UNIVERSITY OF TEXAS AT AUSTIN
RESEARCH REACTOR
LICENSE NO. R-129
DOCKET NO. 50-602

UNIVERSITY OF TEXAS AT AUSTIN
LICENSE RENEWAL APPLICATION
DECEMBER 12, 2011

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

***REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS**

Department of Mechanical Engineering
THE UNIVERSITY OF TEXAS AT AUSTIN

Nuclear Engineering Teaching Laboratory • Austin, Texas 78758
512-232-5370 • FAX 512-471-4589 • http://www.me.utexas.edu/~netl/

December 12, 2011

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

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

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

Sir:

In accordance with direction provided by L. N. Tranh (ADAMS ML110040316), we respectfully request renewal of the University of Texas at Austin TRIGA II nuclear research reactor located at the Nuclear Engineering Teaching Laboratory at the University of Texas at Austin. Enclosed you will find:

1) A completed, updated Safety Analysis Report (SAR)
2) Financial qualifications specified in 10CFR50.33 is incorporated in the SAR Chapter 15;
   a. None of the provision of 10CFR50.33(d) apply.
   b. Based on current budget and expenditures, estimated annual operating costs with the source of funding indicated for the first 5-year period after license renewal is incorporate in Chapter 15.
3) Financial qualifications regarding decommissioning is provided in Chapter 15,
   a. An estimate of decommissioning based on NR guidance,
   b. A statement of intent regarding intent to seek support for decommissioning at the appropriate time,
   c. A description of cost adjustment of decommissioning costs is provided with the current estimate based on the methodology,
   d. Documentation that the University of Texas is a State agency and a State of Texas government licensee under 10CFR50.75€(2)(iv), and that University funding obligations are backed by the State .
4) An Environmental Report
5) The Technical Specifications
6) The University of Texas facility has currently approved programs on file with the NRC for Operator Requalification Program, Emergency Plan, and Physical Security Plan, and does not propose changes in these plans and programs at this time.

A020
NRR

Your attention in this matter is greatly appreciated.

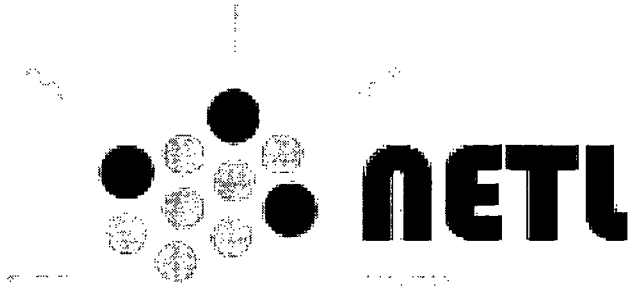
I declare under penalty of perjury that the foregoing is true and correct.

If you have any questions relating to this submission, please feel free to contact me by phone at 512-232-5380 or by email at biegalski@mail.utexas.edu.

Regards,

Steven Biegalski, Ph.D., P.E.
Director, Nuclear Engineering Teaching Laboratory

Cc: Paul Whaley, Associate Director, NETL



Safety Analysis Report

The University of Texas at Austin
Nuclear Engineering Teaching Laboratory
TRIGA Mark II Nuclear Research Reactor

License R-129
Docket 50-602
12 December 2011

The University of Texas at Austin
Nuclear Engineering Teaching Laboratory
10100 Burnet Rd, Bldg 159
Austin, TX 78758

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1. THE FACILITY

This report describes the research reactor operated by the University of Texas at Austin. This report provides the basis for a safety evaluation demonstrating the facility and the reactor does not cause undue risk to the health and safety of the public. This chapter of the Safety Analysis Report reflects and summarizes descriptions and analyses in the individual chapters, and will provide:

- Introduction/overview
- Summary and conclusions on principle safety considerations
- General facility description
- Overview of shared facilities and equipment
- Comparison with similar facilities
- Summary of operations
- Compliance with NWPA of 1982
- Facility modifications & history

1.1 Introduction

The University of Texas operates a 1.1 MW TRIGA II research reactor (with pulsing to a maximum permitted reactivity addition of 2.2% Δk/k) at the Pickle Research Campus (PRC), approximately 10 miles north of the main campus in Austin, Texas. A more complete description of the general facility location and location within the PRC is provided in Chapter 2. This Safety Analysis Report provides information and analysis to demonstrate that there is reasonable assurance operations for an additional 20 year term do not significantly challenge safety. Analysis shows a large margin to thermal hydraulic conditions that might lead to a challenge of fuel cladding using passive, natural convection cooling.

The reactor is located in the Nuclear Engineering Teaching Laboratory (NETL), a building that houses an operating unit of the UT Department of Mechanical Engineering in the Cockerel School of Engineering. The NETL serves a multipurpose role, with the primary function as a "user facility" for faculty, staff, and students of the College of engineering. The facility supports the Nuclear and Radiation Engineering program of the Department of Mechanical Engineering for laboratory exercises in UT courses, undergraduate research, and graduate research. The NETL supports educational programs for other organizations and institutions notably (but not limited to) the Big-12 Consortium and Historically Black College and Universities. The facility supports development and application of nuclear methods for researchers from other universities, industry, and government organizations. The NETL provides nuclear analytic

services to researchers, industry, and other research and industrial laboratories for testing and evaluation of materials. The NETL provides public education through tours and demonstrations.

1.2 Summary and conclusions on principle safety considerations

The decision to build a new TRIGA was based on historical experience with a TRIGA II on main campus. Space considerations on main campus and a well-established infrastructure at the PRC campus led to facility siting.

TRIGA II reactors routinely operate at power levels up to approximately 2 MW with natural convection. At power levels less than 2 MW, fission product inventory is limited enough that emergency planning requirements are somewhat simplified. Therefore, 1.1 MW was initially selected as the maximum steady state license limit providing a large margin to thermal limits and complex emergency planning.

Heat generation in TRIGA fuel produces less than ½ of critical heat flux with natural convection at power levels up to about 2 MW (see Chapter 5). The initial license power level of 1.1 MW provides an extremely large margin to thermal hydraulic limits in passive, natural circulation. The TRIGA ZrH fuel inherently reduces the potential for thermal fission as fuel temperature increases so that temperature increases with operation at power intrinsically limit maximum steady state power level. The TRIGA fuel design retains a large fraction of fission products generated during operation, with stainless steel cladding acting as a passive barrier to release for the fission products that escape the fuel matrix.

The NETL TRIGA shielding was designed to limit personnel exposure rates from radiation generated during reactor operation in accessible areas of the pool and shield structure at 1.5 MW to less than 1 mrem/hr. The maximum dose rate is shown to be at floor level. Current experimental programs at the beam ports limit routine access to the biological shielding surface near the core. Additional shielding information is provided in Chapters 3, 4 and 10.

The principle off-site exposure source term during normal operations is ^{41}Ar, a noble gas with a 110 minute half life. Stack effluent is limited to maintain receptor doses to 109CFR20 limits, as discussed in Chapters 9 and 11. There are no routine liquid releases, and the production of radioactive waste during normal operations is extremely limited (with most radioactive waste held for decay). Accident analysis (Chapter 13) demonstrates potential consequences from postulated scenarios do not result in unacceptable consequences.

The reactor design has many safety features, including a large margin to thermal hydraulic limits, passive cooling, robust shielding, fuel matrix characteristics, and stainless steel

cladding. Buildup of radioactive materials in the facility is controlled by a dynamic confinement and an argon purge system.

1.2 General description of the facility

A. Site

Land development in the area of the current NETL installation began as an industrial site during the 1940's. Following the 1950's, lease agreements between the University and the Federal government led to the creation of the Balcones Research Center. The University became owner of the site and in 1994 the site name was changed to the J.J. Pickle Research Campus (PRC) in honor of retired U.S. Congressman James "Jake" Pickle.

The PRC is a multidiscipline research campus on 1.87 square kilometers. The site consists of two approximately equal areas, east and west. An area of about 9000 square meters on the east tract is the location of the NETL building. Sixteen separate research units and at least five other academic research programs conduct research on the PRC. Adjacent to the NETL site are the Center for Research in Water Resources, the Bureau of Economic Geology, and the Research Office Complex, illustrating the diverse research activities on the campus. A Commons Building provides cafeteria service, recreation areas, meeting rooms, and conference facilities. A more complete description of the environment surrounding the NETL is provided in Chapter 2.

B. Building

[redacted]

One of the primary laboratories contains the TRIGA reactor pool, biological shield structure, and neutron beam experiment area. A second primary laboratory has walls 1.3 meter (4.25 ft) thick for use as a general purpose radiation experiment facility. Other areas of the building include shops, instrument & measurement laboratories, and material handling facilities. An Annex was installed adjacent to the NETL building in 2005, a 24 by 60 foot modular building. The annex provides classroom space and offices for graduate students working at the NETL.

C. Reactor

The largest room in the NETL building is a vault type enclose that serves as a confinement volume for the UT TRIGA nuclear research reactor. The TRIGA Mark II reactor is a versatile and inherently safe research reactor conceived and developed by General Atomics to meet education and research requirements. The UT-TRIGA reactor provides sufficient power and

neutron flux for comprehensive and productive work in many fields including physics, chemistry, engineering, medicine, and metallurgy

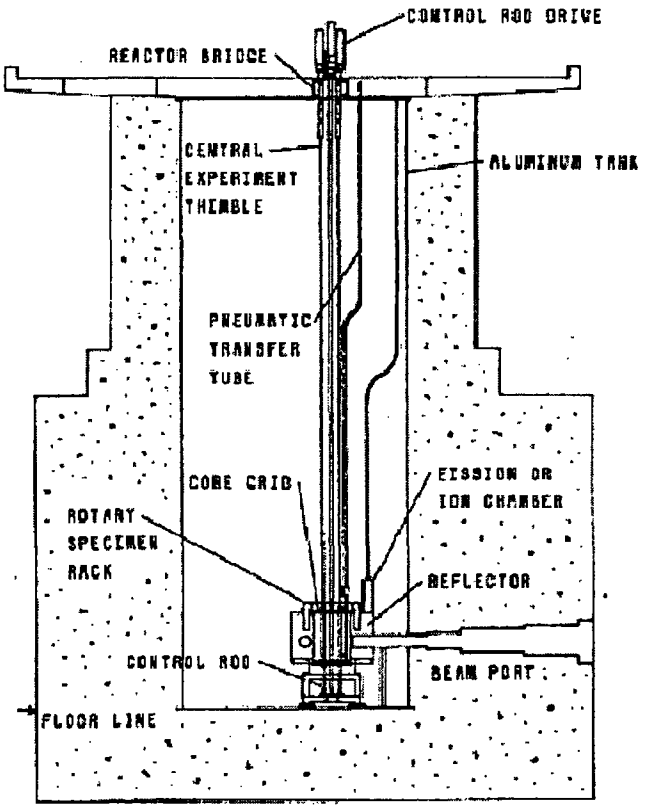

Figure 1.1, UT TRIGA Mark II Nuclear Research Reactor

The NETL UT-TRIGA reactor is an above-ground, fixed-core research reactor. The reactor core is located at the bottom of an 8.2 meter deep water-filled tank surrounded by a concrete shield structure. The water serves as a coolant, neutron moderator, and radiation shield. The reactor core is surrounded by a reflector, a 1 foot thick graphite cylinder.

C.1 Reactor Core.

The reactor core is an assembly of cylindrical fuel elements surrounded by an annular graphite neutron reflector. Fuel elements are positioned by an upper and lower grid plate, with penetrations of various sizes in the upper grid plate to allow insertion of experiments. Each fuel element consists of a fueled region with graphite sections at top and bottom, contained in a thin-walled stainless steel tube. The fuel region is a metallic alloy of low-enriched uranium in a zirconium hydride (UZrH) matrix. Physical properties of the TRIGA fuel provide an inherently safe operation. Rapid power transients to high powers are automatically suppressed without using mechanical control; the reactor quickly and automatically returns to normal power levels. Pulse operation, a normal mode, is a practical demonstration of this inherent safety feature.

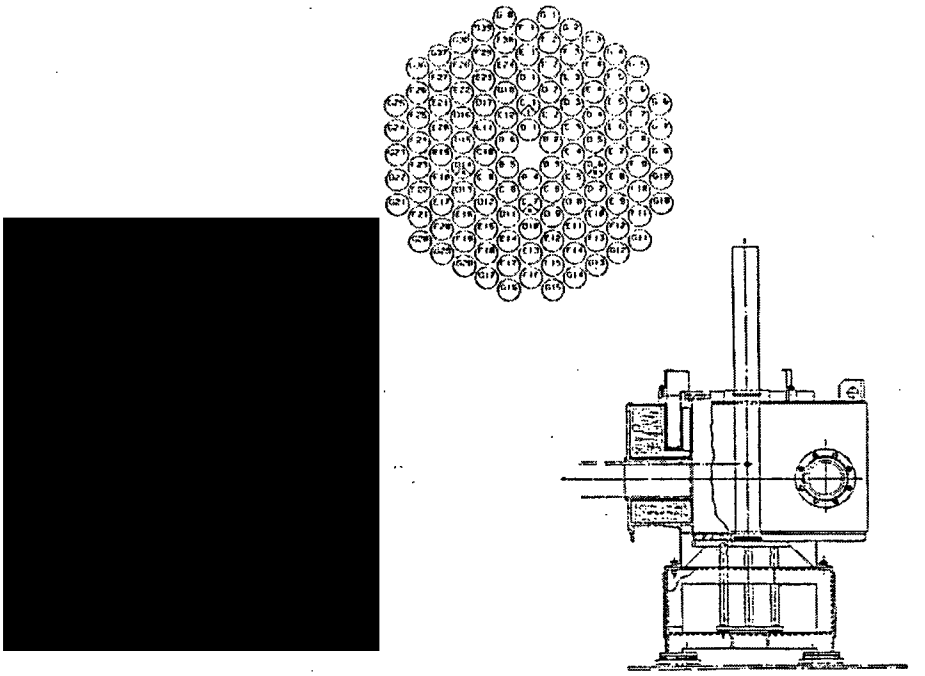

Figure 1.2, Core and Support Structure Details

C.2 Reactor Reflector.

The reflector is a graphite cylinder in an aluminum-canister. A 10" well in the upper surface of the reflector accommodates an irradiation facility, the rotary specimen rack (RSR), and horizontal penetrations through the side of the reflector allow extraction of neutron beams. In 2000 the canister was flooded to limit deformation stemming from material failure in welding joints. In 2004, the reflector was replaced with some modification, including a modification to the upper grid plate for more flexible experiment facilities.

D. Reactor Control.

The UT-TRIGA research reactor can operate continuously at nominal powers up to 1.1 MW, or in the pulsing mode with maximum power levels up to about 1500 MW (with a trip setpoint of 1750 MW) for durations of about 10 msec. The pulsing mode is particularly useful in the study of reactor kinetics and control. The power level of the UT-TRIGA is controlled by a regulating rod, two shim rods, and a transient rod. The control rods are fabricated with integral extensions containing fuel (regulating and shim rods) or air (transient rod) that extend through the lower grid plate for full span of rod motion. The regulating and shim rods are fabricated from B_4C contained in stainless steel tubes; the transient rod is a solid cylinder of borated graphite clad in aluminum. Removal of the rods from the core allows the rate of neutron induced fission (power) in the UZrH fuel to increase. The regulating rod can be operated by an automatic control rod that adjusts the rod position to maintain an operator-selected reactor

power level. The shim rods provide a coarse control of reactor power. The transient rod can be operated by pneumatic pressure to permit rapid changes in control rod position. The transient rod moves within a perforated aluminum guide tube. Details of the control rods are provide in Chapter 4.

The UT-TRIGA research reactor rod control system uses a compact microprocessor-driven control system. The digital control system provides a unique facility for performing reactor physics experiments as well as reactor operator training. This advanced system provides for flexible and efficient operation with precise power level and flux control, and permanent retention of operating data. A more complete description of the rod control system is provided in Chapter 7.

E. Experiment Facilities.

Facilities for positioning samples or apparatus in the core region include cut-outs fabricated in the upper grid plate, a central thimble in the peak flux region of the core, a rotary specimen rack in the reactor graphite reflector, and a pneumatically operated transfer system accessing the core in an in-core section. Beam ports, horizontal cylindrical voids in the concrete shield structure, allow neutrons to stream out away from the core. Experiments may be performed inside the beam ports or outside the concrete shield in the neutron beams. Areas outside the core and reflector are available for large equipment or experiment facilities. A brief description of the facilities follows; a more complete description is provided in Chapter 10.

E.1 Upper Grid Plate 7L and 3L Facilities

The upper grid plate of the reactor contains four removable sections configured to provide space for experiments otherwise occupied by fuel elements (two three-element and two seven-element spaces). Containers can be fabricated with appropriate shielding or neutron absorbers to tailor the gamma and neutron spectrum to meet specific needs. Special cadmium-lined facilities have been constructed that utilize three element spaces.

E.2 Central Thimble

The reactor is equipped with a central thimble for access to the point of maximum neutron flux. The central thimble is an aluminum tube extending through the central penetration of the top and bottom grid plates. Typical experiments using the central thimble include irradiation of small samples and the exposure of materials to a collimated beam of neutrons or gamma rays.

E.3 Rotary Specimen Rack (RSR)

A rotating (motor-driven) multiple-position specimen rack located in a well in the top of the graphite reflector provides for irradiation and activation of multiple samples and/or batch

production of radioisotopes. Rotation of the RSR minimizes variations in exposure related to sample position in the rack. Samples are loaded from the top of the reactor through a tube into the RSR using a specimen lifting device. A design feature provides the option of using pneumatic pressure for inserting and removing samples.

E.4 Pneumatic Tubes

A pneumatic transfer system supports applications using short-lived radioisotopes. The in-core terminus of the system is normally located in the outer ring of fuel element positions, with specific in-core sections designed to support thermal and epithermal irradiations. The sample capsule is conveyed to a sender-receiver station via pressure differences in the tubing system. An optional transfer box permits the sample to be sent and received to three different sender-receiver stations. One station is in the reactor confinement, one is in a fume hood in a laboratory room, and the third operates in conjunction with an automatic sample changer and counting system.

E.5 Beam Port Facilities

Five neutron beam ports penetrate the concrete biological shield and reactor water tank at core level, as shown in Fig.1.3. The beam ports were designed with different characteristics to accommodate a wide variety of experiments. Specimens and/or equipment supporting experiment programs may be placed inside a beam port or outside the beam port in a neutron beam from the beam port.

Shielding reduces radiation levels outside the concrete biological shield to safe values when beam ports are not in use. Beam port shielding is configured with an inner shield plug, outer shield plug, lead-filled shutter, and circular steel cover plate. A neutron beam coming from a beam port may be modified by using collimators, moderators and/or neutron filters. Collimators are used to limit beam size and beam divergence. Moderators and filters are used to change the energy distribution of neutrons in beams (e.g., cold moderator).

E.5 (1) Beam Port 1 (BP1)

BP1 is connected to BP5, forming a through port. The through port penetrates the graphite reflector tangential to the reactor core, as seen in Figure 5-2. This configuration allows introduction of specimens adjacent to the reactor core to gain access from either side of the biological shield, and can provide beams of thermal neutrons with relatively low fast-neutron and gamma-ray contamination.

A reactor-based slow positron beam facility is being fabricated at BP1. The facility (Texas Intense Positron Source) will be one of a few reactor-based slow positron beams in the world. The Texas Intense Positron Source concept includes a copper source, a source transport system, a combined positron moderator/remoderator assembly, a positron beam line and a sample chamber.

E.5 (2) Beam Port 2 (BP2)

BP2 is a tangential beam port, terminating at the outer edge of the reflector. A void in the graphite reflector extends the effective source of neutrons into the reflector for a thermal neutron beam with minimum fast-neutron and gamma-ray backgrounds. Tangential beams result in a "softer" (or lower average-) energy neutron beam because the beam consists of scattered reactor neutrons. BP2 is configured to support neutron depth profiling applications, with a prompt-gamma neutron activation analysis sharing the beam port.

Neutron Depth Profiling (NDP) Some elements produce charged particles with characteristic energy in neutron interactions. When these elements are distributed near a surface, the particle energy spectrum is modulated by the distance the particle traveled through the surface. NDP uses this information to determine the distribution of the elements as a function of distance to the surface.

Prompt-Gamma Neutron Activation Analysis (PGNAA) Characteristic gamma radiation is produced when a neutron is absorbed in a material. PGNAA analyzes gamma radiation to identify the material and concentration in a sample. PGNAA applications include: i) determination of B and Gd concentration in biological samples which are used for Neutron Capture Therapy studies, ii) determination of H and B impurity levels in metals, alloys, and semiconductor, iii) multi-element analysis of geological, archeological, and environmental samples for determination of major components such as Al, S, K, Ca, Ti, and Fe, and minor or trace elements such as H, B, V, Mn, Co, Cd, Nd, Sm, and Gd, and iv) multi-element analysis of biological samples for the major and minor elements H, C, N, Na, P, S, Cl, and K, and trace elements like B and Cd.

E.5 (3) Beam Port 3 (BP3)

BP3 is a radial beam port. BP3 pierces the graphite reflector and terminates at the inner edge of the reflector. This beam port permits access to a position adjacent to the reactor core, and can provide a neutron beam with relatively high fast-neutron and gamma-ray fluxes. BP3 contains the Texas Cold Neutron Source Facility, a cold source and neutron guide system.

Texas Cold Neutron Source. The TCNS provides a low background subthermal neutron beam for neutron reaction and scattering research. The TCNS consists of a cooled moderator, a heat pipe, a cryogenic refrigerator, a vacuum jacket, and connecting lines. The TCNS uses eighty milliliters of mesitylene moderator, maintained by the cold source system at ~36 K in a chamber within the reactor graphite reflector. A three-meter aluminum neon heat pipe, or thermosyphon, is used to cool the moderator chamber. The heat pipe working fluid evaporates at the moderator chamber and condenses at the cold head.

Cold neutrons from the moderator chamber are transported by a 2-m-long neutron guide inside the beam port to a 4-m-long neutron guide (two 2-m sections) outside the beam port. Both neutron guides have a radius of curvature equal to 300 m. All reflecting surfaces are coated with Ni-58. The guide cross-sectional areas are separated into three channels by 1-mm-thick vertical walls that block line-of-sight radiation streaming.

Prompt Gamma Focused-Neutron Activation Analysis Facility The UT-PGAA facility utilizes the focused cold-neutron beam from the Texas Cold Neutron Source. The PGAA sample is located at the focal point of the converging guide focusing system to provide an enhanced reaction rate with lower background at the sample-detector area as compared to other facilities using filtered thermal neutron beams. The sample handling system design permits the study of a wide range of samples and quick, reproducible sample-positioning.

The neutron guide and capillary focusing assembly may be used independent of the TCNS utilization.

E.5 (4) Beam Port 4 (BP4)

BP4 is a radial beam port that terminates at the outer edge of the reflector. A void in the graphite reflector extends the effective source of neutrons to the reactor core. This configuration is useful for neutron-beam experiments which require neutron energies higher than thermal energies. BP4 was configured in 2005 to support student laboratories.

E.5 (5) Beam Port 5 (BP5)

A Neutron Radiography Facility is installed at BP5. Neutrons from BP5 illuminate a sample. The intensity of the exiting neutron field varies according to absorption and scattering characteristics of the sample. A conversion material generates light proportional to the intensity of the neutron field as modified by the sample.

F Other Experiment and Research Facilities

The NETL facility makes available several types of radiation facilities and an array of radiation detection equipment. In addition to the reactor, facilities include a subcritical assembly, various radioisotope sources, machine produced radiation fields, and a series of laboratories for spectroscopy and radiochemistry.

1.3 Overview of Shared Facilities and Equipment

Utilities are provided (underground) by the Pickle research Campus infrastructure. Chill water for HVAC and pool cooling is provided by a central chill water plant. Electrical power is provided by a transformer near the NETL.

1.3.3 Other TRIGA Facilities

The inherent safety of this TRIGA reactor has been demonstrated by the extensive experience of similar TRIGA systems throughout the world. Forty-eight TRIGA reactors are now in operation world-wide, and 31 of these are pulsing reactors. TRIGA reactor installations in the U.S. are reflected in Table 1.1 (shutdown or decommissioned) and 1.2 (currently operating). TRIGA reactors have more than 450 reactor years of operating experience, over 30,000 pulses, and more than 15,000 fuel element years of operation. The safety arises from a large, prompt negative 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. The result is that reactor power excursions are terminated quickly and safely.

The prompt shutdown mechanism has been demonstrated extensively in many thousands of transient tests performed on two prototype TRIGA reactors at the GA Technologies

laboratory in San Diego, California, as well as other pulsing TRIGA reactors in operation. These tests included step reactivity insertions as large as 3.5% Ok/k with resulting peak reactor powers up to 8400 MW(t) on TRIGA cores containing similar fuel elements as are used in this TRIGA reactor.

Because the reactor fuel is similar, the experience and tests form other TRIGA installations apply to this TRIGA system. As a result it has been possible to use accepted safety analysis techniques applied to other TRIGA facilities to update evaluations with regard to the characteristics of this facility.

Table 1.1, SHUTDOWN OR DECOMMISSIONED U.S. TRIGA REACTORS

	thermal power	type	initial crit
GA-TRIGA III	1,500.00	TRIGA MARK III	1/1/1966
TRIGA MK F, NORTHRUP	1,000.00	TRIGA MARK F	1/1/1963
UT TRIGA UNIV TEXAS	1,000	TRIGA MARK I	1/1/1963
BRR UC BERKELEY	1,000	TRIGA MARK III	8/10/1966
TRIGA MK I MICH ST UNIV	250	TRIGA MARK I	3/21/1969
TRIGA COLUMBIA UNIV	250	TRIGA MARK II	1/1/1977
TRIGA PUERTO RICO NUC CTR	2,000	TRIGA CONV	8/1/1960
UI-TRIGA UNIV. ILLINOIS	1,500	TRIGA MARK II	7/23/1969
NRF NEUTRON RAD FACILITY	1,000	TRIGA MARK I	3/1/1977
TRIGA CORNELL	500	TRIGA MARK II	1/1/1962
DORF TRIGA MARK F	250	TRIGA MARK F	1/1/1961
ATUTR	250	TRIGA MARK I	1/1/1989
GA-TRIGA F	250	TRIGA MARK I	7/1/1960
GA-TRIGA I	250	TRIGA MARK I	5/3/1958
UI-TRIGA MK I	100	TRIGA MARK I	8/1/1960
TRIGA, VET. ADMIN.	20	TRIGA MARK I	6/26/1959

Table 1.2, U.S. OPERATING RESEARCH REACTORS USING TRIGA FUEL

	thermal power	type	initial crit
ANN. CORE RES. REACTOR (ACRR)	4,000	TRIGA ACPR	6/1/1967
UC DAVIS/MCCLELLAN N. RAD. CENTER	2,000	TRIGA MARK II	1/20/1990
OSTR, OREGON STATE UNIV.	1,100	TRIGA MARK II	3/8/1967
TRIGA II UNIV. TEXAS	1,100	TRIGA MARK II	3/12/1992
NSCR TEXAS A&M UNIV.	1,000	TRIGA CONV	1/1/1962
UWNR UNIV. WISCONSIN	1,000	TRIGA CONV	3/26/1961
WSUR WASHINGTON ST. UNIV.	1,000	TRIGA CONV	3/13/1961
PSBR PENN ST. UNIV.	1,000	TRIGA MARK CONV	8/15/1955
AFRRI TRIGA	1,000	TRIGA MARK F	1/1/1962

Table 1.2, U.S. OPERATING RESEARCH REACTORS USING TRIGA FUEL

	thermal power	type	initial crit
ANN. CORE RES. REACTOR (ACRR)	4,000	TRIGA ACPR	6/1/1967
GSTR GEOLOGICAL SURVEY	1,000	TRIGA MARK I	2/26/1969
DOW TRIGA	300	TRIGA MARK I	7/6/1967
ARRR	250	TRIGA CONV	7/9/1964
RRF REED COLLEGE	250	TRIGA MARK I	7/2/1968
UCI, IRVINE	250	TRIGA MARK I	11/25/1969
KSU TRIGA MK II	1,250	TRIGA MARK II	10/16/1962
NRAD	250	TRIGA MARK II	10/12/1977
MUTR UNIV. MARYLAND	250	TRIGA MODIFIED	12/1/1960
TRIGA UNIV. UTAH	100	TRIGA MARK I	10/25/1975
UNIV. ARIZONA TRIGA	100	TRIGA MARK I	12/6/1958

1.4 Summary of operations

The UT TRIGA reactor has operated routinely since 1991 except for time required implementing a digital control system as a planned upgrade, and time to replace a failed reflector. The number of days of reactor operation by year is provided in Fig. 1.4A, and the total energy generation per year in Table 1.4B. The reactor is operated to meet demands of experimental programs and service work, with the only limit on operating time associated with personnel availability.

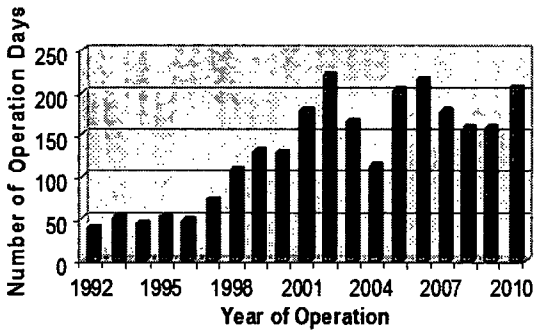

Figure 1.4A, Days of Operation per Year

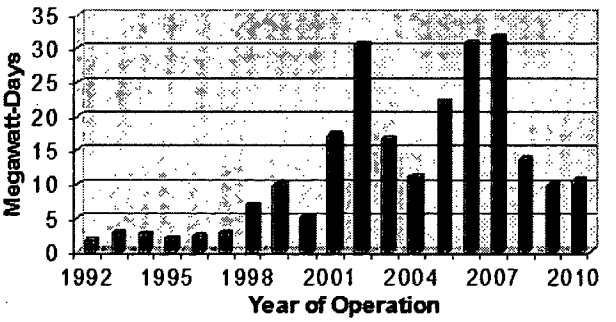

Figure 1.4B, Burnup per Year

1.5 Compliance with NWPA of 1982

Compliance with NWPA of 1982 is assured by the Department of Energy. A copy of the fuels assistance contract is provided in Chapter 15.

1.6 Facility history & modifications

The Department of Mechanical Engineering of the Cockerel College of Engineering at the University of Texas supports a Nuclear and Radiological Engineering program. Development of the nuclear engineering program was an effort of both physics and engineering faculty during the late 1950's and early 1960's. The program subsequently became part of the Mechanical Engineering Department where it currently resides. The program installed and operated the first UT TRIGA nuclear reactor in Taylor Hall on the main campus with initial criticality in August 1963, rated for 10 kilowatts; the license was upgraded for 250 kilowatts operations in 1968. The Taylor Hall reactor operated for 25 years.

In October 1983, planning was initiated for the NETL to replace the original UT TRIGA installation. Construction was initiated December 1986 and completed in May 1989. The NETL facility operating license was issued in January 1992, with initial criticality on March 12, 1992. Dismantling and decommissioning of the first UT TRIGA reactor facility was completed in December 1992.

The original computers supporting the control console have been replaced, and the operating system changed from DOS to a Unix based system. In December 1999 a reflector failure was identified. The reflector was subsequently replaced.

2.0 SITE DESCRIPTION

The site for the TRIGA reactor facility is located in the east tract of the JJ Pickle Research Campus, an area owned and operated by The University of Texas. The Research Center is located in northern Travis County and the City of Austin about 11.6 kilometers north-northwest of The University of Texas at Austin campus. Fig. 2.1 thru 2.4 display the facility locations in relation to surrounding areas. Located near the transition line between hill country and rolling plains, the site is situated about 7.4 kilometers from where the flood controlled Colorado river crosses the transition region and Balcones fault zone. The JJ Pickle Research Campus east and west tracts span part of the inactive fault zone. The east tract is within the transition region to rolling plains. Site location of the TRIGA reactor is in the northeast region of the research center east tract. Adjacent to the north boundary of the research center and near to the eastern boundary, the site location is near the intersection of Braker Lane and Burnet Road. Fig. 2.4 shows the site location within the JJ Pickle Research Campus.

2.1 GENERAL LOCATION AND AREA

Major activities of The University of Texas at Austin, State of Texas government, and City of Austin business district are centered at respective distances of 11.6, 12.6, and 12.9 kilometers to the south-southwest. Distances to air traffic landing facilities in the area are approximately 15 kilometers to the Austin Executive Airport and 16 kilometers to the Breakaway Park Airport. The nearest large commercial airport (Austin-Bergstrom International Airport)is approximately 22 kilometers from the NETL building.

A total area of 1.87 square kilometers is contained within the Research Center area east of Loop 1 (Mopac). The east side of the Center is bounded by a State highway, FM 1325, known as Burnet Road, and the west side is bounded by a Federal highway, US 183. The two tracts are divided by a rail line, formerly the Missouri-Pacific, with 0.93 square kilometers in the east tract and 0.94 square kilometers in the west tract of land. Highway intersections of US 183 with Burnet Road and with Loops 1 and 360 are within two kilometers of the site.

An area of about 9000 square meters in a rectangular shape of 120 meters by 75 meters will comprise the general site location. The 120 meter length is along the north research center boundary. Areas for parking, landscape and access roads are within the general site area. A buffer zone exists between the site area and activities or structures to the east and west. To the west the buffer zone is about 55 meters by 75 meters with parking also about 60 meters by 75 meters. The east buffer region is primarily open space that will provide the access to other development projects north of the general site area.

Figure 2.1, STATE OF TEXAS COUNTIES

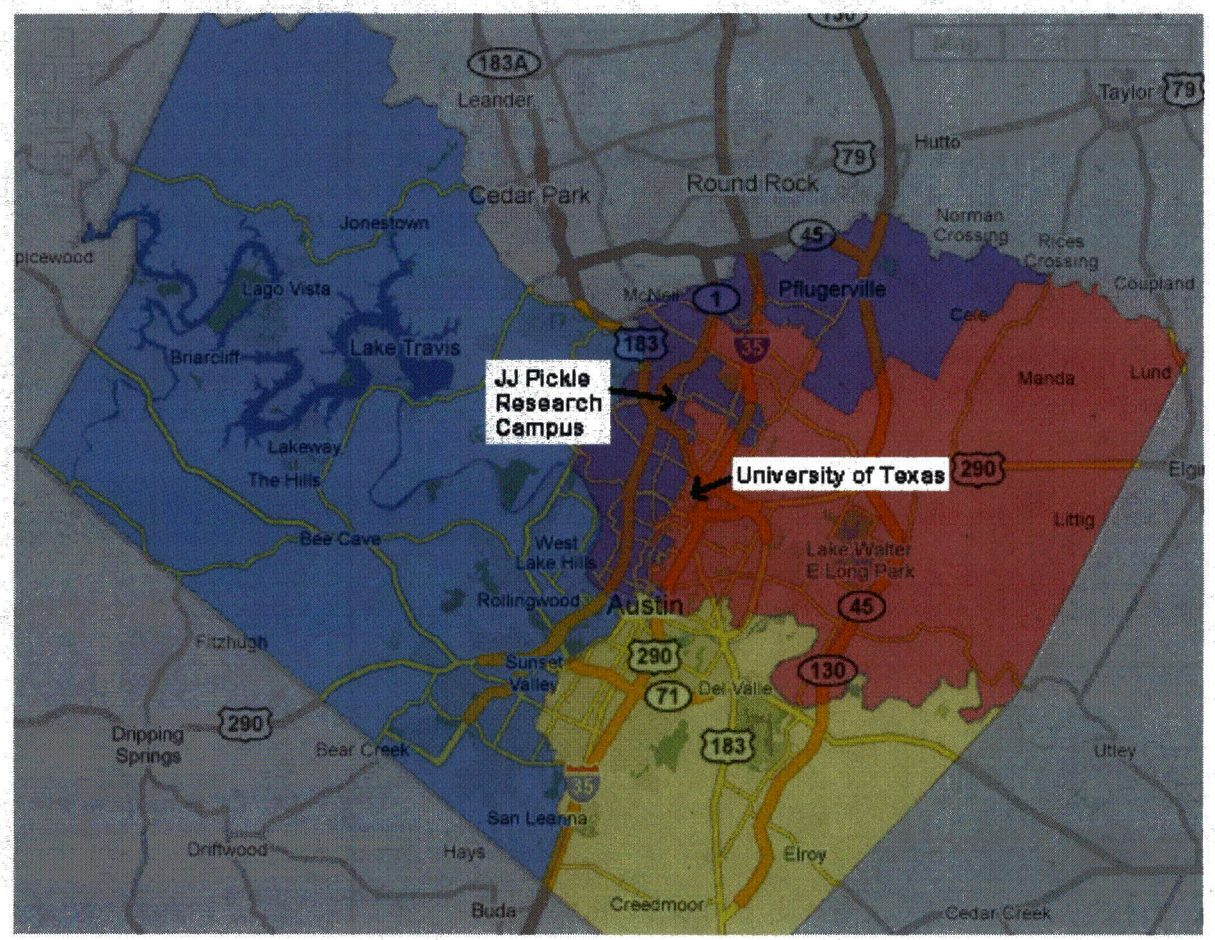

Figure 2.2, TRAVIS COUNTY

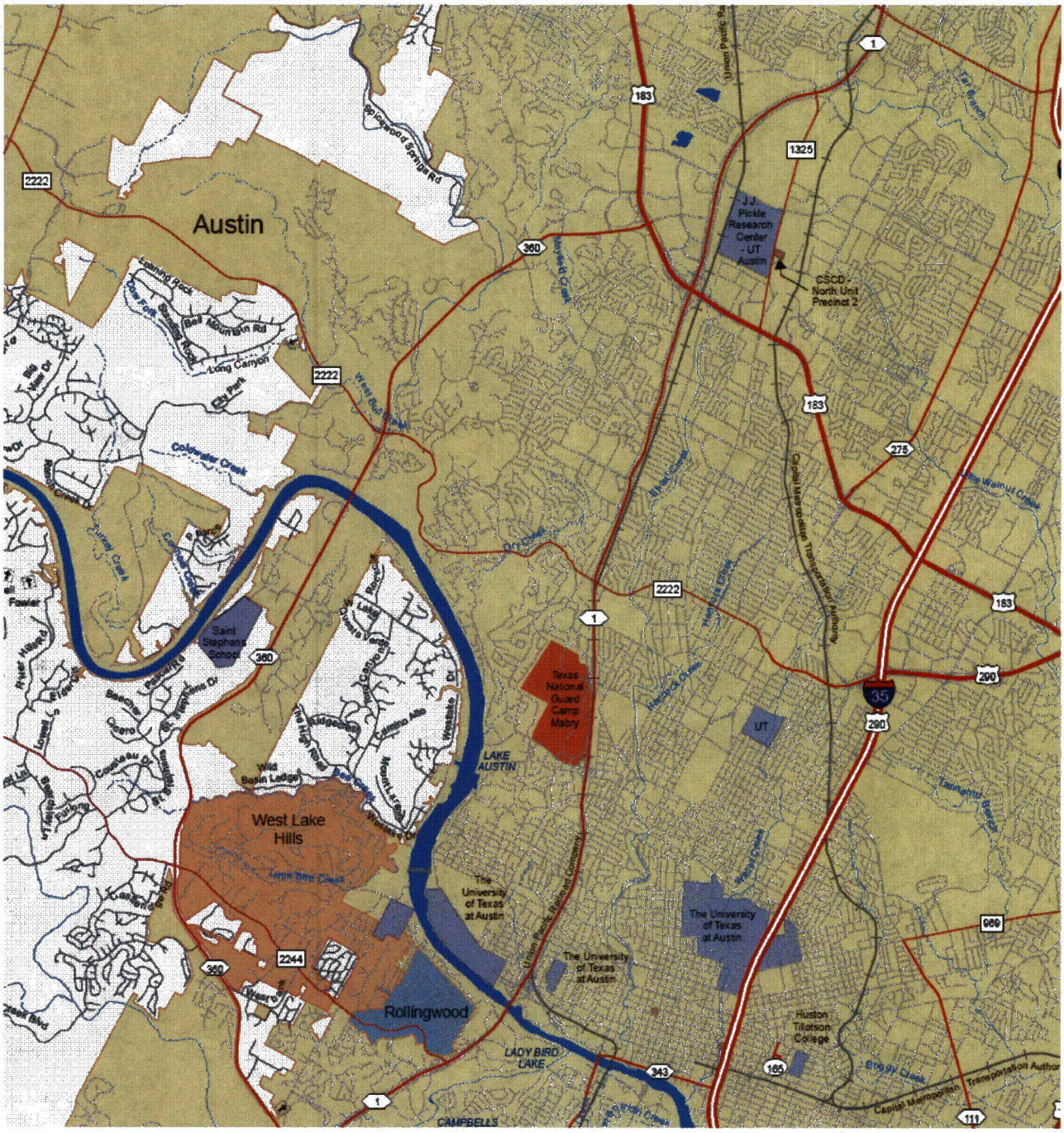

Figure 2.3, CITY OF AUSTIN

Figure 2.4, JJ PICKLE RESEARCH CAMPUS

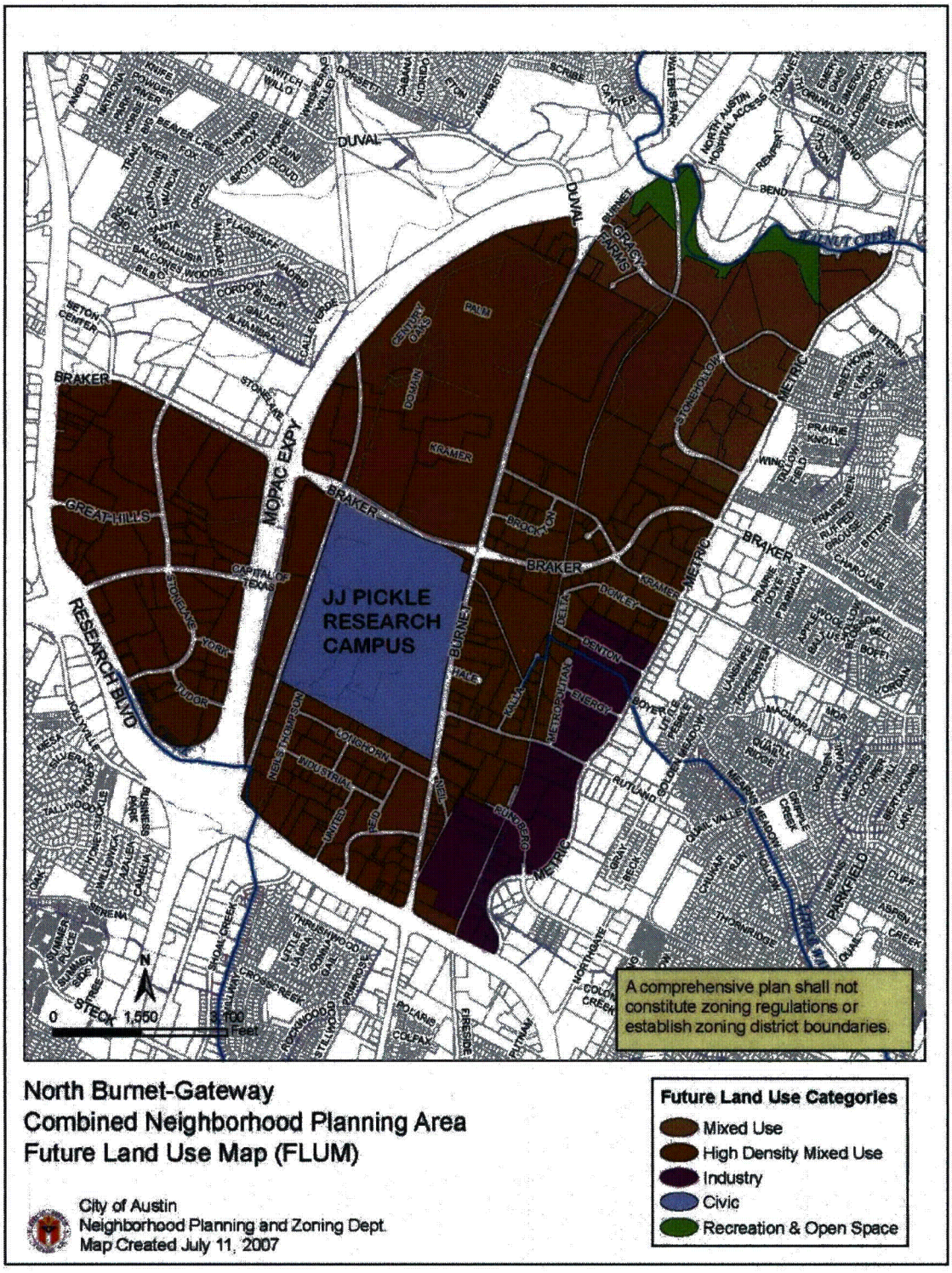

Figure 2.5, LAND USAGE AROUND JJ PICKLE RESEARCH CAMPUS, 2007

Most areas adjacent to the Research Center are developed for mixed commercial and industrial activities including warehouses, manufacturing facilities, and small business parks (see Figure 2-5). Mixed commercial and industrial areas south and east of the Research Center are bounded by highway US 183, highway FM 1325 (Burnet Road), and the Texas New Orleans Railroad to the east. Approximately 2.2 square kilometers of land are enclosed by the area. Much of the remaining area to the west of the Research Center is bounded by highway US 183 and Loop 1 (Mopac) and is residentially and commercially developed, with the Gateway shopping center and multiple apartment complexes. On the southwest side of the intersection of West Braker Road and Loop 1 is the West Pickle Research Building, shown in Fig. 2.4. Immediately north of the JJ Pickle Research Campus east tract is a 2.3 square kilometer commercial complex. Residential areas are located beyond adjoining areas around the JJ Pickle Research Campus with distances from the reactor facility site of 1.2 kilometers to 2.0 kilometers. Few residential structures for either multifamily or single family units are located within a radius of 1.2 kilometers of the reactor site.

2.2 POPULATION AND EMPLOYMENT

Austin is composed primarily of governmental, business, and professional persons with their families. The city has substantial light industry with little heavy industry. Many of the persons in the local labor force are related to activities of the City and its role as a State Capitol, the University and its educational and research programs, or the growing computer-based industries that have established headquarters in the Austin metropolitan area. Travis county has experienced substantial and steady population growth rates over the last several decades. Information on population of the city of Austin and Travis county is contained in Table 2.1.

Since this facility's first criticality in 1992, the Austin population has increased from 466,000 to 790,000 in 2010, a 70% increase. The growth rate slowed down from 2000-2004, and steadily increased from then until 2009. However, according to the 2010 census and predictive data, the growth rate will decrease over the next decade. The 2012 predicted population is 826,235 in Austin and 1,076,119 in Travis County. The annual growth rate in 2010 was 2.11% for Austin and 1.58% for Travis County.

Land usage of the area around JJ Pickle Research Campus is shown in Fig. 2.5. The campus is surrounded by commercial use buildings, including multiple shopping centers. There are a small amount of mixed living areas within several miles of NETL, including apartments and small homes. Population densities for Travis County are listed in Table 2.2 with a map of demarcation lines in Fig. 2.6. Population density in the area containing NETL, zip code 78758, has an average of 5659 people per square mile. This is high compared to other densities in the area because this zip code includes a large tract of residential areas on the far east side. The Research Campus is on the far west side of the zip code, bordering zip code 78759 with 3415 people per square mile.

Table 2.1, AUSTIN AND TRAVIS COUNTY POPULATION TRENDS

Year	City of Austin Total Area Population	Annual Growth Rate	City of Austin Full Purpose Population	City of Austin Limited Purpose Pop	Travis County	Annual Growth Rate	Five County MSA(1)	Annual Growth Rate
1940	87,930				111,053		214,603	
1950	132,459	4.2%			160,980	3.8%	256,645	1.8%
1960	186,545	3.5%			212,136	2.8%	301,261	1.6%
1970	251,808	3.0%			295,516	3.4%	398,938	2.8%
1980	345,890	3.2%			419,573	3.6%	585,051	3.9%
1990	465,622	3.0%			576,407	3.2%	846,227	3.8%
2000	**656,562**	**3.5%**	**639,185**	**17,377**	**812,280**	**3.5%**	**1,249,763**	**4.0%**
2001	669,693	2.0%	654,019	15,674	830,150	2.2%	1,314,344	5.2%
2002	680,899	1.7%	667,705	13,194	844,263	1.7%	1,353,122	3.0%
2003	687,708	1.0%	674,382	13,326	856,927	1.5%	1,382,675	2.2%
2004	692,102	0.64%	678,769	13,333	874,065	2.00%	1,419,137	2.6%
2005	700,407	1.20%	687,061	13,346	893,295	2.20%	1,464,563	3.2%
2006	718,912	2.64%	707,952	10,960	920,544	3.05%	1,527,040	4.3%
2007	735,088	2.25%	724,117	10,971	948,160	3.00%	1,592,590	4.3%
2008	750,525	2.10%	739,543	10,982	978,976	3.25%	1,648,331	3.5%
2009	774,037	3.13%	765,957	8,080	1,008,345	3.00%	1,706,022	3.50%
2010	**790,390**	**2.11%**	**777,953**	**12,437**	**1,024,266**	**1.58%**	**1,716,289**	**0.60%**
2011	**812,025**	**2.74%**	**799,578**	**12,447**	**1,049,873**	**2.50%**	**1,763,487**	**2.75%**
2012	826,235	1.75%	813,776	12,459	1,076,119	2.50%	1,811,983	2.75%
2013	840,695	1.75%	828,223	12,472	1,103,022	2.50%	1,861,812	2.75%
2014	857,508	2.00%	845,024	12,484	1,133,356	2.75%	1,917,667	3.00%
2015	872,515	1.75%	860,018	12,497	1,164,523	2.75%	1,975,197	3.00%
2016	887,784	1.75%	875,274	12,509	1,196,547	2.75%	2,034,453	3.00%
2017	903,320	1.75%	890,798	12,522	1,232,444	3.00%	2,100,572	3.25%
2018	919,128	1.75%	906,594	12,534	1,269,417	3.00%	2,168,841	3.25%
2019	935,213	1.75%	922,666	12,547	1,307,499	3.00%	2,239,328	3.25%
2020	949,241	1.50%	936,682	12,559	1,343,456	2.75%	2,306,508	3.00%
2025	1,022,602	1.50%	1,009,984	12,618	1,538,624	2.75%	2,673,875	3.00%
2030	1,101,633	1.50%	1,089,002	12,631	1,740,812	2.50%	3,062,318	2.75%
2035	1,172,228	1.25%	1,159,584	12,644	1,719,686	2.25%	3,464,732	2.50%
2040	1,232,023	1.00%	1,219,367	12,656	1,921,997	2.00%	3,920,026	2.50%

NOTES: 1) The Five County Austin--Round Rock MSA wholly includes these counties: Bastrop, Caldwell, Hays, Travis and Williamson.

2) Population figures are as of April 1 of each year.

3) Historical and current period population figures for the City of Austin take into account annexations that have occurred.

4) Forecasted population figures for the City of Austin do not assume any future annexation activity.

ZIP Code	Population Dens (pop/square mile)
78701	2611
78702	5141
78703	3999
78704	5737
78705	12482
78717	1215
78719	109
78721	2918
78722	2938
78723	5945
78724	689
78725	133
78726	621
78727	3225
78728	3065
78729	3855
78730	294
78731	3473
78732	486
78733	583
78734	960
78735	604
78736	219
78737	170
78738	299
78739	876
78741	6110
78742	372
78744	167
78745	4627
78746	1468
78747	248
78748	2234
78749	3153
78750	1973
78751	6601
78752	6190
78753	4572
78754	536
78756	4944
78757	5009
78758	5659
78759	3415

Table 2.2, TRAVIS COUNTY 2009 AUSTIN POPULATION DENSITY DISTRIBUTION BY ZIP CODE

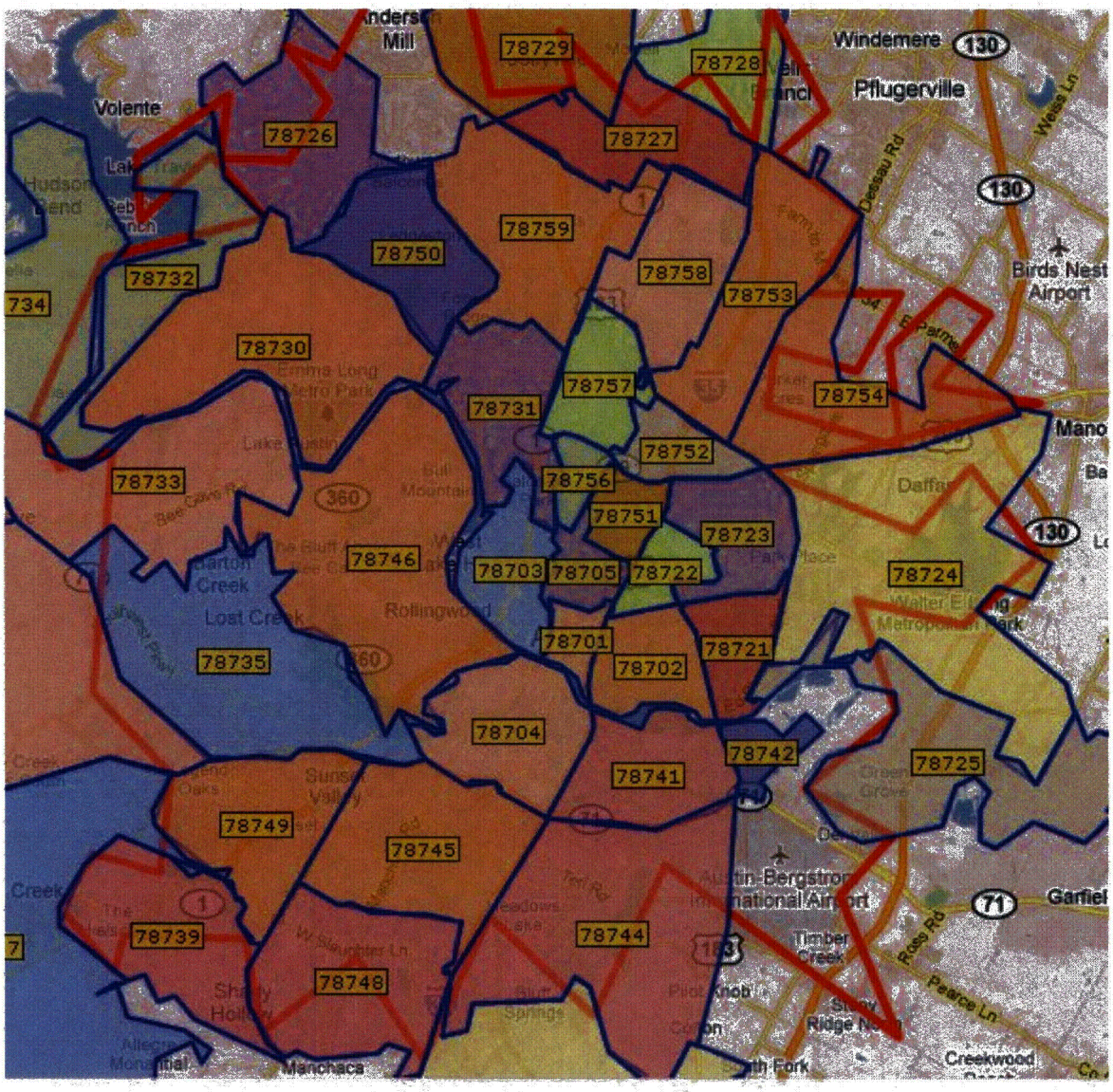

Figure 2.6, 2009 ZIP CODE BOUNDARIES

Research activities at JJ Pickle Research Campus are diverse, and have greatly expanded since the construction of NETL. Research ranges from archeological research on non-vertebrate and vertebrate paleontology to structural engineering to a center for energy and environmental resources. A full list is compiled on the UT Library site[1]. It is difficult to put a number on how many people work at JJ Pickle, since the majority of permanent staff have offices on UT main campus, and most other staff are part time student research assistants. However, at the height of a work day during a semester, there are upwards of 1500 people on Pickle Research Campus.

Immediately adjacent to the NETL building is the geology building (see Fig. 2.4), which houses the Institute for Geophysics, the Bureau of Economic Geology, other research groups, and some administrative offices. Expansion of other activities near the NETL site is possible in the future. Consideration is being given to expanding utility services, and the Texas Advanced Computing Center is undergoing a major expansion near the NETL.

2.3 CLIMATOLOGY

Austin, capital of Texas, is located on the Colorado River where the stream crosses the Balcones Escarpment separating the Texas Hill Country from the Blackland Prairies to the east. Elevations within the City vary from 120 meters to 275 meters above sea level. Native trees include cedar, oak, walnut, mesquite, and pecan.

The climate[2] of Austin is humid subtropical with hot summers. Winters are mild, with below freezing temperatures occurring on an average of less than twenty-five days each year. Rather strong northerly winds, accompanied by sharp drops in temperature, occasionally occur during the winter months in connection with cold fronts, but cold periods are usually of short duration, rarely lasting more than two days. Daytime temperatures in summer are hot, but summer nights are usually pleasant with average daily minima in the low seventies.

Precipitation is fairly evenly distributed throughout the year, with heaviest amounts occurring in late spring. A secondary rainfall peak occurs in September. Precipitation from April through September usually results from thundershowers, with fairly large amounts falling within short periods of time. While thunderstorms and heavy rains have occurred in all months of the year, most of the winter precipitation occurs as light rain. Snow is insignificant as a source of moisture, and usually melts as rapidly as it falls. The City may experience several seasons in succession with no measurable rain fall.

Prevailing winds are southerly throughout the year. Northerly winds accompanying the colder air masses in winter soon shift to southerly as these air masses move out over the Gulf of Mexico.

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[1] "Pickle Research Campus." Univesity of Texas Libraries. Web. 09 June 2011. <http://www.lib.utexas.edu/blsc/>.

[2] NOAA - National Oceanic and Atmospheric Administration, Web, June 2011 <http://www.noaa.gov/>

Climatology data is summarized in Fig. 2.7. Typical Austin wind data are presented in Fig. 2.8[3]. The average length of the warm season (freeze-free period) is 270 days. Climatology and meteorological data is tabulated in Table 2.3 through Table 2.7[4].

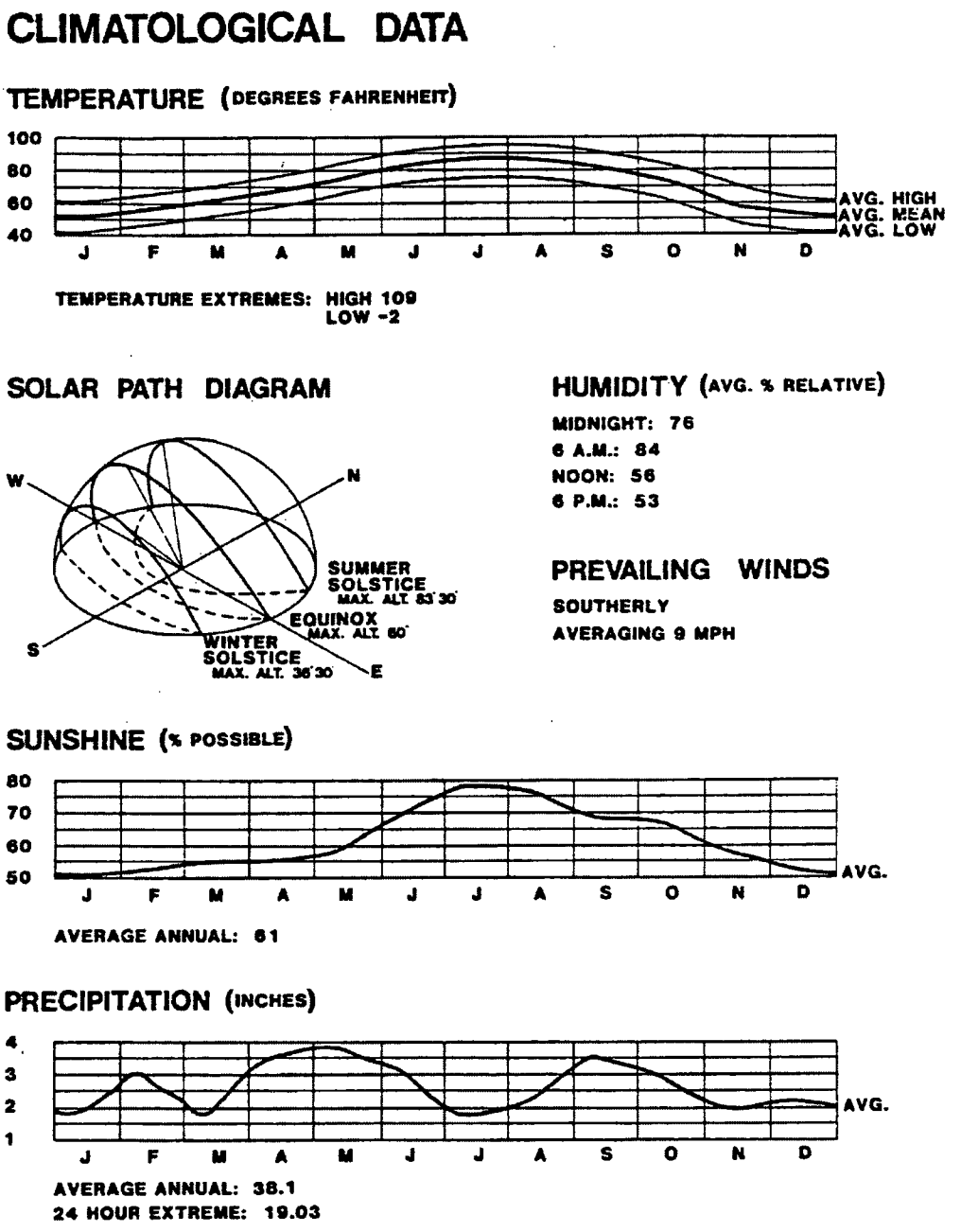

Figure 2.7, AUSTIN CLIMATOLOGY DATA

[3] "Climatography of Texas; Wind Rose-Austin, Texas", National Weather Service, Austin, Texas.

[4] NOAA, op. cit.

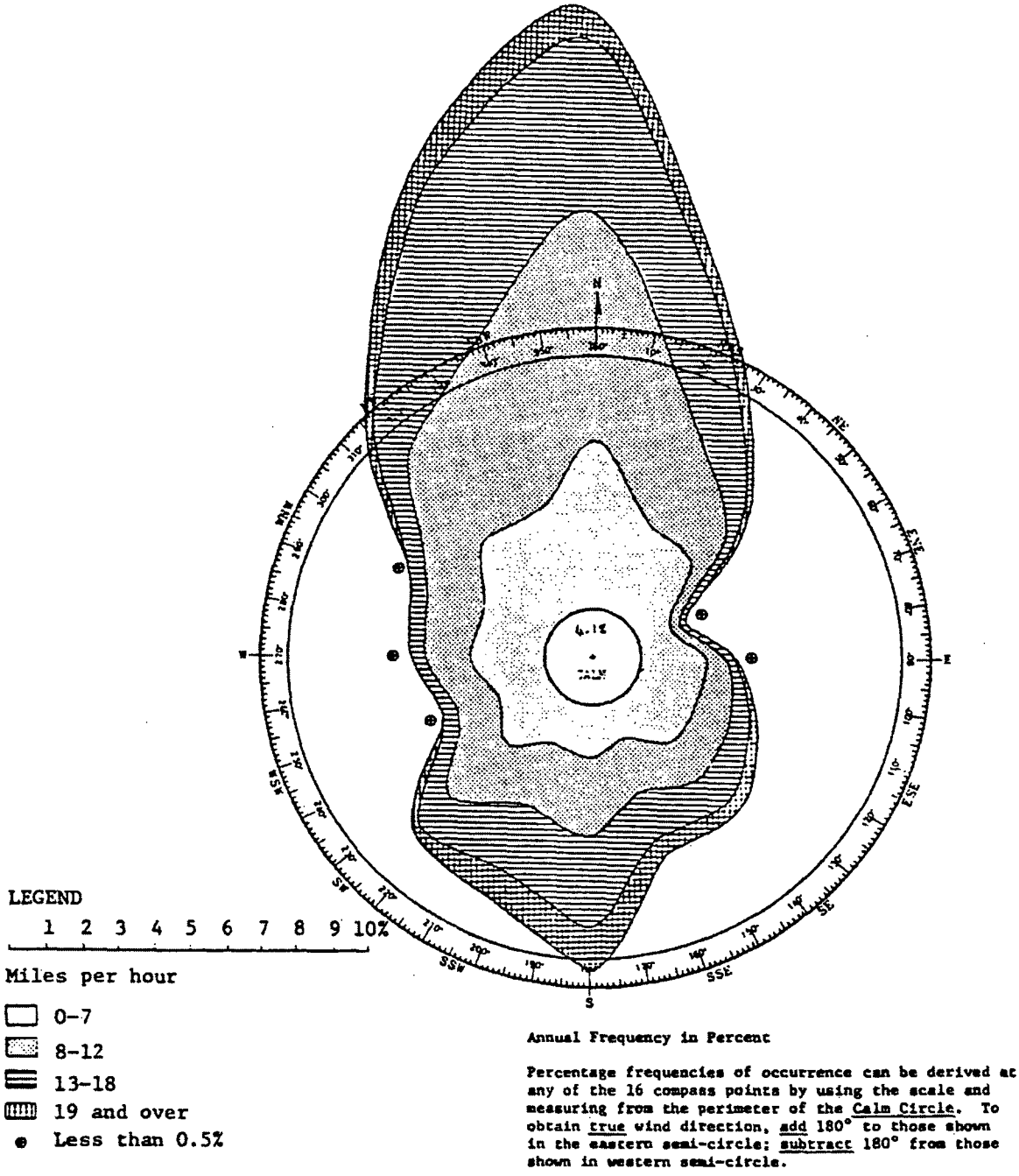

Figure 2.8, AUSTIN WIND ROSE DATA

Destructive winds and damaging hailstorms are infrequent. On rare occasions, dissipating tropical storms effect the City with strong winds and heavy rains. The frequency of tornado type activity and local sightings of tornadoes and funnel clouds is presented in Table 2.8[5]. Recent tropic storm paths are presented in Fig. 2.9 and Fig. 2.10[6].

2.4 GEOLOGY

The northwestern half of Travis county is part of the physiographic province of Texas known as the Edwards Plateau. In Travis County, this is a highly dissected plateau with wooden hills rising in some places more than 150 meters above the drainage pathways. In marked contrast, the southeastern half of the county is gently rolling prairie land which is part of the physiographic province known as the Gulf Coastal Plain. These provinces are separated by the scarp of the Balcones fault zone, which rises 30 to 90 meters above the Coastal Plain. The scarp, however, is not a vertical cliff; it is an indented line of sloping hills leading up from the lower plain to the plateau summit.

The rocks that outcrop in Travis County are primarily of sedimentary origin and of Mesozoic (Cretaceous) and Cenozoic age. They consist largely of limestone, clay, and sand strata which dip southeastward toward the Gulf of Mexico at an angle slightly greater than the slope of the land surface. Therefore, in going from southeast to northwest the outcrops of progressively older formations are encountered, and the rocks lowest in the geologic column have the highest topographic exposure.

At the reactor facility site on the east tract, the geology is of the Austin Group defined as chalk, marly limestone, and limestone with light gray, soft to hard, thin to thick bed, and massive to slightly nodular character. On the west tract, the geology changes to the Edwards Formation of limestone and dolomite with light gray to tan, hard to soft, thin to thick bed, and fine to medium grain character. The separate formations are, respectively, the up and down side of a segment of the Mount Bonnell Fault that passes approximately along the boundary of the east and west Balcones Research Center tracts. Distance to the fault is about 500 meters from the reactor facility site.

The Balcones fault zone, which extends from Williamson County to Uvalde County, extends the full length of Travis County on a line passing through Manchaca, Austin, and McNeil. Here the orderly sequence of formations is replaced by an outcrop pattern controlled by the faults, most of which are normal faults with the down-thrown side toward the coast. Most of the movement of the Balcones Fault zone occurred during the Miocene period. Since no movement has been detected during modern times, this fault is no longer considered active[7]. Location of the Balcones Fault zone and formations in the Austin area are depicted in Fig. 2.11.

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[5] NOAA, op. cit.
[6] NOAA, op. cit.
[7] "Texas Earthquake Information." U.S. Geological Survey Earthquake Hazards Program, Web, June 2011,

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data, and Information Service

Climatography of the United States No. 20 1971-2000

National Climatic Data Center
Federal Building
151 Patton Avenue
Asheville, North Carolina 28801
www.ncdc.noaa.gov

Station: AUSTIN CITY (CAMP MABRY), TX **COOP ID: 410428**

Climate Division: TX 7 **NWS Call Sign: ATT** **Elevation: 621 Feet** **Lat: 30°18N** **Lon: 97°42W**

Table 2.3, 1982 METEOROLOGICAL DATA FOR AUSTIN TEXAS

Temperature (°F)																					
Mean (1)				Extremes										Degree Days (1) Base Temp 65		Mean Number of Days (3)					
Month	Daily Max	Daily Min	Mean	Highest Daily(2)	Year	Day	Highest Month(1) Mean	Year	Lowest Daily(2)	Year	Day	Lowest Month(1) Mean	Year	Heating	Cooling	Max >= 100	Max >= 90	Max >= 50	Max <= 32	Min <= 32	Min <= 0
Jan	60.3	40.0	50.2	90	1971	30	57.3	1990	-2	1949	31	40.7	1979	475	7	.0	@	24.1	.4	6.6	.0
Feb	65.1	44.0	54.6	99	1996	21	62.3	1999	7	1951	2	45.2	1978	319	18	.0	.3	24.4	.3	3.5	.0
Mar	72.5	50.9	61.7	98	1971	28	66.8	1974	18+	1948	12	57.2	1996	163	59	.0	.6	30.2	.0	.8	.0
Apr	78.9	57.6	68.3	98+	2000	23	73.5	1972	31	1940	13	63.0	1997	44	147	.0	1.6	30.0	.0	.0	.0
May	84.8	65.4	75.1	102	1998	7	80.6	1996	43	1954	4	70.5	1976	2	323	.1	7.2	31.0	.0	.0	.0
Jun	90.9	71.1	81.0	108	1998	14	86.4	1998	53	1970	3	77.8	1983	0	495	1.9	20.8	30.0	.0	.0	.0
Jul	95.0	73.4	84.2	109	1954	26	88.0	1998	64+	1970	23	80.1	1976	0	605	4.3	28.0	31.0	.0	.0	.0
Aug	95.6	73.3	84.5	107	2000	31	88.3	1999	61+	1967	13	80.9	1992	0	610	5.6	28.2	31.0	.0	.0	.0
Sep	90.1	68.8	79.5	112	2000	5	84.2	1977	41	1942	27	72.7	1974	2	439	.8	18.2	30.0	.0	.0	.0
Oct	81.4	59.8	70.6	98+	1991	12	73.9	1979	30	1993	31	61.8	1976	32	207	.0	4.4	30.9	.0	@	.0
Nov	70.1	49.3	59.7	91	1951	13	65.6	1973	20	1976	29	52.2	1976	205	51	.0	.0	28.8	.0	.8	.0
Dec	62.3	41.9	52.1	90	1955	25	58.3	1984	4	1989	23	41.8	1983	406	13	.0	.0	26.2	.3	4.9	.0
Ann	78.9	58.0	68.5	112	Sep 2000	5	88.3	Aug 1999	-2	Jan 1949	31	40.7	Jan 1979	1648	2974	11.8	109.3	347.6	1.0	16.6	.0

+ Also occurred on an earlier date(s)

@ Denotes mean number of days greater than 0 but less than .05

Complete documentation available from: www.ncdc.noaa.gov/oa/climate/normals/usnormals.html

Issue Date: February 2004

016-A

(1) From the 1971-2000 Monthly Normals

(2) Derived from station's available digital record: 1930-2001

(3) Derived from 1971-2000 serially complete daily data

U.S. Department of Commerce
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Climate Division: TX 7 **NWS Call Sign: ATT** **Elevation: 621 Feet** **Lat: 30°18N** **Lon: 97°42W**

Table 2.4, HISTORICAL METEOROLOGICAL DATA FOR AUSTIN TEXAS

Precipitation (inches)

	Precipitation Totals									Mean Number of Days (3)				Precipitation Probabilities (1) Probability that the monthly/annual precipitation will be equal to or less than the indicated amount									
	Means/ Medians(1)		Extremes							Daily Precipitation				Monthly/Annual Precipitation vs Probability Levels These values were determined from the incomplete gamma distribution									
Month	Mean	Median	Highest Daily(2)	Year	Day	Highest Monthly(1)	Year	Lowest Monthly(1)	Year	>= 0.01	>= 0.10	>= 0.50	>= 1.00	.05	.10	.20	.30	.40	.50	.60	.70	.80	.90	.95
Jan	1.89	1.39	4.41	1991	9	9.21	1991	.04	1971	7.7	3.8	1.1	.3	.12	.23	.47	.73	1.02	1.35	1.76	2.29	3.01	4.25	5.47
Feb	1.99	2.00	3.05	1958	21	6.56	1992	.03	1999	7.0	3.7	1.5	.3	.21	.36	.64	.92	1.22	1.55	1.94	2.43	3.10	4.20	5.28
Mar	2.14	2.09	2.69	1980	27	6.03	1983	.00	1972	7.9	4.4	1.4	.5	.33	.61	.97	1.27	1.57	1.88	2.22	2.63	3.17	4.01	4.81
Apr	2.51	2.11	3.56	1976	18	8.13	1976	.06	1984	7.2	3.9	1.7	.7	.28	.47	.82	1.17	1.55	1.97	2.46	3.07	3.90	5.28	6.62
May	5.03	5.38	5.55	1979	21	9.49	1995	.73	1998	9.5	6.0	3.1	1.6	1.13	1.60	2.34	3.00	3.66	4.37	5.16	6.11	7.35	9.34	11.21
Jun	3.81	3.05	8.00	1941	7	14.96	1981	.21	1974	7.5	5.2	2.4	1.3	.42	.70	1.23	1.76	2.33	2.97	3.72	4.65	5.92	8.02	10.07
Jul	1.97	1.34	5.20	1936	16	10.54	1979	.00	1993	5.1	3.1	1.2	.5	.04	.15	.40	.67	.99	1.36	1.81	2.38	3.20	4.58	5.96
Aug	2.31	1.30	5.68	1994	9	8.90	1974	.06+	1977	5.2	3.3	1.4	.7	.06	.15	.37	.66	1.01	1.44	1.99	2.72	3.78	5.63	7.52
Sep	2.91	2.45	4.71	1973	26	7.44	1973	.27	1989	7.2	4.4	1.9	.8	.44	.68	1.11	1.51	1.93	2.38	2.91	3.56	4.42	5.83	7.19
Oct	3.97	2.89	6.24	1998	17	12.39	1998	.31	1987	7.4	5.1	2.4	1.2	.39	.67	1.21	1.77	2.36	3.04	3.84	4.84	6.21	8.48	10.71
Nov	2.68	2.64	7.55	2001	15	7.95	2000	.15	1999	8.2	4.3	1.7	.7	.32	.53	.90	1.28	1.68	2.12	2.64	3.28	4.15	5.58	6.98
Dec	2.44	1.78	4.21	1991	20	14.16	1991	.14	1989	7.9	4.0	1.4	.7	.18	.33	.64	.98	1.35	1.78	2.30	2.96	3.87	5.41	6.92
Ann	33.65	33.98	8.00	Jun 1941	7	14.96	Jun 1981	.00+	Jul 1993	87.8	51.2	21.2	9.3	20.94	23.28	26.34	28.71	30.84	32.93	35.11	37.55	40.55	44.95	48.81

\+ Also occurred on an earlier date(s)
\# Denotes amounts of a trace
@ Denotes mean number of days greater than 0 but less than .05
** Statistics not computed because less than six years out of thirty had measurable precipitation

(1) From the 1971-2000 Monthly Normals
(2) Derived from station's available digital record: 1930-2001
(3) Derived from 1971-2000 serially complete daily data

Complete documentation available from:
www.ncdc.noaa.gov/oa/climate/normals/usnormals.html

016-B

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
National Environmental Satellite, Data,
and Information Services

Climatography of the United States No. 20 1971-2000

National Climatic Data Center
Federal Building
151 Patton Avenue
Asheville, North Carolina 28801
www.ncdc.noaa.gov

Station: AUSTIN CITY (CAMP MABRY), TX **COOP ID: 410428**

Climate Division: TX 7 **NWS Call Sign: ATT** **Elevation: 621 Feet** **Lat: 30°18N** **Lon: 97°42W**

Table 2.5, HISTORICAL METEOROLOGICAL DATA FOR AUSTIN TEXAS

Snow (inches)																						
Snow Totals															Mean Number of Days (1)								
Means/Medians (1)					Extremes (2)										Snow Fall >= Thresholds					Snow Depth >= Thresholds			
Month	Snow Fall Mean	Snow Fall Median	Snow Depth Mean	Snow Depth Median	Highest Daily Snow Fall	Year	Day	Highest Monthly Snow Fall	Year	Highest Daily Snow Depth	Year	Day	Highest Monthly Mean Snow Depth	Year	0.1	1.0	3.0	5.0	10.0	1	3	5	10
Jan	.4	.0	#	0	3.9	1985	2	7.5	1985	4	1985	13	#	1985	.3	.1	.1	.0	.0	.2	.1	.0	.0
Feb	.1	.0	#	0	1.2	1985	1	1.2	1985	1+	1985	2	#	1985	.2	.0	.0	.0	.0	.1	.0	.0	.0
Mar	#	.0	0	0	#	1994	9	#+	1994	0	0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Apr	.0	.0	0	0	.0	0	0	.0	0	0	0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
May	.0	.0	#	0	.0	0	0	.0	0	0	0	0	#	1994	.0	.0	.0	0	.0	.0	.0	.0	.0
Jun	.0	.0	0	0	.0	0	0	.0	0	0	0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Jul	.0	.0	0	0	.0	0	0	.0	0	0	0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Aug	.0	.0	#	0	.0	0	0	.0	0	0	0	0	#	1997	.0	.0	.0	.0	.0	.0	.0	.0	.0
Sep	.0	.0	0	0	.0	0	0	.0	0	0	0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Oct	.0	.0	0	0	.0	0	0	.0	0	0	.0	0	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Nov	.1	0	0	0	1.0	1980	25	2.0	1980	#+	1980	26	0	0	.1	.1	.0	.0	.0	.0	.0	.0	.0
Dec	#	.0	0	.0	#	1998	24	#+	1998	#+	1996	16	0	0	.0	.0	.0	.0	.0	.0	.0	.0	.0
Ann	.6	.0	N/A	N/A	3.9	Jan 1985	2	7.5	Jan 1985	4	Jan 1985	13	#+	Aug 1997	.6	.2	.1	.0	.0	.3	.1	.0	.0

+ Also occurred on an earlier date(s) #Denotes trace amounts

@ Denotes mean number of days greater than 0 but less than .05

-9/-9.9 represents missing values
Annual statistics for Mean/Median snow depths are not appropriate

(1) Derived from Snow Climatology and 1971-2000 daily data

(2) Derived from 1971-2000 daily data

Complete documentation available from:
www.ncdc.noaa.gov/oa/climate/normals/usnormals.html

016-C

U.S. Department of Commerce
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Station: AUSTIN CITY (CAMP MABRY), TX **COOP ID: 410428**

Climate Division: TX 7 **NWS Call Sign: ATT** **Elevation: 621 Feet** **Lat: 30°18N** **Lon: 97°42W**

Freeze Data

Spring Freeze Dates (Month/Day)

Temp (F)	Probability of later date in spring (thru Jul 31) than indicated(*)								
	.10	.20	.30	.40	.50	.60	.70	.80	.90
36	3/29	3/21	3/15	3/10	3/06	3/01	2/24	2/18	2/10
32	3/15	3/06	2/28	2/22	2/17	2/12	2/07	1/31	1/23
28	3/06	2/24	2/17	2/10	2/04	1/29	1/22	1/13	12/27
24	2/19	2/09	2/01	1/25	1/17	1/07	0/00	0/00	0/00
20	2/07	1/27	1/18	1/08	12/23	0/00	0/00	0/00	0/00
16	1/05	0/00	0/00	0/00	0/00	0/00	0/00	0/00	0/00

Fall Freeze Dates (Month/Day)

Temp (F)	Probability of earlier date in fall (beginning Aug 1) than indicated(*)								
	.10	.20	.30	.40	.50	.60	.70	.80	.90
36	11/04	11/09	11/13	11/17	11/20	11/23	11/26	11/30	12/06
32	11/15	11/22	11/27	12/02	12/06	12/10	12/15	12/20	12/28
28	11/28	12/06	12/11	12/16	12/21	12/26	12/31	1/07	1/20
24	12/11	12/22	1/01	1/09	1/19	2/02	0/00	0/00	0/00
20	12/19	1/02	1/15	1/29	0/00	0/00	0/00	0/00	0/00
16	1/02	0/00	0/00	0/00	0/00	0/00	0/00	0/00	0/00

Freeze Free Period

Temp (F)	Probability of longer than indicated freeze free period (Days)								
	.10	.20	.30	.40	.50	.60	.70	.80	.90
36	285	276	269	264	259	253	248	241	232
32	323	312	304	297	291	285	278	270	259
28	>365	>365	341	328	319	311	303	294	281
24	>365	>365	>365	>365	>365	347	334	323	311
20	>365	>365	>365	>365	>365	>365	>365	357	335
16	>365	>365	>365	>365	>365	>365	>365	>365	>365

* Probability of observing a temperature as cold, or colder, later in the spring or earlier in the fall than the indicated date.

0/00 Indicates that the probability of occurrence of threshold temperature is less than the indicated probability.
Derived from 1971-2000 serially complete daily data

Complete documentation available from:
www.ncdc.noaa.gov/oa/climate/normals/usnormals.html

016-D

Table 2.6, HISTORICAL METEOROLOGICAL DATA FOR AUSTIN TEXAS

Table 2.7, HISTORICAL METEOROLOGICAL DATA FOR AUSTIN TEXAS

U.S. Department of Commerce
National Oceanic & Atmospheric Administration
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Climatography of the United States No. 20 1971-2000

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Station: AUSTIN CITY (CAMP MABRY), TX **COOP ID: 410428**

Climate Division: TX 7 **NWS Call Sign: ATT** **Elevation: 621 Feet** **Lat: 30°18N** **Lon: 97°42W**

Degree Days to Selected Base Temperatures (°F)

Base	Heating Degree Days (1)												
Below	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Ann
65	475	319	163	44	2	0	0	0	2	32	205	406	1648
60	342	203	63	8	0	0	0	0	0	4	118	277	1015
57	272	152	32	2	0	0	0	0	0	1	78	209	746
55	231	122	19	0	0	0	0	0	0	1	56	171	600
50	146	61	4	0	0	0	0	0	0	0	21	92	324
32	8	0	0	0	0	0	0	0	0	0	0	0	8

Base	Cooling Degree Days (1)												
Above	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Ann
32	566	636	924	1095	1347	1485	1628	1635	1428	1199	833	628	13404
55	65	110	248	409	634	795	915	922	738	490	202	81	5609
57	46	84	201	352	572	735	853	860	678	431	163	60	5035
60	26	53	138	271	480	645	760	767	589	343	113	36	4221
65	7	18	59	147	323	495	605	610	439	207	51	13	2974
70	1	4	20	61	185	345	450	457	298	104	15	2	1942

Growing Degree Units (2)

Base	Growing Degree Units (Monthly)												Growing Degree Units (Accumulated Monthly)											
	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
40	357	451	684	867	1108	1253	1393	1396	1195	959	605	405	357	808	1492	2359	3467	4720	6113	7509	8704	9663	10268	10673
45	233	324	534	717	953	1103	1238	1241	1045	804	459	272	233	557	1091	1808	2761	3864	5102	6343	7388	8192	8651	8923
50	137	214	390	568	798	953	1083	1086	895	650	324	164	137	351	741	1309	2107	3060	4143	5229	6124	6774	7098	7262
55	69	123	257	421	643	803	928	931	745	497	209	82	69	192	449	870	1513	2316	3244	4175	4920	5417	5626	5708
60	29	63	143	279	488	653	773	776	596	353	118	41	29	92	235	514	1002	1655	2428	3204	3800	4153	4271	4312
Base	Growing Degree Units for Corn (Monthly)												Growing Degree Units for Corn (Accumulated Monthly)											
50/86	204	268	424	571	770	872	937	934	819	643	368	235	204	472	896	1467	2237	3109	4046	4980	5799	6442	6810	7045

(1) Derived from the 1971-2000 Monthly Normals
(2) Derived from 1971-2000 serially complete daily data
Note: For corn, temperatures below 50 are set to 50, and temperatures above 86 are set to 86

Complete documentation available from:
www.ncdc.noaa.gov/oa/climate/normals/usnormals.html

016-E

Table 2.8, TRAVIS COUNTY TORNADO FREQUENCIES

Mag: Magnitude
Dth: Deaths
Inj: Injuries
PrD: Property Damage
CrD: Crop Damage

16 TORNADO(s) were reported in **Travis County, Texas** between **01/01/1992** and **02/28/2011**.

Texas

Location or County	Date	Time	Type	Mag	Dth	Inj	PrD	CrD
1 Lago Vista	09/20/1996	07:58 PM	Tornado	F1	0	0	5K	0K
2 Four Pts	05/27/1997	03:11 PM	Tornado	F2	0	0	50K	10K
3 Four Pts	05/27/1997	03:15 PM	Tornado	F1	0	0	5K	0K
4 Lakeway	05/27/1997	03:50 PM	Tornado	F4	1	5	15.0M	0K
5 Lake Travis	08/29/1998	05:45 PM	Tornado	F1	0	0	30K	0
6 Bergstrom Afb	03/16/2000	04:20 PM	Tornado	F0	0	0	0	0
7 Pflugerville	03/16/2000	05:10 PM	Tornado	F0	0	0	0	0
8 Austin	11/15/2001	03:50 PM	Tornado	F1	0	0	100K	0
9 Austin	11/15/2001	04:45 PM	Tornado	F0	0	0	30K	0
10 Bergstrom Afb	11/15/2001	05:30 PM	Tornado	F1	0	0	80K	0
11 Bergstrom Afb	11/15/2001	05:44 PM	Tornado	F0	0	0	15K	0
12 Pflugerville	11/15/2001	06:02 PM	Tornado	F0	0	0	0	0
13 Manor	12/23/2002	07:04 AM	Tornado	F1	0	1	200K	0
14 Austin-bergstrom Arpt	06/08/2004	07:45 PM	Tornado	F0	0	0	150K	0
15 Beecaves	11/16/2004	04:52 PM	Tornado	F0	0	0	0	0
16 Manor	03/25/2005	09:50 PM	Tornado	F1	0	0	100K	0
				TOTALS:	1	6	15.765M	10K

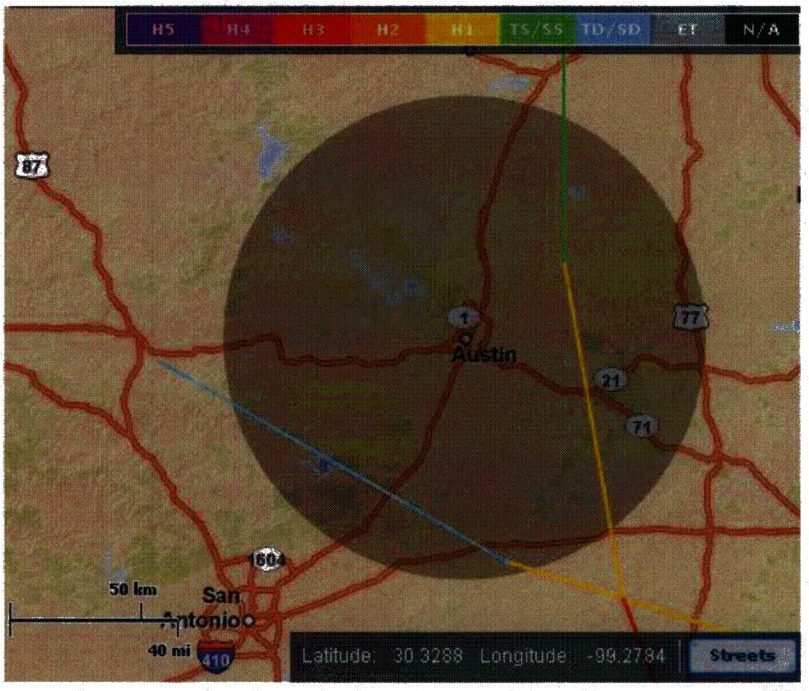


Figure 2.9, TROPICAL STORM PATHS WITHIN 50 NAUTICAL MILES OF AUSTIN, TEXAS
(ALL RECORDED HURRICANES RATED H1 AND UP)

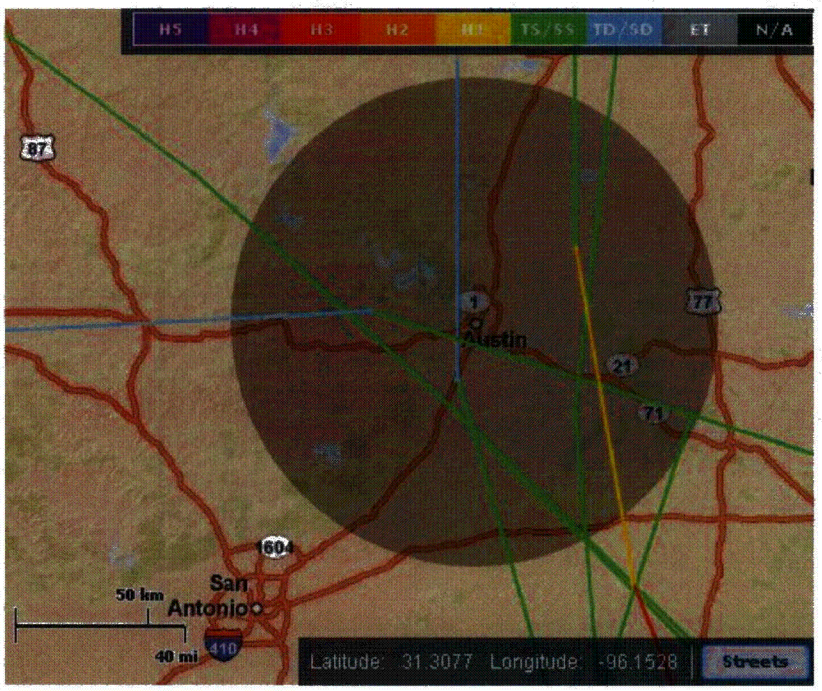


Figure 2.10, TROPICAL STORM PATHS WITHIN 50 NAUTICAL MILES OF AUSTIN, TEXAS
(ALL RECORDED STORMS RATED TROP OR SUBTROP)

2.5 SEISMOLOGY

Thirty three earthquakes of intensity IV or greater have had epicenters in Texas since 1873[8]. The earthquake's intensities were characterized using the Modified Mercalli Scale of 1931. The scale has a range of I thru XII, on which an intensity of I is not felt, an intensity of III is a vibration similar to that due to the passing of lightly loaded trucks, and intensity of VII is noticed by all as shaking trees, waves on ponds, and quivering suspended objects but causes negligible damage to buildings of good design and construction, and an intensity of XII results in practically all works of construction being severely damaged or destroyed. The strongest earthquake, a maximum intensity of VIII, was in western Texas in 1931 and was felt over 1,165,000 square kilometers. Figs. 2.12 and 2.13 show the locations and intensities of earthquakes in Texas. Of these, no damage has ever occurred to local buildings in the Austin area.

