor bay $= 3.12 \times 10^8$ cm$^3$,

$q =$ Volume flow rate from the reactor bay exhaust $1.73 \times 10^5$ cm$^3$/sec

For saturation conditions,

$$V_3 B_3 16 = \frac{S}{\lambda^N + q/v_3} = \frac{4.06 \times 10^5}{9.35 \times 10^{-2} + 0.471 \times 10^{-3}} = 4.31 \times 10^6 \text{ nuclei.} \tag{11.30}$$

Equation 11.30 corresponds to an activity concentration of $3.49 \times 10^{-8}$ $\mu$Ci/cm$^3$, which is negligible. Thus, everywhere but directly above the core, the N–16 activity presents no hazard. Above the core, the direct activity of the core is dominant.

#### 11.1.1.1.4 Routine Release of $^{41}$Ar to Atmosphere

Expressions for the concentration of $^{41}$Ar in the exposure room were presented earlier. Although higher concentrations will prevail with stagnant air in the exposure room, air will not be discharged to the atmosphere under these conditions. When the room is being ventilated, the concentration will be a few percent of the stagnant-air value. When activity is released from the building, the concentration on the ground must be less than the effluent concentration in unrestricted areas set by the NRC. One appropriate method for determining the ground level concentration is to use the following equation for predicting the cloud concentration at ground level at the building,

$$X(0,0,0) = \frac{1}{(0.5)A\bar{u}}(Q), \tag{11.31}$$

where

$X(0,0,0) =$ Concentration at the building, $\mu$Ci/m$^3$,

$Q =$ Activity release rate, $\mu$Ci/sec,

$\bar{u} =$ Mean wind speed, 1 m/sec,

$A =$ Cross sectional area of the reactor bay, 31.2 m$^2$.

Applying a reasonable duty cycle for the operation of the facility, 6 hours per day and five days per week of 1000-kW operation, yields an average $^{41}$Ar discharge rate of

$$\bar{Q} = \frac{6}{24} \cdot \frac{5}{7} \left(0.513 \times 10^{-3}\right) = 0.913 \times 10^{-4} \quad \mu\text{Ci/sec} \tag{11.32}$$

Substitution of $0.913 \times 10^{-4}$ $\mu$Ci/sec for $Q$ in Equation 11.31 gives a value of $X(0,0,0) = 5.85 \times 10^{-6}$ $\mu$Ci/m$^3$.

#### 11.1.1.1.5 Occupational Exposure to $^{41}$Ar from Routine Reactor Operations

The only significant source of $^{41}$Ar that contributes to occupational radiation exposure is that which is generated in, and released from, the reactor tank, regardless of the ventilation or irradiation facility valve configuration. As noted in the previous section, the stack effluent discharge rate for $^{41}$Ar, when the ventilation valves associated with the various irradiation facilities are closed (i.e., the configuration in which the GSTR is routinely operated) is approximately $10^{-4}$ $\mu$Ci/s. With a stack effluent discharge rate of 1 $\mu$Ci/s, a volume flow rate of $5.5 \times 10^5$ cm$^3$/s, and assuming uniform mixing in the reactor bay, the concentration of $^{41}$Ar in the reactor bay would be approximately $1.8 \times 10^{-10}$ $\mu$Ci/cm$^3$. This is well below the 10 CFR 20 listed Derived Air Concentration (DAC) for $^{41}$Ar of $3.0 \times 10^{-6}$ $\mu$Ci/mL.

All operational scenarios show that the concentration of $^{41}$Ar will be significantly less than the DAC. Given the above considerations, any estimated occupational doses must be considered highly conservative upper limits for the total effective dose equivalent (TEDE) due to $^{41}$Ar evolving from the primary tank or irradiation facilities and certainly, in all cases, less than the limits specified in 10 CFR 20.

### 11.1.1.2 Liquid Radioactive Sources

No liquid radioactive material is routinely produced or used in normal operations of the GSTR except for the neutron activation product impurities in the primary coolant. The majority of these impurities are removed by a mechanical filter and demineralizer resin. Non-routine liquid radioactive waste could be generated from decontamination or maintenance activities; however, based on past experience, the quantity and radioactivity concentrations would be small.

Radionuclides and their concentrations in the primary coolant vary depending on reactor power, reactor operating time and time since reactor shutdown, assuming that other variables remain constant. Routine liquid scintillation counting and gamma spectroscopy analysis of primary coolant water samples taken after several hours at 1 MW (equilibrium) reveal the presence of several radioisotopes. It is GSTR policy not to release liquid radioactivity as an effluent; therefore, the primary coolant does not represent a source of exposure to the general public during normal operations. Occupational exposure from liquid sources is also limited because there are few operations which require contact with the primary coolant, and because the majority of the radionuclides produced have short half-lives. In cases where contact is a potential, the primary water could be allowed to decay in order to significantly reduce radioactivity concentrations or protective equipment can be used. Additionally, experience at the GSTR and other TRIGA® reactors has shown that tritium, H–3, is not a significant source of occupational exposure.

#### 11.1.1.3 Solid Radioactive Sources

The major source of radiation and radioactivity from solid sources is the fission product generation in the reactor fuel. [redacted]

### 11.1.2 Radiation Protection Policy

All operations in the GSTR will be planned and conducted such that every reasonable effort will be made to maintain radiation exposures as far below the dose limits as is practical during all phases of operation. This ALARA policy will be of primary concern when working with radioactive materials. In addition, the facility will have an overall philosophy of radioactive waste minimization.

#### 11.1.2.1 Radiation Protection Organization

All reactor staff personnel and experimenters have radiation protection responsibilities. The Reactor Supervisor is responsible for establishing a radiation protection program based on sound engineering and radiation protection principles. He or she shall ensure that sufficient priority and authority is given to radiation protection concerns and that the proper reviews and audits are conducted. The designated Reactor Health Physicist shall implement the radiation protection program and be responsible for the day-to-day activities required by the plan. The Reactor Health Physicist reports to the Reactor Supervisor and has direct access to the Reactor Supervisor and Reactor Administrator. Coordination is maintained with the USGS Central Region safety personnel. The Health Physicist has the freedom and authority to stop operations that are indefensible from a radiation safety view. Radiation safety equipment and supply needs will be evaluated by the Reactor Health Physicist and submitted to the Reactor Supervisor. The Reactor Health Physicist shall have a bachelor's degree or higher in Health Physics (or similar science or engineering field) or a minimum of 3 years experience in radiation safety aspects of research or reactor operations.

All Reactor Operators are responsible for understanding and following the radiation protection program. They are responsible for their actions and the actions of any visitors, contractors, or experimenters under their jurisdiction.

All reactor staff members shall receive radiation protection training on a biennial basis, not to exceed 30 months. The Reactor Health Physicist, in consultation with the Reactor Supervisor, will develop the radiation protection training material. The Reactor Health Physicist will be responsible for coordinating this training.

#### 11.1.2.2 Administrative Controls and Responsibilities

The Senior Reactor Operator in charge of each reactor experiment and the Reactor Health Physicist are responsible to assure that each operation is being performed in compliance with the ALARA policy. It is also the responsibility of the Reactor Health Physicist or a designated substitute to provide radiation protection training, establish and maintain restricted areas and to insure that radiation and contamination levels are not gradually increasing. All reactor staff members have the authority and responsibility to stop any operation that they feel presents an undue radiation safety risk. The Reactor Supervisor or the Reactor Administrator will resolve staff disagreements. Each facility staff member is responsible for insuring that the proper personal dosimetry is worn. The Reactor Supervisor is responsible for insuring that proper radiation protection equipment and supplies are available.

- The Reactor Health Physicist shall perform quarterly reviews, not to exceed 4 months, of occupational exposures to personnel within the reactor facility.
- The Reactor Supervisor will perform an audit of the staff radiation exposures and radioactive material releases from the facility annually, not to exceed 15 months.
- The Reactor Operations Committee shall review the effectiveness of the radiation protection program annually and suggest changes that might reduce overall exposures or releases, not to exceed 15 months.
- Reactor staff members are encouraged to make suggestions and recommendations to the Reactor Supervisor for changes to equipment or procedures that would achieve reductions in radiation exposure.
- The following radiation exposures will be cause for an administrative review, by the Reactor Supervisor and Reactor Administrator, of the facility radiation protection program:
  - Individual acute or monthly whole body exposure of more than 500 mrem.
  - Individual acute or monthly extremity exposure of more than 1000 mrem.
  - Acute or monthly committed dose equivalent to any organ of more than 1000 mrem.
  - Acute whole body exposure of more than 25 mrem to a facility visitor.

#### 11.1.2.3 Health Physics Training

All personnel who routinely enter restricted areas shall receive training in radiation protection sufficient for the work or visit or shall be escorted by an individual who has received such training. No training is required for occasional visitors with escorts or persons in uncontrolled or unrestricted areas. All individuals associated with the health physics functions of the reactor facility are given initial training in the following radiation protection subjects:

1. Rules and regulations of the Nuclear Regulatory Commission and the U.S. Geological Survey.
2. Personnel dosimetry, protective equipment and its proper use.
3. Survey instruments, types of instruments and their proper use.

4. Contamination control and reporting.
5. Waste handling and disposal.
6. Emergency procedures, alarms, and corrective actions.
7. Access control and escort procedures.
8. Radiation hazards and risks; radiological safety principles, policies, and procedures.
9. Posting and labeling requirements; definitions of radiation area, high radiation area and very high radiation area.
10. ALARA program and exposure limits.

Periodic retraining will be performed as specified in Section 11.1.2.1. The periodic training for reactor staff members shall be a condensed version of the items listed above, with a focus on changes in policies, procedures, requirements and facilities. Reactor operators need not repeat radiation protection training subjects that are covered by the Reactor Operator requalification program in that same training cycle.

#### 11.1.2.4 Records

Radiation protection records shall be maintained under the control of the Reactor Health Physicist. All records shall be retained for at least three years. Records that will be kept at least five years are; reportable occurrences, Technical Specification surveillance items, radiation surveys and contamination surveys required by regulations, and reviews and audits. Records of radioactive effluents, environmental monitoring, personnel radiation exposure and locations of inaccessible contamination will be retained for the life of the facility. Logbooks are maintained by the Reactor Health Physicist, detailing the results of wipe surveys, radiation surveys, records of radioactive samples transferred from the reactor facility, and records of radioactive materials removed and/or discharged from the reactor facility. Further records detail CAM filter results, environmental monitoring results, instrument calibrations, radiation safety training, and any unusual radiological occurrences. A file is also maintained of personnel exposures for facility staff and visitors. Whole body (in-vivo) monitoring results are maintained for facility staff.

### 11.1.3 Hazards Associated with the Movement of Materials Activated in the Reactor

Production of radioactive sources in the reactor presents possible radiation hazards that, upon removal, can be both external and internal biological hazards. All materials that are removed from the reactor are monitored with an appropriate survey instrument under the supervision of the Reactor Health Physicist or the designated representative. Radioactive materials must be properly labeled and packaged before removal from the reactor facility. Approval by a reactor staff member must be granted before these radioactive materials may be removed from the reactor facility. This also requires that the recipient possess a proper radioactive material license before the material will be transferred. All transfers of byproduct material will be recorded on facility Radioisotope Request and Receipt (RR&R) forms.

### 11.1.4 Radioactive Material Storage–Reactor Facility

All radioactive materials stored within the reactor facility are located in one of several areas designated for this purpose. [redacted] The Reactor Health Physicist is responsible to insure that these areas are properly marked with the appropriate radiation labels and the contents of the storage facilities are properly maintained. [redacted]. The Reactor Supervisor maintains an inventory of all SNM under the reactor license.

### 11.1.5 ALARA Program

**OBJECTIVES** The objective of the ALARA program is to maintain exposures of radiation and releases of radioactive effluents at levels that are ALARA and within the established limits of the Nuclear Regulatory Commission and U.S. Geological Survey. The occupational dose ALARA objective for the GSTR is an annual limit of one rem and an average of no more than 0.5 rem per year for the maximally exposed worker under normal conditions. The ALARA objective for the GSTR is 50 mrem per year, direct exposure, to any member of the public. The ALARA goal for facility effluents is 10 mrem per year to the nearest public receptor.

**POLICY** All operations in the GSTR will be planned and conducted such that every reasonable effort will be made to maintain radiation exposures as far below the dose limits as is practical during all phases of operation. This ALARA policy will be of primary concern when working with radioactive materials.

**MANAGEMENT COMMITMENT** The ALARA program is supported by the full commitment of the GSTR management. The Reactor Administrator and the Director of the USGS uphold the efforts of the reactor staff to limit radiation exposures to the lowest practicable levels.

**IMPLEMENTATION** It is the responsibility of the Reactor Health Physicist or a designated substitute to implement radiation protection policies. These will follow ALARA policies, provide training, establish and maintain restricted areas and insure that radiation, contamination and effluent levels are not gradually increasing at the GSTR. This person will also perform radiological safety planning as an integral part of operations planning and will submit all equipment and supply needs to the Reactor Supervisor to ensure adequate resources. The Reactor Health Physicist may stop any operation if radiation safety concerns are raised by any member of the staff. The Reactor Health Physicist or his designee shall review personnel exposure records, environmental monitoring, and radiological effluents at least quarterly (not to exceed 4 months) to ensure that the ALARA policy and objectives are met.

**DESIGN FACTORS** The design of experiments and facility changes shall take into account a dose criterion of 10% of the ALARA dose objectives while using conservative occupancy factors and realistic source terms. Designs shall consider the following:

External Radiation Control

1. Shielding
2. Materials of construction

3. Radioactive material processing, storage and disposal
4. Radiation monitoring systems
5. Facility layout for personnel traffic and equipment maintainability and accessibility
6. Systems and devices to control access to high and very high radiation areas

Contamination Control

1. Ventilation and filter systems
2. Confinement to minimize contamination spread
3. Containment to prevent contamination spread
4. Construction materials to ease decontamination
5. Facility layout; flow of personnel and material

Radioactive Effluent Control

1. Control by containment or confinement
2. Gaseous waste disposal capabilities
3. Exhaust system features
4. Containment of insoluble radioactive materials
5. Possible reactions in liquid waste streams
6. Effluent monitoring systems

**OPERATIONS FACTORS** The total effective dose equivalent shall be the quantity of concern when considering the ALARA dose objective, and the impact on external exposures should be considered before any extraordinary contamination controls are performed. The following should be considered for routine and special operations:

Operations Planning

1. Possible mechanical problems and exposure to radiation, contamination and airborne radioactive materials
2. Radioactive decay benefits
3. The feasibility of reducing the radiation levels by draining, flushing, or otherwise decontaminating or relocating the component of interest
4. Location of personnel pathways

5. Assessment of abnormal occurrence response capability
6. Use of portable or temporary shielding
7. Use of portable or temporary ventilation system
8. Preoperational briefing for staff assigned to perform tasks in high radiation areas
9. Performing dry runs on mockup equipment to find possible problems and train staff
10. Use of special communications equipment
11. Availability of sufficient and proper radiation monitoring equipment
12. Personnel dose action levels for management review

Operations

1. Sufficient supervision and surveillance to ensure use of procedures and planned precautions
2. Prompt notification of management when dose limits are approached or unanticipated problems develop
3. Use of properly functioning protective equipment

**REVIEWS** The Reactor Health Physicist shall perform quarterly reviews, not to exceed 4 months, of occupational exposures to personnel within the reactor facility. Annually, not to exceed 15 months, the Reactor Supervisor will perform an audit of the staff radiation exposures and radioactive material releases from the facility. Annually, not to exceed 15 months, the Reactor Operations Committee shall review the effectiveness of the radiation protection program and suggest changes that would reduce overall exposures or releases.

### 11.1.6 Radiation Monitoring

Radiation monitoring shall be performed for the detection and evaluation of occupational and public radiation exposures resulting from facility operations. The monitoring program is under the direction of the Reactor Health Physicist and includes personnel monitoring, area monitoring, contamination monitoring, airborne radioactivity monitoring, liquid radioactivity monitoring, environmental monitoring, and emergency radiation monitoring. The different aspects of the monitoring program are discussed in the following sections. Table 11.1 shows the radiation monitoring equipment used at GSTR facility.

#### 11.1.6.1 Personnel Monitoring

A National Voluntary Laboratory Accreditation Program (NVLAP) dosimetry service is provided for the reactor staff and reactor experimenters. Whole body badges are typically used with both neutron and gamma sensitive elements. Dosimeters are normally processed monthly or quarterly; however, rapid processing is available in cases of suspected overexposure. Gamma sensitive pocket self-reading dosimeters and thermoluminescent dosimeter (TLD) finger rings are used to supplement body badges. The pocket dosimeters are

Table 11.1: Radiation Monitoring Equipment Used in the GSTR Radiation Protection Program

| Item | Location | Function |
|---|---|---|
| Continuous Air Monitor | Reactor Top | Airborne Particulate |
| Continuous Air Monitor | Effluent Stack | Airborne Particulate and Gas |
| Area Radiation Monitors | Various locations in reactor bay | Measure ambient gamma radiation fields |
| Portable Ion Chamber Survey Meters | Reactor Bay and Control room | Measure beta/gamma exposure rates |
| Portable Pancake-Probe GM Survey Meters | Reactor Bay and control room | Measure beta/gamma surface contamination |
| Gamma R Survey Meters | Control Room | Measure gamma exposure rates |
| Neutron Survey Meter | Reactor Bay | Measure neutron dose rates |
| Alpha Survey Meters | Control Room | Measure alpha surface contamination |
| HPGe Gamma Spectroscopy System | Room 157 | Gamma spectroscopy |
| Gas-Flow Proportional Counter | Reception Area | Measure alpha/beta contamination on swipes |
| Hand-and-Foot Monitors | Reception Area | Measure potential contamination on hands and feet prior to leaving radiation restricted areas |
| Direct Reading Pocket Dosimeters | Reception Area | Measure personnel gamma dose |
| TLDs | Various on-site and off-site locations | Measure environmental gamma radiation doses |

Table 11.2: Typical Personnel Monitoring Devices Used at the GSTR

| Type | Dose Radiation Measured | Reading Frequency | |
|---|---|---|---|
| Pocket Ion Chamber | Deep Dose Equivalent | Gamma | As Needed |
| TLD | Deep Dose Equivalent<br>Eye Dose Equivalent<br>Shallow Dose Equivalent | Beta, Gamma<br>Thermal Neutron | Monthly |
| TLD Finger Ring | Extremity Dose Equivalent | Beta, Gamma | Monthly |

Table 11.3: Average Annual Dose Equivalent Incurred by GSTR Staff in CY 2006 (1/1/06 - 12-31-06)

| Type of Dose Incurred | Average Annual Dose Equivalent (mrem) | Highest Individual Annual Dose Equivalent (mrem) | Lowest Individual Annual Dose Equivalent (mrem) |
|---|---|---|---|
| Deep | 137 | 237 | 73 |
| Eye | 141 | 239 | 74 |
| Shallow | 158 | 261 | 74 |
| Extremity | 292 | 430 | 211 |

used for occasional visitors and are issued at the reactor facility under the direction of the Reactor Supervisor. Table 11.2 shows the typical personnel monitoring devices used at the GSTR, and Table 11.3 shows the average annual dose incurred by the GSTR staff in 2006.

#### 11.1.6.2 Area Monitoring

Areas that are potentially radiation areas or high radiation areas are monitored with fixed RAMs or portable instruments. The fixed monitors in the reactor facility have remote readout meters and visual-audible alarms located in Room 152. Selected RAMs are connected to the FPS 24-hour alarm system. A high alarm on any of these detectors will actuate an alarm signal at the FPS office. The RAM monitoring system is operated from 115 VAC power that is connected to the Building 15 emergency power supply to provide a highly dependable power source.

#### 11.1.6.3 Contamination Monitoring

Wipe surveys will be taken at numerous facility locations with each wipe covering approximately 100 $cm^2$ of area. Gross alpha and beta counting will be completed in a suitable gas flow proportional counter or equivalent to determine if loose contamination is present. Limits of 450 pCi/100 $cm^2$ beta activity or 200 pCi/100 $cm^2$ alpha activity will determine which locations need to be decontaminated. Hands, feet and any other potentially contaminated body areas will be monitored for all personnel that have gone into the reactor bay (Room 149). This monitoring will be done with a hand and foot monitor or a sensitive thin-window Geiger-Mueller (GM) instrument. This monitoring must be performed before the individual leaves the reactor facility. Routine monitoring (at least weekly) of radiation levels with a portable instrument shall

be performed for accessible areas in the facility where radioactivity levels may change significantly. These areas are primarily in the reactor bay (Room 149) and include sample storage areas, the demineralizer tank, the demineralizer pre-filter, and the reactor tank perimeter.

Materials, tools and equipment that are used in areas where contamination is likely shall be surveyed before removal from the facility. This survey shall include a wipe test for removable contamination and monitoring for fixed contamination. A removable contamination level above 450 pCi/100 $cm^2$ beta activity or 200 pCi/100 $cm^2$ alpha activity will signify the need for decontamination. Fixed contamination or activation that results in an average beta/gamma level of <2000 pCi/100 $cm^2$ and a maximum level of <6000 pCi/100 $cm^2$ will allow the item to be released for unrestricted use.

#### 11.1.6.4 Airborne Radioactivity Monitoring

Two airborne radioactivity monitors, consisting of a CAM and an $^{41}Ar$ monitor are located in the facility. The CAM is a fixed-filter type, sensitive to beta and gamma emitting particulates collected on a paper filter and a charcoal impregnated filter that is positioned behind the paper filter. The CAM is interlocked with the ventilation system and provides the signal to actuate the emergency ventilation system. A low-level warning and a high-level alarm are provided on the CAM. Both the low-level warning and high-level alarm are visibly and audibly apparent from the reactor console. A strip chart recorder provides a continuous recording of the responses of the monitor. To provide additional effluent information, a continuous $^{41}Ar$ monitor system is located within the reactor facility. A plastic sampling pipe is used to collect a small portion of the normal reactor bay exhaust. This sample is then scanned by a 2 x 3 inch NaI crystal and the signal is fed to either a single channel analyzer or multi channel analyzer calibrated to monitor the $^{41}Ar$ gamma peak. The analyzer output is recorded on a strip chart and displayed on the reactor control console. The output of this counting system is then integrated and converted to the integrated release quantity for $^{41}Ar$ from the reactor facility. This integration is normally done over weekly time periods. A grab sample and multichannel gamma analyzer system is used periodically to determine other reactor-produced radioisotopes that may be present in the exhaust effluent. If radioisotopes other than $^{41}Ar$ are found to be exiting the facility through the exhaust system, calculations will be performed to determine the concentrations in the reactor bay and at the nearest unrestricted public access.