2.6 HYDROLOGY

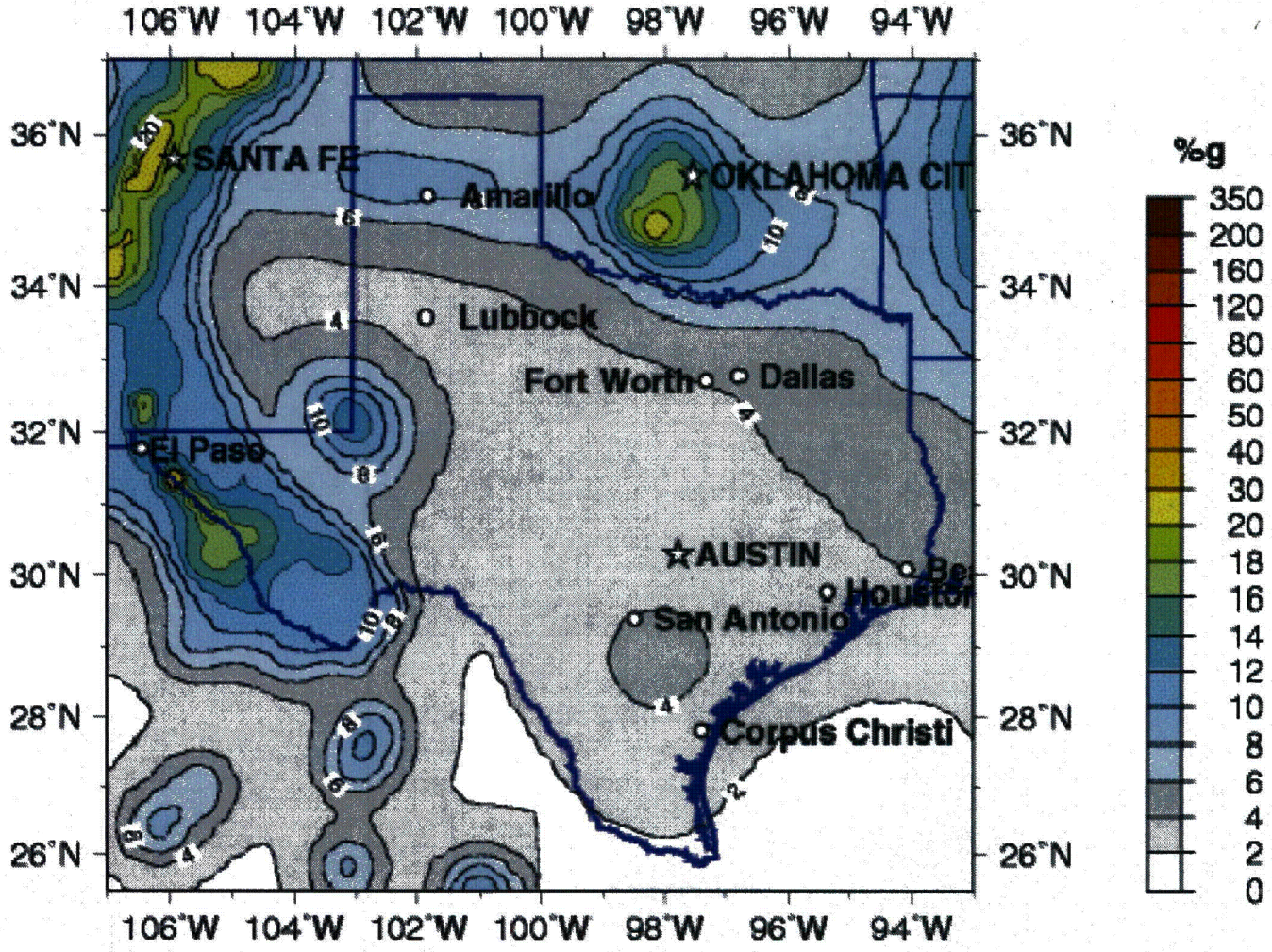
Almost the entire county is drained by the Colorado River and its tributaries. Lake Travis, which is formed by the Mansfield Dam on the Colorado River, is part of the power, flood-control, water conservation, and recreation project of the Lower Colorado River Authority. Other lakes are also operated by the Authority, such as Ladybird Lake and Lake Austin, and are created by Longhorn and Tom Miller dams, respectively. Low level alluvial deposits of the river are commonly saturated with water at relatively shallow depths. Recharge is primarily from the river and local surface contaminations are easily transmitted to this shallow water table.

Ground water from subsurface formation is found in basal Cretaceous sands referred to as the "Trinity" sands. Elevations of the Trinity aquifer range from depths commonly less than 300 meters east of the Balcones Fault Zone to greater than 450 meters to the west of the zone. East of the Mount Bonnell Fault, dolomite and dolomite limestones provide a source of ground water at shallower depths. Access to the Edwards aquifer ranges from 30 meters to 300 meters with natural springs occurring in areas near the Colorado River. Minor aquifers associated with the Glenn Rose Formation supplies small quantities of water west of the Balcones Fault Zone. Water bearing areas in the formation are at varying depths and literally discontinuous. On the Pickle Research Campus east tract, wells drilled for environmental monitoring have produced ground water at depths of less than 15 meters. Fig. 2.14 shows the location of the ground water aquifers.

Water supply for the research campus and wastewater treatment is provided by the City of Austin. Although wells into the aquifers provide substantial water the city supply is filtered river water. Other area municipalities and organizations utilize aquifer water. Control of private wells is the function of county and state Health Departments. Gross beta radioactivity of city water has been measured and is reported in Table 2.9.

[8] "Texas Earthquake Information." U.S. Geological Survey Earthquake Hazards Program, Web, June 2011

Figure 2.11, BALCONES FAULT ZONE

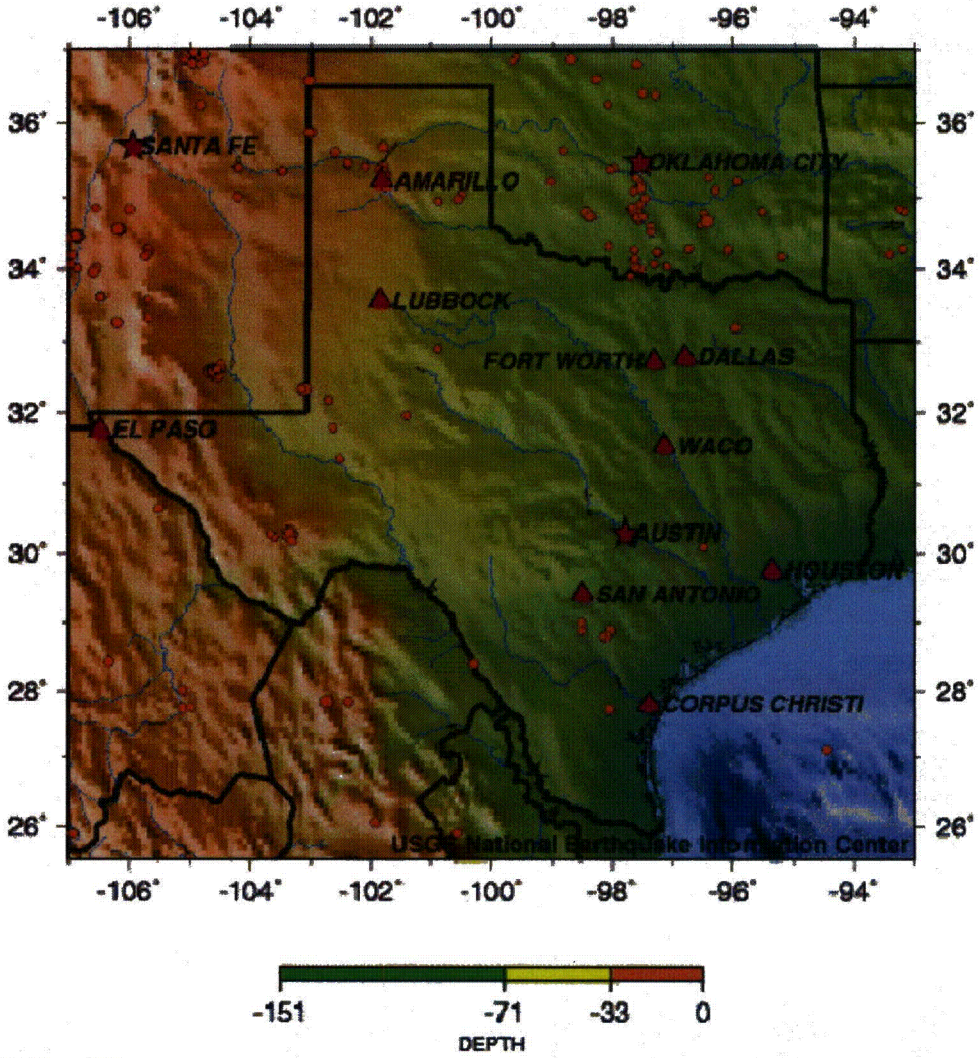


Peak Acceleration (%g) with 2% Probability of Exceedance in 50 Years
site: NEHRP B-C boundary
National Seismic Hazard Mapping Project (2008)

2-8

Figure 2.12, TEXAS EARTHQUAKE DATA

1990 - 2006
Seismicity of Texas



Depth is in kilometers.
Purple Triangles: Cities
Purple Star: Capital City
Circles: Earthquakes (color represents depth range)

Figure 2.13, TEXAS EARTHQUAKE DATA

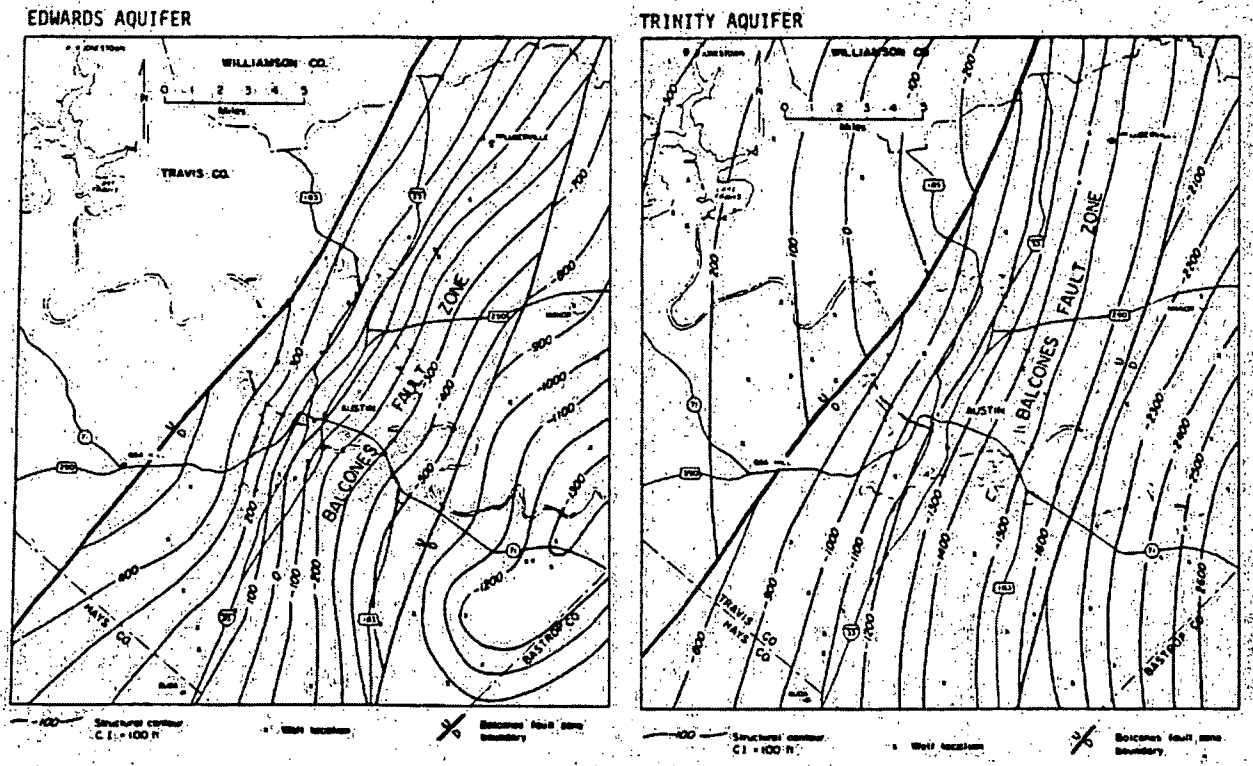


Figure 2.14, LOCAL WATER AQUIFERS

Table 2.9
GROUND WATER ACTIVITY[9]
(gross beta)

Travis County	<4.0 pCi/l
JJ Pickle Research Campus	<4.0 pCi/l

[9] "Water Analysis Report." Texas Department of State Health Services; City of Austin Water and Wastewater, 28 March 2005.

2.7 HISTORICAL

Relocation of the UT TRIGA reactor and related facilities to the JJ Pickle Research Campus site, previously known as the Balcones Research Center, was to help accommodate growth of programs both at the University main campus and at the Research Center site. The facility location at the Research Center is in the north-east corner of the research center site. Reference guidance for site evaluation was ANS 15.7[10].

The original research center site area was operated as a magnesium manufacturing plant by the Federal government in the 1940's. Subsequent arrangements and acquisition by the University would determine activities of the site throughout the 1950's, 1960's and 1970's. Activities at the site were not fully developed prior to the 1980's. University functions or research activities were moved to the site when required accommodations were not available on the main campus. A few functions of the University at the site had resulted in the construction of major facilities suitable for long term use. Other activities at the site have utilized existing structures or other buildings not suited for long term use.

A major program[11] was established in the 1980's to develop the Balcones Research Center site activities. As part of the first phase of development, several major research programs associated with energy and engineering were moved to facilities constructed at the site. Features of the site, before the development activities by the University and after initial development in the 1980's, are illustrated in Fig. 2.15 and 2.17.

Several activities at the Research Center prior to 1980 had been associated with radioactive materials. These activities ranged from the burial of low level radioactive waste materials such as tritium and carbon-14 in the northwest corner of the site, to water transport studies performed in 30 meter diameter surface tanks. Isotopes of cesium-137, cesium-134, and cobalt-60 were present in sludge samples of one of the tanks, but the surface tanks contaminated with radioactive materials used for water transport studies prior to the 1980s were decontaminated and released for unrestricted use in January 1996. Subsequently, the tanks were demolished. The low-level radioactive waste burial site at Pickle Research Campus was released for unrestricted use by the Texas Natural Resource Conservation Commission (now known as the Texas Commission on Environmental Quality) on 06 August 2001. Copies of pertinent documents are on file with UT-Austin EHS[12].

Radioactive waste and other materials at the Research Center site are part of the University broad license for radioactive materials which is managed by the University Environmental, Health, and Safety Department and issued by the Texas Department of State Health Services.

---

[10] "Research Reactor Site Evaluation", American National Standard, ANSI ANS 15.7-1979 (N379).

[11] "Balcones Research Center Project Analysis", Volume I, The University of Texas, 1981.

[12] "Environmental Health and Safety (EHS) | The University of Texas at Austin." The University of Texas at Austin, Web, June 2011, <http://www.utexas.edu/safety/ehs/>

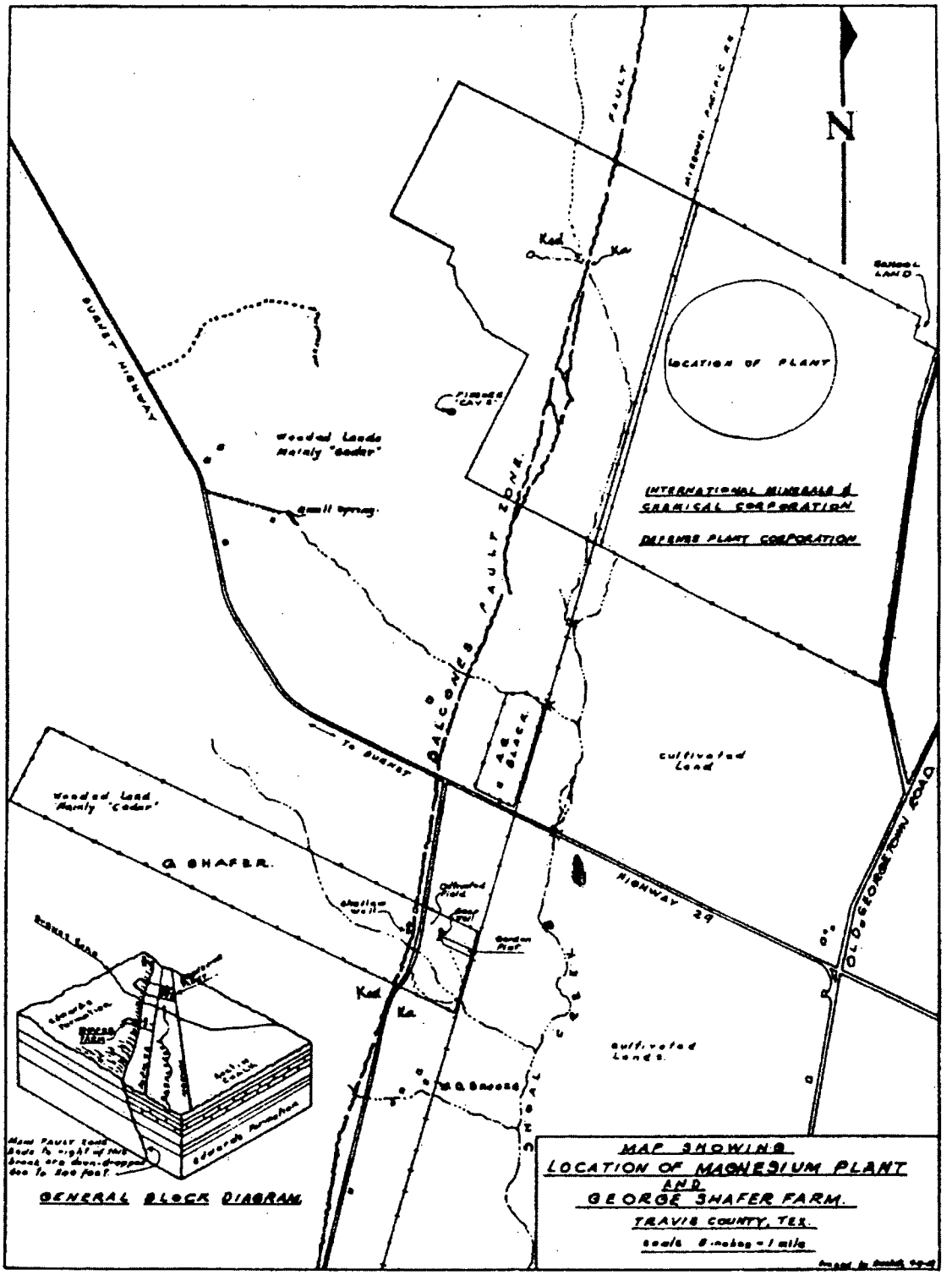


Figure 2.15, RESEARCH CAMPUS AREA 1940

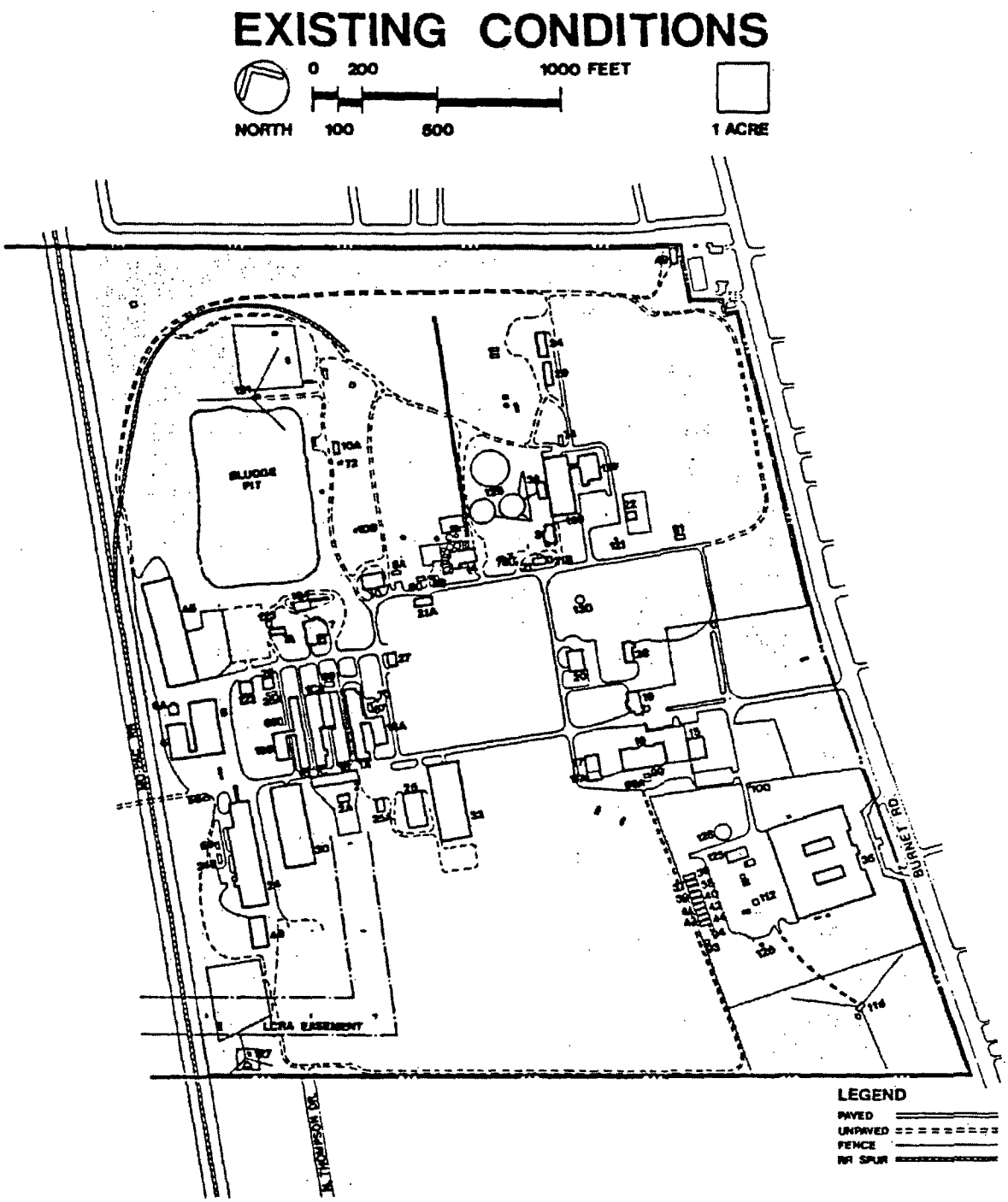


Figure 2.16, PICLKE RESEARCH CAMPUS 1960

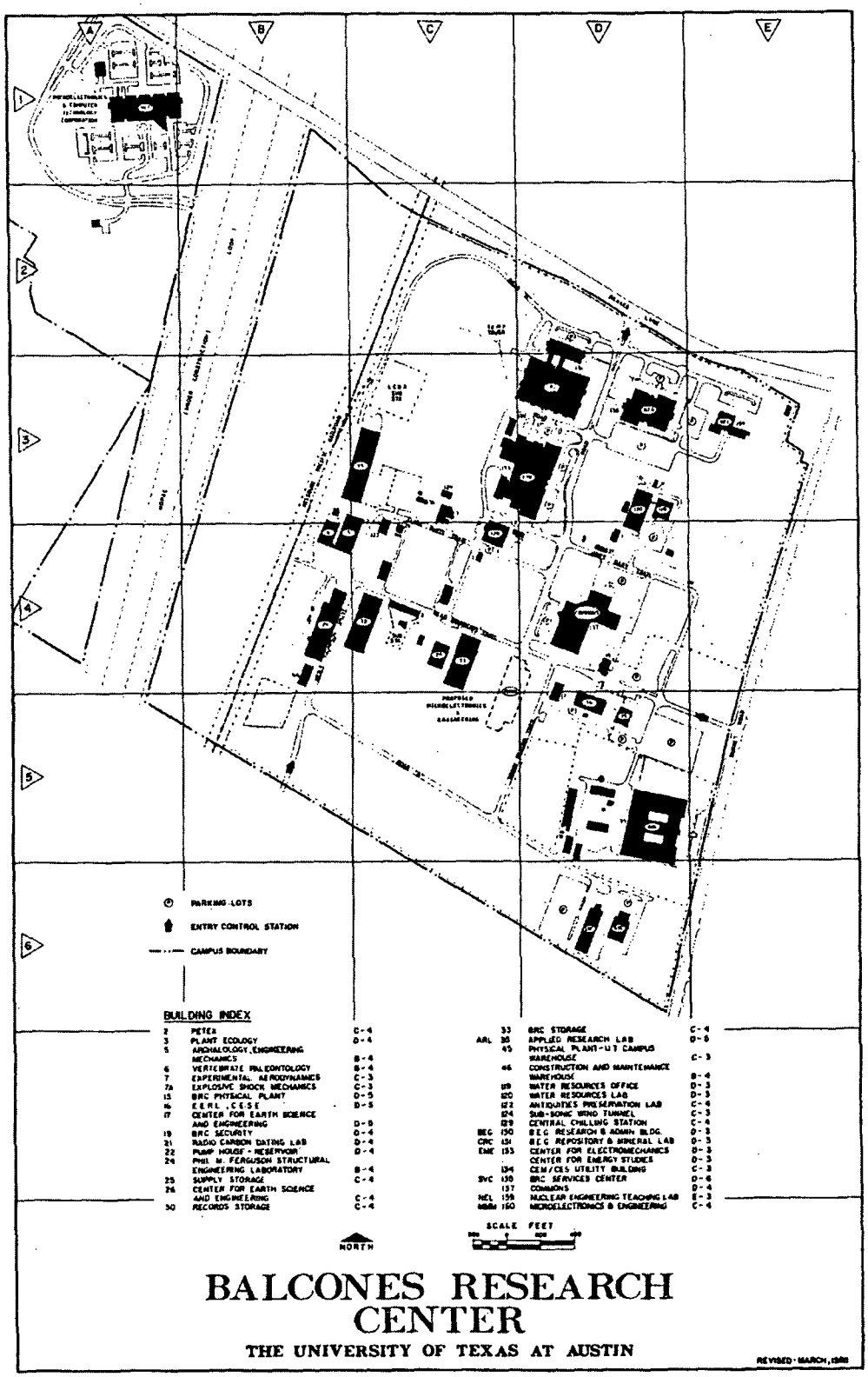
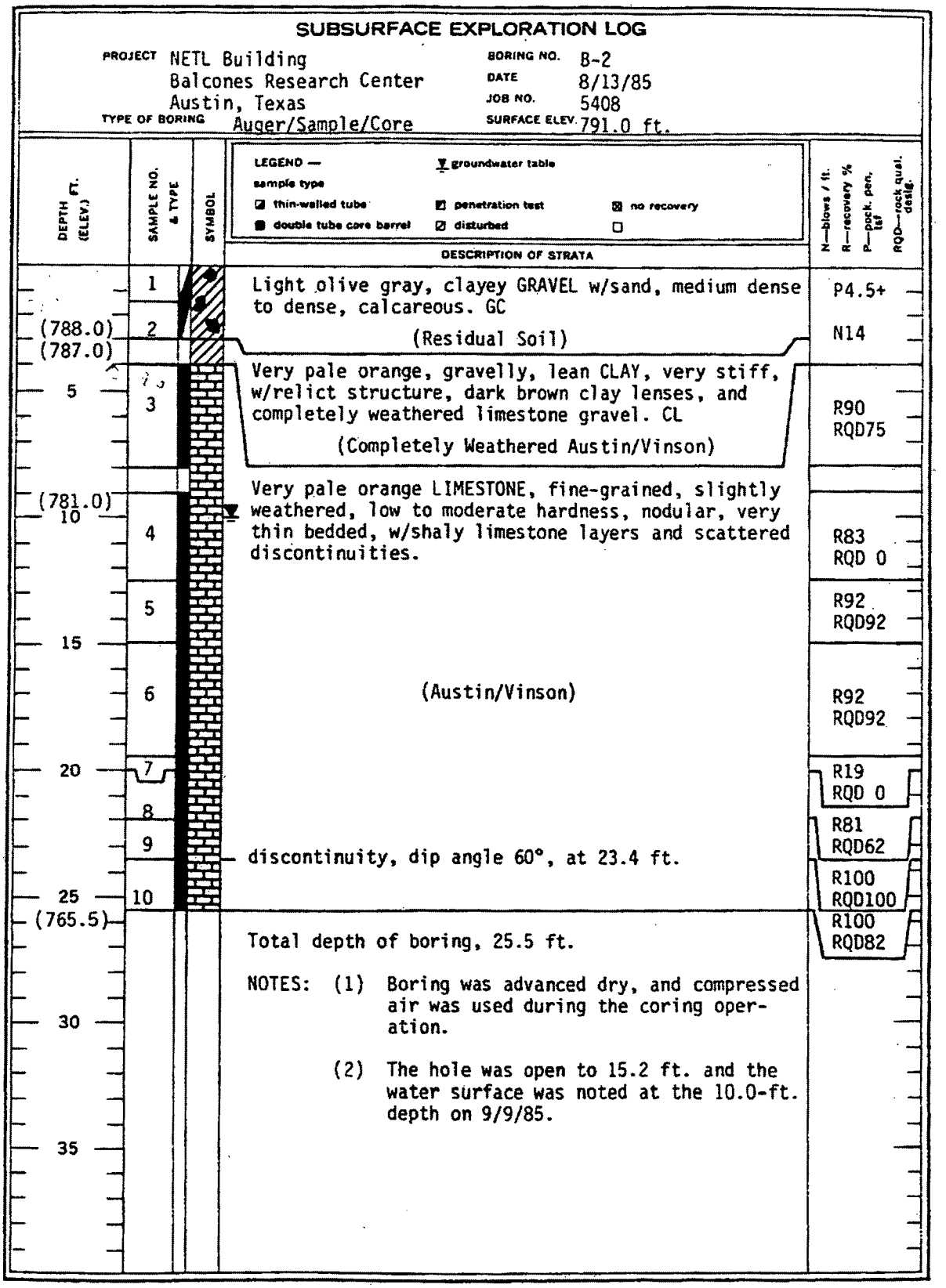


Figure 2.17, BALCONES RESEARCH CENTER 1990


SUBSURFACE EXPLORATION LOG
PROJECT NETL Building
Balcones Research Center
Austin, Texas
TYPE OF BORING Auger/Sample/Core
BORING NO. B-2
DATE 8/13/85
JOB NO. 5408
SURFACE ELEV. 791.0 ft.
LEGEND — sample type
groundwater table
thin-walled tube
penetration test
no recovery
double tube core barrel
disturbed
DEPTH FT. (ELEV.)
SAMPLE NO. & TYPE
SYMBOL
DESCRIPTION OF STRATA
N—blows / ft.
R—recovery %
P—pock. pen. tsf
RQD—rock qual. desig.
1
2
3
4
5
6
7
8
9
10
(788.0)
(787.0)
5
(781.0)
10
15
20
25
(765.5)
30
35
Light olive gray, clayey GRAVEL w/sand, medium dense to dense, calcareous. GC
(Residual Soil)
Very pale orange, gravelly, lean CLAY, very stiff, w/relict structure, dark brown clay lenses, and completely weathered limestone gravel. CL
(Completely Weathered Austin/Vinson)
Very pale orange LIMESTONE, fine-grained, slightly weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.
(Austin/Vinson)
discontinuity, dip angle 60°, at 23.4 ft.
Total depth of boring, 25.5 ft.
NOTES: (1) Boring was advanced dry, and compressed air was used during the coring operation.
(2) The hole was open to 15.2 ft. and the water surface was noted at the 10.0-ft. depth on 9/9/85.
P4.5+
N14
R90
RQD75
R83
RQD 0
R92
RQD92
R92
RQD92
R19
RQD 0
R81
RQD62
R100
RQD100
R100
RQD82
FRANK G. BRYANT & ASSOCIATES, INC.
Austin, Texas

SUBSURFACE EXPLORATION LOG

JOB NO. 5408 DATE 5/17/85 BORING NO. B-1P

40
45
50
55
60
65
70
75
80
85

NOTES:
(1) Boring was advanced dry to the 4.5-ft. depth, and no groundwater was encountered above that depth.
(2) Upon completion of drilling, a piezometer (2-in. I.D. PVC pipe, capped on the bottom, w/the lower 10.0 ft. slotted) was installed w/the bottom at 30.9 ft.
(3) On 6/4/85, the water surface was noted at the 11.7-ft. depth, and the hole was bailed to 30.7 ft.
(4) On 6/5/85, the water surface was noted at the 29.3-ft. depth, and the hole was again bailed to 30.7 ft.
(5) On 6/7/85, the water surface was noted at the 29.1-ft. depth.
(6) On 9/9/85, the water surface was noted at the 9.3-ft. depth.

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SUBSURFACE EXPLORATION LOG

PROJECT: NETL Building
Balcones Research Center
Austin, Texas

BORING NO. B-3
DATE 8/15/85
JOB NO. 5408
SURFACE ELEV. 789.5 ft.

TYPE OF BORING: Auger/Sample/Core

LEGEND — sample type: thin-walled tube; double tube core barrel; groundwater table; penetration test; disturbed; no recovery

DEPTH FT. (ELEV.)	SAMPLE NO. & TYPE	SYMBOL	DESCRIPTION OF STRATA	N—blows / ft. R—recovery % P—pock. pen. tsf RQD—rock qual. desig.
	1		Dusky brown, fat CLAY, stiff, noncalcareous, w/ few fine gravels. CH	P4.5+
	2			P4.5+
(785.6)	3		(Residual Soil)	P4.5+
5	4		Very pale orange, gravelly, lean CLAY, very stiff, w/relict structure, dark brown clay lenses, and completely weathered limestone gravel. CL	
(783.0)			(Completely Weathered Austin/Vinson)	
(780.5)			Very pale orange LIMESTONE, fine-grained, slightly	
10	5		weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.	R100 RQD81
15	6		(Austin/Vinson)	R100 RQD52
20	7			R100 RQD50
(766.5)			Total depth of boring, 23.0 ft.	
25			NOTES: (1) Boring was advanced dry, and compressed air was used during the coring operation.	
30			(2) The hole was open to 18.0 ft. and the water surface was noted at the 9.0-ft. depth on 9/9/85.	
35				

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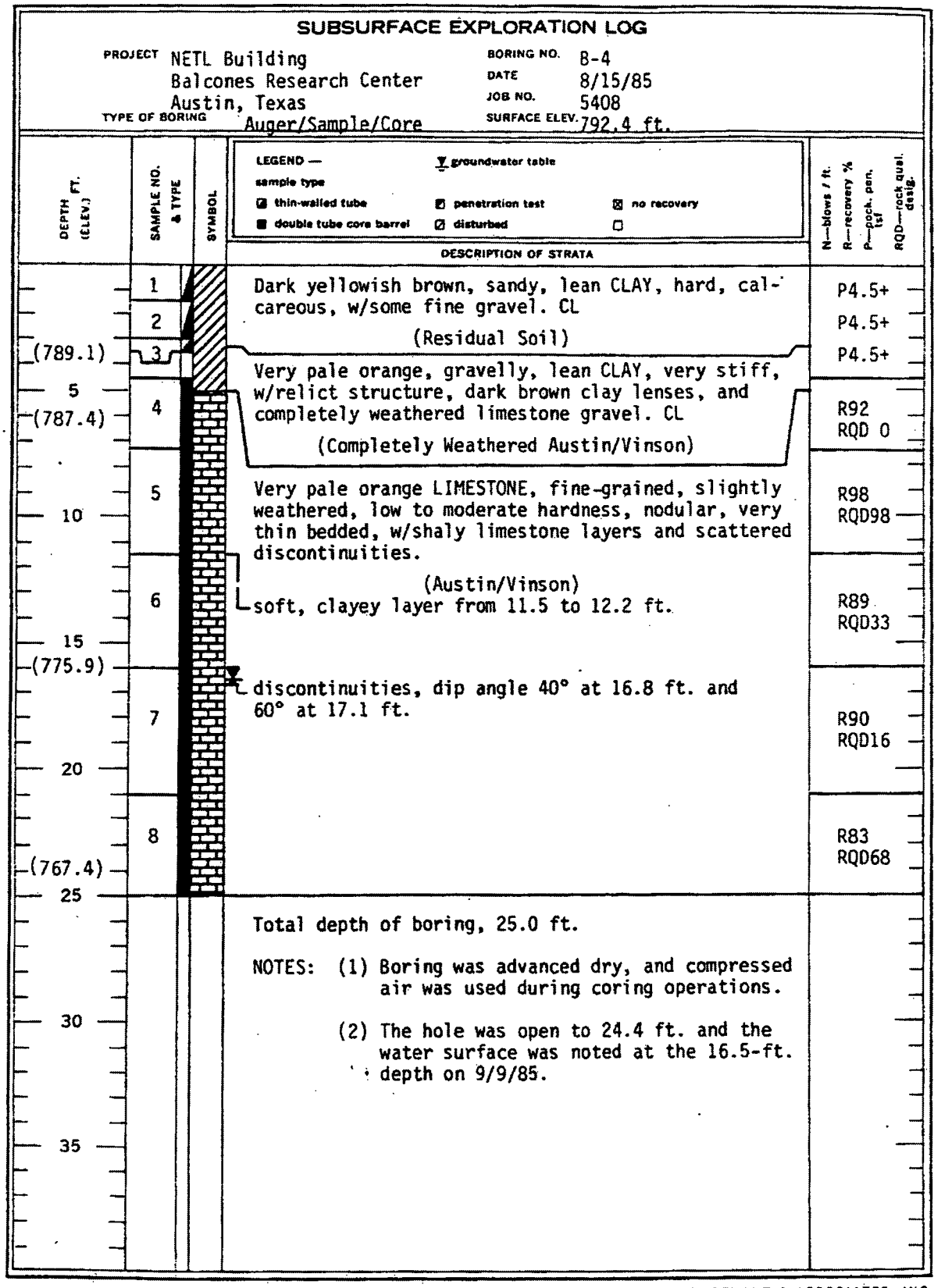

SUBSURFACE EXPLORATION LOG

PROJECT: NETL Building
Balcones Research Center
Austin, Texas
TYPE OF BORING: Auger/Sample/Core

BORING NO. B-4
DATE 8/15/85
JOB NO. 5408
SURFACE ELEV. 792.4 ft.

LEGEND — ▼ groundwater table
sample type
▨ thin-walled tube ▩ penetration test ⊠ no recovery
■ double tube core barrel ▨ disturbed □

DEPTH FT. (ELEV.)	SAMPLE NO. & TYPE	SYMBOL	DESCRIPTION OF STRATA	N—blows / ft. R—recovery % P—pock. pen. tsf RQD—rock qual. desig.
	1		Dark yellowish brown, sandy, lean CLAY, hard, calcareous, w/some fine gravel. CL	P4.5+
	2			P4.5+
(789.1)	3		(Residual Soil)	P4.5+
5			Very pale orange, gravelly, lean CLAY, very stiff, w/relict structure, dark brown clay lenses, and completely weathered limestone gravel. CL	
(787.4)	4		(Completely Weathered Austin/Vinson)	R92 RQD 0
	5		Very pale orange LIMESTONE, fine-grained, slightly weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.	R98 RQD98
10			(Austin/Vinson)	
	6		soft, clayey layer from 11.5 to 12.2 ft.	R89 RQD33
15				
(775.9)			discontinuities, dip angle 40° at 16.8 ft. and 60° at 17.1 ft.	
	7			R90 RQD16
20				
	8			R83 RQD68
(767.4)				
25			Total depth of boring, 25.0 ft.	
			NOTES: (1) Boring was advanced dry, and compressed air was used during coring operations.	
30			(2) The hole was open to 24.4 ft. and the water surface was noted at the 16.5-ft. depth on 9/9/85.	
35				

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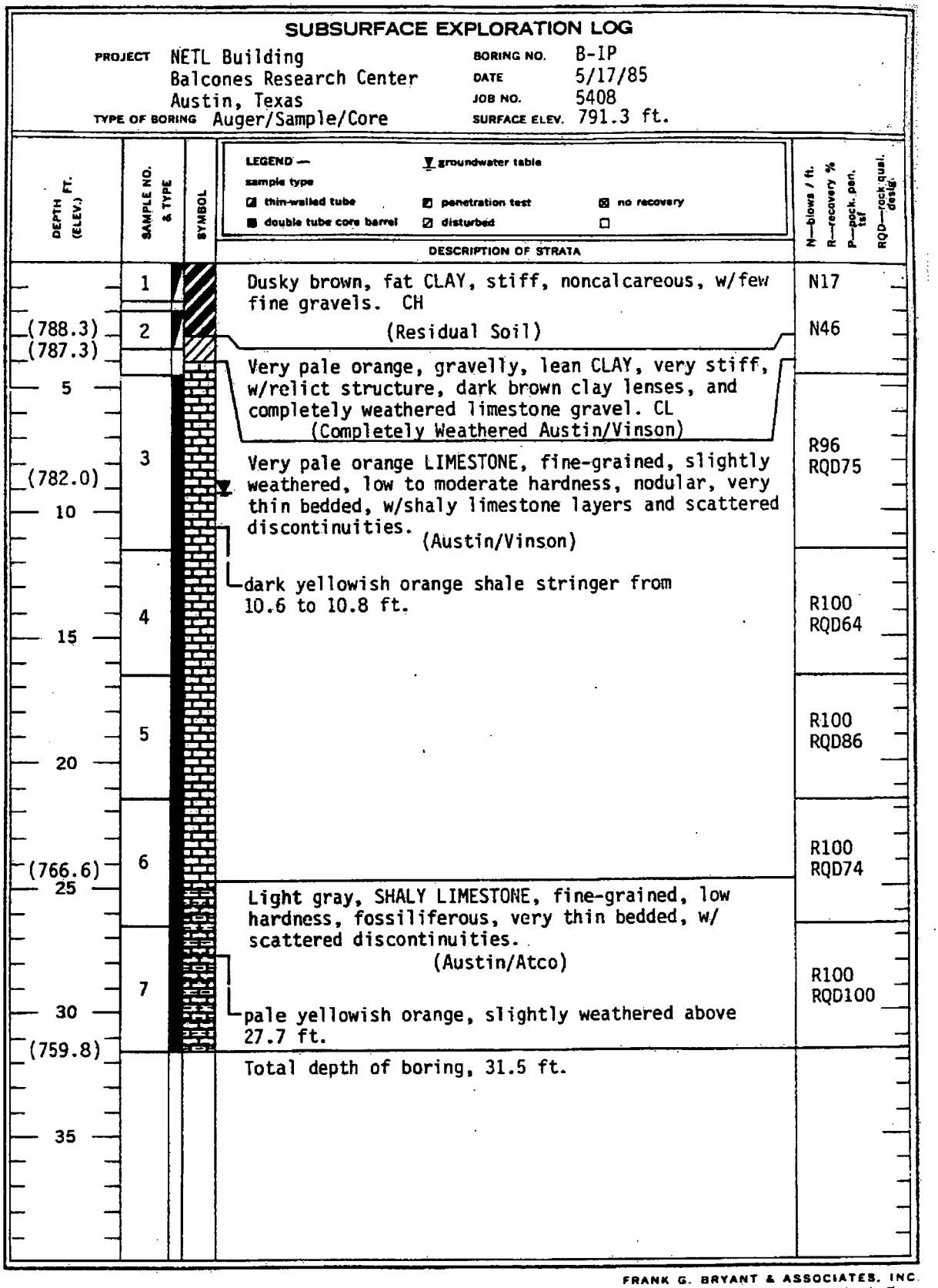

SUBSURFACE EXPLORATION LOG

PROJECT NETL Building
Balcones Research Center
Austin, Texas
TYPE OF BORING Auger/Sample/Core

BORING NO. B-IP
DATE 5/17/85
JOB NO. 5408
SURFACE ELEV. 791.3 ft.

LEGEND — sample type
▽ groundwater table
thin-walled tube
penetration test
no recovery
double tube core barrel
disturbed

DEPTH FT. (ELEV.)	SAMPLE NO. & TYPE	DESCRIPTION OF STRATA	N—blows / ft. R—recovery % P—pock. pen. tsf RQD—rock qual. desig.
	1	Dusky brown, fat CLAY, stiff, noncalcareous, w/few fine gravels. CH	N17
(788.3)	2	(Residual Soil)	N46
(787.3)		Very pale orange, gravelly, lean CLAY, very stiff, w/relict structure, dark brown clay lenses, and completely weathered limestone gravel. CL	
5		(Completely Weathered Austin/Vinson)	
	3	Very pale orange LIMESTONE, fine-grained, slightly weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.	R96 RQD75
(782.0)		(Austin/Vinson)	
10		dark yellowish orange shale stringer from 10.6 to 10.8 ft.	
	4		R100 RQD64
15			
	5		R100 RQD86
20			
	6		R100 RQD74
(766.6)			
25		Light gray, SHALY LIMESTONE, fine-grained, low hardness, fossiliferous, very thin bedded, w/ scattered discontinuities.	
	7	(Austin/Atco)	R100 RQD100
30		pale yellowish orange, slightly weathered above 27.7 ft.	
(759.8)		Total depth of boring, 31.5 ft.	
35			

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Austin, Texas

3.0 DESIGN OF SYSTEMS, STRUCTURES AND COMPONENTS

The Nuclear Engineering Teaching Laboratory (NETL) was built in 1989-1993. The centerpiece of the NETL is a TRIGA Mark II nuclear research reactor. Structures, systems and components (SSC) required for safe operation of the reactor, safe shutdown and continued safe conditions, response to anticipated transients, responses to accidents analyzed in Chapter 13 (Accident Analyses), and control of radioactive material discussed in Chapter 11 (Radiation Protection Program and Waste Management) are identified in Table 3.1. The NETL TRIGA Mk II reactor was originally licensed to operate at power levels up to 1.1 MW, with routine operations up to 950 kW and special operations as required up to 1 MW. Principal functions associated with normal operations include reactor control, heat removal, radiation shielding, gaseous radioactive material control, and shielding. The spectrum of accidents identified for TRIGA and TRIGA fueled reactors in NUCREG/CR-2387 (PNL-4028)[1] includes:

- Excess reactivity addition

  Because of the negative temperature feedback associate with the TRIGA fuel-moderator, core design bounds excess reactivity addition scenarios.
- Metal-water reactions

  Molten metal is required to initiate metal water reactions with zirconium; zirconium melting point (1823°C) exceeds TRIGA fuel temperature limits (1150°C) by a large margin. The maximum temperature that can be achieved in a TRIGA reactor is controlled by design (limiting maximum excess reactivity).
- Lost, misplaced, or inadvertent experiment

  The introduction of a lost, misplaced or inadvertent experiment scenario is controlled by the experiment process (section 10.6), and not by facility design.
- Mechanical rearrangement of core

  Mechanical rearrangement of the core can occur in one of two ways, core crushing or mechanical rearrangement of the core. Core crushing requires the introduction of a large mass over the reactor capable of damaging the reflector and core, and is essentially an operational concern as opposed to a design constraint. Mechanical rearrangement requires an external force (which could be an operationally driven event, or external such as a seismic event), and would result in a decrease in reactivity. Decreasing reactivity does not challenge fuel integrity.
- Loss of coolant accident

  Loss of coolant accident could result from a loss of pool integrity, either a break in the liner or the beam tubes. The design basis for the pool cooling and cleanup system includes specifications to prevent the potential for a piping failure that could siphon a

[1] NUREG/CR-2387 (PNL-4028) Credible Accident Analyses for TRIGA and TRIGA Fueled Reactors (S. C. Hawley, R. L. Kathren, March 1982)

significant amount of water out of the pool. The design basis for the fuel-moderator elements assure that decay heat will not challenge cladding integrity.

- Changes in morphology and ZrH_x composition

  Changes in fuel morphology are driven by temperature changes; design bases to limit fuel morphology issues bound potential accident scenarios.
- Fuel handling

NUREG/CR-2387 identifies nominal core loading of 50 fuel elements; however the UT TRIGA initial criticality required 87 fuel elements. A TRIGA element does not have positive reactivity worth after approximately 6 grams of ^{235}U are burned; as a conservative measure, a maximum burnup of 10 grams is assumed in calculations.

External event modes with potential challenges to each SSC are identified in the Table 3.1. Design criteria for each SSC are provided in section 3.1. Design criteria for and potential impact on required components which are vulnerable to meteorological conditions is provided in section 3.2. Designs to protect against water damage and the impact of potential flooding on structures, system and components which are vulnerable to water intrusion effects are provided in section 3.3. Design criteria for and potential impact on required components which are vulnerable to seismic events is provided in section 3.4.

Table 3.1, SSC Vulnerability

Structure, System, Component	Potential Vulnerability		
	Meteorological	Water	Seismic
Fuel moderator elements			
Control elements			
Core structure			X
Pool, pool cooling, pool cleanup		X	X
Biological shielding			
Reactor Bay/Building	X		X
Ventilation Reactor bay vent, purge HVAC,	X	X	X
Instruments &Controls		X	X
Facility sumps and drains	X	X	X

3.1 Design Criteria for Structures, Systems and Components for Safe Reactor Operation

3.1.1 Fuel Moderator Elements

The TRIGA Mark II nuclear reactor was developed by the General Atomic Division of General Dynamics Corporation for use by universities and research institution as a general-purpose research and training facility. The TRIGA reactor design was based on four [interrelated] principles: safety, simplicity, utility, and cost. General Atomics developed a fuel matrix consisting of zirconium hydride with uranium with a strong negative reactivity response to temperature used in fuel-moderator elements. Since temperature is a function of thermal power and thermodynamic properties (including heat removal time constants), the

temperature response is a feature that inherently limits the maximum achievable power levels under transient and steady state conditions. A complete description of the UT TRIGA Fuel is provided in Chapter 5. The fuel-moderator matrix used at the UT TRIGA is enclosed in stainless steel cladding designed to prevent migration of fission products. The prototype TRIGA reactor attained criticality at General Atomics' John Hopkins Laboratory for Pure and Applied Sciences in San Diego, California on May 3, 1958. The temperature response proved strong enough that pulsing capabilities were developed, using step insertions of large amounts of reactivity through pneumatic removal of a control rod. The performance of uranium-zirconium hydride fuel is substantially independent of uranium content up to 45-w% uranium[2], indicating uranium loading (within a large nominal range of values) is not a design criterion.

Cladding is the principal barrier to fission product release; therefore the design criteria for chemical, mechanical, and thermal conditions require fuel integrity under normal operating and potential accident scenarios. Chemical degradation is limited by establishing a design basis for pool water quality that minimizes corrosion. Mechanical degradation from internal sources is limited by establishing a basis for acceptable morphology and the maximum acceptable internal pressure; mechanical degradation from external sources is limited operationally. The principle cladding failure mechanism is internal pressure generated by temperature; limiting temperatures for pressure are much less than temperatures which could degrade the fuel matrix or cladding directly.

The design criteria for TRIGA fuel is based on pressure generated in the fuel-moderator element. If the cladding temperature is below 500°C, internal pressure will not exceed limits on cladding yield strength at fuel matrix temperatures below 1150°C. If the cladding temperature is greater than 500°C, yield strength of stainless steel cladding is reduced and internal pressure will not exceed limits on cladding yield strength at fuel matrix temperatures below 950°C.

3.1.2 Control Rods

Reactivity is regulated by control rods loaded with boron, described in Chapter 5. Reactor core mechanical design permits control rods to operate in a small set of positions. The positions of the control rods in the core are manipulated by a control rod drive system. The control rods and the control rod drives maintain and control reactor power (i.e., rate of fissions) from shutdown to full power operation, including compensation for temperature increases and fission product poison generated during reactor operation.

Design criteria requires control rods have reactivity capable of establishing and maintaining safe shutdown conditions with the most reactive control rod fully withdrawn, and overcoming negative reactivity effects associated with operations. Design criteria for the control rod drive systems include rod speed adequate to overcome temperature and xenon effects, and fail-safe operation.

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[2] NUREG-1282, Safety Evaluation report on High-Uranium Content, Low-Enriched Uranium Zirconium Hydride Fuels for TRIGA Reactor (Docket No. 50-163)

3.1.3 Core and structural Support

The fuel-moderator elements and control rods are positioned by an upper and lower grid plate. The grid plates establish a geometric array designed to support water moderation and heat removal, and the lower gird plate bears the weight of the fuel-moderator elements. Graphite integral to the elements and a separate, external graphite cylinder surrounding the grid plates reduce neutron leakage. A solid plate directly under the core limits control rod movement down from the fully inserted, preventing potential for the control rod falling out of the core. The reflector assembly rests on a rectangular core support platform fabricated from welded structural aluminum beams. The core support platform is welded to the reactor pool floor. Details of the reflector and core assembly are found in Chapter 5.

Design criteria for the reflector and core array assembly includes mechanical support (stability, strength, and position) as well as cooling and neutronic geometry that assures safe operations and adequate response to accident conditions (adequate cooling, maintenance of shutdown reactivity). Reactor cooling is analyzed in Chapter 5 for normal operations, and Chapter 13 for accident scenarios.

3.1.4 Pool and Pool Support Systems

The reactor core operates by design near the bottom of a large pool of water (Chapter 5). Pool water provides passive cooling for heat removal from the core, moderation of fission energy neutrons required to achieve criticality, and shielding from radiation (produced from the fission process and materials neutron-activated in the core region). The amount of heat produced at the rate of fission at operations below a few kW thermal-powers (and following shutdown) is adequately controlled by convection to pool water, with the heat removed from the pool water by evaporation and conduction to the biological shield. Steady state operation at higher power levels requires active measures to control pool water temperature. A pool cooling system (Chapter 4, 5) is installed to remove heat from the pool water. A pool cleanup system assures the pool water chemistry does not degrade fuel elements.

Design criteria for rector pool includes a depth of water to reduce radiation exposure to acceptable levels, (in conjunction with core cooling geometry) heat transfer characteristics adequate to control pool water temperature during normal and accident conditions. The design criterion for the pool cooling system requires the water temperature can be controlled during operations, with potential for losing pool water inventory in a failure mode controlled. The design criterion for the pool cleanup system is that the water quality can be controlled to acceptable levels.

3.1.5 Biological Shielding

The reactor pool is surrounded by a large concrete biological shield (Chapter 5, 11). The shielding design controls radiation hazard from the fission process (and activated materials). Access to high radiation fields is provided to support experimental programs with beam tubes (Chapter 5, 10) that penetrate the biological shielding. Internal shielding plugs control the

hazard when the beam ports are not in use, active measures provided by experiment controls (Chapter 10) compensate for the increased hazard during utilization.

Design criteria for reactor biological shielding is control of area radiation levels to less than 1 mrem/h.

3.1.6 NETL Building/Reactor Bay

Engineering design, specifications, and construction for the building meet the State of Texas Uniform General Conditions and The University of Texas at Austin Supplementing Conditions[3]. Provisions of the Uniform Building Code[4] and other national codes for mechanical, electrical, and plumbing are applicable to this project. Equipment requirements will apply Underwriter's Laboratories standards or labels, when appropriate, to a piece, type, class, or group of equipment. Other specifications will conform to the standards of the American Society for Testing and Materials (ASTM). Provisions of the Life Safety Code are applicable. One code of importance, the National Fire Protection Code, will determine requirements that relate to fire safety for significant facility operation hazards.

The building site is located on [redacted]. Soil tests of the subsurface set the load capacity [redacted]. Seismic design specifications are Uniform Building Code for zone O. Normal building loads from gravity and wind forces exceed the seismic accelerations for buildings in zone 0; therefore these specifications require no special provisions beyond those of standard building load requirements.

Wind load designs meet requirements of the Uniform Building Code for 70 mph (31.3 m/sec) winds. The specifications include factors for gusts in excess of the wind load criteria. Normal wind and storm conditions are within these design factors.

Building and site draining system design specification were commercial grade, ASTM standards. The sub draining system (French and storm drains) construction includes a granular drainage layer crushed stone meeting ASTM C-33, Grade 67 covering excavated rock surfaces and in the sub-daring trenches for compacted [redacted]. Subdrainage systems were fabricated using American Society of Testing and Materials (ASTM) [redacted]

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[3] [A] Specifications for Nuclear Engineering teaching laboratory, Project No. 102-568, the University of Texas at Austin (09/15/1986)
[B] Construction Administration Manual for Nuclear Engineering teaching laboratory, , Project No. 102-568, the University of Texas at Austin (12/1986)
[C] NETL Project Nos. 1, 2, & #, Project No. 102-568, Amendments; the University of Texas at Austin (12/1986)

[4] Uniform Building Code, International Conference of Building Officials (05/01/1985)

and Vent Pipe and Fittings, [redacted]
and appropriate standards for joining [redacted].

A. Building

Architectural design of the building will develop two separate functional sections, the reactor bay wing and an academic and laboratory wing. Structural design of the building sections is of concrete columns and beams with steel reinforcement. Two floor levels will comprise the academic and laboratory wing. The first level of the reactor bay wing is [redacted] below the mean grade, while the academic wing entry level is 7 feet (2.1 meters) above the mean grade.

The entry floor level (second level) is an administrative and office section. Laboratories will be on the next level (third level). Construction of this wing is reinforced concrete pier and columns with poured beam and slab floors and roof. Exterior walls will consist of concrete tilt panel, metal siding and window units. Interior walls are metal stud frames with gypsum board panels. Doors are solid core wood. Entry way area and door is glass and metal frame. Stairwells at each end of the building wing will provide access to each building level.

The reactor bay wing consists of three basic parts with several types of concrete construction. [redacted] A 4-level section with the HVAC room, control room & offices, shops and facility service/equipment rooms, and staging area are in a section adjacent to the reactor bay. A radiation experiment room with [redacted] is adjacent to the 4-level section. Exterior walls of the reactor bay are concrete and steel construction with tilt panels and attachment columns. The combination of panels and columns set on top of the first level structure forms an integral unit by placement of the panels, then placement of the columns.

Structural concrete and steel columns support slab and beam floors adjacent to the reactor bay. Interior walls are primarily concrete blocks with a few plaster board type walls. The exterior construction of the reactor bay wing is completed by concrete and metal panels. Roof structure is a steel joist system with metal deck, concrete slab, and built-up composition roof that includes fire barrier and thermal insulation.

A room of four walls and a roof of standard density concrete [redacted] thick forms a radiation shield room to complete the reactor bay wing. The room is cast in place with key joints between concrete placements. Tilt panels and composition roof finish the structure. All doors are of hollow metal construction.

B. Reactor Bay

Design of the reactor bay is specified by constraints on the function of the architecture design, access control for physical security, radiation protection for personnel safety, and applicable building code standards.

The reactor pool, shield and primary experiment facilities are located in a reactor bay area that is about 18.3 meters on each side. A total of 4575 cubic meters of volume is enclosed in the reactor bay above the 335 square meters of floor space. Operation control of reactor and of reactor experiment activities is provided by an area located adjacent to the reactor bay. Space in the operation control area is divided into control room, conference room, office, and entry way. Total operation control area (7.3 by 18.3 m) is 134 square meter of floor space and roughly 489 cubic meters of air space. The stairwell in the academic wing provides access to the reactor bay and operation control areas.

[redacted]. The remaining three sides of the reactor bay area are enclosed by exterior walls. Both emergency exits and equipment bay doors on the first level open into the adjacent area within the building from which building exits are accessible.

Two rooms within the reactor bay will enclose reactor support systems. Pool water treatment systems for purification and cooling equipment are on the first level. Auxiliary equipment for experiment systems, such as pneumatic systems, will be in the second level room. Other features of the reactor bay include a five-ton bridge crane and [redacted] fuel storage pits. The storage pits and reactor shield structure are important systems to safely operate and store the reactor fuel materials. However, only the ventilation design for the reactor bay is an engineering safety feature.

3.1.7 Ventilation Systems

Ventilation systems are provide to support general habitability, with two dedicated systems designed to control the buildup of radioactive gas in the reactor bay (the confinement ventilation system and the auxiliary purge system).

Design criteria for the ventilation systems are to control radiation exposure from airborne radionuclides to within acceptable limits during normal operations, and to prevent reactor bay ventilation systems from discharging unacceptable levels of radioactive effluent during accident conditions. A secondary function of the system is to conserve energy required to condition the air when the reactor is not operating.

A control system establishes and manages of differential pressures across spaces to maintain a gradient that manages air flow. The control system is designed to ensure that any potential releases of radioactive materials is directed through a controlled discharge path (Chapter 9).

The reactor bay ventilation system provides fresh air into and an exhaust stream from the reactor bay (Chapter 9, 11). This system has an operational mode that recirculates air if the reactor is not operating to reduce the energy consumed in conditioning the air.

The auxiliary purge system exhausts atmosphere from experimental facilities, where gaseous activation products are expected to occur (Chapter 9, 11).

Effluent pathways for air, liquid, or solid releases of radioactive material provide control of material releases. Control pathways for air and liquid effluents are by way of two rooms, room 4.1M3 and room 1.108. Control of air releases from reactor experiment areas is provided in room 4.1M 3, which contains the air, purge system isolation valve and filter bank. The filter bank normally contains prefilters and one high efficiency particulate filter. The filter bank is configured to accommodate a charcoal filter and additional high efficiency particulate filters, if needed.

There are two principle gases radionuclides produced as a byproduct of reactor operations in quantities of concern. Production of these radionuclides is addressed in Chapter 10.

3.1.8 Instruments and Controls

Reactor instrumentation and controls (including safety system, reactivity control systems, and process, radiation monitoring systems, and process monitoring systems) are designed to be operated and monitored from a central control room.

The design basis for the safety systems is to automatically terminate operations before a safety limit can be exceeded. The design basis for the reactor controls system is to permit reactivity control to (1) maintain safe shutdown under all license conditions, and (2) compensate for transient changes in temperature and xenon over the full range of power operations.

3.1.9 Sumps and Drains

Control of liquid releases that contain radioactive material is provided in room 1.108, which contains storage tanks for collection, processing, storage, or release of liquid effluents. The reactor pool will not release liquid effluents as a part of normal operation.

Design for water runoff in the project vicinity will provide for dispersal of water from local rainfall rates that are frequently sporadic but sometimes torrential. Drainage provisions for the building roof, site landscape, access roadways and subsurface control local runoff. Local flood control includes gravity flow drainage and collection sumps with dual operation pumps. Roof drainage and site runoff are by gravity flow. Separate sumps with pumps control subsurface drainage at the building perimeter and beneath the reactor shield foundation.

3.2 Meteorological Damage

Normal wind and storm conditions are within the design factors established in Uniform Building Code for 70 mph (31.3 m/sec). Hurricanes are not likely to be a direct threat because of the natural dissipation of energy on land. However, tornados are a concern with their extreme wind velocities. Tornado type activity is roughly one event per year per 1000 square miles (2590 sq. kilometers) in the general site area. This activity represents a frequency of one per 2.5×10^5 years for an area of a square with sides of 333 feet (31 meters) representative of the building.

3.3 Water Damage

Gentle slope characteristics in the immediate site vicinity provide an ample gradient of about 3 feet (1 meter) for surface water runoff. A concrete spillway has been constructed to assure drainoff does not concentrate. Mean elevation at the local site is 791 feet (241 meters). Data from the National Flood Insurance Program indicates that no portion of the research campus site is within the 100 or 500 year flood zone. Thus, the only flooding likely will be as a result of local runoff conditions.

The facility has three collection sumps. One sump collects water from the radioactive waste collection system which serves the radioactive labs in the laboratory and office wing, and does not play a role in protection form water intrusion. One sump collects water from French drains installed around the reactor biological shielding/pool foundation. One sump collects water from the truck access ramp and French drains around the building foundations.

Equipment providing services to reactor systems is located in two rooms on the lower level of the reactor building. Makeup water, compressed air, and HVAC chill water are provided from a reactor building lower level room adjacent to the reactor bay. Pool cooling and cleanup are located in a room within the reactor bay structure.

Makeup water is provided by potable water pressure. Service would still be available if the makeup water system were flooded, although water quality could not be monitored. The loss of chill water to fan coil units affect habitability only. The ventilation system damper controls, pool cooling controls, and pulse rod operate using compressed air system. The compressors and air dryer would likely fail if the air compressor room were flooded. The pulse rod would be inoperable with the control rod fully inserted in a safe condition. Pool cooling would be inoperable. Reactor bay air dampers would fail closed. These systems are not required to maintain safe shutdown conditions, but the ventilation is required for reactor operation.

Pool cooling and cleanup pumps could be damaged or rendered inoperable by water intrusion; however, the pool cleanup pump is not required for operation unless chemistry

control is required to maintain pH at acceptable levels, and the pool cooling pump is not required for operations as long as temperatures are acceptable (or operating at less than about 100 kW) or while shutdown. The loss of pool cooling would affect the range of possible operations, but not reactor safety.

In summary, massive water intrusion on the first floor could affect operability of the reactor but would not prevent maintenance of safe shutdown conditions.

3.4 Seismic Damage

The potential for seismic damage is evaluated in three areas, (A) core and structural support, (B) pool and pool cooling, and (3) the building.

A. Core and structural Support

Given (1) the height of the reflector surrounded by a pool of water, (2) the distributed weight of the radial reflector around the core, and (3) the potential motion of fuel elements, hypothetical seismic event is not likely to create any significant acceleration that would not be absorbed by the pool water and/or mitigated by movement of the fuel elements followed by automatic re-centering of the elements in the lower gird plate. NUREG/CR-2387 (PNL-4028) analysis indicates that any disruption of the lattice by mechanical rearrangement would result in negative reactivity, increasing shutdown margin for a seismic event that dislocates, shifts, or otherwise moves fuel elements within the core

B. Pool and pool cooling

An aluminum liner is installed to provide integrity for the reactor pool. Beam ports penetrate the pool wall. However incredible, an earthquake has the potential to cause a loss of pool integrity and therefore is postulated for analysis as a loss of cooling accident. The consequences of a loss of cooling accident are addressed in Chapter 13.

C. Building

A building of good construction should withstand an earthquake acceleration of about 0.75 g. Ground accelerations that exceed this would be rare events in a region in which earthquakes are already infrequent.

4.0 Reactor

This chapter will discuss the reactor core (fuel, control rods, reflector and core support, neutron source, core structure), reactor pool, biological shielding, nuclear design (normal operating conditions, and operating limits), and thermal hydraulic design.

4.1 Summary description

The University of Texas Nuclear Engineering Teaching Laboratory (NETL) is home to a General Atomics' TRIGA Mark II research reactor. This installation follows 25 years (1963-1988) of successful operation of a TRIGA reactor at Taylor Hall on the main campus.

The basic TRIGA design uses U-$ZrH_{1.6}$ fuel clad with stainless steel in natural water convection cooling mode during operation, with a maximum decay heat that can be removed by natural convection of either water or air. The reactor is located in an open pool of purified, light water that serves as a heat sink during operations at power. Nuclear properties and characteristics control heat generation; thermodynamic characteristics of the fuel and the coolant control heat removal and temperature response. Maximum fuel temperature is the principle design constraint. Solubility of hydrogen in the fuel matrix varies with temperature. Consequently, operation at high power levels (i.e., elevated fuel temperature) can cause hydrogen to evolve into space around the fuel matrix; the hydrogen at elevated temperature can generate pressure inside the cladding. Temperature that produces stress greater than the yield strength for the stainless steel cladding is lower than temperature which leads to phase change or melts U-$ZrH_{1.6}$.

TRIGA fuel has a very strong prompt negative fuel temperature coefficient. Fuel mass exceeding critical loading (i.e., excess reactivity) is required to compensate for the negative fuel temperature coefficient, as well as potential experiments, fission product poisons, and fuel burnup. There are several major experiment facilities that could affect core reactivity, as described in Chapter 10. Experiment program requirements vary widely; limits are imposed on the reactivity effects of experiments. The amount of excess reactivity determines the maximum possible power, and therefore the maximum possible fuel temperature.

4.2 Reactor Core

The University of Texas at Austin TRIGA II reactor core is configured in a hexagonal prism volume bounded by aluminum plates at the upper and lower surfaces (grid plates), and surrounded by a cylinder of graphite (aluminum clad) acting as a neutron reflector. Sections of the reflector are cut away to support experimental facilities, including beam ports and a rotating specimen rack. The core assembly is supported by structural aluminum, and includes an aluminum plate that serves to limit downward travel of control elements.

4.2.1 Reactor Fuel

The TRIGA fuel system was developed around the concept of inherent safety, with fuel and cladding designed to withstand all credible environmental and radiation conditions during its lifetime at the reactor site. A TRIGA fuel element consists of (A) a central fueled region containing fuel matrix, bounded by an axial reflector and (B) stainless steel end caps at the top and bottom in a stainless steel envelope (cladding sealed by end cap assemblies).

Design constraints limit internal fuel element pressure as a function of fuel and cladding temperature to prevent cladding rupture. The fuel lattice structure that comprises the NETL TRIGA reactor core contains integral inlet and outlet cooling channels in a geometry which, combined with the thermo-physical properties of the fuel element, assure natural convection is adequate to limit maximum steady state operating temperature. The TRIGA fuel matrix exhibits a large prompt negative temperature coefficient of reactivity. The maximum fuel temperature resulting from sudden insertion of all available excess reactivity would cause power excursion to terminate before any core damage is possible. Limits on core lattice excess reactivity and individual fuel element temperature therefore are interrelated. The maximum possible TRIGA fuel fission product inventory is limited by fissionable material loading. The maximum TRIGA fuel decay heat produced by fission product inventory can be removed by natural convection in air or water.

Handling, transport, and storage of TRIGA fuel elements at the NETL, fresh and irradiated, are described in Chapter 9, Auxiliary Systems.

A. Fuel matrix

A TRIGA fuel element consists of a central fueled region containing fuel matrix, bounded by an axial reflector (with a molybdenum disk as a protective interface between the fuel and the lower graphite/axial reflector, and stainless steel end caps at the top and bottom with a stainless steel cladding.

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. Two limiting temperatures are of interest, depending on the type of TRIGA fuel used. The TRIGA fuel which is considered low hydride, that with an H/Zr ratio of less than 1.5, has a lower temperature limit than fuel with a higher H/Zr ratio. Fig. 4.1 indicates that the higher hydride compositions are single phase and are not subject to the large volume changes associated with the phase transformations at approximately 530°C in the lower hydrides. Also, it has been noted[1] 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 4.1.

[1] GA-3618, Thermal Migration of Hydrogen in Uranium-Zirconium Alloys, Marten U. et. Al., General Dynamics, General Atomics Division (1962)

Graphite dummy elements may be used to fill grid positions in the core. The dummy elements are of the same general dimensions and construction as the fuel-moderator elements. They are clad in aluminum and have a graphite length [redacted]

Table 4.1, TRIGA Fuel Properties

Property	Mark III
Dimensions	
Outside diameter, $D_o = 2r_o$	[redacted]
Inside diameter, $D_i = 2r_i$	[redacted]
Overall length	[redacted]
Length of fuel zone, L	[redacted]
Length of graphite axial reflectors	[redacted]
End fixtures and cladding	[redacted]
Cladding thickness	[redacted]
Burnable poisons	[redacted]
Uranium content	
Weight percent U	[redacted]
^{235}U enrichment percent	[redacted]
^{235}U content	[redacted]
Physical properties of fuel excluding cladding	
H/Zr atomic ratio	[redacted]
Thermal conductivity (W cm^{-1} K^{-1})	[redacted]
Heat capacity [T ≥0°C] (J cm^{-3} K^{-1})	[redacted]
Mechanical properties of delta phase U-ZrH°	
Elastic modulus at 20°C	[redacted]
Elastic modulus at 650°C	[redacted]
Ultimate tensile strength (to 650°C)	[redacted]
Compressive strength (20°C)	[redacted]
Compressive yield (20°C)	[redacted]

A (1) Fabrication

A uranium loaded zirconium hydride was found to produce desired moderating characteristics and acceptably low parasitic neutron absorption with strong temperature feedback and high heat capacity. Feedstock of between [redacted] (or recycled material) are cast in controlled atmosphere, high-temperature induction furnace.[2] [redacted]

Fuel element castings are machined to cylinders of approximately 5 inches in length. A center hole is drilled the length of the cylinder. Additional machining is required for fuel meat to be

[2] TRIGA International: A New TRIGA Fuel Fabrication Facility at CERCA - Gerard Harbonnier, Jean-Claude Ottone, CERCA, Proceedings of the 1997 TRTR Annual meeting

fabricated into instrumented fuel assemblies (IFEs, described below) and fuel element followers. The cylinders are heated in a high temperature electric furnace with a hydrogen atmosphere. The exterior and center surface exposed to hydrogen induces the cylindrical fuel meat to hydride, with a target Zr:H ratio of 1.6. A pure zirconium filler rod is placed in the center hole to maintain nearly uniform thermo-hydraulic properties. Each TRIGA fuel element contains three of these machined pieces.

Instrumented elements have three chromel-alumel thermocouples embedded to about [redacted] from the centerline of the fuel, one at the axial center plane, and one each at [redacted] above and below the center plane. Thermocouple leadout wires pass through a seal in the upper end fixture, and a leadout tube provides a watertight conduit carrying the leadout wires above the water surface in the reactor tank.

Followers are machined to an outer radius of 1.25 in. (0.318 m) and 1.35 in. (0.0343 m) for the transient rod (air filled follower) and the standard rods (fuel followers) respectively.

A (2) Physical Properties

The zirconium-hydrogen system is essentially a simple eutectoid, with at least four separate hydride phases. The delta and epsilon phases are respectively face-centered cubic and face-centered tetragonal hydride phases. The two phase delta + epsilon region exists between $ZrH_{1.64}$ and $ZrH_{1.74}$ at room temperature, and closes at $ZrH_{1.7}$ at 455°C. From 455°C to about 1050°C, the delta phase is supported by a broadening range of H/Zr ratios. Other important properties observed for the delta phase U-ZrH are listed in Table 4.2.

The ratio of Zr-H plays a significant role in determining physical properties. The H:ZR material has a cubic structure in the delta-phase at ratios greater than 1.4. In lower H:Zr ratios (< 1.5) a phase change occurs at about 955°F (535°C) with large density differences between the phases leading to potential for deformation (swelling, and cracking). For hydrogen to zirconium atom ratios greater than 1.5, the matrix is single phase (delta or epsilon) and does not exhibit phase separation with thermal cycling. Thermal diffusion of hydrogen is minimal in higher ratios as well, minimizing potential for deformation from evolution of hydrogen gas. Any hydrogen gas is in equilibrium with the matrix, substantially retained by the cladding, Losses through the cladding from hydrogen migration are about 1% for cladding temperature about 930°F (500°C).

Table 4.2, Physical Properties of High-Hydrogen U-ZrH

Property	Temperature	Value	Units
Thermal Conductivity	93°C – 650°C	0.22	$W\ cm^{-1}{}^{\circ}K^{-1}$
Elastic Modulus	20°C	$9.1x10^6$	psi
	650°C	$6.0x10^6$	psi
Ultimate Tensile Strength	20°C	$2.4\ x10^4$	psi
Compressive Strength	20°C	$6.0\ x10^4$	psi
Compressive Yield	20°C	$3.5\ x10^4$	psi
Heat of Formation	298°C	37.75	$kcal\ g\text{-}mol^{-1}$

At ratios greater than 1.6 there can be a shift to higher density tetragonal. Higher hydride compositions are single phase and are not subject to the large volume changes associated with the phase transformations at approximately 530°C as in the lower hydrides. The stability extends from the minimum on the scale (0°C) to the maximum on the scale (950°C), indicating no volume changes from morphology which might stress cladding occur for a target ratio of 1.6 other than thermal expansion. Significantly, zirconium hydrides at these ratios lack any significant thermal diffusion of hydrogen under isothermal conditions. Under non-isothermal conditions, hydrogen migrates from high temperature regions to low temperature regions, with equilibrium disassociation pressures lower after redistribution. Hydrogen dissociates slightly from the fuel matrix at high temperatures, and is re-absorbed into the matrix at lower temperatures, with the equilibrium hydrogen dissociation pressure a function of both the composition and temperature. The equilibrium hydrogen dissociation pressure is governed by the composition and temperature. For $ZrH_{1.6}$, the equilibrium hydrogen pressure is one atmosphere at about 760°C. Hydrogen dissociation pressures of hydrides are similar in alloys up to about 75 weight per cent uranium. For the delta and epsilon phases, dimensional changes from hydrogen migration are not significant. In the delta phase, equilibrium disassociation pressures are related by:

$$\log P = K_1 + \frac{K_2}{T}$$

With:

P = pressure (atm)
T= temperature (K)
K1= $-3.8415 + 38.6433 \cdot X - 34.2639 \cdot X^2 + 9.28212 \cdot X^3$
K2= $-31.2981 + 23.5741 \cdot X - 6.0280 \cdot X^2$
X= hydrogen to zirconium atom ratio

At a ratio of 1.7 the equilibrium disassociation pressure corresponds to a temperature of about 1400°F (300°C). The density of ZrH decreases as hydrogen ratio increases; from low ratios to the delta phase (H:Zr of 1.5) the density change is high with little change for further increases. Massively hydrided bulk density is reported to be about 2% lower than x-ray diffraction analysis. For TRIGA fuel with a Zr:H ratio of 1:1.6, the uranium density, volume fraction, and weight fraction are related by:

$$\rho_U(A) = \frac{{}^{W}U}{0.177 - 0.125 \cdot {}^{W}U}$$

and

$${}^{W}U = \frac{0.177 \cdot \rho_U(A)}{1 + 0.125 \cdot \rho_U(A)}$$

$$\rho_U(A) = 19.07 \cdot V_f^{U(A)}$$

where

$\rho_U(A)$ = Uranium density
^{W}U = Uranium weight fraction
V_f = volume fraction of uranium in the U-$ZrH_{1.6}$ alloy

Thermal conductivity has been determined from short-pulse heating techniques. Using thermal diffusivity values, density, and specific heat the thermal conductivity of uranium zirconium with a Zr:H ratio of 1:1.6 is 0.042 ± 0.002 $cal^{-1}\ s^{-1}$ cm $°C^{-1}$.

Volumetric specific heat is a function of temperature and composition. Table 4.3 lists values for variations in Zr:H and w% U based on a 0°C reference, showing variation less than 10%.

Table 4.3, U-ZrH Volumetric Specific Heat Capacity (Cp)

ZrH	W% U	Value	Units
U-$ZrH_{1.6}$	8.5	$2.04 + 4.17x10^{-3}$	$W \cdot s \cdot cm^{-3}$
U-$ZrH_{1.7}$	20	$2.17 + 4.36x10^{-3}$	$W \cdot s \cdot cm^{-3}$

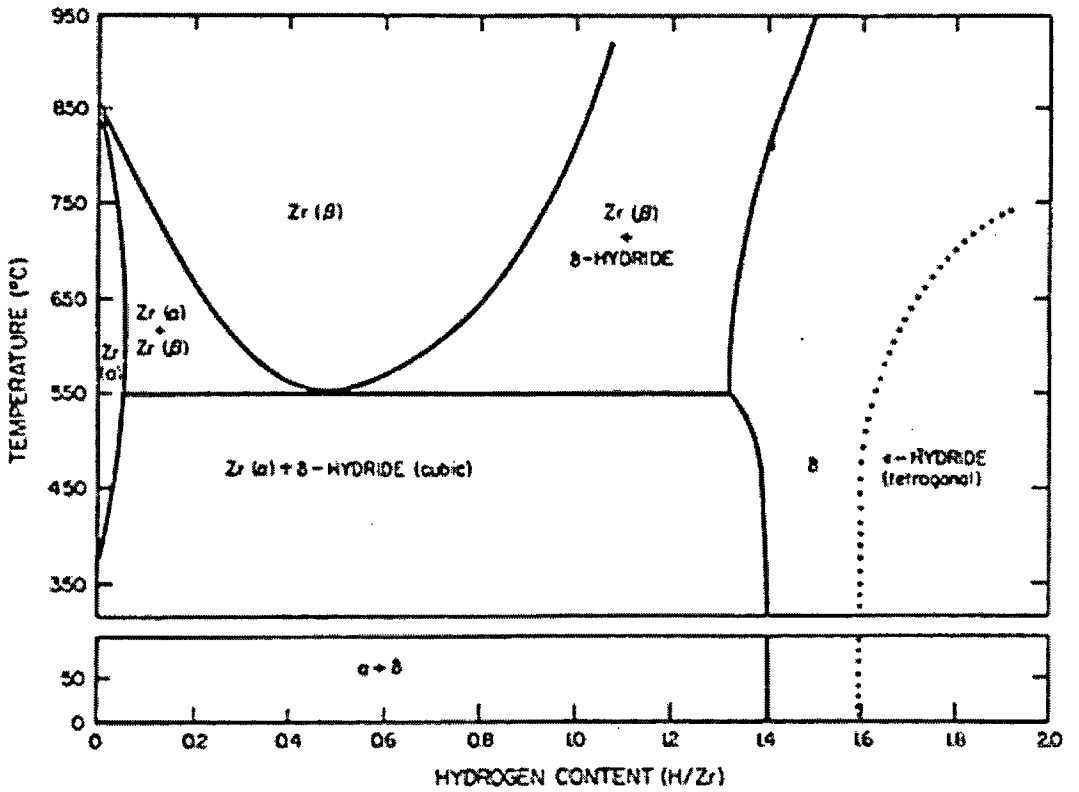

Figure 4.1: H/Zr Phase Diagram

A (3) Operational Properties

The neutronic properties of ZrH are the primary motivation for incorporation in TRIGA fuel development. The morphology of ZrH, in particular hydrogen diffusion in the material, imposes limits during operation. Ultimately, personnel exposure related to TRIGA fuel is limited during normal operations and abnormal events by retaining fission products in the fuel elements. It is well known that zirconium can undergo a reaction with water that releases hydrogen, with subsequent potential for a mixture that can be detonated. Such a reaction has the potential to release a large fraction of fission product inventory of affected fuel elements, but is not likely given characteristics of operation and properties of the fuel matrix. Fuel element changes occur during operation from thermal stress, which can affect fuel performance. Fuel cladding prevents migration of fission products for the fuel element, but in the absence of cladding it is not likely that all fission products will escape the fuel meat. Finally, thermal effects related to fuel matrix from steady state and pulsing operations are considered.

A (4) Neutronic Properties A large fraction of neutron moderation occurs through interactions with hydrogen in the fuel matrix. The zirconium hydride structure has a profound effect on neutron scattering at low energies because of zirconium-hydrogen binding, with distinct lattice energy levels of 0.13 eV and about 0.25 eV found in scattering experiments. Thermal neutrons that interact with hydrogen in the lattice (where neutron energy is below the lattice energies) therefore have potential to gain energy. Because the fission cross section has 1/v dependence in the thermal range, increasing thermal neutron energy decreases fission probability. If fuel temperature increases, thermal excitation creates more of these relatively high-energy lattice centers as indicated in Fig. 4.2a. When the rate of fission is high enough to create elevated fuel temperatures, the elevated fuel temperatures decrease the rate of fission. This phenomenon is responsible for an extremely high feedback of negative reactivity from fuel temperature illustrated in Fig. 4.2b. Maximum possible fuel temperature and maximum theoretical power level are therefore a function of the amount of fuel in the reactor.

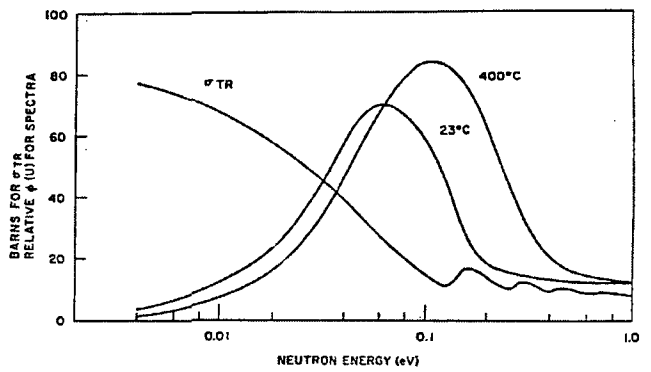

Figure 4.2A, Zr-H Transport Cross Section & TRIGA

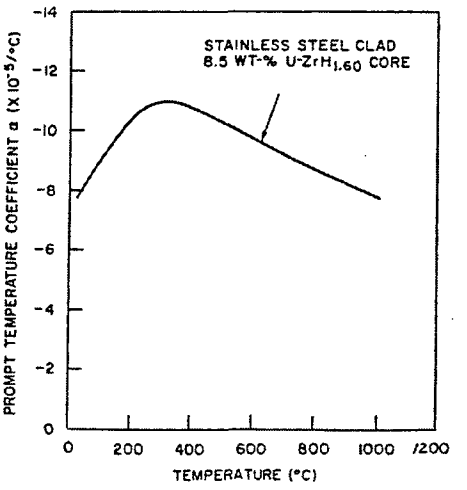

Figure 4.2B, Fuel Temperature Coefficient of

A (5) Fuel Morphology & Outgassing As noted previously, during fuel fabrication the ratio of hydrogen to zirconium is enhanced by thermally induced diffusion in an atmosphere of pressurized hydrogen. During reactor operation, temperature gradients influence hydrogen diffusivity to promote outgassing, bounded by temperature induced pressurization of the hydrogen in free volume of the cladding. Pressure inside the fuel element does not intrinsically pose a challenge to fuel element integrity, and will be considered as part of cladding performance in a later section. At a given temperature, higher H:Zr ratios (in the absence of phase change) exhibit more pressure at a given temperature in a well behaved relationship, shown in Fig. 4.3. Thermal diffusion is accelerated at higher temperatures, but the expansion of free hydrogen gas at higher temperatures also produces more partial gas pressure in the free volume of the element. Calculations performed with a higher mass fraction of uranium result in an increase in the partial pressure of hydrogen by as much as a factor of four.[3]

The fuel rod diameter is on the order of the path length of neutron from generation to absorption, and the mean free path for thermal neutrons within the fuel rod is not large. Consequently, a large fraction of power in a TRIGA fuel element is produced close to the outer surface of the fuel. Fuel rod temperature gradient during normal, steady-state operations is monotonically decreasing from a peak at the center of the fuel rod. Routine power changes occur at a rate that allows quasi-steady state thermal equilibrium, but pulsing operations do not. As a consequence, power distribution and development of temperature gradients in steady-state operations is fundamentally different compared to fast transient (pulsing) operations.

In general, gas pressure during the transient of pulsing operations is expected to be less than during steady state. Diffusion rates are finite, and the diffusion coefficient for thermal diffusion of hydrogen in zirconium[4] (ranging from $4x10^{-5}$ to $2x10^{-8}$ cm^2 s^{-1}, and requiring days to equilibrate) lags the time constant for the temperature changes. The temperature gradient during the transient peaks near the surface of the fuel rod rather than the center, and rapidly vanishes as the system comes to equilibrium. Therefore thermal gradients in pulsing bias hydrogen diffusion towards the center of the fuel rod with only a small region near the surface having a gradient that promotes outgassing. Surface cooling from endothermic gas emission lowers the surface temperature and therefore tends to lower the diffusion constant at the fuel rod surfaces. Re-absorption occurs where hydride surfaces are at relatively lower temperatures. There is evidence that low permeability oxide films on fuel surfaces retard mass transfer. Local heat transfer effects cause the surface temperature to be lower than that which would occur during adiabatic conditions.

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[3] Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 37, No. 10, p. 887-892 (October 2000): *Estimation of Hydrogen Redistribution in Zirconium Hydride under Temperature Gradient*

[4] Congreso Internacional de Metalugia y Materiales, Primeras Jornadas Internacionales de Materiales Nucleares (19 al 23 de Octubre de 2009, Buenos Airesm Argentina; Some Peculiarities of Hydrogen Behavior and Delayed Hydride Cracking in Zirconium Based Reactor Alloys, Shmakov, R.N. Singh

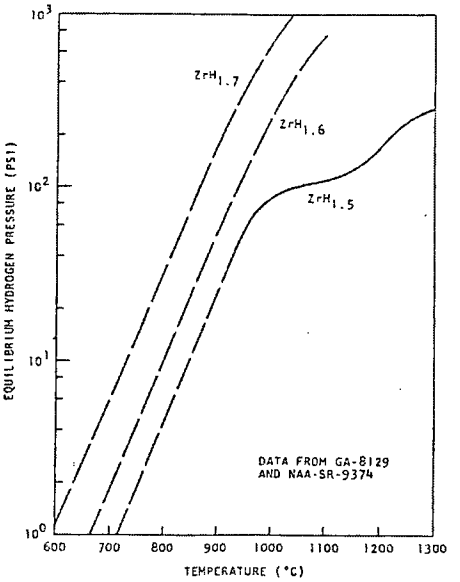


Figure 4.3, Thermal Pressurization in Fuel and Hydriding Ratios

Long term operations with steady state fuel temperatures exceeding 750°C (1023°K) may have time- and temperature-dependent fuel growth.[5] Mechanisms contributing to the growth are identified as fission recoils and gaseous fission products, strongly influenced by thermal gradients. Analysis of steady state operating fuel temperatures is provided in section 4.6, with pulsing operations fuel temperatures in Appendix 4.1.

A (6) Zr water reaction 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 Technologies show that the zirconium hydride systems have a relatively low chemical reactivity with respect to water and air[6]. These tests have involved the quenching with water of both powders and solid specimens of U-ZrH after heating to as high as 850°C, and of solid U-Zr alloy after heating to as high as 1200°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 is retained in even completely unclad U-ZrH fuel.

A (7) Mechanical Effects At room temperature the hydride is like ceramic and shows little ductility. However, at the elevated temperatures of interest for pulsing, the material is found to be more ductile.

[5] General Atomics Technical Report E-117-833

[6] NUREG/CR-2387 Credible Accidents for TRIGA and TRIGA Fueled Reactors, S. C. Hawley,S. C. and Kathren, R. L. , PNL-4208 (1982)

The effect of very large thermal stress 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. During fabrication, a molybdenum disk is placed between the lowest fuel mass and the lower axial-graphite reflector, minimizing potential for interaction that might affect the graphite and cause position changes in fuel meat that has developed surface imperfections. Anticipated mechanical effects from operation of the reactor are not expected to create conditions that challenge fuel performance.

A (8) Fission Product Release Early in development of U-ZrH_x fuel, experiments were performed[7] to determine the potential of the evolution of fission products from the fuel matrix. Zr-U-H alloy foils were irradiated in a materials test reactor and a post irradiation test conducted, with water flowing across the surface of the foil to remove fission products for analysis. The test was performed for 1 day and for 8 days with the total fractional fission product loss calculated to be between 10^{-7} and 10^{-5} from preferential leaching of radionuclides, with gasses evolving from depths of 2.6 μm in the foil, and particulate from 22 Å. Acceptable[8] upper values for release fraction are 1.0×10^{-4} for noble gases and iodine contained within the fuel, and of 1.0×10^{-6} for particulates (radionuclides other than noble gases and iodine). Experiments by General Atomics [Simnad et al., 1976] indicate a value of 1.5×10^{-5} for noble gases, which is in SARs for other reactor facilities [NUREG-1390, 1990].

B. Cladding

The fuel matrix is enveloped by a cylindrical [redacted] 304 stainless steel shell, welded to stainless steel fittings at each end (end caps). The cladding is the principal barrier to release of those fission products that migrate to escape the fuel matrix surface. As noted previously, the free hydrogen in the space within the fuel element pressurizes the interior of the fuel element when fuel temperature is elevated during reactor operations. Power levels are acceptable if they do not result in temperatures that produce stress from the gas pressure that challenges the integrity of the cladding. A cylinder is considered a thin shell if wall thickness is less than about 10% of the radius and the classic equation for hoop stress created by internal pressure is:

$$\sigma_\theta = P \cdot r/t$$

where:

σ_θ is the hoop stress
P is internal pressure
r is inside radius
t is the wall thickness

[7] General Atomic report GA-655, Uranium-Zirconium-Hydride Fuel Elements, Merten, Stone, Wallace (1959)

[8] NUREG/CR-2387, op. cit.

For [redacted], stress is [redacted] times the internal pressure. Figure 4.4A provides temperature dependent ultimate strength and the 0.2% yield, and Figure 4.4B shows where the hoop stress induced by the internal pressure intersects with ultimate strength. This intersection corresponds to a fuel temperature of 950°C for cladding temperatures greater than 500°C.

Therefore, if fuel and cladding temperature remains below 950°C with cladding temperatures greater than 500°C, the stainless steel cladding will not fail from overpressure. For cladding temperatures less than 500°C, hydrogen pressure from peak fuel temperature of 1150°C would not produce a stress in the clad in excess of its ultimate strength. The limiting fuel temperature and pressure is therefore the design basis for the UT TRIGA fuel. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad[9].

There are several reasons why the gas pressure should be *less* for the transient conditions than the equilibrium condition values would predict. For example, the gas diffusion rates are finite; surface cooling is believed to be caused by endothermic gas emission which tends to lower the diffusion constant at the surface. Reabsorption takes place concurrently on the cooler hydride surfaces away from the hot spot. There is evidence for a low permeability oxide film on the fuel surface. Some local heat transfer does take place during the pulse time to cause a less than adiabatic true surface temperature.

[9] "Annual Core Pulse Reactor," General Dynamics, General Atomics Division report GACD 6977 (Supplement 2), Dee. J. B., et. Al.

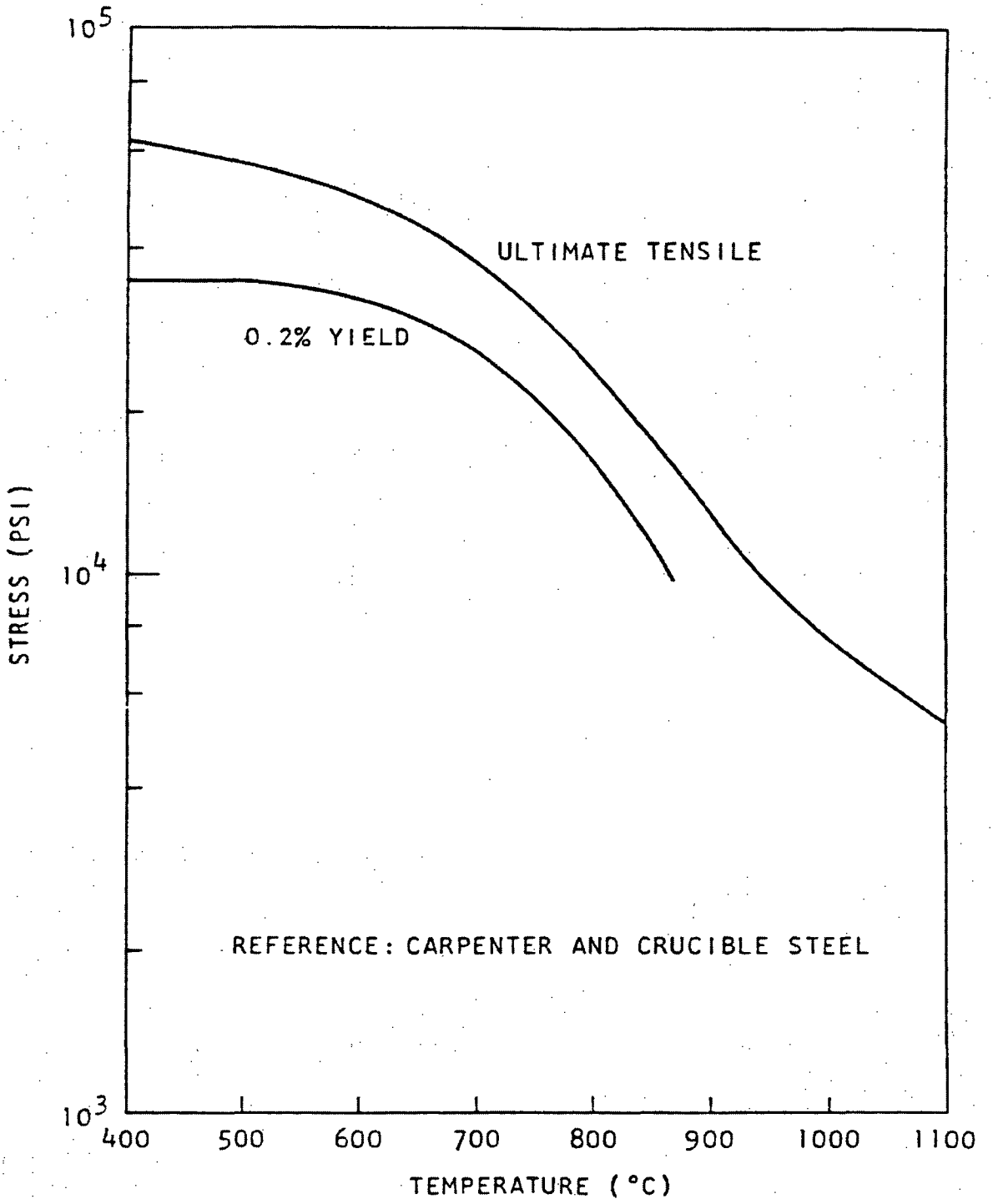


Figure 4.4A, Temperature and Cladding Strength for 0.2% Yield

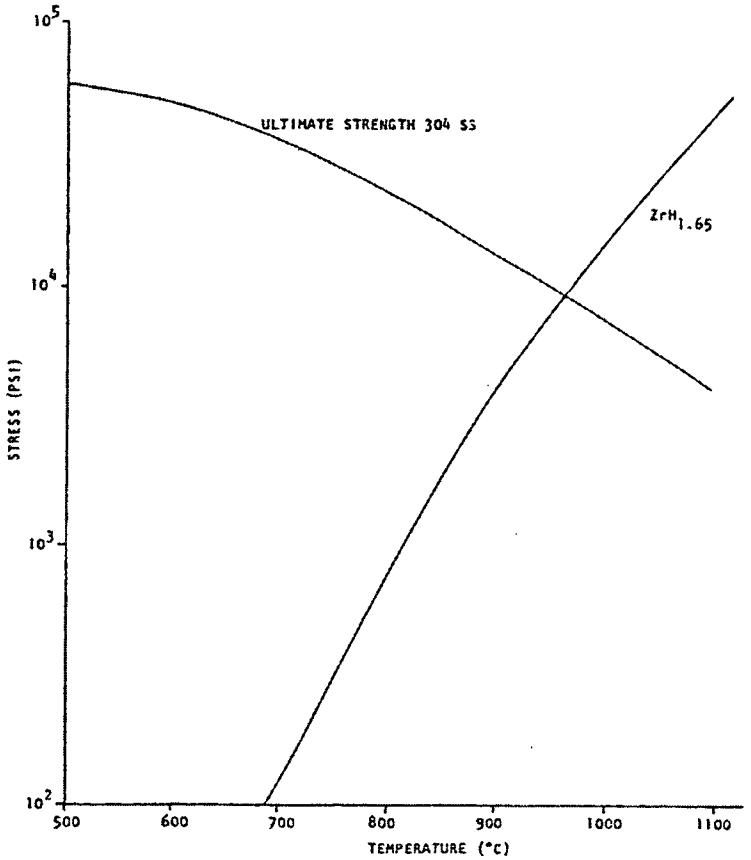


Figure 4.4B, Temperature, Cladding Strength, and Stress

4.2.2 Control Rods and Drive Mechanisms

The control rods and drive mechanisms consist of (A) control rods, (B) standard control rod drives, (C) transient rod drives, (D) control functions, and (E) system operation. The UT TRIGA reactor was installed with 4 control rods, three standard rods magnetically coupled to the control rod drive, and one pulse rod pneumatically coupled to the control rod drive. One of the standard rods, the regulating rod, is capable of being either automatically controlled with instrumentation and control systems described in Chapter 7 or manually from the reactor control console. The other control rods are manually shimmed. Principle design parameters for the control rods are provided in Table 4.4.

A. Control Rods

The standard control rods (regulating and shim) are sealed 304 stainless steel tubes approximately 43 in. (109 cm) long by 1.35 in. (3.43 cm) in diameter in which the uppermost 6.5 in. (16. 5 cm) section is an air void, followed by 15 in. (38.1 cm) of a neutron absorber, solid boron carbide. Standard control rods have a fuel follower attached so that as the control rod is withdrawn from the core the water channel is filled with a fuel element as illustrated in Fig. 4.6. The fuel follower, 15 in. (0.381 cm) of U-$ZrH_{1.6}$ fuel, is immediately below the neutron absorber

of the standard control rods. The bottom 6.5 in. (16.5 cm) of the standard control rod is an air void. Table 4.4 summarizes control rod design parameters.

The transient (also called safety-transient or pulse) rod is a sealed, 36.75 in. (93.35 cm) long by 1.25 in. (3.18 cm) diameter tube containing boron in graphite as a neutron absorber. Below the absorber is an air filled follower section. The absorber section is 15 in. (38.1 cm) long and the follower is 20.88 in. (53.02 cm) long. The transient rod passes through the core in a perforated aluminum guide tube. The tube receives its support from the safety plate and its lateral positioning from both grid plates. It extends approximately 10 in. (25.4 cm) above the top grid plate. Water passage through the tube is provided by a large number of holes distributed evenly over its length. A locking device is built into the lower end of the assembly.

Table 4.4, Summary of Control Rod Design Parameters

Cladding				
Material	Aluminum		SS 304	
OD	1.25 in.	3.18 cm	1.35 in.	3.43 cm
Length	36.75 in.	93.35 cm	43.13 in.	109.5 cm
Wall thickness	0.028 in.	0.071 cm	0.02 in.	0.051 cm
Poison Section				
Material	Boron Carbide			
OD	1.19 in.	3.02 cm	1.31 in.	3.32 cm
Length	15 in.	38.1 cm	14.25 in.	36.20 cm
Follower Section				
Material	Air		U-ZrH1.6	
OD	1.25 in.	3.18 cm	1.31 in	3.34 cm
Length	20.88 in.	53.02 cm		

Control rods are withdrawn out of the core through the upper grid plate; when fully inserted the followers extend down through the lower grid plate. All fuel element position penetrations in the upper grid plate are identical; the lower grid plate (an excerpt in Fig. 4.5, fully described later in Chapter 4) has a set of 11 penetrations in the C and D rings (shaded in gray and black in Fig. 4.5, black representing the current configuration) with the same diameter as the upper grid plate. One of these penetrations in reserved for the central thimble (position A1) while the others are available for use as control rod positions. A safety plate is mounted below the lower grid plate as shown in Fig. 4.6, so that the control rod cannot exit the core region in the downward direction.

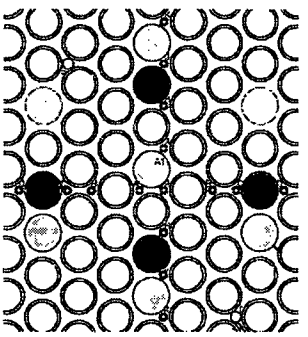


Figure 4.5, Lower Gird Plate Control Rod Positions

Control rod worth is principally a function of control rod dimensions and location, experiment facilities in the core, with lessor influence by fuel and control rod burnup. Estimated control rod from the 1991 preliminary safety analysis report is provide in Table 4.5, along with the worth of each control rod as measured in June 2011. Sections of the control rod are separated and secured by 1-inch magneform fittings.

Table 4.5, Control Rod Information

Rod	Location	Diameter		Estimated (1991)		Current (2011)
		In.	cm.	% Δk/k	$	$
Transient Rod	C Ring	1.25	3.18	2.1	3.00	3.10
Regulating Rod	C ring	1.35	3.43	2.6	3.71	2.82
Shim 1	D ring	1.35	3.43	2.0	2.86	2.52
Shim 2	D ring	1.35	3.43	2.0	2.86	3.07

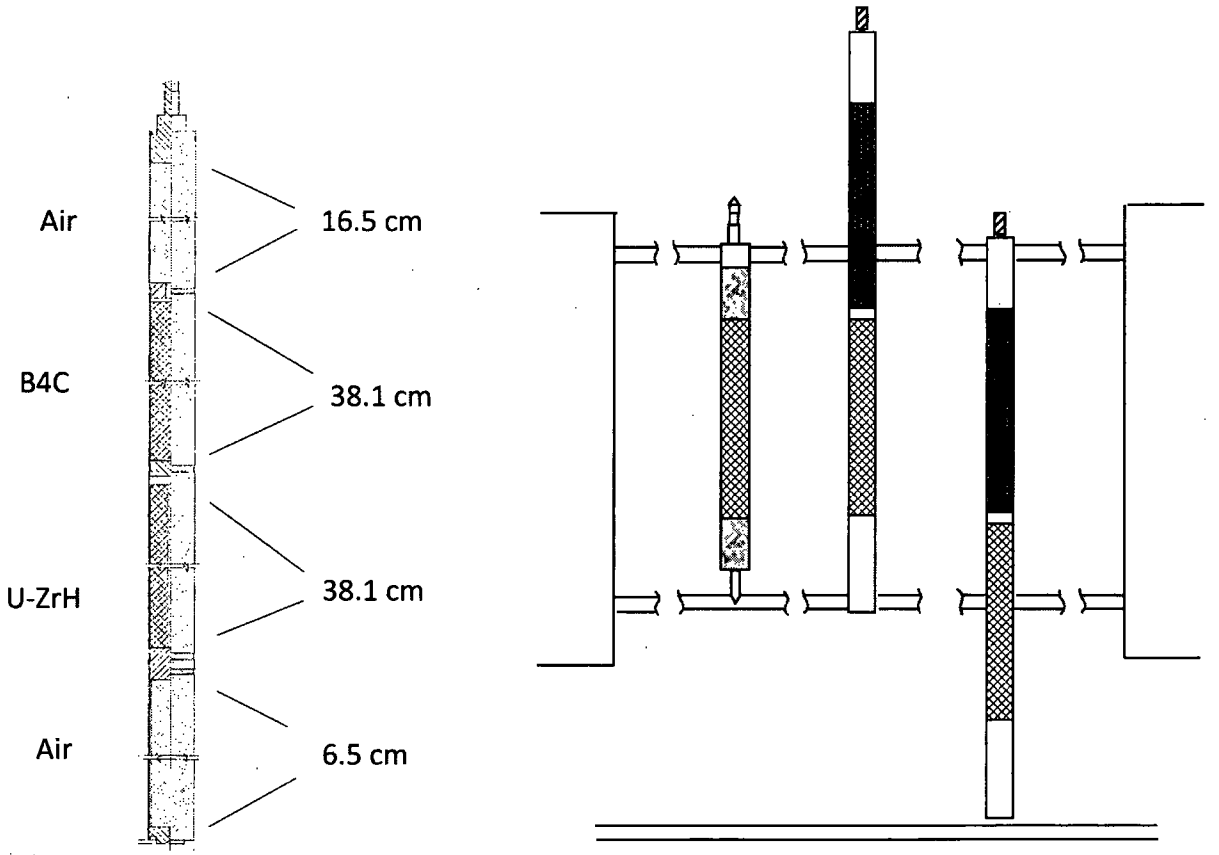


Figure 4.6, Standard Control Rod Configuration

A threaded fitting at the end of each control rod connects to a series of shafts that connect to control rod drive mechanisms mounted on a bridge that spans the reactor pool. The top section of the connecting shafts for standard rods passes through a hole in the bottom of a tube supported by the control rod drive housing. The tube is designed with slots that provide a hydraulic cushion for the rod during a scram, and also prevent the bottom of the control rod from impacting the safety plate.

The shaft is secured to a cylinder that rests on the bottom of the housing when the rod is fully inserted. The top of the cylinder is secured to an iron core, engaged by an electromagnet for fail-safe control. The electromagnet is at the bottom of a small shaft controlled by the control rod drive mechanism. When the electromagnet is energized, the iron core is coupled to the drive unit.

The top section of the transient rod is connected to a single acting pneumatic cylinder which operates on a fixed piston that couples the connecting rods to the drive. The transient rod drive is mounted on a steel frame that bolts to the bridge. Any value from zero to a maximum of 15 in. (38.1 cm.) of rod may be withdrawn from the core; rod travel is limited by administrative control not to exceed to the maximum licensed step insertion of reactivity.

B. Standard Control Rod Drives

The rod drive mechanism for the standard rod drives is an electric stepping-motor-actuated linear drive equipped with a magnetic coupler and a positive feedback potentiometer. A stepping motor drives a pinion gear and a 10-turn potentiometer via a chain and pulley gear mechanism. The potentiometer is used to provide rod position information. The pinion gear engages a rack attached to the magnet draw tube. An electromagnet, attached to the lower end of the draw tube, engages an iron armature. The armature is screwed and pinned into the upper end of a connecting rod that terminates at its lower end in the control rod. When the stepping motor is energized (via the rod control UP/DOWN switch on the reactor control console), the pinion gear shaft rotates, thus raising the magnet draw tube, the armature and the connecting rod will raise with the draw tube so that the control rod is withdrawn from the reactor core. In the event of a reactor scram, the magnet is de-energized and the armature will be released. The connecting rod, the piston, and the control rod will then drop, thus reinserting the control rod.

Stepping motors operate on phase-switched direct current power. The motor shaft advances 200 steps per revolution (1.8 degrees per step). Since current is maintained on the motor windings when the motor is not being stepped, a high holding torque is maintained. The torque versus speed characteristic of a stepping motor is greatly dependent on the drive circuit used to step the motor. To optimize the torque characteristic for the motor frame size, a Translator Module was selected to drive the stepping motor. This combination of stepping motor and translator module produces the optimum torque at the operating speeds of the control rod drives. Characteristic data for the drive indicate a possible travel rate of 33 ipm (1.40 cm/s). Measurements of the actual rate provide a speed of 27 ipm (1.14 cm/s).

C. Transient Control Rod Drive

The safety transient control rod drive is operated with a pneumatics rod drive. Operation of the transient rod drive is controlled from the reactor control console. The transient rod is a scrammable rod operated in both pulse and steady-state modes of reactor operation. During

steady state operation, the transient rod will function as an alternate safety rod with air continuously supplied to the rod. Rod position is thus controlled by operation of an electric motor that positions the air drive cylinder. The position of the transient control rod and its associated reactivity worth will generally dictate removal of the rod as the first step of a startup for steady-state operation. Rod withdrawal speed is about 28 ipm (1.19cm/s).

The transient rod drive is a single-acting pneumatic cylinder with its piston attached to the transient rod through a connecting rod assembly. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air is supplied to the lower end of the cylinder from an accumulator tank when a three -way solenoid valve located in the piping between the accumulator and cylinder is energized. The compressed air drives the piston upward in the cylinder and causes the rapid withdrawal of the transient rod from the core. As the piston rises, the air trapped above it is pushed out through vents at the upper end of the cylinder. At the end of its travel, the piston strikes the anvil of an oil filled hydraulic shock absorber, which has a spring return, and which decelerates the piston at a controlled rate over its last 2 in. (5 cm.) of travel. When the solenoid is de-energized, a solenoid valve cuts off the compressed air supply and exhausts the pressure in the cylinder, thus allowing the piston to drop by gravity to its original position and restore the transient rod to a position fully inserted in the reactor core.

The extent of transient rod withdrawal from the core during a pulse is determined by raising or lowering the de-coupled cylinder, thereby controlling the distance the piston travels when air is applied. The cylinder has external threads running most of its length, which engage a series of ball bearings contained in a ball-nut mounted in the drive housing. As the ball-nut is rotated by a worm gear, the cylinder moves up or down depending on the direction of worm gear rotation.

A ten-turn potentiometer driven by the worm shaft provides a signal indicating the position of the cylinder and the distance the transient rod will be ejected from the core. Motor operation for pneumatic cylinder positioning is controlled by a switch on the reactor control console. The magnet power key switch on the control console power supply prevents unauthorized firing of the transient rod drive.

Attached to and extending downward from the transient rod drive housing is the rod guide support, which serves several purposes. The air inlet connection near the bottom of the cylinder projects through a slot in the rod guide and prevents the cylinder from rotating. Attached to the lower end of the piston rod is a flanged connector that is attached to the rod assembly that moves the transient rod. The flanged connector stops the downward movement of the transient rod when the connector strikes the damp pad at the bottom of the rod guide support. A microswitch is mounted on the outside of the guide tube with its actuating lever extending inward through a slot. When the transient rod is fully inserted in the reactor core, the flange connector engages the actuating lever of the microswitch and indicates on the instrument console that the rod is in the core. In the case of the transient rod a scram signal de-energizes the solenoid valve which supplies the air required to hold the rod in a withdrawn position and the rod drops into the core from the full out position in less than 1 second.

D. Control Functions

Instrumentation and controls provide protective actions through the control rod system, as described in Table 4.6. A trip signal from the reactor protection system or the reactor control systems will deenergize the electro magnets and the pulse rod air solenoid valve previously described which allows gravity to insert the control rods.

Table 4.6, Summary of Reactor SCRAMs

Measuring Channel	Limiting Trip Setpoint		Actual Setpoint
	Steady State	Pulse	
Maximum thermal power	1100 kW	2000 MW	SS - 1050 (NPP/NP) 1080 NM Pulse – 1910 NPP
Power Channel High power	110%	110%	
Detector High Voltage	80%	80%	
High Fuel Temperature		550°C	
Magnet current loss			
Manual Scram			
DAC and CSC watchdog timers			

In addition, the reactor control system (described in Chapter 7) has interlocks to prevent various conditions from developing. Table 4.7 is a summary of the functions.

Table 4.7, Summary of Control Rod Interlocks

INTERLOCK	SETPOINT	FUNCTION/PURPOSE
Source Interlock	2 cps	Inhibit standard rod motion if nuclear instrument startup channel reading is less than instrument sensitivity/ensure nuclear instrument startup channel is operating
Pulse Rod Interlock	Pulse rod inserted	Prevent applying power to pulse rod unless rod inserted/prevent inadvertent pulse
Multiple Rod Withdrawal	Withdraw signal, more than 1 rod	Prevent withdrawal of more than 1 rod/Limit maximum reactivity addition rate (does not apply in automatic flux control)
Pulse Mode Interlock	Mode switch in Hi Pulse	Prevent withdrawing standard control rods in pulse mode
Pulse-Power Interlock	10 kW	Prevent pulsing if power level is greater than 10 kW

These safety settings are conservative in the sense that if they are adhered to, the consequence of normal or abnormal operation would be fuel and clad temperatures well below the safety limits indicated in the reactor design bases. Because of the conservatism in these safety settings, it is reasonable that at some later date less restrictive safety system settings could be assigned in conjunction with upgrading of the reactor to operate at higher steady-state power levels or in the pulsing mode while using the same fuel and core configuration.

Administrative limitations are imposed for the excess reactivity, transient conditions and coolant water temperature as follows:

1) Maximum core excess reactivity of 4.9% Δk/k ($7.00) with a shutdown margin of at least 0.2% Δk/k ($0.29) with the most reactive control rod fully withdrawn,

2) Maximum transient control rod worth of 2.8% Δk/k ($4.00) with a limit of 2.2% Δk/k ($3.14) for any transient insertion, and

3) Core inlet water temperature of 48.9°C.

E. Evaluation of the Control Rod System

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The TRIGA system does not rely on speed of control for reactor safety; scram times for the rods are measured periodically to monitor potential degradation of the control rod system. The inherent shutdown mechanism (temperature feedback) of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady state operation. A scram does not challenge the control integrity or operation, or affect the integrity or operation of other reactor systems.

4.2.3 Neutron Moderator and Reflector (Core Structure)

The UT TRIGA core is supported within a reflector assembly. The reflector assembly supports (A) an upper grid plate, (B) core barrel and reflector, and (C) lower grid plate, shown in Fig. 4.7a/b. The upper and lower grid plates provide alignment and support for the fuel elements.

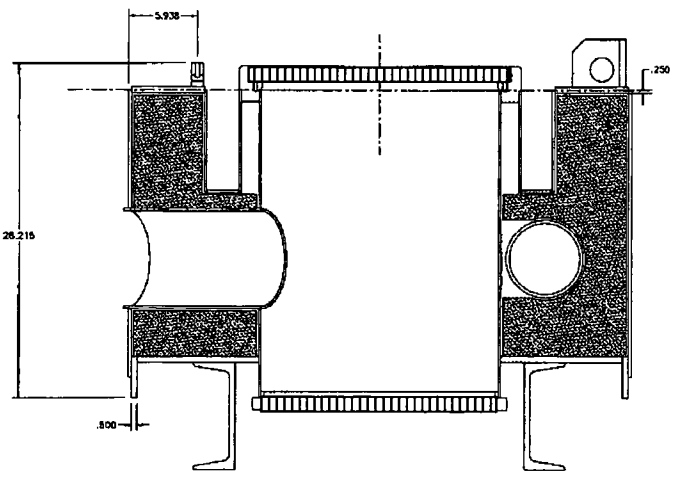


Figure 4.7a, UT TRIGA Core

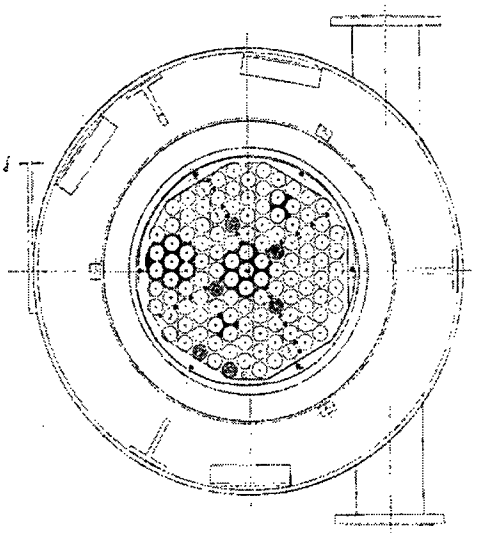


Figure 4.7b, Core Top View

A. Upper grid plate

The upper grid plate provides alignment for fuel elements and control rods, and (in conjunction with the top fuel assembly) space for cooling flow. The top grid plate is fabricated from a circular aluminum plate 5/8 inches (1.59 cm.) thick and 21.6 in. (55.245 cm) diameter, anodized to resist wear and corrosion. The top of the upper grid plate is 59 in. (150 cm.) above the bottom of the pool. [redacted] diameter are established on a triangular pitch of 1.714 in. (4.35 cm), separated by radial fuel arrays integrated on the same pitch, although the radial arrays do not extend to the edge of the core. The holes position the fuel-moderator and graphite dummy elements, the control rods and guide tubes, the pneumatic transfer tube, and the central thimble. Small 0.203 in. (8 mm) holes at various positions in the top grid plate permit insertion of wires or foils into the core to obtain flux data. The flux probe holes are counter sunk/chamfered to (82°) to 0.31 in. (11 mm). The center fuel element position is reserved as an experimental facility. The outermost fuel positions in the radial arrays are not fabricated for fuel insertion. Upper grid plate penetrations are summarized in Table 4.8.

Table 4.8, Upper Grid Plate Penetrations

Penetration Function	Size
Fuel Elements	1.505 in. (3.8227 cm) diameter
3-element	1.2 in. (3.048 cm) radius
6/7-Element	2.2 in. (5.588 cm) radius
Upper grid plate alignment	3/8 in. (0.9525 cm) diameter
Flux probes	0.203 in. (0.5156 cm) diameter

The grid plate is supported by a ring welded to the top inside surface of the reflector container. The ring is fabricated with bosses that hold alignment pins to engage and center the upper grid plate using ⅜ in. (0.953 cm) holes centered along each of the hexagonal faces of the G ring fuel positions.

Fuel positions are indexed by letters denoting a "ring" where elements are collinear with respect to the adjacent radial array fuel positions; A is the central ring position and G is furthest from the center. One radial array is used as a reference position, and the fuel positions range from 1 at the index to the maximum value for the ring, except for the G ring. Since the vertices of the G ring are not used as fuel positions, index numbers for the G ring vertices are not used.

Circular cutouts to replace fuel element positions are fabricated using two different designs, 3-element fuel position facilities and 7-element fuel position facilities (6-element for the facility encompassing the central thimble since the central thimble does not contain fuel). [redacted] The inserts mesh in slots milled in the circular grid plate cutouts; engagement secures the insert. There are two locations fabricated for each design. The 6/7 element facilities permit specimen as large as 4.4 in, (11.8 cm) and the 3 element facilities permit specimen as large as 2.4 in. (6.1 cm).

In addition to the experiment facilities that replace fuel positions, the current core configuration reserves one position for a neutron source, one position for a pneumatic facility, and four positions for control rods. Table 4.9 summarizes fuel element positions displaced or potentially displaced by core equipment. For control rods, only currently used positions are identified; there are alternate positions useable for control rods.

Table 4.9, Displaced Fuel Spaces

Facility	Core Location
[redacted]	[redacted]

B. Reflector

The core is surrounded by a graphite radial reflector for neutron economy. In addition, graphite cylinders are positioned within the fuel cladding above and below the active fuel region.

B (1) Radial Reflector. The radial reflector is a 10.2 in. (25.91 cm) graphite ring with an inner diameter of 21 ⅝ in. (54.93 cm) that is 21 13/16 in. (54.40 cm) tall, surrounded by aluminum. The reflector is fabricated in a top and bottom section. Lifting bosses are located on the surface of the top section (Fig. 4.9a), with flat welded plates tying the top and bottom sections to the

lift points. Angle plate structures are welded on the outer perimeter as points to secure the power level detectors. A 3 inch (7.62 cm.) wide well is fabricated in the top section (Fig. 4.9b), and blocks with threaded penetrations are welded at the inner perimeter of the well to allow securing the rotary specimen rack (an experimental assembly) in the well.

Figure 4.9a, Reflector Top Assembly

Figure 4.9b, Reflector Bottom Assembly

The lower radial reflector is constructed of graphite contained in a welded aluminum canister. The graphite is machined to accommodate three beam ports oriented radial from the center of the reactor core, with one "through port" (Fig. 4.10a) and a 10 in. (25.3 cm.) cylinder cut from the inner surface to allow a 3 inch wide experimental facility surrounding the core.

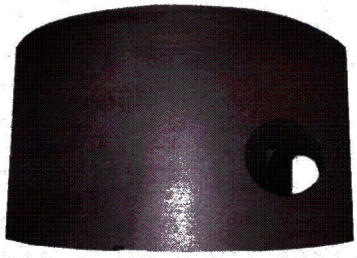


Figure 4.10a, Graphite Reflector, Through Port

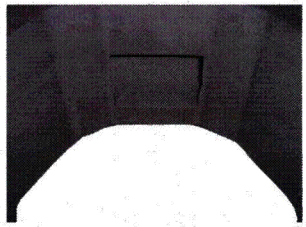


Figure 4.10b, Graphite Reflector Through port Detail

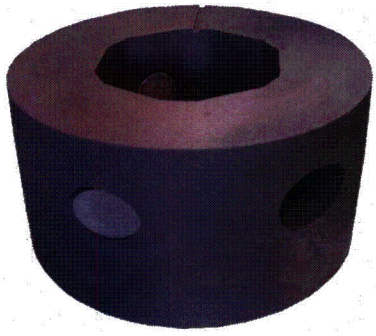


Figure 4.10c, Graphite Reflector, Radial & Piercing-Beam Ports

The through port has a rectangular water-filled cut-out between the core shroud and the beam port penetration (Fig. 4.10b). Aluminum canisters that mate with the beam ports are nested in the reflector in two of the beam ports, one radial and one tangential (Fig. 4.10c, Fig. 4.11a/b). The third beam port (radial) penetrates the core shroud (Fig. 4.11c).

Figure 4.11a, Tangential Beam Port Insert

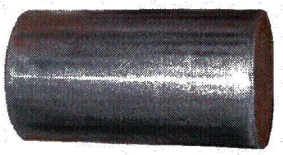


Figure 4.11b, Radial Beam Port insert

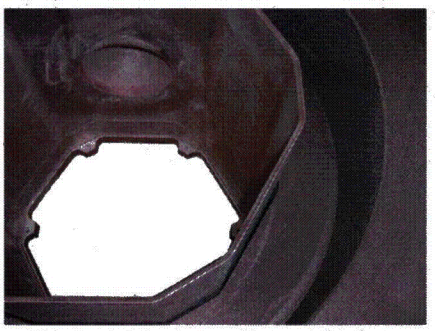


Figure 4.11c, Inner Shroud Surface

B (2) Graphite Rods. Graphite dummy elements may be used to fill grid positions not filled by the fuel-moderator elements or other core compounds. They are of the same general dimensions and construction as the fuel-moderator elements, but are filled entirely with graphite and are clad with aluminum.

B (3) Axial Reflector . Graphite cylinders are placed above and below the fuel in the fuel elements. Fuel element construction was previously discussed.

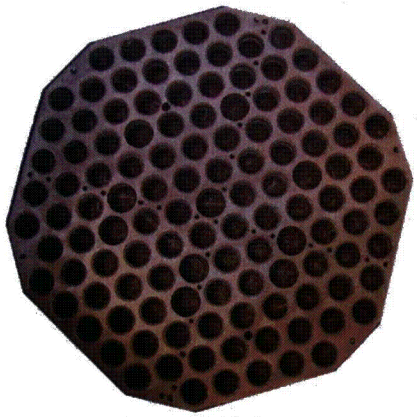
C. Lower grid plate

The lower grid plate (Fig. 4.12) provides alignment for fuel elements and control rods, and (in conjunction with the top fuel assembly) space for cooling flow. The lower (or bottom) grid plate is fabricated from a circular aluminum plate 1.75 inches (3.81 cm.), anodized to resist wear and corrosion. The top of the bottom grid plate is 9.9 in. (25.19 cm.) above the bottom of the pool. The bottom grid plate is fabricated with fuel position penetrations and penetrations matching the flux probe holes on the same center as the upper grid plate, but also contains penetrations that support alignment of the 3, 6, and 7 element facilities (Table 4.10). All but 11 fuel penetrations in the lower grid plate are smaller than the diameter of the fuel element and chamfered to provide a surface supporting triflutes on the bottom of the fuel element elements.

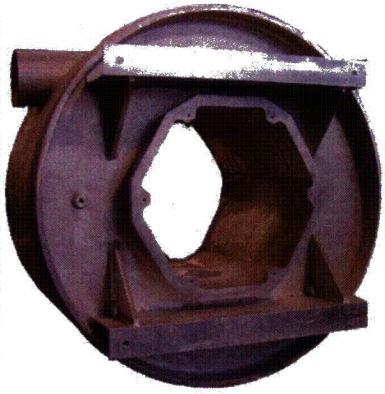
Table 4.10, Lower Grid Plate Penetrations

Penetration Function	Size
Central thimble	1.505
Control Rod	1.505
Flux Hole Probes	8 mm
3-Element Alignment	3/8 in.
Lower grid plate alignment	

Lower Grid Plate Support



Lower Grid Plate



Reflector Canister Bottom View

Grid Plate in Core Shroud

Figure 4.12, Reflector Component and Assembly Views

Ten lower grid plate penetrations are the same diameter as the penetration in the upper grid plate, providing clearance for the central thimble and control rods. Since only 4 controls rods are installed, unused control rod positions (i.e., large diameter holes) can be used for fuel with an adapter to support positioning the fuel above the lower grid plate (Fig. 4.13).

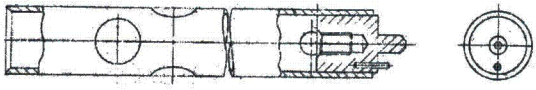


Figure 4.13, Fuel Element Adapter

4.2.4 Neutron Startup Source

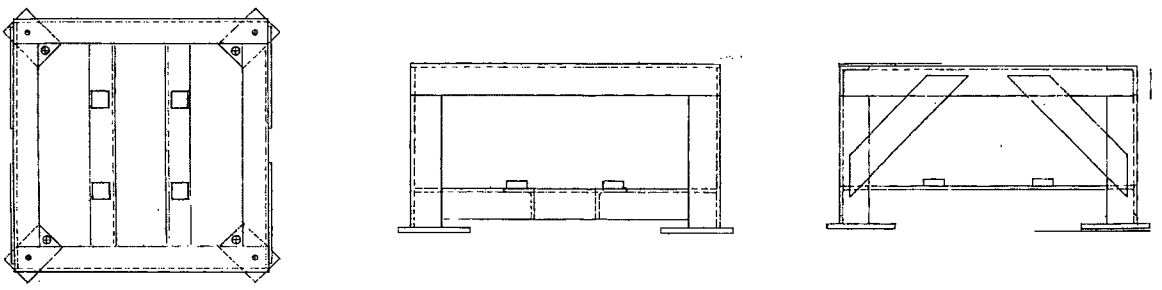
The reactor license permits the use of sealed neutron sources, including a [redacted] is a standard sealed neutron source, encapsulated in stainless steel. The source is maintained in an aluminum-cylinder source holder of approximately the same dimensions as a fuel element. The source holder is manufactured as upper and lower (threaded) sections. The top of the lower section is at the horizontal centerline of the core. A soft aluminum ring provides sealing against water leakage into the cavity. [redacted] The source holder may be positioned in any one of the fuel positions defined by the upper and lower grid plates. The upper end fixture of the source holder is similar to that of the fuel element; the source holder can be installed or removed with the fuel handling tool. In addition, the upper end fixture has a small hole through which one end of a stainless steel wire may be inserted to facilitate handling operation from the top of the tank.

4.2.5 Core support structure

The core support structure includes (A) a platform supporting the reflector and core structure, and (B) a "safety plate" that prevents the control rods in a failure mode from falling out of the core.

A. Core Support Platform

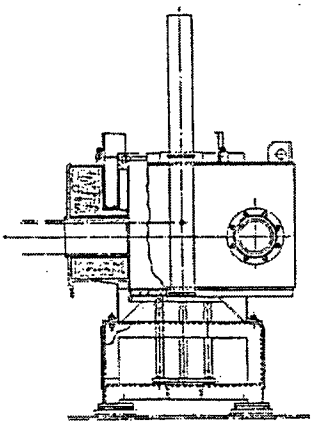
The reflector assembly rests on a platform (Fig. 4.14) constructed of structural angle 6061-T5 aluminum with a 3 in. x 3 in. x ⅜ in. (7.62 cm x 7.62 cm x 0.953 cm) web (Fig. 4.14a/b/c). Aluminum 6061-T651 plate is used for safety plate support pads (¾ in., 1.905 cm), cross braces (⅜ in., 0.953 cm.), and platform support pads (½ in., 1.27 cm.). Angle aluminum is inserted 9 in. (22.86 cm) from two edges to support the safety plate, with angle bracing on the edges perpendicular to the safety plate supports.



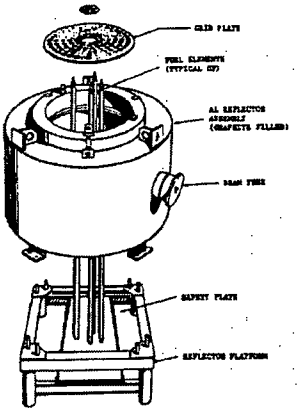
Core Support Top View | Core Support Side View | Core Support Side View

Figure 4.14, Core Support Views

The platform top surface is 30 ¼ in. X 30 ¼ in., with the top surface 16 ¼ in. above the pool floor. Surfaces are matte finished for uniform appearance with shot cleaning and peening using glass beads (MIL-STD -852).



Core and Support Structure Assembly



Core and Support Assembly Isometric

Figure 4.15, Core and Support Structure Views

B. Safety plate

The safety plate (Fig 4.16) limits the distance that a control rod can fall to less than 17.44 in. (44.30 cm) below the top surface of the lower grid plate. The safety plate is an aluminum plate ½ in. (1.27 cm.) thick, 12 in. (30.48 cm) X 13.5 in. (34.29 cm), anodized to resist wear and corrosion (MIL-A-8625 TYPE II, with exception that abrasive and corrosive testing not required).

The top of the safety plate is 7.75 in. (3.05 cm.) above the bottom of the pool. As previously described, the bottom grid plate has a set of through-penetrations for optional placement of control rods. A special adapter is required to support fuel elements when these locations are used for fuel. The adapters have a central alignment pin that fits within holes in the safety plate, and an offset keeper-pin that prevents the adapter from rotating around the central pin.

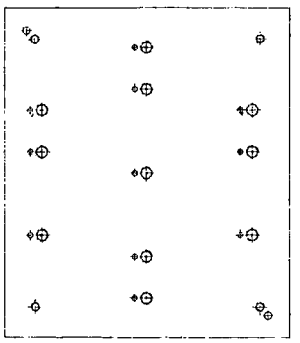


Figure 4.16, Safety Plate

4.3 Reactor Pool

The reactor pool is a 26 foot, 11.5 in. (8.2169 m) tall tank formed by the union of two half-

cylinders with a radius of 6 ½ feet separated by 6 ½ feet (1.9812 m). The bottom of the pool is at the reactor bay floor level. The reactor core is centered on one of the half-cylinders. Normal pool level is 8.179 (26.57 ft.) meters above the bottom of the pool, with a minimum level of 6.5 m (21.35 ft.) required for operations. Volume of water in pool (excluding the reflector, beam tubes and core-metal) is 40.57 m^3 and 32.50 m^3 for the nominal and minimum-required levels. Table 4.11 summarizes reactor coolant system design.

Table 4.11, Reactor Coolant System Design Summary

Reactor Tank	Material	Aluminum plate (6061)
	Thickness	¼ in. (0.635 cm)
	Volume (maximum)	11000 gal (41.64 m^3)
Coolant Lines	Pipes	Aluminum 6061
	Valves	Iron-Plastic Liner, 316 SS Ball and Stem
	Fittings	Aluminum (Victaulic)
Coolant Pump	Type	Centrifugal
	Material	Stainless Steel
	Capacity	250 gpm (15.8 lps)
Heat Exchanger	Type	Shell & Tube
	Materials (shell)	Carbon steel
	Materials (tubes)	304 stainless steel
	Heat Duty	[redacted]
	Flow Rate (shell)	[redacted]
	Flow Rate (tubes)	[redacted]
Typical Heat Exchanger Operating Parameters	Tube Inlet	[redacted]
	Tube Outlet	[redacted]
	Shell Inlet	[redacted]
	Shell Outlet	[redacted]

The pool (Figs. 4.17a/b/c) is fabricated from sheets of 0.25 in. (0.635 cm) 6061 aluminum in 4 vertical sections welded to a ½ in. thick aluminum plate. Full penetration inspection was performed on tank components during fabrication, including 20% of the vertical seam welds, 100% on the bottom welds (internal and external to the pool volume), and 100% on the beam port weld external to the pool volume. A single floor centerline seam weld was used; a sealed channel was welded under the seam and instrumented through a ¼ in. NPT threaded connection to perform a leak test during fabrication. A 2 in. X 2 in. X ¼ in. (square) aluminum channel was rolled and welded to the upper edge of the tank.

Beam port penetrations are fabricated around the core to allow extraction of radiation beams to support experiments. The beam ports are centered 90.2 cm (35 in.) above the pool floor, 7.2 cm (2.83 in.) below the core centerline. The section of the beam ports that are an integral part of the pool include an in-pool section, interface with the pool wall, and a section

extending outside of the pool.

In pool sections are 6 in. (15.4 cm) in diameter, with a 0.635 cm (0.25 in.) wall thickness. The in pool section for BP 1 and 5 is 6 in. (15 cm), while the remaining in-pool beam port sections are much longer. Supports (2 in. X 2 in. X ¼ in. aluminum angle bracket) are welded at the bottom of the pool and directly onto BP 2, 3, and 4 because of the extended lengths. BP 2 and 4 terminate at the outer surface of the reflector, while BP 3 extends into the reflector, terminating at the inner shroud. BP 2 terminates in an oblique cut, and extends approximately 43 cm (16.94 in.) into the pool with the support 12.7 cm (5 in.) from the in-core end. BP 3 extends 73 cm (28.75 in.) into the pool with the support 37.62 cm (14.8125 in.) from the in-pool end. BP 4 extends 43 cm into the pool (16.94 in.) with the support 7.62 cm (3 in.) from the in pool end. Beam port 1 and 5 are aligned in a single beam line. A flight tube inserted into BP 1/5 extends through the reflector near the core shroud; BP 1 and 5 are equipped with a bellows to seal a neutron flight-tube. Beam ports 2, 3, and 4 are sealed at the in-pool end. BP 2 is tangential to the core shroud, offset 34.29 cm (13 ½ in.) from core center rotated 30° with respect to BP 3. Beam port 3 is 90° with respect to BP 1/5, aligned to the center of the core. Alignment of BP 4 is through the core center, rotated 60° from BP 3.

The beam port interface with the pool wall includes a reinforcing flange on the inner pool wall. The flange is 3/8 in. thick, 11 in. in diameter. The flange is welded on the outer diameter to the pool wall and on the inner diameter to the beam port tube.

The beam ports extend approximately 15.24 cm (6 in.) outside of the area define by the pool walls. A stainless steel (304) ring is machined for a slip fit over the extension. The ring is welded to 6 5/8 in. diameter stainless steel pipe (SST 304W/ASTM 312) extending the flight tube for the beam port into the biological shielding.

The floor of the pool has four welded pads for the core and support structure. As noted, the in-pool beam port supports are welded to the pool floor.

Detection of potential pool leakage could occur in a number of ways.

1. Pool water level is maintained approximately 8.1 m above the pool floor, and monitored with an alarm on the control room console. A sudden decrease in pool water will create a condition that alerts the reactor operator at the controls.

2. Losses to evaporation are compensated by makeup water. Makeup water usage is closely monitored, and changes in makeup requirements or increases in makeup water that do not correspond to power level operation are a primary pool-leak indicator.

3. French drains around the reactor pool shielding foundation are collected in a sump, and sampled periodically. Increases in radiation levels from the sump (particularly tritium) could indicate pool leakage.

4.4 Biological Shield

Pool water system and shield structure (Fig. 4.18) design combine to control the effective radiation levels from the operation of the reactor. One goal of the design is a radiological exposure constraint of 1 mrem/hour for accessible areas of the pool and shield system. Dose levels assume a full power operation level of 1.500 megawatts (thermal). Radiation doses above the pool and at specific penetrations into or through the shield may exceed the design goal. The reference case design is a solid structure without any system penetrations. [redacted] Tank assembly is by shop fabrication. A protective layer of epoxy paint and bitumen coal tar pitch with paper provides a barrier between the aluminum pool tank and the reactor shield concrete.

A [redacted] thick foundation pad supports the reactor pool and shield structure. Standard weight concrete, [redacted] comprises the foundation pad. High density concrete, [redacted] Five beam tubes at the level of the reactor provide experimental access to reactor neutron and gamma radiations. Two of the tubes combine to penetrate the complete reactor pool and shield structure from one side to the other side. Special design features of the beam tubes are beam plugs, sliding lead shutters, bolted cover plates, and gasket seal for protection against reactor radiation and coolant leakage when the tubes are not in use. Beam port details are discussed in Chapter 10. A summary of significant component elevations and control functions is provided in Table 4.12.

Table 4.12, Significant Shielding and Pool Levels

Parameter of Interest	Level (meters)	Notes
CONCRETE PAD	[redacted]	[redacted]
FLOOR	[redacted]	[redacted]
SAFETY PLATE	[redacted]	[redacted]
GRID PLATE	[redacted]	[redacted]
CORE BOTTOM	[redacted]	[redacted]
BEAM PORT CL	[redacted]	[redacted]
CORE CL	[redacted]	
CORE TOP	[redacted]	[redacted]
GRID PLATE	[redacted]	[redacted]
MAIN LOWER SHIELDING	[redacted]	[redacted]
TRANSITIONAL CONCRETE STEP	[redacted]	[redacted]
SHIFT TO HIGH DENSITY CONCRETE	[redacted]	[redacted]
MIN CORE LEVEL (TS)	[redacted]	[redacted]
VACUUM BREAKERS	[redacted]	
LOW POOL LEVEL SCRAM	[redacted]	
LOW POOL LEVEL	[redacted]	[redacted]
LOW POOL LEVEL ALARM	[redacted]	[redacted]
NORMAL POOL LEVEL	[redacted]	[redacted]
HIGH POOL LEVEL	[redacted]	[redacted]
HIGH POOOL LEVEL ALARM	[redacted]	[redacted]
TOP OF TOP LEVEL	[redacted]	

4.5 Nuclear Design

The characteristics and operating parameters of this reactor have been calculated and extrapolated using experience and data obtained from existing TRIGA reactors as bench marks in evaluating the calculated data. There are several TRIGA systems with essentially the same

core and reflector relationship as this TRIGA so the values presented here are felt to be accurate to within 5% but, of course, are influenced by specific core configuration details as well as operational details. An operational core of [redacted], 3 fuel followed control rods, and one air followed control rod is to be arranged in 5 rings with a central, water filled hole. Dimension of the active fueled core, approximated as cylinder, 15 in. (0.381 m). The cylinder radius is calculated as the average radius of a hexagonal fuel array with [redacted]

4.5.1 Normal Operating Conditions

Reactivity worth of core components is generally determined by calculation and/or comparison of the reactivity worth associated with the difference in the reactivity worth of control rod positions in the critical condition, component installed and component removed. The 1992 UT SAR provided data indicating estimated worth of the control rods (Table 4.16). Control rod worth is influenced by core the experiment configuration, with significant impact from the large in core irradiation sites. Table 4.13 provides the worth of the control rods in the current configuration (3 element facility in E11, F13, and F14). Change in core configuration require validation that control rod worth is not affected by the experiment facility , or re-establishment of the control rod worth followed by verification that the limiting conditions for operation are met.

Table 4.13 Control Rod Worth

Control Rod	Reference			Current (2011)	
	Position	Worth		Position	Worth
Transient rod	C ring	2.1% Δk/k	$3.00	C-1	$3.10
Regulating rod	C ring	2.6% Δk/k	$3.71	C-7	$2.82
Shim 1	D ring	2.0% Δk/k	$2.86	D-14	$2.52
Shim 2	D ring	2.0% Δk/k	$2.86	D-6	$3.07

4.5.2 Nominal Reactivity Worth Values

Reactivity values for core components based on calculations and observations are provided in Table 4.14, with Technical Specifications values in bold face type. Current values are based on measurements; nominal values are calculations frOm indicated sources.

Table 4.14, Reactivity Values

$	Parameter	TS LIMIT	CURRENT VALUE	NOMINAL VALUE
[redacted]				

[10] Reactor Reference Data Notebook, Safety Analysis report Table 4-5; SAR Table 4-6 indicates CT Fuel $0.90, CT Void -$0.15, PNT Void -$0.10, RSR void -0.20

Table 4.14, Reactivity Values

$	Parameter	TS LIMIT	CURRENT VALUE	NOMINAL VALUE
[redacted]				

4.5.3 Reactor Core Physics

The performance of the TRIGA core was evaluated by General Atomics, as described below. The basic parameter which allows the TRIGA reactor system to operate safely with large step insertions of reactivity is the prompt negative temperature coefficient (Fig. 4.19) associated with the TRIGA fuel and core design. This temperature coefficient allows a greater freedom in steady-state operation as the effect of incidental reactivity changes occurring from the experimental devices in the core is greatly reduced.

[11] 3-Element Experiment Authorization
[12] Significant deviation from values in 3-Element Experiment Authorization (cf. E-Ring ~$0.50 & D-Ring $0.95)

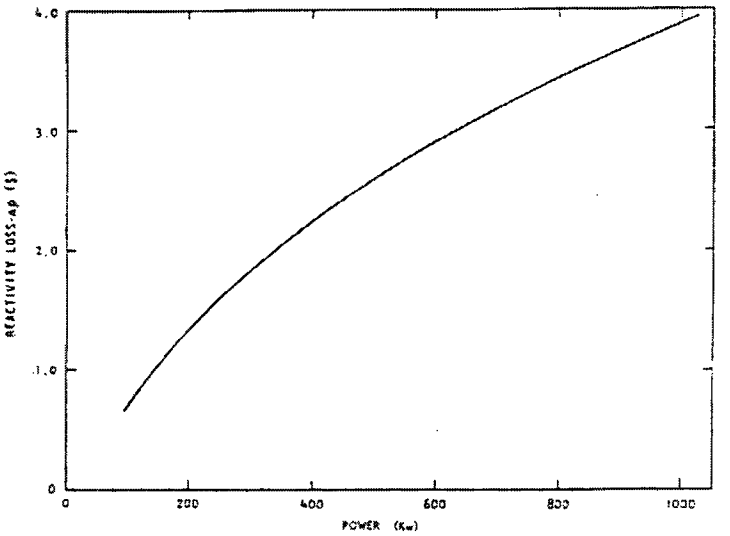


Figure 4.19, Reactivity Loss with Power

A. Reference Calculations

A reference calculation of neutron flux distribution across the core was performed by General Atomics[13]. The calculations were accomplished on an IBM-7090 using General Atomics (diffusion theory based) codes GAMBLE and GAZE, and GAM-I. GAM-I is a fast neutron (using P1 treatment), temperature dependent (using methods developed by Nordhiem) cross section calculations for neutrons above 1 eV. GATHER-I was used to calculated cross sections below 1 eV. Homogenization was accomplished by the transport theory code DSN for group-dependent disadvantage factors (a second homogenization was accomplished for inhomogeneities in cells with control rods). No attempt was made to account for spatial variations in core temperatures. Basic core data for the calculations is provided in Table 4.15, with selected nuclear properties in Table 4.16. The model varies from the UT TRIGA reactor in specification of control rods, with one poison and three aluminum followers, where the UT TRIGA uses one aluminum and three poison followers; since this effects only the homogenization for two discrete cells, the results for core wide parameters is valid. UT TRIGA data is provided in Table 4.17.

Table 4.15, GA-4361 Calculation Model

	Radius		Area	Volume	Volume Fraction
Cell Region	in.	cm	cm^2	cm^3	
U-$ZrH_{1.7}$	0.7175	1.822	10.429	397.34	0.6308
SS Cladding	0.7375	1.873	0.592	22.56	0.0358
Water	0.9032	2.294	5.511	209.98	0.3334
TOTAL	na	na	16.532	629.88	1.0000

[13] *GA-4361, Calculated Fluxes and Cross Sections for the TRIGA Reactors*, G. B. West. August 1963

Table 4.16, Selected TRIGA II Nuclear Properties

Number of cells	80	91
Fuel Temperature	23°C	200°C
1 eV to 10 MeV		
Σ_a	0.00660	0.00675
Σ_f	0.00135	0.00135
Flux/watt	2.46×10^7	2.21×10^7
p[1]	0.9405	0.9481
0 to 1 eV		
Σ_a	0.0873	0.0794
Σ_f	0.0526	0.0472
Flux/watt	1.11×10^7	1.08×10^7
% of fissions	94.6	94.5
v_{ave} cm/s	2.73×10^5	2.94×10^5
E_{ave} eV	0.0391	0.0455

NOTE 1: Resonance escape probability

Table 4.17, UT TRIGA Data

Core Configuration	
Ref Cold Clean Critical Loading	[redacted]
Ref Operational Loading	[redacted]
Actual Initial Criticality	
Fuel element pitch	[redacted]
Coolant volume to cell ratio	[redacted]
Fuel Elements	
Cladding	SS 304
Fuel matrix	U-$ZrH_{1.6}$
Fuel Mass	[redacted]
Uranium fraction	[redacted]
Enrichment	[redacted]
Nuclear Parameters	
Prompt neutron lifetime (ℓ)	41 μs
Effective delayed neutron fraction (β)	0.0070
Prompt negative temperature coefficient (α)	1x10-4 Δk/k°C

B. Prompt Negative Temperature Coefficient

GA Technologies, the designer of the reactor, has developed techniques to calculate the temperature coefficient accurately and therefore predict the transient behavior of the reactor. This temperature coefficient arises primarily from a change in the disadvantage 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 and thus fuel and solid moderator temperatures rise simultaneously. A quantitative calculation of the

temperature coefficient requires knowledge of the energy dependent distribution 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. Since the average chord length in the fuel element is comparable to a mean free path, the probability of escape from the fuel element before capture is increased. In the water the neutrons are rapidly thermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel 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 the core which decreases the ratio of absorptions in the fuel to total cell absorptions as the fuel element temperature is increased. This brings about a shift in the core neutron balance, giving 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 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.

Calculation work on the temperature coefficient made use of a group of codes developed by GA Technologies: GGC-3[14], GAZE-2[15], and GAMBLE-5[16], as well as DTF-IV[17], 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 on tape for all commonly used isotopes, are averaged over a space independent flux derived by solution of the 81 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

[14] General Atomics Report GA-7157, "Users and Programmer Manual for the GGC-3 Multigroup Cross Section Code," General Dynamics, General Atomic Division (1967)
[15] General Atomics Report GA-3152 "GAZE-2: A One-Dimensional, Multigroup, Neutron Diffusion Theory Code for the IBM-7090," Lenihan, S. R., General Dynamics, General Atomic Division (1962)
[16] General Atomics Report GA-818, "GAMBLE-5 – A program for the Solution for the Multigroup Neutron-Diffusion Equations in Two Dimensions, with Arbitrary Group Scattering, for the UNIVAC-1108 Computer," Dorsey, J. P. and R. Foreloch, General Dynamics, General Atomic Division (1967)
[17] USAEC ReportLA-3373, DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scatterings, Los Alamos Scientific Laboratory, new Mexico (1965)

resonance integral work of Adler, Hinman, and Nordhein to generate cross sections for resonance materials which are properly averaged over the region spectrum. 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. The bound hydrogen kernels used for hydrogen in the water were generated by the THERMIDOR code[18] using thermalization work of Nelkin[19]. Early thermalization work by McReynolds et al[20] on zirconium hydride has been greatly extended at GA Technologies[21], and work by Parks resulted in the SIJMMIT t25l 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 Technologies linear accelerator as shown in section 4.2.1 (A).

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 €V, and whose width is adjusted to fit the cross section data of Woods et al. 1281. 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 neutron 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.

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[18] "THERIMIDOR- A FORTRAN II Code for Calculating the Nelkin Scattering Kernel for Bound Hydrogen (A modification of Robespierre),"Gulf General Atomic, Inc. (unpublished data) Brown, H. D., Jr.

[19] "Scattering of Slow Neutrons by Water," Phys. Rev., 11, 741-746, Nelkin, M. S. (1960)

[20] "Neutron Thermalization by Chemically-Bound Hydrogen and Carbon," Proc. 2nd Intl. Conf. Peaceful Used at Energy (A/Conf. 15/F/1540), Geneva, IAEA (1958)

[21] General Atomics Report GA-4490 Neutron Interactions in Zirconium Hydride, Whittenmore, W. L., General Dynamics, General Atomic Division (1964)

Calculations of the temperature coefficient were done in the following steps:

a. 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.

b. A value for k_∞ was computed for each fuel element temperature by transport cell calculations, using the P1 approximation. Comparisons have shown S_4 and S_8 results to be nearly identical. Group dependent disadvantage factors defined as Φ_g^r / Φ_g^c (region : cell) were calculated for each cell region (fuel, clad, and water).

c. 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.

d. The cross sections for an equivalent homogenized core were used in a full reactor calculation to determine the contribution to the temperature coefficient due to the increased leakage of thermal neutrons into the reflector with increasing hydride temperature. This calculation 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 the temperature coefficient for a standard TRIGA core comes from the temperature-dependent disadvantage factor or "cell effect", and ~20% each from Doppler broadening of the ^{238}U resonances and temperature dependent leakage from the core. This produces a temperature coefficient of ~ -0.01%/°C, which is rather constant with temperature.

Because of the prompt negative temperature coefficient a significant amount of reactivity is needed to overcome temperature and allow the reactor to operate at the higher power levels in steady-state operation. Figure 4.19 shows the relationship of reactor power level and associated reactivity loss to achieve a given power level.

4.5.4 Operating Limits

The core-wide operating limits associated with nuclear design are based on spatial distribution of neutron flux that determines the local peak power production. Therefore (A) the peaking factors are required to determine (B) the limiting core configuration. Core reactivity limits (C) are established by Technical Specifications and used as a basis for evaluating performance and capabilities.

A. Core Peaking Factors

The core is generally modeled as a right cylinder. Neutron flux varies along the axis of a cylindrical reactor using periodic Bessel functions. Neutron flux varies radially in a cylindrical reactor using period sine functions. The product of these two functions provides a relationship between average core power and the maximum power at a location within the core. Neutron flux and fission rate also varies significantly across the radius of a TRIGA fuel element; the complexities of the system do not lend themselves to reasonable analytic description.

Core Radial Peaking Factor. Classically, the radial hot-channel factor for a cylindrical reactor (using R as the physical radius and R_e as the physical radius and the extrapolation distance) is given[22] by:

$$F_{R_e}^{N} = \frac{1.202 * \left(R / R_e \right)}{J_1 \left[2.4048 * \left(R / R_e \right) \right]}$$

However, TRIGA fuel elements are on the order of a mean free path of thermal neutrons, and there is a significant change in thermal neutron flux across a fuel element. Calculated thermal neutron flux data[23] indicates that the ratio of peak to average neutron flux (peaking factor) for TRIGA cores under a range of conditions (temperature, fuel type, water and graphite reflection) has a small range of 1.36 to 1.40. Therefore, actual power produced in the most limiting actual case is 14% less than power calculated using the assumption.

Core Axial Peaking Factor. The axial distribution of power in the hottest fuel element is sinusoidal, with the peak power a factor of $\pi/2$ times the average, and heat conduction radial only. The axial factor for power produced within a fuel element is given by:

$$g(z) = 1.514 * \cos\left(\frac{\pi}{2} * \frac{z}{2 * \ell + \ell_{ext}} \right),$$

in which $\ell = L/2$ and ℓ_{ext} is the extrapolation length in graphite, namely, 0.0275 m. The value used to calculate power in the limiting location within the fuel element is therefore 4% higher than power calculated with the actual peaking factor. Actual power produced in the most limiting actual case is 4% less than power calculated using the assumption; therefore calculated temperatures will bound actual temperatures.

Core Local Peaking Factor. The location on the fuel rod producing the most thermal power with thermal power distributed over N fuel rods is therefore:

[22] Elements of Nuclear Reactor Design, 2nd Edition (1983), J. Weisman, Section 6.3
[23] GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors (8/14/1963), G. B. West

$$q''_{max} = \frac{P}{N \cdot D_0 \cdot L}$$

B. Power distribution within a Fuel Element

The radial and axial distribution of the power within a fuel element is given by

$$q'''(r,z) = q'''_{avg} f(r) g(z)$$

in which r is measured from the vertical axis of the fuel element and z is measured along the axis, from the center of the fuel element. The axial peaking factor follows from the previous assumption of the core axial peaking factor, but (since there is a significant flux depression across a TRIGA fuel element) distribution of power produced across the radius of the fuel the radial peaking factor requires a different approach than the previous radial peaking factor for the core. The radial factor within a fuel element is given by:

$$f(r) = \frac{a + cr + er^2}{1 + br + dr^2}$$

in which the parameters of the rational polynomial approximation are derived from flux-depression calculations for the TRIGA fuel[24]. Values for the coefficients are: a = 0.82446, b = -0.26315, c = -0.21869, d = -0.01726, and e = +0.04679. The fit is illustrated in Figure 4.20.

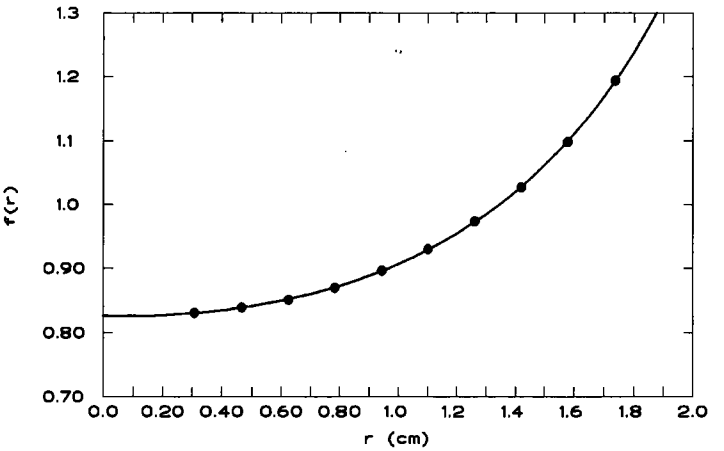

Figure 4.20, Radial Variation of Power Within a TRIGA Fuel Rod.
(Data Points from Monte Carlo Calculations [Ahrens 1999a])

C. Power per rod

The Bernath correlation[25] calculates critical heat flux as:

[24] Report KSUNE -Investigation of the Radial Variation of the Fission-Heat Source in a TRIGA Mark III Fuel Element Using MCNP, Ahrens, C., Department of Mechanical and Nuclear Engineering, Kansas State University, Manhattan, Kansas (1999)

$$\left[\frac{Q}{A}\right]_{BO} = h_{BO} \cdot (T_{WO} - T_B)$$

Where the convection heat transfer coefficient for "burnout" condition is determined by:

$$h_{BO} = 10890 \cdot \left(\frac{D_e}{D_e + D_i}\right) + SLOPE \cdot V$$

With two possible values for the "SLOPE" term:

(1) IF $D_e \leq 0.1$ ft.,

$$SLOPE = \frac{48}{D_e^{0.6}}$$

(2) IF $D_e > 0.1$ ft.:

$$SLOPE = 90 + \frac{10}{D_e}$$

And the burnout wall temperature term is calculated:

$$T_{W,BO} = 57 \cdot \ln(P) - 54 \cdot \frac{P}{P + 15} - \frac{V}{4}$$

The CHF heat flux in is p.c.u./hr-ft^2, the heat transfer coefficient corresponding to the CHF in p.c.u./hr-ft^2-C, is the wall temperature at which CHF occurs in °C, T_b is the local bulk coolant temperature in °C, D_e hydraulic diameter of the coolant passage in feet, D_i is the diameter of the heater surface (heated perimeter divided by π) in feet, P is the pressure in psia, and V is the velocity of the coolant in ft/s. Substituting equivalent terms into the CHF equation results in:

$$\left[\frac{Q}{A}\right]_{BO} = \left(10890 \cdot \left(\frac{D_e}{D_e + D_i}\right) + \frac{48}{D_e^{0.6}} \cdot V\right) \cdot \left(57 \cdot \ln(P) - 54 \cdot \frac{P}{P + 15} - \frac{V}{4} - T_B\right)$$

Where *A* is the flow area and *WP* the wetted perimeter, hydraulic diameter is calculated:

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[25] ANL/RERTE/TM-07-01, Fundamental Approach to TRIGA Steady state Thermal-Hydraulic CHF Analysis

$$D_e = \frac{4 \cdot A}{WP}$$

(1) Wetted perimeter:

$$WP = \frac{1}{2} \cdot \pi \cdot D_{fuel}$$

(2) Flow area:

$$A = PITCH^2 \cdot \frac{\sqrt{3}}{4} - \frac{1}{2} \cdot \pi \cdot \left(\frac{D_{fuel}}{2} \right)^2$$

TRACE calculations completed as described in section 4.6 calculation of thermal hydraulic parameters that are used to calculate critical heat flux using the Bernath correlation (and the ratio of the heat flux to the critical heat flux, CHFR). TRACE calculates CHFR using the Biasi correlation. The results of calculations using heat flux and temperature data for 49°C water at 6.5 m level is provided in Table 4.18. The minimum CHFR versus power level is provided in Figure 4.21. As illustrated, the CHFR values agree well and remain much greater than 2 at power levels up to 22.5 kW per unit cell.

Table 4.18, Critical Heat Flux ratio, Bernath Correlation

kW	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1.5	106.2	93.5	83.3	74.9	68.6	66.1	63.7	61.5	63.5	65.7	67.2	72.5	78.9	86.8	96.8
3.0	61.0	53.4	47.2	42.1	38.5	37.0	35.7	34.4	35.5	36.6	37.2	39.7	42.9	46.8	51.9
4.5	44.3	38.6	34.0	30.2	27.6	26.4	25.5	24.4	25.3	26.0	26.2	27.8	29.7	32.2	35.5
6.0	35.4	30.9	26.9	23.6	21.6	20.8	20.0	19.2	19.7	20.3	20.3	21.3	23.9	24.6	26.7
7.5	29.9	25.8	22.5	19.7	17.9	17.2	16.5	15.8	16.3	16.8	16.6	17.4	19.2	19.6	21.3
9.0	26.0	22.3	19.4	16.9	15.3	14.7	14.1	13.5	13.9	14.2	14.0	14.6	16.0	16.2	17.6
10.5	23.1	19.8	17.1	14.8	13.4	12.9	12.4	11.8	12.1	12.4	12.1	12.5	13.6	13.8	14.8
12.0	20.8	17.8	15.3	13.2	12.0	11.5	11.0	10.5	10.8	11.0	10.7	10.9	11.8	11.8	12.6
13.5	19.1	16.3	13.9	11.9	10.8	10.3	9.9	9.4	9.7	9.9	9.5	9.7	10.4	10.3	10.9
15.0	17.6	15.0	12.8	10.9	9.9	9.4	9.0	8.6	8.8	9.0	8.6	8.6	9.2	9.1	9.5
16.5	16.4	13.9	11.8	10.0	9.1	8.6	8.2	7.8	8.0	8.2	7.8	7.8	7.9	8.0	8.3
18.0	15.4	13.0	11.0	9.3	8.4	8.0	7.6	7.2	7.4	7.5	7.1	7.0	7.0	7.1	7.3
19.5	14.5	12.2	10.3	8.6	7.8	7.4	7.1	6.7	6.8	7.0	6.5	6.4	6.4	6.4	6.5
21.0	14.0	11.8	9.9	8.3	7.5	7.2	6.8	6.5	6.6	6.7	6.3	6.2	6.1	6.1	6.1
22.5	13.7	11.5	9.7	8.1	7.3	7.0	6.6	6.3	6.4	6.6	6.1	6.0	5.9	5.9	6.0

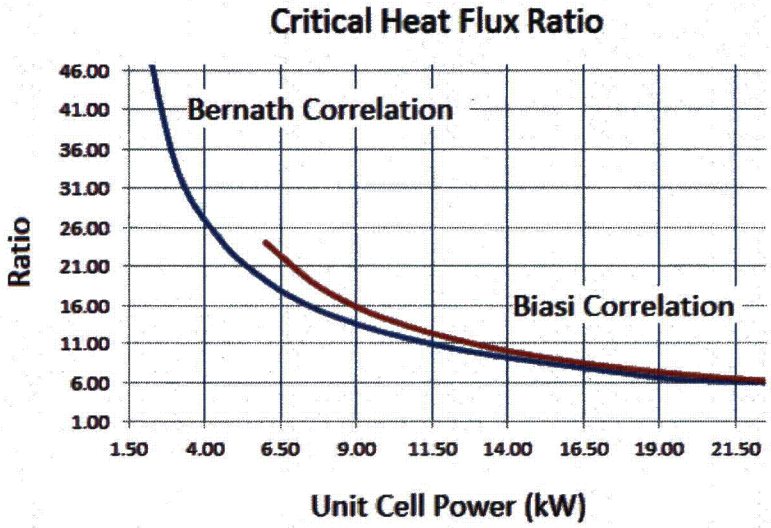

Figure 4.21, Critical Heat Flux Ratio (Bernath and Biasi Correlations)

Thermal hydraulic analysis using TRACE (section 4.6) demonstrates that a TRIGA fuel element operating at about 45 kW has a minimum critical heat flux ratio of 5.9 at a location about 86.7% of the distance of the heated length (38.1 cm) of the fuel. For a core of *N* fuel elements, the fuel element that produces the most power ($P_{PEAK,ROD}$) is related to the core average power level (P_{AVE}) by:

$$P_{PEAK,ROD} = \frac{P_{AVE}}{N} \cdot PF$$

Parametric variations including peaking factors from 1.3 to 2.0 and the number of fuel elements from 85 to 100 are provided in Table 4.19 and Fig. 4.22. With a peaking factor of 2 and 85 fuel elements, a core at 1913 kW would produce 45 kW in the element producing the highest power.

Table 4.19, Core Power, 45 kW Hot Element

Peaking Factor	85	90	100
1.3	2942	3115	3462
1.4	2732	2893	3214
1.5	2550	2700	3000
1.6	2391	2531	2813
1.7	2250	2382	2647
1.8	2125	2250	2500
1.9	2013	2132	2368
2	1913	2025	2250

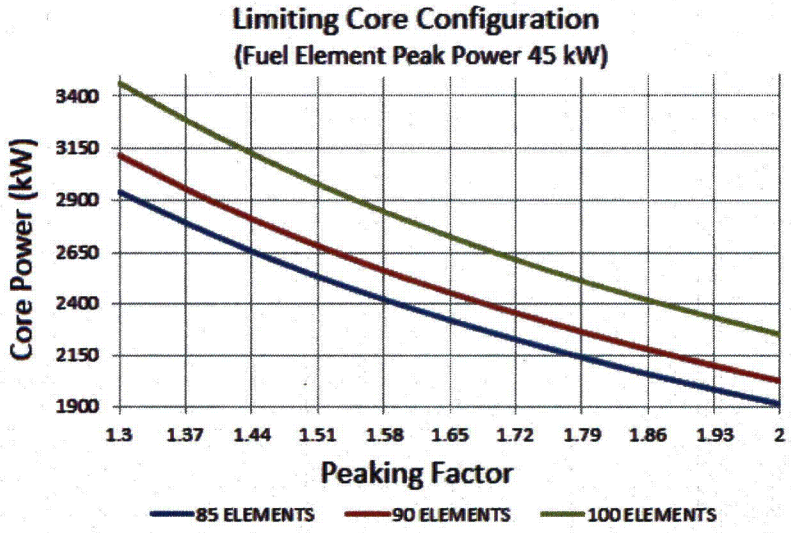

Figure 4.22, Core Power, 45 kW Hot Element

Based on the calculations, 85 fuel elements with a peaking factor of less than 2.0 provides a large margin to thermal hydraulic limits.

4.6 Core Reactivity

As noted in 4.5.1 (A), reactivity worth of material in the core is determined from differential measurements of calibrated control rod worth positions. Verification that the core configuration meets operating limits is similarly determined from the calibrated control rod positions.

As shown in Appendix 4.1, the rapid fuel temperature response from a pulsed reactivity addition terminates the power increase and causes the reactor to stabilize at a power level corresponding to the fuel temperature consistent with Fig. 4.19. Therefore limits on reactivity are based not on the peak pulse power level, but on the final equilibrium power level associated with the reactivity. A polynomial equation calculating the reactivity deficit based on Fig. 4.19 with an R^2 value of 0.99999 is:

$$\delta k = -1.75340^{-12} \cdot P^4 + 6.06670 \cdot 10^{-9} \cdot P^3 - 8.77740 \cdot 10^{-6} \cdot P^2 + 8.45380 \cdot 10^{-3} \cdot P - 0.072937$$

An approximation of the power coefficient of reactivity from 100 kW to 1 MW is therefore:

$$\frac{d}{dP}\delta k = -7.01360^{-12} \cdot P^3 + 1.82001 \cdot 10^{-8} \cdot P^2 - 1.755488 \cdot 10^{-6} \cdot P + 8.45380 \cdot 10^{-2}$$

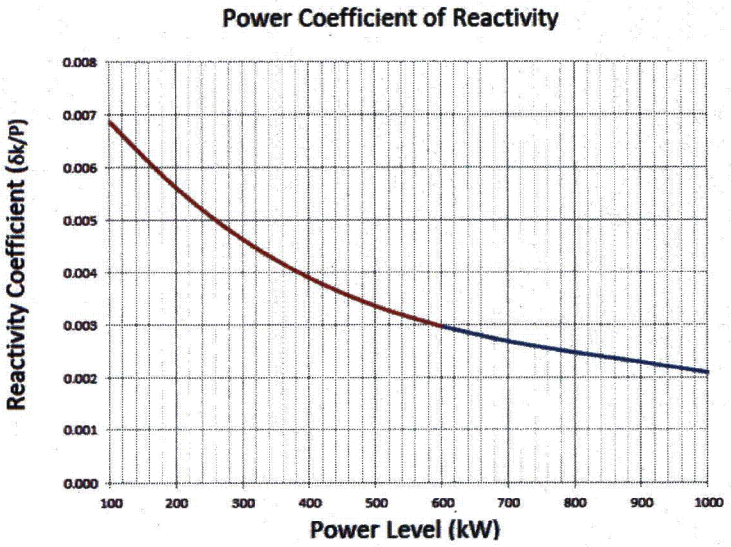

Figure 4.23, Power Coefficient of Reactivity

Therefore a pulse rod worth limited to 2.8% Δk/k ($4.00) will prevent exceeding steady state power level of 1.1 MW following a pulse using the total reactivity worth of the rod.

A limit on pulsed reactivity addition of 2.8% Δk/k ($4.00) provides an adequate safety margin.

Limiting the total experiment worth to 2.1% Δk/k ($3.00) provides additional safety margin in the event of an inadvertent pulse from the removal of all experiments.

Limiting an individual experiment to 1.75% Δk/k ($2.50) ensures that an inadvertent pulse occurring from removal of the experiment at full power operations does not exceed limits.

Limiting moveable experiments to less than 0.7% Δk/k ($1.00) will prevent an inadvertent pulsed reactivity addition leading to prompt critical condition.

Note that prediction of the power coefficient of reactivity beyond the range is not supported because the polynomial passes through a minimum above the maximum data point value. There appears to be a significant difference in response in the power level coefficient comparing low power level data to high power level data.

The operating limits on core reactivity are provided in Table 4.20.

Table 4.20, Reactivity Limits

	% Δk/k	$
Excess reactivity	4.9	7.00
Shutdown margin[1]	0.2	0.182
Moveable experiment worth	0.7	1.00
Single experiment worth	1.75	2.50
Total experiment worth	2.10	3.00

NOTE [1]: most reactive rod fully withdrawn, moveable experiments in the most positive-reactive state

Based on control rod worth values noted in Table 4.13 and calibration data from June 29, 2011, the ability of the control rods to meet the specified limits is demonstrated in Table 4.21. When significant changes to the core configuration are made, verification that the core meets requirements is accomplished including evaluation that the control rod calibration is valid or re-establishing the control rod worth calibration.

Table 4.21, Limiting Core reactivity

Control Rod	Reference		Current (2011)	
	Position	Worth	Position	Worth
Transient rod	C ring	$3.00	C-1	$3.10
Regulating rod	C ring	$3.71	C-7	$2.82
Shim 1	D ring	$2.86	D-14	$2.52
Shim 2	D ring	$2.86	D-6	$3.07
Total Rod Worth		$12.43		$11.51
Critical Reactivity		$5.43		$5.95
		LIMITING		CURRENT
Excess Reactivity		$7.00		$5.56
Shutdown Margin		-$1.72		-$2.85

4.6 Thermal Hydraulic Design

This section provides an independent assessment of the expected fuel and cladding thermal conditions associated with both steady-state and pulse-mode operations with realistic modeling of the fuel-cladding gap. Analysis is based on analysis of limiting conditions applied to a single fuel channel. The relation of the limiting channel to core average power is first defined as spatial power distribution.1

Analysis of pulsed-mode behavior is provided in Appendix 4.1, revealing that film boiling is not expected, even during or after pulsing leading to maximum adiabatic fuel temperatures. Appendix 4.1 reproduces a commonly cited analysis of TRIGA fuel and cladding temperatures associated with pulsing operations. The analysis addresses the case of a fuel element at an average temperature immediately following a pulse and estimates the cladding temperature

and surface heat flux as a function of time after the pulse. The analysis predicts that, if there is no gap resistance between cladding and fuel, film boiling can occur very shortly after a pulse, with cladding temperature reaching 470°C, but with stresses to the cladding well below the ultimate tensile strength of the stainless steel. However, through comparisons with experimental results, the analysis concludes that an effective gap resistance of 450 Btu hr^{-1} ft^{-2} $°F^{-1}$ (2550 W m^{-2} K^{-1}) is representative of standard TRIGA fuel and, with that gap resistance, film boiling is not expected.

Analysis of steady state conditions reveals maximum heat fluxes well below the critical heat flux associated with departure from nucleate boiling. The heat transfer model (A) is discussed, followed by (B) the results.

4.7.1 Heat Transfer Model

The overall heat transfer coefficient relating heat flux at the surface of the cladding to the difference between the maximum fuel (centerline) temperature and the coolant temperature can be calculated as the sum of the temperature changes through each element from the centerline of the fuel rod to the water coolant, where the subscripts for each of the ΔT's represent changes between bulk water temperature and cladding outer surface, (br_0), changes between cladding outer surface and cladding inner surface (r_0r_i), cladding inner surface and fuel outer surface – gap (g), and the fuel outer surface to centerline (r_icl):

$$T_{cl} = T_b + \Delta T_{br_0} + \Delta T_{r_0r_i} + \Delta T_g + \Delta T_{r_icl}$$

Table 4.22: Thermodynamic Values

Parameter	Symbol	Value	Units
Fuel conductivity	k_f	18	W m^{-1} K^{-1}
		14.9	W m^{-1} K^{-1} (300 K)
Clad conductivity	k_g	16.6	W m^{-1} K^{-1} (400 K)
		19.8	W m^{-1} K^{-1} (600 K)
Gap resistance	h_g	2840	W m^{-2} K^{-1}
Clad outer radius	r_0	0.018161	M
Fuel outer radius	r_i	0.018669	M
Active fuel length	L_f	0.381	M
Axial peaking factor	APF	π/2	N/A

A standard heat resistance model for this system is:

$$T_{cl} = T_b + q'' \cdot \left[\frac{1}{h} + \frac{r_0 \cdot \ln\left(r_0 / r_i\right)}{k_c} + \frac{r_0}{r_i \cdot h_g} + \frac{r_0}{2 \cdot k_f} \right]$$

in which r_o and r_i are cladding inner and outer radii, h_g is the gap conductivity, h is the convective heat transfer coefficient, and k_f is the fuel thermal conductivity. The gap conductivity of 2840 W m^{-2} K^{-1} (500 Btu h^{-1} ft^{-2} $°F^{-1}$) is taken from Appendix A.

The convective heat transfer coefficient is mode dependent and is determined in context. General Atomics reports that fuel conductivity over the range of interest has little temperature dependence. Gap resistance has been experimentally determined as indicated in Table 4.25. Temperature change across the cladding is temperature dependent, with values quoted at 300 K, 400 K and 600 K.

4.7.2 Results

TRACE was used to provide data supporting the analysis. TRACE models a unit cell which is composed of the area enclosed the pitch geometry. Since the UT TRIGA uses a hexagonal geometry, the unit cell is an equilateral triangle. Three 30° segments of a fuel element fall within the unit cell, with calculations for heat generation corresponding to ½ of the element. For example, calculations assuming 10 kW for the unit cell give indication of thermal response to an element output of 20 kW.

The TRACE heat source was modeled as a 15 in. (38.1 cm) heat flux simulating fuel, exiting stainless steel with cladding dimensions. Heat distribution was modeled as sinusoidal variation from a maximum at the center to a minimum modified at the end by extrapolation length of thermal neutrons in graphite. Data was calculated for 15 equally spaced nodes across the span of the simulated fuel element (i.e., 0.0127, 0.0381, 0.0635, 0.0889, 0.114, 0.140, 0.165, 0.191, 0.216, 0.241, 0.267, 0.292, 0.318, 0.343, and 0.368 m). TRACE calculations provide cladding temperatures directly, which is a significant portion of the standard heat resistance model, leaving only temperature differences across the gap and fuel matrix, reducing the model to:

$$T_{cl} = T_{cladding} + q'' \cdot \left(\frac{r_0}{r_i \cdot h_g} + \frac{r_0}{2 \cdot k_f} \right)$$

Considering that the terms in parenthesis are all constants, peak temperature at each node analyzed is a function of cladding temperature and heat flux. Cladding temperature and heat flux at the surface of the cladding are both calculated directly by TRACE. Coolant temperature is provided in Table 4.24, the heat fluxes at each of 15 nodes dividing the fuel section are provided in Tables 4.25a/b.

Table 4.24, Coolant Temperature for 49°C 6.5 m Pool

Unit Cell kW	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1.5	323	323	324	325	325	326	326	326	326	326	327	328	329	330	331
3.0	323	324	326	327	327	328	328	328	329	329	330	332	333	335	336

4.5	323	325	327	329	329	329	330	330	331	331	333	335	337	338	340
6.0	324	326	328	330	330	331	331	332	332	333	335	338	337	342	343
7.5	324	326	329	331	332	332	333	333	334	334	337	340	340	345	347
9.0	324	327	329	332	333	334	334	335	335	336	339	342	342	348	350
10.5	324	327	330	334	334	335	335	336	337	337	341	344	345	350	353
12.0	325	328	331	335	335	336	337	337	338	339	343	346	347	353	355
13.5	325	328	332	336	336	337	338	339	339	340	344	348	350	355	358
15.0	325	329	332	337	337	338	339	340	341	341	346	350	352	357	360
16.5	325	329	333	337	338	339	340	341	342	343	347	352	356	360	363
18.0	325	329	334	338	339	340	341	342	343	344	349	354	358	362	365
19.5	326	330	334	339	340	341	342	343	344	345	350	355	360	364	367
21.0	326	330	334	339	340	341	342	343	344	345	351	356	360	365	368
22.5	326	330	335	340	341	341	342	343	344	345	351	356	360	365	368

Table 4.25a, Heat Flux (Nodes 1-9) 49°C 6.5 Pool,

Unit Cell Kw	1	2	3	4	5	6	7	8	9
1.5	-2.72E4	-3.06E4	-3.40E4	-3.74E4	-4.08E4	-4.23E4	-4.38E4	-4.53E4	-4.38E4
3.0	-5.44E4	-6.12E4	-6.80E4	-7.48E4	-8.16E4	-8.46E4	-8.76E4	-9.06E4	-8.76E4
4.5	-8.16E4	-9.18E4	-1.02E5	-1.12E5	-1.22E5	-1.27E5	-1.31E5	-1.36E5	-1.31E5
6.0	-1.09E5	-1.22E5	-1.36E5	-1.50E5	-1.63E5	-1.69E5	-1.75E5	-1.81E5	-1.75E5
7.5	-1.36E5	-1.53E5	-1.70E5	-1.87E5	-2.04E5	-2.11E5	-2.19E5	-2.27E5	-2.19E5
9.0	-1.63E5	-1.84E5	-2.04E5	-2.24E5	-2.45E5	-2.54E5	-2.63E5	-2.72E5	-2.63E5
10.5	-1.90E5	-2.14E5	-2.38E5	-2.62E5	-2.86E5	-2.96E5	-3.06E5	-3.17E5	-3.06E5
12.0	-2.18E5	-2.45E5	-2.72E5	-2.99E5	-3.26E5	-3.38E5	-3.50E5	-3.63E5	-3.50E5
13.5	-2.45E5	-2.75E5	-3.06E5	-3.37E5	-3.67E5	-3.81E5	-3.94E5	-4.08E5	-3.94E5
15.0	-2.72E5	-3.06E5	-3.40E5	-3.74E5	-4.08E5	-4.23E5	-4.38E5	-4.53E5	-4.38E5
16.5	-2.99E5	-3.37E5	-3.74E5	-4.11E5	-4.49E5	-4.65E5	-4.82E5	-4.99E5	-4.82E5
18.0	-3.26E5	-3.67E5	-4.08E5	-4.49E5	-4.89E5	-5.07E5	-5.25E5	-5.44E5	-5.25E5

Table 4.25b, Heat Flux (Nodes 10-15) 49°C 6.5 Pool,

Unit Cell Kw	10	11	12	13	14	15
1.5	-4.23E4	-4.08E4	-3.74E4	-3.40E4	-3.06E4	-2.72E4
3.0	-8.46E4	-8.16E4	-7.48E4	-6.80E4	-6.12E4	-5.44E4
4.5	-1.27E5	-1.22E5	-1.12E5	-1.02E5	-9.18E4	-8.16E4
6.0	-1.69E5	-1.63E5	-1.50E5	-1.36E5	-1.22E5	-1.09E5
7.5	-2.11E5	-2.04E5	-1.87E5	-1.70E5	-1.53E5	-1.36E5
9.0	-2.54E5	-2.45E5	-2.24E5	-2.04E5	-1.84E5	-1.63E5
10.5	-2.96E5	-2.86E5	-2.62E5	-2.38E5	-2.14E5	-1.90E5
12.0	-3.38E5	-3.26E5	-2.99E5	-2.72E5	-2.45E5	-2.18E5

Table 4.25b, Heat Flux (Nodes 10-15) 49°C 6.5 Pool,

Unit Cell Kw	10	11	12	13	14	15
13.5	-3.81E5	-3.67E5	-3.37E5	-3.06E5	-2.75E5	-2.45E5
15.0	-4.23E5	-4.08E5	-3.74E5	-3.40E5	-3.06E5	-2.72E5
16.5	-4.65E5	-4.49E5	-4.11E5	-3.74E5	-3.37E5	-2.99E5
18.0	-5.07E5	-4.89E5	-4.49E5	-4.08E5	-3.67E5	-3.26E5
19.5	-5.50E5	-5.30E5	-4.86E5	-4.42E5	-3.98E5	-3.54E5
21.0	-5.92E5	-5.71E5	-5.23E5	-4.76E5	-4.28E5	-3.81E5
22.5	-6.34E5	-6.12E5	-5.61E5	-5.10E5	-4.59E5	-4.08E5

Table 4.26, Peak Fuel Centerline Line Temperature (K) 49°C 6.5 Pool,

Unit Cell	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1.5	363	369	374	380	385	388	390	393	391	389	387	383	379	375	370
3.0	400	410	420	431	441	445	449	454	450	446	443	434	426	418	409
4.5	434	449	463	478	491	498	504	511	504	499	494	482	470	458	446
6.0	467	485	505	524	541	549	556	562	556	550	543	528	512	496	480
7.5	500	523	545	567	584	591	598	606	598	591	584	568	551	533	514
9.0	531	559	583	603	623	631	640	648	640	631	623	603	584	565	545
10.5	563	592	616	639	661	671	680	690	680	671	661	639	616	594	571
12.0	595	622	350	673	699	710	721	733	721	710	698	673	648	623	598
13.5	622	651	680	709	737	750	762	775	762	750	737	709	680	651	623
15.0	648	680	712	743	775	789	803	816	802	789	775	743	711	680	648
16.5	673	709	743	778	813	828	843	859	843	827	813	777	743	709	673
18.0	698	737	775	813	850	866	883	901	883	866	850	813	775	737	698
19.5	725	765	806	847	888	906	924	942	924	906	887	847	806	765	724
21.0	750	793	838	881	926	945	964	984	964	945	925	881	838	793	750
22.5	775	822	869	916	963	984	1005	1026	1005	984	963	916	869	822	774

TRACE calculation provides thermal response for only a single unit cell which is ½ of a fuel rod, and neutron flux distribution causes power level to vary across the core. For instance, flow rate versus power is provided for each fuel element in Fig. 4.25. Core flow can be calculated by summing the flow rates of individual fuel rods operating at specific power levels, with peaking factors as identified above used to calculate the element power level. Unit cell temperatures are provided in Fig. 4.24 for two unit cell power levels, 10.5 and 22.5 kW.