#### 11.1.6.5 Liquid Radioactivity Monitoring

Water samples are taken periodically to determine radioactivity levels in the reactor tank water, the secondary cooling water, and the reactor tank annulus water, if present. The reactor water is analyzed for tritium concentration and all water samples are analyzed for gross alpha and beta concentrations. The radioactivity of the reactor water is analyzed at least quarterly, not to exceed 4 months. No routine discharges of radioactive liquids occur at the GSTR.

#### 11.1.6.6 Environmental Monitoring

Monitoring of radiation and contamination levels of the environment is performed to assess the effect of effluents from the GSTR. Direct radiation is measured at select locations on the facility site. TLDs are typically used for this purpose because of their high sensitivity. Biennially, not to exceed 30 months, soil and water samples are taken from locations around the facility and analyzed for contamination. Most of the sampling locations are inside the Denver Federal Center boundary; however, a few samples are taken from public or private property. Past environmental sample data, including preoperational data, are available at the facility for comparison with new data.

#### 11.1.6.7 Emergency Radiation Monitoring

Emergency radiation monitoring can be done with equipment provided in Building 15 at the Denver Federal Center. This equipment includes a gamma spectrometer, portable radiation monitors, air sampling equipment, and a low level counting instrument. Additional equipment is available through the Regional Radiation Safety Officer's office. Details are given in the facility Emergency Plan concerning emergency equipment and support.

### 11.1.7 Instrumentation

Radiation monitoring instrumentation is available for both routine and emergency monitoring. This instrumentation includes fixed area monitors, airborne radioactivity monitors, and portable survey meters. Liquid effluent radioactivity monitors are not provided because the facility does not routinely release any contaminated liquids. Routine sampling of the reactor water is performed to check the tritium concentration and the gross alpha and beta activity; an outside contractor normally performs this monitoring. Instruments shall be calibrated at least annually (not to exceed 15 months) in accordance with facility calibration procedures.

#### 11.1.7.1 Portable Survey Meters

The reactor facility is equipped with a variety of portable monitoring instruments, including Geiger counters, ionization chambers, and scintillation detectors. Sufficient instruments are available for detecting gamma, beta, alpha, and neutron radiation. The neutron-sensitive instruments are capable of detecting both fast and slow neutrons. Gamma-sensitive instruments are available with a top range of at least 5 R/hr.

#### 11.1.7.2 Fixed Area Monitors

Five gamma-sensitive fixed area monitors are located in the reactor facility. All of these monitors are normally operational; however, the minimum requirement is that at least one monitor be operating in the reactor bay (Room 149) if the reactor is being operated. The fixed area monitors have both local and remote meter indications and alarms. These monitors interface with the Building 15 evacuation alarm and with FPS. Further details of these functions can be found in the Emergency Plan and the Physical Security Plan.

Table 11.4: Emergency Response Dose Guidance

| Dose Limit | Activity Causing Exposure | Condition |
|---|---|---|
| 5 rem | routine operations | occupational |
| 10 rem | protecting valuable property | lower dose not practical |
| 25 rem | life saving or protection of large populations | lower dose not practical |
| >25 rem | life saving or protection of large populations | volunteers only; person fully aware of risks. |

#### 11.1.7.3 Airborne Radioactivity Monitors

Two airborne radioactivity monitors are provided in the facility. A CAM is required to be operating if the reactor is operating. This instrument pumps air from the reactor bay through paper and charcoal filters and monitors the filters for any radioactive material collected. The CAM has both local and remote meter indications. The CAM also interfaces with the Building 15 evacuation alarm and FPS. Further details of these functions can be found in the Emergency Plan and the Physical Security Plan.

An $^{41}$Ar detector system is also provided to allow determination of both the $^{41}$Ar concentration in the reactor bay air and the total release of $^{41}$Ar to the environment. This system samples reactor bay exhaust air and determines the $^{41}$Ar concentration through use of a single-channel analyzer or a multi-channel analyzer. Experience has shown that the $^{41}$Ar produced and released is highly correlated with the number of megawatt hours of reactor operation. If the $^{41}$Ar instrument is not operational, the $^{41}$Ar levels may be estimated accordingly. Both local and remote indications are provided by the $^{41}$Ar detector system.

### 11.1.8 Emergency Response and Exposure Guidelines

Response to radiological emergencies at the GSTR shall be in accordance with the facility Emergency Plan. Dose limits for workers performing emergency services shall be as given in the Table 11.4. Any person undertaking any emergency operation in which the dose will exceed a total effective dose equivalent of 25 rem should do so only on a voluntary basis and with full awareness of the risks involved. In general, persons for whom radiation exposure presents a lesser risk should be used for emergency operations. Age, lifetime dose, total effective dose equivalent and childbearing status should be taken into account when choosing workers to receive emergency exposures.

### 11.1.9 Declared Pregnant Woman Guidelines

A declared pregnant woman is a woman who has voluntarily informed the reactor health physicist, in writing, of her pregnancy and the estimated date of conception. The occupational exposure of a declared pregnant woman shall be controlled to less than 0.5 rem during the entire pregnancy period.

All staff who will be occupationally exposed to radiation at the reactor facility will be informed of the

definition of a declared pregnant woman and the reduced exposure limit for declared pregnant women.

### 11.1.10 Planned Special Exposures

Planned Special Exposures are not intended to be used at the GSTR facility.

## 11.2 Radioactive Waste Management

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 Health Physics 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.

### 11.2.1 Radioactive Waste Storage

Liquid and solid radioactive wastes are collected and stored in restricted areas or kept under the control of reactor staff members. Disposal of solid radioactive waste from the reactor facility is made through licensed waste disposal facilities. Collection, packaging and labeling of wastes are in accordance with Nuclear Regulatory Commission and Department of Transportation regulations. The Reactor Health Physicist is responsible for proper waste packaging, labeling, disposal and record maintenance.

### 11.2.2 Radioactive Waste Control

At the GSTR, radioactive waste is generally considered to be any item or substance which is no longer of use to the facility and which contains, or is suspected of containing, radioactivity above natural background radioactivity. Equipment and components are categorized as waste by the GSTR staff. When possible, radioactive waste is initially segregated at the point of origin from items that will not be considered waste.

### 11.2.3 Solid Waste

As with most non-power reactors, solid waste is generated from reactor maintenance operations and irradiations of various experiments. The amount of solid waste is generally on the order of 4 cubic feet per year. No solid radioactive waste is intended to be retained or permanently stored on site. Appropriate radiation monitoring instrumentation will be used for identifying solid radioactive waste. Radioactive waste is packaged in metal drums within the GSTR facility. The waste is disposed of properly through a waste disposal broker.

### 11.2.4 Liquid Waste

It is the GSTR's policy not to routinely release radioactive liquid waste. Normal operations of the GSTR do not produce liquid radioactive waste, and if so, the liquid waste is contained locally and disposed of properly.

# References

[1] Henderson, W. J. and P.R. Tunnicliffe, "The production of $^{16}N$ and N-17 in the Cooling Water of the NRX Reactor," NSE 1958 pp.145-150.

[2] Mittl, R. L., and M. H. Theys, "$^{16}N$ Concentrations in EBWR," Nucleonics, March 1961, p.81.

# Chapter 12

# CONDUCT OF OPERATIONS

This chapter describes and discusses the Conduct of Operations at the GSTR. 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 GSTR is the US Geological Survey. However, the Reactor Supervisor is responsible for licensing and reporting information to the NRC. The Reactor Administrator is informed of license issues directly from the Reactor Supervisor.

### 12.1.1 Structure

The management organization of the GSTR, as shown in Figure 12.1, is structured to provide comprehensive and redundant internal oversight of reactor operations and radiation protection programs. It also meets the intent of ANSI/ANS 15.11, Radiation Protection at Research Reactor Facilities. Interaction between the Health Physicists, Reactor Supervisor, and Reactor Operators is constant, although the reporting lines may be separate.

### 12.1.2 Responsibility

The following is a list of various offices/committees/personnel and their associated duties:

- Director, US Geological Survey - chief executive officer of the USGS;
- Reactor Administrator - the Reactor Administrator is responsible for and in charge of the reactor facility. This responsibility includes establishing budgets and policy, reporting to the NRC and accepting responsibility for the safe and legal operation of the facility. The Reactor Administrator has the prime

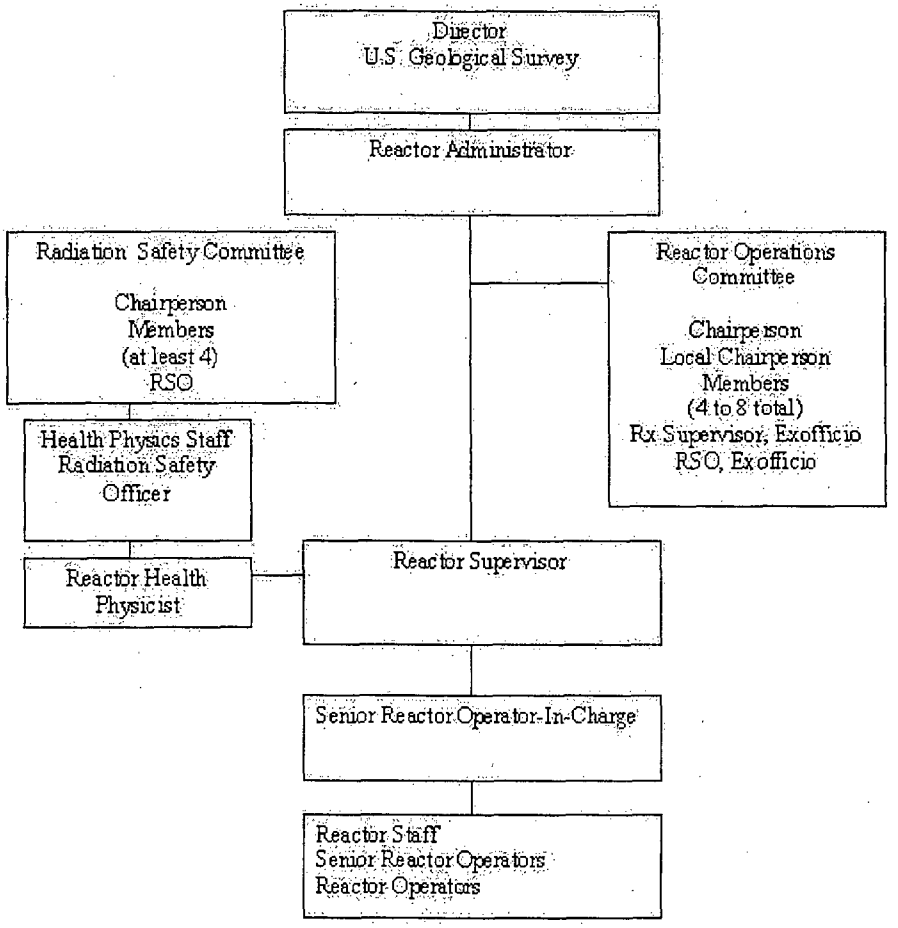


Figure 12.1: GSTR Administrative Organizational and Reporting Line Chart

responsibility for the control of Special Nuclear Material associated with the operation of the reactor. The Reactor Supervisor is directly responsible to the Reactor Administrator.

The Reactor Administrator is guided by the Radiation Safety Officer and the Reactor Operations Committee. The Reactor Health Physicist furnishes advice on the radiological safety of all operations connected with the reactor facility to the Reactor Supervisor and the Health Physicist. The Reactor Operations Committee reviews the safety of the reactor and associated experiments.

- Reactor Operations Committee - this standing committee independently reviews, evaluates, and approves safety standards associated with the operation and use of the US Geological TRIGA® Reactor and its experimental facilities;

- Reactor Supervisor - the Reactor Administrator has delegated authority for all safety and operational aspects of the facility to the Reactor Supervisor. The Reactor Supervisor has been delegated responsibility for the control of Special Nuclear Material associated with the operation of the reactor. The

Reactor Supervisor determines schedules, reviews and classifies all experiments, and works in close liaison with the Reactor Health Physicist and the Reactor Operations Committee.

The Reactor Supervisor is a licensed Senior Reactor Operator, directs the Reactor Operator Training and Requalification Program, is in administrative charge of all operators and technicians and is directly responsible for the maintenance of reactor records and physical inventories.

The Reactor Supervisor has the responsibility to report immediately, or cause to be reported, to the NRC in writing any indication or occurrence of a possible unsafe condition relating to the reactor. This action does not require concurrence of the Committee. However, all Committee members will be advised of the action. All actions of this nature will also be reported to the Reactor Administrator. The Committee may advise on other matters which are brought before it by the Reactor Administrator or the Reactor Supervisor.

- Radiation Safety Committee - the Radiation Safety Committee advises on the radiological safety of any operation of the Geological Survey at the Denver Federal Center involving radioactive materials. The Committee enforces regulations which are a part of the Survey's NRC Broad License. The Committee is available for consultation whenever the RSO or Reactor Supervisor deems it necessary.

- Radiation Safety Officer - the Radiation Safety Officer (RSO) is a permanent member of the Radiation Safety Committee and acts as its Executive Officer. The RSO is responsible for supplying the common services necessary for a safe radiation program within the Denver U.S. Geological Survey and its associated field projects.

  The RSO maintains a central file of records on the use, storage, and disposal of radioactive materials, including radiation inventory records, personnel exposure records, etc.

- Health Physicist - the Reactor Health Physicist (Rx HP) derives radiological authority from the Radiation Safety Committee through the RSO, and the reactor license. Guidance is provided by from the RSO and Reactor Supervisor. Work is performed under the general supervision of the Reactor Supervisor. The Rx HP advises the Reactor Supervisor on operations in the facility and is responsible for radiation safety in all reactor facility spaces. The Rx HP supervises the transfer of all radioactive materials from the reactor facility and is responsible, along with the Reactor Supervisor, for the control of contamination in the reactor area and the proper storage and labeling of all radioactive material and sources. The Rx HP is also responsible for the operation, maintenance and calibration of all the health physics monitoring equipment used by the reactor facility and for the disposal of radioactive waste generated in the reactor facility in accordance with applicable DOT and NRC regulations.

- Reactor Operator - 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. All operators will be licensed by the U.S. Nuclear Regulatory Commission to operate the GSTR. Within the constraints of administrative and supervisory control, a Senior Operator or Operator will be in direct charge of the control console at all times the reactor is operating. Licensed Senior Operators will be delegated the responsibility for the direction of reactor operations, including

control manipulations, reactor maintenance and calibration. They assist the Reactor Supervisor in the supervision and instruction of Reactor Operators.

### 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.

At least two reactor staff members must be on duty whenever the reactor is operating. A Senior Reactor Operator-in-charge will be designated for each reactor operation. The operator at the console shall be an NRC licensed operator and the second person must be able to reach the facility within 15 minutes after being contacted by the console operator. A method of communication must be available so that the operator at the console is able to contact the second person within a 5 minute time period. The Senior Reactor Operator-in-charge of the operation must be one of the two staff members on duty and must be at the facility during reactor start-up, approach to power, recovery from unplanned scram, recovery from unplanned shutdown, recovery from an unscheduled significant power reduction, relocation of any in-core components (other than normal control rod movements) with a reactivity worth greater than one dollar, or fuel movement. The manipulation of the controls of the reactor must be done by a NRC licensed operator or by an individual who is manipulating the controls as part of their training for an operator's license at the facility or their training as a student.

### 12.1.4 Selection and Training of Personnel

The training and qualifications contained in American Nuclear Society 15.4 - Standards for Selection and Training of Personnel for Research Reactors are considered the minimum for personnel at the U.S. Geological Survey TRIGA® Reactor Facility.

#### 12.1.4.1 General Training for Building 15 Personnel

Training for all personnel within Building 15 who work with radioactive materials is specifically discussed in Chapter 11.

#### 12.1.4.2 Initial Reactor Operator Training

For initial training for either a Reactor Operator or a Senior Reactor Operator, the GSTR Operations Manual chapters 1-11 shall be studied and knowledge demonstrated.

### 12.1.5 Radiation Safety

This is covered in Chapter 11.

## 12.2 Review and Audit Activities

The Reactor Operations Committee shall review and approve safety standards associated with the operation and use of the reactor facility. Its jurisdiction shall include all nuclear operations in the facility. The Committee is concerned with those experiments which, by their unusual nature, inherent hazard, or unprecedented complexity, could endanger health, life and property in and about the Nuclear Reactor Facility. These experiments are Class II.

It is not the purpose of the Committee to evaluate or consider conventional administrative or operational functions at the GSTR. Should a particular reactor-oriented experiment contain unusual dangers, which in the event of an accident could endanger the general public or the health, life, and property in and about Building 15, this experiment must be submitted to the Committee for evaluation. Approval[†] of an experiment by the Committee, however, does not obligate the reactor staff to carry it out. Generally, any experiment for which adequate precedent already exists to guarantee safety will not be referred to the Committee. Such experiments are Class I. The day-to-day in-core experiments proceed by approval of the Reactor Supervisor. Questionable experiments are referred by the Reactor Supervisor to the Reactor Operations Committee for approval. Records are kept by the reactor staff listing all in-core experiments for periodic review by the entire Committee to insure that the intent and function of the Operations Committee is being maintained.

The Committee aids the reactor staff with the review and evaluation of any indications that changes are taking place in the core which influence its integrity.

If a situation exists in the reactor which indicates a possibility of unsafe operation, the Committee, by a majority vote, can recommend that the reactor be shutdown. The reactor staff retains the authority to shutdown the reactor without such a recommendation. In the event of a difference of opinion between the staff and Committee, the Committee recommendation to shutdown will govern. The Reactor Administrator will refer immediately to the Office of the Director, U.S. Geological Survey, for appropriate action. Startup after shutdown recommended by the Committee shall have the Committee's approval prior to such startup.

### 12.2.1 Composition and Qualifications

The Reactor Operations Committee shall be composed of at least four and not more than eight voting members, including the Chairman. The Committee shall consist of persons of high-level competency in the fields of physics, chemistry, health and safety or other subject matter related to nuclear reactor operations. At least two and preferably three, of the Committee members will not be present employees of the Department of the Interior and will be drawn primarily from the academic community.

To expedite Committee business, a Committee Chairman and an alternate have been appointed. The Chairman and the Alternate Chairman of the Reactor Operations Committee are listed by name on the Reactor Operations Committee roster.

[†] "Approval" as used in this section shall mean approval for substantive issues, including Class II Experiments, 10 CFR 50.59 changes, license amendments or any other matters designated by the Reactor Administrator, the Reactor Supervisor, or the Reactor Operations Committee. Minor issues will continue to be handled by the reactor staff and the Reactor Supervisor as described elsewhere in the Reactor Operations Manual.

The Committee is appointed by the USGS Director. No definite term of service is specified; but should a vacancy occur in the Committee, the Director will appoint a replacement. The remaining members of the Committee will be available to assist the Director in the selection of new members.

The Reactor Supervisor and the Central Region Radiation Safety Officer are ex officio members of the Committee and the Reactor Supervisor is the only non-voting member of the Committee.

### 12.2.2 Charter and Rules

The Reactor Operations Committee consists of Survey members and non-Survey members, and as specified in the Technical Specifications, the Committee must meet at least semi-annually, not to exceed 7.5 months between meetings. Normally the non-Survey members will attend only every other semi-annual meeting. These full Committee meetings are generally scheduled each spring. Special meetings for the review of specific problems or urgent Class II Experiments will be called whenever necessary.

Criteria have been established for the conduct of the meetings as specified in the Technical Specifications.

A quorum for review, audit, and approval purposes shall consist of not less than one-half of the committee membership, provided that the operating staff does not constitute a majority of the committee membership. The Chairperson or an alternate must be present at all meetings in which the official business of the committee is being conducted. Approvals by the committee shall require an affirmative vote by a majority of the non-Survey members present and an affirmative vote by a majority of the Survey members present.

### 12.2.3 Details of Meetings and Committee Communications

This is covered in the GSTR Technical Specifications.

### 12.2.4 Inactive Status

This is covered in the GSTR Technical Specifications.

### 12.2.5 Review Function

The responsibilities of the Reactor Operations Committee (ROC), or a designated subcommittee thereof, shall include, but are not limited to, the following:

- review and approval of experiments utilizing the reactor facilities;
- review and approval of all changes to the safety analysis report or technical specifications;
- review and approval of all new procedures and substantive changes to existing procedures;
- review all changes to the facility or safety evaluations under 10 CFR 50.59;
- review of the operation and operational records of the facility;

- review of abnormal performance of plant equipment and operating anomalies; and
- review of all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours.

### 12.2.6 Audit Function

The ROC, or a designated subcommittee thereof, shall audit the reactor operations and health physics programs annually. The audit shall include, but not be limited to, the following:

- inspection of reactor operating records;
- inspection of the reactor operating area; and
- reportable occurrences.

## 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 approved by the ROC. The procedures are reviewed by the ROC to ensure that they are appropriate. Minor changes that are needed on short notice may be made by the Reactor Supervisor, with final approvals to be gained from the Health Physicist (for radiation safety items) and Reactor Operations Committee. Any substantive changes to the procedures must be approved by the Reactor Supervisor, the Health Physicist (for radiation safety items) and the Reactor Operations Committee prior to implementation. 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 following two sections list areas that will typically require written procedures.

### 12.3.1 Reactor Operations

The following activities will typically require written procedures:

- Testing and calibration of reactor operating instrumentation and control systems and control rod drives;
- Reactor startup, routine operation, and reactor shutdown;
- Emergency and abnormal conditions, including evacuation, reentry and recovery;
- Fuel loading or unloading;
- Control rod removal and replacement;
- Maintenance operations which may affect reactor safety;

### 12.3.2 Health Physics

The following activities will typically require written procedures:

- Testing and calibration of area radiation monitors and air particulate monitors.