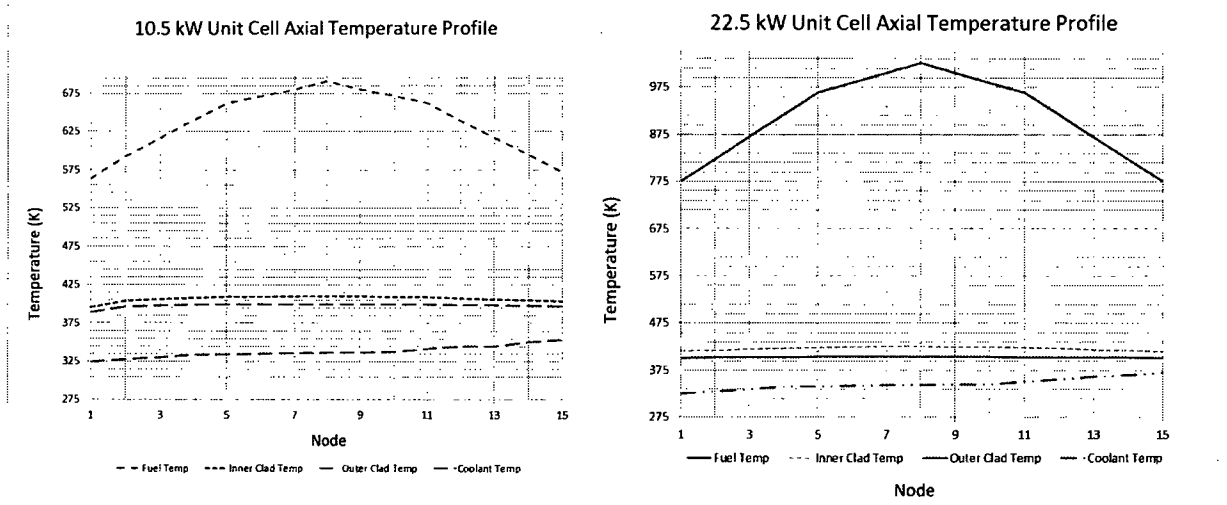

Figure 4.24, Unit Cell Temperature Distribution

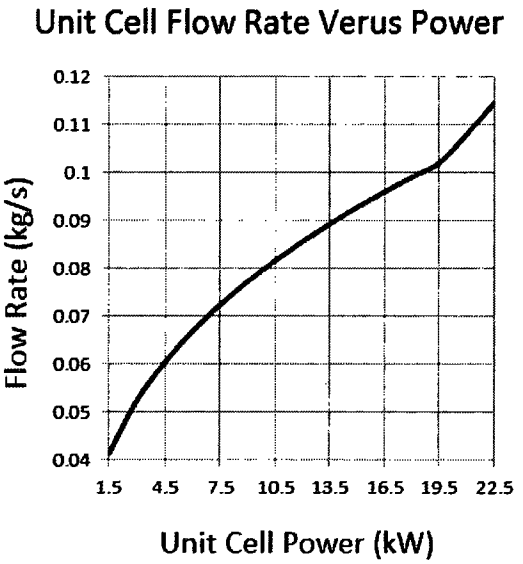

Figure 4.25, Single Rod Flow Cooling Flow Rate versus Power Level 49°C 6.5 Pool,

This discussion is reproduced from Safety Analysis Reports for the University of Texas Reactor Facility (UTA 1991) and the McClellan Nuclear Radiation Center (MNRC 1998).

The following discussion relates the element clad temperature and the maximum fuel temperature during a short time after a pulse. The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4A.1. 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 tend to approach the fuel temperature. Evidence has been obtained experimentally which shows that film boiling has occurred occasionally for some fuel elements in the Advanced TRIGA Prototype Reactor located at GA Technologies [Coffer 1964]. The consequence of this film boiling was discoloration of the clad surface.

Thermal transient calculations were made using the RAT computer code. RAT is a 2-D 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 much later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a markedly reduced threshold for film boiling. Data obtained by Johnson et al. [1961] for transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 Mbtu ft^{-2} hr^{-1} 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} hr^{-1}. 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 Figures 4A.1and 4A.2, respectively, under the assumption that the thermal resistance at the fuel-clad interface was nonexistent. A boiling heat transfer model, as shown in Figure 4A.3, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams [1954] for subcooled boiling and the work of Sparrow and Cess [1962] 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. [1963], Zuber [1959], and Rohsenow and Choi [1961] to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure. 4A.4. Fuel temperature distributions for this case are shown in Figure 4A.5 and the heat flux into the water from the clad is shown in Figure 4A.6. 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 [9] 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 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

400°C to 500°C for TRIGA Mark F fuel elements which have been operated under film boiling conditions [Coffer et al. 1965].

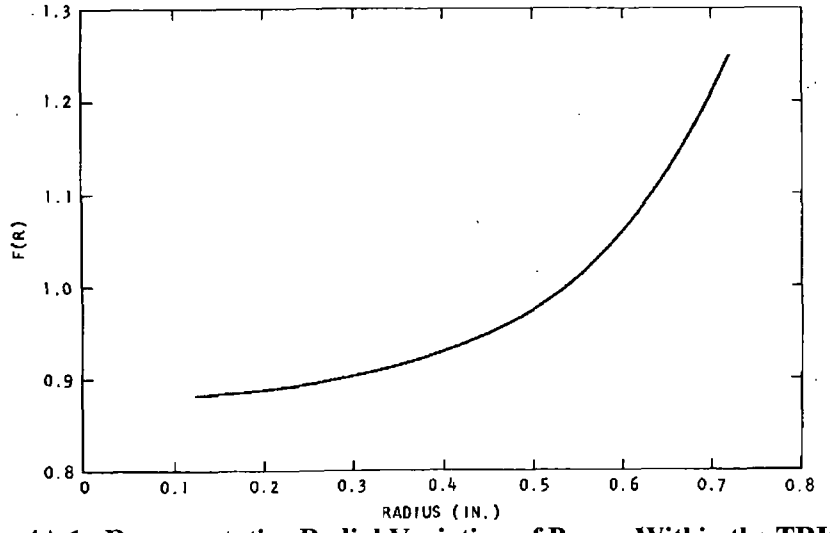

Figure 4A.1. Representative Radial Variation of Power Within the TRIGA Fuel Rod

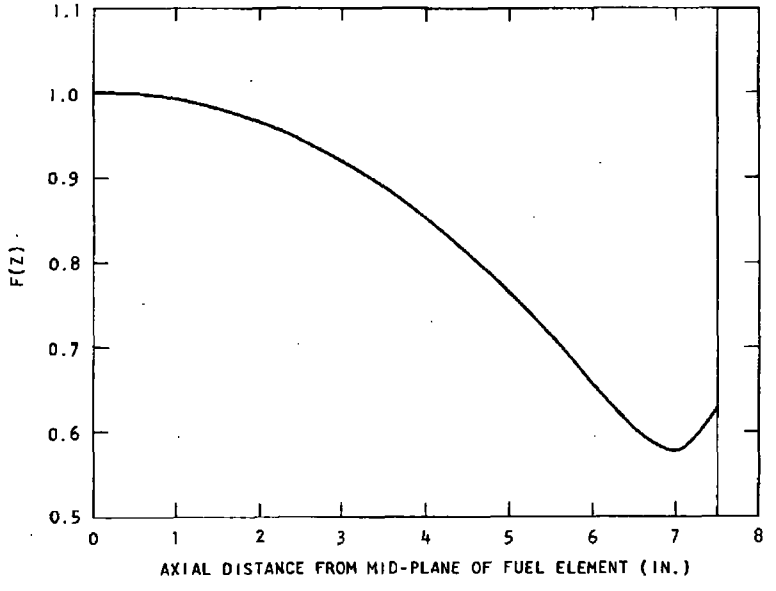

Figure 4A.2, Representative Axial Variation of Power Within the TRIGA Fuel Rod.

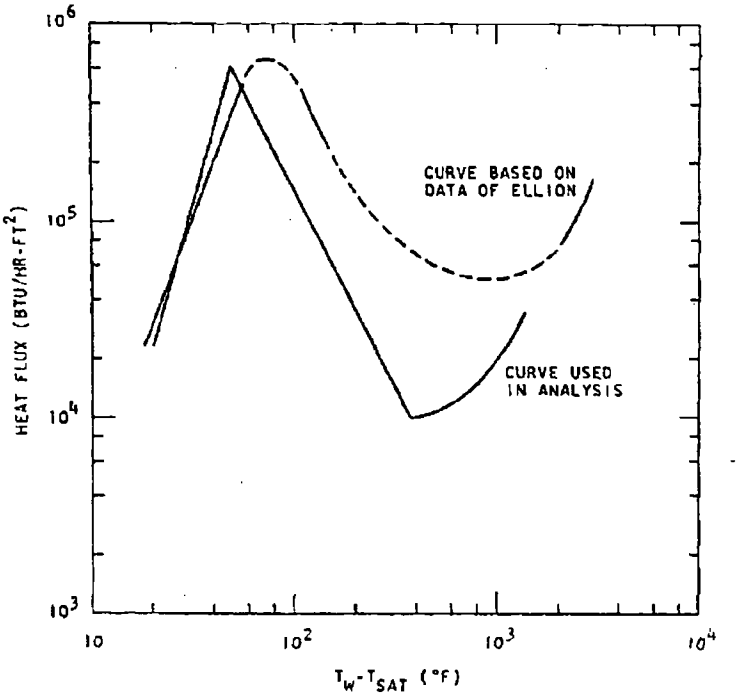

Figure 4A.3, Subcooled Boiling Heat Transfer for Water.

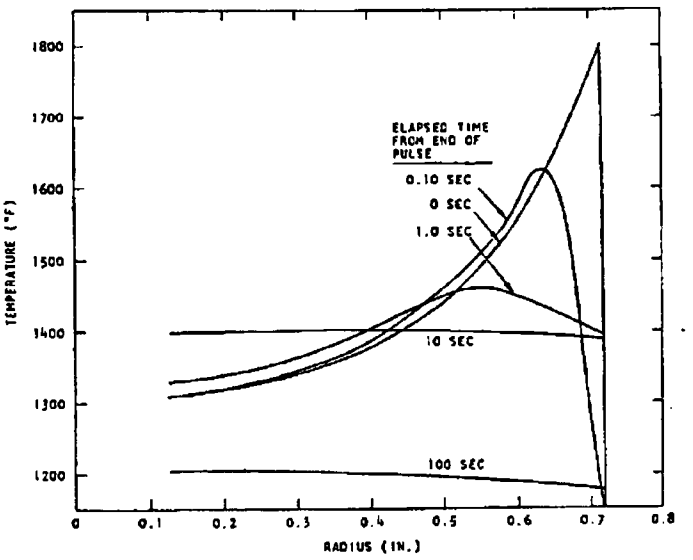

Figure 4A.4, Fuel Body Temperature at the Midplane of a Well-Bonded Fuel Element After Pulse.

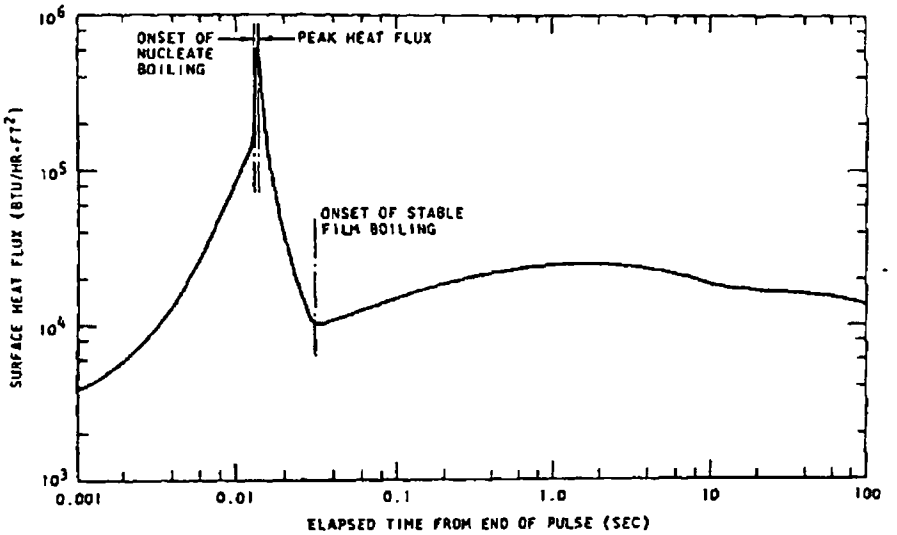

Figure 4A.5, Surface Heat Flux at the Midplane of a Well Bonded Fuel Element After a Pulse.

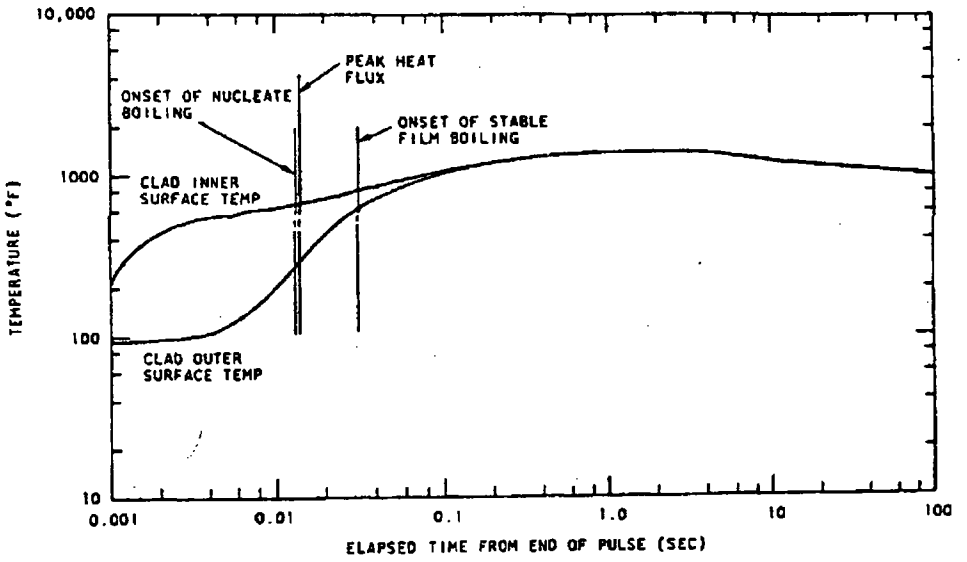

Figure 4A.6, Clad Temperature at Midpoint of Well-Bonded Fuel Element.

The preceding analysis assessing the maximum clad temperatures associated with film boiling

assumed no thermal resistance at 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 by fuel expansion. This gap exists at all temperatures below the maximum operating temperature. (See, for example, Figure 16 in the Coffer report [1965].) The gap thickness varies with fuel temperature and clad temperature so that 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 [Fenech and Rohsenow 1959, Graff 1960, Fenech and Henry 1962] which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction of this application because the basic data required for input are presently not fully known. Instead, several transient thermal computations were made using the RAT code. Each of these was made with an assumed value for 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 1000°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 ft 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 [1960] which has been reported by Spano [1964] 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 ft^{-2} hr^{-1} $°F^{-1}$. 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 is given in Figures 4A.7 through 4A.9. 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 ft^{-2} hr^{-1} $°F^{-1}$ 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 4A.10 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 4A.7 through 4A.9. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu ft^{-2} hr^{-1} $°F^{-1}$ according to the more conservative Bernath correlation at a local water temperature approaching saturation.

This analysis indicates that after operation of the reactor at steady-state power levels of 1 MW(t), or after pulsing to equivalent fuel temperatures, the heat flux through the clad is reduced and therefore reduces the likelihood of reaching a regime where there is a departure from nucleate boiling.

From the foregoing analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C.

As can be seen from Figure 4.7, the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure in the can is less than 59,000 psi, the fuel element.will not undergo loss of containment. Referring to Figure 4.8, and considering U-ZrH fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 12,600 psi. Further studies show that the hydrogen pressure that would result from a transient for which the peak fuel temperature is 1150°C would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad [Dee et al. 1966].

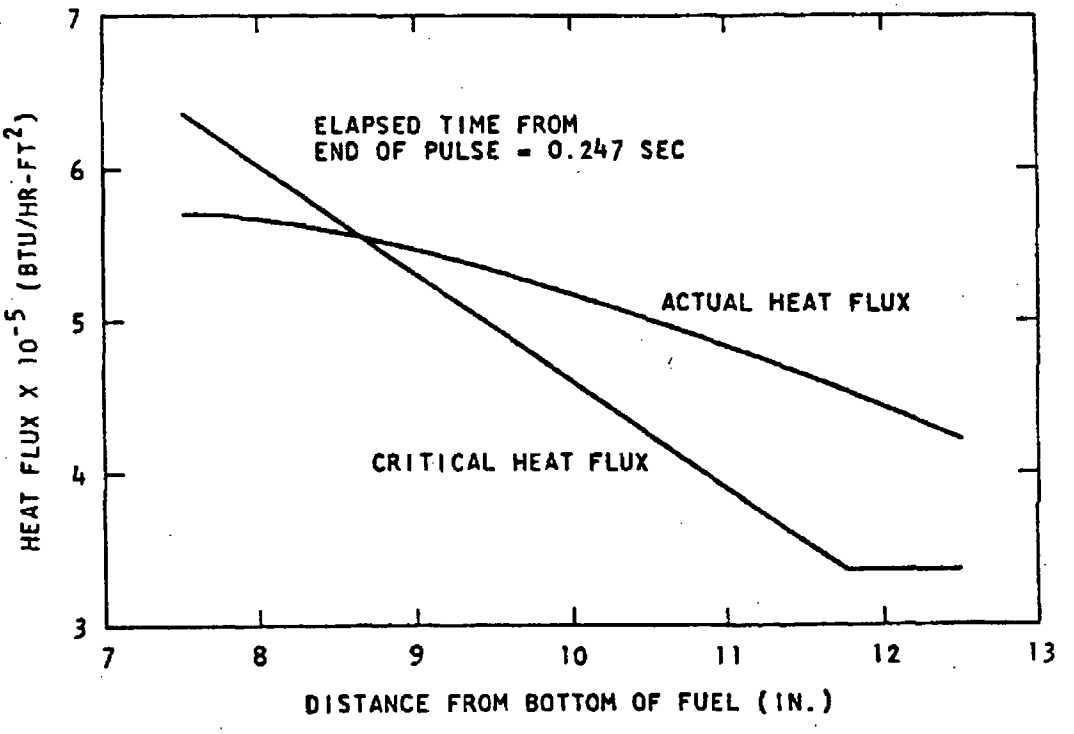

Figure 4A.7, Surface Heat Flux Distribution for Standard Non-Gapped (h_{gap} = 500 Btu/h ft^2 °F) Fuel Element After a Pulse.

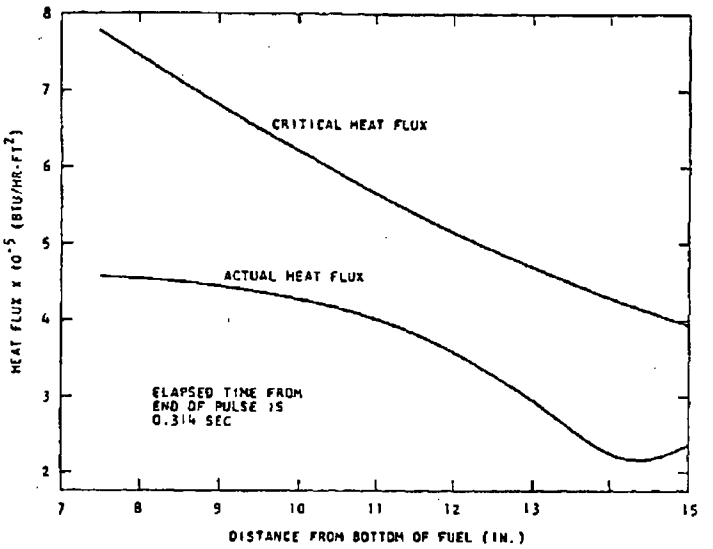

Figure 4A.8, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 375 Btu/h ft^2 °F) After a Pulse.

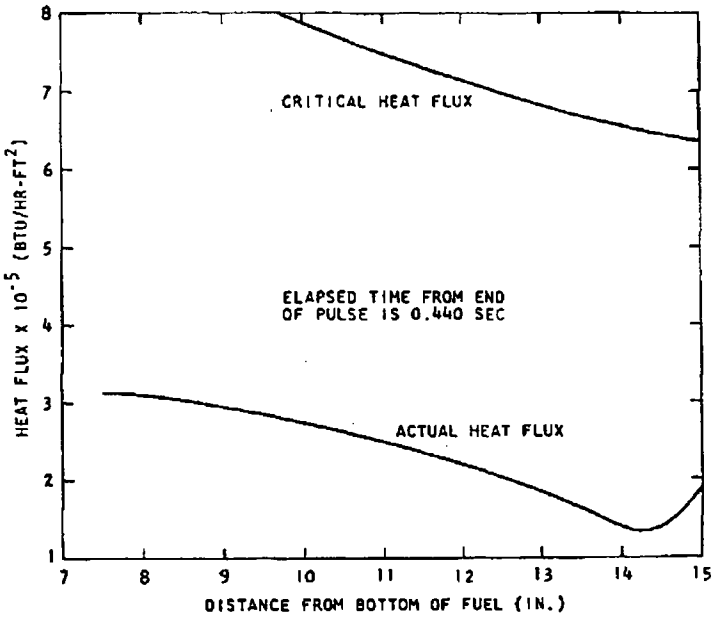

Figure 4A.9, Surface Heat-Flux Distribution for Standard Non-Gapped Fuel Element (h_{gap} = 250 Btu/h ft^2 °F) After a Pulse.

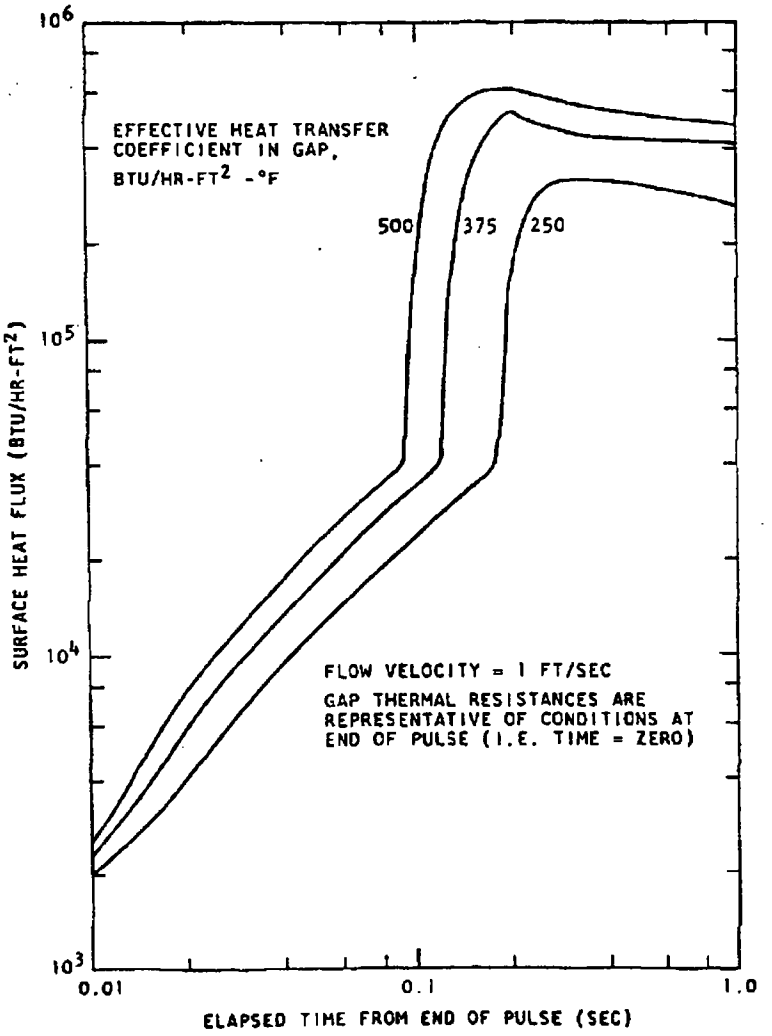

Figure 4A.10, Surface Heat Flux at Midpoint vs. Time for Standard Non-Gapped Fuel Element After a Pulse.

5.0 REACTOR COOLANT SYSTEMS

The TRIGA is designed for operation with cooling provided by natural convective flow of demineralized water in the reactor pool. The suitability of this type of cooling at the power levels for this TRIGA has been demonstrated by numerous TRIGA installations throughout the world.

5.1 Summary Description

The cooling system is composed of three subsystems: the reactor pool, pool cooling and pool cleanup.

The principal function of the reactor pool is to remove fission and decay heat from the fuel, but pool water also serves to:

- provide vertical shielding of radiation from the reactor,
- moderate fission energy neutrons, and
- allow access to the reactor core for maintenance, surveillance, and experimental activities.

Reactor pool functions are accomplished passively. Heat removal occurs by natural circulation. Shielding is provided by the height of the water above the reactor core. Shielding aspects of the pool are discussed in Chapter 11. Approximately 1/3 of the core volume is water, contributing to moderation of fission energy neutrons. Core physics are addressed in Chapter 4. Maintenance, surveillance, and experiment activities are typically performed remotely (i.e., from the pool surface, through the pool water) with long-handled or tethered tools.

When the pool cooling system is operating, pool temperature is controlled by transferring heat from the pool water to a campus chill water system through heat exchanger. The pool cooling system is designed to maintain a higher pressure in the chill water system compared to the pool cooling system, assuring pool water cannot leak into the chill water system. Pool cooling piping is designed with vacuum breakers to prevent potential siphoning through the pool cooling system.

As described in Chapter 4, the fuel is encapsulated in a sealed stainless steel cladding; pool water quality is controlled to assure cladding integrity by the pool cleanup system. The pool cleanup system recirculates pool water through a filter and ion exchanger to remove suspended solids and chemical impurities.

5.2 Reactor Pool

The reactor pool is a [redacted] tall tank formed by the union of two half-cylinders with a radius of [redacted] separating the half-cylinders as

shown in Fig. 5.1A. The bottom of the pool is at the reactor bay floor level. The reactor core is centered on one of the half-cylinders. Normal pool level is 8.01 meters above the bottom of the pool, with a minimum level of 6.5 m required for operations. The volume of water in the pool (excluding the reflector, beam tubes and core-metal) is 40.57 m^3 and 32.50 m^3 for the nominal and minimum-required levels. Basic reactor coolant system data is provided in Table 5.1.

Table 5.1, Reactor Coolant System design Summary

Reactor Tank	Material	Aluminum plate (6061)
	Thickness	¼ in. (0.635 cm)
	Volume (maximum)	11000 gal (41.64 m^3)
Coolant Lines	Pipes	Aluminum 6061
	Valves	Iron-Plastic Liner, 316 SS Ball and Stem
	Fittings	Aluminum (Victaulic)
Coolant Pump	Type	Centrifugal
	Material	Stainless Steel
	Capacity	250 gpm (15.8 lps)
Heat Exchanger	Type	Shell & Tube
	Materials (shell)	Carbon steel
	Materials (tubes)	304 stainless steel
	Heat Duty	[redacted]
	Flow Rate (shell)	[redacted]
	Flow Rate (tubes)	[redacted]
Typical Heat Exchanger Operating Parameters	Tube Inlet	[redacted]
	Tube Outlet	[redacted]
	Shell Inlet	[redacted]
	Shell Outlet	[redacted]

5.2.1 Heat Load

The reactor pool is open at the top (with an argon purge system normally drawing air across the surface) surrounded by concrete. Conduction of heat through the concrete combined with forced convection and evaporation provide ambient cooling adequate to control pool water temperature at low power operations and decay heat removal. At 1 MW operation the reactor is capable of heating up the pool under nominal level of 8.1 m at 20.7°C per hour, and at 2 MW approximately 41°C per hour.

5.2.2 Pool Fabrication

The pool is fabricated from sheets of 0.25 in. (0.635 cm) 6061 aluminum in 4 vertical sections welded to a ½ in. thick aluminum plate. Full penetration inspection was performed on tank components during fabrication, including 20% of the vertical seam welds, 100% on the bottom welds (internal and external to the pool volume), and 100% on the beam port weld external to the pool volume. A single floor centerline seam weld was used; a sealed channel was welded under the seam and instrumented through a ¼ in. NPT threaded connection to perform a leak test during fabrication. A 2 in. X 2 in. X ¼ in. (square) aluminum channel was rolled and welded to the upper edge of the tank.

5.2.3 Beam Ports

Beam port penetrations are fabricated around the core to allow extraction of radiation beams to support experiments. The beam ports are centered 90.2 cm (35 in.) above the pool floor, 7.2 cm (2.83 in.) below the core centerline. The section of the beam ports that are an integral part of the pool include an in-pool section, interface with the pool wall, and a section extending outside of the pool.

In pool sections are 0.1524 m (6 in.) in diameter, with a 0.00635 cm (0.25 in.) wall thickness. The in pool section for BP 1 and 5 is 6 in. while the remaining in-pool beam port sections are longer. Supports (2 in. X 2 in. X ¼ in. aluminum angle bracket) are welded at one end to the bottom of the pool and at the other end directly to BP 2, 3, and 4 to support the weight of the extended lengths. BP 2 and 4 terminate at the outer surface of the reflector, while BP 3 extends into the reflector, terminating at the inner shroud.

BP 2 terminates in an oblique cut, extending approximately 43 cm (16.94 in.) into the pool with the support 12.7 cm (5 in.) from the in-core end. BP 3 extends 73 cm (28.75 in.) into the pool with the support 37.62 cm (14.8125 in.) from the in-pool end. BP 4 extends 43 cm into the pool (16.94 in.) with the support 7.62 cm (3 in.) from the in pool end. Beam port 1 and 5 are aligned in a single beam line. A flight tube inserted into BP 1/5 extends through the reflector near the core shroud; BP 1 and 5 are equipped with a bellows to seal a neutron flight-tube. Beam ports 2, 3, and 4 are sealed at the in-pool end. BP 2 is tangential to the core shroud, offset 34.29 cm (13 ½ in.) from core center rotated 30° with respect to BP 3. Beam port 3 is oriented 90° with respect to BP 1/5, aligned to the center of the core. Alignment of BP 4 is through the core center, rotated 60° from BP 3.

The beam port interface with the pool wall includes a reinforcing flange on the inner pool wall. The flange is 3/8 in. thick, 11 in. in diameter. The flange is welded on the outer diameter to the pool wall. The inner diameter of the flange is welded to the beam port tube.

The beam ports extend approximately 15.4 cm (6 in.) outside of the area define by the pool walls. A stainless steel (304) ring is machined for a slip fit over the 6 in. (15.24 cm) aluminum tube extension. The ring is welded to 6 5/8 in. diameter stainless steel pipe (SST 304W/ASTM 312) extending the flight tube for the beam port into the biological shielding.

Four pads are welded to the pool floor reinforcing the floor for the core and support structure. As noted, the in-pool beam port supports are welded to the pool floor.

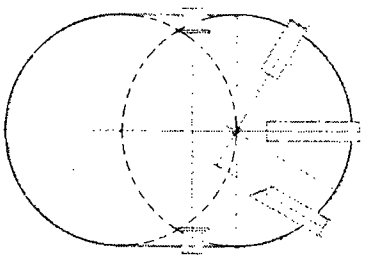

Figure 5.1A, Pool Fabrication

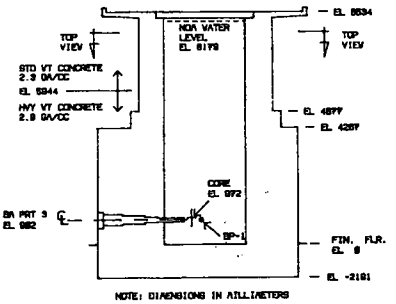

Figure 5.1B, Cross Section

Figure 5.C, Beam Orientation

5.3 Pool Cooling System

The pool cooling system is shown in Fig. 5.2.

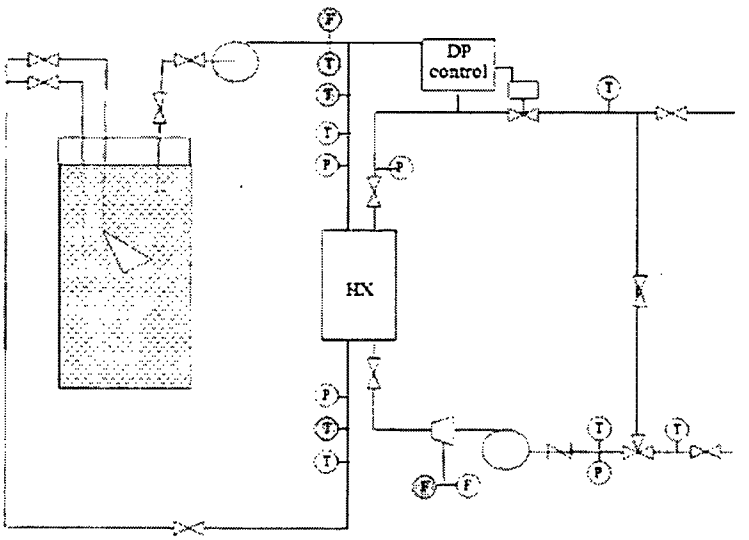

Figure 5.2, Pool Cooling System

5.3.1 Reactor Pool

The reactor pool is open at the top (with an argon purge system normally drawing air across the surface) surrounded by concrete. Conduction of heat through the concrete combined

with forced convection and evaporation provides ambient cooling adequate to control pool water temperature at low power operations. At 1 MW operation the reactor is capable of heating up the pool 20.7°C per hour, at 2 MW approximately 41°C per hour. As noted above, fuel element cooling analysis assumed a maximum temperature of 48.9°C, which could be achieved after operating at the maximum power level for short periods. Therefore a pool cooling system is installed to control pool temperature.

Historically the maximum pool temperature of 48.9°C was established to protect the integrity of ion exchange resin. The reactor pool is normally controlled at about 20°C. In the absence of pool cooling, a temperature rise of 28°C (from 20°C to the maximum permissible pool temperature of 48.9°C) could occur in 1.35 hours at 1 MW, or about 40 minutes at 2 MW. Even without pool cooling, time-limited support for experimental program is possible while still maintaining pool temperature below the limiting value used in analyses.

5.3.2 Pool Heat Exchanger

A tube and shell heat exchanger is installed for heat removal from the reactor pool to the available chilled water system. Design and operating parameters for the heat exchanger are provided in Table 5.1. Heat exchanger capacity is designed to maintain reactor pool temperature at or below the maximum temperature used in heat transfer analysis, 120°F (48.9°C). The stable temperature is maintained by a heat exchanger capacity equivalent to the reactor core thermal output capacity. Other heat losses such as evaporation, or heat gains from the pump, are considered negligible. Heat transfer is defined by:

$$q = U \cdot A \cdot \delta T$$

where U overall heat transfer coefficient (watt/m^2 -°C)
A — surface area for heat transfer (m^2)
δT_m = true mean temperature difference (°C)

The overall heat transfer coefficient of a tube and shell heat exchanger is composed of three terms, the convective heat transfer from the fluid in the tubes to the tube walls, the conductive heat transfer thru the tube wall, and the convective heat transfer from the outside tube wall to the fluid in the shell of the heat exchanger. Based on the outside tube area for heat transfer, the overall heat transfer coefficient is defined as[1]:

[1] Heat Transfer, Holman, JH. P., McGraw-Hill, 4th Edition (1976) pp386-391

$$U_c = \left[\frac{A_0}{A_i \cdot h_i} + \frac{A_0 \cdot ln\frac{r_0}{r_i}}{2 \cdot \pi \cdot k \cdot l} + \frac{1}{h_0} \right]^{-1}$$

Wherer:

A_o is the total outside tube area (m^2)
A_i is the total inside tube area (m^2)
r_i is the tube inside radius (m)
r_o is the tube outside radius (m)
h_i is the convective heat transfer coefficient between fluid in tubes and tube wall (W/m^2-°C)
h_0 is the convective heat transfer coefficient between fluid in shell and tube wall (W/m^2-°C)
k is the conductive heat transfer coefficient in the tube wall (W/m^2-°C)
l is the total tube length in heat exchanger (m)

A correction is applied for fouling of heat exchanger caused by buildup of various deposits. The overall heat transfer coefficient for a fouled heat exchanger is determined by:

$$U_f = \frac{1}{R_f + \frac{1}{U_c}}$$

where R_f is the fouling factor, (non-dimensional). The convective heat transfer coefficient is defined as

$$h = \frac{N_u \cdot k}{d}$$

Where:

Nu is the Nusselt Number
k is the thermal conductivity of the fluid evaluated at the appropriate average temperature (W/m-°C)
d is the tube diameter or applicable hydraulic diameter (m)

The complicated nature of turbulent flow heat transfer is described by a Nusselt number determined by experimental correlation with the Reynolds and Prandtl Numbers. Dittus and Boelter[2] recommend the following relation for fully developed turbulent flow in tubes:

$$Nu_t = 0.023 \cdot Re^{0.8} \cdot Pr^n$$

where parameters are measured inside the tubes

Re is the Reynolds Number based on tube diameter,
Pr is the Prandtl Number at average fluid temperature,
n is 0.4 for heating, 0.3 for cooling.

The relation for the shell side of a baffled cross flow heat exchanger is suggested by Colburn[3] as follows:

$$Nu_t = 0.33 \cdot Re^{0.6} \cdot Pr^{0.33}$$

where parameters are measured outside the tubes and

Re is the Reynolds Number based on tube outside diameter and velocity at minimum shell cross sectional area,
Pr is the Prandtl Number at average fluid temperature.

The product terms, δT_m, are defined consistent with the definition of U and heat exchanger design. The total cross sectional area of the tubes is represented by the heat transfer area, A, as specified by the heat transfer coefficient, U. The true mean temperature difference, 6Tm, is related to the heat exchanger type by a correction factor, F, and a log mean temperature difference, LMTD[4]. The correlation relates a simple single pass heat exchanger with more complex multiple pass baffled units. A relation is defined by

$$\delta T_m = F \cdot LMTD$$

where

F is the correction factor[5], [6],

[2] University of California (Berkeley) Pub. Eng, Dittus, F. W and Boelter, L. M. K., Vol 2, pp 443 (1930)
[3] A method of Correlating Forced Convection Heat Transfer Data and Comparison with Fluid Friction, Colburn, A. P. ,Trans. AIChE, Vol 29, pp 174-210 (1933)
[4] Heat Tansfer, White, op. cit.
[5] Mean Temperature Difference in Design, Bowman, R. A., Mueller, A. C., and Nagle, W. M., Trans. ASME, Vol 62 (1940) pp283-294
[6] Standards, TEMA 3rd Ed., Tubular Heat Exchanger Manufacturers Association New York (1952)

$$LMTD = \frac{T_a - T_b}{ln\frac{T_a}{T_b}}$$

For a counter flow heat exchanger

T_a is ($T_{\text{hot fluid in}}$ - $T_{\text{cold fluid out}}$)

T_b is ($T_{\text{hot fluid out}}$ - $T_{\text{cold fluid in}}$)

Actual heat exchanger capacity is calculated using an energy balance on either the shell or tube fluid. The heat transfer is defined as:

$$T_b = T_{hot\_fluid\_out} - T_{hot\_fluid\_in}$$
$$T_a = T_{hot\_fluid\_in} - T_{hot\_fluid\_out}$$
$$q = C \cdot (T_{in} - T_{out})$$

where

$C = m \cdot c_p$

m is the mass flow rate,

c_p is the fluid specific heat,

T_{in} is the temp of fluid entering heat exchanger,

T_{out} is the temp of fluid exiting heat exchanger.

In the current case T_{out} of either fluid is not known. Only T_{in} (100°F pool water, 48°F coolant water) and the mass flow rate of both fluids are known. To determine T_{out} the effectiveness/NTU method[7,8] is used. The dimensionless parameter called the heat exchanger effectiveness E is defined as

$$\varepsilon = \frac{Actual\_HX}{Max\_HX}$$

where the maximum possible heat transfer is

$$Max_{HX} = c_{min} \cdot (T_{hot\_in} - T_{cold\_in})$$

Substituting (11) for each fluid and (13) into (12) results in

$$\varepsilon = \frac{c_{hot} \cdot (T_{hot\_in} - T_{cold\_in})}{c_{min} \cdot (T_{hot\_in} - T_{cold\_in})}$$

[7]Heat Transfer, White, F. M., Addison-Wesley (1984) pp 512-513

[8] Compact Heat Exhcangers,2nd Ed., Keys, W. and Landon, A. L., McGraw-Hill (1964)

for the hot fluid and

$$\varepsilon = \frac{c_{cold} \cdot (T_{cold\_in} - T_{cold\_in})}{c_{min} \cdot (T_{hot\_in} - T_{cold\_in})}$$

for the cold fluid. The heat exchange effectiveness determined by[9] for a shell and tube heat exchanger with one shell pan and any multiple of tube passes

$$\varepsilon = z \cdot \left[1 + r + B \cdot \frac{1 + e^{-N \cdot B}}{1 - e^{-N \cdot B}}\right]^{-1}$$

Where

R is C_{min}/C_{max}
U is the overall heat transfer defined in (2)
A is the surface area for heat transfer
B is $(1 + r^2)^{1/2}$

Once the effectiveness is calculated and the above used to determine T_{hot} out and T_{cold} out. These may then be used to determine the capacity of the heat exchanger.

Table 5.2, Heat Exchanger, Heat Transfer and Hydraulic Parameters

Component/Parameter	Specification	Value	Units
Tubes	Outside Diameter	[redacted]	in. (cm)
	Wall Thickness	[redacted]	in. (cm)
	Thermal Conductivity	[redacted]	Btu/hr-ft-°F
Flow Area	Tube Side	[redacted]	in^2 (cm^2)
	Shell Side	[redacted]	in^2 (cm^2)
Heat transfer Surfaces	Na	[redacted]	ft^2 (m^2)
Average Prandtl No.	Tube	[redacted]	na
	Shell	[redacted]	na
Average Kinematic Viscosity	Tube	[redacted]	ft^2/s ($m2/^s$)
	Shell	[redacted]	ft^2/s ($m2/^s$)
Reynolds No.	Tube	[redacted]	na
	Shell	[redacted]	na
Corrective Heat Transfer Coefficients	Tube	[redacted]	Btu/hr-ft^2-°F (W/m^2-°C)
	Shell	[redacted]	Btu/hr-ft^2-°F (W/m^2-°C)
Overall Heat Transfer Coefficient	Tube	[redacted]	Btu/hr-ft^2-°F (W/m^2-°C)
	Shell	[redacted]	Btu/hr-ft^2-°F (W/m^2-°C)
Effectiveness	Clean	[redacted]	na
	Fouled	[redacted]	na
LMTD	Na	[redacted]	° F (°C)
Corrective Factor F	Na	[redacted]	na
Capacity	Clean	[redacted]	kW

[9] Compact Heat Exchangers op. cit.

	Fouled	1070	kW

Heat removal capacity and thus pool heat rate is specified by analysis of a tube and shell heat exchanger. Heat removal rate of 1140 kW is expected at a flow rate of 400 gal/min (25.2 liters/sec) of chilled water at 48°F (8.89°C). The presence of fouling in the heat exchanger is considered minimal based on the purity of the two heat exchanger fluids. Capacity is reduced to 1070 kW for a fouling factor of 0.0004.

5.3.3 Secondary Cooling

When the pool cooling system is operating, pool temperature is controlled by transferring heat from the pool water to a campus chill water system in a heat exchanger. The chilled water system is operated by the University for cooling of Pickle Research Campus buildings and equipment through a campus supply loop. At the time of NETL construction, chilling capacity was provided by multiple 1200 ton (4220 kW) units, with 25% of the chilling system capacity of one unit allocated to pool cooling. Construction is currently underway to remove a major load/demand from the shared system; the Texas Advanced Computing Center is expanding, and installing a dedicated cooling system. The PRC chill water supply is also currently planning for system renovations which will expand capacity to meet campus growth and development.

Chill water pumps in the NETL building draw from the campus supply loop and direct flow to the loads at the NETL, including two installations (2 pumps each) supporting building ventilation and air conditioning, and a single pump providing chill water flow for the pool cooling system heat exchanger.

5.3.4 Control System

Chill water flow is normally about 500 gpm. Chill water flow through the heat exchanger is regulated to control pool temperature, sensed in flow to the heat exchanger. If temperature is lower than the control setpoint, a bypass line diverts chill water flow around the heat exchanger. Conversely, if temperature rises above the setpoint, the bypass flow is reduced so that more flow passes through the heat exchange.

The pool cooling system is designed to maintain a higher pressure in the chill water system compared to the pool cooling system, assuring pool water cannot leak into the chill water system. Pool cooling is normally 250 gpm; if pressure at the chill water outlet rises above the pressure at the pool inlet to the heat exchanger, the pool outlet inlet is throttled by a control valve.

5.4 Primary Cleanup System

The primary cleanup system (Fig. 5.3) is designed to use filtration and ion exchange to control water quality for corrosion control, radioactivity control, and optical clarity of the coolant water. Water purity is monitored by analysis of the water conductivity.

The purification skid is located in room [redacted] at about the same level as the reactor core. The skid consists of a pump, flowmeter, filter, resin bed, and instrumentation. The cleanup system is normally operated continuously to provide removal of suspended particles and soluble ions in the coolant water. The system flow rate is about 10 gpm (0.6 lps).

Suction of water from the pool is provided by two inlets in the reactor pool, neither of which extends more than 2 meters below the top of the reactor tank. Valves at the pool surface allow suction from either a subsurface inlet or from a surface skimmer designed to collect and remove floating debris. Accidental siphoning of reactor pool water is prevented by siphon breaks similar to those on the coolant piping. Return flow to the pool is through a subsurface discharge pipe. Valves are provided for isolation of the suction or return lines, and for isolation of system components for maintenance or resin replacement.

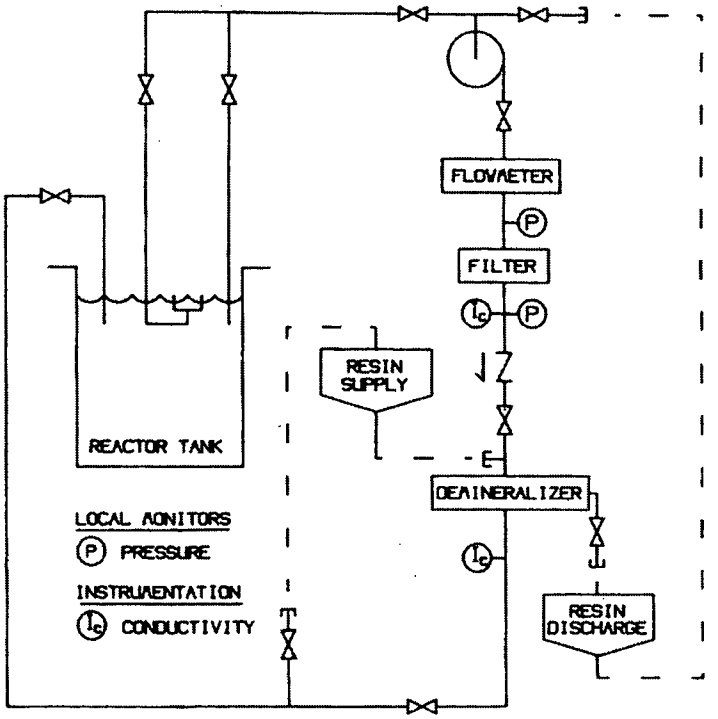

Figure 5.3, Pool Cleanup System

Purification functions of the loop are generated by two components, a filter for removal of suspended materials and a resin bed for removal of soluble elements. Typical filtration is provided with 25 micron filters. Typical ion exchange is provided by 0.85 cubic meters of mixed cation and anion resin. Resin historically used is rated to 120°F; therefore the maximum pool temperature used in analysis is 120°F (48.9°C). Resin performance is monitored as the decrease in conductivity across the demineralizer, measured by inline conductivity cells. Measurements of water conductivity as low as 2.0 micromho per centimeter (or resistance of 1 megohm per centimeter) are maintained by filtration and ion exchange. The conductivity is reduced further by control of materials exposed to the reactor coolant, minimizing dust settling to the pool surface, and occasional cleaning of pool surfaces. Experience has shown that conductivities of 5.0 pmho/cm are sufficient to maintain acceptable limits on corrosion plus good water optical quality and removal of activation products in the water.

Should radioactivity be released from a clad leak or rupture of an experiment, detection of the release would be signaled by the continuous air monitor or by the reactor room area monitors. Based on coolant transport time calculations in the safety analysis section, these monitors should register an increase in coolant radioactivity within approximately 60 seconds of the time of radioactivity release. The transport time is estimated from the time for the coolant exposed in the core to reach the surface of the water where the continuous air monitor will detect a release of radioactivity from the pool water. An alternate indication of radioactive release is provided if a water activity monitor is installed or by a GM detector area monitor.

Experience with this purification equipment in other TRIGA systems has shown that coolant conductivity can be easily maintained at levels of less than five micromhos per centimeter using the materials contained in the coolant system design. Furthermore, this experience has shown that no apparent corrosion of fuel clad or other components will occur if the conductivity of the water does not exceed five micromhos per centimeter when averaged over a 30-day period.

5.5 Makeup Water System

A connection from the domestic (potable) water system to the pool cleanup system provides makeup water to replenish pool inventory losses from evaporation. The potable water header supplies a mechanical filter and a bank of 4 deionizers. Each deionizer is capable of being bypassed, and is instrumented with an indicator that energizes a white lamp if conductivity is greater than 200 kmhos per cm, and a red lamp if conductivity exceeds the setpoint. The deionizers supply lab-spaces and makeup water to the pool cleanup system. A pump recirculates water through the final deionizer and the laboratory distribution header.

A line from the deionizers is routed through shutoff valves and a check valve to a flexible extension in the water treatment room. The flexible extension is equipped with a conductivity monitor and terminated in a quick disconnect fitting that allows physical separation of the two systems except during periods in which the makeup process is operating. When the pool

inventory has decreased from evaporation, the quick disconnect is made up at the suction of the cleanup pump to provide makeup water through the cleanup filter and demineralizer.

5.6 Cooling System Instruments and Controls

Numerous cooling and cleanup system parameters are measured by local sensors in the system lines. Transmitters provide some of the parameters remotely to the control room. Temperature and pressure probes are located on the inlet and outlet lines of the pool water side and chill water side of the heat exchanger. A local indication of flow in the coolant loop is provided by the pressure drop across a venturi in the flow path. Purification loop flow is measured by an in line flow meter. Water pressure before and after the filter in the purification loop is measured locally for indication of filter condition. Parameter monitoring points are illustrated in Fig. 5.2 and 5.3. The parameters that are considered part of the water system instrumentation system are presented in Figure 5.4.

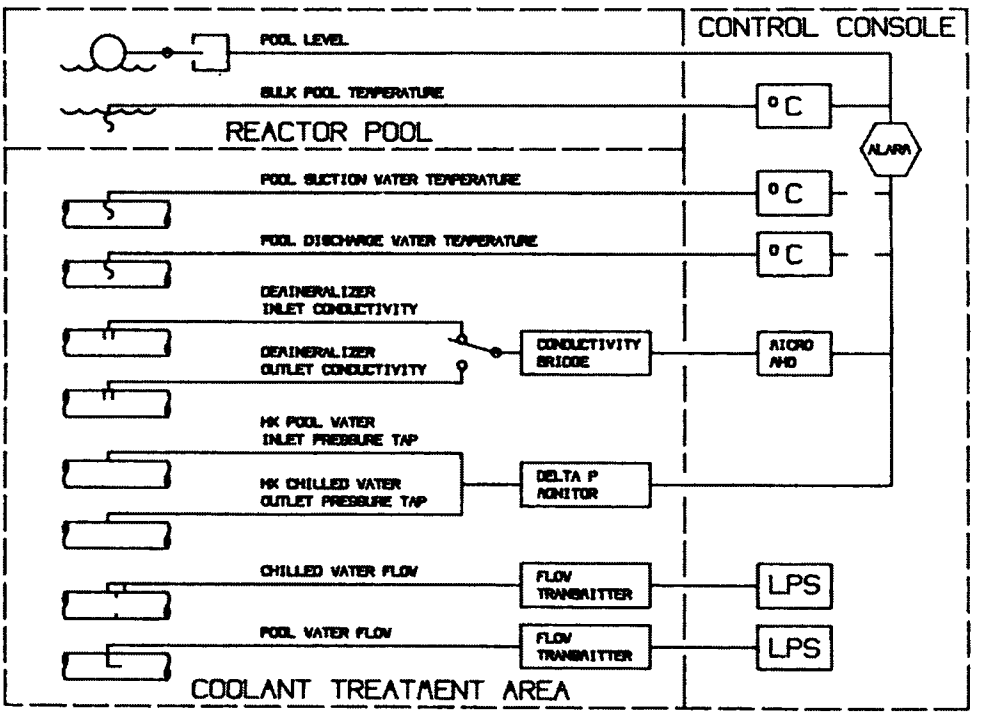

Figure 5.4, Cooling and Cleanup Instrumentation

The cooling system parameters normally available in the control room include coolant temperatures, flow rates, differential pressure status, and pool level. Two temperature probes, one in the pool suction line and one in the line, allow monitoring of heat exchanger cooling function. Typical temperature probes used are resistance temperature detectors (RTD's). Two flow meters, one in the chilled water line and one in the pool water line provide information on system flow rates. A differential pressure monitor provides an alarm if the

pressure at the high pressure point on the heat exchanger tube side is not less than the low pressure point on the shell side. The differential pressure is designed for a difference substantially greater than 7 kilopascals (1 lb/sq. in.).

Water quality is measured by two conductivity cells in the purification loop. The cells are located on inlet and outlet lines of the demineralizer that readout locally in the control room. Typical conductivity cells are composed of two parts, titanium electrodes shielded by thyton for conductivity measurement, and a thermister for temperature compensation. A Wheatstone bridge circuit on the purification skid is connected to the cells. A switch allows selection of either inlet or outlet conductivity.

6.0 ENGINEERED SAFEGUARD FEATURES

As (1) discussed in Chapter 13, Chapter 5, (2) identified in previous analysis, and (3) identified from experience at other TRIGA reactors, emergency core cooling is not required for operations at steady state thermal powers below 1900 kW. No engineered safety features are required for the UT TRIGA II research reactor because the steady state power limit is 1,100 kW.

6.1 References

TRIGA Reactor facility , Nuclear Engineering teaching Laboratory, The University of Texas at Austin SAFETY ANALYSIS REPORT Submitted May 1991

NUCREG-1135, Safety Evaluation report related to the Construction Permit and Operating License for the Research Reactor at the University of Texas, Docket 50-602, May 1985

NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," U.S. Nuclear Regulatory Commission, 1987.

7.0 INSTRUMENTATION AND CONTROL SYSTEM

Design of the instrumentation and control system was intended for TRIGA reactor facilities as a replacement of analog reactor consoles. Initial verification and testing of the design by the manufacturer was a requirement prior to installation at The University of Texas at Austin. An evaluation by the University of the instrument and control console for the TRIGA was part of the initial installation of the console by the vendor. The system development, installation, and initial testing were the responsibility of the vendor, General Atomics.

The system described in this document is a microprocessor-based instrumentation and control system developed by the General Atomics (GA) TRIGA Reactor Division. This system incorporates (1) a digital wide-range neutron power monitor, (2) two analog power safety channels, (3) a variety of state-of-the-art signal conditioners and process controllers, plus (4) a digital data acquisition and control system incorporating a PC compatible computer.

There has been ample historical testing of the digital control system used at this facility. Digital control of research reactors has been accomplished by over twenty facilities across the United States for a number of years. The University of Texas digital TRIGA control system has been operating since 1992.

7.1 DESIGN BASES

The design and manufacture of this system complies with the guidance given in American Nuclear Society and the American National Standards Institute Guide Criteria for the Reactor Safety Systems of Research Reactors (ANSI/ANS 15.15-1978)[1,2]. This standard has served the research reactor community in lieu of the ad hoc application of similar standards for power reactors. Even if single-failure criteria for plant protective actions - not deemed mandatory by ANSI/ANS 15.15 for negligible risk reactors - were applied, the standard has allowed the use of simple redundancy, i.e., the monitoring of the same reactor parameter using independent, redundant equipment, to satisfy the single-failure criteria for the reactor safety system.

There are several advantages in a microprocessor-based system which enhances system safety, reliability, and maintainability over the analog control system used in previous TRIGA reactors:

1. The use of microcomputers allows data (operator input as well as output) to be more efficiently and systematically processed.

2. Several data reductions (such as on-line calculation of the prompt period during a pulse) can be done in near-real-time.

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[1] "Criteria for the Reactor Safety Systems of Research Reactors", American Nuclear Society, American National Standard, ANSI/ANS-15.15-1978.

[2] "Microprocessor Based Research Reactor Instrumentation and Control System", INS-27, Rev. A., GA Technologies, August 1987.

3. On-line self-diagnostics can be performed to determine the state of the system at all times.

4. Operational surveillance and operations data are accommodated with all information gathering and processing done routinely and regularly by the console computers.

The Instrumentation and Control System for the TRIGA reactor[3] is a computer-based design incorporating the use of one multifunction, NM-1000 microprocessor neutron flux monitoring channel and two companion current mode neutron-monitoring safety channels (NP-1000 and NPP-1000). The combination of these two systems provides an independent operating channel and the redundant safety function of percent power with scram. The NM-1000 provides wide range log power and multi-range linear power from source level to full power. The control system logic is contained in a separate control system computer (CSC) with graphics and text displays which are the interface between the operator and the reactor. Another system for data acquisition and control (DAC) functions as the interface point for interface circuitry, process signals and communications. The multifunction NM-1000, NP-1000 and NP-1000 units, and two system microprocessors, the control system computer (CSC) and data acquisition and control system (DAC) are development products of General Atomics. The basic system configuration is shown in Fig. 7.1.

Information from the NM-1000 channel is processed and displayed by the CSC. The NP-1000 and NPP-1000 are independent channels that deliver steady state power level data to the safety system scram circuit, hardwired analog indicators, and to the CSC for processing and display. The NPP-1000 also covers the pulse range. Operating ranges for the neutron channels are shown in Fig. 7.2.

The NM-1000 digital neutron monitor channel was developed for the nuclear power industry and is fully qualified for use in the demanding and restrictive conditions of a nuclear power generating plant. Its design is based on a special GA-designed fission chamber, and low noise ultra-fast pulse amplifier. The NP-1000 and NPP-1000 were developed specifically for use with research reactor safety systems and include several features not usually found in this type of application.

The CSC and its acquisition system, the DAC, manage all control rod movements, accounting for such things as interlocks, and choice of particular operating modes. It also processes and displays information on control rod position, power level, fuel and water temperature, and can display pulse characteristics. The CSC also performs many other functions, such as calibrating control rods, monitoring reactor usage, and historical operating data can be saved for replay at a later date.

[3] "Safety Analysis of Microprocessor Reactor Control and Instrumentation System", The University of Texas at Austin, 1989.

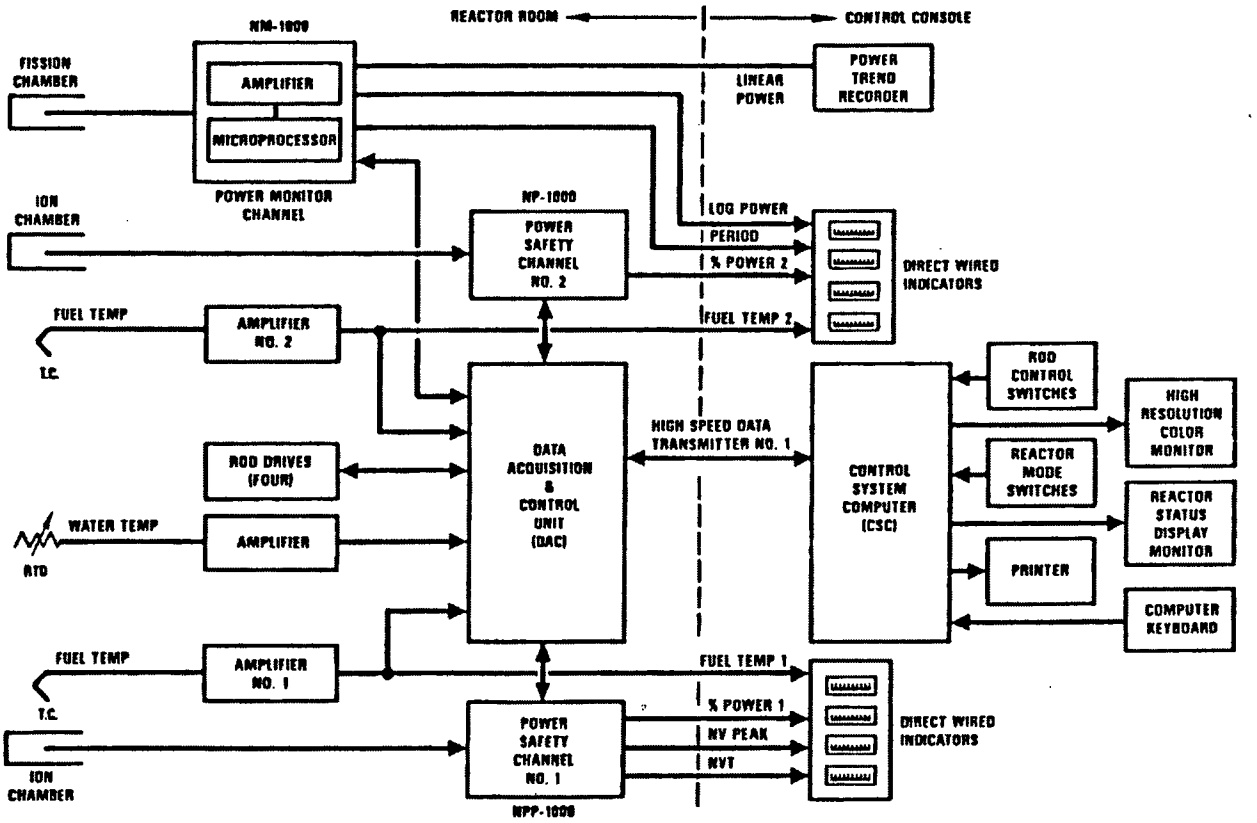


FIGURE 7.1, CONTROL SYSTEM BLOCK DIAGRAM

7.1.1. NM-1000 Neutron Channel

The NM-1000 nuclear channel has multifunction capability to provide neutron monitoring over a wide power range from a single detector. The selectable functions are any or all of the following:

a. Percent power.

b. Wide-range log power.

c. Power rate of change.

d. Multi-range linear power.

For the TRIGA ICS, one NM-1000 system is designated to provide the wide-range log power function and the multi-range linear power function. The wide-range log power function is a digital version of the patented GA 10-decade log power system to cover the reactor power range from below surface level to 150% power and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the Campbelling (root mean square [RMS] signal processing) range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the

full range of power. The microprocessor automatically tests the system to ensure that the upper decades are operable while the reactor is operating in the lower decades and vice versa when the reactor is at high power.

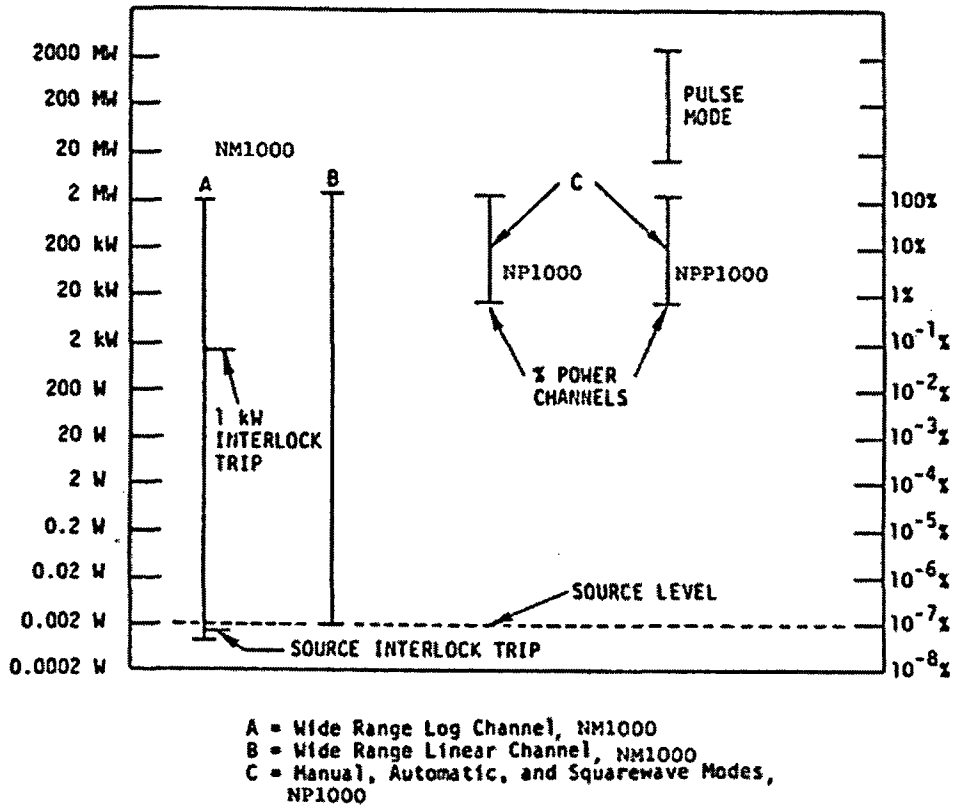


Figure 7.2, NEUTRON CHANNEL OPERATING RANGES

For the multi-range function, the NM-1000 uses the same signal source as for the log function. However, instead of the microprocessor converting the signal into a log function, it converts it into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power. The same self-checking features are included for the log function. The multi-range function is either auto-range or slave to a position switch on the operator's console via the control system computer. A linear power level signal is available for the percent power safety function for 1 to 125%.

The NM-1000 system is contained in two National Electrical Manufacturers Association (NEMA) enclosures, one for the amplifier and one for the processor assemblies. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low voltage power supplies, detector high-voltage power supply, and digital diagnostics and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics (between amplifier and processor), the microprocessor, a control/display module, low-voltage power supplies, isolated 4

to 20 mA outputs, and isolated alarm outputs. Outputs are Class 1E as specified by IEEE. Communication between the amplifier and processor assemblies is via two twisted-pair shielded cables.

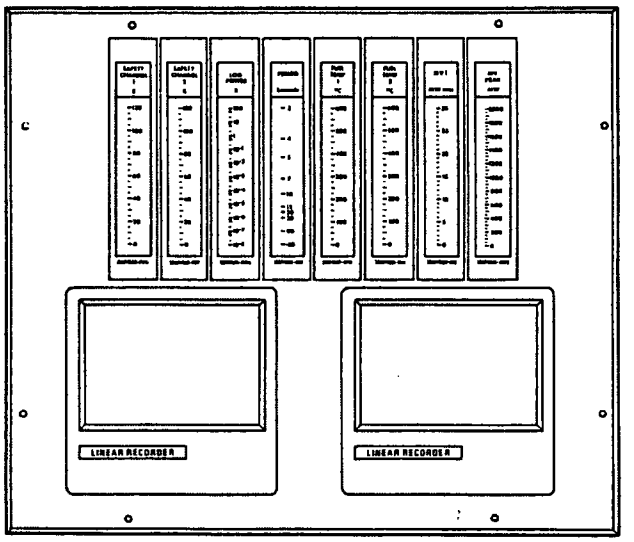


Figure 7.3, Auxiliary Display Panel

The amplifier/microprocessor circuit design employs the latest concepts in automatic on-line self diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system is automatically calibrated and checked (including the testing of trip levels) prior to operation. The checkout data is recorded for future use, and operation cannot proceed without a satisfactorily completed checkout. The accuracy of the channels is + 3% of full scale, and trip settings are repeatable within 1% of full-scale input.

The neutron detector uses the standard 0.2 counts per "nv" fission chamber that has provided reliable service in the past. It has, however, been improved by additional shielding to provide a greater signal-to-noise ratio. The low noise construction of the chamber assembly allows the system to respond to a low reactor shutdown level which is subject to being masked by noise.

7.1.2. NP-1000 Power Safety Channel

The NP-1000 Power Safety Channel is a complete linear percent power monitoring system mounted within one compact enclosure which contains current to voltage conversion signal conditioning, power supplies, trip circuits, isolation devices, and computer interface circuitry. The power level trip circuit is normally hardwired into the scram system and the isolated analog outputs are monitored by the CSC as well as being hardwired to a bar graph indicator.

A special version of the safety channel, the NPP-1000, provides measurement functions for peak pulse power, total pulse energy, automatic gain change and related trip points. The control system automatically selects proper gain setting for steady-state or pulse mode when the operator determines the reactor operating mode. Peak pulse power and total pulse energy are also set by the pulse operation mode.

Both safety channels, the NP-1000 and the NPP-1000, are identical except for the peak and energy circuits. The detector for each safety channel may be either an ionization chamber or self-powered in-core detector.

7.1.3. Reactor Control Console

The layout of the control console is shown in Fig. 7.3. The reactor control console contains several components needed by the operator for reactor control. Included are the following:

a. Reactor control panels.

b. Control System computer (CSC).

c. Two color graphics monitors.

d. Power and temperature meter panels.

e. Disk drive storage and a graphics printer.

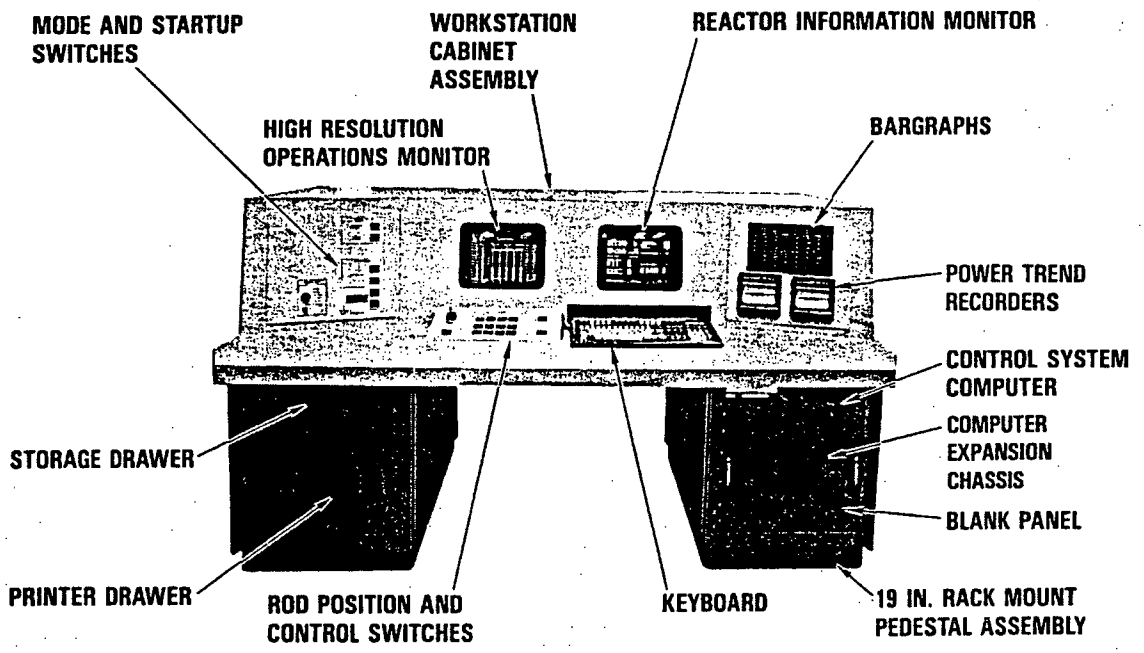


Figure 7.3, LAYOUT OF THE REACTOR CONTROL CONSOLE

A keyboard interface to the system computer is provided for operator control of several system functions. As previously mentioned, the power and period information from the NM-1000 channel and power levels from the NP-1000 and NPP-1000 channels are processed and displayed by the CSC. However, several wide-range channel parameters are also present on linear bar graph meter displays at the console. The NP-1000 and NPP-1000 safety systems are independent, with their own output displays, and connected directly to the control system scram circuit. Thus, wide-range log power, period, multi-range linear power and both percent power channels, have their output displayed on meters as well as on the monitors. This is also true of fuel temperature. Typical layouts of the console panels and video displays are shown in Fig. 7.4 and 7.5.

Functions of the rod control panel are represented in Figure 7.3 and are presented as:

a. Key switch for rod magnet power (also operates "Reactor On" lights).

b. Rod control switches and annunciators.

c. SCRAM-switch for safety function.

d. Annunciation is also provided for reset of the audio channel, as well as for reset of the alarm indicator following alarm clearance.

The CSC provides all of the logic functions needed to control the reactor and augments the safety system by monitoring for undesirable operating characteristics. It displays reactor operational information in a color format on a high-resolution LED monitor for ease of comprehension. Essentially all of the control systems logic contained in previous TRIGA reactor control systems is incorporated into the CSC. However, instead of using electronic circuits and electrical relay circuits, the logic is programmed into the computer.

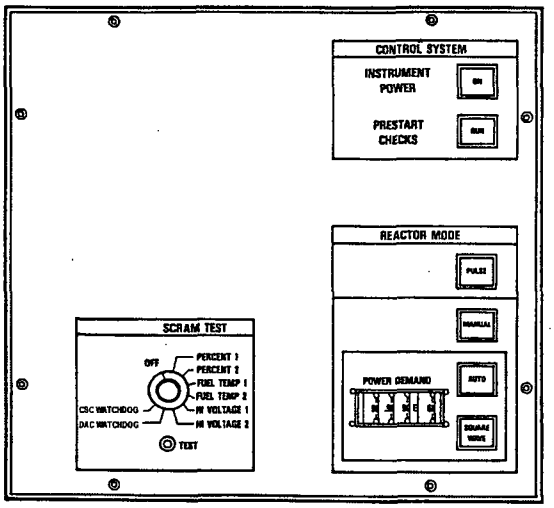
The availability of the computer allows great versatility and flexibility in operationally-related activities aside from the direct control of rod movements. Many other functions can also be performed by the CSC, such as monitoring reactor usage, storing pulse data, reactor operating history and logging operator usage.

Two auxiliary cabinets can be provided to the console for the addition of process instrument readout.

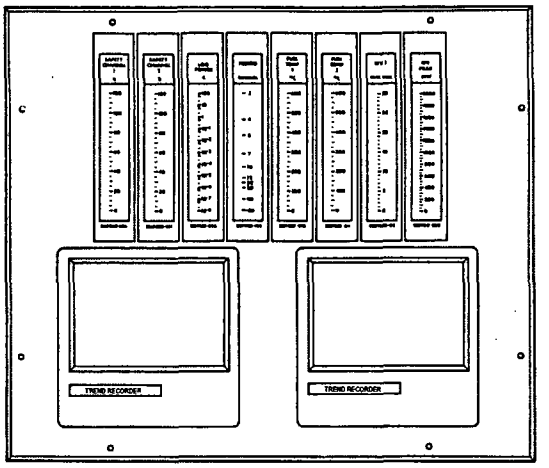
7.1.4. Reactor Operating Modes

There are four standard operating modes: manual, automatic, pulse, and square-wave. The manual and automatic modes apply to the steady-state reactor condition; the pulse and square-wave modes are the conditions implied by their names and require a pulse rod drive. Manual and automatic reactor control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor startup, change in power level, and steady-state operation. The pulse mode generates high-power levels for very short periods of time. High-

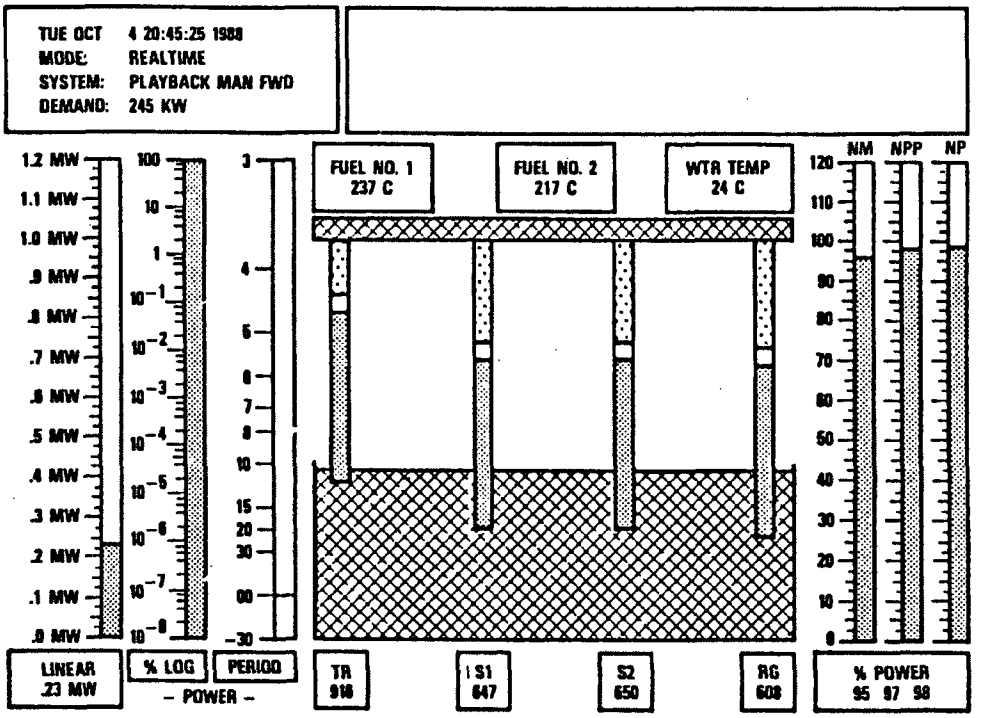
power and low-power pulse mode options are available. The square-wave operation allows the power level to be raised quickly to a desired power level.



Mode Control Panel



Analog Display Panel



graphic display

Figure 7.4, CONSOLE CONTROL PANELS

Figure 7.5, TYPICAL VIDEO DISPLAY DATA

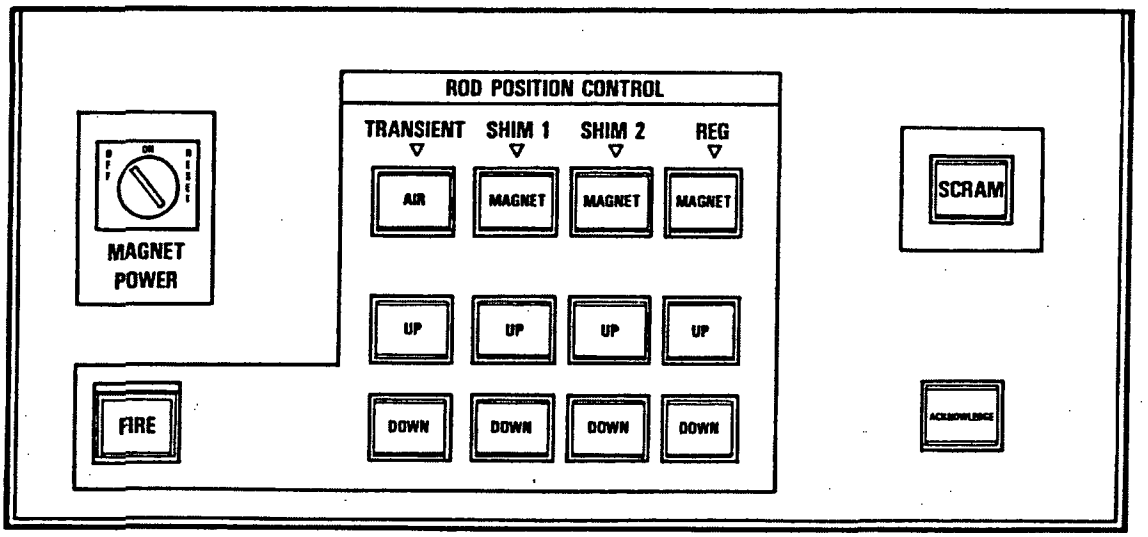


Figure 7.6, ROD CONTROL PANEL

Manual rod control is accomplished by the lighted push buttons on the rod control panel. The top row of annunciators, when illuminated, indicate magnet contact with the armature and magnet current. Depressing any one of the AIR-MAGNET push buttons will interrupt the current to that magnet and extinguish the magnet current on indication. If the rod is above the down limit, the

rod will fall back into the core and the AIR-MAGNET light will remain extinguished until the magnet is driven to the down limit where it again contacts the armature.

The middle row of pushbuttons (UP) and the bottom row (DOWN) are used to position the control rods. Depressing the pushbuttons causes the control rod to move in the direction indicated. Several interlocks prevent the movement of the rods in the up direction under conditions such as the following:

a. Scrams not reset.

b. Magnet not coupled to armature.

c. Source level below minimum count.

d. Two UP switches depressed at the same time.

e. Mode switch in one of the pulse positions.

f. Mode switch in AUTO position (regulating rod only).

There is no interlock inhibiting the down direction of the control rods except in the case of the regulating rod while in the AUTO mode.

Automatic (servo) power control can be obtained by switching from manual operation to automatic operation. All the instrumentation, safety, and interlock circuitry described above applies and is in operation in this mode. However, the regulating rod is now controlled automatically in response to a power level and period signal. The reactor power level is compared with the demand level set by the operator and is used to bring the reactor power to the demand level on a fixed preset period. Logic for the automatic control operation by proportional, integral-differential (PID) control is contained within the digital algorithms of the control system. The purpose of this feature is to automatically maintain the preset power level during long-term power runs. The function of automatic control is provided by the regulating rod with a stepping motor drive.

Reactor control in the pulsing mode consists of establishing criticality at a flux level below 1 kW in the MANUAL mode. This is accomplished by the use of the motor-driven control rods, leaving the transient rod either fully or partially inserted. The mode selector switch is then depressed. The MODE selection switches automatically connect the pulsing chamber to monitor and record peak flux (nv) and energy release (nvt). Pulsing can be initiated from either the critical or subcritical reactor state.

In a square-wave operation, the reactor is first brought to criticality below 1 kW, leaving the transient rod partially in the core. All of the manual instrumentation is in operation. The transient rod is ejected from the core by means of the transient rod FIRE pushbutton. When the power level reaches the demand level, it is maintained much the same as in the automatic mode. Two rods are used, the transient rod to achieve power and the regulating rod to maintain power.

7.1.5. Reactor Scram and Shutdown System

A reactor protective action[4] interrupts the magnet current and results in the immediate insertion of all rods under any of the following:

a. High neutron fluxes from either NP-1000 or NPP1000.

b. High-voltage failure on the NM-1000, NP-1000, or NP1000.

c. High fuel temperature (one out of two).

d. Manual scram.

e. Peak neutron flux or energy (pulse mode).

f. Minimum period (available for use as desired).

g. External safety switches (for experiments).

h. Loss of electrical power to the control console

i. Watchdog circuits for each computer to monitor computer status by updating timers.

All scram conditions are automatically indicated on the monitor. A manual scram will also insert the control rods and may be used for a normal fast shutdown of the reactor. The scram circuit safety function is an independent system that depends on wiring independent of the digital control system functions.

Several conditions of the digital processing system will cause the scram mode condition. Among these are the loss of communication between the two computers, a database timeout condition or failure of a digital input scanner. By updating dual programmable timers, watchdog circuits at periodic intervals, determine the execution status of key elements of the computer digital program.

Two options for reliable operation performance may be installed as necessary. One option for conditions requiring long-term, high power steady-state operation, is configuration of three safety channels with 2 out of 3 logic, allowing one channel to be out of service without requiring reactor shutdown. Another option is an uninterruptable power supply as auxiliary power for the reactor control and monitoring systems for intermittent power failures of periods up to 15 minutes.

[4] "Safety Analysis of Microprocessor Reactor Control and Instrumentation System", The University of Texas at Austin, 1989.

7.1.6. Logic Functions

A simplified control system logic diagram is shown in Fig. 6.7. The two separate flux monitoring safety channels ensure safe operation of the reactor by monitoring the power level and act independently to shut the reactor down if a potentially dangerous condition exists. They provide information to the control system, which consists of three major parts: a reactor control console (RCC), Control System Computer (CSC) and Data Acquisition Computer (DAC). In addition, there are two high resolution LED monitors and a graphics printer. The left-most display monitor contains basic reactor operation control data. The second display monitor provides information on annunciators and special control features. Data from both displays is available for log-records.

The CSC provides the operator with immediate information concerning reactor conditions visually on the monitors. At the same time, the DAC is collecting data from the reactor system and concentrating it into a permanent data base, which is transmitted to the CSC on request and maintained for historical purposes.

During operation of the reactor, the operator's commands to adjust control rod positions are transmitted from the CSC to the DAC to the drive mechanisms. In the automatic mode the DAC controls the position of the rods. The rod control program for automatic operation applies proportional-integral-differential control logic. Digital rod position indication is shown in inches, with a resolution of < 0.1 in. and accuracy equal to or better than + 0.2% of indicated position.

The control rod interface accepts the digital commands from the data acquisition and control system (DAC) to operate the control rod motors. It contains the opto-isolation circuits which send the up-down limits and loss of contact signals to the control rod logic system. An excitation power supply provides a stable reference voltage for the rod position indicator system.

The magnet supply furnishes the required 200 mA needed for the rod magnets to hold control rods in contact with the armature. An opto-isolator detects the absence of magnet current to each drive magnet.

A gamma chamber provides the signal for peak power (nv) and energy release (nvt) in the pulse mode. The nv/nvt amplifier provides the high impedance interface, high voltage and calibration circuits for the pulsing detector.

All of the analog signals and digital signals are routed to the DAC chassis. However, the prime reactor operating signals are also sent directly to the control room. These signals include log power, period, percent power (2), fuel temperature (2), and pulse mode signals for peak and energy.

The DAC system converts the analog signals to a digital equivalent for transmitting along with the digital signals to the CSC in the control room. The DAC chassis receives control instructions from the CSC, via the communication link, which in turn moves the control rods as requested by the operator and causes the individual subsystems to go to the calibrate mode when commanded by the system or operator.

The fuel temperature transmitters are accurate, highly stable units which convert the 0-600°C fuel temperature into a 4-10 mA output signal. A level comparator is included which provides scram capability through an isolated contact state change when the preset level is exceeded.

The water temperature transmitters are standard Resistance Temperature Detector (RTD) transmitters which convert the 0 to 100°C temperature into a 4-20 mA signal. The transmitters have a self-contained power supply.

External switches are provided with terminal strips to terminate and connect various switches to the DAC chassis (beam port open-close, etc.)

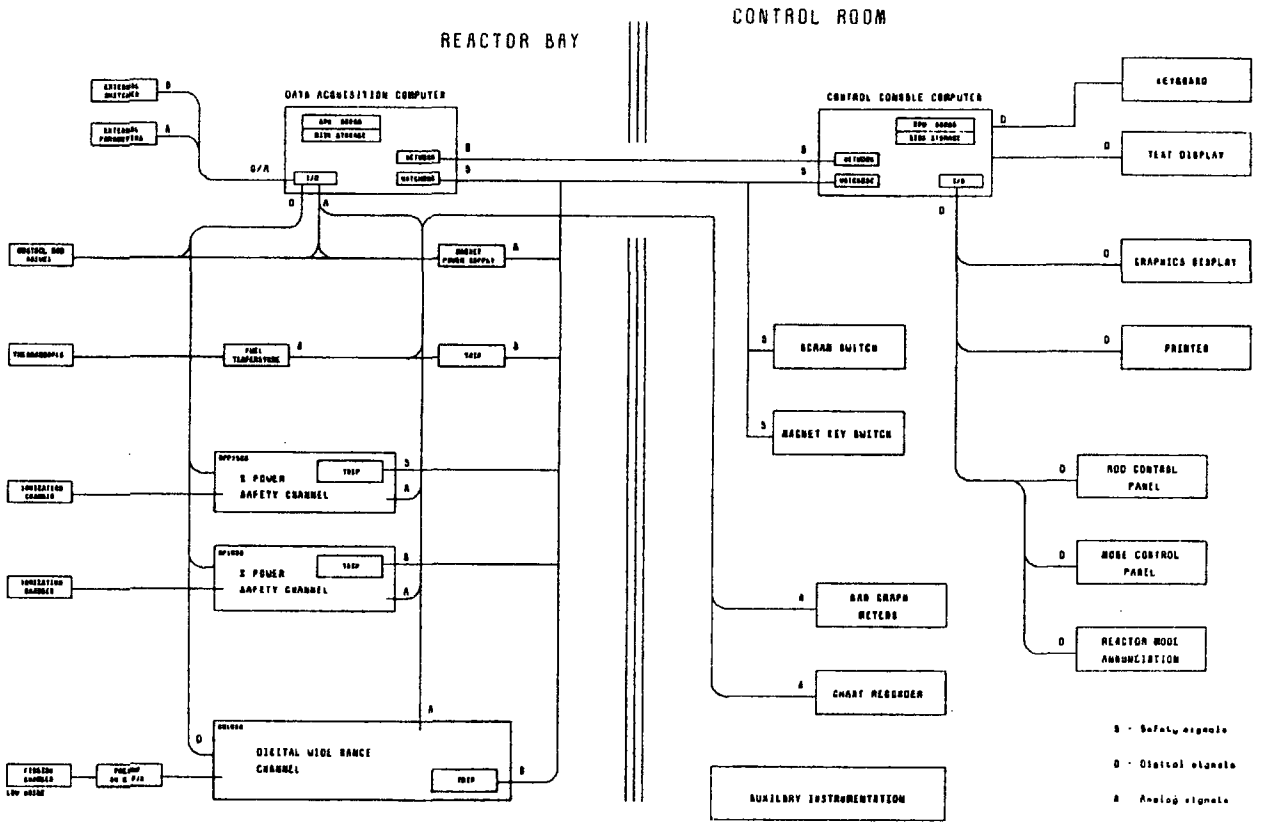


Figure 7.7, LOGIC DIAGRAM FOR CONTROL SYSTEM

7.1.7 Mechanical Hardware

Typical reactor installation are contained in two NEMA enclosure junction boxes, one electronic equipment cabinet, separate stepping motor power supplies installed in the reactor bay, and reactor operator console components installed in the reactor control room.

The control console consists of the components needed by the operator for reactor control. These components include rod control switches and annunciators, the digital rod position indicators, on-line reactor status meters (power and temperature), the control system computer (CSC), reactor operating mode switch panel, LED monitors, printer, disc drives (2) and external switch annunciators (beam port open-close, reactor access, etc.).

Enclosure 1 contains NM-1000 high and low voltage power supplies, a pulse pre-amp with discriminator, an RMS Campbell convertor and a communications module.

Enclosure 2 contains the NM-1000 microprocessor selected to provide the 10-decade log signal and the multi-range linear function from the information provided by the circuits in enclosure 1. The information processed by the microprocessor is 10-decades of log power, rate of power change (period), multi-range linear function, linear percent power from 1 to 125%, level trips from the log and linear percent power, calibrate and failure signals.

Enclosure 3 is a standard rack type equipment enclosure for electronic components. Space in the enclosure provides the terminal strips for connections to the various signal detection systems and the communications to the RCC. The cabinet enclosure includes eight shelves with functional separation between shelves. Power supplies for subsystems are on shelf 1. Shelves 2 and 3 contain, respectively, ac digital and dc digital circuits for processing input or output circuits. Shelf 4 provides several special modules for signal processing. The two power safety channels are positioned on shelf 5. Shelves 6-8 contain the computer. The regulating rod drive translator for the stepping motor drives is contained in a separate, fourth enclosure.

7.2 DESIGN EVALUATION

The TRIGA reactor console[5,6,7] [6,7,8] has developed through the successful operation of many installed facilities throughout the world. Design of the ICS unit incorporates similar basic logic functions proven effective in prior designs. Incorporation of digital electronic techniques in the design to replace analogue circuits is justified by improved performance. Functional self-checks, circuit calibrations, and automated data logging are implemented effectively and efficiently.

A multiphase design, development and installation program by the system manufacturer provided the initial demonstration of the system acceptance by analysis and review. No license modification was found necessary to implement the new digital system in place of the old analog system. The analysis by both the manufacturer and during operation on site determined 1) that there was no increase of the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, 2) that the system does not create the possibility of an accident or malfunction of a different type and 3) no reduction occurs in the margin of safety as defined in the basis for any technical specification.

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[5] "Operation and Maintenance Manual Microprocessor Based Instrumentation System for the University of TRIGA Texas Reactor", E117-1004, General Atomics 1989.

[6] "Operation and Maintenance Manual NM1000 Neutron Monitoring Channel", E117-1000, General Atomics 1989

[7] "Operation and Maintenance Manual NP1000/NPP1000 Percent Power Channel", E117-1010, General Atomics 1989.

8.0 ELECTRIC POWER SYSTEMS

Electric power on the Pickle Research Camus is distributed underground.

The main breaker for the NETL is 3 phase, 480 volts AC (with a 277 tap) rated at 600 amps per phase. 480 VAC power is supplied to:

- HVAC Fans
- Chill water pumps
- Pool cooling pump
- Laboratory vacuum pump
- Laboratory air compressor
- Instrument air compressor
- Crane
- Elevator

277 VAC power is supplied to the reactor bay lighting transformer.

Motor control center and load control center panels are located in a machine room adjacent to the reactor bay on the middle level and upper levels.

An emergency diesel generator operated and maintained by the facilities maintenance on the PRC provides backup power for lighting and sump pumps.

The reactor safety and control systems are failsafe, in that a power supply failure causes the reactor to shutdown. The underground distribution system prevents the potential for most external events affecting the power supply, with exceptions that damage the distribution station.

9.0 AUXILIARY SYSTEMS

9.1 Confinement System

The design of a structure to contain the TRIGA reactor depends on the protection requirements for the fuel elements and the control of exposures to radioactive materials. Fuel elements and other special nuclear materials are protected by physical confinement and surveillance.

The floor of the reactor bay is approximately [redacted]. The lower walls of the reactor bay are cast in place concrete. Above grade, the walls are reinforced, approximately [redacted] precast concrete tilt panels with integral columns and embedded reinforcing steel. The wall panels were then set in place vertically using a crane with space left in between each panel for a structural column and temporarily braced. Next the column forms were placed around reinforcing steel extending from the edges of the panels which was interlaced with additional steel reinforcing internal to the columns. Concrete was then poured into these forms resulting in a finished wall system with columns that resemble a poured in place design rather than the typical tilt panel welded design. [redacted] The roof is sealed using standard tar and gravel techniques. All penetrations in the reactor bay confinement envelope are on the south side, interfacing with the reactor wing offices, machine room spaces, equipment staging area, and confinement (and auxiliary purge) ventilation system.

9.2 HVAC (Normal Operations)

Building environment controls use air handling units for ventilation and comfort with cold and hot water coils for temperature and humidity control. There are two separate HVAC systems with three air handling units, located on the fourth level of the reactor bay wing adjacent to the reactor bay. One unit contains both cold and hot water coils in a single duct system, dedicated to the reactor bay. This system supports confinement functions. The other two units are the cold- and hot-deck components of a double duct system that conditions air in all building zones other than the reactor bay.

Water temperatures of the heating and cooling coils in the air handling units are controlled by set of on-site and off-site systems. The heating system is an on-site boiler unit with a design capacity set by local building (HVAC) requirements. The cooling system is a PRC chilled water treatment plant with design capacity set by overall research campus requirements, with thermostats controlling zone or room temperatures. A local instrument air system provides control air for HVAC systems. Controls and air balancing of the two air handling systems provide user comfort and pressure differentials between the reactor bay (confinement) and adjacent zones, and between the adjacent zones and the academic wing of the building.

The ventilation system is designed to maintain a series of negative pressure gradients with respect to the building exterior and other building areas, with the reactor bay (confinement) at the lowest pressure. Confinement functions of ventilation control the buildup of radioactive materials generated as a byproduct of reactor operations, and isolate the reactor bay in the event that an abnormal release is detected in the reactor areas. Confinement and isolation is achieved by air control dampers and leakage prevention material at doors and other room penetration points.

A conceptual diagram of the system is provided in Fig. 9.1. Manual operation controls for both main and purge air systems are in the reactor control room.

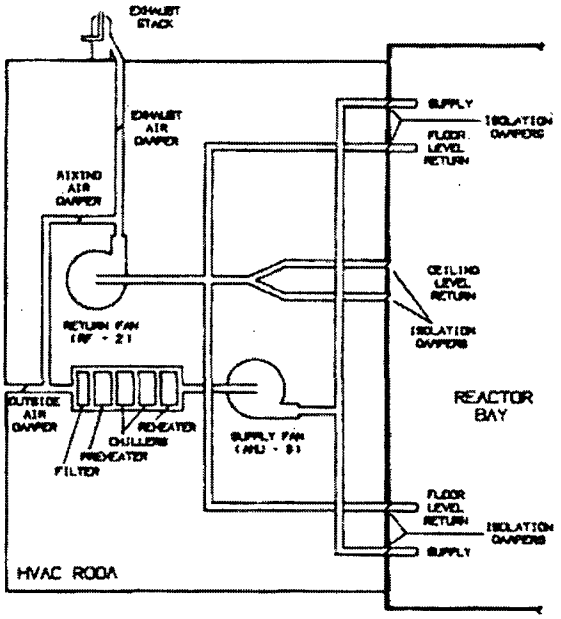

Figure 9.1, Conceptual Diagram of the Reactor Bay HVAC System

An exhaust stack on the roof combines the ventilation exhausts from both the main and the purge systems. As illustrated in Fig. 9.1, the auxiliary purge system discharge is within the HVAC exhaust stack. The auxiliary purge exhaust is a 6 in. (15.24 cm) internal ID and 8.63 in. (21.92 cm) OD. The HVAC exhaust has an 18 in. (45.72 cm).

9.2.1 Design basis

The design goal for HVAC system is to control the reactor bay, adjacent zones and academic wing of the building at a negative pressure difference relative to ambient atmospheric pressure during routine operations. The differential pressures are 0.06: 0.04: 0.03 in. water (0.15: 0.10: 0.80 cm of water). This pressure gradient assures that any radioactive material released during routine operations is discharged through the stack and does not build up in the reactor bay. Release of airborne radioactivity consists mostly of activated ^{41}Ar from routine operation.

During potential accident conditions, sensors initiate confinement system isolation when high levels of radioactivity are detected in reactor bay air, e.g. If a fuel element failure releases fission products or if an experiment with sufficient inventory of radioactive material fails. The reactor room confinement is designed to control the exposure of operation personnel and the public from radioactive material or its release caused by reactor operation. Release criterion is based on Title 10 Chapter 20 of the U.S. Code of Federal Regulations.

Confinement system ventilation has three modes of operation. When the reactor is not operating (quiescent mode), the ventilation system is operated to minimize requirements for conditioning incoming air, in a recirculation mode with a minimal exhaust flow rate and fresh air intake as required to maintain a negative pressure in the reactor bay with respect to adjoining spaces. When the reactor is operating (reactor run mode) the system is operated to generate a rate of air exchange exceeding 2 air volumes (4120 m^3) per hour, maintain a stack velocity, and regulate negative pressure in the reactor bay. In the event that airborne radioactive material exceeding a trip set point is detected, the system is designed to establish a shutdown and isolated condition.

9.2.2 System description

During operating modes supply fans draw air from either the return fan or the environment into a conditioning unit that subcools the air to control humidity then heats the air for habitability/comfort. Air filtration is the typical design for normal HVAC operation with fiberglass roughing filters only. The confinement system uses heating and cooling in a single unit, the remainder of the building HVAC system has air conditioning split into separate hot and cold decks.

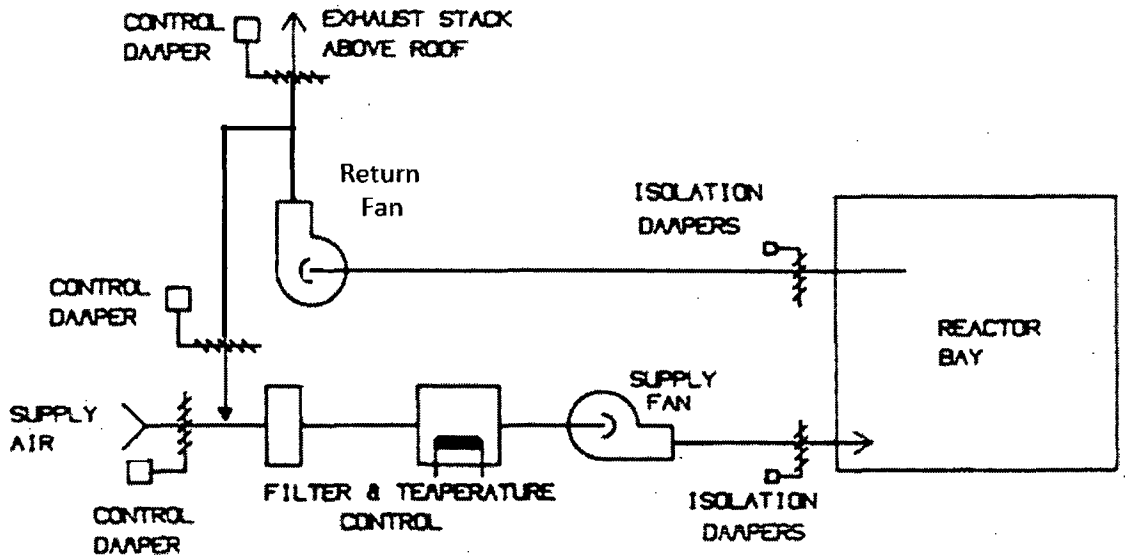

Figure 9.2A, Main Reactor Bay HVAC System

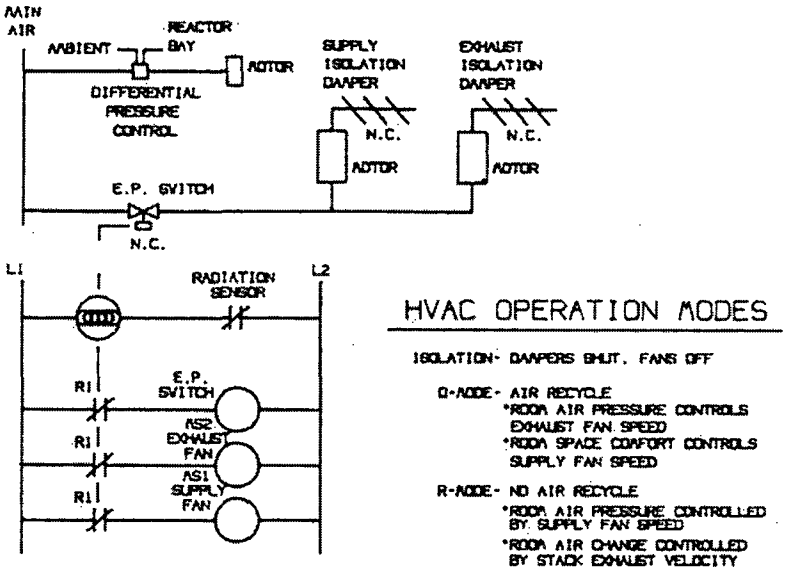

Figure 9.2B, Main Reactor Bay HVAC Control System Control

Table 9.1, Typical Confinement Vent & Purge Parameters

	Duct Velocity				Exit Velocity	
Aux Purge	3900	fpm	20	m/s	35.23	m/s
Confinement Vent	1800	fpm	9	m/s	26.87	m/s
	Flow Rate					
Aux Purge	1100	cfm	0.52	m^3/s		
Confinement Vent	7200	cfm	3.40	m^3/s		

9.2.3 Operational analysis and safety function

Speed of the confinement system supply fan is regulated to produce 0.06 in. water vacuum in the reactor bay by differential pressure control between the reactor bay and a representative ambient external building measurement point. Additional measurement points in ventilation zones adjacent to the reactor bay are used to maintain differential pressure between the reactor bay and adjacent access areas. Supply air is distributed through a rectangular duct near the ceiling and then to distributed ducts down the wall opening near the floor, enhancing mixing and preventing stratification. Air is discharged in parallel duct work (near the ceiling, near the floor) to an exhaust fan. In the reactor run mode the confinement system exhaust fan is controlled to maintain stack velocity designed to exceed the minimum air change specification. Control dampers are located at the supply fan inlet (fresh air intake) and the exhaust fan outlet (discharge to stack), and in a line between the inlet and outlet ducts. Confinement system ventilation discharge is through a stack on the reactor building roof. Schematics of the ventilation system for the reactor bay area and a logic diagram of the ventilation control system sensors and controls are provided in Fig. 9.2A and B.

In the reactor run mode, confinement ventilation is balanced to provide at least reactor-bay 2 air changes per hour and a preset stack flow rate:

(1) One control damper (inlet/outlet duct cross connect) is closed
(2) Supply and exhaust control dampers are open
(3) The exhaust fan is controlled to provide a specified stack velocity
(4) The supply fan is controlled to maintain the reactor bay at nominal 0.06 in. water

In the quiescent mode, the confinement ventilation system is balanced for recirculation flow with a small amount of effluent:

(1) One control damper is throttled open across connecting inlet and outlet ducts,
(2) Inlet and outlet control dampers are set to a minimum open position,
(3) The outlet fan is operated at a constant, minimal speed
(4) The supply fan is controlled to maintain the reactor bay at nominal 0.06 in. water.

In confinement isolation mode:

(1) All isolation dampers are closed
(2) Supply and exhaust fans are secured

Atmospheric dispersion using a stack model requires stack discharge 60 (18.23 m) feet above the ground, and at least 2 and ½ times the height of adjacent structures. The nearest structure is approximately 80 meters from the reactor bay. Ground elevation in the area is 794 feet, with roof elevation at the stack 843 feet, a distance of 49 feet (14.94 m) above grade. The exhaust stack extends 14 feet (4.24 meters) above the roof level so that the stack discharge is 63 feet (19.202 m). The effective release point above the exhaust stack can be calculated from the Bryan - Davidson equation:

$$\Delta h = D \frac{(V_s)^{1.4}}{\bar{\mu}}$$

Where:

Δh is the height of plume rise above release point (m)
D is the diameter of stack (m), confinement vent 0.4012 m^2, auxiliary purge 0.152 m^2
$\bar{\mu}$ is the mean wind speed at stack heght (m/s)
V_s is the effluent vertical efflux velocity (m/s), confinement vent 26.87 m/s, purge 35.23 m/s

The effective stack height for the reactor HVAC confinement vent system (in units of meters) is therefore 40.19/{wind velocity} m above the stack, and the effective stack height for the auxiliary purge system is 22.25/{wind velocity} above the top of the stack at 63 feet (19.202 m). Mixing of the two effluent streams occurs at the exit of the stack.

Pneumatically operated isolation dampers in the confinement system ventilation are located at the supply fan outlet (supply to the reactor bay) and the exhaust fan inlets (return from the reactor bay) near the reactor bay wall penetrations as indicated in Fig. 9.1. Controls close the dampers and secure the fans in response to manual or automatic signal initiated by high airborne particulate radioactivity. Loss of instrument air or loss of control power will cause the dampers to isolate the reactor bay.

9.2.4 Instruments and controls

As indicated, the HAVC control system is controlled by a set of temperature, flow, and differential pressure sensors that develop control signals. The signals are used in variable frequency controllers that regulate fan speed to maintain pressure and temperature.

Control room switches establish the operating mode of the confinement ventilation system. The auxiliary purge system is controlled from the same panel.

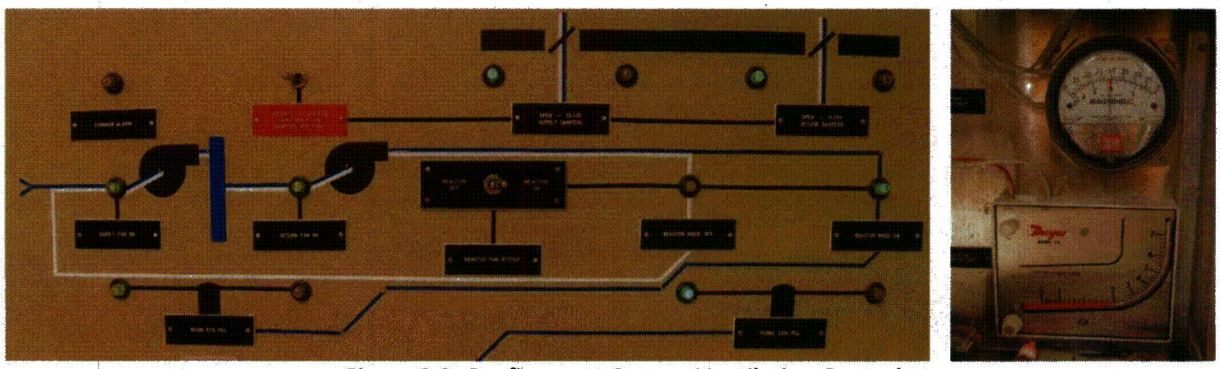

Figure 9.3, Confinement System Ventilation Controls

Alarm indicators on the control panel provide indication that the differential pressures are normal or abnormal. Flow and differential pressure indicators inside the panel provide indication of the zone static pressure, and confinement system and auxiliary purge system velocities.

A continuous air particulate detector located in the reactor bay provides a control signal to initiate confinement isolation when the count rate exceeds a preset level. Indicators at the reactor control console provide alarm level information. A count rate associated with 2,000 pCi/ml detects particulate activity at occupational levels of 10CFR20. The alarm setpoint exceeds the occupational values for any single fission product radionuclide in the ranges of 84-105 and 129-149. Seventy per cent of the particulate radionuclides are also detectable at the reference concentrations within two hours.

9.2.5 Technical Specifications, bases, testing and surveillances

Either the confinement ventilation system or the auxiliary purge system is required to be operating when the reactor is operating to control the buildup of gaseous radioactive material

in the reactor bay. If the confinement ventilation system is operating, instrumentation to initiate confinement isolation on high airborne contamination levels will be operable. The confinement system will be checked periodically to assure proper function. The particulate monitor will be calibrated periodically.

9.3 Auxiliary Purge System

A separate, low volume air purge system is designed to exhaust air that may contain radionuclide products from strategic locations in the reactor bay.

9.3.1 Design basis

The purge system collects and exhausts air from potential sources of neutron activation such as beam tubes, sample transfer systems, rotary specimen rack, and material evolving from the surface of the pool. The purge system filters air in the system through a rough prefilters followed by a high efficiency particulate filter. Design provisions allow for the addition of charcoal filters if experiment conditions or other situations should require the additional protection.

9.3.2 System description

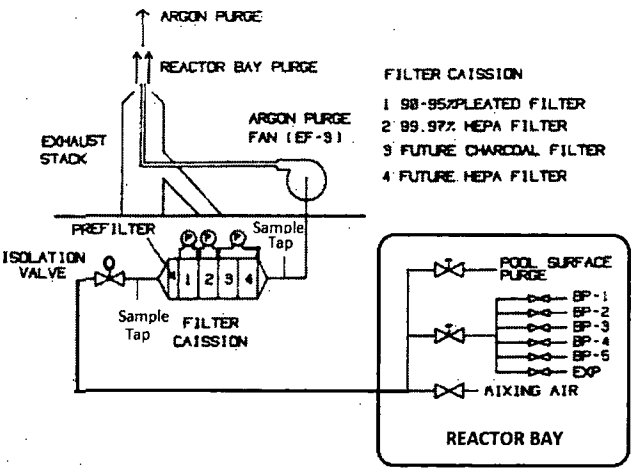

Figure 9.4A, Purge Air System

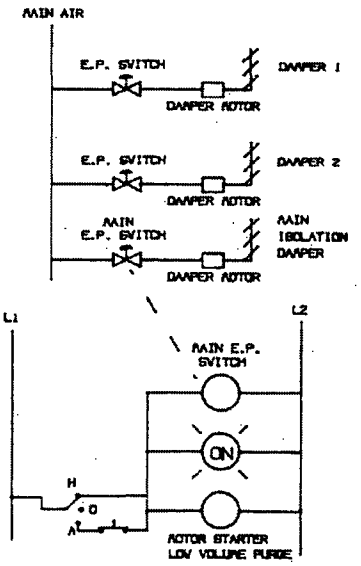

Figure 9.4B, Purge Air Controls

9.3.3 Operational Analysis and Safety Function

The primary nuclide of interest is argon-41. Fig. 9.4A and 9.4B are schematics of the auxiliary purge system and its control logic. Sample ports in the turbulent flow stream of the purge system exhaust provide for measurement of exhaust activities. The isolation damper in the purge system is actuated manually, using the fan control switch. Automatic isolation of the

system is generated by the same particulate radiation monitor as is used by the HVAC confinement ventilation system.

9.3.4 Instruments and controls

The auxiliary purge system is controlled from the same panel as the confinement ventilation system. A flow gage indicates purge stack velocity at the panel. The exhaust point is concentric to the center of the HVAC confinement ventilation exhaust stack.

The auxiliary purge system is monitored by a gaseous effluent radiation detector. The effluent monitor has an alarm setpoint based on ten times the occupational limit or a reference concentration at the ground.

9.3.5 Technical Specifications, bases, testing and surveillances

If the auxiliary purge system is operating, a gaseous effluent monitor will be operating. The auxiliary purge system will have a high efficiency particulate filter. Auxiliary air purge system valve alignment will be checked periodically. The gaseous effluent monitor will be calibrated periodically.

9.4 Fuel storage and handling

Special provisions are necessary for the storage of fuel elements that are not in the core assembly. The design of fuel storage systems requires consideration of the geometry, cooling, shielding, and the ability to account for each of the fuel elements. These storage systems are specially designed racks inside the reactor pool and outside the reactor shield.

Irradiated fuel is manipulated remotely, using a standard TRIGA fuel tool. Irradiated fuel is transferred out of the pool using a transfer cask modeled on the BMI cask TRIGA basket. There are two different loading templates for use with the transfer cask, permitting loading operation either for a single TRIGA fuel element, or to up to three elements. A 5-ton overhead crane is used to move the fuel transfer cask.

9.4.1 Design basis

Stored fuel elements are required to have an effective multiplication factor of less than 0.9 for all conditions of moderation. Fuel handling systems and equipment are designed to allow remote operation of irradiated fuel, thus minimizing personnel exposure.

9.4.2 System description

There is space for a large number of fuel racks inside the reactor pool. The racks are aluminum, suspended from the pool edge by connecting rods. [redacted]

[redacted] To facilitate extra storage, 2 racks may be attached to the same connecting rods by locating one rack at a different vertical level and offsetting the horizontal position slightly. [redacted]

[redacted]

Outside the reactor pool, rack design is intended to fit in special storage wells (Fig. 9.4). [redacted] with [redacted] Water may be added for shielding or cooling. [redacted]

[redacted] Outside the reactor pool, supplemental fuel storage is planned for temporary storage of elements transferred to or from the facility, for isolation of fuel elements with clad damage, emergency storage of elements from the reactor pool and core assembly and routine storage of other radioactive materials. Temporary storage for some reactor components or experiments may also use the fuel storage racks in the reactor pool. Other locations not in the pool will also provide storage for radioactive non-fuel materials.

A fuel transfer cask modeled after the BMI cask TRIGA basket is used to transfer irradiated fuel. A standard TRIGA fuel handling tool is used to remotely grapple irradiated fuel elements

A 5-ton crane is used in conjunction with the fuel handling tool and the transfer cask to allow remote handling of irradiated fuel.

9.4.3 Operational analysis and safety function

Bench mark experiments conducted by TRIGA International indicate minimum mass for criticality requires 64 fuel elements in a favorable geometry.

Pool storage racks do not have the capacity or the geometry to support criticality. Spent fuel storage has a higher fuel density in storage, but does not have the capacity to hold 64 fuel elements, and does not have favorable geometry.

The fuel handling tool has been used successfully at the UT TRIGA reactor, including the original reactor on the main campus as well as the current installation. This design is widely used by TRGIA reactors, with good performance history although the first generation tool occasionally released an element if pressure was not maintained on the tool operator.

The fuel transfer cask is a top loading cask, with no potential for failure or mishandling as exists in a bottom loading cask. The cask does not provide adequate shielding for close-in work, and all handling is conducted remotely.

The crane exceeds load requirements for spent fuel handling by a large margin. There is little potential for failure

9.4.4 Instruments and controls

New fuel storage is in a locked room on the middle level of the reactor bay. A criticality monitor is installed, with neutron and gamma channels. The system has a local indicator directly outside the storage room, and a remote readout in the control room.

9.4.5 Technical Specifications, bases, testing and surveillances

Fuel elements are required to be stored in a configuration with k_{eff} less than 0.8. Irradiated fuel is required to be stored in a configuration where convective cooling by water or air is adequate to maintain temperature below the safety limit.

9.5 Fire protection systems

Active fire protection elements generally have automatic operation, manual response, or personnel action for the intended function. Active elements to be considered include automatic fire detection, automatic fire suppression in labs and office spaces, fire information transmission, manual fire suppression and other manual fire control.

Passive fire protection provides fire safety that does not require physical operation or personal response to achieve the intended function. Passive elements include inherent design features, building physical layout, safety-related systems layout, fire barriers, and construction or component materials, and drainage for control of fire protection runoff water. Penetrations in fire barriers have fire resistant ratings compatible with the purpose of the fire barrier.

9.5.1 Design basis

The goal of fire protection is to provide reasonable assurance that safety-related systems perform as intended and that other defined loss criteria are met[1,2]. For the purpose of fire protection, loss criteria should include protection of safety-related systems, prevention of radioactive releases, personnel protection, minimization of property damage, and maintenance of operation continuity. Three components shall be applied to the fire protection objective. The three components are passive and active fire protection, and fire prevention.

A fire detection, suppression, and information management system is designed to ensure that fires can be detected, suppressed (where possible), and alert response organizations.

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[1] Code of Federal Regulations, Chapter 10 part 20, U.S. Government Printing Office, 1982.
[2] Dorsey, N.E., Properties of Ordinary Water-Substance, Reinhold Publishing Corp., New York p. 537.

Basic design features of the reactor assembly, pool and shield system, and the instrumentation, control, and safety system represent passive fire protection elements. These basic features are sufficient passive protection to protect safety-related systems.

9.5.2 System description

Manual protection consists of manual firefighting actions and the systems necessary to support those actions such as extinguishers, pumps, valves, hoses, and the inspection, maintenance and testing of equipment to assure reliability and proper operation. Other manual actions that are elements of active fire protection include utility control, personnel control, and evacuation. Preplanning and training by facility and emergency personnel ensures awareness of appropriate actions in fire response and possible hazards involved.

Automatic and manual protection systems in the building include several different type systems. In the academic wing of the building automatic protective actions are provided by a sprinkler system with heat sensitive discharge nozzles, detectors for heat and smoke, and ventilation systems dampers. The reactor bay wing uses smoke detectors for areas outside the reactor bay that are radiation areas. The reactor bay ventilation system has smoke detectors that provide a warning of problems within the reactor bay. Although not a strict safety requirement, a gaseous extinguisher system (halon) is installed to protect the reactor instrumentation, control and safety system.

Manual protection equipment includes a dry stand pipe system in each building stairwell. Portable extinguishers such as CO_2, halon and dry chemical are placed in specific locations throughout the building.

Elements of the passive fire protection include the structural construction system and the architectural separation into two separate buildings. Building structural materials are concrete cast in place for foundation, concrete walls, support columns and roof. Steel beam, metal and concrete deck comprises the reactor bay roof. A built-up composition roof with fire barrier materials completes the roof system. The building has pre-cast panels that are cast at the construction site cover 75% of the external perimeter. Metal paneling covers the other 25% of the perimeter. Design and installation of systems and components are subject to the applicable building codes.

The common wall between the academic wing and the reactor bay wing of the building is a fire barrier. Doors between these two building sections and other penetrations such as HVAC chases will conform to applicable codes. Although a few metal stud and plaster board walls have been used in the reactor bay wing, the typical wall system is of concrete block construction.

Design specifications are to meet life-safety requirements appropriate for the conditions. These specifications have requirements for emergency lighting, stairwells and railings, exit doors, and

other building safety features. An emergency shower and eye wash are available in the hallway adjacent to laboratory areas.

Each of the three components of the fire protection program is applied to the design, operation and modification of the reactor facility and components. Fire prevention is primarily a function of operation rather than design.

9.5.3 Operational analysis and safety function

The University of Texas maintains an active fire protection system, with periodic testing and inspections to assure systems are prepared to respond.

The halon system automatically actuates if detectors in two control room ceiling units sense an initiating condition in coincidence. The halon system is equipped with a local alarm to prompt evacuation of the control room prior to system actuation; a manual override can defeat the system if the nature of the event does not require actuation of the system.

Fire suppression is used only in areas where there are no significant quantities of radioactive materials or criticality concerns.

9.5.4 Instruments and controls

A fire alarm panel transmits status and alarm information to the University of Texas Police Department dispatch station and a campus information network monitor.

9.5.5 Technical Specifications, bases, testing and surveillances

There are no Technical Specifications associated with fire protection.

9.5 Communications systems

A communication system of typical telephone equipment provides basic services between the building and other off-site points. Supplemental features to this system, such as intercom lines between terminals or points within the building and zone speakers for general announcements are to provide additional communication within the building.

9.5.1 Design basis

Communications is required to support routine and emergency operations.

9.5.2 System description

The telephone system is installed and maintained by the university. Connection of the main university telephone system is to standard commercial telephone network. Telephones with intercom features are to be located at several locations in the building. Locations include the reactor control room, the reactor bay, and several offices. By use of the intercom feature, each of these locations will be able to access public address speakers in one of several building zones.

A video camera system and a separate intercom system supplement the building telecommunication network. These two systems contribute to safe operation by enhancement of visual and audio communication between the operator and an experimenter. Each system has a central station in the control room with other remote stations in experiment areas.

A public address system allows personnel to direct emergency actions or summon help, as required. A building evacuation alarm system prompts personnel to initiate protective actions. An emergency cell phone is maintained in the control room to compensate for loss of normal communications. A digital radio is kept in the control room to provide emergency communications on first responder and campus frequencies, and to compensate for loss of normal communications.

9.5.3 Operational analysis and safety function

The control room has adequate capabilities to initiate and coordinate emergency response. There are multiple provisions specifically to address failures on normal communications channels.

9.5.4 Instruments and controls

As specified above

9.5.5 Technical Specifications, bases, testing and surveillances

There are no specific Technical Specifications related to communications, but the reactor Emergency Plan specifies communications as indicated above.

9.6 Control, storage, use of byproduct material (including labs)

Experimental facilities in the reactor building include a room with 4' thick walls supporting irradiation programs and a series of laboratories in the lab and office wing.

9.6.1 Design basis

The design basis of the NETL laboratories is to allow the safe and controlled use of radioactive materials.

9.6.2 System description (drawings, tables)

Strategic lab and office wing rooms are equipped with fume hoods and ventilation to control the potential for release of radioactive materials. One room is equipped with two pneumatic transfer systems and a manual port. One system terminates in a fume hood, monitored by a radiation detector. The other system delivers samples within the tube to a detector. The manual port allows samples to be transferred from the reactor bay to the lab without exiting the reactor bay through normal passageways. A more complete description of the associated laboratories is provided in Chapter 10.

9.6.3 Operational analysis and safety function

Engineered controls permit safe handling of radioactive materials.

9.6.4 Instruments and controls

An installed radiation monitor ensures personnel handling samples from the manual pneumatic sample transfer system are aware of the potential exposure.

9.6.5 Technical Specifications, bases, testing and surveillances

There are no specific Technical Specifications related to the laboratories; all operations involved with potential radiation exposure at NETL are managed under the approved reactor Radiation Protection Program.

9.7 Control and storage of reusable components

Several experiment facilities that are used in the core are designed to be removed and inserted as required to support various programs.

9.7.1 Design basis

Management of experiment facilities is designed to minimize potential exposure to personnel.

9.7.2 System description

The 3 element facility, 6 element facility, pneumatic in-core terminals, and central thimble are described in chapter 10. Once irradiated, these facilities are maintained with activated portions in the pool, using pool water as shielding or in other locations typically within the reactor bay

9.7.3 Operational analysis and safety function

Maintaining irradiated facilities under water minimizes potential exposure. Concrete blocks provide temporary shielding as needed.

9.7.4 Instruments and controls

Instruments and controls associated with specific facilities are addressed in Chapter 10.

9.7.5 Technical Specifications, bases, testing and surveillances

The basis for Technical Specifications specific to the pool is in Chapter 5, the basis for experiment in Chapter 10.

9.8 Compressed gas systems

There are two separate compressed air systems use at the UT facility. One system provides air for laboratories and service connections. One system provides control air.

9.8.1 Design basis

Service air is provided to support laboratory and service operations with high capacity applications. Instrument air is intended to support HVAC and reactor operations.

9.8.2 System description

One dual compressor system provides oil free compressed air for laboratories and services. The lab air compressor motor is rated at 30 hp. The other system also uses a dual compressor and motor, with 2-stage compressors. The instrument air compressor provides air to HAVC pneumatic controls, pool cooling flow controls. The laboratory air compressor provides aiur to shops and to the transient rod drive system.

9.8.3 Operational analysis and safety function

The two systems have dual motors and compressors to provide maximum reliability. The two systems are connected through a manual shut off valve, providing maximum flexibility in the event of a system (or associated air dryer) failure.

Failure of the instrument air system will prevent air from supporting control systems. The pulse rod drive system requires air to couple the drive to the rod; a failure will cause the rod to fall into the core. This is a fail-safe condition, causing negative reactivity to be inserted in the core.

Instrument air failure will cause chill water flow control valves to shut, stopping pool cooling. This is a fail-safe condition that prevents potential leakage from the pool to the chill water system. Other operational aspects of this type of event are addressed in Chapter 13.

Instrument air failure will cause isolation dampers in the confinement ventilation system to fail closed, initiating confinement isolation. This is a fail-safe condition, assuring that there is no potential for inadvertent release of radioactive material into the environment in the absence of instrument air.

9.8.4 Instruments and controls

The air compressors and their associated moisture reduction systems are locally controlled. The compressors and air dryers have operating indicators.

9.8.5 Technical Specifications, bases, testing and surveillances

There are no Technical Specifications specifically associated with the compressed air systems.

10.0 EXPERIMENTAL FACILTIES AND UTILIZATION

10.1 Summary Description

The Nuclear Engineering Teaching Laboratory (NETL) experimental facilities support teaching, research, and service work. Multiple courses are taught at NETL that focus on reactor operations, radiation detection, radiochemistry, and health physics. With the reactor facility being the center focus of NETL, many of the nuclear analytical techniques utilize neutrons for materials probing or activation. Isotope production is performed largely for detector calibrations and specialized experiments. In-core experimental facilities are used mostly for activations for neutron activation analysis and for detector calibration related isotope production. Beam port facilities utilize neutrons for either activation or imaging. Laboratory facilities are utilized for radiation detection and measurement along with radiochemistry. The neutron generator facility contains a D-T neutron generator utilized for neutron activation studies. The subcritical assembly is utilized for teaching and neutron source based experiments. The UT Austin TRIGA does not have thermal columns or irradiation rooms associated with the reactor.

List of experimental facilities

1. In core facilities
    a. Central Thimble
    b. Fuel element positions
    c. Pneumatic transfer systems
    d. Three element Facility
    e. 6/7-Element Facility
2. In reflector facilities/Rotary Specimen Rack
3. Automatic transfer facilities
    a. Manual
    b. Automatic
4. Beam ports
5. Cold neutron source
6. Non-reactor experiment facilities
    a. Neutron generator room

b. Sub critical assembly

c. Laboratories

i. Radiochemistry laboratory

ii. Neutron activation analysis laboratory

iii. Radiation detection laboratory

iv. Sample preparation laboratory

v. General purpose laboratory

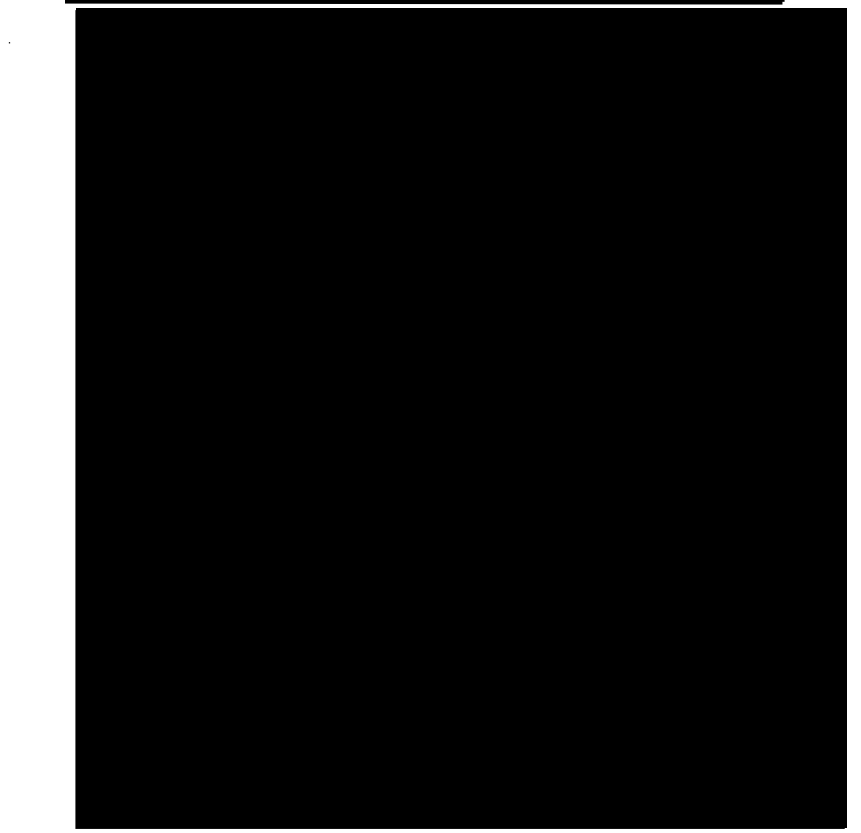
The facility runs experiments in three basic categories: 1) in core irradiations, 2) beam port experiments and non-reactor experiments. The majority of in core experiments are irradiations for neutron activation analysis. Other common in core experiments are irradiations to produce sources for detector calibrations and irradiations for either materials damage or electronics damage studies. Beam port experiments utilize the neutrons for various nuclear analytical techniques from neutron depth profiling to prompt gamma activation analysis to neutron radiography. Non-reactor experiments include those that utilize the D-T neutron generator or other radiation sources.

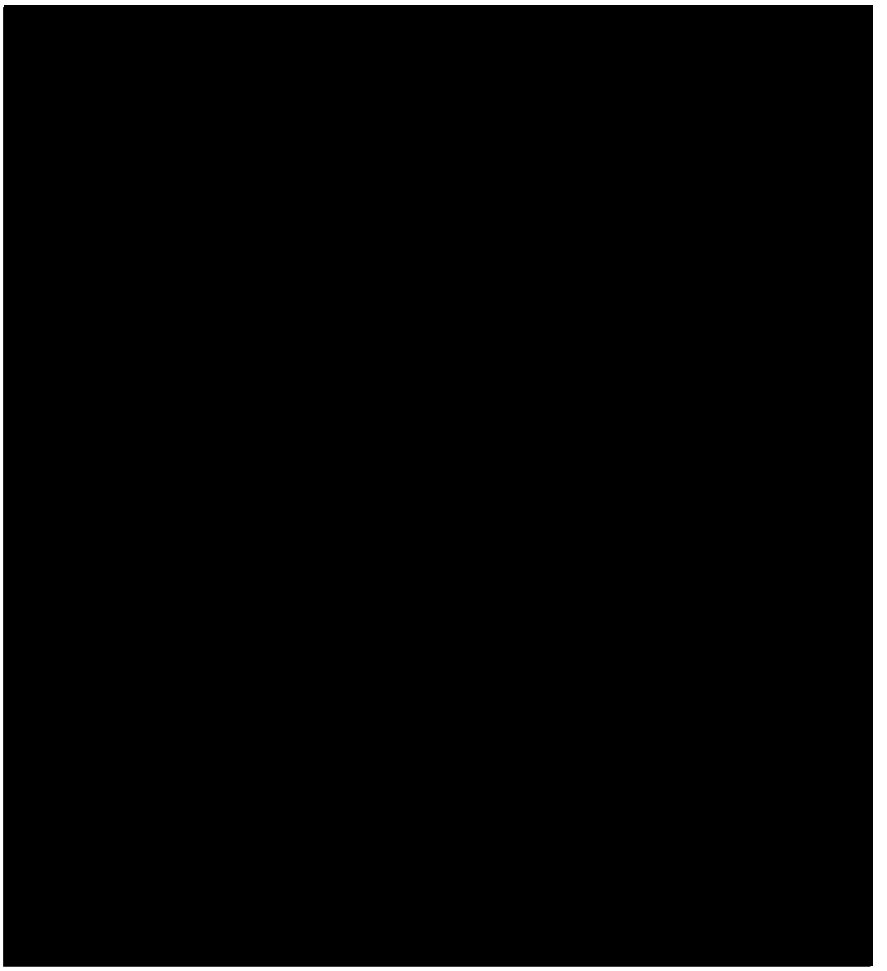
Experimenters work with licensed reactor operators for experiment planning, facility access, and facility utilization. Radiation monitors are placed near unloading points for in core experiments and near beam port facilities. Reactor operators watch neutron monitors adjacent to the reactor core monitor for reactivity perturbations resulting from in core experiments. An ^{41}Ar system monitors the activation of air within the core and beam ports. A continuous air monitor tracks radioactive aerosols that may be produced from experiments or fuel leakage. Access to beam port facilities is directly displayed on the reactor console.

Reactor based experiments and other experiments utilizing radiation sources such as the D-T neutron generator are reviewed by the Reactor Oversight Committee (ROC). A safety analysis is written by the experimenter and often presented in an oral format to the ROC. A ROC subcommittee is nominally formed to review the written safety analysis document. Evaluation criteria include but are not limited to a radiation exposure assessment, core reactivity effects, radiation levels produced, chemical nature of experiment, and heat transfer effects. The subcommittee members then make recommendations to the ROC Chair regarding approval, denial, or recommended changes to the experiment. After a positive review process, the experiment then becomes an approved experiment. Experimenters schedule reactor time utilizing Operation Requests that are reviewed by a senior reactor operator to ensure that the work is an approved experiment.

10.2 In-Core Facilities

In core irradiation facilities include a central thimble, penetrations for flux probes along two perpendicular axes, and four facilities that displace (3, 6, or 7) fuel elements. Cutouts in the upper grid plate accommodate removable plates that position fuel elements when the facilities are not in use. [redacted]





10.2.1 Central Thimble (In-Core Facility)

A. DESCRIPTION.

The central thimble provides access to the maximum neutron flux in the reactor. The central thimble has two modes, normal and beam operation.

Experiment objectives for normal operations maximize activation, gamma irradiation, or reactivity. The central thimble is used to enhance activation or radiation damage. Enhanced activation supports isotope production or neutron activation analysis, while enhanced radiation damage supports radiation damage studies. Experiments or research in reactor kinetics may be performed with the central thimble.

The design of the central thimble permits extraction of a neutron or gamma beam to the bridgework over the pool. Typical beam experiments such as radiography and prompt gamma analysis may be accomplished using the central thimble in the beam mode.

The central thimble consists of an aluminum tube extending through the core. The central thimble provides access to the maximum neutron flux available in the core for sample

irradiation or beam experiments. Samples are placed (normally in an aluminum canister) into the central thimble from the bridge. A threaded cap covers the top of the central thimble when the facility is not in use. Water can be displaced in the central thimble volume above the core with pressurized air to use the central thimble as a beam.

B DESIGN & SPECIFICATIONS

The central thimble is approximately 7.2 m long. The central thimble is assembled from three sections of tubing with the bottom tube sealed on the lower end. Sections are joined with water-tight aluminum or stainless steel connectors with the tube and a sealing sleeve joined and sealed on each side by a large aluminum nut. The bottom two sections (originally used at the UT TRIGA I reactor) are 10 ft. long (3.048 m).

The central thimble extends from the reactor bridge through the (radial) center of the core to approximately 7.5 in. (0.19 m) below the lower grid plate and 8.7 in. (0.22 m) above the safety plate. The central thimble tube outer diameter is 1.5 in. (3.81 cm.), with 1.33 in. (3.38 cm.) inner diameter. There are four ¼ in. (0.00635 m) holes in the central thimble approximately 3 in. (0.762 m) above the upper grid plate to ensure the volume in the core is maintained in a flooded condition. Figure 10.3 illustrates the central thimble union assembly.

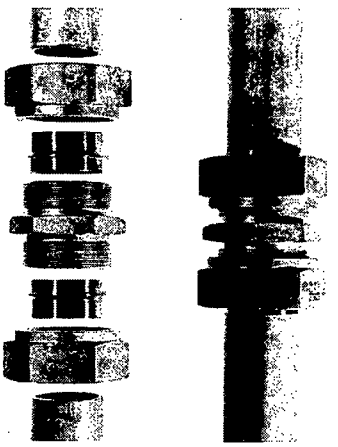


Figure 10.3: Central Thimble Union Assembly

The central thimble tubing is aluminum alloy 6061. The alloy is a precipitation hardening aluminum alloy, containing magnesium and silicon as its major alloying elements. It has good mechanical properties and exhibits good weldability.

The mechanical joint at the lower junction is prefabricated aluminum with a stainless steel sleeve. The upper mechanical joint may be either aluminum or stainless steel.

Aluminum 6061 is a widely used material in aircraft and structural applications. Typical density for 6061 alloy is 2.7 g cm^{-1}. Table 10.1 provides the material composition of Aluminum 6061.

Table 10.1: Composition of Al 6061

Component	Wt. %
Al	95.8 - 98.6
Cr	0.04 - 0.35
Cu	0.15 - 0.4
Fe	(Max) 0.7
Mg	0.8 - 1.2
Mn	(Max) 0.15
Si	0.4 - 0.8
Ti	(Max) 0.15
Zn	(Max) 0.25
Other, total	(Max) 0.15
Other, each	(Max) 0.05

The 6061 alloy has excellent joining characteristics, and good acceptance of applied coatings. The alloy combines relatively high strength, good workability, and high resistance to corrosion.

Aluminum 6061 has a high resistance to corrosion. The central thimble tubing is anodized to further control potential corrosion.

C. REACTIVITY

The original Safety Analysis Report for the UT at Austin TRIGA reactor provided data indicating that replacing the central thimble with a standard fuel element would result in a reactivity change of 0.90% Δk/k ($1.29), and that replacing the central thimble with a void would result in a reactivity change of -0.15% Δk/k (-$0.21). As noted above, voiding of the portion of the central thimble in the core region is prevented passively by design.

D. RADIOLOGICAL ASSESSMENT

Activation of argon dissolved in water will occur in the central thimble region whether the central thimble is installed or not. Radioargon is considered as a normal byproduct of reactor operation. Calculation of argon production and the consequences from normal operations is considered in Chapter 11.

Portions of the central thimble in the core area will become activated, principally minor constituents of 6061 aluminum alloy. A conservative irradiation scheme of 60 years at 2X 10^{13} n cm^{-2} s^{-1} followed by a week of decay using nominal values of 0.7% Fe, 0.4% Cu, 0.35% Cr, and 0.25% Zn results in specific activities identified in Table 10.2.

Table 10.2: Activation Products in Central Thimble 6061 Aluminum Alloy after 60 Year Irradiation

Element	Target Isotope	Concentration	Isotope Produced	Half Life	Activity
Iron	Fe-54	392.1 μg/g	Fe-55	2.7 years	4.889 mCi
	Fe-58	21.78 μg/g	Fe-59	44.53 days	35.7 μCi
Copper	Cu-63	2.740 mg/g	Cu-64	12.7 hours	5.625 μCi
Chromium	Cr-50	146.2 μg/g	Cr-51	27.7 days	6.969 mCi
Zinc	Zn-64	1.187 mg/g	Zn-65	243.9 days	6.255 mCi

The central thimble is normally installed for all operations, and does not create any increased radiological hazards during operations unless the volume above the core is voided for beam experiments. If the central thimble is used in a beam experiment the experiment proposal, review, and approval process will evaluate the need for additional radiological controls.

Portions of the central thimble in the core area will become activated, principally minor constituents of 6061 aluminum alloy. Using values for activation previously calculated, dose rate from the 15 in. (0.381 m) of the tube adjacent to the active fuel region using a point source approximation is estimated approximately 150 mR h^{-1}. However the central thimble can be suspended in the reactor pool indefinitely or removed from the pool using a shielded container.

Radiological hazards associated with materials to be irradiated in the central thimble are evaluated as part of the experiment review and approval process.

E. INSTRUMENTATION

There is no instrumentation associated with the central thimble. Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process.

F. PHYSICAL RESTRAINTS, SHIELDS, OR BEAM CATCHERS

The central thimble facility is shielded by the same materials that shield the reactor core. These include water and concrete.

G. OPERATING CHARACTERISTICS

Isolation from the control rods prevents any potential interaction between the control rods and the central thimble. Maintaining the volume in the core flooded by passive means prevents large reactivity changes associated with voiding and flooding.

The central thimble in the core is a static volume, open to the pool only through the penetrations above the core. The penetrations in the central thimble tube above the upper grid plate core eliminate any possible impact on loss of cooling potential or consequences. Cooling for material in the central thimble is principally through conduction to the water in the core with some thermally induced circulation inside the central thimble.

H. SAFETY ASSESSMENT

The central thimble facility does not carry much risk during reactor operation. Reactivity changes could occur as a result of sample introductions. Such reactivity changes would have to be assessed on an individual basis as part of the experimental review process. However, maximum reactivity additions would not likely be more than a fuel element which would be within the realm of allowed reactivity for a fixed experiment. Negative reactivity changes may occur due to sample introduction into the facility or for water introduction into the facility when it is voided. Sample reactivity has to be assessed on an individual basis and must comply with Technical Specifications. Reactivity changes from water leakage into a voided central thimble facility were calculated to be -0.15% Δk/k (-$0.21) which would not appreciably affect reactor safety.

10.2.2 Fuel Element Positions (In-Core Facilities)

Fuel element positions can be used for in-core irradiation facilities including single fuel element positions and multielement positions incorporated in the grid plate design. Standard facilities used in fuel element positions include in-core terminals for a pneumatic sample transit system and two types of multielement-position irradiation facilities (displacing 3 fuel elements, 6 elements and the central thimble, or 7 elements). Proposals for any other in-core facilities or irradiation of materials in existing facilities are evaluated as part of the experiment review and approval process.

10.2.2.1 Pneumatic Sample Transit System

A. DESCRIPTION.

The pneumatic transit system is used to support neutron activation analysis and isotope production. Major components of the pneumatic sample transit system include:

- In-core terminus assemblies
- Receiver assemblies
- Blower-and-filter assembly
- Valve assembly

- Control assembly
- Specimen capsules

Three different in-core terminals are available for insertion into a core fuel position. Receiving stations are available in the reactor bay and in an adjacent laboratory (either in a fume hood or in an automated counting and analysis station); an additional sample line is available for development. Two capsule sizes are available, a large capsule with an internal volume of 25 cm^3, and a small capsule with an internal volume of 5 cm^3.

B. DESIGN & SPECIFICATIONS.

The system design is a modification to the original, standard General Atomics pneumatic terminal system. Transit lines connect unions at the reactor pool-side to a mechanical switch (used to select the receiving station). Samples can be loaded from and delivered to receiving stations in the reactor bay, a fume hood in 3.102, or a counting station in room 3.102. A line is installed for an additional receiving station, not currently developed; the mechanical switch selects the receiving station. Idle sample transit line unions are capped to prevent intrusion of foreign material into the system. Three in core terminals are available for use in core position G-34; the original, large capsule terminus, a small capsule terminus, and a cadmium lined small capsule terminus.

Sample movement between the loading port and core terminal is provided by a motor-blower assembly, and four valves for air flow direction control (components of the original GA PNT system). Gas flow is designed to recirculate within the system, with losses only at loading stations or system connection points. The large and small systems have separate sample transit lines with a single gas supply and return line supporting both the large and small sample transit systems. Air displacement by CO_2 gas reduces ^{41}Ar production in the system. An air filter in the flow system controls the amount of circulating particulates.

The specimen capsule or "rabbit" is made of polyethylene. The effective available space inside the capsule is 0.56 inch (14.2) in diameter by 3.95 inches (100 cm) in length giving a usable volume of 0.97 cubic inches (15.9 cm^3). The capsule is designed to pass freely in a tube with a curved section no smaller than 2 feet (61 cm) in radius and with an inside tube diameter no smaller than 1.08 inches.

Table 10.3 shows the pneumatic transit system dimensions. The A (Large) system is the original General Atomics pneumatic terminal system. The B(Small) system is the modified system.

Table 10.3 Characteristic Dimension of UT-TRIGA PTS.

Transport System	A (Large)	B (Small)
Terminal point OD	1.25	0.875
Terminal point ID	1.085	0.685
Terminal point tube Thickness	0.0825	0.095
Transport tube OD (aluminum)	1.25	0.875
Transport tube ID (aluminum)	1.12	0.745
Transport tube Thickness	0.065	0.065
Transport bends OD (polyethylene)	1.5	1
Transport bends ID (polyethylene)	1.25	0.75
Transport bends Thickness (polyethylene)	0.125	0.125
Polyethylene transport capsule:	0.985 d X 4.75 l	0.650 d x 2.1S l
Total transport tube length (feet)	90	90
Transit time (seconds)	6	6

One terminal is an aluminum 6061 alloy with the associated radioactive nuclides. The other terminal contains a cadmium liner (for thermal neutron filtering) in addition to the normal aluminum alloy radioactivity.

The pool assembly consists of irradiation terminal and transport tubes to the pool surface. Pool assembly components are made of aluminum (alloy 6061). Tube connections in the pool are nut and ferrule type (aluminum Weatherhead) to seal against water leakage. The standard installation of the GA PNT design consists of aluminum and polyethylene tube. Straight transport sections are 1.25 inch diameter (OD) aluminum (6061) tube. Transport bends are 1.5 inch diameter (OD) polyethylene tubing with two-foot radius curves. Tube connections at the load port in the fume hood are also nut and ferrule type (stainless steel Swedgelock). Tube connections between aluminum and polyethylene transport sections are made with band style hose clamps.

Air lines for the transport system are made of 1.25 (OD) diameter aluminum tube for straight sections and 2.25 (OD) diameter flexible plastic hose for bend sections. All connections are made with band style hose clamps.

Large capsules are high-density polyethylene. High density capsules are reusable several times. Small capsules are fabricated from low-density polyethylene capsule without any reuse to transport the sample capsule.

C. REACTIVITY

Calculations and experiments show that the reactivity effects of the unlined pneumatic transit system are negligible and close to zero. The cadmium lined pneumatic transit system has a reactivity of -0.21% Δk/k (-$0.30). Samples introduced to the pneumatic transit system are

evaluated with regard to reactivity and must be less than the values stated in the Technical Specifications.

D. RADIOLOGICAL ASSEMENT

The pneumatic transit system is constructed of aluminum 6061 alloy. One terminal has an additional cadmium liner. Activation calculations show similar levels to that of the central thimble facility. The cadmium liner activated to ^{107}Cd (6.52 day half-life), ^{109}Cd (461 day half-life), ^{111m}Cd (48.5 minute half-life), ^{113m}Cd (14.1 year half-life), ^{113}Cd (7.7×10^{15} year half-life), ^{115m}Cd (44.6 day half-life), ^{115}Cd (2.228 day half-life), ^{117m}Cd (3.4 hour half-life), and ^{117}Cd (2.49 hour half-life). The Cd liner consists of two sheets of 0.020 inch thick sheets. They line the interior of the irradiation terminal that has in inner diameter of 0.685 inches and a height of 20 inches. This equates to 77.7 g of Cd utilized as a liner in the pneumatic transit system. Table 10.4 lists the activity of the Cd liner after a 30 year irradiation at a flux of 10^{12} n cm^{-2} s^{-1} and a 1 year decay. The dominant activity results from ^{109}Cd. With a half-life of 464 days, ^{109}Cd could be allowed to decay on-site for a number of years prior to disposal.

Table 10.4: Activation of Pneumatic Transit System Cadmium Liner

Isotope	Activity (Ci)	Half Life
Cd-107	0	6.490 h
Cd-109	0.04173	464.0 d
Cd-111m	0	48.5 m
Cd-113m	0	14.1 a
Cd-113	13.33e-15	7.7e15 a
Cd-115m	0	44.6 d
Cd-115	0	53.46 h
In-115m	0	4.486 h
In-115	4.252e-15	5.100e15 a
Cd-117m	0	3.4 h
Cd-117	0	2.49 h
In-117m	0	116.5 m
In-117	0	43.80 m
Sn-117m	868.4e-15	13.61 d

Sample activation levels are assessed on an individual basis.

E. INSTRUMENTATION

Instrumentation supporting the pneumatic transit system includes a control system (located in both the control room and in the laboratory associated with the system) and a radiation monitor in the fume hood near the end terminal. The control system allows the system to be

turned on and off, includes manual and automatic send/retrieve controls, and is attached to a timer. The radiation monitor assesses the activity of samples irradiated in the pneumatic transit system and the readings are displayed in both the laboratory and the control room. An alarm is set to warn experimenters and reactor operators if a high activity samples are measured.

F. PHYSICAL RETRAINTS, SHIELDS, OR BEAM CATCHERS

No special restraints or shields are in place for the pneumatic transit system. The transit line has a bend to prevent streaming. The in-core facility utilizes the same shielding that is in place for the reactor core. Shielded areas are available in the laboratory for sample deposition after irradiation.

G. OPERATING CHARACTERISTICS

The unlined pneumatic transit system may be operated at any licensed power level. However, the cadmium lined pneumatic transit facility is limited to a power of 500 kW due to temperature constraints. This limit is to prevent the polyethylene sample rabbits from softening in the facility and becoming fixed in place. Temperature measurements in the terminals at 500 kilowatts and 950 kilowatts were made with a thermocouple. Approximately 30 minutes is required to create steady-state temperatures. Peak temperatures in the standard terminals are 52.5°C and 72 °C at the two respective power levels. Higher temperatures of 83°C and 120 °C occur in the Cd version of the irradiation terminal.

Neutron flux measurements with gold foils and three threshold foils were made to characterize the facility. Results of the measurements are shown in Table 10.5 and demonstrate the operational difference between the two irradiation terminals. Absorption of neutrons by the Cd liner changes the cadmium ratio for a sample from 5.06 to 0.99.

Table 10.5: Flux Measurements in Pneumatic Transit System at 100 kW

	($n\ cm^{-2} s^{-1}$)		
	Thermal	Epithermal	Cd Ratio
No cd	7.8×10^{11}	1.3×10^{10}	5.06
W cd	1.80×10^{9}	1.1×10^{10}	0.99

H. SAFETY ASSESSMENT

Air displacement by CO_2 gas reduces ^{41}Ar production in the system. An air filter in the flow system controls the amount of circulating particulates. As a result, operation of the system without samples causes a minimal radiological risk. Samples need to be evaluated on a case by case basis. In the event of a sample with unexpected high radiation levels, a radiation monitor with an automated alarm will alert experimenters.

With regard to nuclear reactivity, the facility itself is well within Technical Specification requirements. Calculations and measurements on routine samples show reactivity levels less than 0.035% Δk/k ($0.05).

10.2.2.2 Three Element Irradiator

A. DESCRIPTION.

The three element facility is typically used to generate radioisotopes for research or neutron activation analysis.

The three element experiment facility displaces thee fuel elements. The three element facility consists of modifications to the upper and lower grid plate, a fixture for aligning and manipulating the three element canister, and the three element canister. Since the bulk of the upper grid plate supporting the position of three fuel elements is removed, an adapter is required to position fuel elements the facility is not in use. The three-element facility is designed to be rotated (either manually or motor driven) to minimize spatial variations in fluence when required, using a reach-rod or other attachment extended to the bridge.

The facility requires ballast in the form of a metal liner. A lead liner is used for a normal, predominantly thermal neutron irradiation. A cadmium liner is used when reduced thermal neutron flux and enhanced epithermal irradiation is desired.

B. DESIGN & SPECIFICATIONS.

B (1) Upper and Lower Grid Plate Modifications. The upper grid plate has two positions where a three element irradiation canister can be inserted. The positions are fabricated by machining a 2.062 in. (0.052375 m) diameter hole in the upper grid plate centered at a center point between three fuel elements. A hole is fabricated in the lower grid plate centered on the three fuel element positions for alignment.

The alignment fixture is composed of two plates (that interface with the upper and lower grid plates) attached by rods. The lower plate is a disk with lobes corresponding to each of the three fuel element positions. A pin extends through the plate. On the bottom, the pin fits into the centered-penetration in the lower grid plate previously described. A recess in the bottom of the three element canister fits over the pin in the top of the plate. The plate acts as a bearing surface for rotation of the canister. The upper plate is roughly triangular with truncated apexes, and is machined into two separate thicknesses. The thicker, center section of the upper plate has extrusions that mater with vacant fuel penetration holes in the upper grid plate around a center hole for insertion of the canister. The triangular section

extends over fuel positions adjacent to the three element vacancy, and circular cutouts provide clearance for adjacent fuel element cooling channels. Additional holes are drilled around the central hole to provide cooling flow for the three element canister.

B (2) Alignment Frame. The three element facility uses an alignment frame that fits into the core grid location. The alignment frame provides position control, vertical and lateral support, of the irradiation canister. Components of the grid alignment frame consists of the base plate, an alignment pin for the canister, three vertical rods for the frame structure, a top plate for the placement of the irradiation canister, and a fitting for use when the canister is not present. The three element assembly rests on the lower grid plate, and is ballasted to be negatively buoyant. The submerged weight of the three element facility is less than the weight of the three elements it displaces. Although theoretically the all of the three element space could be fully occupied by sample material, flux depression considerations prevent such usage.

Structure rods of the alignment frame prevent the irradiation canister from contacting the adjacent fuel elements during insertion and removal of the irradiation canister into the frame. A 0.5-inch diameter pin.at the base of the frame aligns the irradiation canister and provides a bearing for the rotation of the canister. The rods are welled into the upper and lower plates. At the top of the frame is a 2-inch diameter hole within which the canister rotates. Coolant holes in grid alignment frame provide for cooling of the irradiation canister. A closure fitting is placed on the irradiation assembly frame when a tube is not in place. This fitting minimizes coolant by passing the fuel and prevents inadvertent reactivity insertion into the three-element grid location in the reactor core.

The three element facility positions are in fuel element positions D-05, E-06 and E-07 and fuel element positions D-17, E-22, and E-23. The three element facilities are isolated from potential control rod positions by at least one fuel element position from traditional positions for the pulse and regulating rods, and two fuel rods in the case of the shim rods. One three element facility is two elements from the outer edge of the core, the other is one fuel element from the outer edge of the core.

The D05/E06/E07 three-element facility is close to the radial extension form the center to a power level channel. Experiments have demonstrated that the facility is sufficiently isolated from the leakage neutron path reaching the detector as to not excessively affect the power level signal.

The D17/E22/E23 three-element facility lies in a quadrant of the core between two power level detectors, is closer to the core center, and is sufficiently isolated as to have a minimal effect on leakage neutrons.

B (3) Three Element Facility Canister. The facility uses a sealed canister with a usable space 1.527 in. (0.038786 m) in diameter. A component assembly diagram is provided in Figure 10.4; a rod with an end fitting similar to a fuel element is secured to the top cap for

handling with the fuel tool, and a rod with a tapered end is secured to the bottom for alignment in the lower grid plate penetration. The three element canister outer diameter is 1.875 in. (0.047625 m). The canister wall is 0.1 in. (0.000254 m). The inner liner is 1.625 in. (0.041275 m) outer diameter, and 1.527 in. (0.038786 m) Inner diameter. Overall length from the bottom of the canister to the top of the threaded fitting at the top of the canister (i.e., excluding the handling and alignment pins and the end cap) is 50.375 in. (1.279525 m) with the length of the usable volume 48.125 in. (1.22375 m).

A threaded cap for the top fitting contains two o-ring seals, a pressure relief valve, a gas valve or vent port, and an attachment anchor for remote handling of the canister. Seals for the protection of both expansion and compression pressures in the canister consist of two o-ring seals, one a radial seal and one an end seal. The double seal design should provide extra protection against water leakage into the canister. Two holes in the cap allow venting and purging of the canister gases. One cap hole is the vent line. The other hole contains a pressure relief valve set at a differential pressure of about 2 psig. During sample irradiation the position of the canister is at a depth of about 20 feet of water. At the irradiation position the canister pressure with 20 feet of water will change about 12 psig relative to the loading condition at a pressure of one atmosphere. A threaded hole at the center of the canister cap is for the attachment of a canister-handling device. The type of attachment rod utilized depends on canister handling requirements. One type of attachment is a rod with a fitting for remote attachment with the fuel-handling tool. Routine movements of the canister in the reactor pool and core can then be made with the fuel-handling tool.

When the facility is in use, lobes of the vacated fuel element position are open. The geometry of fuel elements surrounding the three-element facility causes significant potential for variations in exposure based on the position of the material to be irradiated. Therefore the capability to rotate the canister was designed into the facility alignment fixture.

The facility is ballasted with either lead or cadmium. Ballast of approximately 0.0625 in. (0.001588 m) thick is placed between the canister and an inner liner. The liner layer of Cd or Pb wraps twice around the internal aluminum tube, extends almost the full length of the canister, about 46.75 inches, and includes an equivalent end disk at the bottom end. Each layer of the cadmium or lead liner folds over the bottom disks. The two vertical layers of the cadmium liner overlap-while the two vertical layers of lead do not overlap.

With the exception of the ballast, the three element facility is manufactured from aluminum 6061 alloy. Activation of the aluminum components are expected to be similar to the specific acidity described for the central thimble, except that (1) the three element facility does not have the previous irradiation history from the earlier UT Austin Mark I reactor, and (2) the three element positions are at lower flux positions as compared to the central thimble. The lead and

cadmium used as ballast in the three element facility are at least 99.9% pure. Neither the lead nor the cadmium has potential for significant chemical activity in contact with aluminum.

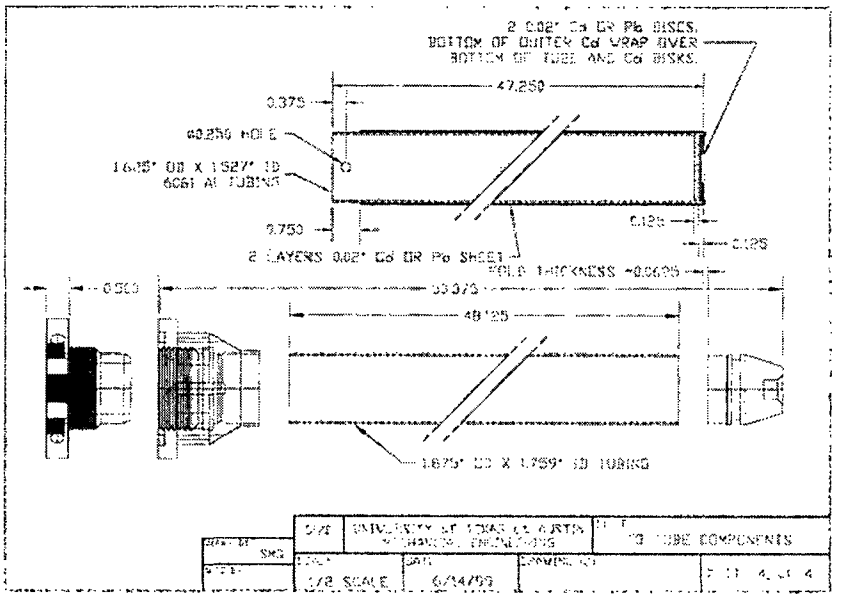


Figure 10.4: Three Element Irradiator

C. REACTIVITY

Removal of three fuel elements for placement of the three element irradiation has a significant effect on core reactivity. The reactivity change has been measured with a control rod bank configuration and with a configuration of one and two control rods full out. The average change in reactivity of the configurations to remove the three fuel elements was $2.30 with a minimum of $2.08 and maximum of $2.47. When three fuel elements are removed for placement of the three element facility, recalibration of control rod worth curves is necessary.

Experiments with the three element irradiator canister require that it remain in the core during operation. Insertion of the irradiator into the core or out of the core must be conducted when the reactor is in a shutdown state. However, the facility may be rotated in place during irradiation. The reactivity effect of canister rotation has a non-measurable effect on reactivity. Only a redistribution of the liner absorbing material is capable of causing the rotational reactivity to change. Unless accident conditions such as mechanical or thermal damage redistribute the neutron absorber materials, the rotation reactivity will remain effectively zero to within a few cents. Estimates of the three element irradiation canister reactivity were made prior to initial tests of the canister. Some of these reactivity estimates were made from measurements with similar equipment at another research reactor facility and include extrapolation of measurements made on similar experiment components such as the two irradiation terminals of the Pneumatic Transfer System.

The reactivity limit for a single moveable experiment must be less than $1.00. As a moveable experiment the three element irradiation canister must be less than $1.00 of reactivity to meet this constraint of the present Technical Specifications. Classification of the three element experiment system as a moveable experiment would be a conservative condition since the total reactivity of the canister will occur only during an insertion, removal, or an unknown type of accident that occurs with the reactor at critical conditions. The total available reactivity change of the three element irradiation canister will not occur with the reactor at critical conditions.

C (1) Reactivity Calculation

Estimates of the experiment facility reactivity were insufficient to determine the operating requirements for the T3 canister with a neutron absorption liner made of Cd. Several calculations were done to develop the final design constraints for the neutron absorption liner. Measurements with the fi nal design were made prior to acceptance of the irradiation system.

Calculations with MCNP(4a), a Monte Carlo particle transport computer code, were made to develop a better evaluation of the canister component reactivity. Previous test measurements and several test core configurations were useful to benchmark the calculation with the measurements. Agreement of the benchmark measurements and MCNP(4a) calculations were adequate to pursue installation and test of the three element irradiation canister. The irradiation canister analysis focused on the most significant reactivity conditions that occur with various configurations of the installation of the cadmium liner version of the system. Development of the MCNP(4a) analysis proceeded in three steps. The first step was a calculation of several reactor core conditions for which measurements were available to compare the experiment and calculation results. The second step was an analysis of the irradiation canister reactivity with a full-length liner of neutron absorber and a short version with a six-inch long neutron absorber. A final step was a calculation of the most plausible accident condition that is flooding of the irradiation canister volume with water.

Calculations project the total three element irradiation canister reactivity worth will change by $1.08 as the absorbing liner changes from a zero-length liner to a full-length liner. Calculation error is as much as 10 to 15%. Although this result exceeds the $1.00 constraint the calculation of the net reactivity available from insertion and removal of the system with the liner does not exceed the limit. Calculations indicate that the three element irradiation canister without any neutron absorbing liner will create a positive reactivity of $0.16. This condition represents the competitive process of neutron leakage from the core and neutron moderation and absorption by the water in the location of the canister.

The goal of the MCNP canister calculation was to determine whether the full length Cd liner in the canister would exceed the conservative constraint of $1.00 for the system worth as a moveable experiment. Initial test measurements in the core did not support the less than $1.00

conclusion. The MCNP calculation predicts the reactivity worth of the irradiation canister with a full-length Cd liner will be less than one dollar. The canister reactivity with Cd liner reactivity was -$0.89 ± $0.12.

A flooding accident with the canister in the core will decrease reactivity by increasing neutron absorption. The MCNP result for the flooded canister condition calculates the negative reactivity change by $0.56 to -$1.45 ± $0.12. Flooding of a canister with a neutron absorption liner will exceed the $1.00.

C (2) Reactivity Measurements

Two measurements of the reactivity of the three element irradiation canister with the full-length cadmium liner found the reactivity worth was -$0.92 with the control rods in a bank configuration. Measurements of the core reactivity were also made with two conditions of the control rods full out. Control rod configuration measurements both decrease and increase the canister worth in the range of $0.89 to $0.96. In the flooded condition the canister negative reactivity worth increases by $0.24 to -$1.16. Extreme positions of the control rods do not significantly change the flooded condition result. These measurement results are consistent with MCNP calculations for the two canister non-flood and flood conditions.

D. RADIOLOGICAL ASSESSMENT

Activation of aluminum 6061 was discussed in the section describing the central thimble.

An average neutron flux was calculated based on a nominal value of $2x10^{11}$ n cm^{-2} s^{-1} with an assumed irradiation at 2 MW, 8 hours per day each week, 11 months each year. With an average neutron flux of $4.37x10^{10}$ n cm^{-2} s^{-1} irradiation over 40 years followed by 1 week of decay, 61.6 pCi per gram of lead 205 is produced.

A similar irradiation of cadmium produces the activities noted in Table 10.6.

Table 10.6: Activity of Three Element Irradiator Cd Liner

Isotope	Activity	Half Life	1 m Dose Rate
Cd-107	1.274 pCi g^{-1}	6.490 h	
Cd-109	40.10 µCi g^{-1}	464.0 d	7.5 µR h^{-1} g^{-1}
Cd-113	10.00e-18 Ci g^{-1}	9.300e15 a	
Cd-115	69.93 µCi g^{-1}	53.46 h	
In-115m	76.34 µCi g^{-1}	4.486 h	
In-115	3.189e-18 Ci g^{-1}	5.100e15 a	
Sn-117m	40.97 nCi g^{-1}	13.61 d	

If the canister is filled with air, ^{41}Ar may be produced. Assuming Argon is 1.28% of the mass of air, with the mass of air as 1.3 kg m^{-3}. Irradiation and decay under the same scheme above

followed by release to the reactor bay results in an atmospheric activity concentration of 128.3 $\mu R\ h^{-1}\ m^{-3}$ resulting from ^{41}Ar.

Based on 50.374 length of the air volume in the canister at 1.527 in. diameter, the canister volume is 0.001512 m^3. It should be noted that the neutron flux value utilized in this calculation is the maximum possible in the reactor (neutron flux is about a factor of five less at the three element irradiator position), is further reduced by the ballast (lead or cadmium), and not constant across the container.

To minimize the potential for the production of ^{41}Ar the canister is flushed with dry nitrogen prior to insertion into the reactor.

E. INSTRUMENTATION

There is no instrumentation associated with the three element facility. Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process.

F. PHYSICAL RESTRAINTS, SHIELDS, or BEAM CATCHERS

No special restraints or shields are in place for the three element facility. The facility is entirely under water during irradiation with no possible radiation streaming outside the reactor pool. The in-core facility utilizes the same shielding that is in place for the reactor core. Shielded areas are available in the reactor bay area for sample deposition after irradiation.

G. OPERATING CHARACTERISTICS

The three element irradiator is a widely utilized facility for in-core irradiations. The lead lined canister is utilized for thermal neutron irradiation at powers up to the maximum licensed power. The cadmium lined facility is utilized for epithermal neutron activation experiments at power levels up to 500 kW. Irradiations are conducted by loading the three element irradiator into the core when the reactor is in a shutdown system. The facility may be rotated during irradiation, but is not inserted or removed during reactor operation.

H. SAFETY ASSESSMENT

H (1) Cooling

Grid holes beneath each fuel element are the coolant flow source for each fuel channel. A provision has been made to also provide coolant channel flow by water convection around the three element canister assembly. The core grid frame contains two holes for each of the three

fuel element positions that make up the experiment facility. Six holes in the grid frame bottom plate provide coolant flow to the three element canister assembly. The bottom fitting of the three element canister contains fins to enhance the heat transfer to the coolant. Coolant flows past the cooling fins along the length of the three element canister assembly. Six holes in the top plate of the core grid frame provide an exit path for coolant flow around the assembly. Generation of heat by the three element canister is substantially less than that of the adjacent fuel element channels. Thermal neutron reaction rates in the neutron absorption liner are a substantial source of heat. Cooling of the three element canister is an important design consideration to protect canister components, specifically samples or materials, from thermal damage. An estimate of the potential temperatures in the three element canister was found by examination of the measurements made with the two PTS irradiation terminals.

H (2) Temperature

The physical design of the cylindrical irradiation canister with internal aluminum cylindrical insert provides a O.072-inch gap. The cylindrical gap prevents the mechanical rearrangement of the absorber material. Thermal redistribution of the materials depends on the melting point for the three materials of the irradiation canister. The irradiation canister is made of 6061 aluminum alloy. The aluminum has a melting temperature of 660 °C. By comparison the liner materials of lead and cadmium have much lower melting temperatures of 327 and 321 °C, respectively. Reactor fuel elements at nominal conditions of full power operation produce maximum fuel temperatures of roughly 450 °C with a respective element cladding temperature of about 140 °C. Heat from neutron activation reactions in the lead or cadmium liner material will produce higher temperatures in the irradiation canister than that of a canister without the liner. Experiments with the pneumatic transit system irradiation terminals found the aluminum terminal without a cadmium liner to have a 500-kilowatt temperature of 54 °C and a 950-kilowatt temperature of 72 °C. The aluminum terminal with a cadmium liner was found to have a 500-kilowatt temperature of 85 °C and a 950-kilowatt temperature of 120 °C. Experiments with low-density polyethylene demonstrate that some material deformation begins at a temperature of 90-95 °C. The temperature limit recommendation for continuous use of polyethylene is a function of the polyethylene density and ranges from 60 to 200 °C. The test location for the pneumatic transit system irradiation terminals was in the reactor core G-ring and neutron fluxes are a factor of 1.8 less than those measured for the three element irradiator. Thus, an equilibrium temperature adjustment by a factor of 1.8, assuming all other heating and cooling factors remain about the same, can be made for the neutron flux difference between the three element irradiator location in rings D and E and the G-ring location of the pneumatic transit system. Estimates of the potential irradiation canister temperatures indicate that the temperatures will not approach the melting temperatures of the lead or cadmium material. The equilibrium temperatures that occur at one hour at full power could exceed 200 °C. These temperatures may cause damage to polyethylene sample capsules and other materials that are irradiated in the canister.

H (3) Pressure

Air pressure relief for excessive pressure buildup in the canister is a design feature to protect the canister from rupture.

Yield stress for the T6 6061-aluminum alloy of the irradiation canister is 30,000 psi. A limit for the canister operating pressures has been set at 250 psi. This limit includes a safety factor of two and a strength reduction for the heat treatment from T6 to TO. Design of the top fitting controls the pressure with a double O-ring *seal* and two 1/8 inch valves, a pressure relief valve and a manual fill valve.

Temperature changes on the three element canister during irradiation and the evolution of gases from experiment materials in the canister will change the ambient pressure. A relief valve has been chosen with a set-point of two to three psi. At pressures less than the setpoint the canister gas inventory will remain constant. A double O-ring seal protects against leakage into the canister. As a constant volume device the canister pressure is readily found from the gas law, PV=nRT. The number of moles of gas, n, the volume, V, and the gas constant R are all constants. For the purpose of the analysis the canister to liner gap is 20 cm^3 and the canister volume is 2400 cm^3. At the operating depth of the canister the external pool water pressure is 10 psi. The differential pressure at the relief valve must exceed the pressure due to water and the pressure setting of the valve. During normal canister operation a change of the air temperature from 300 K° to 350 °K will increase the pressure in the canister by 2.45 psi or about 5 psi per 100 °C. This pressure increase will go to zero as the canister cools following an irradiation.

A change in the number of moles of gas in the canister could also occur. Two source conditions can occur that will increase the gas content in the canister. These potential sources of gas production are vaporization of the water in liquid samples and the evolution of gas by radiation of the polyethylene. Other sources may be present if volatile materials are part of the experiment

Evaporation of water by heating vials of liquid samples will create a total change of 1 cm^3 of liquid to gas. Conversion of 1 cm^3 of water to gas produces 1000 cm^3 of gas. The resultant canister pressure change is +8.18 psi per cm^3 of water vaporized assuming it is distributed over the entire canister volume. The pressure increase should neutralize following cooling of the canister. Irradiation of hydrocarbon materials has the potential to produce 0.1 cm^3 of gas per gram per megarad. The release rate for polyethylene capsule materials is much less, 0.02 cm^3per gram per megarad. If the fast neutron and gamma ray dose in the canister is 1,500 megarad/hour at 1 megawatt the potential gas release from the polyethylene capsules is 30 cm^3 per gram or about 750 cm^3 for an irradiation of 25 sample capsules in a two hour 500 kilowatt operation. This gas production represents a pressure increase of 4.6 psi. This is not a

significant pressure change in the canister although it may cause the canister to vent all the pressure that exceeds the relief valve setting. If sample materials in the capsules are hydrocarbon materials the pressure could be five times greater.

Most of the gas release in the breakdown of polyethylene and other hydrocarbon materials is hydrogen. A purge of the canister atmosphere prior to irradiation with carbon dioxide or nitrogen gas will reduce the available oxygen and eliminate the air activation of argon.

H (4) LOCA potential

The canister is completely submerged during irradiation, and does not offer any leakage path for pool water.

10.2.2.3 6/7 Element Irradiator

A. DESCRIPTION

The 6/7 element irradiator is a large in-core facility to perform neutron irradiations. It is located in the seven-element cutout in the top grid plate of the reactor as shown in Figure 10.2. The facility may be placed in the middle of the core removing 6 fuel elements and the central thimble or it may be placed in the location that overlays part of the outer three fuel rings. It has largely been utilized for irradiation of circuit boards and irradiation of samples for neutron activation analysis.

B. DESIGN AND SPESIFICATIONS

The irradiation can is composed of 6061-T6 aluminum and contains a O.O8 in (2 mm) thick borated aluminum (B) liner. The inner diameter of the irradiation can is 2.25-in. The boron concentration is 4.5% by weight in the 1100 series aluminum alloy. The boron is enriched to greater than 95% ^{10}B, which is the boron isotope with a high thermal neutron cross-section. The design of the irradiation can is very similar to that of the cadmium and lead lined three element facilities described above. The total height of the facility is approximately 52 inches. This height is intended to elevate the stainless steel fittings, a purge valve, and a relief valve above the reactor top grid plate and thereby minimize activation of these components.

The second component is a separate, hollow lead cylinder that is clad with 6061-T6 aluminum. This lead sleeve surrounds the main irradiation can. 6061-T6 aluminum is once again used for this component. The sleeve resembles a thick, hollow cylinder. The outside diameter of the irradiation can is slightly smaller than the inside diameter of the sleeve. When inserted into the middle of the sleeve, the can rests on a pin that is connected to the base of the sleeve. This pin has been designed to accept the 3L facilities previously mentioned. The pin assembly also includes six holes to allow pool coolant to pass through the center of the sleeve. A small gap

exists for the coolant to pass between the can and the sleeve when the can is being used. Three pegs have been built into the top of the sleeve which center the irradiation can when it is in place. The sleeve has been designed to be removable. An eye bolt attached to the top of what resembles the handle of a bucket is used to raise and lower the sleeve.

The connector box is a small, aluminum can which sits approximately 3-ft above the irradiation can. The can and box are connected by an aluminum tube. The tube is for passing electrical wires from the box into the irradiation can. The box is designed to allow for electrical connectors to mate on its inside which isolates electrical wire that is not activated by wire that has been activated during irradiation. From the top of the box extends Tygon tubing to pass through the remainder of the electrical wires to the top of the pool.

C. REACTIVITY.

A MCNPX model of the TRIGA was used to calculate the reactivity of the seven element facility. The base calculation had the seven element location empty with the three element location filled with fuel. The reactivity effect of the change from the fuel configuration with the three element irradiator to the seven element fuel configuration is -$1.28. The perturbation caused by the addition of the lead sleeve is +$0.08. When adding the irradiation can to the assembly, the total experiment worth is $0.25. Therefore, the reactivity of the experiment is far less than $1.00. Reactivity worth of individual experiments in the facility have to be evaluated on an individual basis.

D. RADIOLOGICAL ASSESSMENT

From a radiological perspective, the seven element irradiator is similar to three element irradiator. However, the seven element irradiator does not have a cadmium liner that activates. In its places is boron (95% ^{10}B) liner. The primary absorption reaction is $^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}\alpha$. This reaction does not have a radioactive product, so activation hazards from the boron are minimal. Aluminum activation is similar to that of the other facilities analyzed. Experiments within the seven element irradiator require analysis on an individual basis.

E. INSTRUMENTATION

There is no instrumentation associated with the seven element irradiation facility. Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process.

F. PHYSICAL RESTRAINTS, SHIELDS OR BEAM CATCHERS

No special restraints or shields are in place for the seven element irradiation facility. The facility is entirely under water during irradiation with no possible radiation streaming outside the reactor pool. The in-core facility utilizes the same shielding that is in place for the reactor core. Shielded areas are available in the reactor bay area for sample deposition after irradiation.

G. OPERATING CHARACTERISTICS

Operation of the seven element facility for electronics damage facility is at 1 kW of power or less. The facility allows for electronics to be powered during irradiation through a curved water tight tube going to the pool surface. The facility allows for direct monitoring of electronics as it undergoes irradiation and reaches a point of failure.

H. SAFETY ASSESSMENT

H (1) Temperature (Fuel)

Fuel temperature measurements at 1-kW show the fuel temperature to be +/- 1 °C of the pool water temperature, which was recorded as 20.7 °C. As the reactor is operating at 1-kW, the maximum temperature in anyone fuel pin in the reactor is significantly below the maximum allowable temperatures for the outside clad temperature of greater or less than 500 °C.

H (2) Temperature (Lead)

Calculations were performed to ensure that the lead in the sleeve would not reach near melting temperatures even at a reactor power of 1 MW. The temperature was calculated to be less than 40 °C with a coolant inlet temperature of 25 °C and an inlet velocity of 0.15 m s^{-1}. The melting point of lead is 325 °C. A collision heating (+F6) tally was utilized with the MCNPX model to determine energy deposition in the lead sleeve. Since the temperature increase was so minor, thermal expansion of the lead and aluminum clad are neglected. Additionally, a 1/16 inch gap was added into the design as the distance between the outside edge of the sleeve and the hole in the tap grid plate to prevent the sleeve from becoming stuck in the tap grid plate.

H (3) Pressure (irradiation Can)

Through the aluminum tube and Tygon tubing, the irradiation can is open to atmosphere. Therefore, no internal pressurization will occur.

H (4) Pressure (Lead Sleeve)

The lead sleeve consists of two aluminum tubes that are attached together by two end caps. The lead was not poured within the two tubes completely to the top the sleeve allowing for an air gap. Since the temperature does not rise within the fuel and the energy deposition in the lead is so small, the pressurization of the air within the lead sleeve is negligible and not a risk.

H (5) Mass

The lead sleeve weighs less than 60 pounds. The seven elements that the sleeve replaces weigh approximately 56 pounds. The weight of the lead sleeve is distributed as one single, circular area of 3.874-inches in diameter whereas the weight of each of the fuel elements is distributed on a much smaller area of the grid plate. The irradiation can and connector box are slightly negatively buoyant and do not contribute a significant amount to the total additional mass of the system. The mass of the lead sleeve, irradiation can, and connector box are not a risk.

H (6) Structural

Both lead sleeve and irradiation can are at risk for being dropped onto the top of the core which could cause structural damage to the reactor. To minimize the risk, both sleeve and can are lowered as closely to the side of the pool wall as possible before being maneuvered over the reactor at the height of the top grid plate. The sleeve is stored on the underwater table when not in use and is tied to the top of the pool to keep it from toppling over. Likewise, the irradiation can is stored on the inside of the reactor pool and tied to the top of the tank for storage.