## 12.4 Required Actions

This is covered in the GSTR Technical Specifications.

## 12.5 Reports

This is covered in the GSTR Technical Specifications.

## 12.6 Records

This is covered in the GSTR Technical Specifications and Chapter 11.

## 12.7 Emergency Planning

The GSTR Emergency Response Plan contains detailed information concerning the GSTR response to emergency situations. The GSTR 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 GSTR. Detailed implementing procedures are referenced in this plan. This approach provides the GSTR emergency staff the flexibility to cope with a wide range of emergency situations without requiring frequent revisions to the plan.

The Reactor Supervisor is responsible for emergency preparedness planning for the GSTR facility, including updating of emergency plans and procedures and coordinating plans with other organizations. The Reactor Supervisor is also responsible for coordinating the activities of the radiological assistance teams. Implementation of the plan on a day-to-day basis is the responsibility of the Reactor Supervisor. 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 GSTR Physical Security Plan contains detailed information concerning the GSTR security measures. The plan provides the GSTR 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 Reactor Supervisor. Implementation of the plan on a day-to-day basis during hours of operations is also the responsibility of the Reactor Supervisor.

## 12.9 Quality Assurance

Quality assurance can be found throughout the operating and health physics procedures. It is not called out as a separate document.

## 12.10 Operator Training and Requalification

This Reactor Operator training and requalification program is designed to satisfy the requirements of the NRC's rules contained in 10 CFR 55. It also generally complies with the requalification program in ANSI/ANS 15.4, Selection and Training of Personnel for Research Reactors. The Reactor Supervisor is responsible for the requalification exams.

The requalification training program was established to maintain the proficiency of the operating organization through training exercises, instruction periods, and reviews covering those items and equipment which relate to the safe operation of the facility. While the program will be continuous in nature, certain benchmarks have been established to aid in the administration of the requalification plan. They are:

1. Annual written examinations,
2. Annual operating examinations and
3. Stipulated minimum tasks which must be performed during the requalification period of two years (not to exceed 30 months).

### 12.10.1 Written Examination

An annual (not to exceed 15 months) written examination, designed to demonstrate competence in accordance with the guidance given in 10 CFR 55.41 or 55.43 will be given to all NRC licensed Operators and Senior Operators. The corrected written annual examination will document competence and hence will be preserved for each licensed Senior Reactor Operator and Reactor Operator. A successful repeat written examination for those areas of inadequacy will document the sufficiency of retraining.

Table 12.1: Subject areas for RO and SRO requalification exams

| Column 1 | Column 2 |
|---|---|
| Reactor Theory Fundamentals | Facility License |
| General Design Features | Technical Specifications |
| Primary System Features | Radiation Hazards |
| Secondary System Features | Fuel Movement |
| Operating Characteristics | Assessment of Normal, Abnormal, |
| Reactivity Controls & Insts. | and Emergency Conditions. |
| Control & Safety Systems | |
| Facility Procedures | |
| Radiation Monitoring | |
| Radioactive Material Handling | |

Since the annual written examination will be prepared and corrected by the Reactor Supervisor who is also required to hold a Senior Operator License, these functions will be deemed adequate to demonstrate and document continuing competence.

The subject areas in column 1 of Table 12.1 will be sampled by the Reactor Operators' written examination and the subject areas in columns 1 and 2 will be sampled by the Senior Reactor Operators' written examination.

### 12.10.2 Operating Examination

An annual (not to exceed 15 months) operating examination, designed to evaluate operating competence, will be given to all NRC licensed Senior Reactor Operators and Reactor Operators. Any weakness shown by the operating examination shall be the focus of subsequent retraining.

A memorandum will be written and retained to document operating examinations and any related retraining activities. Operating examination tasks will typically be selected from activities such as reactor checkouts, reactor startups, major power changes, manual or automatic operations, square wave operations, pulse operations, power calibrations, or control rod calibrations. The operating examinations may be conducted during normal experiment operations and will be administered by the Reactor Supervisor. These functions will be deemed adequate to demonstrate and document continuing operational competence of the Reactor Supervisor.

### 12.10.3 Reactor Operations - On-the-Job-Training

Each Senior Reactor Operator and Reactor Operator will be required to perform certain minimum tasks during the requalification period (2 years, not to exceed 30 months) to insure continued competence in the areas listed in 10 CFR 55.59. These minimums are listed in Table 12.2.

Table 12.2: Minimum tasks required during requalification period

| | | Reactor Operator | Senior Reactor Operator |
|---|---|---|---|
| a. | Reactivity Control Manipulations | 10 | 10 |
| b. | Daily Startup and Shutdown Checklists | 10 | 10 |
| c. | Participate in Monthly Checklists | 1 | 2 |
| d. | Participate in Pulse or Square Wave Operations | 1 | 1 |
| e. | Participate in Annual Emergency Drills | 1 | 1 |
| f. | Review all (1) facility design changes (2) procedure changes (3) facility license changes | within 20 days of issuance | within 20 days of issuance |
| g. | Review emergency procedures | 2 | 2 |

### 12.10.4 Evaluation

The examinations will be graded by the Reactor Supervisor. The following written exam criteria will be used:

1. A grade of 80% or above on each section indicates satisfactory competence. No remedial action is required.

2. A grade below 80% on any section signifies weakness in that area; therefore, remedial work and/or lectures on the deficient area will be required.

3. An overall grade of less than 70% on the annual written examination will require the operator or Senior Operator be removed from licensed duties and placed on an accelerated training program until satisfactory performance is achieved.

The Reactor Supervisor will routinely observe and evaluate the performance of licensed operators and Senior Operators. This evaluation will include their actions during actual or simulated abnormal and emergency conditions. If an unsatisfactory performance is determined as a result of the Reactor Supervisor's evaluation, the licensed operator or Senior Operator will be removed from licensed duties and be placed on an accelerated training program. Licensed duties will not be resumed until satisfactory performance has been demonstrated by the operator.

### 12.10.5 Records

The corrected annual examination will be maintained in the Reactor Operator/Senior Reactor Operator License File. Documentation of an Operator/Senior Reactor Operator's On-The-Job Training will be as follows:

1. Table 12.2, Parts a, b, c, d, and e will be contained within normal facility operating records.

2. Table 12.2, Part f: A routing sheet will be used with all facility design changes, procedure changes and facility license changes which will require the initials of all licensed Operators and Senior Operators, certifying their review and understanding of the change. A copy of this routing sheet will be maintained in the Operator Requalification Program file.

3. Table 12.2, Part g: A review and critique sheet will be prepared by the Reactor Supervisor, normally in the same time frame with scheduled emergency drills, to document that each Reactor Operator and Senior Operator has reviewed the contents of the Reactor Operations Manual, Section 7, Emergency Procedures and is aware of any special problems or deficiencies noted in the last drill. A copy of this review sheet will be maintained in the Operators Requalification Program file.

4. An annual statement by the Reactor Supervisor will be inserted in each Reactor Operators or Senior Reactor Operators file certifying continuing competency based on the (1) successful completion of the annual examinations plus any remedial work if required and (2) the actual performance of licensed duties of Table 12.2.

### 12.10.6 Inactive Status

When a licensed operator or Senior Operator has not actively performed licensed functions for a duration of at least four hours in any calendar quarter, he/she shall perform licensed functions for a minimum of six hours under the direction of a licensed Senior Reactor Operator, including at least one startup and shutdown checklist. A letter shall be sent to the NRC certifying this on-the-job training and that the qualifications and status of the licensee are valid. Resumption of unsupervised operation will not occur until after this training and certification have been performed.

## 12.11 Startup Plan

This is not applicable.

## 12.12 Environmental Reports

The staff has determined that this renewal application involves no significant hazards consideration, no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this renewal application meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this renewal application.

# References

[1] Hazards Summary Report for a 1,000 KW Mk-I Reactor at the Geological Survey, Denver, Colorado, December 1966.

[2] Technical Specifications GSTR

[3] Radiation Safety Committee - Radiation Protection Procedures, 2004.

[4] Standards for Selection and Training of Personnel for Research Reactors, American Nuclear Society 15.4.

# Chapter 13

# ACCIDENT ANALYSIS

## 13.1 Introduction

In about 1980, the U.S. Nuclear Regulatory Commission requested an independent and new 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®-fueled Reactors. 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 GSTR at a power level of 1 MW are established in Chapter 4. In this chapter, it was assumed that three different TRIGA® fuel types might be used in the GSTR: 8.5 wt% stainless steel-clad fuel with 20% enrichment, 8.0 wt% aluminum-clad fuel with 20% enrichment, and 12.0 wt% stainless steel-clad fuel with 20% enrichment.

The fuel temperature is a limit on operation in both steady-state and pulse modes. This limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element cladding material. The strength of the stainless steel cladding as a function of temperature sets the upper limit on the fuel temperature for stainless steel clad fuel. Phase stability of the ZrH matrix sets the upper limit on the fuel temperature for aluminum clad fuel. Fuel temperature limits of 1,100 °C for stainless steel clad fuel and 535 °C for aluminum clad fuel have been set to preclude the loss of clad integrity.

Nine credible accidents for research reactors were identified in NUREG-1537 as follows:

- MHA;
- Insertion of excess reactivity;
- Loss of coolant accident (LOCA);
- Loss of coolant flow;
- Mishandling or malfunction of fuel;

- Experiment malfunction;
- Loss of normal electrical power;
- External events; and
- Mishandling or malfunction of equipment.

This chapter contains analyses of postulated accidents that have been categorized into one of the above nine groups. Some categories contain accidents that do not appear applicable or credible for the GSTR, but this is acknowledged in a brief discussion of the category. Some categories contain analyses of more than one accident even though one is usually limiting in terms of impact. Any accident having significant radiological consequences was included.

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)

#### 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 GSTR, the MHA has been determined to be 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 the GSTR, with three different possible fuel types, a 12 wt% fuel element was chosen as the irradiated element since it contains the most $^{235}U$ and, hence, the highest inventory of fission products. The failed fuel element was assumed to have been operated at the highest core power density for a continuous period of one year at 1 MW. This results in all of the halogens and noble gases (except $^{85}Kr$) reaching their saturated activities.

This is the most severe accident for a TRIGA® reactor and is analyzed to determine the limiting or bounding potential radiation doses to the reactor staff and to the general public in the unrestricted area. A less severe,

but more credible accident, involving this same single element having a cladding failure in water will also be analyzed. This latter accident more correctly falls into the mishandling or malfunction of fuel accident category and will be addressed there.

During the lifetime of the GSTR, used fuel within the core may be moved to new positions or removed from the reactor. Fuel elements are moved only during periods when the reactor is in a shutdown condition. Also, the GSTR is very rarely operated continuously at 1 MW for a period longer than 12 hours, let alone a period of one year. Nevertheless, this MHA has been analyzed for the GSTR.

The following scenario has been chosen for analysis:

- This scenario assumes that the noble gas and halogen fission products instantly and uniformly mix with the reactor bay air. The fission products that have been released to the reactor bay air are then exhausted at the stack ventilation rate of 800 cfm ($3.78 \times 10^5$ cm$^3$/sec). The path for this release is assumed to be through the emergency exhaust stack. The air is assumed to be discharged at 21 feet above ground, at the exit of the exhaust stack. The reactor bay free volume is assumed to be 12,500 cubic feet. The exhaust system takes 15.6 minutes to expel one reactor bay volume of air (3.84 room changes per hour). The time to discharge 95% of the fission product gases from the reactor bay is 47 minutes, but this analysis conservatively assumes that all fission product gases are released instantaneously in a single pulse discharge. Similarly, it is conservatively assumed that the gas concentration in the reactor bay undergoes no dilution during the maximum assumed stay time of 5 minutes.

#### 13.2.1.2 Accident Analysis and Determination of Consequences

As stated earlier, it is assumed that the GSTR is fueled with 12 wt% fuel elements, 100 fuel elements in the core, and that the reactor has operated continuously at 1 MW for a period of one year. Thus, all halogens and all noble gases (except $^{85}$Kr) are at their saturation activity. The highest-power density fuel element fails and releases the noble gases and halogens to the cladding gap. This highest-power-density element has a conservative power density of 22 kW. The fission product inventory of halogen and noble gases are given in Table 13.1 for this element. The inventory assumes a saturated activity is present and is based upon the fission yield for each isotope.

Considerable effort has been expended to measure and define the fission product release fractions for TRIGA® fuels. Data on this aspect of fuel performance are reported. Using these data, GA developed a conservative correlation for fission product release to be

$$e = 1.5 \times 10^{-5} + 3.6 \times 10^{3} \exp\left\{\frac{-1.34 \times 10^{4}}{\mathrm{T} + 273}\right\}. \tag{13.1}$$

At an average fuel temperature of 350 °C, this release fraction is $1.66 \times 10^{-5}$. This fuel temperature (350 °C) is much higher than the actual expected average fuel temperature of 240 °C, and results in a release fraction about 11% higher.

Once the fission products are released to the gap, this activity is released when the cladding fails. If the

release is in air (MHA), then this activity is released directly into the reactor bay air. If the release occurs in the pool water, then the fission products must migrate through the water before being released to the reactor bay air. Once released into the reactor bay air, a further reduction of the halogen activity is expected to occur due to plateout on the surfaces of the bay.

Thus, the fraction ($w$) of the fission product inventory released from a single fuel element which reaches the reactor bay air and, subsequently, the atmosphere in the unrestricted environment is:

$$w = efgh, \tag{13.2}$$

where:

$e$ = the fraction released from the fuel to the fuel-cladding gap;

$f$ = the fraction released from the fuel-cladding gap to the reactor bay air (if no water is present), or to the pool water (if water is present);

$g$ = the fraction released from the pool water to the reactor bay air ($g$=1.0 when no water is in the pool); and

$h$ = the fraction released from the reactor bay air to the outside unrestricted environment, due to plateout in the reactor bay.

For the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor bay to the outside environment. This value is based on historical usage and recommendations, where other sources recommend a 50% release of the halogens from the gap to the air. Further references apply a natural reduction factor of 50% due to plateout in the reactor bay. Combining the 50% release from the gap with the 50% plateout results in the 25% total release. However, this value appears to be quite conservative, as some references quote a 1.7% release from the gap rather than 50%.

For the accident in air, 100% of the noble gases are assumed to be available for release to the unrestricted environment.

For the accident in water, it is assumed that most of the halogens released from the cladding gap remain in the water and are removed by the demineralizer. A small fraction, 5%, of the halogens is assumed to escape from the water to the reactor bay air. Combining this with the 50% release from the gap to the water, the result is that 2.5% of the halogens in the gap are released to the reactor bay. Again, 50% of these plateout in the reactor bay before release to the outside environment. For the noble gases in water, 100% are assumed to be available for release to the unrestricted environment.

The experience at Three Mile Island, along with recent experiments, indicate that the 50% halogen release fraction is much too large. Possibly as little at 0.06% of the iodine reaching the cladding gap may be released into the reactor bay due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble that elemental iodine.

The values for these various release fractions are given in Tables 13.2 and 13.3.

Table 13.1: Saturated Activities for Highest Power Density 12 wt% Fuel Element

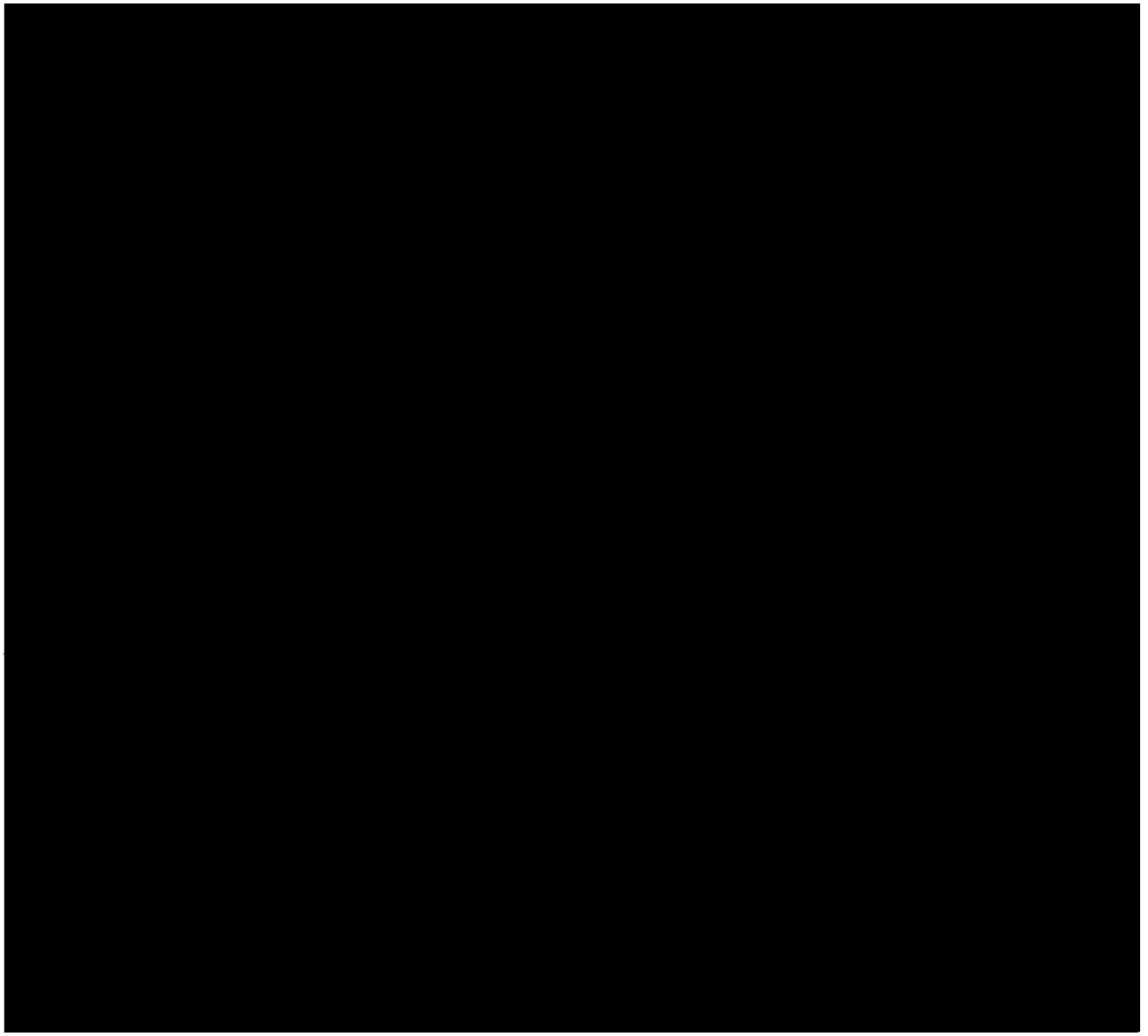

Table 13.2: Release Fraction Components

| **Fission product** | $f$ **No pool water** | $f$ **With pool water** | $g$ **No pool water** | $g$ **With pool water** | $h$ |
|---|---|---|---|---|---|
| Noble gas | 1.0 | 1.0 | N/A | 1.0 | 1.0 |
| Halogens | 0.5 | 0.5 | N/A | 0.5 | 0.5 |

Table 13.3: Total Release Fraction

| Fission product | $w$ No pool water | $w$ With pool water |
|---|---|---|
| Noble gas | $1.66 \times 10^{-5}$ | $1.66 \times 10^{-5}$ |
| Halogens | $4.15 \times 10^{-6}$ | $2.08 \times 10^{-6}$ |

For the GSTR, the prevailing wind is from the west, blowing to the east. The minimum distance to the unrestricted environment (350 m) is to the east, the minimum distance to the nearest public residence (640 m) is to the north, and a public school is about 720 m to the east. For this accident, therefore, it was assumed that the wind is blowing from west to east and all recipients are east.

The methodology of the DOE HOTSPOT computer code for areas outside of the reactor bay and uniform dispersion with 10 CFR 20 Appendix B conversion factors was used. The results of these analyses are provided in Table 13.6.

Furthermore, for calculations beyond the reactor bay, it was assumed that all of the fission products were released to the unrestricted area by a discharge pulse, which would maximize the dose rate to persons exposed to the plume during the accident. Calculations inside the reactor bay assumed uniform distribution of the released fission products within the $> 3.1 \times 10^8$ cc volume of the bay.

It was also assumed that the receptor breathing rate was $3.33 \times 10^{-4}$ $m^3$/sec (NRC "light work" rate) and that the longest isotope retention category was applicable.

Calculations for personnel inside the reactor bay assumed that all of the fission product gases released were instantly and uniformly distributed within the reactor bay. The exposures for personnel in the reactor bay for short stay-times (up to 5 minutes)was calculated by conservatively assuming that the fission product concentration is constant for that time period. The isotope concentrations in terms of DAC values and DAC-hr exposures during a 2-minute stay time are given in Table 13.4 below. Values for 5 minute stay times are 2.5 times higher than the 2 minute stay time values since the fission product gas concentration is assumed to be constant during this exposure period.

Since a non-stochastic exposure of 1 Annual Limit of Intake (ALI) gives a Committed Dose Equivalent (CDE) of 50,000 mrem for the target organ (thyroid for radioiodine) and a stochastic exposure of 2000 DAC-Hr results in a TEDE of 5000 mrem, the $CDE_{Thyroid}$ and TEDE for 2-minute and 5-minute stay times in the reactor bay are shown in Table 13.5.

The results of the HOTSPOT calculations for the two scenarios (no water vs water in reactor tank) are shown in Table 13.6. As seen from the tables, no water in the reactor gives the highest doses to the general public at any distance, as is expected since there is no capture of fission products by the water. The scenario with water in the reactor tank gives the lowest doses at any given distance since the capture and retention of fission products in the water is significant. Also, as expected, the doses were highest at the shortest distance (i.e., near the reactor bay). As shown in Table 13.6, in all cases doses for the general public and occupational workers were all well below the annual dose limits specified by 10 CFR 20.