The reactor power is no more than 1 kW for electronic component testing. There is no noticeable increase in the fuel temperature at this power level above the bulk pool water temperature. With no increase in temperature and both the coolant pump and the diffuser nozzle off, there is no flow through the core and no risk for flow blockage. At these temperatures, there is no risk for phase change of coolant either.

All of the components of the lead sleeve, irradiation can, and connection box are fixed together by aluminum welds or tube fittings. The risk for any component of these parts to separate and become a hazard is negligible.

All of the materials in this experiment are sealed water-tight either by welding, fasteners, gaskets, or a combination of these methods. Each of the components (lead sleeve, irradiation can, and connector box) are leak tested prior to being utilized for any experiment requiring the reactor. Therefore, any part of the electronic components under test have no interaction with

the reactor that would cause any material hazard. No hazardous chemicals are used in this experiment or materials that are flammable.

10.2.3 Rotary Specimen Rack

A. DESCRIPTION

The rotary specimen rack (RSR) is used to support neutron activation analysis and isotope production. The rotary specimen rack consists of an air-filled water-tight canister enclosing a sample rack and pinion drive assembly attached to a sample rack. The sample rack is assembled from an upper and lower ring attached to tubes. A ring-drive and an indexing mechanism allow samples to be placed in each position. The pinion gear drive shaft housing is a dry tube from the pool bridge to the rotary specimen rack housing.

Sample vials are inserted and removed through curved dry tubes. Curvature minimizes radiation leakage through the dry tubes. Both a manual and an automatic dry tube are installed, but infrastructure supporting use of the automatic dry tube has not been developed. An electro-mechanical operator attached to a cable is available to support insertion and removal of sample vials. The cable is coiled on a spool operated with a reel. The automatic dry tube is designed to use pneumatic pressure to remove and insert samples.

Rotation can be performed manually or with an installed drive motor, powered from the same source as the pool lights. Rotating samples during a long irradiation evenly distributes the neutron fluence received by each sample.

B. DESIGN SPECIFICICATIONS

The RSR housing is a cylindrical canister with an internal diameter of 22 in, and an outer diameter of 27 2/7 in. Specimen positions are 1.23 in. (3.18 cm.) in diameter by 10.80 in. (27.4 cm.) in depth. Figure 10.5 illustrates the RSR which basically forms a ring outside the reactor core. There are ports for loading of samples as well a drive shaft for rotating the samples.

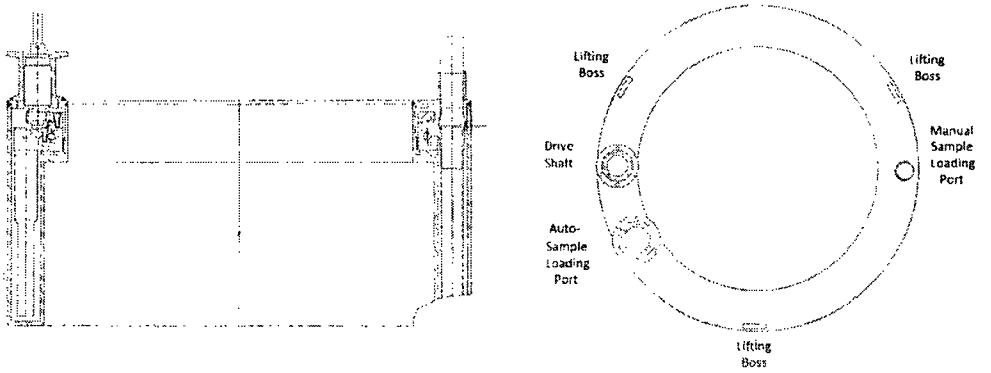


Figure 10.5 Rotary Specimen Rack Diagram

The RSR contains raceways that are supports for sample rotation as illustrated in Figure 10.6. These raceways are manufactured from titanium forgings. There are two concentric raceways with a ball bearing assembly interface.

The inner raceway has an inner diameter of 22 inches, with an outer diameter of 24 ½ inches. The inner raceway is manufactured by welding a 0.38 inch tall by 1 ¼ inch wide ring (ID 22, OD 24 ½) to a 1.12 inch tall by 0.56 inch wide ring (23.12 OD, 22 in ID). Ball bearings (0.045 in. radius) are spaced by four cylindrical titanium separators; separators in contact with the bearings are slightly smaller than the center separators.

The outer raceway provides the second bearing surface and supports the specimen tubes. The router raceway is a ring 1.88 inches tall by 5 5/8 inch wide (21 ½ inch ID by 27 7/8 inch OD). The bottom section of the ring, supporting the specimen tubes, is 0.50 inches tall. There are 40 holes supporting specimen tubes 1.38 inches in diameter equally spaced on a 26.312 inch diameter circle. The upper section is formed from a ring 2 5/8 inches thick (24 1/8 in. OD by 21 ½ in. ID) to accept a spur gear. A spur gear is secured to the top of the outer raceway.

Table 10.7: Rotary Specimen Rack Gears

Item	Spur	Pinion
Teeth	200	10
Width	0.5	0.5
Pitch	23.873	1.194
Pressure angle	20°	20°
Center Distance	12.5335	12.5335
Gear OD	23.992	1.550

Gears are used to drive the RSR rotation mechanism. These are fabricated from aluminum 60601 T-6. Gear specifications are provided in Table 10.7.

The overall length of all specimen tubes is 11.44 in., with the wall thicknesses of 0.058 in. The top of the tubes are flared 45° to 1.62 OD. Position 1 is in two sections. The top section is 5.5 in. tall, with 1 3/8 in. OD. The bottom section OD is 1 in. Positions 2 through 40 have an OD of 1 3/8 inches. The bottom of the cylinder is penetrated by 3 ½ inch holes in the wall at 120° intervals. The bottom of tube is terminated with a ring 0.06 inches thick that has a ¾ inch centered hole.

Figure 10.7 illustrates the RSR rotation control box. The RSR position for loading is indicated in the index dial. Controls are available for manual RSR rotation of automated sample rotation. The direction of automated sample rotation may also be set.

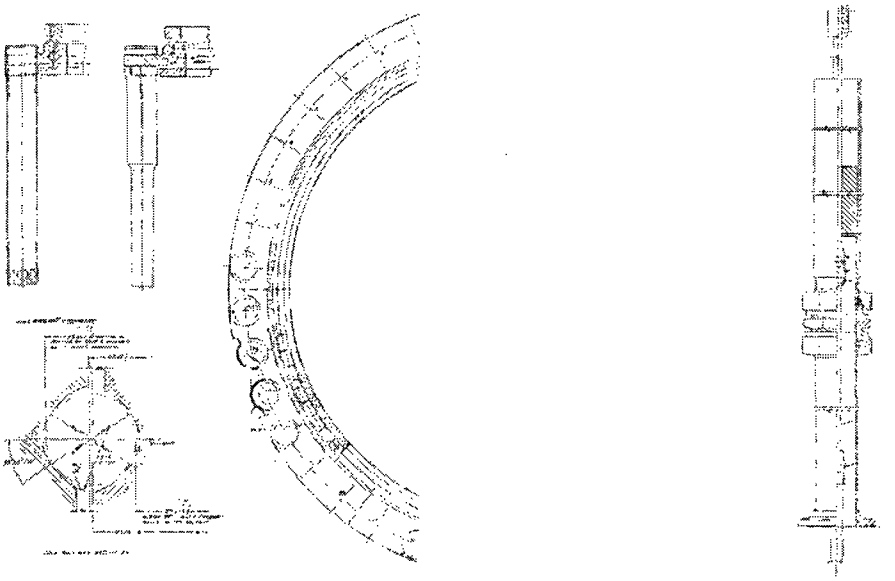


Figure 10.6: Rotary Specimen Rack Raceway Geometry

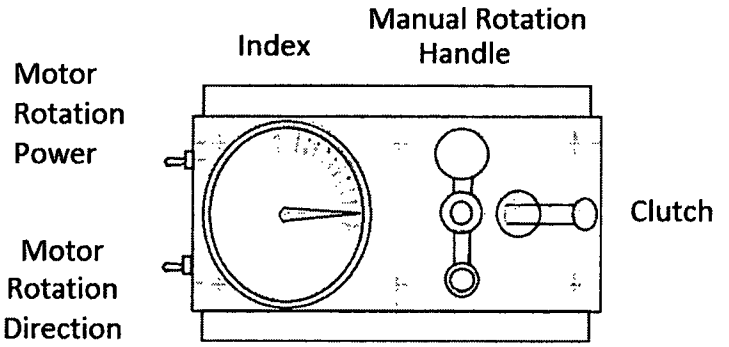


Figure 10.7: Rotary Specimen Rack Rotation Control Box

C. REACTIVITY

The RSR is located outside the reactor core. Along with the graphite reflector and water, the RSR facility affects the reflection of neutrons back into the reactor. However, the facility does not largely affect reactivity due to its proximity to the reactor core. Reactivity worth of individual experiments need to be assessed on an individual basis.

D. RADIOLOGICAL ASSESSMENT

The neutron flux at full reactor power within the RSR facility is *c.a.* 1×10^{12} n cm^{-2} s^{-1}. As such activation rates are less than the three element and seven element facilities analyzed above. The facility does not have a cadmium liner like the three element irradiator, so there is no cadmium activation hazard to assess.

E. INSTRUMENTATION

There is no instrumentation associated with the RSR facility. Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process.

F. PHYSICAL RESTRAINTS, SHIELDS OR BEAM CATCHERS

No special restraints or shields are in place for the RSR. The facility is entirely under water during irradiation. The sample loading tube has a bend to prevent streaming. The in-core facility utilizes the same shielding that is in place for the reactor core. Shielded areas are available in the reactor bay area for sample deposition after irradiation.

G. OPERATING CHARACTERISTICS

The RSR is commonly operated for neutron activation and isotope production experiments. Operation during irradiations is typically in the range of 100 kW to 1 MW. The facility has a strong thermal component to the neutron flux and is utilized for thermal activation. Multiple samples are inserted for simultaneous irradiation. Sample removal is often hours after the irradiation is finished to allow for decay of short-lived radionuclides.

H. SAFETY ASSESMENT

The RSR facility is external to the reactor core and physically isolated from the fuel. The sample loading tube goes to the pool surface and would prevent over pressurization of the facility. Radiological effects and reactivity effects of samples need to be assessed on an individual basis.

10.3 Beam Ports

A. DESCRIPTION

Access to horizontal neutron beams is created by five beam tubes penetrating the reactor shield structure. All beam tubes are 6 inch diameter tubes originating at or in the reactor reflector. One tangential beam tube is composed of a penetration in the reactor reflector assembly with extensions through both sides of the reactor shield. A second tangential beam tube penetrates and terminates in the reactor reflector. The two remaining tubes are oriented radial to the reactor core.

The beam ports, shown in Figure 10.8, provide tubular penetrations through the concrete shield and reactor tank water, making beams of neutrons (or gamma radiation) available for

experiments. The beam ports also provide an irradiation facility for large sample specimens in a region close to the core. Beam port diameters near the core are 6 inches (15.2 cm). The five beam ports are divided into two categories: tangential beam ports and radial beam ports.

B. DESIGN AND SPECIFICATIONS

Two tangential beam ports penetrate the graphite reflector, thecoolant water, and the concrete shield. A hole is drilled in the graphite tangential to the outer edge of the core. One beam port terminates at the tangential point to the core. The other beam tubes extend both directions from the reflector and out opposite sides of the reactor shield.

The two radial beam ports penetrate the concrete shield structure and the coolant water. One radial port terminates at the outer edge of the reflector. The second radial port also terminates at the outer edge of the reflector. However, a hole drilled in the graphite reflector extends the effective source of the radiations to the reactor core region.

C. REACTIVITY

The beam port facilities are external to the reactor core and pose minimal influence on core reactivity. Experiments utilizing beam port facilities require analysis on an individual basis. In order to limit reactivity effects, a collimator and/or filter assembly placed within a beam port may have no portion closer than 2 feet from the outer edge of the core (fuel). Because of this distance, reactivity changes due to insertion of suc collimators and filters are negligible. If a sample or other material is inserted closer than 2 feet from the outer edge of the core, its reactivity worth shall be calculated and verified as part of reactor startup. Any single sample estimated to be worth more than $0.20 shall be secured. Insertion or removal of samples inside a beam port requires prior approval by a Senior Reactor Operator and notification of the Reactor Operator at the time of the action.

D. RADIOLOGICAL ASSESSMENT

Experiments may be conducted within the beam ports tubes or external to shielding. In the case of internal beam port experiments, neutron fluxes can reach up to 10^{12} n cm^{-2} s^{-1}. External neutron beam fluxes range from 10^{6} to 10^{8} n cm^{-2} s^{-1} depending on the shielding and filtering in place. Internal beam tube activations close to the core can reach levels similar to those assessed for the in-core facilities above. External neutron beam fluxes may reach hazardous levels, but they need to be assessed on an individual experiment basis.

E. INSTRUMENTATION

There is no instrumentation associated with the beam port facilities. Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process. Radiation monitors around the reactor bay area are affected by beam port operation and are indicative of experimental conditions. However, these monitors are not directly associated with the beam ports.

F. PHYSICAL RESTRAINTS, SHIELDS, OR BEAM CATCHERS

A step is incorporated into each beam port to prevent radiation streaming through the gap between the beam tube and shielding plug. The inner section of each beam port is an aluminum pipe 6 inches (15.2 cm) in diameter. The outer section of beam ports 1, 2 and 4 consists of a steel pipe 8 inches (20.3 cm) in diameter.

Beam ports 3 and 5 have three outer sections with 8 inch, 12 inch, and 15.25 inch diameters. A lead shield ring in the shield structure provides a "shadow" shield for the 15.25 inch beam port section. Special shielding reduces the radiation outside the concrete to a safe level when the beam port is not in use. The shielding is provided in four sections as follows:

1. inner shield plug,
2. outer shield plug,
3. lead-filled shutter,
4. door.

The inner shield plug consists of graphite cylinder, backed with a 0.125-inch (0.32-cm) sheet of boral and 5 inches (12.7 cm) of lead, sandwiched between two 1.25 inch (3.2 cm) thick steel plates. Beam ports 1, 2, and 4 have a section of graphite 6 inch (15.2 cm) in diameter. Beam ports 3 and 5 have the same configuration as the other beam ports, except that the graphite portion is 6 inch (15.2 cm) in diameter, with a change to 8 inch (20.3 cm) in diameter to provide graphite shielding in the 6 inch and 8 inch portions of the tube. Two rollers are provided to facilitate the insertion and removal of the inner shield plugs. To help guide the shield plug over the steps in the beam tube during insertion, the inner end of the plug is cone-shaped. A threaded hole is provided in the outer end of the plug for attaching the beam tube plug-handling tool. The graphite sections are encased in an aluminum canister.

The outer shield plug is wooden and is 8 inch (20.3 cm) in diameter and 42 inch (1.07 m) long for beam ports 1, 2, and 4. Beam ports 3 and 5 have a wooden shield plug for the outer portion of the tube that has a length of 48. inch (1.22 m) and diameter of 15 inch (38.1 cm) for the outer portion of the tube. A handle on the outer end of this plug is provided for manual handling. The plug is equipped with an electrical circuit consisting of a position switch mounted in the front of the plug and an electrical connector at the rear of the plug. The switch can be actuated only by the inner plug when the inner plug is installed in the beam tube.

A physical contact between the inner and outer shield plug, and an electrical connection between the outer plug and the beam tube are part of an installation status circuit. The circuit monitors the plug configuration or other experiment shield conditions. Information on the console for each beam tube indicates the plug or beam tube status.

The lead-filled shutter and lead-lined door provide limited gamma shielding when the plugs are removed. The shutter is contained in a rectangular steel housing recessed in the outer surface of the concrete shield. The shutter is -10 inch (25.4 cm) in diameter and 9.5 inch (24.1 cm) thick for beam ports 1, 2, and 4. Beam ports 3 and 5 have a shutter that is 15.25 inch (38.7 cm) in diameter and 9.5 inch (24.1 cm) thick. The shutter is operated by a removable push rod on the face of the shield structure and can be moved even with the shutter housing door is closed. In the open position, a section of the shutter consisting of pipe of equal diameter to the outer portion of the beam tube is aligned with the beam port and the outer shield center plug to facilitate insertion or removal of the beam plugs. The shutter housing is equipped with a steel cover plate lined with 1.25 inch (3.2 cm) of lead for additional shielding. A removable cover

plate provides easy access to the beam port. The plate can be bolted shut so that the seal would prevent loss of shielding water if the beam tube should develop a serious leak.

While in use, each beam port has controlled access through concrete walls that serve as shielding and via locked gates. The gates have sensors that alert reactor operators to opening while the reactor is in operation. Beam stops are in place for each beam when the shutter is in the open position.

G. OPERATING CHARACTERISTICS

Neutron beam experiments typically utilize radiation for nuclear analytical techniques. Facility usage has included positron production through neutron irradiation of copper, neutron depth profiling, prompt gamma activation analysis, and neutron radiography. Reactor operation for such experiments is nominally at full power, but can range to lower powers. For good counting statistics, beam port experiments normally last on the order of hours and can take up an entire day of operation. Experiments on multiple beam port facilities may be run simultaneously.

H. SAFETY ASSESSMENT

The main concern of the beam port facilities is that a puncture within the beam port walls into the reactor pool area could cause drainage of the pool system. As a result placement of sharp object, explosive material, or material with high chemical reactivity are limited within the facility. Inflatable plugs may be placed in the beam ports to seal them and minimize loss of coolant.

Experiments performed within the beam port facilities shall not change the cooling channel configuration of the reactor core and will produce negligible additional heating of the core. Thus, no thermal-hydraulic change will occur within the reactor core due to routine neutron beam port usage.

Heating loads to the beam ports due to collimators, neutron filters, or other materials inserted at a distance no closer than 2 feet from the outer edge of the core will be negligible. If a sample or other material is inserted closer than 2 feet from the outer edge of the core, the heating rate shall be calculated and the capacity of the beam port to cool by normal flows of air or water shall be demonstrated to the satisfaction of a supervisory Senior Reactor Operator. Encapsulation of samples shall be sufficient to prevent encapsulation failure due to heating.

Mechanical stresses resulting from the weight of collimator and/or filter pieces inserted no closer than 2 feet from the outer edge of the reactor core will cause no deviations from nominal design conditions because the beam ports are embedded into the concrete shield at distances 2 feet and greater from the outer edge of the core. Any experiment inserted in a

beam port closer than 2 feet to the outer edge of the core must be designed such that weight on the 2 feet section is less than 100 pounds.

10.4 Cold Neutron Source

A. DESCRIPTION

The Texas Cold Neutron Source Facility is located at beam port 3. It consists of the Texas Cold Neutron Source (TCNS), a curved neutron guide system, a converging neutron guide system, a prompt gamma activation analysis system, and extensive shielding.

B. DESIGN AND SPECIFICATIONS

The TCNS consists of a vacuum system, a cryorefrigerator, an aluminum thermosyphon (a.k.a. heat pipe), and a neon cooled moderator chamber. The purpose of the TCNS is to maintain the temperature of the moderator chamber, filled with mesitylene (1, 3, 5-tri-methylbenzene, C9H12), at a temperature of approximately 45 °K when the reactor is operating at 950 kW and at 36 °K when the reactor is shutdown. The moderator chamber is made of aluminum and is cylindrical in shape (3.75 cm radius and a height of 2 cm). The mesitylene, that has a freezing temperature of 228.3 °K, serves to moderate incoming thermal neutrons produced in the reactor core and effectively shift their energies to the subthermal region. The neutrons approach the frozen mesitylene temperature as they travel through the moderator. It is expected that a large fraction of the neutrons entering the moderating medium will exist at a lower energy once they exit the chamber.

The mesitylene temperature is maintained through the use of a gravity driven thermosyphon that uses neon as its working fluid to transfer heat from the moderator to a copper heat exchanger. In turn the copper heat exchanger is coupled to a cold-head that is cryogenically cooled by a helium cryorefrigerator that maintains a temperature of approximately 17 °K when the reactor is operating at 950 kW and 15 °K when the reactor is shutdown.

The TCNS is currently equipped with a Cryomech model AL230 helium cryorefrigerator that is capable of removing 25 W at 20 °K as shown in Figure 10.9. The cryorefrigerator keeps the coldhead at its target temperature by way of its increased capability and range. The cryorefrigerator consists of a compressor package and a cold-head. The cold-head (Figure 10.9), is vertically inserted into a Cryomech designed vacuum box shown in Figure 10.10. It is an expansion device capable of reaching cryogenic temperatures. An extra silicon diode has been installed in order to get more accurate cold-head temperature measurements.

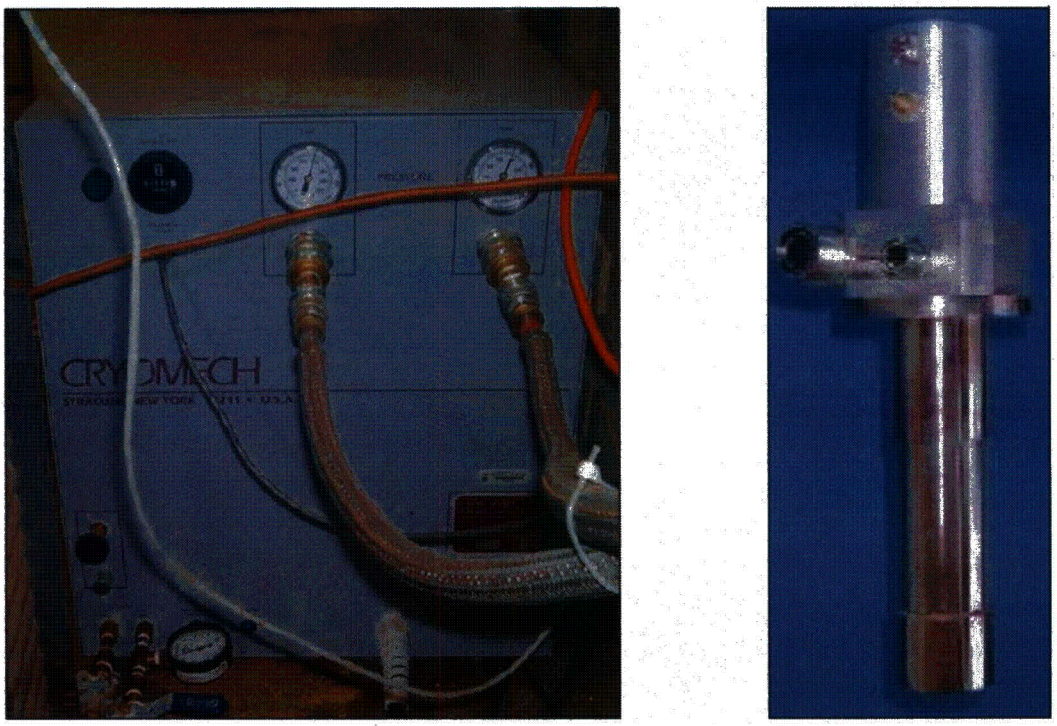


Figure 10.9: Al230 Cryomech Cryorefrigerator and Cold Head

The cold-head consists of two groups of parts; the motor assembly and the base tube assembly. A heat exchanger, made of oxygen free high conductivity (OFHC) copper, is attached to the bottom of the 304 stainless steel tube assembly. The volume of the newly installed OFHC copper block is significantly larger than that of the former heat exchanger. The increased volume of the OFHC copper heat exchanger increased the contact area between itself and the thermosyphon condenser area. The increase in contact area aids in balancing the surface heat flux at the condenser and evaporator ends of the neon thermosyphon. Since the heat transport rate is approximately equal in each section one can transform the surface heat flux at the heat input side to a lower or higher heat flux at the heat output side because the transformed heat flux varies inversely as the ratio of the surface areas [57]. This heat flux property is important when the heat flux associated with the fixed heat source is either too high or too low to be accommodated by the cold-head. The copper heat exchanger is in direct contact with the neon thermosyphon that acts to keep the mesitylene chamber at its target temperature. The moderator, thermosyphon, mesitylene and neon transfer lines are encased within a stainless steel vacuum jacket as shown in Figure 10.11

The neon contained within the thermosyphon, through use of a two phase transformation, transfers the heat generated by the moderator, due to gamma-ray heating (calculated to be less than 2 W), to the end where the cold-head is located. The two phase transformation performed by the neon consists of condensation and subsequent vaporization.

Figure 10.10: Cryomech Cold-Head and Vacuum Box

C. REACTIVITY

The TCNS is external to the reactor core on Beam Port 3. Studies have shown that this facility has a minimal impact on core reactivity.

D. RADIOLOGICAL

At the end of the TCNS beam line the thermal equivalent neutron flux was measures at 1×10^7 n cm^{-2} s^{-1} when the reactor is operating at 950 kW. With the shielding in place, the dose rate surrounding the facility is *c.a.* 1 mrem/hr. The neutron beam line can be turned on and off when via the remote controlled boral shutter.

E. INSTRUMENTATION

The TCNS is equipped with several sensors that are used to measure the various temperatures and pressures associated with the TCNS. Five temperature sensors are used in conjunction with the TCNS to monitor temperature changes and six other sensors are used to monitor pressure changes. Three type "E" Chromel-Constantan thermocouples (TC1, TC2, and TC3) are attached to the mesitylene moderator chamber and two silicon diodes (SD1 and SD2) are located in the vicinity of the cold-head. TC1 is located on the flat face of the moderator chamber closest to the core while TC2 and TC3 are located on the flat face of the moderator furthest from the core.

TC1, TC2, and TC3 are all IOTech Model DBK81 – Built-in Cold Junction Compensation thermocouples. These temperature sensors support up to 7 thermistors per board. Their measuring capabilities support 0.1 degree of precision and 0.5 degree of accuracy from 270 °K to 650 °K. All three sensors connect to an IOTech Model DAQ2000 16-bit 200ksps ADC (64k 5 µsec conversion) that in turn plugs into the system computer's backplane.

Figure 10.12: Silicone Diode and Heater Relative to Cold-Head

SD1 is located on the copper heat sink and SD2 is located on an aluminum yoke that is wrapped around the thermosyphon effectively holding it in place mated to the heat exchanger as shown in Figure 10.12. SD1 is the digital temperature indicator and controller for the Scientific Instruments Model 9650 heater. The silicon diode temperature sensor is capable of measuring temperatures from 1.5 °K to 450 °K with 0.1 °K accuracy of 0.1 degree or better from 1.5 °K to 35 °K and 0.5 °K from 35 °K to 450 °K. The heater provides 60 W of heating (30 V @ 2 A) and connects to the computer through a GPIB interface. SD2 is the temperature indicator and controller for the Scientific Instruments Model 9600 heater. The diode's operation range is 1.5 °K to 450 °K and has a selected sensor excitation current of 100 μA that is can be switched to 10 μA. The heater provides 25 W of heating (25 V @ 1 A) and connects to the computer through a RS-232C serial port.

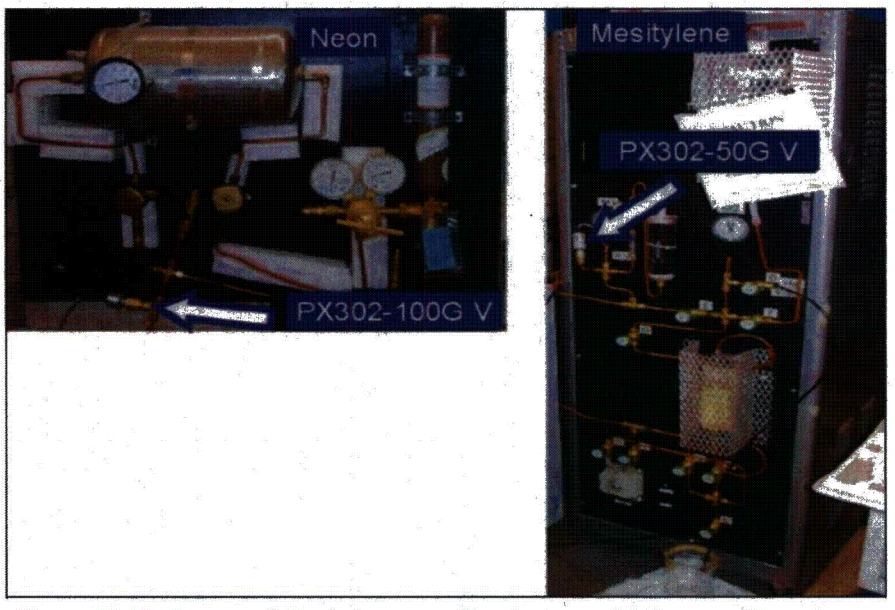


Figure 10.13: Neon and Mesitylene Handling System with Pressure Transducers

The vacuum levels are monitored by an ion gauge (IG) model IGT 274 Bayard-Albert and three model CGT 275 convectron gauges (CG1, CG2, and CG3). Two diaphragm IOTech Model DBK16 pressure transducers (PX302- 100G V and PX302-50G V) are used to measure manometric pressures in psig. PX302-100G V is located on the neon handling system feed line while PX302-50G V is positioned on the mesitylene handling system feed line (Figure 10.13). Each transducer connects to the DAQ2000. Up to 16 DBK16s can be connected to a single DAQ2000 channel. It should be noted that the pressure transducer located on the neon handling system can only record pressures of 100 psig (689 kpa) or less and the transducer on the mesitylene handling system can only record pressures of 50 psig (345 kpa) or less.

The IG and CG1 are located on the right face of the vacuum box. Both the IG and CG1 are used to monitor the evacuated volume in the vacuum box. CG2 is located to the left of the vacuum box between the Leybold manufactured Turbotronik/NT 50 turbo-molecular pump and the roughing pump that are used to obtain the required vacuum level (Figure 3.13). CG3 is placed

with the vacuum pump used to evacuate the curved neutron guide. The convectron gauges are capable of reading 10^{-4} torr to 990 torr. All of the vacuum sensors are connected to an extended capability vacuum gauge controller (307-VGC) that has an operating range of 5×10-12 torr to 760 torr. The 307-VGC connects to the system computer through an RS-232C serial port.

The TCNS vacuum system is also equipped with two remote control gate valves (GV1 and GV2) model DN 63 and DN 16 that are manufactured by the Swiss company VAT. The gate valves are used for isolating the vacuum system during TCNS startup and shutdown procedures. GV1 is located between the vacuum box and the turbo-molecular pump and GV2 is located between the turbomolecular pump and the mechanical pump as shown in Figure 3.13. Both valves are pneumatically actuated and have position indicator switches at each extent. The gate valves are monitored and controlled by a Keithley PDISO-8 that contains 8 optically isolated inputs and 8 electromechanical relay outputs with 3A ratings. The PDISO-8 plugs into the system computer backplane.

F. PHYSICAL RESTRAINTS, SHIELDS, OR BEAM CATCHERS

The TCNS system has an array of restraints, shields, and beam catchers. Figure 10.14 shows these shielding structure surrounding the TCNS with materials including boral, polyethylene, borated polyethylene, Boroflex, Lithoflex, concrete, lead, and Li_2CO_3 powder. The borated materials, Li based materials, and polyethylene are intended for neutron shielding. The lead is primarily a gamma-ray shield. The concrete is in place for both neutron and gamma-ray shielding.

G. OPERATING CHARACTERISTICS

If the TCNS has not been operated recently, the evacuated volume around the moderator chamber and neutron channels should have a nitrogen atmosphere of less than 650 torr. The moderator chamber and filling lines should be filled with low pressure (~1-2 psig or ~7-14 kpa) helium. Mesitylene should be stored in its reservoir with all valves on the mesitylene handling system shut. The thermosyphon valve should be in the off position from the neon-reservoir which should have a pressure neon atmosphere of about 145 psig (1 MPa). At this time, the vacuum system should be shut off and the instrumentation system may or may not be turned off.

If the TCNS has been operated recently, the evacuated volume around the moderator chamber and neutron channels should be evacuated to less the 10^{-4} torr.

H. SAFETY ANALYSIS

If during startup the heat transport rate is too high, the copper heat exchanger temperature may not significantly rise above that of the condenser. Therefore, if the neon in the thermosyphon is originally frozen more condensate will continue to freeze as melting and vaporization occurs in the evaporator end. Since the liquid in the evaporator will not be replenished as long as the condensate in the condenser remains frozen, the evaporator and mesitylene chamber will begin to overheat which will cause an unwanted buildup in pressure towards the bottom of the thermosyphon. In order to avoid this situation, care should be taken to optimize the thermal resistance between the heat exchanger and the thermosyphon during startup. Freeze-out can be avoided by fully insulating the condenser against heat loss and allowing the thermosyphon condenser temperature to rise above neon's critical point of 24.5 °K. This will allow the liquid neon to replenish the vaporized neon in the evaporator section and keep the mesitylene from melting too fast.

The vapor within the thermosyphon typically reaches sonic velocity during startup and thus the drag force at the liquid-vapor interface may be relatively high. If the entrainment limit is not greater than the sonic limit the neon liquid will be entrained by the neon vapor and will therefore lead to evaporator dry out and overheating since the liquid return rate to the evaporator will be reduced. This type of failure will not cause any type of pressure buildup within the thermosyphon but will affect the ability of the TCNS to keep the moderator frozen. However, as long as the actual heat transport rate is equal to the sonic limit and the entrainment limit is greater than the sonic limit, entrainment can be avoided. Entrainment may also be avoided by adding a non-condensable gas to the vapor space. The non-condensable gas, during startup, will limit the effective condenser heat rejection area by occupying most of the vapor condenser area while the neon vapor is at a low pressure. By occupying the vapor space, the noncondensable gas also raises the thermal resistance between the condenser and heat exchanger and thus decreases the ability of freeze-out to occur.

None of the failure mechanisms presented here increases the probability of an accident, involving the use of the TCNS, to occur. Each of the above mentioned failures fall within the limits and capabilities previously evaluated.

10.5 Non-reactor experiment facilities

The NETL maintains a number of facilities related to nuclear radiation and detection. These facilities are utilized for teaching, research, and service work.

10.5.1 Neutron generator room

The NETL houses a neutron generator room that has 3 foot thick concrete walls, floor and ceiling. The room currently is utilized for operation of a Thermo Scientific MP 320 D-T neutron generator and other neutron based experiments. Figure 10.15 shows that this is a compact neutron generator designed for portability. The MP 320 has a flux of 1×10^8 n s^{-1} and has a pulse rate of between 250 Hz to 20 kHz. The fast neutron source uses a deuterium-tritium reaction to produce 14 MeV neutrons.

The system is paired with an ORTEC GMX50P4-83 n-type HPGe detector. The detector is specially equipped with an integrated heater for annealing the HPGe crystal after damage from fast neutrons. The MP 320 provides an output to synchronize gamma-ray spectrum acquisition with the neutron pulses. For this setup, two MCAs are utilized so that spectra will be acquired during the neutron pulse (prompt) and between the pulses (delayed).

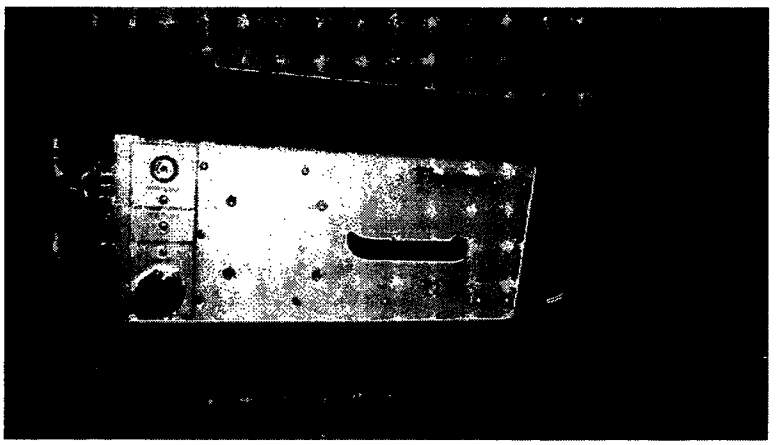


Figure 10.15: Thermo MP 320 Neutron Generator at NETL

10.5.2 Subcritical assembly

Cylindrical subcritical assemblies of graphite and polyethylene are utilized for student laboratory experiments with neutron sources and a subcritical ^{235}U assembly. The plutonium-beryllium neutron sources and uranium dioxide used in the polyethylene subcritical assembly may be stored and used in the room containing the reactor, but are licensed separately from the reactor. The subcritical core and moderator assemblies are products of Lockheed Nuclear Products. Figure 10.16 Illustrates the subcritical facilities.

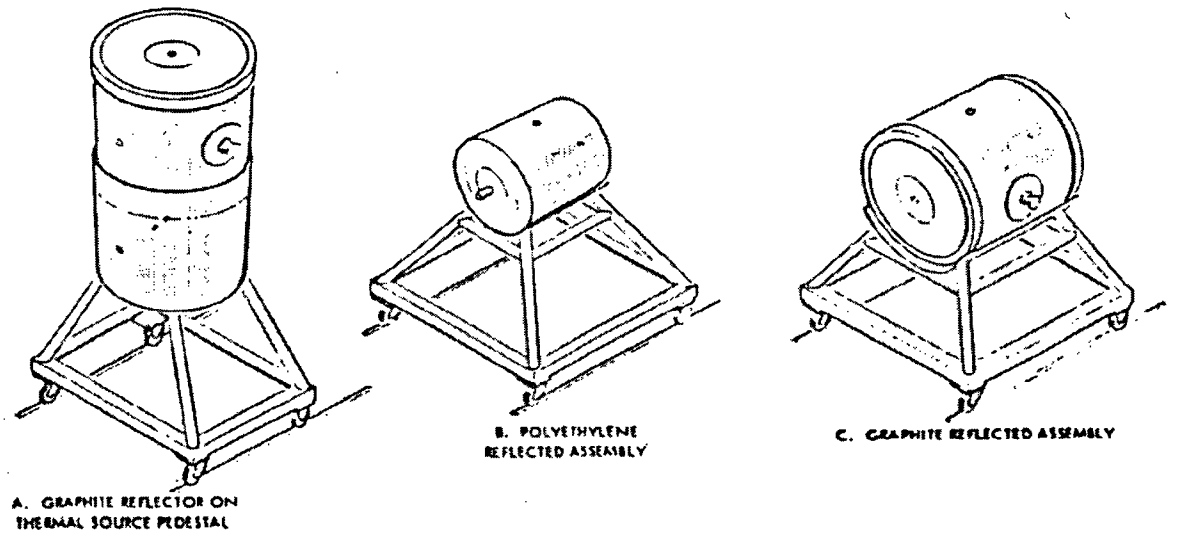


Figure 10.16 Subcritical Assemblies

The subcritical polyethylene core is a cylinder 10 inches in diameter and 14 inches long. Reflector assemblies can be assembled with or without the fueled core. Dimensions of the cylindrical reflector assemblies are 30 inch diameter by 34 inch length for the graphite moderator and 22 inch diameter by 25 inch length for the polyethylene moderator. An additional graphite moderator cylinder 30.5 inches high by 24 inch diameter is available for neutron source moderation.

10.5.3 Laboratories

10.5.3.1 Radiochemistry laboratory

The radiochemistry laboratory focuses on work utilizing open nuclear sources. It contains a fume hood along with laboratory equipment to support radiochemistry experiments. Wet chemistry experiments and radioactive gas experiments are often conducted in this facility. Nuclear detection equipment including alpha spectroscopy, beta-gamma coincidence spectroscopy, and standard NaI(Tl) detectors are currently utilized in the laboratory.

10.5.3.2 Neuron Activation Analysis Laboratory

A neutron activation analysis laboratory contains a terminal for the pneumatic transit system. The laboratory includes a glove box utilized for sample handling and houses the terminal for the manual pneumatic transit system. The laboratory contains shielded areas for neutron activation analysis samples and HPGe detectors for gamma-ray spectral acquisitions.

10.5.3.3 Radiation detection laboratory

The radiation detection laboratory is utilized for gamma-ray spectroscopy as well as laboratory classes. It is one of the larger laboratories with benches that may be utilized for a wide variety of radiation detection experiments. Multiple HPGe detectors are in the facility that are utilized for measurement of long-lived radionuclides. This laboratory is primarily utilized for experiments with sealed nuclear sources.

10.5.3.4 Sample preparation laboratory

The sample preparation laboratory is utilized for sample packaging and recording. It has a fume hood for experiments. It has a clean bench, high precision scale, and ovens for sample drying. Radioactive materials are not utilized in this laboratory to prevent contamination of samples being prepared for experiments.

10.5.3.5 General purpose laboratory

The general purpose laboratory is utilized for radioactive sample based experiments along with non-radioactive material experiments. The laboratory includes work benches and storage cabinets.

10.6 Experiment Review

The Reactor Oversight Committee (ROC) oversees the nuclear reactor and approval of experiments. The ROC ensures that the experiment follows ALARA protocols and does not violate any Technical Specifications. In addition a general safety analysis is performed. Experimenters are required to submit a document describing their experiment and address the items identified in Table 10.8.

The ROC reviews the safety analysis report with respect to facility Technical Specifications, public safety, experimenter safety, protection of the facility, and ALARA principles. Experimental proposals may be accepted, rejected, or have suggested modifications. The ROC may also require additional analysis to support the safety assessment of the experiment. Once

an experiment is approved, experimenters may schedule experiments through an Operations Request. An Operations Request requires the approval of a Senior Reactor Operator prior to being conducted.

Table 10.8: Items to be Addressed in Safety Analysis for Experiments

Topic	Description
Description and Purpose of Experiment	This section shall include a general review of the experiment. A purpose and goals should be identified.
Experimental Requirements	This section identifies the facilities and operational requirements for the facility.
Experiment Facility and Location	Identify the specific facility and location within the reactor.
Maximum Reactor Power	Describe the maximum power at which the experiment will be conducted (for pulse experiments the reactivity insertion should be identified as well).
Maximum Operation Time	Provide a conservative estimate of the time at power required for the experiment.
Physical Experiment Effects	This section describes the reactor effects.
Reactivity	Conservatively based reactivity calculations should be performed. Identify worst case scenarios for the experiment and calculate the reactivity effect of these cases.
Thermal Hydraulic and Experiment Temperature	Identify heat transfer concerns that will occur in experiment. If there appears to be any heat transfer concerns, conservative calculations should be made to calculate maximum temperatures in the fuel and in the experiments.
Mechanical Stress	Mechanical stress issues should be identified. Calculations should support conclusions based on possible pressure increases or other mechanical stresses.
Material Evaluation	The materials in the experiment should be identified and classified.
Radioactivity	Activation calculations should be performed. Based on these calculations, health physics

Table 10.8: Items to be Addressed in Safety Analysis for Experiments

Topic	Description
	concerns should be addressed. If radioiodine or radiostrontium are produced, calculations should be compared to maximum values stated in the Technical Specifications.
Material Hazards	This relates to specific material hazards.
Trace Element Impurities Which May Represent a Significant Radiological Hazard	Identify elements which may activate to produce radiation hazards.
High Cross-Section Elements	Identify high cross-section elements and address reactivity and radioactivity concerns.
Flammable, Volatile, or Liquid Materials	Identify flammable, volatile, or liquid materials. If such materials are in the experiment, address containment issues and estimate consequences of worst case accident scenario.
Explosive Chemicals	Identify explosive chemicals within the experiments. Address safety concerns and make sure quantities are less than those stated in the Technical Specifications.
Radiation Sensitive Materials Which When Exposed to Radiation Exhibit Degradation of Mechanical Properties, Decomposition, Chemical Changes, or Gas Evolution	Identify materials that suffer from radiation effects. Special concern should be placed on materials that emit hydrogen or other combustible gasses upon being irradiated. Also address possible degradation of sample containment during irradiation.
Toxic Compounds	Identify toxic compounds and chemicals within the experiment. Address safety concerns.
Cryogenic Liquids	Identify cryogenic liquids within each experiment and address safety concerns.
Unknown Materials	Sometimes samples are analyzed via various nuclear techniques. In such cases the makeup of samples may not be entirely known. Try to estimate the bounds of experimental sample compositions and address safety concerns.
Experiment Classification	Experiments are identified as being Class A, B, or C. 1) Class A experiments require a senior operator (Class A, SRO) to direct an activity of

Table 10.8: Items to be Addressed in Safety Analysis for Experiments

Topic	Description
	experiment. 2) Class B experiments require only an operator and if necessary an experimenter (Class B, RO) to perform the experiment, with an SRO available. 3) Class C experiments are all non-reactor experiments.

11.0 RADIATION PROTECTION AND WASTE MANAGEMENT

This chapter deals with the overall NETL radiation protection program and the corresponding program for management of radioactive waste. The chapter is focused on identifying the radiation sources which will be present during normal operation of the reactor and upon the many different types of facility radiation protection programs carried out to monitor and control these sources. This chapter also identifies expected radiation exposures due to normal operation and use of the reactor.

11.1 Radiation Protection

The purpose of the NETL radiation protection program is to allow the maximum beneficial use of radiation sources with minimum radiation exposure to personnel and the general public. Requirements and procedures set forth in this program are designed to meet the fundamental principle of maintaining radiation exposures As Low As Reasonably Achievable (ALARA).

11.1.1 Radiation Sources

The radiation sources present at the NETL can be categorized as airborne, liquid, or solid. Airborne sources consist mainly of argon-41 due largely to neutron activation of air dissolved in the reactor's primary coolant. Liquid sources include mainly the reactor primary coolant. Solid sources are more diverse, but are typical of a research reactor facility. Such sources include the fuel in use in the core, irradiated fuel in storage, and fresh unirradiated fuel. In addition, other solid sources are present such as the neutron startup source, irradiated experiment materials, items irradiated as part of normal reactor use, various check, reference, and calibration sources and a limited amount of solid waste.

11.1.1.1 Airborne Radiation Sources

During normal operation of the NETL reactor, airborne radioactivity is almost exclusively Ar-41.

11.1.1.1.1 Production of Ar-41 in the Reactor Room

Production of Ar-41 in the pool water can be found by determining the concentration of Ar-40 in the water and multiplying by the volume of water irradiated, the Ar-41 production cross section, and the thermal neutron flux. From information obtained from Dorsey[1], one sees that the Ar-40 concentration in water at typical core inlet temperature is approximately 7.1×10^{15} atoms cm^{-3}. Given the volume of water in the core is 18500 cm^3, the effective cross section for production of Ar-41 is 0.661×10^{-24} cm^2, and thermal neutron flux of 2.4×10^{13} n cm^{-2} s^{-1} at the central thimble at 1.1 MW is assumed to be the uniform flux across the entire core, a conservative Ar-41 production rate is approximately 2.1×10^9 atom s^{-1}. Assuming continuous operation at 1.1 MW, the equilibrium activity of Ar-41 in the pool water is 2.1×10^9 Bq.

Likewise, the production of Ar-41 in experimental facilities can be found by multiplying the concentration of Ar-40 in air by the volume of air irradiated, the Ar-41 production cross section, and the thermal neutron flux. The natural concentration of argon in air is 0.93% which equates (at STP) to $2.5x10^{17}$ argon-40 atoms cm^{-3}. The effective air volume of the beam tubes is $5.9x10^{5}$ cm^{-3} and the average thermal neutron flux in the beam tubes is $1x10^{11}$ n cm^{-2} s^{-1}. This results in an argon-41 production rate in the beam tubes of $9.7x10^{9}$ atom s^{-1}. The effective air volume of the rotary specimen rack (RSR) is $3.3x10^{4}$ cm^{-3} and the average thermal neutron flux in the RSR is $6x10^{12}$ n cm^{-2} s^{-1}. This results in an argon-41 production rate in the RSR of $3.3x10^{10}$ atom s^{-1}. Assuming continuous operation at 1.1 MW, the equilibrium activity of Ar-41 in the experimental facilities is $4.3x10^{10}$ Bq.

At equilibrium, the production of Ar-41 in the poo waterl and experimental facilities is equal to the removal of Ar-41 from the pool water and experimental facilities. Assuming this removal is exclusively diffusion of Ar-41 into the air of the reactor room and assuming all this activity diffuses uniformly into the volume of the reactor room ($4.12x10^{9}$ cm^{3}), the Ar-41 activity concentration would be $3.0x10^{-4}$ µCi cm^{-3} which is 100 times the DAC value of $3x10^{-6}$ µCi cm^{-3}. As Ar-41 is a noble gas, assuming a semi-infinite cloud model, the dose rate in the reactor room would be approximately 320 mrem hr^{-1} during extended 1.1 MW operations due to airborne Ar-41. While this would be a high radiation area, exposures to this airborne radiation source can easily be controlled by personnel monitoring and procedural control over access to the reactor room. However, in reality, all the experimental facilities are not utilized simultaneously (resulting in less volume of air for Ar-41 production) and a facility ventilation system exchanges the room air mitigating this potential exposure. Additionally, due to the utilization trends at the NETL, extended 1.1 MW operations are not the norm. Operational experience has shown that airborne argon-41 is not a significant contribution to occupational dose at the NETL.

11.1.1.1.2 Radiological Impact of Ar-41 Outside the Operations Boundary

Argon-41 is the only routine effluent from the NETL. A conservative estimate of effluent concentration outside the facility is to calculate the ground level concentration at the building using:

$$X(0,0,0)= Q/(0.5)(A)(\bar{u})$$

where

X(0,0,0) = Ground level concentration at the building in µCi m^{-3}

Q = Activity release rate in µCi s^{-1}

A = Cross sectional area of the reactor building (256 m^{2})

$\bar{u}$ = Mean wind speed (assumed as 1 m s^{-1})

Q is determined by multiplying the activity concentration in the reactor room (3.0×10^{-4} μCi cm^{-3}) by the volume release rate of the stack (3.9×10^{6} cm^3 s^{-1}). Thus Q = 1170 μCi s^{-1} and X(0,0,0) = 9.1 μCi m^{-3} = 9.1×10^{-6} μCi cm^{-3}. While this concentration is about 900 times the effluent concentration limit of 1×10^{-8} μCi cm^{-3}, this is based on a very conservative calculation based on continuous operation at 1.1MW. In reality, operations are not continuous and are not always at full power. Measured Ar-41 releases over the past several years shows an average annual Ar-41 release of less than 6 Ci per year (0.2 μCi s^{-1}). Using a 6 Ci per year release rate in the above equation gives a ground level concentration at the building of 1.6×10^{-3} μCi m^{-3} = 1.6×10^{-9} μCi cm^{-3} which is well below the effluent concentration limit.

Determination of radiation dose to the general public from airborne effluents may also be carried out using several computer codes recognized by regulatory authorities. One such method involves use of the Clean Air Assessment Package - 1988 (CAP88-PC). Application of this code to the very conservatively projected Ar-41 releases from continuous 1.1MW operation at the NETL predicts a dose to the maximally exposed individual of approximately 66 mrem per year. Applying the code to the more reasonable release rate of 6 Ci per year predicts a dose to the maximally exposed individual of less than 0.02 mrem per year.

11.1.1.2 Liquid Radioactive Sources

Liquid radioactive material routinely produced as part of the normal operation of the NETL includes miscellaneous neutron activation products in the primary coolant. Many of these activation products are deposited in the mechanical filter and the demineralizer resins. Therefore, these materials are dealt with as solid sources. Non-routine liquid radioactive waste could result from decontamination or maintenance activities (i.e., filter or resin changes). The amount of this type of liquid waste is expected to remain small, especially based on past experience. There are also various liquid radioactive materials used as reference or calibration standards for instruments. However, these materials tend to be low volume and low activity. A [redacted] liquid analytical samples produces liquid radioactive sources. However, these materials too are typically low volume and low activity. Thus, the primary liquid radioactive source at the NETL is the primary coolant.

11.1.1.2.1 Radioactivity in the Primary Coolant

Nitrogen-16 is produced by fast neutron activation of oxygen-16 in the water of the primary coolant. The oxygen density in water is approximately 3.3×10^{22} atoms cm^{-3}. Given the volume of water in the core is 18500 cm^3, the effective cross section for production of N-16 is 2.1×10^{-29} cm^2, and neutron flux of 1×10^{13} n cm^{-2} s^{-1} in the energy range of interest at 1.1 MW is assumed to be the uniform flux across the entire core, a conservative N-16 production rate is approximately 1.3×10^{11} atom s^{-1}. Assuming continuous operation at 1.1 MW, the equilibrium

activity of nitrogen-16 in the core region is $1.3x10^{11}$ Bq. At equilibrium, the production of N-16 in the core region is equal to the removal of N-16 from the pool. As the N-16 tends to stay in solution and the half-life of N-16 is 7.1s, the primary removal mechanism from the pool is decay.

The N-16 from the core region moves through the reactor tank by natural convection. Assuming the water containing the N-16 continues upward to the surface of the pool at the coolant flow velocity through the core (17 cm s^{-1}), it will traverse the distance to the surface (640 cm) in about 38 seconds. In that time period, substantial radioactive decay will have occurred resulting in $3.2x10^{9}$ Bq actually reaching the surface. Assuming the N-16 that makes it to the surface of the pool spreads out into a uniform disk of 2 meter diameter, the calculated dose rate at 1 meter above the surface of the water would be about 90 mrem hr^{-1}. Exposures to this liquid radiation source can easily be controlled by personnel monitoring and procedural control over access to the area of the surface of the reactor pool. However, in reality, due to the utilization trends at the NETL, extended 1.1 MW operations are not the norm. Operational experience has shown that nitrogen-16 is not a significant contribution to occupational dose at the NETL.

11.1.1.2.2 N-16 Radiation Dose Rates from Primary Coolant

Nitrogen-16 is produced by fast neutron activation of oxygen-16 in the water of the primary coolant. The oxygen density in water is approximately $3.3x10^{22}$ atoms cm^{-3}. Given the volume of water in the core is 18500 cm^3, the effective cross section for production of N-16 is $2.1x10^{-29}$ cm^2, and neutron flux of $1x10^{13}$ n cm^{-2} s^{-1} in the energy range of interest at 1.1 MW is assumed to be the uniform flux across the entire core, a conservative N-16 production rate is approximately $1.3x10^{11}$ atom s^{-1}. Assuming continuous operation at 1.1 MW, the equilibrium activity of nitrogen-16 in the core region is $1.3x10^{11}$ Bq. At equilibrium, the production of N-16 in the core region is equal to the removal of N-16 from the pool. As the N-16 tends to stay in solution and the half-life of N-16 is 7.1s, the primary removal mechanism from the pool is decay. The N-16 from the core region moves through the reactor tank by natural convection. The time it takes for the N-16 to move to the surface of the tank, T, is given by the ratio of the volume above the core region ($4x10^{7}$ cm^3) to the rate at which the activated coolant is flowing into that volume ($8x10^{3}$ cm^3 s^{-1}). Thus, T is equal to 5000 s. By the time the N-16 would reach the surface of the tank, it has decayed to background. Therefore, an equilibrium concentration of N-16 in the primary coolant will never be reached. Thus, the N-16 becomes a radiation source below the surface of the reactor tank. As it takes 5000s for the coolant exiting the core to reach the surface 6.4m above, the vertical velocity of the coolant is approximately 1.3 cm s^{-1}. After ten half-lives (71s), the activity would be reduced by approximately three orders of magnitude. In 71 seconds, the N-16 would move upward approximately 92 cm. Additional time spent moving upward results in additional decay. Thus it is assumed any significant contribution to dose at the surface of the tank results from N-16 activity approximately 5.5m below the surface of the tank. As a conservative case, the dose rate from a disk source of 2 meter diameter with total activity equal to the equilibrium N-16 activity located 5.5m below the surface of the tank is calculated to be approximately 170 mrem hr^{-1} at the surface of the tank

without taking into account the shielding provided by the 5.5m of water. The tenth value thickness of water for N-16 photons is approximately 1m. Thus, even taking into account a buildup factor of approximately an order of magnitude for this thickness of water, the dose rate would be attenuated by approximately four orders of magnitude due to the shielding provided by the water resulting in actual dose rates from N-16 near background at the surface of the tank.

11.1.1.3 Solid Radioactive Sources

The solid radioactive sources associated with the NETL program are summarized in the following table. Because the actual inventory of reactor fuel and other radioactive sources continuously changes as part of the normal operation, the information in the table is to be considered representative rather than an exact inventory.

Although solid waste is included in the preceding table, more information on waste classification, storage, packaging and shipment is included in Section 11.2.

11.1.1.3.1 Shielding Logic

Although not a solid source of radioactivity itself, shielding is involved in reducing radiation levels from many solid sources and therefore the basic logic used for the reactor shielding is included here. The logic and bases used for the NETL shielding design originated from General Atomic developed source terms for 1.5MW operation. Shielding was designed for a surface dose rate of no more than 1 mrem hr^{-1}.

Operational experience has shown the shield performs as designed. As the irradiated fuel is the most significant solid radioactive source at the NETL, as long as it remains within the reactor shield structure, no significant occupational radiation exposure is expected.

11.1.2 Radiation Protection Program

The radiation protection program for the NETL is executed with the goal of limiting radiation exposures and radioactivity releases to levels that are as low as reasonably achievable without seriously restricting operation of the facility for purposes of education, research, and service. The program is executed in coordination with The University of Texas at Austin, Office of Environmental Health and Safety, Radiation Safety Office. The program has been reviewed and approved by the Reactor Oversight Committee for the facility. The program was developed following the guidance of ANSI 15.11 *Radiation Protection at Research Reactor Facilities* and designed to meet the requirements of 10CFR20. Some aspects of the program deal with radioactive materials regulated by the Texas Department of State Health Services (TDSHS) under license L00485 and the program has been reviewed by the Radiation Safety Committee

which has responsibility for administering the radiation protection program under the TDSHS license.

11.1.2.1 Management and Administration

11.1.2.1.1 Level 1 Personnel

Level 1 represents the central administrative functions of the university and the Cockrell School of Engineering. The University of Texas at Austin is composed of 16 separate colleges and schools; the Cockrell School of Engineering manages eight departments with individual degree programs. The Nuclear Engineering Teaching Laboratory (NETL) is one of several education and research functions within the School.

President, The University of Texas at Austin

The President is the individual vested by the University of Texas System with responsibility for the University of Texas at Austin.

Executive Vice President and Provost

Research and educational programs are administered through the Office of the Executive Vice President and Provost. Separate officers assist with the administration of research activities and academic affairs with specific management functions delegated to the Dean of the Cockrell School of Engineering and the Chairman of the Mechanical Engineering Department.

Dean of the Cockrell School of Engineering

The Dean of the Cockrell School of Engineering reports to the Provost. The School consists of 8 departments and undergraduate degree programs and 12 graduate degree programs.

11.1.2.1.2 Level 2 Personnel

The Nuclear Engineering Teaching Laboratory operates as a unit of the Department of Mechanical Engineering at The University of Texas at Austin. Level 2 personnel are those with direct responsibilities for administration and management of resources for the facility, including the Chair of the Mechanical Engineering Department, the NETL Director and Associate Director. Oversight roles are provided at Level 2 by the Radiation Safety Committee, the Radiation Safety Officer and the Reactor Oversight Committee.

Chair, Department of Mechanical Engineering

The Chairman reports to the Dean of the Cockrell School of Engineering. The Department manages 8 areas of study, including Nuclear and Radiation Engineering.

Director, Nuclear Engineering Teaching Laboratory (NETL Director)

Nuclear Engineering Teaching Laboratory programs are directed by an engineering faculty member with academic responsibilities in nuclear engineering and research related to nuclear applications. The Director is a member of the Cockrell School of Engineering, and the Department of Mechanical Engineering.

Associate Director

The Associate Director is responsible for safe and effective conduct of operations and maintenance of the TRIGA nuclear reactor. Other activities performed by the Associate Director and staff include neutron and gamma irradiation service, operator/engineering training courses, and teaching reactor short courses. In addition to Level 3 staff, an Administrative Assistant and an Electronics Technician report to the Associate Director. Many staff functions overlap, with significant cooperation required.

Safety Oversight

Safety oversight is provided for radiation protection and facility safety functions. A University of Texas Radiation Safety Committee is responsible programmatically for coordination, training and oversight of the University radiation protection program, with management of the program through a Radiation Safety Officer. Nuclear reactor facility safety oversight is the responsibility of a Reactor Oversight Committee.

Radiation Safety Committee

The Radiation Safety Committee reports to the President and has the broad responsibility for policies and practices regarding the license, purchase, shipment, use, monitoring, disposal and transfer of radioisotopes or sources of ionizing radiation at The University of Texas at Austin. The Committee meets at least three times each calendar year. The Committee is consulted by the Office of Environmental Health and Safety concerning any unusual or exceptional action that affects the administration of the Radiation Safety Program.

Radiation Safety Officer

A Radiation Safety Officer holds delegated authority of the Radiation Safety Committee in the daily implementation of policies and practices regarding the safe use of radioisotopes and sources of radiation as determined by the Radiation Safety Committee. Radiation Safety Officer responsibilities are outlined in *Radioactive Materials License Commitments for The University of Texas at Austin*. The Radiation Safety Officer has an ancillary function reporting to the NETL Director as required on matters of radiological protection. The Radiation Safety Program is administered through the University Office of Environmental Health and Safety. A NETL Health Physicist (Level 3) manages daily radiological protection functions at the NETL, and reports to

the Radiation Safety Officer as well as the Associate Director. This arrangement assures independence of the Health Physicist through the Radiation Safety Officer while maintaining close interaction with NETL line management.

Reactor Oversight Committee (ROC)

The Reactor Oversight Committee evaluates, reviews, and approves facility standards for safe operation of the nuclear reactor and associated facilities. The ROC meets at least semiannually. The ROC provides reports to the Dean on matters as necessary throughout the year and submits a final report of activities no later than the end of the spring semester. The ROC makes recommendations to the NETL Director for enhancing the safety of nuclear reactor operations. Specific requirements in the Technical Specifications are incorporated in the committee charter, including an audit of present and planned operations. The ROC is chaired by a professor in the Cockrell School of Engineering. ROC membership varies, consisting of ex-officio and appointed positions. The Dean appoints at least three members to the Committee that represent a broad spectrum of expertise appropriate to reactor technology, including personnel external to the School.

11.1.2.1.3 Level 3 Personnel

Level 3 personnel are responsible for managing daily activities at the NETL. The Reactor Supervisor and Health Physicist are Level 3.

Reactor Supervisor

The Reactor Supervisor function is incorporated in a Reactor Manager position, responsible for daily operations, maintenance, scheduling, and training. The Reactor Manager is responsible for the maintenance and daily operations of the reactor, including coordination and performance of activities to meet the Technical Specifications of the reactor license. The Reactor Manager plans and coordinates emergency exercises with first responders and other local support (Austin Fire Department, Austin/Travis County EMS, area hospitals, etc.). The Reactor Manager, assisted by Level 4 personnel and other NETL staff, implements modifications to reactor systems and furnishes design assistance for new experiment systems. The Reactor Manager assists with initial experiment design, fabrication, and setup. The Reactor Manager provides maintenance, repair support, and inventory control of computer, electronic, and mechanical equipment. The Administrative Assistant and Reactor Manager schedule and coordinate facility tours, and support coordination of building maintenance.

Health Physicist

The Health Physicist function is incorporated into a Laboratory Manager position, responsible for radiological protection (Health Physics), safe and effective utilization of the facility (Lab Management), and research support. Each of these three functions is described below. The

Laboratory Manager is functionally responsible to the NETL Associate Director, but maintains a strong reporting relationship to the University Radiation Safety Officer and is a member of the Radiation Safety Committee. This arrangement allows the Health Physicist to operate independently of NETL operational constraints in consideration of radiation safety.

- Health Physics: NETL is a radiological facility operating in the State of Texas under a facility operating license issued by the Nuclear Regulatory Commission (NRC). Radioactive material and activities associated with operation of the reactor are regulated by the NRC, and the uses of radioactive materials at the NETL not associated with the reactor are regulated by the Texas Department of State Health Services (TDSHS). The NETL Health Physicist ensures operations comply with these requirements, and that personnel exposures are maintained ALARA. One or more part-time Undergraduate Research Assistants (URA) may assist as Health Physics Technicians.

- Lab Management: The lab management function is responsible for implementation of occupational safety and health programs at the NETL. The Laboratory Manager supports University educational activities through assistance to student experimenters in their projects by demonstration of the proper radiation work techniques and controls. The Laboratory Manager participates in emergency planning for NETL and the City of Austin to provide basic response requirements and conducts off-site radiation safety training to emergency response personnel such as the Hazardous Materials Division of the Fire Department, and Emergency Medical Services crews.

- Research Support: The mission of The University of Texas at Austin is to achieve excellence in the interrelated areas of undergraduate education, graduate education, research and public service. The Laboratory Manager and research staff supports the research and educational missions of the university at large, as well as development or support of other initiatives. The Laboratory Manager is responsible for coordinating all phases of a project, including proposal and design, fabrication and testing, operation, evaluation, and removal/dismantlement. Researchers are generally focused on accomplishing very specific goals, and the research support function ensures the NETL facilities are utilized in a safe efficient manner to produce quality data. The Laboratory Manager obtains new, funded research programs to promote the capabilities of the neutron beam projects division for academic, government and industrial organizations and/or groups.

11.1.2.1.4 Level 4 Personnel

Reactor Operators and Senior Reactor Operators (RO/SRO) operate and maintain the reactor and associated facilities. An RO/SRO may operate standard reactor experiment facilities as directed by the Reactor Supervisor.

11.1.2.1.5 Other Facility Staff

In addition to the line management positions defined above, NETL staff includes an Administrative Assistant, an Electronics Technician, and variously one or more Undergraduate Research Assistants assigned either non-licensed maintenance support (generally but not necessarily in training for Reactor Operator licensure) or to support the Laboratory Manager as Health Physics Technicians and/or research support.

11.1.2.2 Health Physics Procedures and Document Control

Operation of the radiation protection program is carried out under the direction of the Health Physicist using formal NETL health physics procedures. These procedures are reviewed for adequacy by the Health Physicist and others as appropriate, and are approved by the Facility Director for submission to the Reactor Oversight Committee for review and approval. The original copy of the procedures is maintained by and the distribution of the procedures is managed by the Reactor Supervisor. A current copy is maintained in the reactor control room. The procedures are reviewed periodically and changes are made as necessary. While not intended to be all inclusive, the following list provides an indication of typical radiation protection procedures used in the NETL program:

- Radiation Monitoring - Personnel
- Radiation Monitoring – Facility
- NETL ALARA Program
- Radiation Protection Training
- Radiation Monitoring Equipment
- Radioactive Material Control
- Radiation Work Permits

11.1.2.3 Radiation Protection Training

Individuals who do not have formal training in radiation safety must attend the University's radiation worker training course. The course is approximately eight hours in length. Alternatively, the course may be conducted via computer or over the Internet, or by using video instruction. If these methods of training are used the course will include the same topics as those included in a live course. The Radiation Safety Officer may waive the course if the

individual can provide evidence of equivalent training and/or experience. If the Radiation Safety Officer waives the course, the individual must take the radiation worker refresher course.

The radiation worker refresher course is approximately one hour in length and addresses topics specific to the University such as dosimetry, waste disposal, purchasing, emergency procedures, operating procedures, record keeping, as well as a basic review of radiation safety techniques. Alternatively this course may be conducted via computer or over the Internet, or by using video instruction. If these methods of training are used the course will include the same topics as those included in a live course.

Upon successful completion of either course, credit is posted to the individual's electronic training history in the campus-wide training database. If requested, the successful graduate is issued a certificate of completion.

Radiation safety courses are taught by senior staff of the Radiation Safety Office. At the Nuclear Engineering Teaching Laboratory (NETL), comparable, site-specific radiation worker training is taught by the NETL health physicist. If necessary or desired, outside training specialists may be utilized to present the courses. Subjects covered in the radiation worker training include, but are not limited to the following:

- Atomic Structure and Radioactivity
- Interactions of Radiation with Matter
- Quantities and Units of Radiation
- Basic Principles of Radiation Protection
- Safe Handling of Radioactive Materials and Sources
- Radiation Detection Instruments and Surveys
- Dosimetry
- Waste Disposal
- Purchasing and Receiving Radioactive Materials
- Regulations
- Emergency Procedures
- Record Keeping

The Radiation Safety Officer may also require radiation workers to be trained in other areas, such as general hazard communication (Texas Hazard Communication Act) and laboratory safety. The Radiation Safety Office shall maintain records of course attendance and course credit.

11.1.2.4 Audits of the Radiation Protection Program

Review and audit of the radiation protection program is conducted at least annually by a technically competent person appointed by the Reactor Oversight Committee. The annual radiation protection program audit normally covers areas such as health physics training for NETL staff and users, health physics procedures, personnel monitoring, environmental monitoring, effluent monitoring, operational radiological surveys, instrument calibration, radioactive waste management and disposal, radioactive material transportation, and a review of unusual occurrences. The audit reports are sent to the ROC for review and follow-up action.

11.1.2.5 Health Physics Records and Record Keeping

Radiation protection program records such as radiological survey data sheets, personnel exposure reports, training records, inventories of radioactive materials, environmental monitoring results, waste disposal records, instrument calibration records and many more, are maintained by the Health Physicist. The records will typically be retained for the life of the facility either in hard copy, or on photographic or electronic storage media. Records for the current and previous year are typically retained in the health physicist's office. Other records may be retained in long-term storage. Radiation protection records are reviewed by the health physicist prior to filing. Radiation protection records are used for developing trend analysis, particularly in the personnel dosimetry area, for keeping management informed regarding radiation protection matters, and for reporting to regulatory agencies. In addition, they are used for planning radiation protection related actions, e.g., radiological surveys to preplan work or to evaluate the effectiveness of decontamination or temporary shielding efforts.

11.1.3 ALARA Program

The objectives of the ALARA program are to maintain exposures to ionizing radiation and releases of radioactive effluents at levels that are as low as reasonably achievable (ALARA) within the established dose equivalent and effluent release limits of the appropriate regulatory authority. The management of the NETL does not desire to limit the ability of researchers to perform experiments and participate in reactor operations. However, the management is firmly and unequivocally committed to keeping exposures to personnel and the general public ALARA. The NETL Health Physicist is the individual given explicit responsibility and authority for implementation of the radiation protection and ALARA programs.

In support of ALARA, local occupational dose limits (whole body) have been established as follows:

1. An annual limit, which is the more limiting of:

    a. the total effective dose equivalent being equal to 1 rem (10 mSv); or

    b. the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 1 rem (10 mSv).

2. The annual dose limits to the lens of the eye, to the skin, and to the extremities, which are:

    a. an eye dose equivalent of 1.5 rem (15 mSv), and

    b. a shallow dose equivalent of 5 rem (50 mSv) to the skin or any extremity.

These dose limits may only be exceeded by written permission of the NETL director who will assign a new individual local dose limit for the person.

Procedures provide for a review of all experiments and reactor operations and maintenance activities for radiological considerations by the Health Physicist and Reactor Supervisor.

11.1.4 Radiation Monitoring and Surveying

The radiation monitoring program for the NETL is structured to ensure that all three categories of radiation sources (airborne, liquid and solid) are detected and assessed in a timely manner. To achieve this, the monitoring program is organized such that two major types of radiation surveys are carried out: namely, routine radiation level and contamination level surveys of specific areas and activities within the facility, and special radiation surveys necessary to support non-routine facility operations.

11.1.4.1 Monitoring for Radiation Levels and Contamination

The routine monitoring program is structured to make sure that adequate radiation measurements of both radiation fields and contamination are made on a regular basis. This program includes but is not limited to the following:

Typical surveys for radiation fields:

- Weekly surveys in restricted areas
- Monthly surveys of exterior walls and roof
- Quarterly surveys of non-restricted areas

- Surveys required for certain incoming radioactive materials packages
- Surveys to determine radiological impact of non-routine operations

Typical surveys for contamination:

- Weekly surveys in restricted areas
- Monthly surveys of reactor room roof
- Quarterly surveys of exterior of facility
- Quarterly surveys in non-restricted areas
- Surveys required for certain incoming radioactive materials packages
- Surveys to determine radiological impact of non-routine operations

11.1.4.2 Radiation Monitoring Equipment

Radiation monitoring equipment used in the NETL is summarized below. Because equipment is updated and replaced as technology and performance requires, the equipment listed should be considered representative rather than an exact listing.

Table 11.2, Representative Radiation Detection Instrumentation

Vendor	Model	Range	Purpose/Function
Bicron	Frisk-Tech	0–500,000 cpm	Portable Contamination Survey Instrument
Bicron	Micro-Rem	0–20 mrem/hr	Portable Radiation Survey Instrument
Eberline	R0-2A	0–50 R/hr	Portable Radiation Survey Instrument
Ludlum	12-4	0–10 rem/hr	Portable Neutron Survey Instrument
Eberline	RM-14S	0–5,000,000 cpm	Portable Contamination Survey Instrument
Various PICs		0–200 mrem	Personnel dosimetry
Canberra	Dosicard	N/A	Personnel dosimetry
Victoreen	450B	0–5 R/hr	Portable Radiation Survey Instrument
Eberline	E600	0–1000 R/hr	Extendable Radiation Survey Instrument
Ludlum	375 Dual	0.1–1,000 mrem/hr	Criticality Monitor
Berthold	LB-1043	N/A	Hand/Foot Monitor
Protean	WPC 9550	N/A	Gas Flow Proportional Counter
Wallac	1409	N/A	Liquid Scintillation Counter
P.R.M.	AR-1000	N/A	Ar-41 CAM
Ludlum	333-2	N/A	Particulate CAM
Eberline	RMSII	0.1–10000 mR/hr	Area Radiation Monitor

11.1.4.3 Instrument Calibration

Radiation monitoring instrumentation is calibrated according to written procedures developed from the guidance of industry standards such as ANSI N323A *Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments*. A calibration sticker shall be attached to all calibrated instruments showing the last calibration date, the initials of the person who performed the calibration, and the next calibration due date. The NETL Health Physicist shall maintain all instrument calibration records.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control depends on many different factors including facility design features, operating procedures, training, proper equipment, etc. Training and procedures have been discussed previously under the section dealing with the NETL's radiation protection program. Therefore, this section will focus on design features such as shielding, ventilation, containment and entry control devices for high radiation areas, and will also include protective equipment, personnel dosimetry, and estimates of annual radiation exposure. A description of the dosimetry records used to document facility exposures and a summary of exposure trends at the NETL will also be presented.

11.1.5.1 Shielding

The biological shielding around the NETL reactor is the single biggest design feature in controlling radiation exposure during operation of the facility. The shielding is based on TRIGA® shield designs used successfully at many other similar reactors. The shield has been designed with beam ports to allow extraction of radiation from the core for use in research, education, and service work. When beam port shielding is removed, additional control measures are needed to control radiation exposure. Restricting access to the areas of elevated radiation levels and/or additional shielding are typically used to control radiation exposure. Radiation survey data and the ALARA principle are used determine the appropriate control measures for new configurations as necessary.

11.1.5.2 Containment

Containment of radioactivity within the NETL is primarily a concern with respect to experiments being irradiated in the various irradiation facilities and with the reactor fuel. Containment of fission products within the fuel elements is achieved by maintaining the integrity of the fuel's cladding, which is accomplished by maintaining the fuel and cladding temperatures below specified levels. Containment of other radionuclides generated during use of the irradiation facilities is achieved through strict encapsulation procedures for samples and strict limits on what materials will be irradiated. To further improve containment and minimize the potential release of radioactivity from experiments irradiated in the in-core pneumatic transfer system, the terminal where samples are manually loaded and unloaded is located inside a fume hood. The hood maintains an in-flow of air to prevent the release of radioactivity to the surrounding area.

11.1.5.3 Entry Control

For security purposes, the entire NETL facility perimeter is access controlled. In addition, restricted areas within the NETL are access controlled with unescorted access granted only to trained radiation workers. [redacted]
Most of the restricted areas within the NETL are not high radiation areas. However, in areas which are known high radiation areas, additional measures are in place to control access. The beam port enclosures are the areas typically controlled due to high radiation areas. Entryways to the beam port enclosures are normally locked. When the beam port shutter is open (creating the high radiation area), a conspicuous visible signal is activated at the entryway. If a beam port enclosure entryway is opened, a signal is sent to the control console immediately notifying the reactor operator.

11.1.5.4 Personal Protective Equipment

Typical personal protective equipment used in the NETL radiation protection program consists of anti-contamination items (gloves, lab coats, coveralls, etc.) used when working with unsealed sources of radiation. Other than Ar-41, no airborne radioactive material is expected during normal operation. Thus, no respiratory protection program has been implemented.

11.1.5.5 Representative Annual Radiation Doses

Regulation 10CFR.20.1502 requires monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the limits prescribed in 10CFR20.1201. The regulation also requires monitoring of any individuals entering a high or very high radiation area within which an individual could receive a dose equivalent of 0.1 rem in one hour. According to Regulatory Guide 8.7, if a prospective evaluation of likely doses indicates that an individual is not likely to exceed 10 percent of any applicable limit, then there are no requirements for recordkeeping or reporting. Likewise, Regulatory Guide 8.34 indicates that, if individual monitoring results serve as confirmatory measures, but monitoring is not required by 10CFR20.1502, then such results are not subject to the individual dose recordkeeping requirements of 10CFR20.2106(a) even though they may be used to satisfy 10CFR20.1501 requirements.

The following table lists recent occupational exposures at the NETL. There have been no instances of any exposures in excess of 10 percent of the above limits. Thus, retrospectively, only confirmatory monitoring is required and 10CFR20.2106(a) recordkeeping requirements do not apply, so long as there are no significant changes in the facility, operating procedures, or occupational expectations.

Table 11.3, Representative Occupational Exposures

Numbers of persons in annual-dose categories

Year	Immeasurable	< 0.1 rem	0.1-0.5 rem	> 0.5 rem
2010	13	5	0	0
2009	6	7	0	0
2008	4	9	0	0
2007	8	5	3	0
2006	4	10	2	0
2005	15	22	0	0

Although it appears monitoring of workers is not required, it is the policy of the NETL to monitor workers and members of the public for radiation exposure. Anyone entering a restricted area within the NETL is monitored for radiation exposure with a dosimeter and/or radiation survey and occupancy time data. Although the NETL is likely exempt from record keeping requirements of 10CFR20.2106(a), records of this monitoring are maintained.

11.1.5.5.2 Personnel Dosimetry Devices

Personnel dosimetry devices are available to provide monitoring of all radiation categories likely to be encountered. Direct reading dosimeters (pocket ion chambers or electronic dosimeters) are used by personnel and visitors when in restricted areas. OSL dosimeters with neutron capabilities are assigned to personnel who regularly work in restricted areas. TLD extremity dosimeters are assigned to personnel where extremity exposure may be the dominant issue. The OSL and TLD dosimeters are provided and processed by a NVLAP accredited vendor. Uptakes of radioactive material are not expected during normal operations. Thus, no internal dosimetry program has been implemented.

11.1.6 Contamination Control

Radioactive contamination is controlled at the NETL by using written procedures for radioactive material handling, by using trained personnel, and by operating a monitoring program designed to detect contamination in a timely manner. While there are no accessible areas of the NETL that are routinely grossly contaminated, personnel are trained in contamination detection and control, methods for avoiding contamination, and procedures for handling, storing, and disposing of identified contaminated material. After working in contaminated areas, personnel are required to perform surveys to ensure that no contamination is present on clothing, shoes, etc., before leaving the work location. Activities that are likely to create significant contamination may have special work procedures applied such as a Radiation Work Permit. Contamination events are documented in a special survey report.

11.1.7 Environmental Monitoring

The NETL has routinely performed environmental radiation monitoring throughout its operational history. While many different types of samples have been collected and analyzed, to date there has been no indication that NETL operations have significantly impacted the environment and there are no trends in environmental data which indicate that future impacts

will occur. This result is consistent with expectations for a facility of this type. With the exception of Ar-41, there are virtually no pathways for radioactive materials from the NETL to enter the unrestricted environment during normal facility operations. However, the NETL environmental monitoring program has been structured to provide surveillance over a broad range of environmental media even though there is no credible way the facility could be impacting these portions of the environment. The current environmental monitoring program consists of the following basic components which may change from time to time to meet program objectives:

- Direct gamma radiation measurements performed monthly around the perimeter of the facility.

- Integrated gamma dose measurements using dosimeters located at the perimeter and in the general area of the facility which are exchanged quarterly.

- Ground water sample obtained quarterly from under the reactor structure.

- Monthly contamination monitoring on the roof of the reactor building.

- Quarterly contamination monitoring at the perimeter and in the general area of the facility.

Results of this monitoring are reviewed and records are maintained as part of the radiation monitoring program. In addition, the Texas Department of State Health Services conducts environmental monitoring independently of the NETL program. The TDSHS monitoring program includes quarterly integrated gamma dose using dosimeters at locations around the facility and ground water samples from near the facility. Reports from the TDSHS monitoring are made available to the NETL for comparison with in-house results.

11.2 Radioactive Waste Management

The NETL routinely generates very modest quantities of radioactive waste due to the type of program carried out at the facility and to the fact that a conscious effort is made to keep waste volumes to a minimum. Much of the waste that is generated consists of radioactive materials with a relatively short half-life. Thus, much of the radioactive waste generated at the NETL is held in a restricted area and allowed to decay to background levels and then disposed as non-radioactive waste. Radioactive waste that is not decayed in storage is typically transferred to the university Radiation Safety Office for appropriate disposal.

11.2.1 Radioactive Waste Management Program

The objective of the radioactive waste management program is to ensure that radioactive waste is minimized, and that it is properly handled, stored and disposed of. The NETL health

physicist is responsible for administering the radioactive waste management program. Written procedures address handling, storing and disposing of radioactive waste. The radioactive waste management program is audited as part of the oversight function of the Reactor Oversight Committee. Waste management training is part of both the initial radiation protection training and operator requalification training. Radioactive waste management records are maintained by the health physicist. As stated previously, minimization of radioactive waste is a policy of the NETL. Although there are no numerical volume goals set due to the small volume of waste generated, the health physicist and the reactor supervisor periodically assess operations for the purpose of identifying opportunities or new technologies that will reduce or eliminate the generation of radioactive waste.

11.2.2 Radioactive Waste Controls

At the NETL, radioactive waste is generally considered to be any item or substance which is no longer of use to the facility and which contains radioactivity above the established natural background radioactivity. Because NETL waste volumes are small and the nature of the waste items is limited and reasonably repetitive, there is usually little question about what is or is not radioactive waste. Equipment and components are categorized as waste by the reactor operations staff or health physics staff, while standard consumable supplies like plastic bags, gloves, absorbent material, disposable lab coats, etc., automatically become radioactive waste if detectable radioactivity above background is found to be present. When possible, radioactive waste is initially segregated at the point of origin from items that will not be considered waste. Screening is based on the presence of detectable radioactivity using appropriate monitoring and detection techniques and on the projected future need for the items and materials involved. All items and materials initially categorized as radioactive waste are monitored a second time before packaging for disposal to confirm data needed for waste records, and to provide a final opportunity for decontamination/reclamation of an item. This helps reduce the volume of radioactive waste by eliminating disposal of items that can still be used.

11.2.2.1 Gaseous Waste

Gaseous waste is not created at the NETL under normal operations. Although Ar-41 is released from the NETL stack, this release is not considered to be waste in the same sense as the solid waste which is collected and disposed of by the facility. The Ar-41 is usually classified as an effluent which is a routine part of the normal operation of the NETL reactor.

11.2.2.2 Liquid Waste

Because normal operations create only small volumes of liquid which contain radioactivity, it is typically possible to convert the liquids to a solid waste form. In limited cases, larger volumes of radioactive liquid waste could be generated. In these cases, decay in storage or disposal by the sanitary sewer in accordance with 10CFR20 may be required.

11.2.2.3 Solid Waste

As with most research reactors, solid waste is routinely generated from reactor maintenance operations and irradiations of various experiments. Average annual solid radioactive waste volume produced at the NETL is approximately [redacted]. However, as mentioned previously, much of this waste contains radioactive material with a relatively short half-life. Thus, much of this solid waste is held in a restricted area until it has decayed to background levels of radioactivity. Once decayed and surveyed to confirm background levels of radioactivity, the waste is disposed as non-radioactive. The remaining solid waste which contains radioactive materials with a relatively long half-life typically amounts to approximately two cubic feet per year. Appropriate radiation monitoring instrumentation will be used for identifying and segregating solid radioactive waste. Solid radioactive waste to be held for decay is typically packaged in plastic bags, labeled appropriately, and moved to a designate storage area within a restricted area. Solid radioactive waste to be transferred for disposal is packaged according to USDOT, waste processor, and disposal site requirements as applicable and is temporarily stored in a restricted area until transfer for disposal. No solid radioactive waste is intended to be retained or permanently stored on site.

11.2.2.4 Mixed Waste

As mixed waste has in addition to being radioactive, the characteristic of being chemically hazardous and falling under RCRA regulations, great care is taken at the NETL to avoid generating mixed waste whenever possible. However, generation of mixed waste cannot be completely avoided. The University of Texas at Austin is considered a RCRA "Large Quantity Generator." Thus, any mixed waste generated at the NETL must be disposed within 90 days. Processes that may generate mixed waste are reviewed with the intent of modifying the process or substituting materials were appropriate to minimize the mixed waste generated. In many cases, the mixed waste contains radioactive materials with a half-life such that decay to background levels within the 90-day disposal requirement is possible. Where decay is not an option, the mixed waste is packaged appropriately and transferred to the university Radiation Safety Office for disposal.

11.2.2.5 Decommissioning Waste

There is no intention of decommissioning the NETL in the near future. Thus, there is no expectation of decommissioning waste being generated.

11.2.3 Release of Radioactive Waste

Controlled releases of radioactive waste to the environment are not a routine occurrence at the NETL. However, there is the possibility of infrequent releases of liquid waste to the sanitary sewer in compliance with applicable regulations. The typical release of radioactive waste from the NETL is via transfer of solid waste to the university Radiation Safety Office for appropriate disposal.

12 CONDUCT OF OPERATIONS

12.1 ORGANIZATON

This chapter describes and discusses the Conduct of Operations at the University of Texas TRIGA. 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. License is used in Chapter 12 in reference to reactor operators and senior reactors subject to 10CFR50.55 requirements.

12.1.1 Structure

12.1.1.1 University Administration

Fig. 12.1 illustrates the organizational structure that is applied to the management and operation of the University of Texas and the reactor facility. Responsibility for the safe operation of the reactor facility is a function of the management structure of Fig. 12.1[1]. These responsibilities include safeguarding the public and staff from undue radiation exposures and adherence to license or other operation constraints. Functional organization separates the responsibilities of academic functions and business functions. The office of the President administers these activities and other activities through several vice presidents.

12.1.1.2 NETL Facility Administration

The facility administrative structure is shown in Fig. 12.2. Facility operation staff is an organization of a director and at least four full time equivalent persons. This staff of four provides for basic operation requirements. Four typical staff positions consist of an associate director, a reactor supervisor, a reactor operator, and a health physicist. One or more of the listed positions may also include duties typical of a research scientist. The reactor supervisor, health physicist, and one other position are to be full time. One full time equivalent position may consist of several part-time persons such as assistants, technicians and secretaries. Faculty, students, and researchers supplement the organization. Titles for staff positions are descriptive and may vary from actual designations. Descriptions of key components of the organization follow.

[1] "Standard for Administrative Controls" ANSI/ANS - 15.18 1979

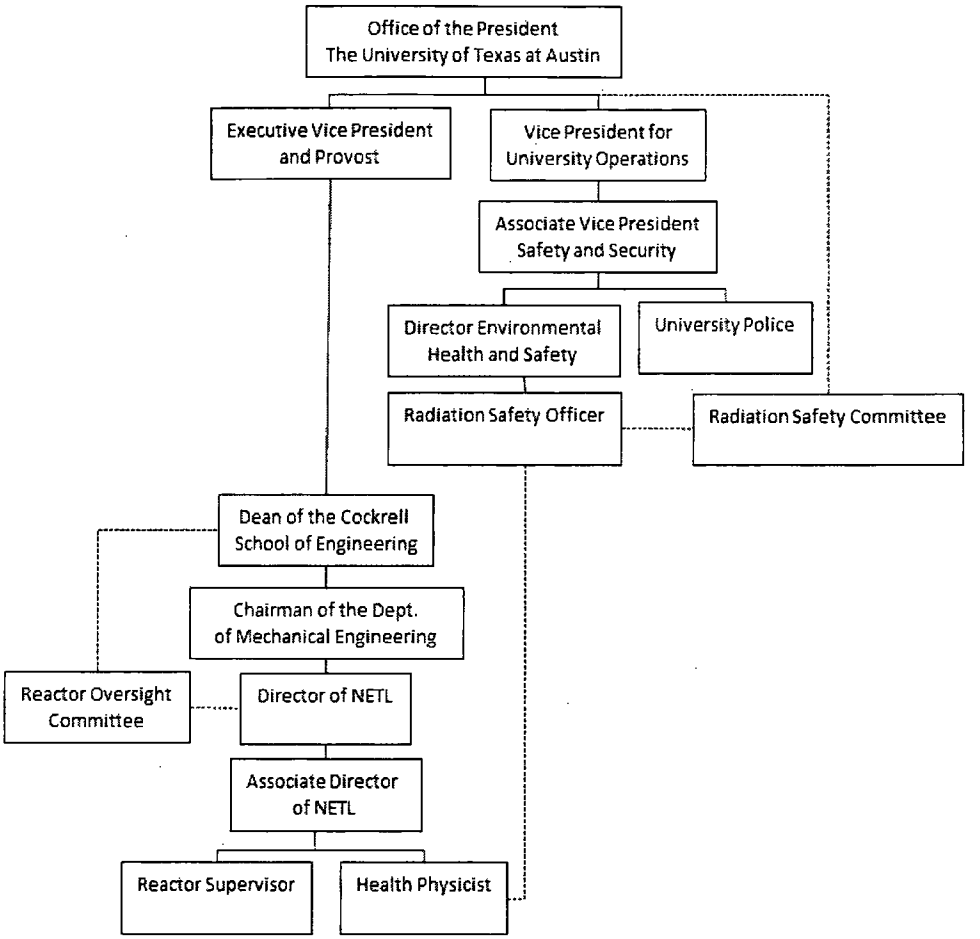


Figure 12.1, University Administration

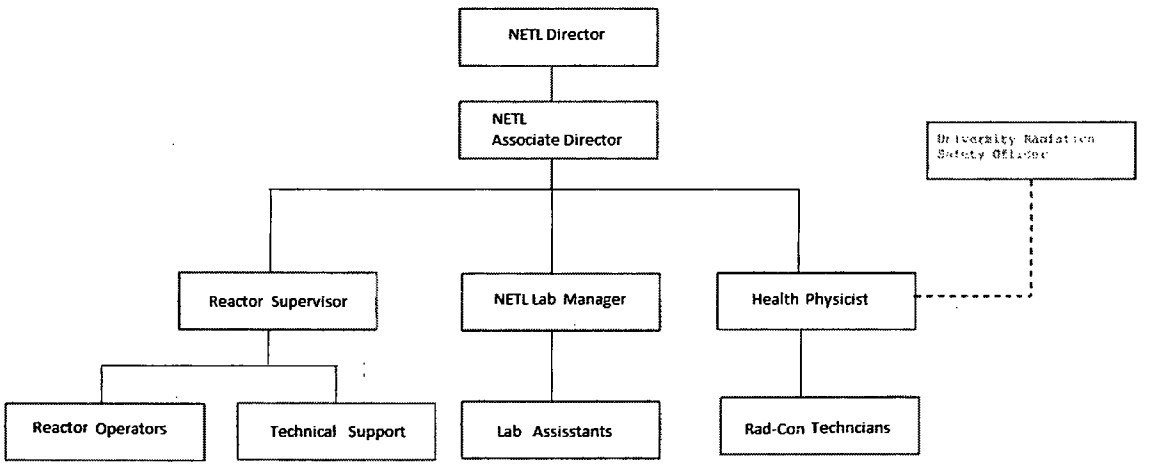


Figure 12.2, NETL Facility Administration

12.1.2 Responsibility

12.1.2.1 Executive Vice President and Provost

Research and academic educational programs are administered through the Office of the Executive Vice President and Provost. Separate officers assist with the administration of research activities and academic affairs with functions delegated to the Dean of the Cockrell School of Engineering and Chairman of the Mechanical Engineering Department.

12.1.2.2. Vice President for University Operations

University operations activities are administered through the Office of the Vice President for Operations. This office is responsible for multiple operational functions of the University including university support programs, human resources, campus safety and security, campus real estate, and campus planning and facilities management.

12.1.2.3 Associate Vice President Campus Safety and Security

The associate vice president for campus safety and security oversees multiple aspects of safety and security on campus including environmental health and safety, campus police, parking and transportation, fire prevention, and emergency preparedness.

12.1.2.4 Director of Nuclear Engineering Teaching Laboratory

Nuclear Engineering Teaching Laboratory programs are directed by a senior classified staff member or faculty member. The director oversees strategic guidance of the Nuclear Engineering Teaching Laboratory including aspects of facility operations, research, and service work. The director must interact with senior University of Texas at Austin management regarding issues related to the Nuclear Engineering Teaching Laboratory.

12.1.2.5 Associate Director of Nuclear Engineering Laboratory

The Associate Director performs the day to day duties of directing the activities of the facility. The Associate Director is knowledgeable of regulatory requirements, license conditions, and standard operating practices. The associate director will also be involved in soliciting and carrying out research utilizing the reactor and other specialized equipment at the Nuclear Engineering Teaching Laboratory.

12.1.2.6 Reactor Oversight Committee

The Reactor Oversight Committee is established through the Office of the Dean of the Cockrell School of Engineering of The University of Texas at Austin. Broad responsibilities of the committee include the evaluation, review, and approval of facility standards for safe operation.

The Dean shall appoint at least three members to the Committee that represent a broad spectrum of expertise appropriate to reactor technology. The committee will meet at least twice each calendar year or more frequently as circumstances warrant. The Reactor Oversight Committee shall be consulted by the Nuclear Engineering Teaching Laboratory concerning unusual or exceptional actions that affect administration of the reactor program.

12.1.2.7 Radiation Safety Officer

A Radiation Safety Officer acts as the delegated authority of the Radiation Safety Committee in the daily implementation of policies and practices regarding the safe use of radioisotopes and sources of radiation as determined by the Radiation Safety Committee. The Radiation Safety Program is administered through the University Environmental Health and Safety division. The responsibilities of the Radiation Safety Officer are outlined in The University of Texas at Austin Manual of Radiation Safety.

12.1.2.8 Radiation Safety Committee

The Radiation Safety Committee is established through the Office of the President of The University of Texas at Austin. Responsibilities of the committee are broad and include all policies and practices regarding the license, purchase, shipment, use, monitoring, disposal, and transfer of radioisotopes or sources of ionizing radiation at The University of Texas at Austin.

The President shall appoint at least three members to the Committee and appoint one as Chairperson. The Committee will meet at least once each year on a called basis or as required to approve formally applications to use radioactive materials. The Radiation Safety Committee shall be consulted by the University Safety Office concerning any unusual or exceptional action that affects the administration of the Radiation Safety Program.

12.1.2.9 Reactor Supervisor

Reactor operation at the Nuclear Engineering Teaching Laboratory is directed by a Reactor Supervisor. Responsibilities of the Reactor Supervisor include control of license documentation, reactor operation, equipment maintenance, experiment operation, and instruction of persons with access to laboratory areas.

Activities of reactor operators with USNRC licenses will be subject to the direction of a person with a USNRC senior operator license. The Reactor Supervisor shall be qualified as a senior

operator. This person is to be knowledgeable of regulatory requirements, license conditions, and standard operating practices.

12.1.2.10 Health Physicist

Radiological safety of the Nuclear Engineering Teaching Laboratory is monitored by a health physicist, who will be knowledgeable of the facility radiological hazards. Responsibilities of the health physicist will include calibration of radiation detection instruments, measurements of radiation levels, control of radioactive contamination, maintenance of radiation records, and assistance with other facility monitoring activities.

Activities of the health physicist will depend on two conditions. One condition will be the normal operation responsibilities determined by the director of the facility. A second condition will be communications specified by the radiation safety officer. This combination of responsibility and communication provides for safety program implementation by the director, but establishes independent review. The health physicist's activities will meet the requirements of the director and the policies of an independent university safety organization.

12.1.2.11 Laboratory Manager

Laboratory operations and research support is provide by a designated Laboratory Manager. The function is typically combined with the Health Physicist position.

12.1.2.12 Reactor Operators

Reactor operators (and senior reactor operators) are licensed by the USNRC to operate the UT TREIGA II nuclear research reactor. University staff and/or students may be employed as reactor operators.

12.1.2.13 Technical Support

Staff positions supporting various aspects of facility operations are assigned as required.

12.1.2.14 Radiological Controls Technicians

Radiological Controls Technicians are supervised by the Health Physicist to perform radiological controls and monitoring functions. Radiological Controls Technicians are generally supported as Undergraduate Research Assistant positions.

12.1.2.15 Laboratory Assistants

Laboratory Assistants are supervised by the Laboratory Manager to perform laboratory operations and analysis. Laboratory Assistants are generally supported as Undergraduate Research Assistant positions.

12.1.3 Staffing

Operation of the reactor and activities associated with the reactor, control system, instrument system, radiation monitoring system, and engineered safety features will be the function of staff personnel with the appropriate training and certification[2].

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as Reactor Supervisor. The Supervisor may be on call if capable of arriving at the facility within thirty minutes and cognizant of reactor operations. The Reactor Supervisor shall directly supervise:

a. All fuel element or control rod relocations or installations within the reactor core region, and subsequent initial startup and approach to power.

b. Relocation or installation of any experiment in the core region with a reactivity worth of greater than one dollar, and subsequent initial startup and approach to power.

c. Recovery from an unscheduled shutdown or significant power reductions,

d. All initial startup and approach to power following modifications to reactor safety or control rod drive systems.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations. All activities that require the presence of licensed operators will also require the presence in the facility complex of a second person capable of performing prescribed written instructions.

Only the Reactor Operator at the controls or personnel authorized by, and under direct supervision of, the Reactor Operator at the controls shall manipulate the controls. Whenever the reactor is not secured, operation of equipment that has the potential to affect reactivity or power level shall be manipulated only with the knowledge and consent of the Reactor Operator at the controls. The Reactor Operator at the controls may authorize persons to manipulate reactivity controls who are training either as (1) a student enrolled in academic or industry

[2] "Selection and Training of Personnel for Research Reactors", ANSI/ANS -15.4 - 1970 (N380)

course making use of the reactor, (2) to qualify for an operator license, or (3) in accordance the approved Reactor Operator requalification program.

Whenever the reactor is not secured, a second person (i.e., in addition to the reactor operator at the control console) capable of initiating the Reactor Emergency Plan will be present in the NETL building. Unexpected absence of this second person for greater than two hours will be acceptable if immediate action is taken to obtain a replacement.

Staffing required for performing experiments with the reactor will be determined by a classification system specified for the experiments. Requirements will range from the presence of a certified operator for some routine experiments to the presence of a senior operator and the experimenter for other less routine experiments.

12.1.4 Selection and Training of Personnel

12.1.4.1 Qualifications

Personnel associated with the research reactor facility[3] shall have a combination of academic training, experience, skills, and health commensurate with the responsibility to provide reasonable assurance that decisions and actions during all normal and abnormal conditions will be such that the facility and reactor are operated in a safe manner.

12.1.4.2 Job Descriptions

Qualifications for University positions are incorporated in job descriptions, summarizing function and scope. The typical description includes title, duties, supervision, education, experience, equipment, working conditions, and other special requirements for the job position. Student employment is typically under the general description of Undergraduate or Graduate Research Assistant, with minimal specification to accommodate a wide range of jobs.

12.1.4.2.1 Facility Director

A combination of academic training and nuclear experience will fulfill the qualifications for the individual identified as the facility director. A total of six years' experience will be required. Academic training in engineering or science, with completion of a baccalaureate degree, may account for up to four of the six years' experience. The director is generally a faculty member with a Ph.D. in nuclear engineering or a related field.

[3] ANS/ANSI-15.4, op. cit.

12.1.4.2.2 Associate Director

A combination of academic training and nuclear experience will fulfill the qualifications for the individual identified as the facility director. Academic training in engineering or science, with operating and management experience at a research reactor is required. The Associate Director will be qualified by certification as a senior operator and is typically a person with at least one graduate degree in nuclear engineering or a related field.

12.1.4.2.3 Reactor Supervisor

A person with special training to supervise reactor operation and related functions will be designated as the reactor supervisor. The reactor supervisor will be qualified by certification as a senior operator as determined by the licensing agency. Additional academic or nuclear experience will be required as necessary for the supervisor to perform adequately the duties associated with facility activities. The supervisor is typically a person with at least one graduate degree in nuclear engineering or a related field.

12.1.4.2.4 Health Physicist

A person with a degree related to health, safety, or engineering, or sufficient experience that is appropriate to the job requirements will be assigned the position of health physicist. A degree in health physics or similar field of study and some experience is preferred. Certification is not a qualification, but work towards certification should be considered a requirement.

12.1.4.3.4 Laboratory Manager

Laboratory operations and research support id provide by a designated Laboratory Manager. The function is typically combined with the Health Physicist position.

12.1.2.12 Reactor Operators

Reactor operators (and senior reactor operators) are licensed by the USNRC to operate the UT TREIGA II nuclear research reactor. Training and requalification requirements are indicated below.

12.1.2.13 Technical Support

Staff positions supporting various aspects of facility operations are assigned as required. Selection, qualification and training are on a case by case basis.

12.1.2.14 Radiological Controls Technicians

Radiological Controls Technicians training is provided in the Radiation Protection Program.

12.1.2.15 Laboratory Assistants

Laboratory Assistants are supervised by the Laboratory Manager to perform laboratory operations and analysis, with specific training requirements related to job responsibilities. .

12.1.5 Radiation Safety

Protection of personnel and the general public against hazards of radioactivity and fire is established through the safety programs of the University Safety Office. Safety programs at the reactor facility supplement the university programs so that appropriate safety measures are established for the special characteristics of the facility[4] [5].

Safety programs are operated as a function of the Vice President for University Operations and include a radiation safety organization as presented in Fig. 12.1. Radiation protection at the reactor facility is the responsibility of the Reactor Supervisor, Health Physicist, or a designated senior operator in charge of operation activities. The person responsible for radiation protection at the reactor facility will have access to other individuals or groups responsible for Radiological safety at the University. Contact with the Radiation Safety Officer will occur on an as needed basis and contact with the Reactor Oversight Committee will occur on a periodic basis. Responsibility includes the authority to act on questions of radiation protection, the Acquisition of appropriate training for radiation protection and the reporting to management of problems associated with radiation protection. Radiological management policies and programs are described in Chapter 11.

12.2 REVIEW AND AUDIT ACTIVITES

The review and audit process is the responsibility of the Reactor Oversight Committee (ROC).

12.2.1 Composition and Qualifications

The ROC shall consist of at least three (3) members appointed by the Dean of the Cockrell School of Engineering that are knowledgeable in fields which relate to nuclear safety. The university radiological safety officer shall be a member or an ex-officio member. The committee will perform the functions of review and audit or designate a knowledgeable person for audit functions.

[4] "Radiological Control at Research Reactor Facilities", ANSI/ANS-15.11 1977(N628)
[5] "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents", ANSI/ANS - 15.12 1977(N647)

12.2.2 Charter and Rules

The operations of the ROC shall be in accordance with an established charter, including provisions for:

a. Meeting frequency (at least twice each year, with approximately 4-8 month frequency).

b. Quorums (not less than one-half the membership where the operating staff does not contribute a majority).

c. Dissemination, review, and approval of minutes.

d. Use of subgroups.

12.2.3 Review Function

The responsibilities of the Reactor Safeguards Committee to shall include but are not limited to review of the following:

a) All new procedures (and major revisions of procedures) with safety significance

b) Proposed changes or modifications to reactor facility equipment, or systems having safety significance

c) Proposed new (or revised) experiments, or classes of experiments, that could affect reactivity or result in the release of radioactivity

d) Determination of whether items a) through c) involve unreviewed safety questions, changes in the facility as designed, or changes in Technical Specifications.

e) Violations of Technical Specifications or the facility operating licensee

f) Violations of internal procedures or instruction having safety significance

g) Reportable occurrences

h) Audit reports

Minor changes to procedures and experiments that do not change the intent and do not significantly increase the potential consequences may be accomplished following review and approval by a senior reactor operator and independently by one of the Reactor Supervisor, Associate Director or Director. These changes should be reviewed at the next scheduled meeting of the Reactor Oversight Committee.

12.2.4 Audit Function

The audit function shall be a selected examination of operating records, logs, or other documents. Audits will be by a Reactor Oversight Committee member or by an individual appointed by the committee to perform the audit. The audit should be by any individual not directly responsible for the records and may include discussions with cognizant personnel or observation of operations. The following items shall be audited and a report made within 3 months to the Director and Reactor Committee:

a. Conformance of facility operations with license and technical specifications at least once each calendar year.

b. Results of actions to correct deficiencies that may occur in reactor facility equipment, structures, systems, or methods of operation that affect safety at least once per calendar year.

c. Function of the retraining and requalification program for reactor operators at least once every other calendar year.

d. The reactor facility emergency plan and physical security plan, and implementing procedures at least once every other year.

12.3 PROCEDURES

Written procedures shall govern many of the activities associated with reactor operation. Activities subject to written procedures will include:

a) Startup, operation, and shutdown of the reactor

b) Fuel loading, unloading, and movement within the reactor.

c) Control rod removal or replacement.

d) Routine maintenance, testing, and calibration of control rod drives and other systems that could have an effect on reactor safety.

e) Administrative controls for operations, maintenance, conduct of experiments, and conduct of tours of the Reactor Facility.

f) Implementing procedures for the Emergency Plan or Physical Security Plan.

Written procedures shall also govern:

a) Personnel radiation protection, in accordance with the Radiation Protection Program as indicated in Chapter 11

b) Administrative controls for operations and maintenance

c) Administrative controls for the conduct of irradiations and experiments that could affect core safety or reactivity

A master Procedure Control procedure specifies the process for creating, changing, editing, and distributing procedures. Preparation of the procedures and minor modifications of the procedures will be by certified operators. Substantive changes or major modifications to procedures, and new prepared procedures will be submitted to the Reactor Oversight Committee for review and approval. Temporary deviations from the procedures may be made by the reactor supervisor or designated senior operator provided changes of substance are reported for review and approval.

Proposed experiments will be submitted to the reactor oversight committee for review and approval of the experiment and its safety analysis[6], as indicated in Chapter 10. Substantive changes to approved experiments will require re-approval while insignificant changes that do not alter experiment safety may be approved by a senior operator and independently one of the following, Reactor Supervisor, Associate Director, or Director. Experiments will be approved first as proposed experiments for one time application, and subsequently, as approved experiments for repeated applications following a review of the results and experience of the initial experiment implementation.

12.4 REQUIRED ACTIONS

This section lists the actions required in the event of certain occurrences.

12.4.1 Safety Limit Violation

In the event that a Safety Limit is not met,

a. The reactor shall be shutdown, and reactor operations secured.

b. The Reactor Supervisor, Associate Director, and Director shall be notified

c. The safety limit violation shall be reported to the Nuclear Regulatory Commission within 24 hours by telephone, confirmed via written statement by email, fax or telegraph

d. A safety limit violation report shall be prepared within 14 days of the event to describe:

[6] ANSI/ANS 15.6, op. cit.

1. Applicable circumstances leading to the violation including (where known) cause and contributing factors
2. Effect of the violation on reactor facility components, systems, and structures
3. Effect of the violation on the health and safety of the personnel and the public
4. Corrective action taken to prevent recurrence

e. The Reactor Oversight Committee shall review the report and any followup reports

f. The report and any followup reports shall be submitted to the Nuclear Regulatory Commission.

g. Operations shall not resume until the USNRC approves resumption.

12.4.2 Release of Radioactivity Above Allowable Limits

Actions to be taken in the case of release of radioactivity from the site above allowable limits shall include a return to normal operation or reactor shutdown until authorized by management if necessary to correct the occurrence. A prompt report to management and license authority shall be made. A review of the event by the Reactor Oversight Committee should occur at the next scheduled meeting. Prompt reporting of the event shall be by telephone and confirmed by written correspondence within 24 hours. A written follow up report is to be submitted within 14 days.

12.4.3 Other Reportable Occurrences

In the event of a reportable occurrence, as defined in the Technical Specifications, and in addition to the reporting requirements,

a. The Reactor Supervisor, the Associate Director and the Director shall be notified
b. If a reactor shutdown is required, resumption of normal operations shall be authorized by the Associate Director or Director
c. The event shall be reviewed by the Reactor Oversight Committee during a normally scheduled meeting

12.5 REPORTS

This section describes the reports required to NRC, including report content, timing of reports, and report format. Refer to section 12.4 above for the reporting requirements for safety limit violations, radioactivity releases above allowable limits, and reportable occurrences. All written reports shall be sent within prescribed intervals to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

12.5.1 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to licensing authorities within three months following the end of each prescribed year. Each annual operating report shall include the following information:

a. A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests or experiments, or both, that are significantly different from those performed previously, including conclusions that no new or unanalyzed safety questions were identified.

e. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

f. A summarized result of environmental surveys performed outside the facility.

g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

12.5.2 Other or Special Reports

A written report within 30 days to the chartering or licensing authorities of:

a. Permanent changes in the facility organization involving Director or Supervisor.

b Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

12.6. RECORDS

Records of the following activities shall be maintained and retained for the periods specified below[7]. The records may be in the form of logs, data sheets, electronic files, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

12.6.1. Lifetime Records

Lifetime records are records to be retained for the lifetime of the reactor facility. (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

a. Gaseous and liquid radioactive effluents released to the environs.

b. Offsite environmental monitoring surveys required by Technical Specifications.

c. Events that impact or effect decommissioning of the facility.

d. Radiation exposure for all personnel monitored.

e. Updated drawings of the reactor facility.

12.6.2 Five Year Period

Records to be retained for a period of at least five years or for the life of the component involved whichever is shorter.

a. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least one year).

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Reactor facility radiation and contamination surveys where required by applicable

[7] "Records and Reports for Research Reactors", ANSI/ANS - 15.3-1974 (N399).

regulations.

f. Experiments performed with the reactor.

g. Fuel inventories, receipts, and shipments.

h. Approved changes in operating procedures.

i. Records of meeting and audit reports of the review and audit group.

12.6.3 One Training Cycle

Training records to be retained for at least one license cycle are the requalification records of licensed operations personnel. Records of the most recent complete cycle shall be maintained at all times the individual is employed.

12.7 EMERGENCY PLANNING

Emergency planning is guided by an NRC approved Emergency Plan following the general guidance set forth in ANSI/ ANS15.16, Emergency Planning for Research Reactors. The plan specifies two action levels, the first level being a locally defined Non-Reactor Specific Event, and the second level being the lowest level FEMA classification, a Notification of Unusual Event. Procedures reviewed and approved by the reactor Oversight Committee are established to manage implementation of emergency response.

12.8 SECURITY PLANNING

Security planning is guided by an NRC approved Security Plan. The plan incorporates compensatory measures implemented following security posture changes initiated post 9/11. The Plan and portions of the procedures are classified as Safeguards Information. Security procedures implementing the plan, approved by the Reactor Oversight Committee, are established.

12.9 QUALITY ASSURANCE

Objectives of quality assurance (QA) may be divided into two major goals. First is the goal of safe operation of equipment and activities to prevent or mitigate an impact on public health and safety. Second is the reliable operation of equipment and activities associated with education and research functions of the University. The risk or potential release of radioactive materials is the primary impact on public health and safety, and may be divided into direct risks and indirect risks. Direct risks are activities such as waste disposal, fuel transport and decommissioning that introduce radioactive materials into the public domain. Indirect risks are accident conditions created by normal or abnormal operating conditions that generate the potential or actual release of radioactive materials from the controlled areas of a facility.

Quality assurance program procedures have been developed that apply to items or activities determined to be safety-related follows the guidelines of Reg. Guide 2.5[8] [9]. Specific procedures apply to fuel shipment and receipt, a general procedure guides unspecified safety related activities.

12.10. OPERATOR REQUALIFICATION

Regulatory requirements and standards provide guidance for requalification training. Specific regulatory requirements are found in 10CFR55 for the licensing of operators and senior operators with regulations for requalification set forth in section 55.59. Standards for the selection and training of facility personnel and reactor operators are available. Specific regulations in the form of two sets of license conditions also apply to the facility personnel and reactor operators. One set of conditions for the facility license, 10CFR 50.54, applies to facility personnel. The other set of conditions for individual licenses, 10CFR 55.53 applies to operators and senior operators.

An NRC approved UT TRIGA Requalification Plan is used to maintain training and qualification of reactor operators and senior reactor operators. License qualification by written and operating test, and license issuance or removal, are the responsibility of the U.S. Nuclear Regulatory Commission. No rights of the license may be assigned or otherwise transferred and the licensee is subject to and shall observe all rules, regulations and orders of the Commission. Requalification training maintains the skills and knowledge of operators and senior operators during the period of the license. Training also provides for the initial license qualification.

Active status of any licensee requires successful participation in the UT Operator Requalification program. A process is in place to manage re-establishment of active status where conditions of an active license status are not met.

The program addresses training by lectures, instruction, discussion and self-study. The program addresses training topics. The program establishes requirement for a biennial schedule of activities. The program addresses on the job training. The program requires:

a. Observation at least once each year of a satisfactory understanding of the reactivity control system and knowledge of operating procedures.

b. Each operator or senior operator will review facility design changes, procedure changes and license changes as they occur or once each 6 to 8 months.

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[8] "Quality Assurance Requirements for Research Reactors", Nuclear Regulatory Guide 2.5 (77/05).

[9] "Quality Assurance Program Requirements for Research Reactors," ANSI/ANS - 15.8 - 1976 (N402).

c. A review of the contents of abnormal and emergency procedures will be done by each operator or senior operator at 6 to 8 month intervals so that at least 3 reviews occur during the two year training cycle.

The program addresses performance evaluation of on annual examination and periodic observations, including methods to address deficiencies identified in evaluation. The program addresses records to be generated, including required information and retention schedule.

12.11 STARTUP PROGRAM

Startup and testing of the Balcones Research Center TRIGA facility was completed in 1992, therefore a startup plan is not applicable.

12.12 ENVIRONMENTAL REPORT

The Environmental Report is provided as a separate document.

13.0 ACCIDENT ANALYSIS

This chapter provides information and analysis to demonstrate that the health and safety of the public and workers are not challenged by equipment malfunctions or other abnormalities in reactor performance. The analysis demonstrates that facility design features, limiting safety system settings, and limiting conditions for operation ensure that unacceptable radiological consequences to the general public, facility personnel or the environment will not occur as a result of credible accidents. Reference values for physical properties and values used in analysis are provided in 13.1. An overview of accident scenarios is provided in 13.2, followed by detailed analyses.

13.1 Notation and Fuel Properties

Tables 13.1-13.3 identify physical characteristics of the TRIGA Mark II fuel. Table 13.4 identifies the assumptions and design basis values used in the accident analyses.

Table 13.1. Neutronic Properties of TRIGA MkII $ZrH_{1.6}$ Fuel Elements.

Property	Symbol	Value
Effective delayed neutron fractions	β	0.007
Effective neutron lifetime	ℓ	43 μsec
Temperature coefficient of reactivity	α	-0.000115 K^{-1}

Source: West et al. (1967).

Table 13.2, Dimensions of TRIGA MkII $ZrH_{1.6}$ Fuel Elements.

Property of Individual Element	Symbol	Value
Length of fuel zone	L_f	[redacted]
Fuel radius	r_i	[redacted]
Clad outside radius	r_o	[redacted]
Fuel volume	V_f	[redacted]
Clad volume	V_c	[redacted]
Fuel mass	M_f	[redacted]
Clad mass	M_c	[redacted]
Wt. Fraction U in fuel	x_u	[redacted]
Wt. Fraction $ZrH_{1.6}$ in fuel	x_m	[redacted]

Table 13.3, Thermal and Mechanical Properties of TRIGA MkII $ZrH_{1.6}$ Fuel Elements and Type 304 Stainless Steel Cladding.

Property	Symbol	Value	Temp.
Fuel			
Density	ρ_f	[redacted]	
Thermal conductivity	k_f	[redacted]	All
Heat capacity, c_{pf} = 340.1 + 0.6952T(°C)	c_{pf}	[redacted]	0 °C
Cladding			
Density	ρ_c	[redacted]	300 K
Thermal conductivity	k_c	[redacted]	300 K
		[redacted]	400 K
		[redacted]	600 K
Heat capacity	c_{pc}	[redacted]	300 K
		[redacted]	400 K
Yield strength		[redacted]	400 °C
Tensile strength		[redacted]	400 °C

Source: fuel properties from Simnad (1980); cladding properties from Incropera and DeWitt (1990) and from Metals Handbook (1961).

Table 13.4, UT TRIGA Core-Conditions Basis for Calculations.

Parameter	Value
Steady state maximum power, P_o	1,100 kW
Fuel mass per element	2.367 kg
Heat capacity per element at T (°C)	$805.0 + 1.646T$ (J K^{-1})
Minimum number of fuel elements, N	83
Core radial peaking factor	2
Axial peaking factor	$\pi/2$
Excess reactivity	\$4.00 (2.8% Δk/k)
Maximum pulsing reactivity insertion	\$3.00 (2.1% Δk/k)
Excess reactivity at 500 kW maximum power[a]	\$1.16 (0.81% Δk/k)
Fuel average temperature at 500 kW maximum	285 °C

[a]Source: Data from GA Torrey Pines TRIGA reactor

13.2 Accident Initiating Events and Scenarios

Three accident scenarios were identified in the initial licensing of the University of Texas TRIGA reactor in 1992: maximum hypothetical accident (fuel element failure in air), insertion of excess of reactivity, and loss of coolant. The current accident analysis substantially reprises the original, with updates to the methodology based on current standards.

NUREG/CR-2387 (*Credible Accident Analyses for TRGIA and TRIGA Fueled Reactors*, Hawley & Kathren, 1982) was the definitive work in identifying and evaluating the spectrum of accidents to be addressed for TRIGA reactors, addressing seven scenarios:

- Excess reactivity insertion
- Metal-water reactions
- Lost/misplaced or inadvertent experiments
- Mechanical rearrangement of the core
- Loss of coolant accident
- Changes in fuel morphology and ZrH_x composition
- Fuel handling accident

NUREG-1537 (*Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*) provides guidance for format and contents as well as a standard review plan. The spectrum of accidents specified in NUREG-1537 includes:

- Maximum Hypothetical Accident

  A fuel handling accident is considered to lead to the maximum hypothetical accident, with consequences analyzed in section 13.3.

- Insertion of Excess Reactivity

  Excess reactivity insertion accidents are analyzed for the UT TRIGA in section 13.4. Rapid insertion of reactivity into a TRIGA reactor is a designed feature of the fuel performance. Thus, most plausible reactivity accidents do not subject the fuel to conditions more severe than normal operating situations. Insertion of excess reactivity is considered for two sets of initial conditions. First, the maximum reactivity addition of 2.8% Δk/k/k ($4.00) from operations below feedback range is considered with respect to maximum fuel temperature. A second initial condition is considered where the 2.8% Δk/k/k ($4.00) reactivity addition occurs with the core operating at a power level equivalent to the balance of the core excess reactivity Analysis demonstrates that maximum fuel temperature does not exceed acceptable limits. An administrative limit for experiments of $3.00 assures that experiment removal cannot exceed the analysis by a large margin.

- Loss of Coolant

  Loss of coolant accident is analyzed in 13.5. A loss of coolant accident is analyzed to demonstrate that maximum fuel temperature does not exceed acceptable limits. Cooling in air is considered and the results are compared to limiting fuel temperatures. Doses from scattered radiation from the uncovered core are analyzed.

- Loss of Coolant Flow

  Loss of coolant flow is analyzed in 13.6.

- Mishandling or Malfunction of Fuel

  Transport of fission products released in the pool is significantly affected by pool water. Failure during operation would occur under water, leading ultimately to atmospheric release of fission products. A failure of an element immersed in pool water would be partially retain and/or retard migration of gaseous material and substantially retain particulate material. Consequently, the failure in water is bounded by the failure in air. Fuel handling accident (the maximum hypothetical accident) is considered in section 13.3.

  NUREG/CR-2387 identifies two categories of specific fuel failures, changes in fuel morphology and metal-water reactions. Changes in fuel morphology and ZrH_x composition for fuel used by the UT TRIGA reactor is not credible for accident scenarios, as described in Chapter 4. As noted in the NUREG and in Chapter 4, significant metal water reactions are not possible at TRIGA operating temperatures.

- Loss of Normal Electric Power

  A loss of normal electric power is analyzed in section 13.7.

- External Events

  External events are analyzed in section 13.8. External events are considered with respect to potential mechanical rearrangement of the core (specified in NUREG/CR-2387). As described in Chapter 4, the core support structure is secured to the floor, the core is surrounded by a canister of graphite, and fuel elements are positioned in the core by the upper and lower grid plate approximately 10 in. thick. There is no credible scenario that would disturb the core lattice or structure while simultaneously retaining fuel elements in a critical geometry.

- Mishandling or malfunction of experiment

  Experiment mishandling or malfunction is described in section 13.9. Lost/misplaced or inadvertent experiments; administrative controls on experiments as described in Chapter 10 require an assessment of personnel and facility hazard, with specific limits on potential hazard to personnel and the facility

13.3 Maximum Hypothetical Accidents, Single Element Failure in Air

The maximum hypothetical accident for a TRIGA reactor is the failure of the encapsulation of one fuel element, in air, resulting in the release of airborne fission products to the reactor bay and the environment. Failure in air could result from a fuel-handling accident or from a failure during operation in the core following a loss of coolant accident. This section addresses potential consequences, should failure occur in air.

13.3.1 Assumptions

- Fuel mass of 1 metric ton is used in burnup calculations; results are scaled to actual fuel mass in determining fission product inventory.

- Continuous operation at specified power levels is assumed until end of useful fuel life (burnup of 10 grams ^{235}U). At the end of useful life, one week of regular operations (8 hours per day) is assumed. Radioisotope inventory decay calculations begin 5 minutes after termination of power operations based on the loss of pool water scenario; fuel handling after shutdown requires a substantially longer decay time for practical reasons.

- The fraction of noble gases and iodine contained within the fuel that is actually released is assumed to be 1.0×10^{-4}. This is a very conservative value prescribed in NUREG 2387 [Hawley and Kathren, 1982] and may be compared to the value of 1.5×10^{-5} measured at General Atomics [Simnad et al., 1976] which is used in SARs for other reactor facilities [NUREG-1390, 1990].

- The fractional release of particulates (radionuclides other than noble gases and iodine) is assumed to be 1.0×10^{-6}, a very conservative estimate used by Hawley and Kathren (1982).

- The reactor bay free air volume is 4120 m^3; 10% of this volume is not credited in dilution calculations.

- Radioisotopes specified in NURGE/CR-2387 with limits specified in 10CFR20 Appendix B are used in consequence analysis, including iodine, noble fission product gases, and cesium and strontium. Halogen (bromine) was analyzed in the 1992 UT SAR, and is therefore included in this analysis. The relevant information from 10CFR Appendix B is provided in Table 13.5.

Table 13.5, Relevant 10CFR20 Appendix B Values

Noble Gas & Iodine Radioisotopes				Particulate Radioisotopes			
Isotope	ALI μCi	DAC μCi/ml	EL μCi/ml	Isotope	ALI μCi	DAC μCi/ml	EL μCi/ml
Br80	2.0E+05	8.0E-05	3.0E-07	Cs131	3.0E+04	1.0E-05	4.0E-08
Br80m	2.0E+04	6.0E-06	2.0E-08	Cs132	4.0E+03	2.0E-06	6.0E-09
Br82	4.0E03	2.0E-06	5.0E-09	Cs134m	1.0E+05	6.0E-05	2.0E-07

Table 13.5, Relevant 10CFR20 Appendix B Values

Noble Gas & Iodine Radioisotopes				Particulate Radioisotopes			
Isotope	ALI μCi	DAC μCi/ml	EL μCi/ml	Isotope	ALI μCi	DAC μCi/ml	EL μCi/ml
Br83	6.0E+04	3.0E-05	9.0E-08	Cs135	1.0E+03	5.0E-07	2.0E-09
Br84	6.0E+04	2.0E-05	8.0E-08	Cs135m	2.0E+05	8.0E-05	3.0E-07
I125	6.0E+01	3.0E-08	3.0E-10	Cs136	7.0E+02	3.0E-07	9.0E-10
I128	1.0E+05	5.0E-05	8.0E-04	Cs137	2.0E+02	6.0E-08	2.0E-10
I129	9.0E+00	4.0E-09	4.0E-11	Cs138	6.0E+04	2.0E-05	8.0E-08
I130	7.0E+02	3.0E-07	2.0E-10	Sr85	2.0E+03	6.0E-07	2.0E-09
I131	5.0E+01	2.0E-08	2.0E-10	Sr85m	6.0E+05	3.0E-04	9.0E-07
I132	8.0E+03	3.0E-06	2.0E-08	Sr87m	1.0E+05	5.0E-05	2.0E-07
I133	3.0E+02	1.0E-07	1.0E-09	Sr89	1.0E+02	6.0E-08	2.0E-10
I134	5.0E+04	2.0E-05	6.0E-08	Sr90	4.0E+00	2.0E-09	6.0E-12
I135	2.0E+03	7.0E-07	6.0E-09	Sr91	4.0E+03	1.0E-06	5.0E-09
Kr79		2.0E-05	7.0E-08	Sr92	7.0E+03	3.0E-06	9.0E-09
Kr81		7.0E-04	3.0E-06				
Kr83m		1.0E-02	5.0E-05				
Kr85		1.0E-04	7.0E-07				
Kr85m		2.0E-05	1.0E-07				
Kr87		5.0E-06	2.0E-08				
Kr88		2.0E-06	9.0E-09				
Xe125		2.0E-05	7.0E-08				
Xe127		1.0E-05	6.0E-08				
Xe129m		2.0E-04	9.0E-07				
Xe131m		4.0E-04	2.0E-06				
Xe133		1.0E-04	5.0E-07				
Xe133m		1.0E-04	6.0E-07				
Xe135		1.0E-05	7.0E-08				
Xe135m		9.0E-06	4.0E-08				
Xe138		4.0E-06	2.0E-08				

- The 10FR20 appendix B Annual Limit on Intake (ALI) and Derived Air Concentration (DAC) values include the effects of the ingrowth of daughter radionuclides produced in the body by the decay of the parent nuclide (10CFR20, Appendix B, *Notation*, Table 1), and therefore daughters are not calculated or considered separately.

13.3.2 Analysis

Analysis of the maximum hypothetical accident begins with (A) a discussion of calculation's for fission product generated in the reactor, (B) methods and strategy for calculating the UT TRIGA fission product inventory, (C) fraction of fission products released from a single fuel element. The calculation of fission product inventory is used to evaluate the impact with respect to the 10CFR20 (D) Annual Limit on Intake, (E) Derived Air Concentration, and (F) Effluent Limits. The results are reviewed (F), concluding that measures prescribed by the Radiation Protection

Program would be required for worker protection in the worst case scenario, and that the effluent limits are met.

A. Radionuclide Inventory Buildup and Decay, Theory

Consider a mass of ^{235}U yielding thermal power P (kW) due to thermal-neutron induced fission. The fission rate is related to the thermal power by the factor $k = 3.12 \times 10^{13}$ fissions per second per kW. Consider also a fission product radionuclide, which is produced with yield Y, and which decays with rate constant λ. It is easily shown that the equilibrium activity A_∞ (Bq) of the fission product, which exists when the rate of creation by fission is equal to the rate of loss by decay, is given by $A=\lambda \cdot N$. Here it should be noted that the power must be small enough or the uranium mass large enough that the depletion of the ^{235}U is negligible. Starting at time $t = 0$, the buildup of activity is given by:

$$A(t) = A_\infty * \left(1 - e^{-\lambda * t}\right)$$

For times much greater than the half-life of the radionuclide, and for times much less than the half-life, $A(t) = A_\infty * \lambda * t$. If the fission process ceases at time t_1, the specific activity at later time t is given by

$$A(t) = A_\infty * \left(1 - e^{-\lambda * t}\right) * e^{-\lambda * (t - t_1)}$$

Consider the fission product ^{131}I, which has a half-life of 8.04 days ($\lambda = 0.00359\ h^{-1}$) and a chain (cumulative) fission product yield of about 0.031. At a thermal power of 1 kW, the equilibrium activity is about $A_\infty = 9.67 \times 10^{11}$ Bq (26.1 Ci). After only four hours of operation, though, the activity is only about 0.37 Ci. For equilibrium operation at 3.5 kW, distributed over 81 fuel elements, the average activity per element would be $26.1 \times 3.5 \div 81 = 1.13$ Ci per fuel element. The worst case element would contain twice this activity. With a release fraction of 1.0×10^{-4}, the activity available for release would be about $1.13 \times 2 \times 1.0 \times 10^{-4} = 2.26 \times 10^{-4}$ Ci. This type of calculation is performed by the ORIGEN ARP code for hundreds of fission products and for arbitrary times and power levels of operation as well as arbitrary times of decay after conclusion of reactor operation. The code accounts for branched decay chains. It also may account for depletion of ^{235}U and ingrowth of ^{239}Pu, although those features were not invoked in the calculations reported here because of minimal depletion in TRIGA fuel elements.

B. Fission Product Inventory Calculations

When burnup for TRIGA fuel containing 8.5% uranium enriched to [redacted] ^{235}U reaches about 6 grams ^{235}U, the element does not have enough net positive reactivity to contribute to criticality, and is removed from service. Since end of fuel element life is about 6 grams burnup, a 10 gram burnup is used in calculations to maximize potential fission product inventory.

SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design developed and maintained by Oak Ridge National Laboratory under contract with NRC and DOE to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs. A SCALE depletion sequence (code input, Appendix 13.1) was used to generate inventories of radioactive fission products for operation at steady state power until the target burnup was achieved. The sequence uses KENO VI to develop a reactor specific (geometry and materials) flux, with SCALE integrating calculations of flux averaged cross sections by several modules in sequence accounting for various factors that influence interaction rate, such as resonance self-shielding. Core and reflector geometry used to model the core is described in Chapter 4. Flux average cross sections are then used by ORIGEN S to calculate fission product generation and depletion. ORIGEN defaults to 1 metric ton of heavy metal (i.e., uranium) for calculations; the default value was used to simulate fission product inventory buildup with negligible burnup.

ORIGEN ARP (code input, Appendix 13.2) was used to determine the fission product inventory following specified decay intervals based on the depletion code output data. Burnup calculations were performed to evaluate (1) the core inventory for nominal 1 MW operations, (2) core inventory for nominal 2 MW power level, which may alternately be considered as the value for a single fuel element operating at twice the 1 MW core average power level, and (3) core inventory for 3.5 MW power level, or the a peak value for a 2 MW core average with a peaking factor of 1.5. The number of days to achieve each burnup was determined manual by iterations of the code to find an end point ^{235}U mass for 1 MTU that is 75% of the original value. SCALE burnup calculations are limited to a specified number of days to limit errors in calculation for large burnup values; the TRIGA burnup is not large, and the maximum number of days per calculation step was changed in the input for the TRIGA core, but still resulted in a different number of steps for each burnup value. Parameters of the calculations are provided in Table 13.6.

Table 13.6, SCALE T-6 Sequence Continuous Burnup Parameters

MW	DAYS	STEPS	START MASS	END MASS	RATIO
1	41000	104	1.97E+05	1.47E+05	7.46E-01
2	20475	54	1.97E+05	1.47E+00	7.46E-06
3.5	11625	20	1.97E+05	1.47E+05	7.48E-01

While long lived radionuclides should reasonably be represented by continuous operations at these intervals, the irradiation schedule is not representative of NETL operations. The facility is not staffed for continuous operations, and radioisotopes that have half-lives on the order of a few hours to days are not well represented. Therefore, a schedule for 1 working week (5 days, 8 hour operations at the specified power level followed by 16 hours of decay) was added to each irradiation following the continuous burnup interval. Fission product inventories calculated by ORIGEN are provided in Tables 13.7A and 13.7B for the specified gaseous and particulate fission products.

Table 13.7A, 1 MTU Gaseous Fission Product Inventory for 3.5 kW Case (Ci)

	s	m	h		D					
	1	30	1	8	1	7	30	90	180	365
br80										
br80m										
br82										
br83										
br84										
i125										
i128										
i129										
i130										
i131										
i132										
i133										
i134										
i135										
kr79										
kr81										
kr83m										
kr85										
kr85m										
kr87										
kr88										
xe125										
xe127										
xe129m										
xe131m										
xe133										
xe133m										
xe135										
xe135m										
xe138										.0

Table 13.7B, 1 MTU Particulate Fission Product Inventory (Ci)

	s	m	h		d					
	1	30	1	8	1	7	30	90	180	365
cs131										
cs132										
cs134m										
cs135										
cs135m										
cs136										

Table 13.7B, 1 MTU Particulate Fission Product Inventory (Ci)

	s	m	h		d					
	1	30	1	8	1	7	30	90	180	365
cs137										
cs138										
sr85										
sr85m										
sr87m										
sr89										
sr90										
sr91										
sr92										

C. Fission Product release

The calculated fission product inventory for 1 MTU is scaled by the ratio of the mass of a fuel element to a metric ton to determine the fission product inventory for a single fuel element. Most of the fission products generated during operation are trapped in the fuel matrix, only a fraction of the inventory has enough mobility to escape. The fraction that escapes is calculated using release fractions provided by NUREG/CR-2387 applied to each radionuclide identified. The NUREG considers noble gas, iodine, cesium, and strontium as the isotopes significant to consequence analysis; other refractories are neglected as they do not contribute significantly to potential exposure. Release inventories are provided in Tables 13.8A and 13.8B for gaseous and particulate fission products.

Table 13.8A. Gaseous Fission product Release from Single Element (μCi)

	S	M	H		D					
	1	30	1	8	1	7	30	90	180	365
br80	1.5E-4	5.4E-3	8.7E-5	1.2E-3	1.9E-6	3.5E-16	0.0	0.0	0.0	0.0
br80m	9.0E-5	4.3E-3	7.7E-5	1.1E-3	1.8E-6	3.3E-16	0.0	0.0	0.0	0.0
br82	1.7E-1	8.9	1.7E-1	7.5	1.1E-1	2.0E-1	7.7E-8	2.0E-18	0.0	0.0
br83	3.1E2	1.5E4	2.7E2	1.4E3	2.7E-1	1.2E-20	0.0	0.0	0.0	0.0
br84	5.5E2	1.6E4	1.7E2	2.4E-1	3.9E-12	0.0	0.0	0.0	0.0	0.0
i125	1.1E-10	1.1E-10	1.1E-10	1.1E-10	1.1E-10	9.7E-11	7.4E-11	3.7E-11	1.3E-11	1.5E-12
i128	2.7E-1	1.2E-1	5.0E-2	8.3E-8	2.3E-19	0.0	0.0	0.0	0.0	0.0
i129	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4	1.4E-4
i130	1.5	1.5	1.4	9.1E-1	3.7E-1	3.0E-5	1.1E-18	0.0	0.0	0.0
i131	1.7E3	1.7E3	1.7E3	1.6E3	1.6E3	8.6E2	1.2E2	6.6E-1	2.8E-4	3.1E-11
i132	2.5E3	2.5E3	2.5E3	2.4E3	2.1E3	4.5E2	3.1	7.2E-6	2.5E-14	0.0
i133	3.9E3	3.9E3	3.8E3	3.0E3	1.7E3	6.4	6.6E-8	9.5E-29	0.0	0.0
i134	4.5E3	4.1E3	3.4E3	1.4E1	5.0E-5	0.0	0.0	0.0	0.0	0.0
i135	3.6E3	3.5E3	3.3E3	1.4E3	2.6E2	5.2E-6	2.7E-31	0.0	0.0	0.0
kr79	1.9E-10	1.9E-10	1.9E-10	1.6E-10	1.2E-10	4.2E-12	7.7E-17	3.3E-29	0.0	0.0
kr81	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11	5.7E-11

kr83m	3.1E2	3.1E2	3.0E2	7.2E1	1.0	4.2E-7	3.5E-7	2.1E-7	1.0E-7	2.4E-8
kr85	1.0E2	1.0E2	1.0E2	1.0E2	1.0E2	1.0E2	1.0E2	9.9E1	9.7E1	9.4E1
kr85m	7.7E2	7.2E2	6.7E2	1.9E2	1.6E1	8.4E-11	0.0	0.0	0.0	0.0
kr87	1.5E3	1.1E3	8.7E2	1.1E1	1.8E-3	0.0	0.0	0.0	0.0	0.0
kr88	2.0E3	1.8E3	1.6E3	2.2E2	4.5	7.0E-18	0.0	0.0	0.0	0.0
xe125	9.4E-13	9.2E-13	9.0E-13	6.5E-13	3.4E-13	3.4E-16	5.0E-26	0.0	0.0	0.0
xe127	3.7E-8	3.7E-8	3.7E-8	3.7E-8	3.6E-8	3.2E-8	2.0E-8	6.5E-9	1.2E-9	3.5E-11
xe129m	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.0E-5	5.9E-6	9.8E-7	9.0E-9	8.0E-12	4.3E-18
xe131m	2.0E1	2.0E1	2.0E1	2.0E1	2.0E1	1.8E1	7.2	2.9E-1	1.5E-3	3.1E-8
xe133	3.9E3	3.9E3	3.9E3	3.9E3	3.7E3	1.6E3	7.8E1	2.8E-2	1.9E-7	4.5E-18
xe133m	4.2E1	4.2E1	4.2E1	4.1E1	3.8E1	5.4	3.7E-3	2.1E-11	9.0E-24	0.0
xe135	3.6E3	3.6E3	3.6E3	2.9E3	1.3E3	5.7E-3	3.8E-21	0.0	0.0	0.0
xe135m	4.7E2	3.8E2	3.4E2	1.4E2	2.7E1	5.4E-7	0.0	0.0	0.0	0.0
xe138	3.6E3	8.3E2	1.9E2	1.0E-8	3.1E-29	0.0	0.0	0.0	0.0	0.0

Table 13.8B. Particulate Fission Product Release from Single Element

	S	M	H		D					
	1.0	3.0E1	1.0	8.0	1.0	7.0	3.0E1	9.0E1	1.8E2	3.7E2
cs131	2.7E-10	2.7E-10	2.7E-10	2.7E-10	2.5E-10	1.5E-10	3.0E-11	4.1E-13	6.5E-16	1.2E-21
cs132	1.0E-5	9.9E-6	9.9E-6	9.6E-6	8.9E-6	4.2E-6	3.6E-7	5.9E-10	3.9E-14	9.8E-23
cs134m	3.3E-2	3.0E-2	2.6E-2	3.9E-3	8.7E-5	3.7E-22	0.0	0.0	0.0	0.0
cs135	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4	1.1E-4
cs135m	2.0E-3	1.4E-3	9.2E-4	1.7E-6	6.1E-12	0.0	0.0	0.0	0.0	0.0
cs136	9.3E-2	9.3E-2	9.3E-2	9.1E-2	8.8E-2	6.1E-2	1.8E-2	7.7E-4	6.7E-6	3.9E-10
cs137	7.3	7.3	7.3	7.3	7.3	7.3	7.3	7.3	7.2	7.2
cs138	3.9E1	2.9E1	1.7E1	8.9E-4	2.0E-12	0.0	0.0	0.0	0.0	0.0
sr85	1.4E-9	1.4E-9	1.4E-9	1.4E-9	1.4E-9	1.3E-9	1.0E-9	5.4E-10	2.1E-10	2.9E-11
sr85m	7.8E-10	5.7E-10	4.2E-10	3.1E-12	1.6E-16	0.0	0.0	0.0	0.0	0.0
sr87m	2.2E-6	2.0E-6	1.7E-6	2.4E-7	4.7E-9	5.1E-27	0.0	0.0	0.0	0.0
sr89	2.7E1	2.7E1	2.7E1	2.7E1	2.7E1	2.4E1	1.8E1	7.8	2.3	1.8E-1
sr90	7.1	7.1	7.1	7.1	7.1	7.1	7.1	7.0	7.0	6.9
sr91	3.3E1	3.2E1	3.1E1	1.7E1	5.5	3.1E-5	1.7E-22	0.0	0.0	0.0
sr92	3.4E1	3.0E1	2.6E1	3.3	5.1E-2	4.9E-21	0.0	0.0	0.0	0.0

D. ALI Consequence Analysis

Regulatory Guideline 8.34, *Monitoring Criteria and Methods To Calculate Occupational Radiation Doses*, provides methodology to determine potential dose rates from ingestion of, or immersion in, radionuclides using data in 10CFR20 Appendix B. The ALI is used to determine potential consequences from an ingestion of a radionuclide. If the radionuclide inventory is less than one 10CFR20 Appendix B "Annual Limit on Intake" (ALI), then it is not physically possible to exceed the annual limits for worker exposure. If the available radionuclide release exceeds an ALI, then it is necessary to examine the fraction of the inventory to which individuals will be exposed. The ratio of a radionuclide inventory to the ALI value determines the fraction of the limit subsumed

by a single radionuclide. The sum of the ratios for all radionuclides bounds the consequences, with a sum-value less than 1 indicating a total value less than the ALI value and a total value greater than 1 exceeding the ALI value.

Table 13.9A, Fraction of Gaseous Fission Product Inventory to 10CFR20 ALI

	s	M	h		d					
	1	30	1	8	1	7	30	90	180	365
br80	1.2E-5	2.1E-2	4.2E-2	3.3E-1	1.0	7.0	3.0E1	9.0E1	1.8E2	3.7E2
br80m	7.5E-10	2.7E-8	4.4E-10	6.0E-9	9.5E-12	1.8E-21	0.0	0.0	0.0	0.0
br82	4.5E-9	2.1E-7	3.8E-9	5.6E-8	8.9E-11	1.7E-20	0.0	0.0	0.0	0.0
br83	4.4E-5	2.2E-3	4.3E-5	1.9E-3	2.7E-5	5.0E-5	1.9E-11	5.1E-22	0.0	0.0
br84	5.2E-3	2.5E-1	4.4E-3	2.3E-2	4.5E-6	1.9E-25	0.0	0.0	0.0	0.0
i125	9.2E-3	2.7E-1	2.8E-3	4.0E-6	6.4E-17	0.0	0.0	0.0	0.0	0.0
i128	1.8E-12	1.8E-12	1.8E-12	1.8E-12	1.8E-12	1.6E-12	1.2E-12	6.1E-13	2.1E-13	2.5E-14
i129	2.7E-6	1.2E-6	5.0E-7	8.3E-13	2.3E-24	0.0	0.0	0.0	0.0	0.0
i130	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5
i131	2.1E-3	2.1E-3	2.0E-3	1.3E-3	5.3E-4	4.3E-8	1.5E-21	0.0	0.0	0.0
i132	3.4E1	3.4E1	3.4E1	3.3E1	3.1E1	1.7E1	2.3	1.3E-2	5.5E-6	6.3E-13
i133	3.1E-1	3.1E-1	3.1E-1	3.0E-1	2.6E-1	5.6E-2	3.9E-4	9.0E-10	3.1E-18	0.0
i134	1.3E1	1.3E1	1.3E1	9.8	5.8	2.1E-2	2.2E-10	3.2E-31	0.0	0.0
i135	9.0E-2	8.1E-2	6.7E-2	2.8E-4	1.0E-9	0.0	0.0	0.0	0.0	0.0
br80	1.8	1.7	1.6	7.0E-1	1.3E-1	2.6E-9	1.3E-34	0.0	0.0	0.0
SUMS:	48.8	48.6	48.4	43.6	37.2	17.2	2.4	0.13	2.1E-5	1.5E-5

Table 13.9B, Fraction of Particulate Fission Product Inventory to 10CFR20 ALI

	s	M	h		d					
	1	30	1	8	1	7	30	90	180	365
cs131	9.1E-15	9.1E-15	9.1E-15	8.9E-15	8.5E-15	5.1E-15	9.9E-16	1.4E-17	2.2E-20	3.9E-26
cs132	2.5E-9	2.5E-9	2.5E-9	2.4E-9	2.2E-9	1.1E-9	9.0E-11	1.5E-13	9.7E-18	2.5E-26
cs134m	3.3E-7	3.0E-7	2.6E-7	3.9E-8	8.7E-10	3.7E-27	0	0	0	0
cs135	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7	1.1E-7
cs135m	1.0E-8	6.8E-9	4.6E-9	8.6E-12	3.0E-17	0	0	0	0	0
cs136	1.3E-4	1.3E-4	1.3E-4	1.3E-4	1.3E-4	8.7E-5	2.6E-5	1.1E-6	9.6E-9	5.6E-13
cs137	3.7E-2	3.7E-2	3.7E-2	3.7E-2	3.7E-2	3.7E-2	3.7E-2	3.6E-2	3.6E-2	3.6E-2
cs138	6.5E-4	4.8E-4	2.9E-4	1.5E-8	3.3E-17	0	0	0	0	0
sr85	7.2E-13	7.2E-13	7.2E-13	7.2E-13	7.1E-13	6.6E-13	5.2E-13	2.7E-13	1.0E-13	1.4E-14
sr85m	1.3E-15	9.5E-16	7.0E-16	5.1E-18	2.7E-22	0	0	0	0	0
sr87m	2.2E-11	2.0E-11	1.7E-11	2.4E-12	4.7E-14	5.1E-32	0	0	0	0
sr89	2.7E-1	2.7E-1	2.7E-1	2.7E-1	2.7E-1	2.4E-1	1.8E-1	7.8E-2	2.3E-2	1.8E-3
sr90	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.7	1.7
sr91	8.3E-3	8.0E-3	7.8E-3	4.4E-3	1.4E-3	7.7E-9	4.3E-26	0	0	0
sr92	4.9E-3	4.3E-3	3.8E-3	4.7E-4	7.2E-6	7.0E-25	0	0	0	0
SUMS:	2.1	2.1	2.1	2.1	2.1	2.0	2.0	1.9	1.8	1.8

ALI values are exceeded for the 3.5 MW case; data from all cases is provided graphically in Fig. 13.1. The gaseous radionuclide inventory is shown to be greater than the ALI for approximately 25-40 days following the release, while the particulate radionuclide inventory remains above the ALI for all cases, principally driven by ^{90}Sr.

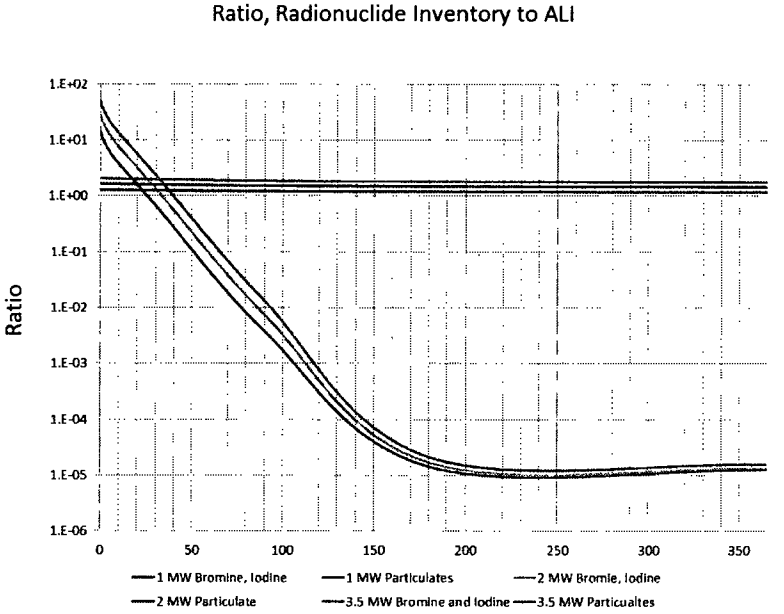

Figure 13.1, Ratio of Radionuclide Inventory to ALI

This analysis is extremely conservative in neglecting transport to personnel. There is no conceivable scenario where all radionuclide inventories are delivered to a single individual, and any reduction in the amount of uptake to the individual reduces the uptake.

This analysis is conservative in assuming a burnup of 10 grams ^{235}U and a continuous operating history. A slightly more realistic assumption of 6 grams burnup reduces the inventory of the long lived ^{90}Sr by approximately 60%, and a less aggressive operating schedule reduces shorter lived radionuclides considerably.

This analysis is conservative in assuming that the radionuclide inventory is maintained in one location for the duration of the analysis, and does not consider any removal of the inventory from the receptor through normal atmospheric transport such as simple settling of particulate matter or removal from the reactor bay by the HVAC system, wind driven exchange of building air, or active cleanup processes. A reduction in inventory reduces the ALI ratio.

Finally, this analysis does not consider any compensatory or mitigating actions in response to the release. The reactor Radiation Protection Program requires monitoring and control of exposure, and with a maximum hypothetical ALI ratio of 1.8 for ^{90}Sr, measures to reduce and control exposure to an individual by a factor of approximately 2 for particulate radionuclides are easily

achievable by passive measures or active processes such as dilution in the reactor bay air or filtering.

Therefore although the ALI values in the reactor bay are exceeded for the maximum hypothetical accident, control of personnel exposure under the Radiation Protection Program to the radionuclides released into the reactor bay is adequate to manage personnel dose within limits of 10CFR20.

E. DAC Consequence Analysis

The escaping fission product inventory is assumed to mix with reactor bay atmosphere. Nominal free volume of the reactor bay is 4120 m^3; 10% of the nominal volume is assumed occupied by equipment or materials. The radionuclide inventory is therefore assumed to be distributed in 3719 m^3. The 10CFR20 "Derived Air Concentration" (DAC) is used to limit potential consequences for workers based on the radionuclide inventory released into a volume of air. In a manner similar to the ALI analysis described above, consequences of exposure to a mixture of radionuclides are evaluated based on the derived air concentration in 10CFR20 Appendix B with the results in Table 13.10A and 13.10B.

Table 13.10A, Fraction of Instantaneous Gaseous Fission Product Inventory to 10CFR20 DAC[1]

	s	m	h		d					
	1	30	1	8	1	7	30	90	180	365
br80	6.1E-10	2.2E-8	3.5E-10	4.9E-9	7.7E-12	1.4E-21	0	0	0	0
br80m	4.8E-9	2.3E-7	4.1E-9	6.1E-8	9.6E-11	1.8E-20	0	0	0	0
br82	2.8E-5	1.4E-3	2.8E-5	1.2E-3	1.7E-5	3.3E-5	1.2E-11	3.3E-22	0	0
br83	3.4E-3	1.6E-1	2.9E-3	1.5E-2	2.9E-6	1.3E-25	0	0	0	0
br84	8.9E-3	2.6E-1	2.7E-3	3.9E-6	6.2E-17	0	0	0	0	0
i125	1.1E-12	1.1E-12	1.1E-12	1.1E-12	1.1E-12	1.0E-12	8.0E-13	4.0E-13	1.4E-13	1.6E-14
i128	1.7E-6	7.5E-7	3.3E-7	5.4E-13	1.5E-24	0	0	0	0	0
i129	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5	1.1E-5
i130	1.6E-3	1.6E-3	1.5E-3	9.8E-4	4.0E-4	3.2E-8	1.2E-21	0	0	0
i131	2.7E1	2.7E1	2.7E1	2.7E1	2.5E1	1.4E1	1.90	1.1E-2	4.5E-6	5.1E-13
i132	2.7E-1	2.7E-1	2.7E-1	2.6E-1	2.2E-1	4.9E-2	3.4E-4	7.8E-10	2.7E-18	0
i133	1.3E1	1.2E1	1.2E1	9.60	5.60	2.1E-2	2.1E-10	3.1E-31	0	0
i134	7.3E-2	6.6E-2	5.4E-2	2.3E-4	8.2E-10	0	0	0	0	0
i135	1.70	1.60	1.50	6.5E-1	1.2E-1	2.4E-9	1.2E-34	0	0	0
kr79	3.1E-15	3.1E-15	3.1E-15	2.6E-15	1.9E-15	6.9E-17	1.2E-21	5.3E-34	0	0
kr81	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17	2.6E-17
kr83m	1.0E-5	1.0E-5	9.8E-6	2.3E-6	3.3E-8	1.4E-14	1.1E-14	7.0E-15	3.4E-15	7.6E-16
kr85	3.2E-4	3.2E-4	3.2E-4	3.2E-4	3.2E-4	3.2E-4	3.2E-4	3.2E-4	3.1E-4	3.0E-4
kr85m	1.2E-2	1.2E-2	1.1E-2	3.1E-3	2.6E-4	1.4E-15	0	0	0	0
kr87	9.6E-2	7.4E-2	5.6E-2	7.2E-4	1.2E-7	0	0	0	0	0
kr88	3.3E-1	2.9E-1	2.6E-1	3.6E-2	7.3E-4	1.1E-21	0	0	0	0
xe125	1.5E-17	1.5E-17	1.5E-17	1.1E-17	5.5E-18	5.6E-21	8.2E-31	0	0	0
xe127	1.2E-12	1.2E-12	1.2E-12	1.2E-12	1.2E-12	1.0E-12	6.6E-13	2.1E-13	3.8E-14	1.1E-15

Table 13.10A, Fraction of Instantaneous Gaseous Fission Product Inventory to 10CFR20 DAC[1]

	s	m	h		d					
	1	30	1	8	1	7	30	90	180	365
xe129m	1.8E-11	1.8E-11	1.8E-11	1.7E-11	1.6E-11	9.5E-12	1.6E-12	1.5E-14	1.3E-17	7.0E-24
xe131m	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.6E-5	1.4E-5	5.8E-6	2.3E-7	1.3E-9	2.5E-14
xe133	1.3E-2	1.3E-2	1.3E-2	1.2E-2	1.2E-2	5.2E-3	2.5E-4	9.0E-8	6.1E-13	1.5E-23
xe133m	1.4E-4	1.4E-4	1.4E-4	1.3E-4	1.2E-4	1.7E-5	1.2E-8	6.8E-17	2.9E-29	0
xe135	1.2E-1	1.2E-1	1.2E-1	9.5E-2	4.1E-2	1.8E-7	1.2E-25	0	0	0
xe135m	1.7E-2	1.4E-2	1.2E-2	5.2E-3	9.6E-4	1.9E-11	0	0	0	0
xe138	2.9E-1	6.7E-2	1.5E-2	8.4E-13	2.5E-33	0	0	0	0	0
SUMS:	3.6	3.5	3.5	3.1	2.6	1.2	1.6	9.1E-3	2.8E-4	2.6E-4

Table 13.10B, Fraction of Instantaneous Particulate Fission Product Inventory to 10CFR20 DAC [1]

	s	M	h		D					
	1	30	1	8	1	7	30	90	180	365
cs131	7.4E-15	7.4E-15	7.4E-15	7.2E-15	6.8E-15	4.1E-15	8.0E-16	1.1E-17	1.8E-20	3.2E-26
cs132	1.3E-9	1.3E-9	1.3E-9	1.3E-9	1.2E-9	5.7E-10	4.9E-11	7.9E-14	5.2E-18	1.3E-26
cs134m	1.5E-7	1.3E-7	1.2E-7	1.8E-8	3.9E-10	1.7E-27	0	0	0	0
cs135	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8	5.8E-8
cs135m	6.8E-9	4.6E-9	3.1E-9	5.8E-12	2.0E-17	0	0	0	0	0
cs136	8.3E-5	8.3E-5	8.3E-5	8.2E-5	7.9E-5	5.5E-5	1.6E-5	6.9E-7	6.0E-9	3.5E-13
cs137	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.3E-2	3.2E-2
cs138	5.2E-4	3.9E-4	2.4E-4	1.2E-8	2.7E-17	0	0	0	0	0
sr85	6.5E-13	6.5E-13	6.5E-13	6.4E-13	6.4E-13	5.9E-13	4.6E-13	2.4E-13	9.3E-14	1.3E-14
sr85m	7.0E-16	5.1E-16	3.8E-16	2.8E-18	1.5E-22	0	0	0	0	0
sr87m	1.2E-11	1.1E-11	9.4E-12	1.3E-12	2.6E-14	2.8E-32	0	0	0	0
sr89	1.2E-1	1.2E-1	1.2E-1	1.2E-1	1.2E-1	1.1E-1	8.0E-2	3.5E-2	1.0E-2	8.0E-4
sr90	9.5E-1	9.5E-1	9.5E-1	9.5E-1	9.5E-1	9.5E-1	9.5E-1	9.5E-1	9.4E-1	9.3E-1
sr91	9.0E-3	8.7E-3	8.4E-3	4.7E-3	1.5E-3	8.3E-9	4.6E-26	0	0	0
sr92	3.1E-3	2.7E-3	2.4E-3	2.9E-4	4.5E-6	4.4E-25	0	0	0	0
SUMS:	1.12	1.12	1.12	1.11	1.11	1.09	1.06	1.02	0.985	0.963

NOTE[1]: DAC limits are based on 2000 hours of exposure over a year; these tables compare the instantaneous value of airborne radionuclides, and not the 2000 hour exposure period. Integration of the instantaneous values over a year evaluates compliance with DAC limits.

The DAC values are exceeded for gaseous fission product concentration for about 40 days following the accident in the 3.5 MW case. Atmospheric particulate activity remains elevated above the DAC value because of the long-lived strontium (as with the ALI) for about ½ of the year in the 3.5 MW case; however, the particulate values for the 2 and the 1 MW cases do not exceed the DAC values at any time, as indicated in Fig. 13.2.

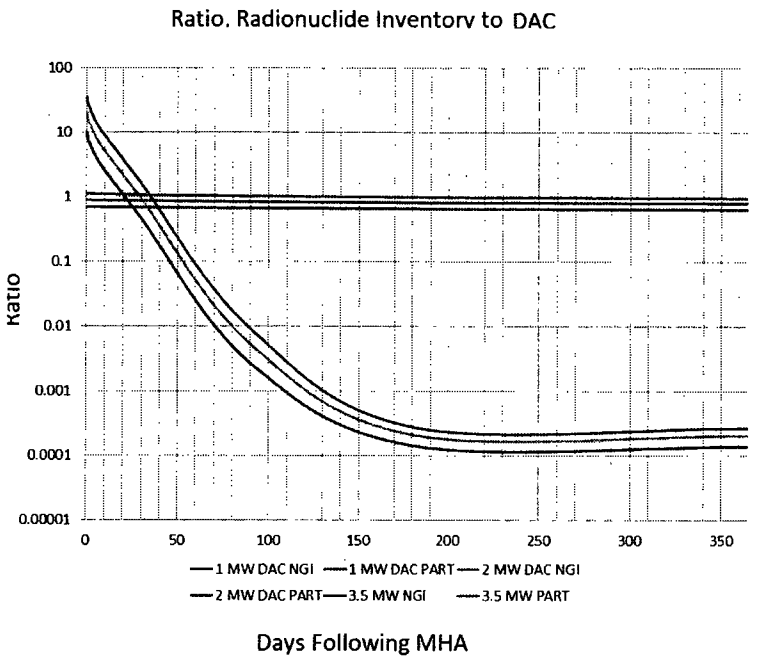

Figure 13.2, Ratio of Radionuclide Concentration to 10CFR 20 DAC Values

DAC values apply to continuous exposure over a year. Concentrations averaged using time interval weighting over a year following the event for all three cases are provided in Table 13.11.

Table 13.11, DAC Ratios for All Cases

		1 MW		2 MW		3.5 MW	
		Gaseous	Particulate	Gaseous	Particulate	Gaseous	Particulate
Seconds	1	10	0.69	20	0.89	36	1.1
Minutes	30	10	0.69	20	0.89	35	1.1
Hours	1	10	0.69	20	0.89	35	1.1
	8	8.9	0.69	18	0.89	31	1.1
Days	1	7.4	0.69	15	0.89	26	1.1
	7	3.3	0.69	6.6	0.88	12	1.1
	30	0.45	0.68	0.91	0.86	1.6	1.1
	90	2.7E-03	0.66	5.3E-03	0.83	9.1E-03	1.0
	180	1.4E-04	0.65	2.1E-04	0.81	2.8E-04	0.98
	365	1.4E-04	0.64	2.0E-04	0.8	2.6E-04	0.96
Weighted average		0.11	0.67	0.25	0.86	0.46	1.10

As in the ALI analysis, this analysis is conservative in assuming a burnup of 10 grams ^{235}U and a continuous operating history. A slightly more realistic assumption of 6 grams burnup reduces the inventory of the long lived ^{90}Sr by approximately 60%, interpolating between the particulate ratios indicates that the DAC ratio is 1 at or below 3 MW, and a less aggressive operating schedule reduces shorter lived radionuclides considerably.

This analysis is conservative in assuming 10% of the volume is occupied by equipment. Increasing the volume decreases nuclide concentration.

This analysis is conservative in assuming that the radionuclide inventory is maintained in one location for the duration of the analysis, and does not consider any removal of the inventory from the reactor bay through normal atmospheric transport, either simple settling of particulate matter or removal from the reactor bay by natural or active cleanup processes. The reactor bay HVAC control system is designed to automatically secure ventilation on detecting a preset level of airborne contamination, and there is some delay before the radionuclides buildup to the trip level. During this interval the reactor bay continues to exhaust by design 34.3 m^3 s^{-1}, and an actual 57.2 m^3 s^{-1}. A reduction in fission product inventory reduces the DAC ratio.

Finally, this analysis does not consider any compensatory or mitigating actions in response to the release. The reactor Radiation Protection Program requires monitoring and control of exposure, and with a maximum hypothetical DAC ratio of 1.14 for ^{90}Sr that dominates the particulate analysis, measures to reduce and control exposure to an individual by a factor of approximately 2 for particulate radionuclides are easily achievable by passive measures or active processes such as dilution in the reactor bay air or filtering.

Therefore although the DAC values in the reactor bay are exceeded for the 2 and 3.5 MW case of the maximum hypothetical accident under extremely conservative assumptions, access control to manage personnel exposure under the Radiation Protection Program is adequate to maintain personnel dose within limits of 10CFR20.

F. Effluent release Consequence Analysis

The radionuclide concentration in the reactor bay atmosphere following the maximum hypothetical accident is compared to the effluent limit, assuming the radionuclide inventory is not transported from confinement and is only removed through decay.

Table 13.12, Reactor Bay Atmosphere Following MHA Compared to Effluent Limit

	1 MW		2 MW		3.5 MW	
TIME	NGI	PART	NGI	PART	NGI	PART
1 s	2.29E+02	2.29E+02	2.94E+02	2.94E+02	3.67E+02	3.67E+02
30 m	2.29E+02	2.29E+02	2.94E+02	2.94E+02	3.67E+02	3.67E+02
1 h	2.29E+02	2.29E+02	2.93E+02	2.93E+02	3.67E+02	3.67E+02
8 h	2.29E+02	2.29E+02	2.92E+02	2.92E+02	3.65E+02	3.65E+02
1 d	2.28E+02	2.28E+02	2.92E+02	2.92E+02	3.64E+02	3.64E+02
7 d	2.27E+02	2.27E+02	2.90E+02	2.90E+02	3.60E+02	3.60E+02
30 d	2.24E+02	2.24E+02	2.84E+02	2.84E+02	3.51E+02	3.51E+02
90 d	2.20E+02	2.20E+02	2.75E+02	2.75E+02	3.36E+02	3.36E+02
180 d	2.16E+02	2.16E+02	2.70E+02	2.70E+02	3.27E+02	3.27E+02
365 d	2.13E+02	2.13E+02	2.65E+02	2.65E+02	3.20E+02	3.20E+02

The results demonstrate that the reactor bay atmosphere cannot be discharged to the environment in the absence of mitigating factors. However, individuals are not directly exposed to effluent releases. The environment dilutes the radionuclide concentration in atmospheric dispersion.

F (1) Atmospheric Dispersion

Standard plume modeling is used to assess dilution of contaminants at the exit of the reactor bay. A standard approach assumes a Gaussian distribution for the dispersion of contaminants perpendicular to wind-driven motion of material in a plume. The Gaussian dispersion parameters are a function of atmospheric stability and the distance of plume travel. The *Workbook of Atmospheric Dispersion Estimates* (D. B. Turner, 2nd Ed., 1994) reports dispersion parameters determined experimentally for urban areas, tabulated below.

Table 13.13: BRIGGS URBAN DISPERSION PARAMETERS

	σ_y (meters)				σ_z (meters)			
x,km	A-B	C	D	E-F	A-B	C	D	E-F
0.01	3.19	2.20	1.60	1.10	2.41	2.00	1.40	0.79
0.02	6.37	4.38	3.19	2.19	4.85	4.00	2.79	1.58
0.03	9.54	6.56	4.77	3.28	7.31	6.00	4.18	2.35
0.04	12.70	8.73	6.35	4.37	9.79	8.00	5.57	3.11
0.05	15.80	10.90	7.92	5.45	12.30	10.00	6.95	3.86
0.06	19.00	13.00	9.49	6.52	14.80	12.00	8.33	4.60
0.07	22.10	15.20	11.00	7.59	17.40	14.00	9.70	5.33
0.08	25.20	17.30	12.60	8.66	20.00	16.00	11.10	6.05
0.09	28.30	19.50	14.10	9.73	22.60	18.00	12.40	6.76
0.10	31.40	21.60	15.70	10.80	25.20	20.00	13.80	7.46

Dispersion parameters are used to develop a conversion factor (χ/Q) at each distance from the center of the release relating a contamination release rate (Q_0, contaminant released per second) to a concentration (N_x) at the specified location.

$$Q_0 \cdot \frac{\chi}{Q_x} = N_x$$

The release of the radioactive inventory (Q_0) can be characterized as the product of the nuclide concentration being released (N_0) and the volumetric flow rate ($\dot{w}$ volume per second), and the product of the nuclide and the decay constant (λ) is the activity of the radionuclide. Therefore, where $\dot{w}$ is the volumetric flow rate (in units consistent the nuclide concentration) the equation can be written as:

$$A_0 \cdot \frac{\chi}{Q_x} \cdot \dot{w} = A_x$$

Two cases are considered. The reactor bay ventilation is automatically secured on detection of airborne radioactive material in the reactor bay, but the auxiliary purge system override is used to re-initiate purge flow. Therefore the first case considers that the auxiliary purge system discharges the reactor bay effluent continuously through a HEPA filter and the building stack at the normal flow rate (0.52 m^3 per second). In the second case, the auxiliary purge system remains secured trough the event. In the second case the discharge is not through the stack, but through normal building aspiration processes as a result of environment (wind) driven differential pressures developed across the building acting on vents and other building penetrations.

F (2) CASE I:

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, addresses releases from stacks. The equation for ground-level relative concentration at the plume centerline for stack releases is given as:

$$\chi/Q = \frac{1}{\pi \cdot \bar{U}_h \cdot \sigma_y \cdot \sigma_z} \cdot e^{\frac{-h_e^2}{2 \cdot \sigma_z}}$$

Where:

$\bar{U}_h$ is the wind speed applicable to the release height
h_e is the effective stack height
σ_y and σ_z are Gaussian dispersion coefficients for distance and height of the release

For the case of auxiliary purge system operation,

- Effective stack height is calculated in Chapter 9 as 1.71/{wind velocity} m above the top of the stack at 63 feet (19.202 m);
- A high efficiency particulate filter is required in operation of the auxiliary purge system, and is credited in analysis;
- The auxiliary purge system operates at a nominal 1100 cfm (0.52 m^3/s, 5.2E5 cm^3/s); this flow rate is used in the dilution calculation;
- Removal rate for air from the reactor bay in days is calculated:

$$r = \frac{0.52 \frac{cm^3}{s} \cdot \frac{3600s}{1h} \cdot \frac{24h}{1d}}{4120m^3 \cdot \frac{1E6cm^3}{m^3}}.$$

For a limiting case, and the wind speed is assumed to be 1, and the χ/Q was calculated for each associated set of dispersion parameters (σ_y, σ_z), with the results provided in Table 13.14; the maximum χ/Q value that provides the least dispersion is 0.001416 (Class C, 0.02 km).

Table 13.14, Calculated χ/Q Values

km	A-B	C	D	E-F
0.01	0.000541	0.000388	8.11E-05	6.54E-07
0.02	0.001193	0.001331	0.000843	0.000123
0.03	0.001092	0.001416	0.001309	0.000483
0.04	0.00088	0.001233	0.001377	0.000812
0.05	0.0007	0.001026	0.001285	0.001008
0.06	0.000558	0.000854	0.001148	0.001093
0.07	0.000454	0.000709	0.001015	0.001106
0.08	0.000374	0.000598	0.000887	0.001079
0.09	0.000313	0.000507	0.000783	0.00103
0.1	0.000266	0.000437	0.000689	0.000973

Effluent limit values are exceeded by gaseous fission products for about 1 day. However, the effluent limit values are bases on a continuous discharge over a year, and the total annual average is well within limits.

Table 13.15, Reactor Bay Atmosphere Following MHA Compared to Effluent Limit

	1 MW		2 MW		3.5 MW	
TIME	NGI	PART	NGI	PART	NGI	PART
1 s	1.19	5.07E-05	1.53	6.49E-05	2.71	8.11E-05
30 m	1.16	5.06E-05	1.53	6.48E-05	2.66	8.10E-05
1 h	1.13	5.06E-05	1.52	6.48E-05	2.63	8.10E-05
8 h	0.859	5.05E-05	1.33	6.46E-05	2.30	8.06E-05
1 d	0.633	5.05E-05	1.11	6.45E-05	1.92	8.04E-05
7 d	0.258	5.02E-05	0.493	6.40E-05	0.856	7.96E-05
30 d	3.34E-02	4.95E-05	6.68E-02	6.28E-05	0.117	7.75E-05
90 d	2.01E-04	4.85E-05	3.95E-04	6.08E-05	6.80E-04	7.42E-05
180 d	1.46E-05	4.77E-05	2.16E-05	5.94E-05	2.85E-05	7.20E-05
365 d	1.41E-05	4.68E-05	2.07E-05	5.82E-05	2.73E-05	7.04E-05

In all cases where the auxiliary purge system is operating and the confinement ventilation system is secured, 10CFR20 effluent limits are met in the maximum hypothetical accident.

F (2) CASE II:

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, addresses releases through vents or other building penetrations. REGGUIDE 1.145, section 1.3.1, considers three different effects for decreasing concentrations of an effluent release from vents or building openings: fundamental atmospheric dispersion, effects of the building itself on atmospheric mixing characteristics, and the effects of

the building and plume meandering. The REGGUIDE provides three different formulae to determine relative concentration values, and directs the use of the highest calculated value of the first two formulae (building effects and basic atmospheric dispersion); the third formula addresses mitigation (i.e., reduced concentration of contaminants) caused by turbulent mixing from building wake effects and plume meandering. Equation (3) applies to neutral and stable atmospheric conditions (Class D, E, F, G), where the 10 meter wind speeds are slow enough that the effects are significant (less than 6 m s^{-1}).

$$\chi/Q = \frac{1}{\overline{U}_{10} \cdot \left(\pi \cdot \sigma_y \cdot \sigma_z + \frac{A}{2}\right)}$$

$$\chi/Q = \frac{1}{\overline{U}_{10} \cdot 3 \cdot \pi \cdot \sigma_y \cdot \sigma_z}$$

$$\chi/Q = \frac{1}{\overline{U}_{10} \cdot \pi \cdot \Sigma_y \cdot \sigma_z}$$

Plume meander and building wake effects (mixing effects) from the building at distances less than 800 meters from the release use a coefficient, Σ_y in equation (3), which is the product of a correction factor (*M*) and the dispersion coefficient, σ_y. The correction factor is presented graphically in the REGGUIDE for stability classes D, E, F, and G; each class has a constant value from the minimum wind speed to 2 m s^{-1}, and decreases linearly from the maximum value at 2 m s^{-1} to 0 at 6 m s^{-1}. REGGUIDE guidance effectively allows the use of calculated plume meander factors (*M*) greater than 3, where winds are less than 6 m s^{-1}. Or (for winds < 6 m s^{-1}, where $M >$ 3):

$$\chi/Q = \frac{1}{\overline{U}_{10} \cdot M \cdot \sigma_y \cdot \sigma_x}$$

Table 13.16, Calculated Plume Meander Factor (M) for < 6 m s^{-1} Winds

Class	0.77 m s^{-1}	2.57 m s^{-1}	4.37 m s^{-1}
D	2	1.8575	1.4075
E	3	2.715	1.815
F	4	3.5725	2.2225
G	6	5.2875	3.0375

The minimum 10 meter dispersion parameters in Table 13.17 and the lowest correction factor (M) for the applicable category are provided in Table 13.16. The χ/Q for each stability class calculated for each equation in REGUIDE 1.45, using the minimum values for σ_y, σ_z, and M, are reported in Table 13.18.

Table 13.17, Minimum Dispersion Parameters by Stability Class

	A-B	C	D	E-F
σ_y	3.19	2.2	1.6	1.1
σ_z	2.41	2	1.4	0.79
M			2	3

Table 13.18, Minimum χ/Q by Stability Class

	A-B	C	D	E-F
RG 1.45 (1)	0.004164	0.004351	0.004484	0.004572
RG 1.45 (2)	0.013801	0.024114	0.047368	0.122098
RG 1.45 (3)	na	na	0.071051	0.122098

The limiting value for χ/Q is 0.122.

F (3) Source Term Release Rate

As described in Chapter 9, the reactor bay ventilation system is designed to provide at least 2 air changes per hour (2.29 m^3 s^{-1}, 2.29E6 cm^3 s^{-1}), and actually produces about 5 air changes per hour. Also described, a control system secures ventilation when the atmospheric contamination the reactor bay reaches a fraction of a DAC. Effluent is then driven by building leakage. Building leakage with the HVAC system secured is driven by pressure differential across porous barriers developed by winds, with the low pressure developed from a building wake. The ASHRAE Handbook of Fundamentals (2009) suggests a simple model for building leakage (SAE units) is:

$$Q = 2160 \cdot A \cdot \sqrt{\Delta P}$$

Where

Q is building leakage in cfm
2160 is a conversion factor
A is the "net open crack area of the room"
Δ*P* is differential pressure between the room and the surrounding environment

All door openings to the reactor bay are facing the same direction, all doors exit to buffer areas designed to support differential pressures. There is essentially no potential for differential pressure at the reactor bay openings from environmental conditions.

The equipment hatch has two hinged doors, 66 in. X 132.5 in. sealed in the center. All three personnel doors are 36 in. X 72 in. Therefore the total linear perimeter of all openings with the exception of the HVAC system is 1177 in. The HVAC system ductwork (separate supply and return) is approximately 2 ft. X 3 ft. at the inlet, 3 ft. XC 3 ft. at the outlet. The perimeter of the duct work is therefore 264 in. If a large ⅛ in. gap is assumed at all openings, the open surface

area for all doors is 1.02 ft^2. The duct surface area with the HVAC system operating is 15.00 ft^2, with the HVAC system secured is 0.22 ft^2.

Total area open for air flow with the system operating is the sum of the HVAC surface area and the door fittings, 16.02 ft^2. Total open area with the system secured is the leakage around the dampers, 0.22 ft^2. The ratio of surface areas in the two conditions is 1.37%. It is assumed that the pressure drop across the HVAC fan is at least a factor of 10 greater (than a pressure difference from ambient conditions) when the fan is operating. The ratio of flow rate while the HVAC system is operating can reasonably be expected to be reduced by a factor consisting of the ratio of the two specified areas and the square root of the ratio of the differential pressures driving flow, or reduction by a factor of 0.434%.

Leakage around system dampers and door-penetrations is not measured, but all openings are equipped with gaskets. While the ventilation system is a confinement system and not containment, reactor bay openings are equipped with rubber seals which are periodically checked for function. Therefore the simple model is likely to overestimate building leakage by a large margin.

The most limiting atmospheric dispersion factor (0.122098) and the conservative estimate of the building leakage factor (0.004343) provide a reduction in the airborne concentration of fission products as indicated in Table 13.10A and 13.10B as they are released (from the reactor bay) by a factor of 5.30E-4.

The concentration of the reactor bay fission product inventory (reduced by the minimum atmospheric dispersion in transit to unrestricted areas) is compared to the effluent limit, with the results provided in Table 13.19.

Table 13.19, Effluent Limit Ratio to Release Concentrations

		1 MW		2 MW		3.5 MW	
		Gaseous	Particulate	Gaseous	Particulate	Gaseous	Particulate
Seconds	1	0.5594	0.1214	1.1183	0.1559	1.9546	0.1946
Minutes	30	0.5488	0.1214	1.0977	0.1559	1.9191	0.1946
Hours	1	0.5425	0.1214	1.0849	0.1554	1.8963	0.1946
	8	0.4741	0.1214	0.9481	0.1548	1.6576	0.1935
Days	1	0.3956	0.1209	0.7912	0.1548	1.3835	0.1930
	7	0.1766	0.1204	0.3526	0.1538	0.6167	0.1909
	30	0.0239	0.1188	0.0483	0.1506	0.0838	0.1861
	90	1.45E-04	0.1167	2.84E-04	0.1458	4.91E-04	0.1782
	180	1.06E-05	0.1145	1.54E-05	0.1432	2.07E-05	0.1734
	365	1.01E-05	0.1129	1.48E-05	0.1405	1.96E-05	0.1697
Weighted average		0.0056	0.1145	0.0112	0.1429	0.0196	0.1734

In all cases for the maximum hypothetical accident, when the HVAC system is secured the annual effluent concentration limit is met.

The most significant conservatism in this analysis is the assumption that meteorological conditions maintain the lowest possible dilution factor for a year. This is obviously not supported in reality; any changes to meteorological conditions will increase dilution and reduce the concentration of the effluent.

As in the ALI analysis, this analysis is conservative in assuming a burnup of 10 grams ^{235}U and a continuous operating history. A slightly more realistic assumption of 6 grams burnup reduces the inventory of the long lived ^{90}Sr by approximately 60%, interpolating between the particulate ratios indicates that the DAC ratio is 1 at or below 3 MW, and a less aggressive operating schedule reduces shorter lived radionuclides considerably.

This analysis is conservative in assuming less than 100% dilution in the reactor bay volume, with 10% of the volume is occupied by equipment. Increasing the volume decreases nuclide concentration.

This analysis is conservative in assuming that the radionuclide inventory is not decreased in the transport from the reactor bay to the environment.

The reactor bay HVAC control system is designed to automatically secure ventilation on detecting a preset level of airborne contamination, and there is some delay before the radionuclides buildup to the trip level. During this interval the reactor bay continues to exhaust by design 34.3 $m^3 s^{-1}$, and an actual 57.2 $m^3 s^{-1}$. A reduction in reactor bay inventory reduces the radionuclide inventory to be released.

The 1% release value is unrealistically conservative. Ambient flow from a building is motivated by a pressure difference that is the result of wind. All reactor bay penetrations are on the south side of the building, and there are no access points on the east, west or north sides. Therefore any wind developed pressure will be constant across the openings, and there is no differential pressure to develop flow.

Therefore although the DAC values in the reactor bay are exceeded for the 2 and 3.5 kW case of the maximum hypothetical accident under extremely conservative assumptions, dilution of radionuclides at the point of release ensures 10CFR20 Appendix B effluent limits are met.

13.3.3 Results and Conclusions

Although the initial radionuclide release inventory exceeds the Annual Limit on Intake, there is no conceivable means of delivering the total inventory to a single worker. Although the radionuclide inventory in the reactor bay exceeds the limiting DAC values in the limiting case,

access control as required under the Radiation Protection Program would prevent exposure of any individual exceeding 10CFR20 limits. Effluent limits are met.

13.4 Insertion of Excess Reactivity

Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA reactor. Characteristics of a slow (ramp) reactivity insertion are less severe than a rapid transient since temperature feedback will occur rapidly enough to limit the maximum power achieved during the transient. Analyses of plausible accident scenarios reveal no challenges to safety limits for the TRIGA. The fuel-integrity safety limit, according to Simnad (1980), may be stated as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150°C for pulsing, stainless steel U-$ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500°C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950°C. ....*

Two reactivity accident scenarios are presented. The first is the insertion of 2.8% Δk/k ($4.00) reactivity at zero power (i.e., less than 1 kW) by sudden removal of a control rod. The second is the sudden removal of the same 2.8% Δk/k ($4.00) reactivity with the core operating at the maximum power level permitted by the balance of core excess reactivity. Maximum excess reactivity permitted is 4.9% Δk/k ($7.00); if 2.8% Δk/k ($4.00) is allocated to the reactivity transient, then 2.1% Δk/k ($3.00) supports power operation; 880 kW operation corresponds to a 2.1% Δk/k ($3.00) reactivity deficit. Analysis shows that in neither scenario does the peak fuel temperature exceed the temperature limit. The nearest approach occurs if the reactor is operating at a steady power of 880 kW, an action prevented both by administrative requirements and by interlocks, but there is adequate margin to the temperature limit for cladding that has a temperature less than 500°C. Chapter 5 shows that steady state temperatures are much less than 500°C.

13.4.1. Initial Conditions, Assumptions, and Approximations

The following conditions establish an extremely conservative scenario for analysis of insertion of excess reactivity:

- For the first scenario, the reactor is critical below 1 kW, with reactor and coolant ambient (zero power) temperature 27°C.

- For the second scenario, the reactor is operating at a steady state power level supported by core excess reactivity minus 2.8% Δk/k ($4.00) reserved for pulsing.