Table 13.4: Concentrations and Exposures from Gaseous Fission Product Releases

Table 13.5: Occupational $CDE_{Thyroid}$ and TEDE in the Reactor Bay Following a Single Element Failure in Air and Water

| Reactor Bay Occupancy (minutes) | $CDE_{Thyroid}$ (no water) (mrem) | TEDE (no water) (mrem) | $CDE_{Thyroid}$ (water) (mrem) | TEDE (water) (mrem) |
|---|---|---|---|---|
| 2 | 2800 | 180 | 1400 | 140 |
| 5 | 6900 | 460 | 3500 | 350 |

Table 13.6: Radiation Doses to Members of the General Public Following a Single Element Failure

| Distance (m) | $CDE_{Thyroid}$ (no water) (mrem) | TEDE (no water) (mrem) | $CDE_{Thyroid}$ (water) (mrem) | TEDE (water) (mrem) |
|---|---|---|---|---|
| 10 | 180 | 7.1 | 10 | 1.3 |
| 50 | 15 | 0.59 | 0.84 | 0.11 |
| 100 | 3.8 | 0.15 | 0.21 | 0.027 |
| 150 | 1.7 | 0.065 | 0.092 | 0.012 |
| 200 | 0.94 | 0.037 | 0.051 | 0.0065 |
| 250 | 0.60 | 0.023 | 0.033 | 0.0041 |
| 300 | 0.41 | 0.016 | 0.023 | 0.0028 |
| 350 (fence) | 0.30 | 0.012 | 0.017 | 0.0021 |
| 640 (residence) | 0.091 | 0.0035 | 0.005 | 0.00058 |
| 720 (school) | 0.072 | 0.0028 | 0.0039 | 0.00045 |

## 13.2.2 Insertion of Excess Reactivity

### 13.2.2.1 Accident Initiating Events and Scenarios

The most credible generic accident is the inadvertent rapid insertion of positive reactivity that could, if large enough, produce a transient resulting in fuel overheating and a possible breach of cladding integrity. Operator error or failure of the automatic power level control system could cause such an event due to the uncontrolled withdrawal of a single control rod, or possibly even the withdrawal of more than one control rod. In a separate scenario, a large reactivity insertion was postulated to create fuel cladding temperatures that might cause a metal-water reaction, but for many reasons this accident is not considered to be a safety risk in TRIGA® reactors.

### 13.2.2.2 Accident Analysis and Determination of Consequences

#### 13.2.2.2.1 Maximum Reactivity Insertion

Raising the temperature of TRIGA® fuel has a strong, prompt negative reactivity effect, which can overcome a rapid reactivity insertion such as that produced by the rapid removal of the transient rod. The quantity that causes this effect is the prompt negative temperature coefficient of the fuel, discussed in Chapter 4. There is a limit to the protection provided by this feedback, since the peak fuel temperature attained

increases with the magnitude of the inserted reactivity. The Nordheim-Fuchs model was used to compute the maximum reactivity pulse that can occur without exceeding the safety limit on fuel temperature established in Chapter 4. In the Nordheim-Fuchs model, it is assumed the transient is so rapid that 1) the temperature rise is adiabatic, and 2) delayed neutrons can be neglected. Thus, the model is given by the following set of coupled differential equations:

$$\frac{\mathrm{dP}}{\mathrm{dt}} = [\rho - \beta]\frac{P}{\lambda}, \tag{13.3}$$

$$\frac{\mathrm{dT}}{dt} = \frac{\mathrm{P}}{\mathrm{C(T)}}, \text{ and} \tag{13.4}$$

$$\rho(\mathrm{T}) = \rho_0 - [\alpha \mathrm{T}]\,[\mathrm{T} - \mathrm{T_0}], \tag{13.5}$$

where

$P$ = the reactor power,

$\rho(T)$ = the time-dependent reactivity,

$l$ = the neutron lifetime,

$\lambda$ = the effective delayed neutron fraction,

$T$ = the core-average temperature,

$T_0$ = the initial temperature,

$\rho_0$ = the reactivity insertion,

$\beta$ = the temperature-dependent prompt reactivity feedback coefficient, and

$C$ = the temperature-dependent whole-core heat capacity.

The quantity of interest is $\Delta T$, the difference between the maximum and initial values of the core-average fuel temperature. From the definition of $\Delta T$, the peak fuel temperature can be found to be

$$T_{\mathrm{peak}} = T_0 + PF\,[\Delta T] = T_0 + PF\,[T - T_0] \tag{13.6}$$

where $\mathrm{T_0}$ is the initial temperature and $PF$ is the total peaking factor.

In the above equations, the desire is to find the value of $\rho_0$ that yields a $T_{\mathrm{peak}} = 1,100°\mathrm{C}$. In the Nordheim-Fuchs model, at the end of the reactivity burst, one gets the maximum fuel temperature ($T_{\mathrm{peak}}$), and the total inserted reactivity is

$$\rho = 2\,[\beta - \rho_0] \tag{13.7}$$

Thus, one gets the equation:

Table 13.7: Maximum Reactivity Insertion and Related Quantities for Various Fuels and Burnups

| Fuel type | Peak Fuel Temperature Limit (°C) | Heat Capacity for total core (watt·sec/°C) | Prompt Negative Temperature Coefficient ($\alpha$k/k °C) | Reactivity (\$) $T_{peak} = 1,000$°C | Core Average Temp °C |
|---|---|---|---|---|---|
| 12/20 | 1100 | $92500 + 123T$ | $11 \times 10^{-5}$ $+8 \times 10^{-8}T$ | \$3.49 | 278 |

$$\alpha\,[T - T_0] + 2\,[\beta - \rho_0] = 0 \tag{13.8}$$

Knowing the total peaking factor ($PF$) and the peak temperature, one can find the core-average temperature, $T$. With given values for $\beta$ and initial fuel temperature ($T_0$), and the expression for $\alpha$, one can calculate $\rho_0$.

The following input values were used for all calculations here:

$\alpha =$ the temperature dependent heat capacity of the core;

$\beta = 0.007$;

$l = 44$ $\mu$sec;

$T_0 = 20$ °C;

$PF = 3.85$ (conservative for 12 wt% 20% enriched [12/20] fuel).

Using these values for $T_0$ and $PF$, the core-average temperature that corresponds to a peak fuel temperature of 1,100 °C is 304 °C for 12/20 fuel.

The case where the peak fuel temperature was 1,000 °C for 12/20 fuel was also calculated. For this case, the core-average temperature was 278 °C for 12/20 fuel.

The maximum fuel temperature was not found to change with reasonable variations in $l$. The value of $\beta$ is well known. The value of $T_0$ is the nominal zero-power fuel temperature. The value of $PF$ is the largest total peaking factor expected for the highest-loaded fuel type in the core.

The reactivity insertion limit is shown in Table 13.7.

A \$3.49 reactivity insertion for 12/20 fuel would yield a worst case peak fuel temperature of 1,000 °C. This is conservative because actual pulsing data shows peak fuel temperatures in 12/20 fuel that are less than the predicted values. The maximum peak temperature measured in a 12/20 fuel element was 345 °C from a \$2.31 pulse. The theory presented above predicts this pulse would have a peak fuel temperature of 514 °C.

#### 13.2.2.2.2 Uncontrolled Withdrawal of a Control Rod

The GSTR routinely operates with one core configuration where all but two or three grid positions are occupied by fuel elements. The worths of the four control rods vary little with routine changes in fuel loading.

Table 13.8: Control Rod Data for Normal Configuration Core

| Control Rod | Total Worth ($) | Rod position at $T = T_0$ (% withdrawn) | Rod position at P = 1 MW (% withdrawn) | Total rod withdrawal time (sec) | Reactivity insertion rate ($/sec) |
|---|---|---|---|---|---|
| Transient | 2.36 | 44 | 62 | 37.0 | 0.107 |
| Shim 1 | 2.56 | 44 | 62 | 48.9 | 0.079 |
| Shim 2 | 2.47 | 44 | 62 | 49.0 | 0.078 |
| Regulating | 3.88 | 44 | 62 | 35.4 | 0.171 |

Good experimental data exist for the control rod worths and rod withdrawal times for the normal core configuration. Each fueled-follower control rod has an active length of [redacted] inches. Data for this configuration are given in Table 13.8.

Operator error or failure of the automatic power level control system could cause one of the control rods to be driven out, starting either at low power or high power levels. The most reactive rod is the regulating rod and this rod also has one of the fastest withdrawal times. This rod was considered in this accident.

To analyze this accident, the one-delay group model was used with the prompt jump approximation. For a linear increase in reactivity (a ramp input), this model gives:

$$\frac{P}{P_0} = e^{-\lambda t}\left\{\frac{\beta}{[\beta - \gamma t]}\right\}^a, \tag{13.9}$$

where:

$P$ = final power level;

$P_0$ = initial power level;

$\beta$ = total delayed neutron fraction = .007;

$\lambda$ = one group decay constant ($\text{sec}^{-1}$) = .405 $\text{sec}^{-1}$;

$t$ = time (sec);

$\gamma$ = linear insertion rate of reactivity ($\delta$k/k sec); and

$a = 1 + \lambda\$/\gamma$.

For the case with an initial power level of 100 watts, and the trip setpoint at 1.08 MW, the reactor power was calculated to reach the trip setpoint in about 5.24 seconds. Assuming it takes 0.5 seconds for the signal to cause actual release of the rods, the peak reactivity inserted would be \$0.88. This is considerably less than the limiting reactivity insertion of \$3.49, given in Section 13.2.2.2.1, and, thus, should produce no adverse safety effects.

For the case with the initial power at 1 MW, and the trip setpoint again at 1.08 MW, the reactor power was calculated to reach the trip setpoint in 0.45 seconds. Again, allowing for 0.5 seconds for the rods to be released, the peak reactivity inserted would be \$0.145. This is well below the limiting reactivity insertion of \$3.49.

#### 13.2.2.2.3 Uncontrolled Withdrawal of All Control Rods

For this accident, all four control rods were assumed to be withdrawn simultaneously. For this accident to happen, there must be multiple failures in the control system.

The initial power level was assumed again to be 100 watts, and the trip setpoint was 1.08 MW. The total reactivity insertion rate was equal to the sum of the four rod insertion rates: \$0.419/sec. The trip setpoint was reached in about 2.03 seconds, and again assuming 0.5 seconds elapsed before the rods were released, the total reactivity inserted was \$0.85. This is still well below the limiting reactivity insertion of \$3.49.

#### 13.2.2.2.4 Metal-Water Reactions

Although metal-water reactions have occurred in some reactor accidents or destructive tests, the evidence from these events and laboratory experiments shows that a dispersed liquid metal is required for a violent chemical reaction to occur. The conditions for a solid metal-water reaction are not readily achievable in a reactor system such as the GSTR.

Water quench tests on TRIGA® fuel have been conducted to fuel temperatures as high as 1,200 °C without significant effect. Since the operating temperatures at 1 MW do not approach this temperature, this effect does not represent a safety risk. The only credible way in which temperatures high enough to allow metal-water reactions can be created in a TRIGA® reactor is through a large reactivity insertion. The limits set on excess reactivity preclude this.

### 13.2.3 Loss of Coolant Accident (LOCA)

#### 13.2.3.1 Accident Initiating Events and Scenarios

Loss of coolant from the GSTR could occur primarily through one credible scenario: pumping water from the reactor tank. This scenario is analyzed as part of this section.

#### 13.2.3.2 Accident Analysis and Determination of Consequences

#### 13.2.3.2.1 Pumping Water from the Reactor Tank

The outlet for the primary water is located about 4 feet below the normal tank water level. This line and the primary water inlet line each have a syphon break about 22 inches below the normal tank water level. The intake for the water purification system is at the normal water level surface, and if the water level drops about 3 inches or more, this system will lose suction. Thus the normal water systems could only pump water down about 22 inches below the normal tank water level, and could not accidentally pump the tank dry.

Even if the siphon break was clogged, the water system could not pump the water level down more than about 4 feet in a tank that normally has about 24 feet of water in it.

#### 13.2.3.2.2 Reactor Tank Failure

A loss of coolant accident from a reactor tank failure is not credible [redacted] Even if the tank developed a leak, leakage through the concrete would be at a very slow rate and the tank low level alarm would warn the operator of water loss.

There are two immediate considerations of a catastrophic loss of water: cooling of the reactor core, and dose rates around the uncovered core. The consequences of each of these will be discussed in the following sections.

##### 13.2.3.2.2.1 Cooling of Reactor Core

If all coolant is suddenly lost when the reactor has been operating at full power for a considerable period, the primary concern is the maximum fuel temperature that might be reached and whether or not the fuel cladding will remain intact. This accident has been analyzed in numerous reports previously. All of these reports reach the same conclusion: natural convective air cooling of the fuel will keep the maximum fuel temperature well below the temperature for cladding failure if the reactor operates at a maximum power level of 1.5 MW or below. In NUREG/CR-2387, this accident was also examined and the conclusion was that, for a reactor such as the GSTR, a loss of coolant accident was not credible.

##### 13.2.3.2.2.2 Radiation Levels from the Uncovered Core

If the reactor core suddenly becomes uncovered after sustained operation at full power, high radiation doses might be expected from the uncovered core. Radiation doses at three locations are calculated here: at the grating over the top of the reactor, at a point in the reactor bay, and at the nearest fence boundary to the unrestricted area. These latter two doses would be scattered doses from the reactor bay roof.

The basic assumption is that the reactor has been operating at a maximum power level of 1 MW for one continuous year, and then the water is instantly lost. This is a very conservative assumption, since there is no conceivable way the GSTR could be operated continuously, 24 hours per day, at 1 MW for one year. The GSTR normally operates on an 10-hour-per-day shift for 5 days per week.

The total fission product activity as a function of time after shutdown was determined using the standard equation:

$$A(t) = 1.4 \times 10^6 \left[t^{-0.2} - (t+T)^{-0.2}\right] \text{ (Ci)} \tag{13.10}$$

where:

$P$ = reactor thermal power (MW);

$t$ = time after shutdown (days); and

Table 13.9: Total Fission Product Activity and Source Strength After Shutdown

$T$ = operating time (days).

For this calculation: $P = 1$ MW and $T = 365$ days.

The total fission product activity will be calculated at 5 different times after shutdown: 10 seconds, 1 hour, 1 day, 1 week, and 1 month.

The reactor core was modeled as a cylinder of radius 25 cm and a height of 38.1 cm. This assumes that the fuel is filling all grid positions from the B ring through the G ring of the core, which has been the case for the normal core configuration since about 1995. Therefore, the source term for this analysis ($S_v$) was determined by dividing the total activity by the volume of this cylinder:

$$S_v = \frac{3.7 \times 10^{10} A}{\pi R^2 h} \left(\gamma \text{cm}^{-3} \text{sec}^{-1}\right), \tag{13.11}$$

where:

$R$ = core radius (cm) = 25 cm;

$h$ = core height (cm) = 38.1 cm; and

$3.7 \times 10^{10}$ converts curies to gamma rays, assuming gamma rays of 1 MeV per disintegration. The total activity and the source strength are shown in Table 13.9.

#### 13.2.3.2.2.3 Dose Rate Directly Above Core

The reactor core, shutdown and drained of water, was treated as a uniform bare cylindrical source of 1-MeV photons. Its dimensions were taken to be equal to those of the active core lattice. No accounting was made of sources other than fission product decay gammas, and no credit was taken for attenuation through the fuel element end pieces and the upper grid plate. The first of these assumptions is optimistic, the second conservative, and the net effect is conservative. The conservative assumption of a uniformly distributed source of 1-MeV gammas was balanced by not assuming any buildup in the core.

The dose rate was calculated at a point on the axis of the core cylinder at a distance of 762 cm (25 feet) from the top of the core. This is the distance from the top of the core to a point about 3 feet above the tank cover grating.

Table 13.10: Dose Rates on GSTR Reactor Top After a Loss of Pool Water Accident Following 1-MW Operation

| Time After Shutdown | Effective Dose Equivalent Rate (rem/h) |
|---|---|
| 10 seconds | $1.00 \times 10^4$ |
| 1 hour | $1.52 \times 10^3$ |
| 1 day | $1.19 \times 10^3$ |
| 1 week | $6.38 \times 10^2$ |
| 1 month | $3.50 \times 10^2$ |

The dose rate was determined from the equation:

$$\phi = \left[\frac{S_v R^2}{4\mu_c a^2}\right]\left[1 - e^{-\mu_c h}\right](\gamma/\text{cm}^2\text{sec})\,. \tag{13.12}$$

where:

$R$ = core radius (25 cm);

$h$ = core height (38.1 cm);

$\mu_c$ = core attenuation coefficient (0.28/cm);

$a$ = distance from top of core to dose point (610 cm); and

$S_v$ = source strength.

The dose conversion factor, $K$, for effective dose equivalent per unit photon fluence was obtained from ICRP 51, Table 2. This has been calculated for photons incident on an anthropomorphic phantom from various geometries. The worst case (highest dose factor) was for the anterior to posterior geometry, and this was used for this case. For 1-MeV photons, this value of $K$ was $4.60 \times 10^{-12}$ $S_v$ cm$^2$. The dose rate is then calculated by multiplying the gamma flux times this $K$ value, converting to rem (factor of 100) and converting to a time base of one hour by multiplying by a factor of 3600.

The results are given in Table 13.10 and are in agreement with the University of California-Davis and Torrey Pines TRIGA® reactors, when corrections are made for the operating power levels.

#### 13.2.3.2.2.4 Dose Rate From Scattered Radiation in Reactor Bay

The purpose of this section is to calculate the dose rate to a person in the reactor bay not in the direct beam from the exposed core, but subject to scattered radiation from the reactor bay ceiling. The GSTR reactor bay is not normally occupied during reactor operations. The dose point was chosen to be 3 feet above the reactor bay floor at a point in the SE corner of the room. This point is 16 feet from the center of the reactor tank and is the farthest distance one can get from the edge of the reactor and remain in the reactor bay. The ceiling is about 16 feet above the reactor top. The ceiling is assumed to be a concrete slab. The concrete slab

assumption gives the worst case scattering, but it should be noted that the roof over the reactor is partially, but not completely, a concrete slab. Therefore, in reality the scattering will not be as great as calculated because the radiation from the unshielded core will undergo less interaction with the roof.

The following equation was used to find the flux at the dose point from scattered radiation:

$$\phi = \frac{6.03 \times 10^{23} \rho Z I_0 C \frac{\partial \sigma}{\partial \Omega}}{Ax^2 \left\{ \mu_0 + \mu_1 \left[ \frac{\cos \theta_0}{\cos \theta_1} \right] \right\}} \left( \gamma / \mathrm{cm}^2 \mathrm{sec} \right), \tag{13.13}$$

where:

$\rho =$ density of the scattering material (concrete) = 2.35 g/cm$^3$;

$Z/A =$ ratio of the average atomic number to the atomic mass = 0.5;

$I_0C =$ incident beam times the cross section of the beam ($\gamma$/sec);

$x =$ distance from scattering point to dose point = 757 cm;

$\mu_0 =$ attenuation coefficient in scattering material for incident photons = 0.146/cm;

$\mu_1 =$ attenuation coefficient in scattering material for scattered photons = 0.282;

$\theta_0 =$ incident angle, measured from normal to the scatterer = 0°;

$\theta_1 =$ scattered angle, measured from normal to the scatterer = 40.1°; and

$\partial\sigma/\partial\Omega =$ differential Klein-Nishina scattering cross section (cm$^2$).

It was assumed that the source photons which reached the top of the reactor tank were incident normally to the concrete roof ($\theta_0 = 0$) at a point directly over the core. Thus:

$$I_0 C = S_0 \omega = \frac{\pi R^2 S_v \omega}{\mu_c}, \tag{13.14}$$

where $S_0$ is the strength of a point source equal to the total radioactivity and placed at a point which is one mean free path from the top of the reactor.

The expression $\omega$ is the fractional solid angle subtended by the equivalent point source to the top of the reactor tank. Thus:

$$\omega = \frac{1}{2} \left\{ 1 - \frac{z}{(z^2 + r^2)^{1/2}} \right\}, \tag{13.15}$$

where:

$r =$ radius of reactor tank = 121.9 cm; and

$z =$ distance from equivalent point source to top of tank = 610 cm.

Table 13.11: Scattered Radiation Dose Rates in the GSTR Reactor Bay After a Loss of Pool Water Accident Following 1-MW Operation

| Time after shutdown | Effective Dose Equivalent Rate (3rem $h^{-1}$) |
|---|---|
| 10 seconds | 43100 |
| 1 hour | 6560 |
| 1 day | 5130 |
| 1 week | 2750 |
| 1 month | 1510 |

The energy of the scattered photons is given by:

$$E = \frac{E_0}{1\frac{E_0(1-\cos\beta)}{0.51}}, \tag{13.16}$$

where

$E_0$ is the incident photon energy (1 MeV) and $\theta_1$ is the scattering angle = 50.9°. In this case, $\theta_0 = 0°$. For this case, $\beta = 129.1°$ and $E_0 = 0.238$ MeV.

The differential scattering cross section is given by:

$$\frac{\partial\sigma}{\partial\Omega} = \frac{r_e^2}{2\left\{\frac{E}{E_0} - \frac{(E\sin\beta)^2}{E_0^2} + \left(\frac{E}{E_0}\right)^3\right\}}(\text{ cm})^2, \tag{13.17}$$

where $r_e$ is the classical electron radius, $2.818 \times 10^{-13}$ cm. For this case,

$$\frac{\partial\sigma}{\partial\Omega} = 1.827 \times 10^{-25}\ (\text{cm})^2, \tag{13.18}$$

As before, the ICRP 51, Table 2 dose factors were used. Again, the anterior to posterior geometry was used as it gave the largest dose factor for 0.238-MeV scattered photons. This dose factor was $1.15 \times 10^{-12}$ Sv·cm$^2$. Unit conversion factors of 100 Rem/Sv and 3600 s/hr were also applied.

The results of the calculations for the scattered radiation into the reactor bay are given in Table 13.11.

#### 13.2.3.2.2.5 Dose Rate from Scattered Radiation at the Eastern Federal Center Fence

The purpose of this section is to calculate the dose rate to a person at the eastern Federal Center fence not in the direct beam from the exposed core, but subject to scattered radiation from the reactor bay ceiling. The dose point was chosen to be 3 feet above the ground at the eastern Federal Center fence. This is the closest point a member of the public would be able to occupy. The distance to this point from the center of the reactor bay ceiling above the reactor tank (the scattering point) is about 350 m.

The calculational methodology is exactly the same as that used in Section 13.2.3.2.2.2. Values used were the

Table 13.12: Scattered Radiation Dose Rates at the Federal Center East Fence After a Loss of Pool Water Accident Following 1-MW Operation

| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) No attenuation in reactor wall | Effective Dose Equivalent Rate (mrem/h) Including attenuation in reactor wall |
|---|---|---|
| 10 seconds | 0.936 | $3.62 \times 10^{-4}$ |
| 1 hour | 0.142 | $5.51 \times 10^{-5}$ |
| 1 day | 0.111 | $4.31 \times 10^{-5}$ |
| 1 week | 0.0598 | $2.31 \times 10^{-5}$ |
| 1 month | 0.0327 | $1.27 \times 10^{-5}$ |

same except as indicated below:

$\mu_1 = 0.250\ \text{cm}^{-1}$;

$\theta_1 = 87.6°$;

$E = 0.329$ MeV;

$\beta = 92.4°$;

$x = 3.5 \times 10^4$ cm;

$\partial\omega/\partial\Omega = 1.55 \times 10^{-25}$; and

$K = 1.45 \times 10^{-12}$ Sv/cm$^2$sec.

If credit is taken for attenuation in the scattered beam as it passes through the reactor bay wall, then the scattered flux is multiplied by an additional factor of $e^{-\mu x}$, where $\mu$ = attenuation coefficient for concrete for 0.329-MeV photons (0.250/cm) and $x$ is the distance through the 12" thick wall (30.5 cm).

The results of this calculation are given in Table 13.12.

## 13.2.4 Loss of Coolant Flow

### 13.2.4.1 Accident Initiating Events and Scenarios

Loss of coolant flow could occur due to failure of a key component in the reactor primary or secondary cooling system (e.g., a pump), loss of electrical power, or blockage of a coolant flow channel. Operator error could also cause loss of coolant flow.

The GSTR tank holds ~8,000 gallons of water, or about $1.74 \times 10^4$ kg of water. At a steady-state power level of 1 MW, the bulk water temperature would increase adiabatically at a rate of about 0.47°C/min. Under these conditions, the operator has ample time to reduce the power and place the heat-removal system back into operation before a high temperature is reached in the reactor bulk water. The GSTR has visual

indications available to indicate that the primary water pump is off, the secondary water pump is off, and/or the cooling tower fans are off. These will allow the operator to observe an abnormal condition, along with the primary water temperature alarm setpoint of 58 °C.

#### 13.2.4.2 Accident Analysis and Determination of Consequences

##### 13.2.4.2.1 Loss of Coolant Flow Without Immediate Operator Action

If the GSTR was operated without coolant flow for an extended period of time, and there was no heat removal by the reactor coolant systems, voiding of the water in the core could occur and the water level in the reactor tank would decrease because of evaporation. The sequence of events postulated for this very unlikely scenario is as follows:

- The reactor would continue to operate at a power level of 1 MW (provided the rods were adjusted to maintain power) and would heat the tank water at a rate of about 0.47 °C/min until the tank water reached the bulk water high temperature alarm setpoint. This setpoint is at 58 °C. The normal bulk water temperature, when operating at 1 MW, ranges from 35 to 45 °C, depending on the outside air temperature. Thus, it would take 32 minutes or longer to reach the alarm setpoint. Assuming the operator did not notice this alarm and did not take any corrective action, the bulk water would continue to rise above 60 °C. It would then take an additional 85 minutes for the water in the tank to reach the 100 °C temperature. At this time, voids in the core would cause power oscillations and the negative void coefficient would cause a reduction in power if control rods were not adjusted to maintain power; and
- If it is assumed that the operator or automatic control system maintained power at 1 MW, and still assuming that the system is adiabatic except for the evaporation process, about 440 gallons or 1,596 kg per hour would be vaporized. [redacted]. The reactor tank water level alarm would actuate on the control console by the time the water level was 24" below the top lip of the tank, within one hour of the initiation of water boiling. If that alarm was ignored, the reactor would shut down as the water level dropped past the top of the fuel due to the lack of a sufficient neutron moderator. It is considered inconceivable that such an operating condition would go undetected. Water level and temperature alarms would certainly alert the operator. Also, as the water level decreases, [redacted]. Because of all of these factors, the reactor would be shut down unless cooling flow and level could be promptly restored.

### 13.2.5 Mishandling or Malfunction of Fuel

#### 13.2.5.1 Accident Initiating Events and Scenarios

Events which could cause accidents at the GSTR in this category include

1. Fuel handling accidents where an element is dropped underwater and damaged severely enough to breach the cladding,

Table 13.13: Occupational $CDE_{Thyroid}$ and TEDE in the Reactor Bay Following a Single Element Failure in Air and Water

| Reactor Bay Occupancy (minutes) | $CDE_{Thyroid}$ (no water) (mrem) | TEDE (no water) (mrem) | $CDE_{Thyroid}$ (water) (mrem) | TEDE (water) (mrem) |
|---|---|---|---|---|
| 2 | 2778 | 182 | 161 | 104 |
| 5 | 6915 | 455 | 403 | 261 |

2. Simple failure of the fuel cladding due to a manufacturing defect or corrosion, and

3. Overheating of the fuel with subsequent cladding failure during steady-state or pulsing operations.

Event number 2 is by far the most likely and it is the only of the three events that has occurred in almost 40 years of operation of the GSTR.

#### 13.2.5.2 Accident Analysis and Determination of Consequences

All three scenarios mentioned in the previous paragraph result in a single fuel element failure in water. In the unlikely event that this failure occurred in air, it would be the MHA analyzed in Section 13.2.1.2.

At various points in the lifetime of the GSTR, fuel elements are moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is shutdown.

Assumptions for this accident are almost exactly the same as those used for the MHA, except for one thing: the presence of the pool water contains most of the halogens and, thereby, reduces the halogen dose contribution.

The assumptions for this accident and the method of analysis of this accident were described in Section 13.2.1.2.

The results for this accident are given in Tables 13.13 to 13.14.

The results of this accident show that under all possibilities the radiation doses to the general public are well below the annual limits in 10 CFR 20, with the maximum dose being 7.1 mrem TEDE at 10 meters (the fence just west of the reactor bay) for the worst case scenario. The occupational radiation doses to workers in the reactor bay are also well below the occupational annual limits in 10 CFR 20, with the maximum dose being 455 mrem TEDE for a 5-minute exposure. Five minutes is more than enough time for workers to move the 30 feet or less to evacuate the room if such an accident were to occur.

Since most of the halogens released from the fuel element will be retained in the primary water, the majority of this activity will end up in the demineralizer tank. The exposure rate from the demineralizer tank can be estimated by:

$$DR = 6CEN(\text{R/h at 1 foot}) \tag{13.19}$$

where:

Table 13.14: Radiation Doses to Members of the General Public Following a Single Element Failure in Water

| Distance (m) | $CDE_{Thyroid}$ (mrem) | TEDE (mrem) |
|---|---|---|
| 10 | 10 | 1.3 |
| 50 | 0.84 | 0.11 |
| 100 | 0.21 | 0.027 |
| 150 | 0.092 | 0.012 |
| 200 | 0.051 | <0.01 |
| 250 | 0.033 | <0.01 |
| 300 | 0.023 | <0.01 |
| 350 (fence) | 0.017 | <0.01 |
| 640 (residence) | 0.005 | <0.01 |
| 720 (school) | 0.004 | <0.01 |

$C$ = number of halogen curies retained in the demineralizer tank (Ci);

$E$ = energy of gamma rays (MeV) = 1; and

$N$ = number of gamma rays per disintegration = 1.

From Table 13.1, the total saturated activity of the halogens is 5,899 curies. Of this, $1.22 \times 10^{-4}$ is released to the gap, 0.5 of the gap activity is released to the water, and 0.95 of this remains in the water. Thus, the number of curies retained in the demineralizer tank is 0.342 Ci. The assumption that the average energy of the gamma rays from the halogens is 1 MeV is conservative. Thus, Dose Rate = 2.0 R/h at 1 foot.

Surrounding the sides of the GSTR demineralizer tank is 0.5 inch of lead (no shielding on top). With the linear attenuation factor for this lead is equal to about 0.428/cm for 1-MeV gammas, the overall attenuation factor for the lead shield is about 0.581. This reduces the exposure rate to about 1.16 R/h at one foot.

A fuel loading error is another potential way that a fuel element might overheat and result in a cladding failure. The GSTR Technical Specifications require that if a core containing aluminum-clad fuel is used, the aluminum-clad fuel must be in the outer two rings of the core grid. Loading aluminum-clad fuel in the inner rings of the core could potentially result in cladding failure due to overheating of the element during accident conditions.

## 13.2.6 Experiment Malfunction

### 13.2.6.1 Accident Initiating Events and Scenarios

Improperly controlled experiments involving the GSTR could potentially result in damage to the reactor, unnecessary radiation exposure to facility staff and members of the general public, and unnecessary releases of radioactivity into the unrestricted area. Mechanisms for these occurrences include the production of excess amounts of radionuclides with unexpected radiation levels, and the creation of unplanned pressures in irradiated materials. These materials could subsequently vent into the irradiation facilities or into the

reactor bay causing damage from the pressure release or an uncontrolled release of radioactivity. Other mechanisms for damage, such as large reactivity changes, are also possible.

#### 13.2.6.2 Accident Analysis and Determination of Consequences

There are two main sets of procedural and regulatory requirements that relate to experiment review and approval. These are the GSTR Operating Procedures and the GSTR Technical Specifications. These requirements are focused on ensuring that experiments will not fail, and they also incorporate requirements to assure that there is no reactor damage and no radioactivity releases or radiation doses which exceed the limits of 10 CFR 20, should failure occur. For example, the detailed procedures call for the safety review and approval of all reactor experiments.

Safety related reviews of proposed experiments require the performance of specific safety analyses of proposed activities to assess such things as generation of radionuclides and fission products (e.g., radioiodines), and to ensure evaluation of reactivity worth, chemical and physical characteristics of materials under irradiation, corrosive and explosive characteristics of materials, and the need for encapsulation. This process is an important step in ensuring the safety of reactor experiments and has been successfully used for many years at research reactors to help assure the safety of experiments placed in these reactors. Therefore, this process is expected to be an effective measure in assuring experiment safety at the GSTR.

In the GSTR Technical Specifications, a limit of \$1.00 has been placed on the reactivity worth of non-secured experiments. This is well below the maximum reactivity limit of \$3.49 established in Section 13.2.2 and would result in fuel temperatures well below the safety limit.

A limit of \$3.00 has also been placed on the reactivity worth of any single experiment. This is below the maximum reactivity limit of \$3.49 established in Section 13.2.2 and would result in fuel temperatures below the safety limit. A further limit on the reactivity worth of multiple experiments has been set at \$5.00.

The GSTR procedures require that the reactor be shut down before any experiments with a worth >\$1.00 are moved. The Technical Specifications require that the GSTR shutdown margin be at least \$0.55 with the most reactive rod withdrawn. The regulating rod is the most reactive rod, with a worth of \$3.88. Therefore with all the rods inserted, the reactor must be shutdown by at least \$4.45. If a single experiment with a positive worth of \$3.00 was inserted into the core, the reactor would still have a shutdown margin of at least \$1.45. Insertion of negative worth experiments would make the reactor further shutdown, increasing the shutdown margin.

Limiting the generation of certain fission products in fueled experiments also helps to assure that occupational radiation doses as well as doses to the general public, due to experiment failure with subsequent fission product release, will be within the limits prescribed in 10 CFR 20. A limit of 1.5 curies of $^{131}$I through $^{135}$I is specified in the GSTR Technical Specifications. This amount of iodine isotopes is very small compared to the approximately 5400 curies which are present in the single fuel element failure analyzed in Section 13.2.1 (failure in air) and Section 13.2.5 (failure in water). In both cases, the occupational doses and the doses to the general public in the unrestricted area due to radioiodine are within 10 CFR 20 limits. Therefore, limiting experiments to 1.5 curies of radioiodine will result in projected doses far below the 10 CFR 20 limits.

Projected damage to the reactor from experiments involving explosives varies significantly depending on

Table 13.15: Material Strengths

| Material | Yield Strength ($\times 10^3$ psi) | Ultimate Strength ($\times 10^3$ psi) | Density ($g/cm^3$) |
|---|---|---|---|
| Stainless Steel (type 304) | 35 | 85 | 7.98 |
| Aluminum (alloy 6061) | 40 | 45 | 2.739 |
| Polyethylene | 1.7 | 1.4 | 0.923 |

the quantity of explosives being irradiated and where the explosives are placed relative to critical reactor components and safety systems. If in the reactor tank, the GSTR Technical Specifications limit the amount of explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, to quantities less than 25 milligrams. Also, the Technical Specifications state that the pressure produced upon detonation of the explosive must have been calculated and/or experimentally demonstrated to be less than the design pressure of the container. The following discussion shows that the irradiation of explosives up to 25 milligrams could be safely performed if the containment is properly chosen.

A 25-milligram quantity of explosives, upon detonation, releases approximately 25 calories (104.6 joules) of energy, with the creation of 25 $cm^3$ of gas. For the explosive TNT, the density is 1.654 $g/cm^3$, so that 25 mg represents a volume of 0.015 $cm^3$. If the assumption is made that the energy release occurs as an instantaneous change in pressure, the total force on the encapsulation material is the sum of the two pressures. For a 1 $cm^3$ volume, the energy release of 104.2 joules represents a pressure of 1,032 atmospheres. The instantaneous change in pressure due to gas production in the same volume adds another 25 atmospheres. The total pressure within a 1 $cm^3$ capsule is then 1,057 atmospheres for the complete reaction of 25 mg of explosives.

Typical construction materials of capsules are stainless steel, aluminum, and polyethylene. Table 13.15 lists the mechanical properties of these encapsulation materials.

Analysis of the encapsulation materials determines the material stress limits that must exist to confine the reactive equivalent of 25 mg of explosives. The stress limit in a cylindrical container with thin walls is one-half the pressure times the ratio of the capsule diameter-to wall thickness. This is the hoop stress. The hoop stress is 2 times the longitudinal stress, and hence hoop stress is limiting. Thus:

$$\sigma_{\max} = \frac{pd}{2t}, \tag{13.20}$$

where:

$\sigma_{\max}$ = maximum hoop stress in container wall;

$p$ = total pressure within the container;

$d$ = diameter of the container; and

$t$ = wall thickness of the container.

Table 13.16: Container Diameter-to-Thickness Ratio

| Material | d/t |
|---|---|
| Stainless Steel (type 304) | 4.5 |
| Aluminum (alloy 6061) | 5.1 |
| Polyethylene (low density) | 0.22 |

When evaluating an encapsulation material's ability to confine the reactive equivalent of 25 mg of explosives, the maximum stress in the container wall is required to be less than or equal to the yield strength of the material:

$$\frac{pd}{2t} \leq \sigma_{\text{yield}}, \tag{13.21}$$

where $\sigma_{\text{yield}}$ is the yield strength. Solving this equation for $d/t$ provides an easy method of evaluating an encapsulation material:

$$\frac{d}{t} \leq \frac{2\sigma_{\text{yield}}}{p}. \tag{13.22}$$

Assuming an internal pressure of 1,057 atmospheres (15,538 psi), the maximum values of $d/t$ for the encapsulation materials are displayed in Table 13.16. The results indicate that a polyethylene vial is not a practical container since its wall thickness must be at least 4.5 times the diameter. However, both the aluminum and the stainless steel make satisfactory containers.

As a result of the preceding analysis, a limit of 25 mg of TNT-equivalent explosives is deemed to be a safe limitation on explosives which may be irradiated in facilities located inside the reactor tank, provided that the proper container material with appropriate diameter and wall thickness is used.

## 13.2.7 Loss of Normal Electrical Power

### 13.2.7.1 Accident Initiating Events and Scenarios

Loss of electrical power to the GSTR could occur due to many events and scenarios that routinely affect commercial power.

### 13.2.7.2 Accident Analysis and Determination of Consequences

Since the GSTR does not require emergency backup systems to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. A backup power system is present at the GSTR that mainly provides power to the reactor console and the control instrumentation. This backup system consists of a ~3-KVA battery-powered inverter. The system will provide emergency power immediately after the loss of regular electrical power. It will continue to supply power for a period of up to

a few days. Numerous hand-held battery-powered 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 during reactor operations requires that an orderly shut down is to be initiated by the operator on duty. The backup power supply will allow monitoring of the orderly shut down and verification of the reactor's shutdown condition.

## 13.2.8 External Events

### 13.2.8.1 Accident Initiating Events and Scenarios

Hurricanes, tornadoes, and floods are virtually nonexistent in the area around the GSTR. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. In addition, seismic activity in the Lakewood area is relatively low compared to other areas in the United States.

### 13.2.8.2 Accident Analysis and Determination of Consequences

There are no accidents in this category that would have more on-site or off-site consequences than the MHA analyzed in Section 13.2.1, and, therefore, no additional specific accidents are analyzed in this section.

## 13.2.9 Mishandling or Malfunction of Equipment

### 13.2.9.1 Accident Initiating Events and Scenarios

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 (SAR Chapter 7). TRIGA® fuel also incorporates a number of safety features (SAR Chapter 4) which, together with the features designed into the control system, assure safe reactor response, including in some cases reactor shutdown.

Malfunction of confinement or containment systems would have the greatest impact during the MHA, if used to lessen the impact of such an accident. However, no safety considerations at the GSTR depend on confinement or containment systems. Rapid leaks of liquids have previously been addressed in Section 13.2.3. Although no damage to the reactor occurs as a result of these leaks, the details of the previous analyses provide a more comprehensive explanation.

### 13.2.9.2 Accident Analysis and Determination of Consequences

Since there were no credible initiating events identified, no accident analysis was performed for this section and no consequences were identified.

# References

[1] Credible Accident Analyses for TRIGA® and TRIGA®-fueled Reactors, NUREG/CR-2387, PNL-4028, April 1982.

[2] Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content, NUREG-1537, Part 1, February 1996.

[3] "The Calculations of Distance Factors for Power and Test Reactor Sites," J.J.DiMunno et al, TID-14844, U.S. Atomic Energy Commission, March 1962.

[4] Regulatory Guide 3.33, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant," U.S. Nuclear Regulatory Commission, April 1977.

[5] Regulatory Guide 3.34, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant," U.S. Nuclear Regulatory Commission, July 1979.

[6] Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, June 1974 (Also see Regulatory Guide 1.3 on BWRs).

[7] "A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Nuclear Facilities," J.C. Elder et at, LA-10294-MS, Los Alamos National Laboratory, January 1986.

[8] Nuclear Power Reactor Safety, E.E. Lewis, John Wiley and Sons, 1977, p. 521.

[9] Nuclear Engineering, Theory and Technology of Commercial Nuclear Power, R.A. Knief, Hemisphere Publishing, 1992, pp. 353, 431.

[10] "Fuel Elements for Pulsed TRIGA® Research Reactors," M.T. Simnad et al, Nuc. Tech. Vol. 28, January 1976

[11] "The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuel," M.T. Simnad, General Atomic Report E-117-833, February 1980.

[12] HOTSPOT 2.05: Health Physics Code System for Evaluating Accidents Involving Radioactive Materials, Lawrence Livermore National Laboratory, Livermore, California, November 4, 2003.

[13] "Kinetic Behavior of TRIGA Reactors," G.B. West et al, GA-7882, General Atomics, March 31, 1967.

[14] Dynamics of Nuclear Reactors, David Hetrick, U. of Chicago Press, 1971.

[15] Stationary Neutron Radiography System, Final Safety Analysis Report, McClellan Air Force Base, Sacramento, CA, August 1990.

[16] Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, GA-9064, General Atomics, January 5, 1970.

[17] Safety Analysis Report for the TRIGA Reactor Facility, University of Texas at Austin, May 1991.

[18] Introduction to Nuclear Engineering, 2nd Edition, John R. Lamarsh, Addison-Wesley, 1983, p. 72.

[19] "Data for Use in Protection Against External Radiation," International Commission on Radiation Protection, ICRP Report No. 51, Pergamon Press, March 1987.

[20] Safety Analysis Report, Revision 2, McClellan Nuclear Radiation Center, McClellan Air Force Base, Sacramento, CA, April 1998.

[21] Introduction to Nuclear Engineering, 2nd Edition, Richard Stephenson, McGraw-Hill, 1958, pp. 209-213.

# Chapter 14

# TECHNICAL SPECIFICATIONS

Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual Technical Specifications, are included for informational purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

## 14.1 Definitions

**Audit:** A quantitative examination of records, procedures or other documents.

**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.

**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.

**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.

**Channel Test:** A channel test is the introduction of a signal into the channel for verification that it is operable.

**Confinement:** Confinement means an enclosure on the reactor bay which controls the movement of air into it and out through a controlled path.

**Control Rod:** A control rod is a device fabricated from neutron absorbing material or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. 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. It may have a fueled-follower section. Its position may be varied manually or by the servo-controller.
2. Shim Rod: A shim rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section.
3. Transient Rod: The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It may have a voided-follower.

**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.

**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 ($k_{eff} = 1$).

**Experiment:** Any operation, hardware, or target (excluding devices such as detectors) 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 core 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.
2. Movable Experiment: A movable experiment is one that is not secured.

**Experiment Safety Systems:** Experiment safety systems are those systems, including their associated input channel(s), which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

**Fuel Element:** A fuel element is a single TRIGA® fuel rod.

**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.

**Measured Value:** The measured value is the value of a parameter as it appears on the output of a channel.

**Irradiation Facilities:** Irradiation facilities shall mean vertical tubes, rotating specimen rack, pneumatic transfer system, sample holding dummy fuel elements and any other in-tank irradiation facilities.

**Operable:** A system or component shall be considered operable when it is capable of performing its intended function.

**Operating:** Operating means a component or system is performing its intended function.

**Pulse Mode:** Pulse mode shall mean any operation of the reactor with the mode selector in the pulse position.

**Reactor Facility:** The physical area defined by the area that contains the Denver Federal Center Building 15, between North Center Street, 1st Street, 2nd Street, and South Center Street.

**Reactor Operation:** Reactor operation is any condition wherein the reactor is not secured or shut down. Performance of routine subcritical surveillance is not considered operating.

**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.

**Reactor Secured:** The reactor is secured when:

1. It contains insufficient fissile material or moderator present in the reactor core, adjacent experiments or control rods, to attain criticality under optimum available conditions of moderation and reflection with no $^{135}Xe$ present; or,
2. A combination of the following exist:
   (a) The reactor is shut down;
   (b) No experiments or irradiation facilities in the core are being moved or serviced that have, on movement or servicing, a reactivity worth exceeding the maximum value of one dollar; and
   (c) 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.

**Reactor Shutdown:** The reactor is shut down if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included.

**Reference core condition:** The condition of the core when it is at ambient temperature (cold, 18 °C–25 °C) and the reactivity worth of xenon is negligible ($< 0.30$ dollars).

**Review:** A qualitative examination of records, procedures or other documents.

**Safety Channel:** A safety channel is a measuring channel in the reactor safety system.

**Scram time:** Scram time is the elapsed time between the initiation of a scram and the instant that the control rod reaches its fully-inserted position.

**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" denotes permission, neither a requirement nor a recommendation.

**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.

**Shutdown Reactivity:** Shutdown reactivity is the measured reactivity with all control rods inserted. The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

**Square-Wave Mode (S.-W. Mode):** The square-wave mode shall mean any operation of the reactor with the mode selector in the square-wave position.

**Steady-State Mode (S.-S. Mode):** Steady-state mode shall mean operation of the reactor with the mode selector in the manual or auto position.

**Substantive Changes:** Substantive changes are changes that would provide a significant decrease in the safety of an action or event.

**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.

## 14.2 Safety Limits and Limiting Safety System Setting

### 14.2.1 Safety Limit-Fuel Element Temperature

**Applicability.** This specification applies to the temperature of the reactor fuel.

**Objective.** The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding shall result.

**Specifications.**

1. The temperature in an aluminum-clad TRIGA® fuel element shall not exceed 995 °F (535 °C) under any mode of operation.
2. The temperature in a stainless-steel clad TRIGA® fuel element shall not exceed 1,830 °F (1,000 °C) under any mode of operation.

**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 of the integrity of the fuel element cladding could arise from a build-up 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 dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the aluminum-clad TRIGA® fuel element is based on data which indicate that the zirconium hydride will undergo a phase change at 995 °F (535 °C). This phase change can cause severe distortion in the fuel element and possible cladding failure. Maintaining the fuel temperature below this level will prevent this potential mechanism for cladding failure.(SAR 4.5.4).

The safety limit for the stainless-steel clad TRIGA® fuel is based on data including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1,830 °F (1,000 °C) (SAR 4.5.3.1).

### 14.2.2 Limited Safety System Setting

**Applicability.** This specification applies to the scram settings which prevent the safety limit from being reached.

**Objective.** The objective is to prevent the safety limits from being reached.

**Specifications.** The limiting safety system setting shall be a steady state thermal power of 1.1 MW when there are at least 100 fuel elements in the core (including fuel-followed control rods) and a steady state thermal power of 0.1 MW if there are less than 100 fuel elements in the core and any of the core fuel elements are aluminum-clad.

**Basis.** The limiting safety system setting is a total core thermal power, which, if exceeded shall cause the reactor safety system to initiate a reactor scram. This setting applies to all modes of operation. In steady-state operation up to 1.0 MW, ample margins exist between this setting and the safety limits of 535 °C and 1000 °C for aluminum-clad and stainless-steel clad fuel, respectively, as long as the aluminum-clad fuel is restricted to the F and G rings of the core assembly (SAR 4.5.3.1). Thermal and hydraulic calculations indicate that stainless-steel clad TRIGA® fuel may be safely operated up to power levels of at least 1.9 MW with natural convection cooling (SAR 4.5.3.3).

High fuel temperatures are experienced during pulse transients, initiated from low power. The peak fuel temperature reached from pulse operations is controlled by limiting the energy released in the "tail" of the pulse. The pulse tail duration is limited by rod drop time settings that are tested prior to each day of pulsing operations. The peak fuel temperature estimates are obtained from calculations based on an adiabatic reactor kinetics model. Results of this model are very conservative compared to the measured values (SAR 4.5.4.2).

## 14.3 Limiting Conditions of Operation

### 14.3.1 Reactor Core Parameters

#### 14.3.1.1 Steady-state Operation

##### 14.3.1.1.1 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 at least \$0.55 with:

1. Irradiation facilities and experiments in place and the highest-worth, non-secured experiment in its most reactive state;
2. The most reactive control rod fully-withdrawn; and
3. The reactor in the "cold, clean" condition where there is no $^{135}$Xe poison present and the core is at ambient temperature.

**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 should remain in the fully-withdrawn position.

##### 14.3.1.1.2 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 shall not exceed \$6.80.

**Basis.** The nominal total rod worth of the normal core configuration is \$11.27. Subtracting the shutdown margin (\$0.55) and the nominal rod worth of the most reactive rod (\$3.88), the core excess reactivity remaining shall not be allowed to exceed a value of \$6.80.

#### 14.3.1.2 Pulse Mode Operation

**Applicability.** This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

**Objective.** The objective is to assure that the fuel temperature safety limit shall not be exceeded.

**Specifications.** The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical stop on the transient rod, such that the reactivity insertion shall not exceed $3.00.

**Basis.** The fuel temperature rise during a pulse transient has been estimated conservatively by adiabatic models. These models accurately predict pulse characteristics for operation of TRIGA® cores and should be accepted with confidence, relying also on information concerning prompt neutron lifetime and prompt temperature coefficient of reactivity. These parameters have been established for TRIGA® cores by calculations and have been confirmed in part by measurements at existing facilities. In addition, the calculations rely on flux profiles and corresponding power densities which have been calculated for a variety of TRIGA® cores.

In this manner, it is estimated conservatively that reactivity insertions up to $3.00 in operational cores should produce pulse transients with maximum fuel temperatures no greater than 325 °C in aluminum-clad fuel (in the F and G rings) and 800 °C in stainless-steel clad fuel; i.e., a safety margin of 200 °C with respect to the safety limit of the fuel is maintained in either case, allowing for any uncertainties in measurements and/or calculations (SAR 13.2.2.2.1).

#### 14.3.1.3 Core Configuration Limitations

**Applicability.** This specification applies to mixed cores of aluminum-clad and stainless-steel clad types of fuel.

**Objective.** The objective is to assure that the fuel temperature safety limit shall not be exceeded due to power peaking effects in a mixed core.

**Specifications.** Alumimum-clad fuel shall only be loaded in the F and G rings of the core. If aluminum-clad fuel is present in the core and there are less than 100 fuel elements in the core (including fuel follower control rods), then reactor power shall be limited to a steady state limit of 0.1 MW.

**Basis.** The limitation of power peaking effects ensures that the fuel temperature safety limit shall not be exceeded in an operational core. The number of fuel elements in the core is a primary consideration in the peak fuel temperature of each fuel element (SAR 4.5.1.2).

#### 14.3.1.4 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 0.10 inch for stainless-steel clad fuel or 0.50 inch for aluminum-clad fuel;
3. A cladding defect exists as indicated by release of fission products; or

4. Visual inspection identifies significant bulges, pitting, or corrosion.

**Basis.** Gross failure or obvious, significant 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).

## 14.3.2 Reactor Control And Safety System

### 14.3.2.1 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. Physical damage is apparent to the rod or rod drive assembly and it does not respond to normal control rod motion signals; or
2. The scram time exceeds 1 second for the shim and regulating rods or 2 seconds for the transient rod.

**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 (SAR 13.2.2.2.1).

### 14.3.2.2 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 power 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 power measuring channels listed in Table 14.1 are operable.

Table 14.1: Minimum Measuring Channels

| Measuring Channel | Effective Mode | | |
|---|---|---|---|
| | S.-S. | Pulse | S.-W |
| Power Level (NP1000 and NPP1000) | 2 | - | 2 |
| Nvt-Circuit | - | 1 | - |
| Power level (NM1000) | 1 | - | - |

**Basis.** The power level monitors assure that the reactor power level is adequately monitored for steady-state, square wave and pulse modes of operation (SAR 7.2.3.1). The specifications on reactor power level indication are included in this section, since the power level is directly related to the fuel temperature.

#### 14.3.2.3 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 Table 14.2 and interlocks described in Table 14.3 are operable.

Table 14.2: Minimum Reactor Safety Channels

| Safety Channel | Function | Effective Mode | | |
|---|---|---|---|---|
| | | S.-S. | Pulse | S.-W. |
| Power Level | SCRAM @ 1.1 MW(t) or less | 2 | - | 2 |
| Preset timer | SCRAM (<15 sec) | - | 1 | - |
| Console Scram Button | SCRAM | 1 | 1 | 1 |
| High Voltage | SCRAM @ loss of nominal operating voltage to required power level channels | 2 | 1 | 2 |
| Watchdog scrams | Scram upon lack of response in DAC or CSC computer (one scram circuit per computer) | 2 | 2 | 2 |

Table 14.3: Minimum Interlocks

| Interlock | Function | Effective Mode | | |
|---|---|---|---|---|
| | | S.-S. | Pulse | S.-W. |
| NM1000 Power Level Channel | Prevents control rod withdrawal @ less than $10^{-7}$% power | 1 | - | - |
| Transient Rod Cylinder | Prevents application of air unless fully inserted | 1 | - | - |
| 1 kW Pulse Interlock | Prevents pulsing above 1 kW | - | 1 | - |
| Shim, and Regulating Rod Drive Circuits | Prevents simultaneous manual withdrawal of two rods | 1 | - | 1 |
| Shim, and Regulating Rod Drive Circuits | Prevents movement of any rod except transient rod | - | 1 | - |

**Basis.** The power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to

shut down the system if an unsafe or abnormal condition occurs. The high voltage scram ensures that the required power measuring channels have sufficient high voltage as required for proper functioning of their power level scrams. The interlock to prevent startup of the reactor at count rates less than $10^{-7}$% power assures that the startup is not initiated unless a reliable indication of the neutron flux level in the reactor core is available. The interlock to prevent the initiation of a pulse above 1 kW is to assure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady-state mode. The interlock to prevent withdrawal of the shim, safety or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The interlock to prevent simultaneous withdrawal of two control rods is to limit reactivity insertion rate from the standard control rods.

### 14.3.3 Reactor Primary Tank Water

**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 ion exchanger resin integrity.

**Specifications.** The reactor primary water shall exhibit the following parameters:

1. The tank water level shall be at least16 feet above the top of the core;
2. The bulk tank water temperature shall not exceed 140 °F (60 °C);
3. The conductivity of the tank water shall be less than 5 $\mu$S when averaged over a one month period;
4. The pH of the tank water shall be in the range of 4.5 to 7.5 if there is any aluminum-clad fuel in the core.
5. When aluminum-clad fuel is in the core, the reactor shall not be operated if the tank water level is more than 24" below the top lip of the reactor tank.
6. These specifications are not required to be met if the reactor core has been defueled.

**Basis.** The minimum height of 16 feet of water above the top of the core guarantees that there is sufficient water for effective cooling of the fuel and that the radiation levels at the top of the reactor are within acceptable levels (SAR 4.4, 4.5.3.1.1, and 11.1.1.1). The bulk water temperature limit is necessary to ensure that the ion exchange resin does not undergo severe thermal degradation. Experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control of corrosion (NUREG-1537). The minimum water level of no more than 24" below the top lip of the reactor tank ensures sufficient cooling water during the design reactor tank leak of 350 gpm for the aluminum-clad fuel to cool to safe levels after a reactor shutdown.

### 14.3.4 This section intentionally left blank.

### 14.3.5 Ventilation System

**Applicability.** This specification applies to the operation of the facility 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.**

1. The reactor shall not be operated unless the facility ventilation system is operating and the reactor bay pressure is maintained negative with respect to surrounding areas by at least 0.1" water pressure. This does not apply to short periods of time (not to exceed 2 hours) for system troubleshooting, maintenance and movement of personnel or equipment through open doors.
2. The reactor bay ventilation system shall operate in the emergency mode, with all exhaust air passing through a HEPA filter, whenever a high level continuous air monitor alarm is present due to airborne particulate radionuclides in the reactor bay.

**Basis.** The worst-case maximum total effective dose equivalent is well below the 10 CFR 20 limit for individual members of the public. This has been shown to be true for scenarios where the ventilation system continues to operate during the MHA and where the ventilation system does not operate during the MHA. (SAR 13.2.1). Therefore, operation of the reactor for short periods while the reactor bay underpressure is not maintained because of testing or reactor bay open doors, does not compromise the control over the release of radioactive material to the unrestricted area nor should it cause occupational doses that exceed those limits given in 10 CFR 20 (SAR 11.1.1.1.5). Moreover, radiation monitors in the building, independent of the ventilation system, will give warning of high levels of radiation that might occur during operation of the reactor (SAR 11.1.6).

### 14.3.6 This section intentionally left blank.

### 14.3.7 Radiation Monitoring Systems and Effluents

#### 14.3.7.1 Radiation Monitoring Systems

**Applicability.** This specification applies to the radiation monitoring information which must be available to the Reactor Operator during reactor operation.

**Objective.** The objective is to specify the minimum radiation monitoring 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 radiation monitoring channels listed in Table 14.4 are operating, except for time periods of up to 2 hours for repair and maintenance, provided:

Table 14.4: Minimum Radiation Monitoring Channels

| Radiation Monitoring Channels | Number |
|---|---|
| Continuous Air Particulate Radiation Monitor | 1 |
| Area Radiation Monitor | 1 |

1. The ventilation system is operating; and

2. The continuous air particulate radiation monitor is operating.

Each channel shall have a readout in the control room and be capable of sounding an audible alarm which can be heard in the reactor control room.

**Basis.** 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 (SAR 11.1.6).

#### 14.3.7.2 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 $4.8 \times 10^{-6} \mu$Ci/ml at the point of discharge.

**Basis.** If $^{41}$Ar is continuously discharged at $4.8 \times 10^{-6} \mu$Ci/ml, measurements and calculations show that $^{41}$Ar released to the publicly accessible areas under the worst-case weather conditions would result in an annual TEDE of 5 mrem (SAR 11.1.1.1.1). This is only 50% of the applicable limit of 10 mrem.

### 14.3.8 Limitations on Experiments

#### 14.3.8.1 Reactivity Limits

**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. Movable experiments shall have reactivity worths less than $1;

2. The reactivity worth of any single secured experiment shall be less than $3.00;

3. Total experiment worth of all experiments shall be less than $5.00;

**Basis.** The worst event which could possibly arise is the sudden removal of a movable experiment immediately prior to, or following, a pulse transient of the maximum licensed reactivity insertion. Limiting the worth of the movable experiment to less than $1 will assure that the additional increase of transient power and temperature is slow enough for the high power scram to be effective (SAR 7.2.3.1 and 13.2.2).

The worst event which may be considered in conjunction with a single secured experiment is its sudden removal while the reactor is operating in a critical condition at a low power level.

This is equivalent to pulse-mode operation of the reactor. Hence, the reactivity limitation for a single secured experiment is the same as that for pulsing (SAR 13.2.2.2.1 and 14.3.1.2).

#### 14.3.8.2 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, or nitroglycerin, in quantities greater than 25 milligrams shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities less than or equal to 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container; and

2. Each fueled experiment shall be controlled such that the total inventory of $^{131}$I–$^{135}$I in the experiment is no greater than 1.5 curies and the total inventory of $^{90}$Sr in the experiment is no greater than 5 millicuries.

**Basis.** This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials (SAR 13.2.6.2). The 1.5-curie limitation on $^{131}$I–$^{135}$I, and the 5 millicurie limit on $^{90}$Sr, assure that in the event of a failure of a fueled-experiment involving total release of the iodine, the dose in the reactor bay and in the unrestricted area will be considerably less than that allowed by 10 CFR 20 (SAR 13.2.6).

#### 14.3.8.3 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 (except fueled experiments) 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 bay 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 130 °F 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 GSTR will result in exceeding the total dose limits to an individual as specified in 10 CFR 20.

### 14.3.9 This section intentionally left blank.

## 14.4 Surveillance Requirements

### 14.4.1 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. A channel calibration shall be made of the power level monitoring channels by the calorimetric method semi-annually.
2. The total reactivity worth of each control rod shall be measured following any significant change in core or control rod configuration.
3. The shutdown reactivity shall be determined prior to each day's operation, prior to each operation extending more than one day, or following any significant change in core or control rod configuration.

4. The core excess reactivity shall be determined prior to each day's operation or following any significant change in core or control rod configuration.
5. All fuel elements shall be inspected for deterioration and measured for length and bend at five year intervals or every 500 pulses.
6. The core shutdown margin shall be determined at an annual frequency.
7. These checks are not required if the reactor core has been defueled.

**Basis.** Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

### 14.4.2 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. The control rods shall be visually inspected for damage or deterioration biennially.
2. The scram time shall be measured semi-annually.
3. The transient rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated as necessary, semi-annually.
4. A channel check of each of the reactor safety system channels for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day.
5. A channel test of each item in Table 14.2 and 14.3 in section 14.3.2.3, other than power measuring channels, shall be performed semi-annually.
6. These checks are not required if the reactor core has been defueled.

**Basis.** Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

### 14.4.3 Reactor Primary Tank Water

**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 alarm shall be performed semi-annually.
2. A channel check of the reactor tank water temperature alarm shall be performed quarterly. A channel calibration of the reactor tank water temperature system shall be performed annually.
3. The reactor tank water conductivity shall be measured monthly unless the reactor tank is drained.
4. The reactor tank water pH shall be measured quarterly if Al clad fuel is in the core, and the reactor tank is filled.

**Basis.** Experience has shown that the frequencies of checks on systems which monitor reactor primary water can adequately keep the tank water at the proper level and maintain water quality at such a level to minimize corrosion and maintain safety.

### 14.4.4 This section intentionally left blank.

### 14.4.5 Ventilation System

**Applicability.** This specification applies to the reactor bay 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 bay ventilation system's ability to maintain a negative pressure in the reactor bay 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 check of the reactor bay ventilation system's ability to automatically switch to the emergency mode upon actuation of the CAM high alarm shall be performed quarterly.

**Basis.** Experience has demonstrated that checks of the ventilation system on the prescribed frequencies are sufficient to assure proper operation of the system and its control over releases of radioactive material.

### 14.4.6 This section intentionally left blank.

### 14.4.7 Radiation Monitoring System

**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 check of the radiation monitoring systems in section 14.3.7.1 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 particulate monitor shall be performed monthly.
3. A channel calibration of the radiation monitoring systems in section 14.3.7.1 shall be performed annually.

**Basis.** 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.

### 14.4.8 Experimental Limits

**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 routine 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 Section 14.3.8 by the Reactor Supervisor or Reactor Operations Committee in full accord with Section 14.6.2.3 of these Technical Specifications, and the procedures which are established for this purpose.

**Basis.** Experience has shown that experiments which are reviewed by the staff of the GSTR and the Reactor Operations Committee can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

### 14.4.9 This section intentionally left blank.

## 14.5 Design Features

### 14.5.1 Site and Facility Description

**Applicability.** This specification applies to the U.S. Geological Survey 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 fence surrounding the reactor building and the reactor building itself. The unrestricted area is that area outside the reactor building and the fence surrounding the reactor building.

2. Building 15 houses the TRIGA® reactor and other research laboratories of the U.S. Geological Survey.

3. The reactor facility shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor bay and release them from vertical level at least 21 feet above ground level.

4. Emergency controls for the ventilation systems shall be located in the reactor control room.

**Basis.** The reactor building and site description are strictly defined (SAR Chapter 2). The facility is designed such that the ventilation system will normally maintain a negative pressure in the reactor bay with respect to the outside atmosphere so that there will be no uncontrolled leakage to the unrestricted environment. Controls for normal and emergency operation of the ventilation system are located in the reactor control room. 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 (SAR 9.1 and 13.2.1).

### 14.5.2 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 convective water flow.

2. The tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks 14 feet above the top of the core or higher.

3. A tank water level alarm shall be provided to indicate loss of coolant prior to the tank level dropping 24 inches below the top lip of the tank.

4. A bulk tank water temperature alarm shall be provided to indicate high bulk water temperature prior to the temperature exceeding 140 °F (60 °C).

5. These specifications are not required to be met if the reactor core has been defueled.

**Basis.**

1. This specification is based on thermal and hydraulic calculations which show that the TRIGA® core can operate in a safe manner at power levels up to 1.9 MW with natural convection flow of the coolant water (SAR 4.5.3.3).

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 drop to a level no less than 14 feet from the top of the core (SAR 5.2).

3. Loss-of-coolant alarm caused by a water level drop to no more than 24 inches below the top lip of the tank provides a timely warning so that corrective action can be initiated. This alarm is located in the control room (SAR 5.2).

4. The bulk water temperature alarm provides warning so that corrective action can be initiated in a timely manner to protect the quality of the ion exchange resin. The alarm is located in the control room (SAR 7.2.3.2).

## 14.5.3 Reactor Core and Fuel

### 14.5.3.1 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 shall be an arrangement of TRIGA® uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

2. The TRIGA® core assembly may consist of stainless-steel clad fuel elements (8.5 to 12.0 wt% uranium), aluminum-clad fuel elements (8.0 wt% uranium), or a combination thereof. All aluminum-clad fuel elements must be located in the F or G rings.

3. 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, and startup sources. The core may also contain two separated experiment positions in the D through E rings, each occupying a maximum of three fuel element positions.

4. Core grid positions may be empty (water filled).

5. The reflector, excluding experiments and irradiation facilities, shall be graphite, water, or a combination of graphite and water. A reflector is not required if the core has been defueled.

**Basis.**

1. Standard TRIGA® cores have been in use for years and their characteristics are well documented. Analytic studies performed at GSTR for a variety of mixed fuel arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR 4.2).

2. The core will be assembled in the reactor grid plate which is located in a tank of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of irradiation facility radiation requirements (SAR 4.2).

#### 14.5.3.2 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.**

1. The shim 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. These rods may incorporate fueled followers.

2. The transient control rod shall have scram capability and contain borated graphite or boron, with its compounds in a solid form as a poison in an aluminum or stainless steel cladding. The transient rod drive mechanism shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate an aluminum-or air-follower.

**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. Control rods that are fuel-followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled-followers has the additional advantage of reducing flux peaking in the water-filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for rapid withdrawal from the reactor core which results in a reactor pulse. The nuclear behavior of the air- or aluminum-follower, which may be incorporated into the transient rod, is similar to a void. (SAR 4.2.2).

#### 14.5.3.3 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.**

1. a. Aluminum-clad TRIGA® fuel. The individual unirradiated aluminum-clad fuel elements shall have the following characteristics:

(a) Uranium content: nominally 8.0 wt% enriched to a nominal 20% $^{235}U$;

(b) Hydrogen-to-zirconium atom ratio nominally 1.0 to 1; and

(c) Cladding is aluminum of a nominal 0.030 inch thickness.

2. Stainless-steel clad TRIGA® fuel. The individual unirradiated standard TRIGA® fuel elements shall have the following characteristics:

(a) Uranium content: nominal range of of 8.5 to 12.0 wt% enriched to a nominal 20% $^{235}U$;

(b) Hydrogen-to zirconium atom ratio nominally between 1.6 to 1 and 1.7 to 1; and

(c) Cladding: 304 stainless steel, nominal 0.020 inches thick.

**Basis.**

1. A nominal uranium content of 8 wt% in an aluminum-clad TRIGA® element is less than the traditional stainless-steel clad element design value of 8.5 wt%. Such an decrease gives a lower power density. The nominal hydrogen-to-zirconium ratio of 1.0 to 1 could result in a phase change of the ZrH if fuel temperature is allowed to exceed 535 °C. Although this would not necessarily cause a rupture of the fuel cladding, it would cause distortion and stressing of the cladding.

2. A maximum nominal uranium content of 12 wt% in a standard TRIGA® element is about 50% greater than the lower-loaded nominal value of 8.5 wt%. Such an increase in loading would result in an increase in power density of less than 50%. An increase in local power density of 50% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.7 to 1 could result in a maximum stress under accident conditions to the fuel element cladding of about a factor of 1.5 greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the cladding stress during an accident would not exceed the rupture strength of the cladding.

### 14.5.4 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.

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 design values.

**Basis.** The limits imposed are conservative and assure safe storage (NUREG-1537).

## 14.6 Administrative Controls

### 14.6.1 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 radiation exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations. The minimum qualification for all members of the reactor operating staff shall be in accordance with ANSI/ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors."

#### 14.6.1.1 Structure

The reactor administration shall be related to the USGS and USNRC structure as shown in Figure 14.1.

#### 14.6.1.2 Responsibility

The following specific organizational levels, and responsibilities shall exist:

1. USGS Director (Level 1): The Director is accountable for ensuring that all regulatory requirements, including implementation and enforcement, are in accordance with all requirements of the USNRC and the Code of Federal Regulations.
2. Reactor Administrator (Level 2): The Reactor Administrator is responsible to the Director and is responsible for guidance, oversight, and management support of reactor operations.
3. Reactor Supervisor (Level 3): The Reactor Supervisor reports to the Reactor Administrator and is responsible for directing the activities of the Reactor Operators and Senior Reactor Operators and for the day-to-day operation and maintenance of the reactor.
4. Reactor Operator and Senior Reactor Operator (Level 4): The Reactor Operator and Senior Reactor Operator report to the Reactor Supervisor and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor related equipment.

#### 14.6.1.3 Staffing

1. The minimum staffing when the reactor is operating shall be:
   (a) A Licensed Operator in the control room;
   (b) A second facility staff person present or on call; and
   (c) On call means an individual who:
      i. Can be reached by an available communication method within 5 minutes and

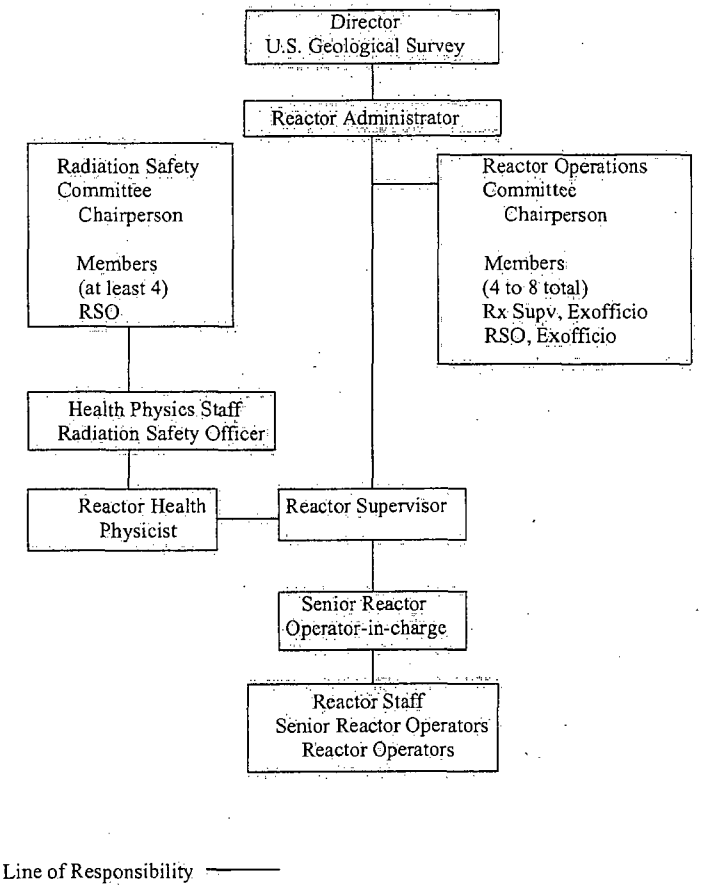


Figure 14.1: Administrative Structure

ii. Is capable of getting to the reactor facility within 15 minutes under normal conditions.

(d) A SRO shall be reachable by any communication method and capable of getting to the reactor facility within 15 minutes under normal conditions.

2. Events requiring the direction of a Senior Reactor Operator

(a) Initial approach to critical for each day's first critical operation;

(b) Reactor start-up and approach to power;

(c) All fuel or control-rod movements within the reactor core region;

(d) Relocation of any in-core components (other than normal control rod movements) or irradiation facility with a reactivity worth greater than one dollar;

(e) Recovery from unplanned or unscheduled shutdown or an unscheduled significant power reduction; and

#### 14.6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall be in accordance with ANSI/ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors."

### 14.6.2 Review And Audit

The ROC shall have primary responsibility for review and audit of the safety aspects of reactor facility operations.

#### 14.6.2.1 Composition and Qualifications

The Reactor Operations Committee shall be composed of at least four voting members, including the Chairman. All members of the Committee shall be knowledgeable in subject matter related to reactor operations. To expedite Committee business, a Committee Chairman has been appointed. The Chairman of the Reactor Operations Committee is listed by name on the Reactor Operations Committee roster.

The Committee is appointed by the Director, U.S. Geological Survey. No definite term of service is specified; but should a vacancy occur in the Committee, the Director will appoint a replacement. The remaining members of the Committee will be available to assist the Director in the selection of new members. The Reactor Supervisor and the Radiation Safety Officer are ex officio members of the Committee and the Reactor Supervisor is the only non-voting member of the Committee.

#### 14.6.2.2 Charter and Rules

The Reactor Operations Committee consists of USGS members and non-USGS members, and the Committee must meet at least semi-annually.

Criteria have been established for the conduct of the meetings and a charter for the Committee is written in the USGS Survey Manual.

A quorum for review, audit, and approval purposes shall consist of not less than one-half of the committee membership, provided that the operating staff does not constitute a majority of the committee membership. The Chairperson or an alternate must be present at all meetings in which the official business of the committee is being conducted. Approvals[†] by the committee shall require an affirmative vote by a majority of the non-Survey members present and an affirmative vote by a majority of the Survey members present.

#### 14.6.2.3 Review and Audit Function

Semiannual meetings will be held to review and audit reactor operations. These meetings will also include inspections of the reactor facility and audits of reactor records by the Committee. New Class I Experiments

[†] "Approval" as used in this section shall mean approval for substantive issues, including Class II Experiments, 10 CFR 50.59 changes, license amendments or any other matters designated by the Reactor Administrator, the Reactor Supervisor, or the Reactor Operations Committee. Minor issues will continue to be handled by the reactor staff and the Reactor Supervisor as described elsewhere in this Reactor Operations Manual.

approved by the Reactor Supervisor will be discussed. Any new Class II Experiments pending, or any business which requires Committee approval will be handled in the following manner.

1. The Reactor Supervisor will request a meeting of Survey members.

2. If the experiment is approved by them, the Reactor Supervisor will communicate the recommendations of the local group, along with the proposed experimental details to Non-Survey Members. The Reactor Supervisor shall provide adequate detail, either verbal, or in writing to the out-of-town members to allow an informed decision. The decisions, with comments and discussions, shall be documented for future review and audit purposes.

3. Approval to proceed will be given by the Reactor Operations Committee Chairman, who will sign the experiment application document to indicate Committee approval.

4. If the experiment cannot be approved in this manner, the experiment application document will be distributed to all Committee members for further discussion and will be placed on the agenda of the next full Committee meeting. Normally the non-Survey members will attend only every other semi-annual meeting. These full Committee meetings are generally scheduled each spring. Special meetings for the review of specific problems or urgent Class II Experiments will be called whenever necessary.

A Quarterly Report will be prepared by the Reactor Supervisor for the members of the Committee. This report will be in sufficient detail to allow members of the Committee to review the safety standards associated with the operation and use of the reactor facility. Each Quarterly Report will include a synopsis of all experiments approved during the period to insure that the intent and function of the Committee, as mentioned in the Technical Specifications is being maintained.

#### 14.6.2.4 Inactive Member Status

Committee members will be considered on inactive status should prolonged absences from their normal business address not allow them to participate in routine Committee business. The Committee Chairperson is responsible to insure the function of the Committee is not diluted so as to be ineffective by such actions and will recommend the appointments of alternates if the need arises.

The Reactor Supervisor will insure that any members who are unable to attend a Reactor Operations Committee meeting, or are on inactive status due to other commitments, will be kept informed of all Committee business.

### 14.6.3 Radiation Safety

The Reactor Supervisor, in coordination with the Reactor Health Physicist, 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 should use the guidelines of the ANSI/ANS 15.11, "Radiation Protection at Research Reactor Facilities."

### 14.6.4 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. Procedures shall be in effect for the following items:

1. Performing experiments and maintenance;
2. Startup, operation and shutdown of the reactor;
3. Emergency situations;
4. Core changes and fuel movement;
5. Control rod removal and replacement;
6. Performing maintenance which may affect reactor safety;
7. Administrative controls;
8. Power calibration; and
9. Radiation protection.

Substantive changes to the above procedure shall be made only with the approval of the ROC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the Reactor Supervisor and documented by the Reactor Supervisor within 90 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved and documented by the Reactor Health Physicist within 90 days of implementation.

### 14.6.5 Required Actions

#### 14.6.5.1 Actions to Be Taken in Case of Safety System Setting Violation

In the event a safety system setting limit (steady state power level of 1.1 MW) is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the Reactor Operations Committee;
2. An immediate notification of the occurrence shall be made to the Reactor Administrator, ROC Chairperson; and
3. Reports shall be made to the USNRC in accordance with Section 14.6.6.2 of these Technical Specifications. The written report (required within 14 days) shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the ROC for review and submitted to the NRC when authorization is sought to resume operation of the reactor.

#### 14.6.5.2 Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 14.6.6.2 Other than a Safety System Setting Violation

For all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 14.6.6.2, except a safety system setting violation, the following actions shall be taken:

1. The reactor shall be secured and the Reactor Supervisor notified;

2. Operations shall not resume unless authorized by the Reactor Supervisor;

3. The Reactor Operations Committee shall review the occurrence at their next scheduled meeting; and

4. Where appropriate, a report shall be submitted to the NRC in accordance with Section 14.6.6.2 of these Technical Specifications.

### 14.6.6 Reports

#### 14.6.6.1 Annual Operating Report

An annual report covering the previous calendar year shall be created and submitted by the Reactor Supervisor to the USNRC 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 shutdowns, 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 percent 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 percent of that allowed.

#### 14.6.6.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made by the Reactor Supervisor to the NRC as follows:

1. A report within 24 hours by telephone or fax to the NRC Operations Center followed by a written report within 14 days that describes the circumstances associated with any of the following:
   (a) Any accidental release of radioactivity above applicable limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;
   (b) Any violation of a safety limit;
   (c) Operation with a safety system setting less conservative than specified in the Technical Specifications.;
   (d) Operation in violation of a Limiting Condition for Operation;
   (e) Failure of a required reactor safety system component which could render the system incapable of performing its intended safety function unless the failure is discovered during maintenance tests or periods of reactor shutdown;
   (f) An unanticipated or uncontrolled change in reactivity greater than $1.00;
   (g) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; or
   (h) A release confirmed to be fission products greater than 10CFR20 from a fuel element;
2. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:
   (a) Permanent changes in the facility organization involving Level 1-2 personnel;
   (b) Significant changes in the transient or accident analyses as described in the Safety Analysis Report;

### 14.6.7 Records

#### 14.6.7.1 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, data 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. Reactor Operations Committee meetings and audit reports.

#### 14.6.7.2 Records to be Retained for at Least One Training Cycle

1. Records of retraining and requalification of Reactor Operators and Senior Reactor Operators shall be retained for at least one training cycle.
2. Records of the most recently completed training cycle for an individual shall be maintained at least as long as that individual is employed at the facility.

#### 14.6.7.3 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; and
4. Drawings of the reactor facility.

# Chapter 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

In fiscal year 2008, the funding level from the USGS for the entire USGS Reactor Project is projected to total $411,100. As shown in Figure 15.1, the projected project expenses for the past 20 years have generally ranged from about $290,000 to $431,000. Salary expenses generally increase each year and that increase depends significantly on the stability of the staff, potential promotions, and changes in cost of living. Staff salaries follow the rules and regulations formulated by the U.S. Congress, Office of Personnel Management, Department of Interior, and U.S. Geological Survey.

Salary costs include direct salaries and benefits; however, project expense numbers do not include the infrastructure costs such as electrical power, heating, air conditioning, and most building maintenance. Salary costs typically consist of ~85% of the annual expenses shown in Figure 15.1. The project cost for FY2009 is estimated to be $491,000, and the subsequent 5 year project costs are estimated with a 6% annual inflation increase.

| | |
|---|---|
| FY2010 | $520,460 |
| FY2011 | $551,687 |
| FY2012 | $584,789 |
| FY2013 | $619,876 |
| FY2014 | $657,069 |

The institutional funding for the reactor facility is provided by the USGS through the U.S. Department of Interior, through congressional appropriations. Within the USGS, funding support for the reactor comes from USGS program assessments and user fees. The USGS programs that typically support the reactor are

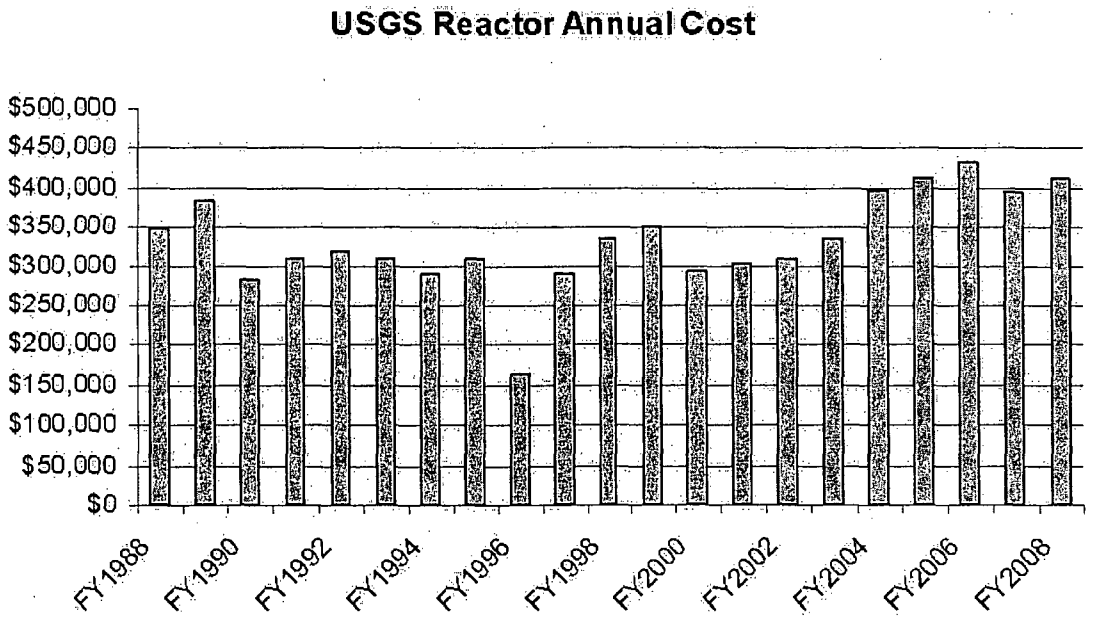


Figure 15.1: Current and Past Expenses for the USGS Reactor Facility

the Minerals Assessment Program, Coastal and Marine Geology Program, and Energy Resources Program. It is expected that these same programs will continue to support the reactor operation for the foreseeable future.

The Department of the Interior funding details are given in Appendix 15–A.

## 15.3 Financial Ability to Decommission the Facility

The USGS is a federal bureau within the U.S. Department of Interior. Therefore, in accordance with the provisions of 10 CFR 50.75(e)(2)(iv) the funds needed for decommissioning will be requested through appropriate federal funding channels and will be obtained sufficiently in advance of decommissioning to prevent delay of required activities.

The decommissioning alternative planned for the GSTR after permanent cessation of operation is DECON. The alternative will require that the facility be decontaminated sufficiently to allow unrestricted release and future use of the reactor site.

In memorandum dated March 1990, the USGS estimated the cost of decommissioning at $1.0 million. In 1995, the cost of decommissioning was estimated to be $1.227 million based on inflation and increases in radioactive waste disposal costs. The 1995 estimate was updated in 2000 to be $2.146 million. The most current estimated cost of decommissioning the GSTR is estimated at $4.25 million as of April, 2005.

The details of the March 2006 decommissioning cost estimate are as follows:

| | |
|---|---|
| Planning, calculations and inventories | $102,926 |
| Fuel transportation to DOE site | $171,543 |
| Dismantling, decontamination and disposal | $2,524,286 |
| USGS preparation and miscellaneous expenses | $171,543 |
| Contingency allowance (25%) | $742,574 |
| Total Decommissioning Cost Estimate | $3,712,872 |

These values assume that the four reactor staff members remain on site to assist in the decommissioning effort and that their salaries and overhead costs are not included in the above estimate.

# APPENDIX 15–A

Table 1: Budget Authority and Receipts for 2007, 2008, and 2009 (in millions of dollars)

| | 2007 Actual[a] | 2008 Estimate | 2009 Request | Change from 2008 |
|---|---|---|---|---|
| **BUDGET AUTHORITY** | | | | |
| Total Current Appropriations *(w/o supps)* | 10,976 | 11,112 | 10,724 | -388 |
| Supplementals | 108 | 249 | 0 | -249 |
| Total Current Appropriations *(w/ supps)* | 11,084 | 11,361 | 10,724 | -637 |
| Permanent Appropriations | 4,715 | 5,523 | 5,994 | +471 |
| Total | 15,799 | 16,884 | 16,718 | -166 |
| *[Net discretionary BA]* | *[10,987]* | *[11,250]* | *[10,625]* | *[-624]* |
| **RECEIPTS** | | | | |
| Outer Continental Shelf | 6,763 | 11,120 | 10,159 | -961 |
| Onshore Mineral Leasing | 3,922 | 4,568 | 5,496 | +927 |
| Other Offsetting Receipts | 1,582 | 1,674 | 1,590 | -84 |
| Other Receipts | 722 | 806 | 917 | +111 |
| Total | 12,989 | 18,169 | 18,162 | -7 |

[a]Excludes one-time disaster supplementals totaling $28.0 million

The entire report can be found online at
http://www.doi.gov/budget/2009/09Hilites/2009_Highlights_Book.pdf.

# References

[1] Memorandum to file; March, 1990; USGS Reactor Supervisor

[2] Memorandum to file; March, 1995; USGS Reactor Supervisor

[3] Memorandum to file; April, 2000; USGS Reactor Supervisor

[4] Memorandum to file; November, 2008; USGS Reactor Supervisor

# REACTOR OPERATOR REQUALIFICATION PROGRAM

## USGS TRIGA Reactor

### 1 Program

The requalification training program was established to maintain the proficiency of the operating organization through training exercises, instruction periods, and reviews covering those items and equipment which relate to the safe operation of the facility. While the program will be continuous in nature, certain benchmarks have been established to aid in the administration of the requalification plan. They are: (1) annual written examinations, (2) annual operating examinations and (3) stipulated minimum tasks which must be performed during the requalification period of two years.

### 2 Written Examination

An annual (not to exceed 14 months) written examination, designed to demonstrate competence in the areas listed in 10 CFR 55.41 or 55.43 will be given to all NRC licensed Senior Operators and Operators. Any weakness shown by the annual written examination will represent areas in which the licensee shall receive appropriate retraining. For a Senior Reactor Operator, the retraining for demonstrated weaknesses shall consist normally of study assignments or lectures or both. For a Reactor Operator, the retraining for demonstrated weaknesses shall consist of lectures together with study assignments. The corrected written annual examination will document competence and hence will be preserved for each licensed Senior Reactor Operator and Reactor Operator. For any examples of demonstrated weaknesses, a successful repeat written examination for those areas of inadequacy will document the sufficiency of retraining.

Since the annual written examination will be prepared and corrected by the Reactor Supervisor; who is also required to hold a Senior Operator License, these functions will be deemed adequate to demonstrate and document continuing competence.

The subject areas in Column 1 below will be sampled by the Reactor Operators' written examination and the subject areas in columns 1 and 2 will be sampled by the Senior Reactor Operators' written examination.

| Column 1 | Column 2 |
|---|---|
| Reactor Theory Fundamentals | Facility License |
| General Design Features | Technical Specifications |
| Primary System Features | Radiation Hazards |
| Secondary System Features | Fuel Movement |
| Operating Characteristics | Assessment of normal, |
| Reactivity Controls & Insts. | abnormal, and emergency |
| Control & Safety Systems | conditions. |
| Facility Procedures | |
| Radiation Monitoring | |
| Radioactive Material Handling | |

## 3 Operating Examination

An annual (not to exceed 14 months) operating examination, designed to evaluate operating competence, will be given to all NRC licensed Senior Reactor Operators and Reactor Operators. Any weakness shown by the operating examination shall be the focus of subsequent retraining.

A memorandum will be written and retained to document operating examinations and any related retraining activities. Operating examination tasks will typically be selected from activities such as reactor checkouts, reactor startups, major power changes, manual or automatic operations, square wave operations, pulse operations, power calibrations, or control rod calibrations. The operating examinations may be conducted during normal experiment operations and will be administered by the Reactor Supervisor. These functions will be deemed adequate to demonstrate and document continuing operational competence of the Reactor Supervisor.

## 4 Reactor Operations - On-the-Job-Training

Each Senior Reactor Operator and Reactor Operator will be required to perform certain minimum tasks during the requalification period (2 years) to insure continued competence in the areas listed in 10 CFR 55.59. These minimums are listed in Table 1.

## 5 Evaluation

1. The examinations will be graded by the Reactor Supervisor. The following written exam criteria will be used:

   (a) A grade of 80% or above on each section indicates satisfactory competence. No remedial action is required.

Table 1: Minimum tasks required by each SRO and RO during requalification period

| | | Senior Reactor Operator | Reactor Operator |
|---|---|---|---|
| a. | Reactivity Control Manipulations | 10 | 10 |
| b. | Daily Startup and Shutdown Checklists | 10 | 10 |
| c. | Participate in Monthly Checklists | 2 | 1 |
| d. | Participate in Pulse or Square Wave Operations | 1 | 1 |
| e. | Participate in Annual Emergency Drills | 1 | 1 |
| f. | Review all (1) facility design changes (2) procedure changes (3) facility license changes | within 20 days of issuance | within 20 days of issuance |
| g. | Review emergency procedures | 2 | 2 |

(b) A grade below 80% on any section signifies weakness in that area; therefore, remedial work and/or lectures on the deficient area will be required.

(c) An overall grade of less than 70% on the annual written examination will require the operator or senior operator be removed from licensed duties and placed on an accelerated training program until satisfactory performance is achieved.

2. The Reactor Supervisor will routinely observe and evaluate the performance of licensed operators and senior operators. This evaluation will include their actions during actual or simulated abnormal and emergency conditions. If an unsatisfactory performance is determined as a result of the Reactor Supervisor's evaluation, the licensed operator or senior operator will be removed from licensed duties and be placed on an accelerated training program. Licensed duties will not be resumed until satisfactory performance has been demonstrated by the operator.

## 6 Records

1. The corrected annual examination will be maintained in the Reactor Operator/Senior Reactor Operator License File.

2. Documentation of an Operator/Senior Reactor Operator's On-The-Job Training will be as follows:

   (a) Table 1, Section III, Parts a, b, c, d, and e will be contained within normal facility operating records.

   (b) Table 1, Part f: A routing sheet will be used with all facility design changes, procedure changes and facility license changes which will require the initials of all licensed Operators and Senior Operators, certifying their review and understanding of the change. A copy of this routing sheet will be maintained in the Operator Requalification Program file.

   (c) Table 1, Part g: A review and critique sheet will be prepared by the Reactor Supervisor, normally in the same time frame with scheduled emergency drills, to document that each Reactor Operator and Senior Operator has reviewed the contents of the Reactor Operations Manual, Section 7,

Emergency Procedures and is aware of any special problems or deficiencies noted in the last drill. A copy of this review sheet will be maintained in the Operators Requalification Program file.

(d) An annual statement by the Reactor Supervisor will be inserted in each Reactor Operators or Senior Reactor Operators file certifying continuing competency based on the (1) successful completion of the annual examinations plus any remedial work if required and (2) the actual performance of licensed duties as noted in Part 4-B.

3. When a licensed operator or senior operator has not actively performed licensed functions for a duration of at least four hours in any calendar quarter, he/she shall perform licensed functions for a minimum of six hours under the direction of a licensed Senior Reactor Operator, including at least one startup and shutdown checklist. A letter shall be sent to the NRC certifying this on-the-job training and that the qualifications and status of the licensee are valid. Resumption of unsupervised operation will not occur until after this training and certification have been performed.

# Environmental Report

USGS TRIGA Reactor

December 4, 2008

# Contents

# ENVIRONMENTAL REPORT

## 1 Introduction

This environmental report is prepared in accordance with 10 CFR 51 as part of the nuclear reactor license renewal at the USGS. The GSTR is a light-water cooled and moderated reactor using uranium fuel. The reactor has been operated since 1969 at steady-state thermal power levels up to 1000 kW. The reactor is housed on the Denver Federal Center, within the city limits of Lakewood, CO. A full description of the reactor is contained in the Facility Safety Analysis Report, License R-113, Docket 50-274. The GSTR supports research scientists specializing in geological and other science disciplines in addition to the use of the reactor as an education and training tool.

There are no safety considerations dependent on the duration of operations at the GSTR facility. Because of licensed power and operating history of the facility, there are no fuel burn up or material damage issues to be considered.

## 2 Proposed Actions

We propose to continue operating the GSTR as we have done since 1969. The GSTR has a nearly 40 year history of safe and reliable operations. The reactor was initially licensed in 1969 to operate at steady-state thermal power levels up to 1000 kW with pulsing operations allowed up to an insertion of \$3.00[†]. With the application to extend the operating license beyond the initial 40 years, authorization is sought to continue to operate the GSTR at a maximum steady-state thermal power of 1000 kW, with pulsing operations allowed up to an insertion of \$3.00.

## 3 Impact of the Proposed Actions on the Environment

The GSTR is operated solely for educational and research purposes which benefit the community, the country and the environment. Specific benefits include:

[†]Core reactivity is measured in arbitrary units of dollars and cents. The parameter is actually a dimensionless measure of the reactor's ability to produce neutrons.

## 3.1 Nuclear Education

### 3.1.1 Nuclear Engineering

Nuclear engineering is the principle discipline concerned with the safe release, control, and utilization of all types of energy from nuclear sources. Energy is needed to meet the world's technological needs and to maintain a suitable standard of living. Nuclear reactors are used to produce radioisotopes for diagnosis and therapy of disease, to produce research radio chemicals, and to provide energy sources for medical devices such as pacemakers and for probes to outer space. The engineering of safe nuclear power sources is vital to the future growth of the world.

The Colorado School of Mines (CSM) has a recently-established Nuclear Engineering Department. The CSM currently offers nuclear education at a graduate study level, providing a Masters of Science or Doctorate degree to successful candidates. All CSM Nuclear Engineering students are educated in fundamental nuclear engineering courses. One strength of the CSM Nuclear Engineering curriculum is the nearby access of the GSTR research reactor facility. The CSM program builds upon the foundations of mathematics, physics, thermal hydraulics, material science, radiation protection, radiation transport, interaction of radiation with matter and applied computer science. On top of this foundation, the GSTR serves as a training site for the nuclear engineering students. At the GSTR, the students gain practical experience in reactor operations, reactor safety, environmental concerns, health physics and interactive decision-making.

### 3.1.2 Health Physics

Health physics is a professional discipline based upon the scientific knowledge of, and the practical means for, radiation protection. The objective of a health physicist is to protect people and the environment from unnecessary exposure to radiation. Thus, the basic tenets of radiation must be understood, radiation knowledge explored, practical problems evaluated, radiation effects established and risk measurements relative to effect derived and implemented.

Health Physics students from the Colorado State University (CSU) Health Physics curriculum enter professional careers in health physics and medical physics, and the GSTR provides a valuable component to their education. The reactor provides the students with hands on, real world laboratory experience. It is at the reactor where the health physicist or medical physicist of the future learns how to monitor high intensity radiation sources for safety, how to implement emergency plans and how to monitor for environmental radiation.

### 3.1.3 Community

High school science classes use the GSTR for education and projects. The reactor is used by the Boy Scouts of America for earning the nuclear merit badge. Frequent tours and experiments are provided for junior high and high school students interested in science.. Members of the staff of the GSTR are active participants in training city and county emergency planning groups and periodically host emergency training, drills, and exercises.

## 3.2 Support of Scientific Programs

In addition to the educational programs, the GSTR offers support of research programs not only within the USGS but also for organizations throughout the country.

### 3.2.1 Nuclear Research Programs

The GSTR is supported by associated facilities, including specially equipped laboratories for neutron activation analysis and neutron-based geochronology. Extensive use of the reactor is made by researchers in the physical sciences as well as plant and animal sciences. Over the years, neutron activation analysis and geochronology work performed for geologists and chemists around the country has constituted the bulk of the scientific support work at the GSTR.

## 3.3 Education for Future Energy Needs

The demand for nuclear graduates supporting research into new-generation plants, engineering and operating staff, and health physicists at current (2008) levels is not being met. As the nuclear industry becomes more active, it is essential that the education infrastructure expand to ensure an educated nuclear workforce. The Colorado School of Mines is the only institution in the Rocky Mountain States with an active nuclear education program supported by a reactor; the GSTR.

# 4 Unavoidable Environmental Risks

Some low-level environmental risks cannot be eliminated. They include the use of nuclear fuel, the production of minimal gaseous effluents, the generation of some liquid and solid radioactive wastes, some waste heat, and some radiation exposure of personnel to radiation. None of these are considered significant with respect to environmental impact although each is individually assessed. They are:

## 4.1 Nuclear Fuel Cycle

The GSTR is designed for nominally [redacted] TRIGA fuel elements closely packed in a cylindrical core approximately [redacted]. Each fuel element contains uranium enriched less than 20% $^{235}$U in a ZrH matrix. The 8.5 to 12.0 weight % fuel contains up to [redacted] grams of $^{235}$U. Burnup is determined by power level and operating history, but historically 1000 kW operations has resulted in an average of approximately 42 MWD per year, depleting about 52 grams of $^{235}$U annually. Fuel burnup data for a four-years calculation is presented in Table 1.

Almost 40 years of experience at the GSTR shows that 1 MWD of burnup for TRIGA fuel results in a reactivity loss of about \$0.02. The data also indicate steady state operation at 1.0 MW results in xenon worth of approximately \$0.10 at 4 hours, \$0.70 at 8 hours (normal work day), and \$3.26 at 40 hours (steady state). Fuel temperature negative temperature reactivity during 1000 kW operations will require approximately \$2.85 of excess reactivity. Therefore, 4 hours of operation will require excess reactivity of \$2.95, and 8 hours of operation will require excess reactivity of \$3.55. The proposed maximum excess reactivity limit of \$6.80 permits routine operating times of 8 hours per operation for about 4 years. After these 4 years, it will be necessary to replace fuel in order to support operations. Therefore, fuel element replacement adequate to compensate for burnup will be required about five times over the new license period, and fuel removed will require storage at the facility until disposal.

The GSTR has a small quantity of usable fuel (fresh and lightly burned) in storage. Useful fuel inventory (excluding not-useful elements or elements not likely to be used) is capable of adding about 245 grams of $^{235}$U to the core, adequate to replace about half of the quantity of $^{235}$U expected to be depleted by burnup over the 20-year renewed license period. [redacted]. Additional pool storage racks are available but not currently installed, and it is possible to fabricate additional racks on demand. Therefore, spent fuel storage at the GSTR is adequate to accommodate anticipated needs.

Table 1: Projected Burnup and Reactivity

| Year | $^{235}U$ (g) Burned | $\delta k$ ($) | Lost Excess $\delta k$ ($) |
|---|---|---|---|
| 1 | 21 | $0.84 | $5.96 |
| 2 | 42 | $1.68 | $5.12 |
| 3 | 63 | $2.52 | $4.28 |
| 4 | 84 | $3.36 | $3.44 |

Since fuel is already present at the GSTR facilities with storage and utilization adequately managed, the only impacts of continued operation will be utilization of fuel elements in storage and the acquisition of additional fuel at some point during the 20-year period, requiring ultimate disposal of the fuel as spent fuel.

## 4.2 Releases of Solid Radioactive Waste

The major portion of solid radioactive wastes includes clean-up resins from the demineralizer systems and filters used in treating water for the demineralizer system. Other solid radioactive wastes include absorbent paper, plastic gloves, spent samples, some contaminated laboratory apparatuses, spent standards, etc.

The reactor pool is maintained at low chemical contamination by a demineralizer. The resin is exhausted every two to three years, with about 3 cu ft replaced. The resin is aggregated for disposal as solid radioactive waste until a significant quantity can be collected for disposal, and decays significantly during aggregation. Based on historical values, in no case is total activity of any radionuclide expected to exceed 400 $\mu$Ci for a shipment. Some routine maintenance activities result in radioactive waste

The USGS holds a radioactive materials license issued by the NRC, and is required to have a radiation protection program approved by the NRC. Activities using radioactive material at the USGS are conducted under the NRC materials license; the GSTR is one source of radioisotopes and activated material supporting USGS research programs. The bulk of solid radioactive waste generated by the reactor during the conduct of research and experiments under the NRC materials license is held for decay, with a small quantity infrequently disposed of through burial.

The recent history of solid waste transfer from the GSTR is contained in Table 2.

Table 2: Solid Radioactive Waste Transfer

| Year | Quantity Transferred |
|---|---|
| 1999-2000 | 15 cu ft |
| 2001-2002 | 15 cu ft |
| 2003-2004 | 7.5 cu ft |
| 2005-2006 | 7.5 cu ft |
| 2007-2008 | 7.5 cu ft |

## 4.3 Releases of Liquid Wastes

Liquid radioactive wastes are disposed of by evaporating the water and disposing of the residue as solid waste. The only routine liquid waste created is excess water from the ion exchange resin replacement operation. Very infrequent collection of contaminated water from under the reactor tank occurs. Again, this water is evaporated and the residue is disposed of as solid waste. There are typically no liquid discharges related to reactor operation.

## 4.4 Release of Radioactive Gases

In practice, $^{41}Ar$ and $^{3}H$ gases are the only gaseous radioactive effluents emitted from the reactor. The $^{41}Ar$ release rate is continuously monitored by a detector located within the effluent emission stack. Chapter 11.1.1.1.1 of the Facility Safety Analysis Report shows that at full power continuous operations, the maximum off-site annual dose would be only 17 mrem (0.017 mSv) from an annual release of 25 Ci, well within applicable limits. The $^{3}H$ release is from evaporation of the reactor tank water which is monitored. The $^{3}H$ concentration in the tank water is analyzed periodically and the $^{3}H$ release is appropriately calculated.

The GSTR staff routinely monitors for other potential effluent noble gases, halogens, or particulate matter. None have been detected in normal operations. During accident conditions, detection of airborne particulate releases would prompt reactor shutdown and filtered ventilation.

## 4.5 Radiation Exposure of Personnel

The minimum reliable dose for the personnel monitoring system is 10 mrem. Only the GSTR staff are normally monitored at the facility, so the number of dosimeters is small. The personnel exposure records for the last 20 years of operation are provided in Table 3.

The GSTR has an active ALARA policy in place. In essence, the program attempts to achieve, through engineering controls and thorough planning, detailed procedures to minimize radiation exposures to as low as possible. The occupational dose ALARA objective for the GSTR is 1 rem annual limit and an average of no more than 0.5 rem per year for the maximally exposed worker under normal conditions. The ALARA objective for the GSTR is 50 mrem per year, direct exposure, to any member of the public. The ALARA goal for facility effluents is 10 mrem per year to the nearest public receptor.

## 4.6 Environmental Radiation Exposure

Since there are no horizontal beam tubes at the GSTR, there is no potential for exposure from the direct beam to anyone outside the facility from such beams. Two possible vertical beams are infrequently used and access to the reactor bay roof is restricted access.

Facility radiation levels are periodically checked, including verification that the radiation levels in areas adjacent to the reactor bay (direct and scattered radiation) are either less than the levels required for a restricted area, or the areas are controlled as restricted areas.

Sources of environmental radiation exposure include leakage through the biological shielding, activated material in the primary coolant system, and scattered radiation from extracted beams. Environmental Radiation Exposures around the reactor are monitored with fixed dosimeters. For the period 1998-2008, exposure rates outside of the reactor bay during full power operation averaged 9.78 mR/hour, with background approximately 0.02 mR/hour. The most recent 5-year average (2002-2007) power history resulted in. an annual average of 8.87 mrem at the highest point at the west fence around the reactor bay.

Table 3: Numbers of persons in annual-dose categories

| Year | Immeasurable[a] | <0.1 rem | 0.1-0.5 rem | >0.5 rem |
|---|---|---|---|---|
| 2007 | 1 | 2 | 3 | - |
| 2006 | - | 2 | 3 | - |
| 2005 | - | - | 4 | - |
| 2004 | 1 | 4 | - | - |
| 2003 | 3 | 4 | - | - |
| 2002 | 1 | 3 | 1 | - |
| 2001 | - | 5 | - | - |
| 2000 | 4 | 1 | - | - |
| 1999 | 3 | 1 | - | - |
| 1998 | 1 | 2 | 1 | - |
| 1997 | - | 3 | - | - |
| 1996 | 3 | - | - | - |
| 1995 | - | 4 | - | - |
| 1994 | 3 | 1 | - | - |
| 1993 | - | 3 | 1 | - |
| 1992 | - | 2 | 2 | - |
| 1991 | - | 1 | 3 | - |
| 1990 | - | 3 | 1 | - |
| 1989 | 1 | 3 | - | - |
| 1988 | 1 | 3 | - | - |
| 1987 | - | 4 | - | - |

## 4.7 Secondary Cooling

Heat generated in the reactor is passed via primary coolant loop through a plate type heat exchanger where the heat is transferred to a secondary loop. The secondary cooling loop rejects heat through a standard, commercially supplied forced draft cooling tower. The cooling tower discharge path is well within the restricted access area of the GSTR. Secondary water is potable water and is not treated with any chemicals.

### 4.7.1 Heat

The heat rejection from the GSTR reactor cooling tower is insignificant in contrast to the large capacity heating and air conditioning units that support the buildings on the Denver Federal Center.

### 4.7.2 Makeup Water

Since the secondary cooling system relies on air cooling to reject heat produced by the reactor to the environment, some water is lost to (minimally) drift and (principally) evaporation. Therefore, water is periodically required to be added to make up for losses. The rate of evaporation varies according to environmental conditions, and is compensated for (as needed) by a float-actuated makeup valve that opens and closes to maintain the secondary water tank level within a specified range.

The cooling tower draft contains high humidity, which is quickly dissipated by environmental conditions. There is no impact on the environment from the increased water utilization.

There is no measurable environmental impact associated with operation of the secondary cooling system.

# 5 Additional Environmental Benefits

## 5.1 Provision of Short Half Life Radioisotopes

The availability of a nuclear reactor provides researchers the opportunity to use short half life radioisotopes unavailable to users that do not have access to an on campus reactor. For example, $^{24}$Na is a typical sodium isotope used for radiography techniques. $^{24}$Na has a 15 hour half-life. Thus, $^{24}$Na can be held for radioactive decay for one week when all the radioactivity has dissipated. In contrast, use of a long-lived radioisotope would require radioactive waste disposal in a permitted site

## 5.2 Public Awareness of Environmental Energy Alternatives

The GSTR facility and staff provides an open forum for the education about alternative energy sources. An informed public can make informed decisions. Because of the GSTR, the USGS employs staff with an expertise in nuclear science. This expertise is used to advise on radiological safety and alternative energy sources and issues germane to the local community. Such issues include regulations, radiation safety and environmental control for nearby universities, colleges and schools, industry and resolution of legal issues regarding ionizing radiation.

# 6 Alternatives to the Continued Operation of the Reactor

There is no comparable alternative facility. If the reactor is not relicensed, the quality of science for the USGS and education for Colorado nuclear engineers and health physicists will be diminished. Research projects will come to a halt. The forward progress of nuclear science technology will be decreased.

# 7 Relationship Between Local Short Term Uses and Long Term Benefits

The short term use of the GSTR centers around the earth science research of the USGS, education of nuclear engineers, health physicists, research scientists and the general education of the students and community about nuclear energy and radioisotopes.

The long term contribution that the GSTR provides comes from the many contributions to society made by students and scientists to the country. Numerous novel techniques have been developed over the past forty years by USGS scientists at the GSTR. Some of these are widely used throughout the world. The GSTR serves as radiation science incubator of ideas and products.

The continued operation of the GSTR is not an irreversible commitment. Changes in programs, extent of operations, and potential decommissioning are all equally possible at any time in the future.

# 8 Analysis

The GSTR is an important research and education facility. It is an integral part of the USGS research activities and is an essential tool for neutron-based research. It has no significant adverse environmental

impact. Radiation exposures to members of the local public are insignificant and unmeasurable when related to the variation in natural radiation in the same area.

The GSTR is already in operation. New capitalization funds are not necessary. It is the most prudent use of taxpayer's money to continue operation of the reactor. At this point in time, initial capital investment costs have been paid off. All technology, science, education and services rendered now are at minimal cost. Thus the resultant benefit/cost ratio is very high.

## 9 Long Term Effects on the Environment

At the end of its useful life, the GSTR site will be returned to general use for the Federal Government. When finished, the fuel rods will be sent to a DOE facility where the unspent uranium will be recovered and the radioactive byproducts recovered for commercial use or packaged and shipped for disposal through commercial radioactive waste disposal brokers.

The long term effects on the environment from renewing the operating license for the GSTR are insignificant.