- Maximum pulsed reactivity insertion is 2.8% ($4.00)

- The time over which heat is generated and causes fuel temperature to rise is much shorter than heat transfer time constants for removal of the heat, so that analysis uses adiabatic conditions

- The reactivity addition is assumed to be instantaneous and greater than the delayed neutron fraction so that the contribution of delayed neutrons is small and therefore neglected in analysis

- A control rod interlock preventing pulsing operations from power levels greater than a maximum of 10 kW is not credited

- Operation at 880 kW with the pulse rod decoupled and fully inserted is assumed to be possible

- The core is assumed to contain 85 fuel elements to maximize the power generated in the hottest fuel element

- Hot channel factors calculated in Chapter 5 are used

13.4.2 Computational Model for Power Excursions

As noted in Chapter 4, TRIGA fuel has a strong negative temperature coefficient. Operating at power causes fuel to heat up, and the increase in temperature then contributes negative reactivity. The temperature increase is nearly instantaneous as fission products transfer kinetic energy to the fuel matrix, increasing the average kinetic energy/temperature compared to the heat transfer time constants for fuel and coolant. Rapid changes in core reactivity are therefore nearly adiabatic until the system has time to respond. Large, nearly instantaneous reactivity additions, pulses, in a TRGIA reactor are therefore characterized by a power excursion, fuel heatup, and power reduction (associated with the heatup) over a short time interval. The contribution of delayed neutrons is limited by the transient time interval.

Temperature is related to energy (power over an interval) through the specific heat capacity. Specific heat capacity of the fuel material (J/kg°K) at temperature T (°C) is given by:

$$c_{p.f} = c_0 + c_1 \cdot T$$

Where c_o is 340.1 and c_1 is 0.6952. With there are N fuel elements in the core, each with mass m_f(kg), then the core heat capacity (α, units of J/°K) is given by

$$C = \left[N \cdot m_f \right] \cdot \left(c_0 + c_1 \cdot T \right) \equiv C_0 + C_1 \cdot T$$

The following relationships are for the Fuchs-Nordheim model, modified by Scalletar, for power excursions, as described for the TRIGA reactor by West et al. (1967). The inhour equation for the prompt critical condition (if the delayed neutron time constants do not play a significant role in time dependent behavior) reduces to:

$$\frac{dP(t)}{dt} = \frac{\rho - \beta}{\ell} \cdot P(t)$$

Where ℓ is the prompt neutron lifetime (41 μs for TRIGA II reactors). The rise in temperature is related to the change in reactor power from some initial power level ($P(t_0)$) to a power level at some incremental time after the initial power ($P(t)$):

$$C \cdot \delta t = \left(P(t + \delta t) - P(t_0) \right) \cdot \delta t$$

The reactivity from fuel temperature ($\Delta k/k$) is characterized by the temperature coefficient (α) and the temperature (T):

$$\delta k = \alpha \cdot T$$

Leading to:

$$\frac{dP}{dT} = \frac{\left(\delta k - \alpha \cdot T \right) \cdot \left(C_o + C_1 \cdot T \right)}{\ell \cdot \left(P - P_0 \right)} \cdot P$$

On integration (where T=0 at $P=P_0$):

$$\ell \cdot \left\{ \left(P - P_0 \right) - P_0 \cdot \ln \frac{P}{P_0} \right\} = T \cdot \left(\delta k \cdot C_0 - \left(\alpha \cdot C_0 - C_1 \cdot \delta k \right) \cdot \frac{T}{2} - \alpha \cdot C_1 \cdot \frac{T^2}{3} \right)$$

Minimum and maximum temperatures occur at pulse initiation and after completion. For $P=P_0$,

$$T \cdot \left(\delta k \cdot C_0 - \left(\alpha \cdot C_0 - C_1 \cdot \delta k \right) \cdot \frac{T}{2} - \alpha \cdot C_1 \cdot \frac{T^2}{3} \right) = 0$$

Which simplifies to:

$$T^2 + \left[\frac{C_0}{C_1} - \frac{\delta k}{\alpha}\right] \cdot \frac{3}{2} \cdot T - 3 \cdot \frac{\delta k \cdot C_0}{\alpha \cdot C_1} = 0$$

With the quadratic equation solution:

$$T = \frac{1}{2}\left\{\left[\frac{C_0}{C_1} - \frac{\delta k}{\alpha}\right] \cdot \frac{3}{2} \pm \sqrt{\left(\left[\frac{C_0}{C_1} - \frac{\delta k}{\alpha}\right] \cdot \frac{3}{2}\right)^2 + 4 \cdot 3 \cdot \frac{\delta k \cdot C_0}{\alpha \cdot C_1}}\right\}$$

Only the positive half of the solution has physical meaning. Although there are nonlinear terms in the model, calculation of temperature change as a function of temperature shows a nearly linear response.

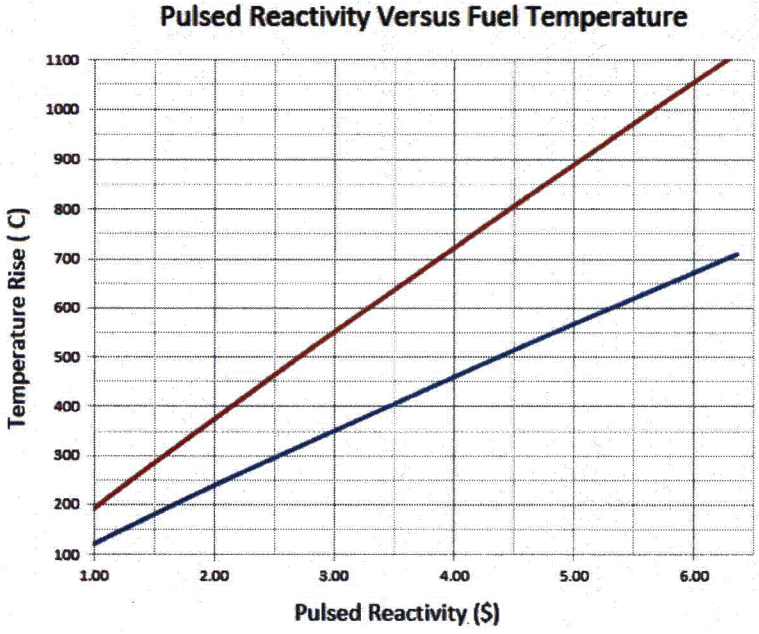

Figure 13.3, Fuel Temperature and Pulsed Reactivity

The average core temperature rise in response to a reactor pulse was calculated and tabulated in Table 13.20, along with the maximum temperature rise based on a radial peaking factor of $\pi/2$ as provided in Chapter 4. Maximum permissible pool temperature during reactor operations is 48.9°C; fuel temperatures that result from pulsed operations at the limiting pool temperature are provided in Table 13.20 based on the increase in fuel temperature caused by the pulse over pool temperature.

Table 13.20, Low Power Pulsed Reactivity Response

Rho	$	$\Delta T_{Fuel,Ave}$	$\Delta T_{Fuel,Peak}$	$T_{Fuel,Peak}$
0.00700	1.000	123	193	242
0.00786	1.123	137	216	264
0.00882	1.260	153	241	290

Table 13.20, Low Power Pulsed Reactivity Response

Rho	$	$\Delta T_{Fuel,Ave}$	$\Delta T_{Fuel,Peak}$	$T_{Fuel,Peak}$
0.00990	1.414	171	269	318
0.01111	1.588	192	301	350
0.01247	1.782	214	336	385
0.01400	2.000	239	375	424
0.01572	2.245	267	419	468
0.01764	2.521	298	468	516
0.01981	2.829	332	522	570
0.02223	3.176	370	582	631
0.02495	3.565	413	649	697
0.02801	4.002	460	723	772
0.03144	4.492	513	806	855
0.03529	5.042	571	898	947
0.03962	5.660	637	1000	1049
0.04447	6.353	709	1114	1163

At the maximum permissible pulsed reactivity insertion of 2.8% Δk/k, peak fuel temperature is 772°C, approximately 18.7% below the safety limit of 950°C with cladding temperature above 500°C and 32.9% below the safety limit of 1150°C with cladding temperature below 500°C. It should be noted that in pulsing from low power operations, the cladding temperature is determined by the pool water temperature so that the 1150°C safety limit applies.

The second case assumes 2.8% Δk/k ($4.00) of reactivity is reserved for a pulse, and the reactor is operating at the maximum steady state power level that can be supported by the balance of the excess reactivity (2.1% Δk/k, $3.00). Power level is conservatively assumed to be 880 kW. Assuming an 85 element core (initial criticality for the UT TRIGA) and a π/2 peaking factor, the hottest element produces 17.3 kW. As noted in Chapter 4, the temperature difference across the fuel matrix is calculated by:

$$\Delta T = q'' \cdot \frac{r_0}{2 \cdot k_f}$$

Where q'' is the heat flux across the outer cladding surface, k_f is the fuel conductivity (18 W m^{-1} K^{-1}) and r_o is the fuel diameter. The temperature difference from fuel center to the outer surface of the element is 195°C. The temperature difference for 17.3 kW from bulk water to the inner surface of the cladding was calculated in Chapter 4 to be 12°C. Fuel temperature is therefore 207°C. Using data previously calculated for temperature rise from pulse reactivity values, the peak fuel temperatures were calculated for pulsed reactivity values from $1.00 to approximately $4.5 and reported in Table 13.21.

Table 13.21, Initial Power 880 kW Pulsed Reactivity Response

Δk/k/k	$	$\Delta T_{Fuel,Ave}$	$\Delta T_{Fuel,Peak}$	$T_{Fuel,Peak}$
0.00700	1.000	123	193	400
0.00786	1.123	137	216	423
0.00882	1.260	153	241	448
0.00990	1.414	171	269	476
0.01111	1.588	192	301	508
0.01247	1.782	214	336	543
0.01400	2.000	239	375	582
0.01572	2.245	267	419	626
0.01764	2.521	298	468	675
0.01981	2.829	332	522	729
0.02100	3.000	351	551	758
0.02223	3.176	370	582	789
0.02495	3.565	413	649	856
0.02801	4.002	460	723	930
0.03144	4.492	513	806	1013

Pulsing to $3.00 from 880 kW, the hot channel has a margin of 20.1 below the safety limit of 950°C with cladding temperature above 500°C and 34.0% below the safety limit of 1150°C with cladding temperature below 500°C. Pulsing from $4.00 the margin to the safety limit of 950°C with cladding temperature above 500°C is only 2%, but the margin to the safety limit of 1150°C with cladding temperature below 500°C is 19.1%. Chapter 4 shows that for steady state operations the cladding temperature is below the fuel temperature, well below 500°C, so that the 1150°C limit and the 19.1% margin applies.

The postulated scenarios do not result in fuel damage, but physical aspects of system prevent these scenarios from occurring. It is not possible to achieve full power operation with the pulse rod fully inserted; since the pulse rod is partially withdrawn with air applied to the pulse solenoid, it physically cannot be pulsed. Although not required to ensure the safety of the reactor, an interlock prevents pulsing from power levels greater than a maximum of 10 kW.

13.4.3 Results and Conclusions

Insertion of the maximum possible reactivity of $4.00 without initial temperature feedback (i.e., fuel temperature is too low to limit core available reactivity) results in a peak hot spot well below the safety limit. Insertion of the $4.00 maximum possible reactivity with the reactor operating at power providing initial temperature feedback results in a peak hot spot fuel temperature well below the safety limit for cladding temperature greater than 500°C.

13.5 Loss of Reactor Coolant Accident

Although total loss of reactor pool water is considered to be an extremely improbable event, calculations have been made to determine (1) the maximum fuel temperature rise and (2) the maximum radiation dose that could be expected to result from such an event taking place after long-term operation at power levels up to 2 MW.

A TRIGA fuel element (with 8.5% uranium) useful life ends about 6 grams of burnup; a 10 gram burnup is used for the end of life as a conservative assumption. Slightly more than 1 gram is depleted per MWD of burnup, corresponding to 10 MWD per element. Calculations were performed using the T-6 depletion sequence SCALE to determine decay heat at shutdown from TRIGA fuel operated to 10 MWD per element (input file in Appendix I). TRIGA specific cross section libraries were generated as part of the sequence for use in determining the gamma source term from fission product decay. Scale is a comprehensive modeling and simulation suite for nuclear safety analysis and design developed and maintained by Oak Ridge National Laboratory under contract with NRC and DOE to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs. ORIGIN ARP was used to determine time dependent decay heat and gamma energy-spectrum intensity. Fuel temperature is calculated using the decay heat and first principles modeling of cooling. Radiation dose rates for receptor locations are modeled using MCNP with the gamma spectrum as a source term.

Discharge flow rate from a tank at atmospheric pressure (Streeter, V. L., E. B. Wylie, and K. W. Bedford, 1998, *Fluid Mechanics*. McGraw-Hill, Inc. 9ed, Daugherty, R. L., J. B. Franzinin, and E. J. Finnemore. 1985; *Fluid Mechanics with Engineering Applications*. Mc Grawe Hill, Inc. 8ed) is given by:

$$Q = a \cdot C \cdot \sqrt{2 \cdot g \cdot h}$$

Where:

- a is the diameter of a circular (drain) opening
- C is the loss coefficient associated with the opening
- h is the water height, subscripted i for initial and f for final
- g is the acceleration of gravity

Flow from a tank with a constant cross sectional area A is also characterized by:

$$Q = -A \cdot \frac{dh}{dt}$$

The time to drain a tank open to atmosphere between an initial level (H_i) and a final level (H_f) is calculated by substituting the differential into the first equation and integrating between the initial and final heights, with the result:

$$t = \frac{A}{a \cdot C} \cdot \left(\sqrt{H_i} - \sqrt{H_f} \right) \cdot \sqrt{\frac{2}{g}}$$

As described in Chapter 4, the pool has a composite surface are of a circle with radius of 39 in. (0.9906 m) and a 39 in. X 78 in. (0.9906 m X 1.9626 m) rectangle. Normal pool height is 8.1 m, with a reactor scram at 7.8 m. The loss coefficient is a dimensionless number between 0 and 1.0, with high turbulence as in a sharp edge losing more (61%) than from a short tube (80%). Since the discharge from the pool through a beam port travels through about 3 m with multiple abrupt changes in diameter, significant additional loss can be expected; for conservatism, a loss factor of 0.61 is assumed. If a beam port shears and falls completely out of the flow path while the beam port shutter is open and no shielding or obstructions to flow are in the beam line, a minimum of 5.0 minutes will be required to drain the pool coolant from 7.8 m to the top of the active fuel (47.25 in, 1.200 m above the pool floor). Therefore cooling analysis assumes a decay time of 5 minutes prior to uncovering fuel. Reduced shielding capability occurs as the water falls, but normal levels are adequate for full power operations and most of the radiation exposure source term during operation is from fissions, falling by a factor of about 0.053 at shutdown. Since (1) shielding requirements are significantly reduced and (2) the calculation of the time to drain the pool to the top of the reactor core, a 5 minute decay time is assumed for source term calculation.

This section demonstrates under extraordinarily conservative assumptions that maximum fuel temperature reached in a loss of coolant accident is well below any safety limit for TRIGA reactor fuel. Conservatism notwithstanding, the margin between computed temperature and design limits is sufficiently great to accommodate a design margin of at least a factor of two. Limiting design basis parameters and values for cooling consideration are addressed by Simnad (1980) as follows:

> *Fuel-moderator temperature is the basic limit of TRIGA reactor operation. This limit stems from the out-gassing of hydrogen from the ZrH_x and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. A fuel temperature safety limit of 1150°C for pulsing, stainless steel U-$ZrH_{1.65}$ ... fuel is used as a design value to preclude the loss of clad integrity when the clad temperature is below 500°C. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950°C. There is also a steady-state operational fuel temperature design limit of 750°C based on consideration of irradiation- and fission-product-induced fuel growth and deformation....*

13.5.1 Initial Conditions, Assumptions, and Approximations

The following conditions establish the scenario for analysis of the loss of coolant accident.

- The reactor is assumed to have been operating with 100 fuel elements for infinite time at power P_o = 2,000 kW when coolant is lost.
- Decay heat is calculated using a SCALE depletion sequence (T-6) based on core burnup to 10 MWD per element followed by decay over intervals
- Coolant loss is assumed to occur in 5 minutes.
- Reactor shutdown is assumed to occur with initiation of coolant loss.
- Decay heat is from fission product gamma and x rays, beta particles, and electrons. Effects of delayed neutrons are neglected.
- Thermal power is distributed across the core with a radial peak-to-average ratio of $\pi/2$. In individual elements, thermal power is distributed axially according to a sinusoidal function.
- Cladding and gap resistance are assumed to be negligible, i.e., cladding temperature is assumed to be equal to the temperature at the outside surface of the fuel matrix.
- Cooling of the fuel occurs via natural convection to air at inlet temperature T_i = 300°K. Radiative cooling and conduction to the grid plates are neglected.
- Heat transfer in the fuel is one dimensional, i.e., axial conduction is neglected, and fuel is assumed to be uniform in thermodynamic and physical properties.
- Heat transfer in the fuel is treated as pseudo-steady-state behavior, i.e., at any one instant, heat transfer is described by steady-state conduction and convection equations.[1]

13.5.2 Heat Transfer to Air

Fundamental relationships between buoyancy driven differential pressure and pressure losses from friction provide a means to calculate fuel and cladding temperature that results from the decay heat source, related by:

$$\delta P_b = \delta P_f + \delta P_e + \delta P_i + \delta P_a$$

Where

[1] See Todreas & Kazemi (1990) or El-Wakil (1971) for steady-state conduction equations.

δP_b is the buoyancy force

δP_f is the pressure difference from friction developed across the fuel

δP_e is the pressure difference across the exit restriction

δP_i is the pressure difference across the inlet restriction

δP_a is the pressure drop from acceleration\

A. Buoyancy Forces

If ρ_i and ρ_o are respectively the densities of air at the inlet and outlet temperatures,[2] the distance between the center of the zone (1/2 the fuel length L_f) in which the air is heated (inlet temperature) and the center of the zone in which the air is cooled by full mixing (outlet temperature) is 10 hydraulic diameters above the core exit, the distance from the heated length to the core exit is L_t, the acceleration of gravity (9.8 m s^{-2}) is g, the buoyancy pressure difference is given by:

$$\Delta p_b = (\rho_i - \rho_o) \cdot g \cdot \left(\frac{L_f}{2} + L_t + 10 \cdot D_h \right)$$

B. Friction Losses

Friction losses across the lower unheated length, heated length and upper unheated length are given by:

$$\delta P_f = f_{F,i} \cdot \frac{4 \cdot L_i}{D_e} \cdot \frac{W^2}{2 \cdot g \cdot \rho_i \cdot A_{c,i}^2} + f_{F,f} \cdot \frac{4 \cdot L_f}{D_e} \cdot \frac{W^2}{2 \cdot g \cdot \rho_f \cdot A_{c,f}^2} + f_{F,e} \cdot \frac{4 \cdot L_e}{D_e} \cdot \frac{W^2}{2 \cdot g \cdot \rho_e \cdot A_{c,e}^2}$$

Where

L_x is the length of the x component

$f_{F,x}$ is the friction factor ($23.46/R_e$) for the x section

(x is lower, heated, and upper lengths)

$A_{c,x}$ flow area for the x section per element

C. Losses from Flow Restrictions

Inlet and exit losses are calculated by:

[2] Density at 1 atm, for air as an ideal gas, is given by ρ (kg/m^3) = 353.0/T(°K). Heat capacity, from 300 to 700 °K is 1030 J/kgK ± 3% (Incropera and DeWitt, 1990).

$$\delta P_e + \delta P_i = \frac{W^2 \cdot k_i \cdot \left[\frac{A_c}{A_i}\right]^2}{2 \cdot g \cdot \rho_i \cdot A_{c,f}^2} + \frac{W^2 \cdot k_e \cdot \left[\frac{A_c}{A_e}\right]^2}{2 \cdot g \cdot \rho_e \cdot A_{c,f}^2}$$

Acceleration losses are given by:

$$\delta P_a = \left(\frac{1}{\rho_i} - \frac{1}{\rho_o}\right) \cdot \frac{w^2}{g \cdot A_c^2}$$

Using the definition for Reynolds number and values for elevation previously described:

$$\left[\frac{0.700}{\rho_1} - \frac{0.149}{\rho_2}\right] \cdot 10^{-4} \cdot w^2 + \left[\frac{0.153 \cdot \mu_1}{\rho_1} + \frac{0.153 \cdot \mu_{ave}}{\rho_{ave}} + \frac{0.153 \cdot \mu_2}{\rho_2}\right] \cdot 10^{-2} w +$$

$$\left[1.25 \cdot \rho + 1.25 \cdot \rho_1 - 1.25 \cdot \rho_0\right] = 0$$

Where flow w is in lb/h and viscosity μ is in units of lb/h-ft. the properties for air for use in the equation are expressed as:

$$\rho_x = \frac{40}{T_x}$$

And μ is:

$$\mu_x = 5.739x10^{-3} + 7.601x10^{-5} \cdot T_x - 1.78x10^{-8} \cdot T_x^2$$

Where T is in units of °R, the heat transfer coefficient is calculated through:

$$N_u = 6.3 \qquad \text{Ra} \leq 1000$$

$$N_u 0.806 \cdot R_a^{0.2976} \qquad \text{Ra} > 1000$$

Where the Nusselt number is:

$$N_u = \frac{h \cdot D_e}{k}$$

The Rayleigh number is:

$$R_a = \frac{D_e \cdot \rho^2 \cdot g \cdot \beta \cdot \delta T \cdot c_p}{\mu \cdot k \cdot L}$$

Thermal conductivity (from a least squares fit to data presented by Etherington) is:

$$k = 2.377x10^{-4} + 2.995x10^{-5} \cdot T + 14.738^{-9} \cdot T^2$$

The specify heat capacity for air (also least squares fit, Etherington) is:

$$c_p = 2.413x10^{-1} - 1.780x10^{-6} \cdot T + 1.018x10^{-8} \cdot T^2$$

Volumetric expansion coefficients is β, δT is the temperature rise over the channel. The expression for the Nusselt number was developed from the work of Sparrow, Loeffler, and Hubbard for laminar flow between triangular arrays of heated cylinders. The parameters derived above were used as input data for a General Atomics 2-dimensional transient-heat transport compute code used for calculating the systems temperatures after a loss of pool water. Maximum temperature reached by the fuel are plotted as a definition of operating power density in Fig. 13.4A for several cooling or delay times between reactor shutdown and loss of coolant from the core.

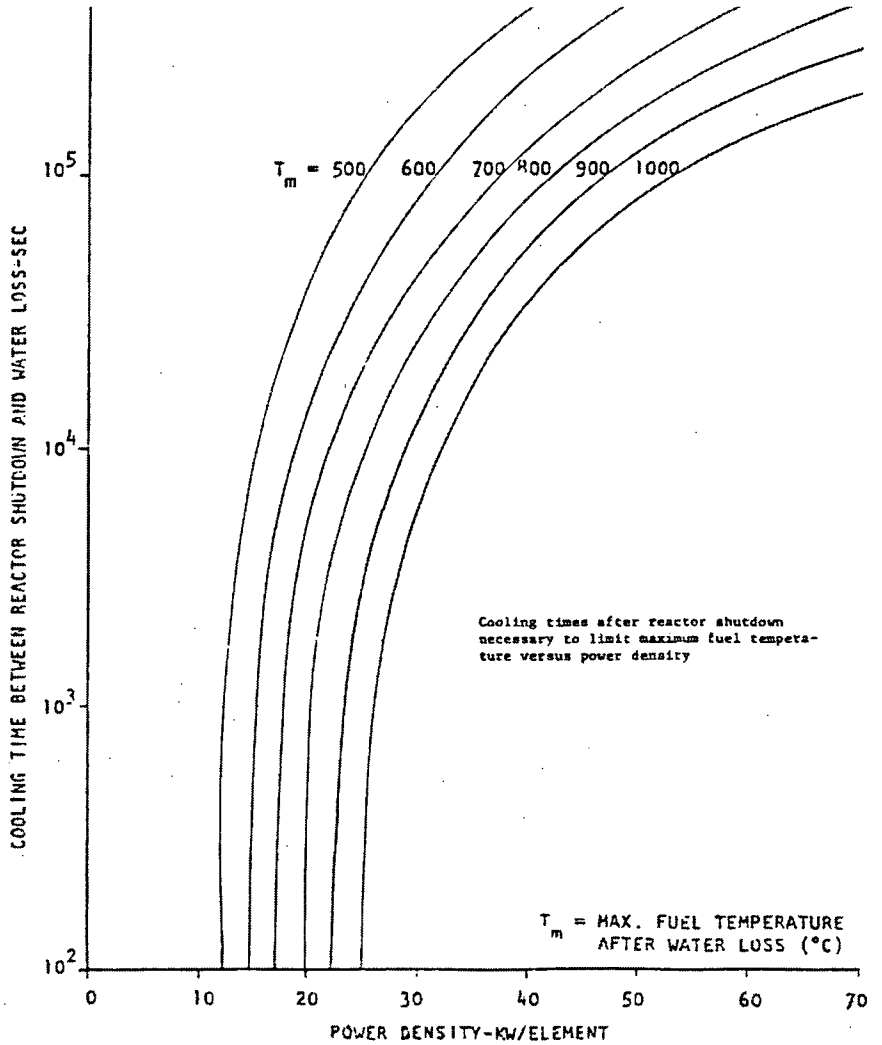

Figure 13.4A, Cooling Time

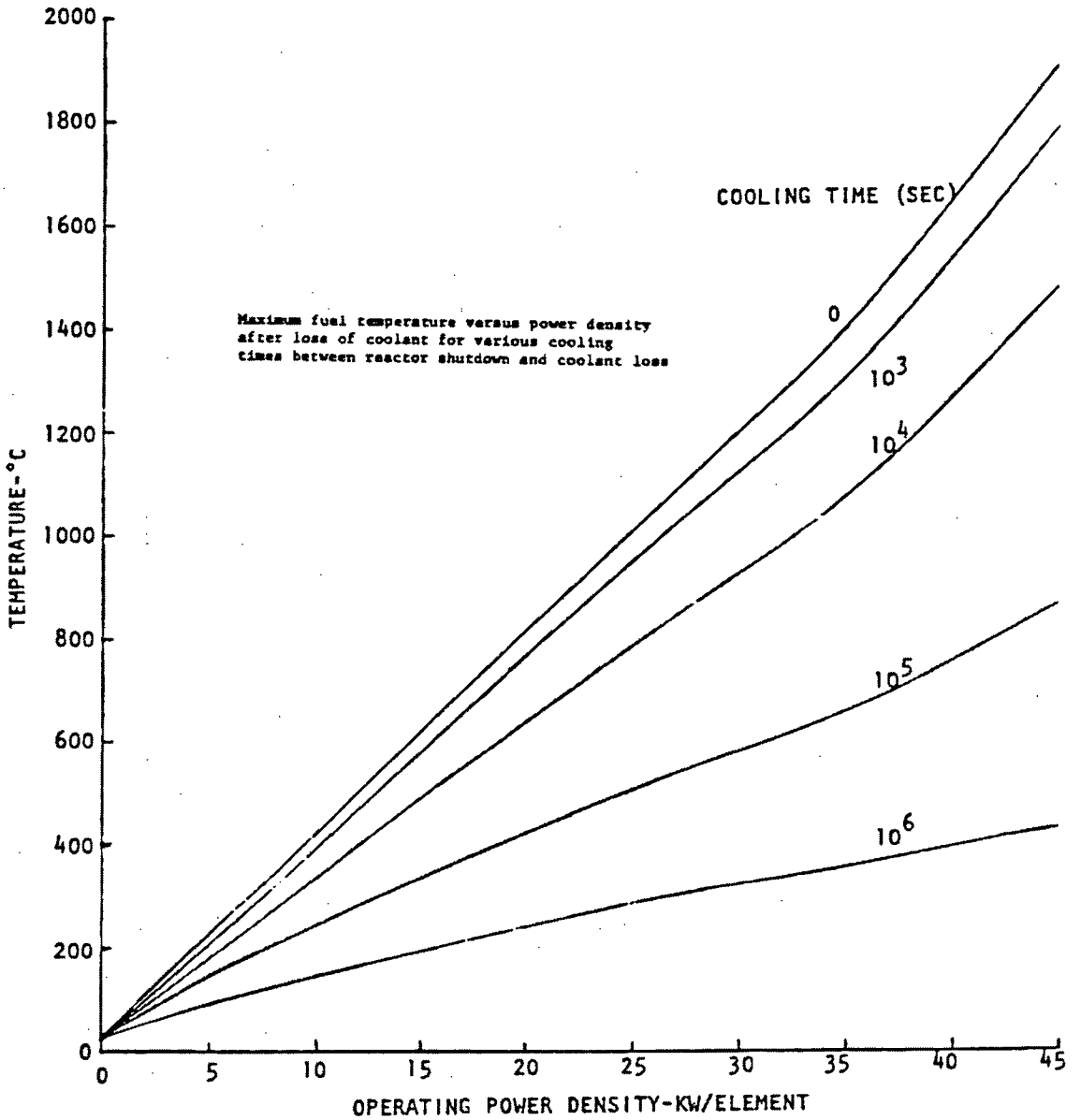

Figure 13.4B, Cooling Time and Power Density

For reactor operation with maximum power density of 18 kW/element or less, loss of coolant water immediately upon reactor shutdown would not cause the maximum fuel temperature to exceed 750°C. Operation at maximum power densities above 18 kW/element will not cause fuel temperatures above 750° C if coolant loss occurs sometime after shutdown, with the decay time required depending on power density. Therefore, minimum 5 minutes to effect loss of pool water adds additional margin.

In Fig. 13.4B, data was developed to show time required for natural convective cooling after reactor shutdown to produce temperatures less than a given value. For instance, temperature less than 950°C after operating with a maximum power density of 27 kW/element requires a

shutdown interval of 3730 s (1.04 h) after shutdown, when decay heat will be low enough that air cooling is adequate. A 65 minute delay time applies to power density corresponding to a 90 element core, and is negligible for power density in a 100 element core.

13.5.7 Radiation Levels from the Uncovered Core

Although there is only a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered dose rates from an uncovered core following 1,000, 2,000 and 3,500 kW operations are calculated. This section describes calculations of on-site and off-site radiological consequences of the loss-of-coolant accident. Extremely conservative assumptions are made in the calculations, namely, operation at 2,000 kW for one year followed by instant and simultaneous shutdown and loss of coolant. The SCALE depletion sequence (previously referenced for decay heat calculation) is used to generate TRIGA specific cross section libraries for use in ORIGIN ARP for operation over the life of the core (10 MWD per element). Gamma-ray source strengths, by energy group, are determined by an ORIGEN ARP calculation. Radiation transport calculations use the MCNP code.

Table 13.22, Gamma Source Term

	1	30	1	8	1	7	30	90	365
MeV	Sec	Min	Hours		Days				
0.01	1.80E19	7.11E18	5.94E18	3.25E18	2.32E18	1.33E18	8.57E17	5.48E17	2.58E17
0.025	9.09E18	3.30E18	2.79E18	1.63E18	1.18E18	5.47E17	3.39E17	2.19E17	1.02E17
0.0375	6.62E18	2.74E18	2.40E18	1.61E18	1.30E18	5.86E17	2.70E17	1.58E17	7.54E16
0.0575	2.17E18	8.30E17	7.01E17	4.15E17	3.09E17	1.55E17	8.93E16	5.46E16	2.58E16
0.085	2.12E18	7.93E17	6.07E17	3.31E17	2.61E17	1.32E17	5.93E16	3.90E16	1.90E16
0.125	1.60E18	5.90E17	5.34E17	3.89E17	3.05E17	8.03E16	3.22E16	2.07E16	1.02E16
0.225	9.64E17	3.48E17	2.87E17	1.69E17	1.38E17	5.86E16	3.01E16	1.57E16	5.99E15
0.375	6.54E17	2.58E17	2.06E17	1.37E17	9.93E16	3.15E16	8.72E15	3.34E15	1.60E15
0.575	5.68E7	2.09E17	1.76E17	1.05E17	7.69E16	3.90E16	1.76E16	5.19E15	9.38E14
0.85	5.59E17	2.60E17	2.34E17	1.58E17	1.26E17	7.20E16	5.23E16	3.27E16	6.30E15
1.25	2.69E17	9.11E16	6.72E16	1.70E16	8.41E15	2.61E15	7.44E14	2.02E14	9.37E13
1.75	1.49E17	5.21E16	4.22E16	1.97E16	1.57E16	1.06E16	3.02E15	1.76E14	3.49E13
2.25	5.82E16	1.57E16	1.08E16	1.70E15	5.79E14	1.71E14	9.76E13	7.98E13	4.10E13
2.75	3.70E16	9.62E15	7.18E15	1.10E15	6.39E14	4.50E14	1.30E14	6.62E12	9.61E11
3.5	1.43E16	1.13E15	8.01E14	3.10E13	8.51E12	6.19E12	1.86E12	1.62E11	5.42E10
5	4.53E15	1.63E14	9.41E13	2.73E12	2.26E10	3.99E6	3.80E6	3.39E6	2.04E6
7	5.22E14	4.27E10	1.50E10	2.14E9	4.30E7	8.78E2	8.59E2	8.16E2	7.15E2
9.5	1.02E12	8.30E1	8.22E1	8.22E1	8.22E1	8.17E1	7.98E1	7.58E1	6.63E1

Modeling of the reactor core (Appendix 13.2) was performed using approximate geometry described in Fig. 13.5 and 13.6. The TRIGA reactor core is approximated as a right circular cylinder with the outer diameter of the G ring and a fuel region 0.381 m (15 in.) high. Axial zones are defined above and below the fuel, and at the grid plate elevations. The zones are described in Table 13.23, including the height of the zone and identification of materials in

locations defined by fuel positions (FUEL POS) and materials outside the fuel positions (CHANNEL).

Table 13.23, Height/Thickness Dimensions of Unit Cell

Zone	THICKNESS/LENGTH			CHANNEL	FUEL POS
1	LOWER GRID PLATE	3.27	cm	AL	VOID, SS
2	LOWER ELEMENT	12.70	cm	VOID	
	(1) End Cap (Lower)	5.09	cm	VOID	VOID, SS
	(2) Graphite	8.36	cm	VOID	Gr, SS
3	FUEL	38.10	cm	VOID	FUEL, SS
4	UPPER ELEMENT	11.58	cm	VOID	
	(1) Graphite	8.36	cm	VOID	Gr, SS
	(2) End Cap (Upper)	3.22	cm	VOID	VOID, SS
5	UPPER GRID PLATE	1.59	cm	AL	VOID, SS

Mass fractions of material components are calculated assuming a unit cell based on the fuel element pitch. A unit cell is the total area defined by the section of three fuel elements that lie with in the area formed by connecting three fuel center points (Table 13.24, Unit cell Area). Materials within the unit cell are eight fuel, graphite (assumed to have the same cross section as fuel), cladding, or void. The areas are listed in Table 13.24.

Table 13.24, Unit Cell Areas

UNIT CELL		
Unit cell area	8.2071	cm^2
Fuel	4.7886	cm^2
Cladding	0.3397	cm^2
Channel void	3.0788	cm^3

Materials in the volumes described by the heights/thicknesses Table 13.23 and the areas in Table 13.24 are homogenized based on material characterizations in Table 13.25.

Table 13.25, Material Characterization

COMPONENT	VALUE	UNIT
FUEL		
U^{235}	38.00	G
U^{238}	156.87	G
Zr	2052.38	G
H	45.36	G
SS 304		
Fe	0.6993	%

Cr	0.1900	%
Ni	0.1000	%
Mn	0.0200	%
Si	0.0100	%
P	0.0005	%
S	0.0003	%
Graphite	2.25	g/cc
Aluminum	2.70	g/cc
SS 304	8.03	g/cc

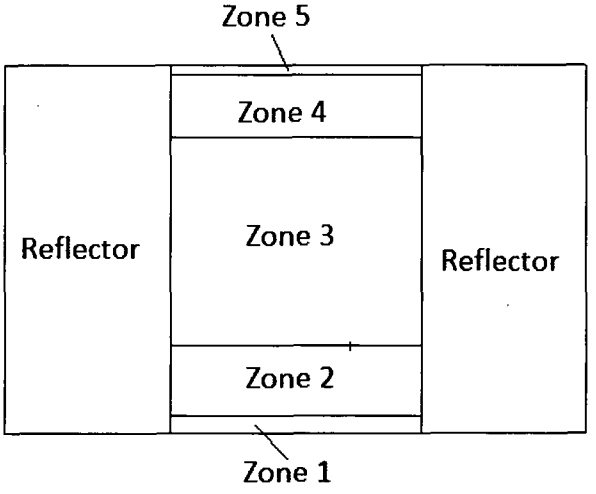

Figure 13.5, Core Model

Biological shielding is approximated as a two-section concrete cylinder based on dimensions in Chapter 4. The structure was simplified as rectangular for this calculation, and the top deck neglected.

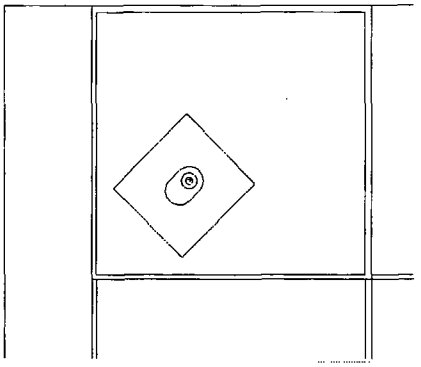

Figure 13.6A, Bay Model Top View

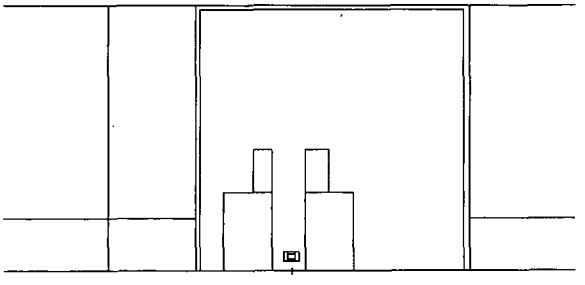

Figure 13.6B, Bay Model Cross Section

The site boundary is about 75 m at its nearest approach to the north wall of the reactor bay (87.5 m from the core center), with a fence erected 70 m from the reactor bay wall (82.5 m from

the core center). Receptor locations for dose calculations inside the reactor bay were set at 1 foot from (1) the ground floor personnel door, (2) the center of the truck door, (3/4) in line with the core at the north and west walls, (5) the top floor personnel door, and directly over the core. Receptor locations for dose calculations outside the reactor bay were set 1 foot outside the walls of the reactor bay in line with the core on the three sides with exterior faces. Additional points were set 80 and 90 meters from the core center.

The building geometry is simplified to single thickness walls, and the floor structures are neglected. The colocation boundary extends about 4 meters into the ground below the reactor bay, and spherically to approximately 700 meters.

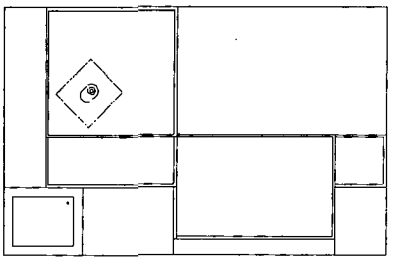

Figure 13.7A, Building Model

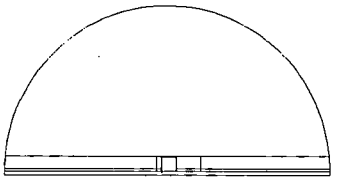

Figure 13.7B, MCNP Side View

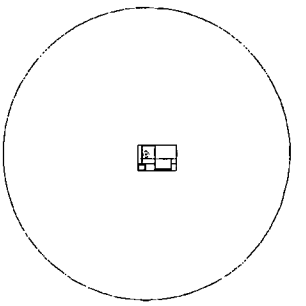

Figure 13.7C, Top View

The results of the calculation are provided in Table 13.26.

Table 13.26, Post LOCA Doses

	Sec	Min.	hours		days				
	1	30	1	8	1	7	30	90	365
	R/h								
Lower bay door	3.66	0.28	0.228	0.084	0.064	0.106	0.042	0.025	0.010
Lower bay east wall	3.64	0.28	0.207	0.101	0.065	0.111	0.045	0.023	0.011
Lower bay west wall	4.75	0.35	0.264	0.120	0.083	0.123	0.058	0.032	0.013
Mid-truck door	4.65	0.36	0.286	0.112	0.087	0.154	0.077	0.042	0.013
Top deck over core	14801	948	754	301	206	324	135	73	28
Top deck door	26.58	1.82	1.590	0.704	0.489	0.720	0.311	0.190	0.068
	mR/h								
Outside east wall	0.906	0.0712	0.0529	0.0107	0.0079	0.0165	0.0067	0.0035	0.0015
Outside west wall	1.547	0.1062	0.0923	0.0341	0.0204	0.0374	0.0144	0.0083	0.0026
Outside north wall	1.035	0.0678	0.0590	0.0232	0.0107	0.0167	0.0069	0.0047	0.0040
Approx. Parking Lot	2.475	0.0732	0.0659	0.0145	0.0151	0.0180	0.0070	0.0053	0.0019
Approx. Fence Line	1.615	0.0722	0.0540	0.0121	0.0132	0.0134	0.0063	0.0056	0.0023

13.5.8 Results and Conclusions

Although a loss of pool water is considered to be an extremely improbable event, calculations show the maximum fuel temperature that could be expected to result from such an event (after long-term operation at full power of 2,000 kW is 750°C, well below any safety limit for TRIGA reactor fuel.

Maximum possible dose rates resulting from a complete loss of pool water permit mitigating actions. The area surrounding the reactor is under control of the University of Texas, and exposures outside the reactor bay environment can be limited by controlling access appropriately. The University of Texas has complete authority to control access to campus locations.

13.6 Loss of Coolant Flow

13.6.1 Initialing 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, blockage of a coolant flow channel, or operator error.

The UT TRIGA reactor pool tank holds 40.57 m^3 (10717 gallons) of water, or about 40570 kg of water. At a steady-state power level of 1 MW, the bulk water temperature would increase adiabatically at a rate of about 20.74°C MW^{-1} h^{-1}.

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. Control console instruments indicate pool temperature, heat exchanger inlet and outlet temperature. Alarms are provided for heat exchanger low differential pressure (pool to chill water), pool water temperature, and abnormal water level (hi or low). A reactor scram occurs at low-low water level. These indicators allow the operator to observe an abnormal condition and make corrections or secure operations, and prevent operating the reactor with low water pool water level.

13.6.2 Accident Analysis and Determination of Consequences

If the UT TRIGA 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 0.35°C m^{-1} for

approximately 66 minutes until the tank water reached the maximum allowed operating temperature. It is considered inconceivable that such an operating condition with the attendant alarms and indications would not be undetected.

If it is assumed that the operator or automatic control system continued to maintain power at 1 MW, and assuming that the system is adiabatic except for the evaporation process, pool water would evaporate until the pool low level scram setpoint is reached and the reactor would shutdown.

13.7 Mishandling or Malfunction of Fuel

13.7.1 Initiating Events and Scenarios

Events which could cause accidents at the UT TRIGA in this category include:

- Simple failure of the fuel cladding due to a manufacturing defect or corrosion) and
- Fuel handling accidents where an element is dropped underwater and damaged severely enough to breach the cladding,
- Overheating of the fuel with subsequent cladding failure during steady-state or pulsing operations.

In the experience at UT, cladding failures from manufacturing defects occur before the element has enough operating history to generate a significant quantity of fission products.

13.7.2 Analysis

Gaseous fission product releases in water are delayed or partially retained (because of gas solubility) slowed (in the case of gas). Particulate fission product releases are substantially retained. Therefore a cladding failure under water is bounded by cladding failure in air, the maximum hypothetical accident.

13.8 Experiment Malfunction

13.8.1 Accident Initiating Events and Scenarios

Improperly controlled experiments involving the UT TRIGA reactor 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 room causing damage from the pressure release or an uncontrolled release of radioactivity. Other mechanisms for damage, such as large reactivity changes, are also possible.

13.8.2. 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 UT Reactor Procedures and the 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.

A. Administrative Controls

Safety related reviews of proposed experiments require the performance of specific safety analyses of proposed activities to assess such things as generation of radio nuclides and fission products, 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 UT TRIGA reactor.

B. Reactivity Considerations

A Technical Specifications limit of $1.00 has been placed on the reactivity worth of non-secured experiments. This is designed to prevent an inadvertent pulse by experiment manipulation, and is well below the maximum reactivity limit analyzed in the insertion of excess reactivity of 13.4.

A Technical Specifications limit of $1.00 has been placed on the reactivity worth of any single experiment. This is designed to prevent an inadvertent pulse by experiment manipulation while operating at power, and is well below the maximum reactivity limit analyzed in the insertion of excess reactivity of 13.4. Since these experiments are secured the transient that occurs from removal while operating at power will be less severe, and reactor protective systems are expected to terminate operations.

A Technical Specifications limit of $3.00 has been placed on the reactivity worth of all experiments during an operation. Removal of all experiments while operating is bounded by the positive reactivity addition analysis.

C. Fueled Experiment Fission Product Inventory

Limiting the generation of certain fission products in fueled experiments ensures 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. DAC ratio, as previously used, indicates the radionuclide concentration to which an exposed individual can receive 5 rem TEDE in a 2000 hour exposure. The DAC ratio for the activity of a specific nuclide (A_x) of an element distributed in a volume (V) is defined by:

$$F_x = \frac{A_x / V}{DAC_x}$$

The sum of the fractions for all nuclides determines an effective DAC fraction which meets DAC requirements if the sum is less than or equal to 1. For a fission product distribution yield across an element, if the yield is defined as Y% then the fraction can be calculated:

$$F_x = \frac{A_E / V \cdot Y\%}{DAC_x}$$

Therefore the total DAC fraction for the element is calculated:

$$F = \frac{A_E}{V} \cdot \sum_x \frac{Y\%}{DAC_x}$$

For a target DAC fraction, activity can be calculated:

$$A_E = F \cdot \frac{V}{\sum_x \frac{Y\%}{DAC_x}}$$

The ORIGEN source term calculations were used to calculate fractional fission product yields for iodine and strontium. The calculation assumes a 5 minute decay time after the reactor is shut down until the source term calculations are initiated; this is conservative from a practical perspective in considering the removal process. The weighted elemental yield fraction, and the weighted yield normalized to reactor bay volume is provide in Table 13.27.

Table 13.27, Calculations Supporting Limits on Fueled Experiments

isotope	Isotope Yield	Isotope DAC	Weighted Yield
i125	6.6E-15	3.0E-8	2.2E-7
i128	1.6E-5	5.0E-5	3.3E-1
i129	8.7E-9	4.0E-9	2.2
i130	9.2E-5	3.0E-7	3.1E2
i131	1.0E-1	2.0E-8	5.2E6
i132	1.5E-1	3.0E-6	5.1E4
i133	2.4E-1	1.0E-7	2.4E6
i134	2.8E-1	2.0E-5	1.4E4
i135	2.2E-1	7.0E-7	3.2E5
			8.0E+06
		VOL/SUM	4.66E2
sr85	1.41E-11	6E-7	2.36E-5
sr85m	7.65E-12	3E-4	2.55E-8
sr87m	2.20E-8	5E-5	4.39E-4
sr89	2.67E-1	6E-8	4.45E6
sr90	6.96E-2	2-9	3.48E7
sr91	3.28E-1	1E-6	3.28E5
sr92	3.35E-1	3E-6	1.12E5
			3.97E7
		VOL/SUM	9.35E1

For a 2-hour evacuation period, the DAC fraction is 1000; therefore a total iodine activity of 4.66E5 μCi will allow an individual to meet the annual 10CFR20 dose limits for radiation workers assuming a 2-hour evacuation period, and 9.32E5 μCi will allow an individual to meet the annual 10CFR20 dose limits for radiation workers assuming a 1-hour evacuation period. Similarly, a 9.35E4 μCi strontium inventory is acceptable for a 2-hour evacuation period. Therefore, limiting experiment radioiodine and strontium inventories in experiments will assure that there is adequate time for taking corrective actions.

D. Explosives

Projected damage to the reactor from experiments involving explosives varies significantly depending on 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 UT TRIGA reactors 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 13.28 lists the mechanical properties of these encapsulation materials.

Table 13.28, Material Strengths

Material	Yield Strength (Kpsi)	Ultimate Strength (Kpsi)	Density (g/cm^3)
Stainless Steel (304)	35	85	7.98
Aluminum (6061)	40	45	2.739
Polyethylene	1.7	1.4	0.923

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:

$$\sigma_{max} = \frac{p \cdot d}{2 \cdot t}$$

Where

σ_{max} is the maximum hoop stress in the container wall
p is the total pressure in the container
d is the diameter of the container, and
t is the container wall thickness

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:

$$\frac{p \cdot d}{2 \cdot t} \leq \sigma_{yield}$$

Solving this equation for d/t provides an easy method of evaluating an encapsulation material:

$$\frac{d}{t} \leq \frac{2 \cdot \sigma_{yield}}{p}$$

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 13.28. 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.

Table 13.29, Container Diameter to Thickness Ratio

Material	*d/t*
Stainless Steel (304)	4.5
Aluminum (6061)	5.1
Polyethylene	0.22

13.9 Loss of Normal Electric Power

13.9.1 Initiating Events and Scenarios

Loss of electrical power to the UT TRIGA reactor could occur due to many events and scenarios that routinely affect commercial power.

13.9.2 Accident Analysis and Determination of Consequences

Since the UT TRIGA does not require emergency backup systems to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. Backup power for lighting is provided by an emergency diesel on the Pickle Research Campus, and there are emergency exit lights and hand-held battery-powered lights 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.

13.10 External Events

13.10.1 Accident Initiating Events and Scenarios

Hurricanes, tornadoes, and floods are virtually nonexistent in the area around the UT TRIGA reactor. Therefore, these events are not considered to be viable causes of accidents for the reactor facility. In addition, seismic activity in the area as indicated in Chapter 2 is acceptably low.

13.10.2 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 previously analyzed, and, therefore, no additional specific accidents are analyzed in this section.

13.11 Experiment Mishandling or Malfunction

13.11.1 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 UT TRIGA depend on confinement or containment systems. Loss of pool water was previously addressed. Although no damage to the reactor occurs as a result of these leaks, the details of the previous analyses provide a more comprehensive explanation.

13.10.2 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.

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'     Mixture Compositions
' ------------------------------------------------------------------
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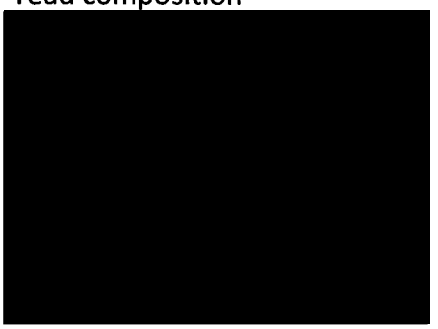

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' Aluminum for sheet then smeared for RSR volume
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                      6012  20.0
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 time=days
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 title="decay heat (watts)"
 units=watts
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[redacted]0)"
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 media 1 1 11 -12
 media 5 1 12
 media 5 1 13
 media 3 1 30 -20
 media 4 1 20 -10 -11 -13
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 media 4 1 11 -10
 media 3 1 20 -11
 boundary 20
unit 3
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 hexprism 10   2.177   31.909   -31.909
 media 3 1 10
 boundary 10
' ***************
unit 4
com="UNIT 4: STANDARD CONTROL ROD"
Com="aluminum above boron region"
 cylinder 10   1.619   31.909   19.05 ORIGIN X=0 Y=0 Z=20
Com="Boron region"
 cylinder 11   1.619   19.05   -19.05 ORIGIN X=0 Y=0 Z=20
Com="al spacer between boron and fuel"
 cylinder 12   1.619  -19.05   -21.59 ORIGIN X=0 Y=0 Z=20
[redacted]
 media  7 1 10 -11  -12
 media 10 1 11 -12
 media  7 1 12
 media  1 1 13 -14
 media  2 1 14
 media  4 1 20 -10 -11 -12 -13
 media  3 1 30 -20
 boundary 30
' ***************
Unit 5
' com="UNIT 5: PULSE ROD"
Com="aluminum above boron region (inside 10)"
 cylinder 10   1.519   31.909   19.05 ORIGIN X=0 Y=0 z=38.1
Com="Boron region (inside 11)"
 cylinder 11   1.519   19.05   -19.05 ORIGIN X=0 Y=0 Z=38.1
Com="Aluminum spacer (inside 12)"
 cylinder 12   1.519  -19.05   -21.59 ORIGIN X=0 Y=0 Z=38.1
Com="Air in follower (inside 13)"
 cylinder 13   1.519  -21.59   -64.61 ORIGIN X=0 Y=0 Z=38.1
Com="Cladding (inside 20, not in 10, 11, 12, 0r 13)"
 cylinder 20   1.59   31.909  -64.61 ORIGIN X=0 Y=0 Z=38.1
Com="Control rod cell boundary (water inside 30, not in 20)"
```

```
hexprism 30   2.177  31.909 -31.909
'
media  7 1 10
media 10 1 11
media  7 1 12
media  9 1 13
media  7 1 20 -10 -11 -12 -13
media  3 1 30 -20
boundary 30
' ***************
```

[redacted]

```
media 6 1  40  -24  -11  -50  -51  -52  -53
media 9 1  50  40  -11
media 9 1  51  40  -11
media 9 1  52  40  -11
```

```
 media 9 1  53   40   -11
' RSR
 media 9 1 20  -11
 media 7 1 21  -20
 media 7 1 22  -21
 media 8 1 23  -22
 media 7 1 24  -23
 boundary 40
end geometry
read array
ara=1 nux=15 nuy=15 nuz=1 typ=shexagonal
 fill
' ------------------------------------------------------------
' INITIAL CRITICALITY 3/16/1992
' LHS                                    RHS
' BOTTOM OF ARRAY
' AX are apexes, W water, F fuel, G graphite, PR pulse rod,
' RR reg rod, SX shim rod
' S source is ' not modeled - position is water filled
```

```
' TOP OF ARRAY
' ------------------------------------------------------------
```

```
 end fill
end array
end data
end model
end
#shell
copy ft71f001 "%RTNDIR%"\TRIGA.ft71
end
```

```
'This SCALE input file was generated by
'OrigenArp Version  6.1 Compiled on Thu Oct  7 11:31:00 2010
#shell
copy "C:\NEW SCALE CALCS\TRIGA GAMMA SOURCE TERM.f71" "ft71f001"
end
#origens
0$$ a11 71 e t
Decay Case
3$$ 21 1 1 0 a16 2 a33 18 e t
35$$ 0 t
54$$ a8 1 a11 0  e
56$$ a2 10 a6 1 a10 0 a13 -21 a15 3 a17 2 e
57** 0 a3 1e-05 e
95$$ 0 t
case 1
0 MTU
60** 0.001042 0.003125 0.009375 0.028125 0.041667 0.125 0.375 1 3 9
61** f0.05
65$$
'Gram-Atoms  Grams  Curies  Watts-All  Watts-Gamma
 3z  1  0  0  3z  3z  3z  6z
 3z  1  0  0  3z  3z  3z  6z
 3z  1  0  0  3z  3z  3z  6z
81$$ 2 0 26 1 e
82$$ 2 2 2 2 2 2 2 2 2 e
83**
 1.1000000e+07 8.0000000e+06 6.0000000e+06 4.0000000e+06 3.0000000e+06
 2.5000000e+06 2.0000000e+06 1.5000000e+06 1.0000000e+06 7.0000000e+05
 4.5000000e+05 3.0000000e+05 1.5000000e+05 1.0000000e+05 7.0000000e+04
 4.5000000e+04 3.0000000e+04 2.0000000e+04 0.0000000e+00 e
t
56$$ 0 0 a10  1 e t
56$$ 0 0 a10  2 e t
56$$ 0 0 a10  3 e t
56$$ 0 0 a10  4 e t
56$$ 0 0 a10  5 e t
56$$ 0 0 a10  6 e t
56$$ 0 0 a10  7 e t
56$$ 0 0 a10  8 e t
56$$ 0 0 a10  9 e t
56$$ 0 0 a10  10 e t
54$$ a8 1 a11 0  e
56$$ a2 5 a6 1 a10 10 a15 3 a17 2 e
```

```
57** 9 a3 1e-05 e
95$$ 0 t
Case 2
0 MTU
60** 27 30 90 180 365
61** f0.05
65$$
'Gram-Atoms  Grams  Curies  Watts-All  Watts-Gamma
 3z  1  0  0  3z  3z  3z  6z
 3z  1  0  0  3z  3z  3z  6z
 3z  1  0  0  3z  3z  3z  6z
81$$ 2 0 26 1 e
82$$ 2 2 2 2 2 e
83**
 1.1000000e+07 8.0000000e+06 6.0000000e+06 4.0000000e+06 3.0000000e+06
 2.5000000e+06 2.0000000e+06 1.5000000e+06 1.0000000e+06 7.0000000e+05
 4.5000000e+05 3.0000000e+05 1.5000000e+05 1.0000000e+05 7.0000000e+04
 4.5000000e+04 3.0000000e+04 2.0000000e+04 0.0000000e+00 e
t
56$$ 0 0 a10  1 e t
56$$ 0 0 a10  2 e t
56$$ 0 0 a10  3 e t
56$$ 0 0 a10  4 e t
56$$ 0 0 a10  5 e t
56$$ f0 t
end
=opus
LIBUNIT=21
TYPARAMS=NUCLIDES
UNITS=WATTS
LIBTYPE=ALL
TIME=DAYS
NPOSITION=22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 end
end
=opus
LIBUNIT=21
TYPARAMS=NUCLIDES
UNITS=CURIES
LIBTYPE=ALL
TIME=DAYS
NPOSITION=22 23 24 25 26 27 28 29 30 31 end
end
#shell
copy ft71f001 "C:\NEW SCALE CALCS\TRIGA GAMMA SOURCE TERM.f71"
```

del ft71f001
end

```
c      Created on: Tuesday, May 25, 2010 at 16:20
c ********************** CELL CARDS ************************
   1    0      75 :-66
```

```
c
  12    8  -1.6104  -75 -33 66       $ DIRT BELOW EVEYTHING
  13    9 -0.001205 -75  45          $ AIR ABOVE EVERTHING
  14    8  -1.6104  -75 (13 :31 :-2 :-14 )-75 33 -34  $ DIRT OUTSIDE FTPRT
  15    9 -0.001205 -75 (13 :31 :-2 :-14 )-75 34 -45  $ AIR OUTSIDE FTPRT
c
  16    8  -1.6104  14 -18 6 -13 -34  33    $ DIRT WEST OF BAY
  17    9 -0.001205 14 -18 6 -13  34 -45    $ AIR WEST OF BAY
c
  18    8  -1.6104  22 -25 2 -6 -34  33     $ DIRT SOUTH OF RX WING
  19    9 -0.001205 22 -25 2 -6  34 -45     $ AIR SOUTH OF RX WING
c
  20    8  -1.6104  25 -29 2 -4 -34  33     $ DIRT SOUTH OF OFFICE WING
  21    9 -0.001205 25 -29 2 -4  34 -45     $ AIR SOUTH OF OFFICE WING
c
  22    8  -1.6104  29 -31 2 -6 -34  33     $ DIRT SOUTH OF STAIRS
  23    9 -0.001205 29 -31 2 -6  34 -45     $ AIR SOUTH OF STAIRS
c
  24    8  -1.6104  26 -31 11 -13 -34  33   $ DIRT NORTH OF OFFICE WING
  25    9 -0.001205 26 -31 11 -13  34 -45   $ AIR NORTH OF OFFICE WING2
c ABOVE BUILDING
  26    9 -0.001205 26 -29 4 -11  43 -45    $ AIR OVER OFFICE WING
  27    9 -0.001205 25 -26 4  -6  43 -45    $ Accounts for wall interface
  28    9 -0.001205 29 -31 6 -11  43 -45    $ AIR OVER OFFICE WING STAIR
```

```
  29    9 -0.001205 14 -22 2  -6  41 -45    $ AIR OVER N-GEN ROOM
c  INSIDE BUILDING
  30   10    -2.3   14 -22 2 -6 33 -41 #31   $ n-gen room shell
  31    9 -0.001205 15 -21 3 -8 33 -40      $ n-gen room volume
c
  32   10    -2.3   18 -26 6 -10 33 -45 #33  $ rx wing shell
  33    9 -0.001205 19 -25 7 -10 33 -44      $ rx wing volume
c
  34   10    -2.3   (26 -29 4 -11 33 -43) #36  $ office wing shell
  35   10    -2.3   (25 -26  4  -6 33 -43)
  36    9 -0.001205 (26 -28 5 -10 33 -42)
c
  37   10    -2.3   29 -31 6 -11 33 -43 #38  $ office wing stairwell she
  38    9 -0.001205 29 -30 7 -10 33 -42      $ office wing stairwell vol

c ********************  SURFACE CARDS ********************
```

[redacted]

```
c
  14      px -609.6   $ n-gen room walls
  15      px -487.68  $ n-gen room interior
```

[redacted]

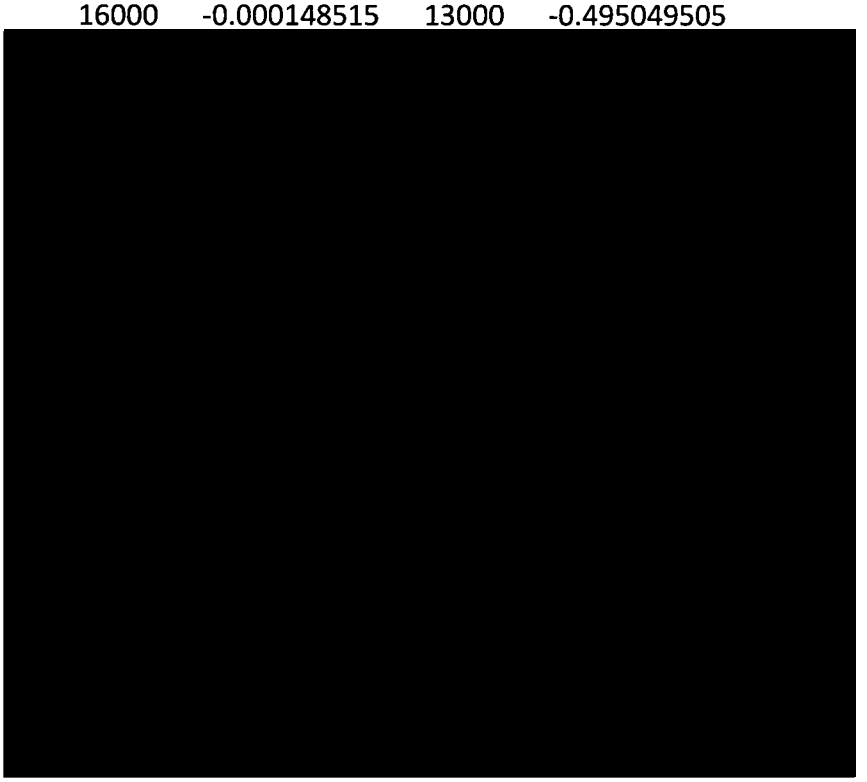
```
  27      px 1880     $ office wing wall
```

```
  28      px 4260     $ wall
  29      px 4300.92  $ lab wing wall
  30      px 4840     $ building wall
  31      px 4876.8   $ building
c
  32      pz -365     $ concrete pad/foundation 12 ft
  33      pz 0        $ bay floor  ALREADY SURFACE 12
  34      pz 366      $ ground at side of building
  35      pz 183      $ ground at parking lot
  36      pz 365      $ office wing basement ceiling
  37      pz 396      $ office wing 1st floor floor
  38      pz 762      $ office wing 1st floor ceiling
  39      pz 792      $ office wing 2nd floor floor
  40      pz 945      $ n-gen room ceiling
  41      pz 1240.526 $ n-gen room roof 633.984 cm < bay
  42      pz 1158     $ office 2nd floor ceiling
  43      pz 1183.4   $ office 2nd floor roof 411.48 cm < bay
  44      pz 1840.52  $ bay roof
  45      pz 1865.92  $ bay roof
C  POOL AND POOL WALL CORE CENTER (x,y) = (655,655)
  46   1  px -99
  47   1  px 99
  48   1  py 0
  49   1  py -99
  50   2  cz  99
  51   2 c/z -70 -70 99
  52   1  pz 853.44
  53   1  px -99.6
  54   1  px 99.6
  55   2  cz 99.6
  56   2 c/z -70 -70 99.6
c  LOWER SHIELD (pz=0 to pz=548.64 )
  57   1  px -342.9
  58   1  px  342.9
  59   1  py  316.26
  60   1  py -415.26
  61   1  pz  548.64
c  STEM  (pz=548.64 to pz=853.44 )
  62   1  px -190.5
  63   1  px  190.5
  64   1  py  190.5
  65   1  py -289.56
c
```

```
  66      pz -500
C
```

```
c
  75      s  655 655 0 21000

*tr1 655 655 0 135 315 90 45 45 90 90 90 0  $ 45 degree at (655,655)
*tr2 655 655 0
mode  p
nps 10000
C    LGP        3.378563
m1    26000     -0.346163366     24000     -0.094059406
      28000     -0.04950495      25000     -0.00990099
      14000     -0.004950495     15000     -0.000222772
      16000     -0.000148515     13000     -0.495049505
```



```
      28000     -0.061516567     25000     -0.012303313
      14000     -0.006151657     15000     -0.000276825
```

```
    16000    -0.00018455      13000    -0.372531019
c  composition of radial reflector
m6   6000.            -1  $MAT
C     material: nominal soil d=1.6104 g/cm^3; .05 bound water content
c                              .20 free  water content
m8   1000.    -0.02331  $MAT
    8000.    -0.55922 14000.   -0.22259 13000.   -0.06528
    26000.   -0.04015 20000.   -0.02915 19000.   -0.0208
    11000.   -0.02272 12000.   -0.01678 18000.   -0.0128
m9   6000.    -0.000124  $ Air 0.001205 g/cc
    7014.    -0.755268 8016.   -0.231781 18000.  -0.012827
m10  1001.    -0.0221  $ Normal concrete 2.3 g/cc
    6012.    -0.002484 8016.   -0.57493  11023.  -0.015208
    12000.   -0.001266 13027.  -0.019953 14000.  -0.304627
    19000.   -0.010045 20000.  -0.042951 26000.  -0.006435
m11  1000.    -0.003585  $ Barite concrete 2.8 g/cc (up to 3.5 g/cc)
    8000.    -0.311622 12000.  -0.001195 13000.  -0.004183
    14000.   -0.010457 16000.  -0.107858 20000.  -0.050194
    26000.   -0.047505 56000.  -0.4634
imp:p          0 1 36r         $ 1, 36
c ********************** RUN CARDS **********************
sdef  cel=9  erg=d1 axs=0 0 1 pos=655 655 98.91 rad=d2 ext=d3
SI2 0 25.4
si3 -19.05 19.05
c
c ** 3.5 MW 0.12E-04 days
si1 a
      1.000E-02  2.500E-02  3.750E-02  5.750E-02  8.500E-02
      1.250E-01  2.250E-01  3.750E-01  5.750E-01  8.500E-01
      1.250E+00  1.750E+00  2.250E+00  2.750E+00  3.500E+00
      5.000E+00  7.000E+00  9.500E+00
sp1
      1.798E+19  9.090E+18  6.621E+18  2.174E+18  2.124E+18
      1.599E+18  9.635E+17  6.540E+17  5.683E+17  5.588E+17
      2.685E+17  1.486E+17  5.819E+16  3.698E+16  1.432E+16
      4.528E+15  5.217E+14  1.020E+12
c
F5:p  25 25 200 1
f15z:p 566 1500 1  566 1800  1  566 4900  1  566 8000  1
    566 9000 1  566 10000 1  566 15000 1  566 20000 1
c
c fs15  4  10  13  15  18  22  26  52  61
c
```

```
fm5  9.749E17
fc5  Tally multiplied by 2.708E14 photons/s  per kW
     times 3600 s/h to yield Sv/h per kilowatt
fm15  9.749E17
fc15  Tally multiplied by 2.708E14 photons/s  per kW
     times 3600 s/h to yield Sv/h per kilowatt
c    ---------------------------------------------
c    Ambient dose conversion (Sv cm^2) - ICRP 51, 1987
de0  0.01 0.015 0.02 0.03 0.04 0.05 0.06 0.08 0.10
     0.15 0.20  0.30 0.40 0.50 0.60 0.80 1 1.5
     2 3 4 5 6 8 10
df0  0.0769E-12 0.846E-12 1.01E-12  0.785E-12  0.614E-12
     0.526e-12  0.504E-12 0.532E-12 0.611E-12  0.890E-12
     1.18e-12   1.81E-12  2.38E-12  2.89E-12   3.38E-12
     4.29e-12   5.11E-12  6.92E-12  8.48E-12   11.1E-12
     13.3e-12   15.4E-12  17.4E-12  21.2E-12   25.2E-12
c    ---------------------------------------------
Phys:p  10 1 1
prdmp  3j 3 1j
print
```

15.0 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate a Nuclear Research Reactor

The University of Texas is a State owned entity, as documented in Appendix 15.1. UT has operated a TRIGA nuclear research reactor since 1967. In 1998, UT decided to decommission a 250 kW TRIGA located on the main campus and construct a new 1.1 MW TRIGA on the Pickle Research Campus. The PRC facility has operated successfully, continuously since granted a facility operating license in 1991. Recent facility budgeting and expenditures was used to develop an estimate of operating costs and income for the next five years (Appendix 15.2).

15.2 Financial Ability to Decommission the Facility

The University of Texas intends to renew the facility operating license. Whenever a decision is made to terminate operations and decommission the facility, the university will seek legislative appropriations of funds from the State of Texas, as indicated in indicated in Appendix 15.2.

15.3 Bibliography

NUREG/CR-1756 "Technology, Safety, and Costs of Decommissioning Reference Nuclear Research and Test Reactors," U.S. Nuclear Regulatory Commission, March 1982; Addendum, July 1983.

EXCERPTS FROM THE TEXAS EDUCATION CODE FOR THE GOVERNMENT OF THE UNIVERSITY OF TEXAS SYSTEM AND RULES 10501 AND 20201 FROM THE RULES AND REGULATIONS OF THE BOARD OF REGENTS OF THE UNIVERSITY OF TEXAS SYSTEM FOR THE GOVERNMENT OF THE UNIVERSITY OF TEXAS SYSTEM

EDUCATION CODE

TITLE 3. HIGHER EDUCATION

SUBTITLE C. THE UNIVERSITY OF TEXAS SYSTEM

CHAPTER 67. THE UNIVERSITY OF TEXAS AT AUSTIN

SUBCHAPTER A. GENERAL PROVISIONS

Sec. 67.01. DEFINITIONS. In this chapter:

(1) "University" means the University of Texas at Austin.

(2) "Board" means the board of regents of The University of Texas System.

Acts 1971, 62nd Leg., p. 3159, ch. 1024, art. 1, Sec. 1, eff. Sept. 1, 1971.

Sec. 67.02. THE UNIVERSITY OF TEXAS AT AUSTIN. The University of Texas at Austin is a coeducational institution of higher education within The University of Texas System. It is under the management and control of the board of regents of The University of Texas System.

Acts 1971, 62nd Leg., p. 3160, ch. 1024, art. 1, Sec. 1, eff. Sept. 1, 1971.

EDUCATION CODE

TITLE 3. HIGHER EDUCATION

SUBTITLE C. THE UNIVERSITY OF TEXAS SYSTEM

CHAPTER 65. ADMINISTRATION OF THE UNIVERSITY OF TEXAS SYSTEM

SUBCHAPTER A. GENERAL PROVISIONS

Sec. 65.02. ORGANIZATION. (a) The University of Texas System is composed of the following institutions and entities:

(1) The University of Texas at Arlington, including:

(A) The University of Texas Institute of Urban Studies at Arlington; and

(B) The University of Texas School of Nursing at Arlington;

(2) The University of Texas at Austin, including:

(A) The University of Texas Marine Science Institute;

(B) The University of Texas McDonald Observatory at Mount Locke; and

(C) The University of Texas School of Nursing at Austin;

(3) The University of Texas at Dallas;

(4) The University of Texas at El Paso, including The University of Texas School of Nursing at El Paso;

(5) The University of Texas of the Permian Basin;

(6) The University of Texas at San Antonio, including the University of Texas Institute of Texan Cultures at San Antonio;

(7) The University of Texas Southwestern Medical Center at Dallas, including:

(A) The University of Texas Southwestern Medical School at Dallas;

(B) The University of Texas Southwestern Graduate School of Biomedical Sciences at Dallas; and

(C) The University of Texas Southwestern Allied Health Sciences School at Dallas;

(8) The University of Texas Medical Branch at Galveston, including:

(A) The University of Texas Medical School at Galveston;

(B) The University of Texas Graduate School of Biomedical Sciences at Galveston;

(C) The University of Texas School of Allied Health Sciences at Galveston;

(D) The University of Texas Marine Biomedical Institute at Galveston;

(E) The University of Texas Hospitals at Galveston; and

(F) The University of Texas School of Nursing at Galveston;

(9) The University of Texas Health Science Center at Houston, including:

(A) The University of Texas Medical School at Houston;

(B) The University of Texas Dental Branch at Houston;

(C) The University of Texas Graduate School of Biomedical Sciences at Houston;

(D) The University of Texas School of Health Information Sciences at Houston;

(E) The University of Texas School of Public Health at Houston;

(F) The University of Texas Speech and Hearing Institute at Houston; and

(G) The University of Texas School of Nursing at Houston;

(10) The University of Texas Health Science Center at San Antonio, including:

(A) The University of Texas Medical School at San Antonio;

(B) The University of Texas Dental School at San Antonio;

(C) The University of Texas Graduate School of Biomedical Sciences at San Antonio;

(D) The University of Texas School of Allied Health Sciences at San Antonio; and

(E) The University of Texas School of Nursing at San Antonio;

(11) The University of Texas M. D. Anderson Cancer Center, including:

(A) The University of Texas M. D. Anderson Hospital;

(B) The University of Texas M. D. Anderson Tumor Institute; and

(C) The University of Texas M. D. Anderson Science Park; and

(12) The University of Texas Health Science Center--South Texas, including The University of Texas Medical School--South Texas, if established under Subchapter N, Chapter 74.

(b) The University of Texas System shall also be composed of such other institutions and entities as from time to time may be assigned by specific legislative act to the governance, control, jurisdiction, or management of The University of Texas System.

Added by Acts 1973, 63rd Leg., p. 1186, ch. 435, Sec. 1, eff. Aug. 27, 1973.
Amended by Acts 1989, 71st Leg., ch. 644, Sec. 2, eff. June 14, 1989; Acts 2001, 77th Leg., ch. 325, Sec. 1, eff. Sept. 1, 2001.
Amended by:

Acts 2009, 81st Leg., R.S., Ch. 1341, Sec. 5, eff. June 19, 2009.

SUBCHAPTER B. ADMINISTRATIVE PROVISIONS

Sec. 65.11. BOARD OF REGENTS. The government of the university system is vested in a board of nine regents appointed by the governor with the advice and consent of the senate. The board may provide for the administration, organization, and names of the institutions and entities in The University of Texas System in such a way as will achieve the maximum operating efficiency of such institutions and entities, provided, however, that no institution or entity of The University of Texas System not authorized by specific legislative act to offer a four-year undergraduate program as of the effective date of this Act shall offer any such four-year undergraduate program without prior recommendation and approval by a two-thirds vote of the Texas Higher Education Coordinating Board and a specific act of the Legislature.

Acts 1971, 62nd Leg., p. 3144, ch. 1024, art. 1, Sec. 1, eff. Sept. 1, 1971. Amended by Acts 1973, 63rd Leg., p. 1188, ch. 435, Sec. 2, eff. Aug. 27, 1973; Acts 1989, 71st Leg., ch. 644, Sec. 3, eff. June 14, 1989.

SUBCHAPTER C. POWERS AND DUTIES OF BOARD

Sec. 65.31. GENERAL POWERS AND DUTIES. (a) The board is authorized and directed to govern, operate, support, and maintain each of the component institutions that are now or may hereafter be included in a part of The University of Texas System.

(b) The board is authorized to prescribe for each of the component institutions courses and programs leading to such degrees as are customarily offered in outstanding American universities, and to award all such degrees. It is the intent of the legislature that such degrees shall include baccalaureate, master's, and doctoral degrees, and their equivalents, but no new department, school, or degree-program shall be instituted without the prior approval of the Coordinating Board, Texas College and University System.

(c) The board has authority to promulgate and enforce such other rules and regulations for the operation, control, and management of the university system and the component institutions thereof as the board may deem either necessary or desirable. The board is specifically authorized and empowered to determine and prescribe the number of students that shall be admitted to any course, department, school, college, degree-program, or institution under its governance.

(d) The board is specifically authorized to make joint appointments in the component institutions under its governance. The salary of any person who receives such joint appointment shall be apportioned to the appointing institutions on the basis of services rendered.

(e) The board is specifically authorized, upon terms and conditions acceptable to it, to accept, retain in depositories of its choosing, and administer gifts, grants, or donations of any kind, from any source, for use by the system or any of the component institutions of the system.

(f) No component institution which is not authorized to offer a four-year undergraduate program shall offer a four-year undergraduate program without the specific authorization of the legislature.

(g) The board by rule may delegate a power or duty of the board to a committee, officer, employee, or other agent of the board.

Acts 1971, 62nd Leg., p. 3145, ch. 1024, art. 1, Sec. 1, eff. Sept. 1, 1971. Amended by Acts 1971, 62nd Leg., p. 3360, ch. 1024, art. 2, Sec. 37, eff. Sept. 1, 1971; Acts 1983, 68th Leg., p. 5010, ch. 900, Sec. 1, eff. Aug. 29, 1983; Acts 1995, 74th Leg., ch. 213, Sec. 1, eff. May 23, 1995.

. . . .

Rule 10501 Delegation to Act on Behalf of the Board (last amended 2/5/10)

1. Title

Delegation to Act on Behalf of the Board

2. Rule and Regulation

Sec. 1 Identification of Significant Contracts or Documents. Institutional presidents and executive officers at U. T. System Administration are responsible for identifying contracts, agreements, and other documents that are of such significance to require the prior approval of the Board of Regents. Each such matter so identified

shall be presented to the Board by the Chancellor as an agenda or docket item at a meeting of the Board.

Sec. 2 Compliance with Special Instructions. All authority to execute and deliver contracts, agreements, and other documents is subject to these *Rules and Regulations* and compliance with all applicable laws and special instructions or guidelines issued by the Chancellor, an Executive Vice Chancellor, and/or the Vice Chancellor and General Counsel. Special instructions or guidelines by the Chancellor, an Executive Vice Chancellor, or the Vice Chancellor and General Counsel may include without limitation instructions concerning reporting requirements; standard clauses or provisions; ratification or prior approval by the Board of Regents or the appropriate Executive Vice Chancellor; review and approval by the Office of General Counsel; and recordkeeping.

Sec. 3 Contracts or Agreements Requiring Board Approval. The following contracts or agreements, including purchase orders or vouchers and binding letters of intent or memorandums of understanding, must be submitted to the Board for approval or authorization.

3.1 Contracts Exceeding $1 Million. All contracts or agreements, with a total cost or monetary value to the U. T. System or any of the institutions of more than $1 million, unless exempted in Section 4 below. The total cost or monetary value of the contract includes all potential contract extensions or renewals whether automatic or by operation of additional documentation. For purposes of this Rule, all contracts with unspecified amounts of payments with a term of greater than four years are presumed to have a total value of greater than $1 million.

3.2 Contracts with Foreign Governments. Contracts or agreements of any kind or nature, regardless of dollar amount, with a foreign government or agencies thereof, except affiliation agreements and cooperative program agreements, material transfer agreements, sponsored research agreements and licenses, or other conveyances of intellectual property owned or controlled by the Board of Regents prepared on an approved standard form or satisfying the requirements set by the Office of the General Counsel, or agreements or contracts necessary to protect the exchange of confidential information or nonbinding letters of intent or memorandums of understanding executed in advance of definitive agreements each as reviewed and approved by the Vice Chancellor and General Counsel.

3.3. Contracts Involving Certain Uses of Institution Names, Trademarks, or Logos. Except as specifically allowed under

existing contracts entered into between the Board of Regents and nonprofit entities supporting a U. T. System institution, agreements regardless of dollar amount that grant the right to a non-U. T. entity to use the institutional name or related trademarks or logos in association with the provision of a material service or in association with physical improvements located on property not owned or leased by the contracting U. T. System institution.

3.4 Contracts with Certain Officers. Agreements, regardless of dollar amount, with the Chancellor, a president, a former Chancellor or president, an Executive Vice Chancellor, a Vice Chancellor, the General Counsel to the Board, or the Chief Audit Executive are subject to the applicable provisions of *Texas Education Code* Section 51.948.

3.5 Insurance Settlements.

(a) Settlements in excess of $1 million must have the approval of the Board.

(b) Settlement claims from insurance on money and securities or fidelity bonds of up to $1 million shall be approved by the Executive Vice Chancellor for Business Affairs.

(c) If a loss is so extensive that partial payments in excess of $1 million are necessary, the Chancellor is delegated authority to execute all documents related to the partial payment or adjustment. Final settlement of claims in excess of $1 million will require approval by the Board.

3.6 Settlement of Disputes. Settlements of any claim, dispute or litigation for an amount greater than $1 million require approval. The settlement may also be approved by the appropriate standing committee of the Board of Regents. The Vice Chancellor and General Counsel shall consult with the institution's president and appropriate Executive Vice Chancellor, or Vice Chancellor with regard to all settlements in excess of $150,000 that will be paid out of institutional funds.

Sec. 4 Contracts Not Requiring Board Approval. The following contracts or agreements, including purchase orders and vouchers, do not require prior approval by the Board of Regents regardless of the contract amount.

4.1 Construction Projects. Contracts, agreements, and documents relating to construction projects previously

approved by the Board of Regents in the Capital Improvement Program and Capital Budget or Minor Projects.

4.2 Construction Settlements. All settlement claims and disputes relating to construction projects to the extent funding for the project has been authorized.

4.3 Intellectual Property. Legal documents, contracts, or grant proposals for sponsored research, including institutional support grants, and licenses or other conveyances of intellectual property owned or controlled by the Board of Regents as outlined in Rule 90105 of these Rules.

4.4 Replacements. Contracts or agreements for the purchase of replacement equipment or licensing of replacement software or services associated with the implementation of the software.

4.5 Routine Supplies. Contracts or agreements for the purchase of routinely purchased supplies.

4.6 Group Purchases. Purchases made under a group purchasing program that follow all applicable statutory and regulatory standards for procurement.

4.7 Approved Budget Items. Purchases of new equipment or licensing of new software or services associated with the implementation of the software, identified specifically in the institutional budget approved by the Board of Regents.

4.8 Loans. Loans of institutional funds to certified nonprofit health corporations, which loans have been approved as provided in The University of Texas System Administration Policy UTS166, *Cash Management and Cash Handling Policy* and The University of Texas System Administration Policy UTS167, *Banking Services Policy* concerning deposits and loans.

4.9 Certain Employment Agreements. Agreements with administrators employed by the U. T. System or any of the institutions, so long as such agreements fully comply with the requirements of *Texas Education Code* Section 51.948 including the requirement to make a finding that the agreement is in the best interest of the U. T. System or any of the institutions.

4.10 Energy Resources. Contracts or agreements for utility services or energy resources and related services, if any,

which contracts or agreements have been approved in advance by the Chancellor or the Chancellor's delegate.

4.11 Library Materials. Contracts or agreements for the purchase or license of library books and library materials.

4.12 Athletic Employment Agreements. Contracts with athletic coaches and athletic directors except those with total annual compensation of $250,000 or greater, as covered by Rule 20204.

4.13 Bowl Games. Contracts or agreements related to postseason bowl games, subject to a requirement that the contract or agreement has been submitted to the Executive Vice Chancellor for Academic Affairs and is in a form acceptable to the Vice Chancellor and General Counsel.

4.14 Property or Casualty Losses. Contracts or agreements with a cost or monetary value to the U. T. System or any of the institutions in excess of $1 million but not exceeding $10 million associated with or related to a property or casualty loss that is expected to exceed $1 million may be approved, executed, and delivered by the Chancellor. The Chancellor shall consult with the institutional president, if applicable.

4.15 Health Operations. Contracts or agreements for the procurement of routine services or the purchase or lease of routine medical equipment, required for the operation or support of a hospital or medical clinic, if the services or equipment were competitively procured.

4.16 Increase in Board Approval Threshold. An institution's dollar threshold specified in Section 3.1 may be increased to up to $5 million by the Vice Chancellor and General Counsel, after consultation with the General Counsel to the Board of Regents, if it is determined that the institution has the expertise to negotiate, review, and administer such contracts. Unless approved in advance by the Vice Chancellor and General Counsel, any increase will not apply to contracts or agreements designated as Special Procedure Contracts by the Vice Chancellor and General Counsel.

4.17 Group Employee Benefits. Contracts or agreements for uniform group employee benefits offered pursuant to Chapter 1601, *Texas Insurance Code*.

Sec. 5 Signature Authority. The Board of Regents delegates to the Chancellor or the president of an institution authority to execute and

deliver on behalf of the Board contracts and agreements of any kind or nature, including without limitation licenses issued to the Board or an institution. In addition to other primary delegates the Board assigns in the Regents' *Rules and Regulations*, the Board assigns the primary delegate for signature authority for the following types of contracts.

5.1 System Administration and Systemwide Contracts. The Board of Regents delegates to the Executive Vice Chancellor for Business Affairs authority to execute and deliver on behalf of the Board contracts or agreements:

(a) affecting only System Administration,

(b) binding two or more institutions of the U. T. System with the concurrence of the institutions bound, or

(c) having the potential to benefit more than one institution of the U. T. System so long as participation is initiated voluntarily by the institution.

5.2 Contracts Between or Among System Administration and Institutions. The Board of Regents delegates to the Executive Vice Chancellor for Business Affairs authority to execute on behalf of the Board contracts or agreements between or among System Administration and institutions of the U. T. System for resources or services. Any such contract or agreement shall provide for the recovery of the cost of services and resources furnished.

5.3 Contracts with System Administration or Between or Among Institutions. The Board of Regents delegates to the president of an institution authority to execute on behalf of the Board contracts or agreements with System Administration or between or among institutions of the U. T. System for resources or services. Any such contract or agreement shall provide for the recovery of the cost of services and resources furnished.

5.4 Contracts for Legal Services and Filing of Litigation. The Board of Regents delegates to the Vice Chancellor and General Counsel authority to execute and deliver on behalf of the Board contracts for legal services and such other services as may be necessary or desirable in connection with the settlement or litigation of a dispute or claim after obtaining approvals as may be required by law. Litigation to be instituted under these contracts on behalf of the Board, System Administration, or an institution of U. T. System must

have the prior approval of the Vice Chancellor and General Counsel.

5.5 Settlements of Disputes. Except as provided in Section 5.6 below, the Board of Regents delegates to the Vice Chancellor and General Counsel authority to execute and deliver on behalf of the Board agreements settling any claim, dispute, or litigation. The Vice Chancellor and General Counsel shall consult with the institutional president and the appropriate Executive Vice Chancellor or Chancellor with regard to all settlements greater than $150,000 that will be paid out of institutional funds. Settlements greater than $1,000,000 will require the approval of the Board as outlined in Section 3.5 above. The Vice Chancellor and General Counsel shall consult with the Office of External Relations with respect to settlement of will contests and other matters relating to gifts and bequests administered by that Office.

5.6 Construction Settlements. The Board of Regents delegates authority to execute all documents necessary or desirable to settle claims and disputes relating to construction projects to the System Administration or institution official designated in the construction contract to the extent funding for the project has been authorized.

5.7 Assurance of Authority to Act. The officer or employee executing any document on behalf of the Board of Regents shall be responsible for assuring that he or she has authority to act on behalf of the Board and that such authority is exercised in compliance with applicable conditions and restrictions. Documents executed on behalf of the Board pursuant to authority granted under these *Rules and Regulations* shall not require further certification or attestation.

5.8 Institutional Agreements for Dual Credit. The Board of Regents delegates the authority to approve and execute dual credit partnership agreements for the academic institutions to the Executive Vice Chancellor for Academic Affairs.

Sec. 6 Delegation Process. The primary delegate identified in these *Rules and Regulations* or in an official Board action may further delegate his or her delegated authority to a secondary delegate unless otherwise specified. Any such further delegation of authority must be made in writing and the primary delegate shall permanently maintain, or cause to be maintained, evidence of all such delegations. A secondary delegate of the primary delegate may not further delegate such authority.

6.1 Delegate's Responsibilities. The primary delegate identified in these *Rules and Regulations* as authorized to execute and deliver on behalf of the Board of Regents various types of contracts, agreements, and documents shall maintain, or cause to be maintained, necessary and proper records with regard to all contracts, agreements, and documents executed and delivered pursuant to such delegated authority, in accordance with any applicable records retention schedule or policy adopted by the Board, the U. T. System Administration, or the institution.

Sec. 7 Actions of the Board as Trustee. Authority delegated by the Board of Regents in these *Rules and Regulations* includes actions that may be taken by the Board in its capacity as trustee of any trust to the extent such delegation is permitted by law.

Sec. 8 Power to Authorize Expenditures. No expenditure out of funds under control of the Board shall be made and no debt or obligation shall be incurred and no promise shall be made in the name of the System or any of the institutions or of the Board of Regents by any member of the respective staffs of the U. T. System or any of the institutions except:

8.1 In accordance with general or special budgetary apportionments authorized in advance by the Board of Regents and entered in its minutes; or

8.2 In accordance with authority specifically vested by the Board of Regents in a committee of the Board; or

8.3 In accordance with authority to act for the Board of Regents when it is not in session, specifically vested by these *Rules and Regulations* or by special action of the Board.

Sec. 9 Power to Establish Policies. No employee of the U. T. System or any of the institutions, as an individual or as a member of any association or agency, has the power to bind the System or any of the institutions unless such power has been officially conferred in advance by the Board of Regents. Any action which attempts to change the policies or otherwise bind the System or any of the institutions, taken by any individual or any association or agency, shall be of no effect whatsoever until the proposed action has been approved by the president of an institution concerned, if any, the appropriate Executive Vice Chancellor, and the Chancellor, and ratified by the Board.

Sec. 10 Exceptions. This Rule does not apply to any of the following:

10.1 UTIMCO. Management of assets by UTIMCO, which is governed by contract and the provisions of Rule 70101, 70201, 70202, and 70401 of these *Rules and Regulations*.

10.2 Acceptance of Gifts. The acceptance, processing, or administration of gifts and bequests, which actions are governed by Rule 60101, 60103, 70101, and 70301 of these *Rules and Regulations* and applicable policies of the Board of Regents.

10.3 Statutory. Any power, duty, or responsibility that the Board has no legal authority to delegate, including any action that the Texas Constitution requires be taken by the Board of Regents.

3. Definitions

Settlement - the amount of the settlement shall mean the amount that might be reasonably expected to be recoverable by the U. T. System or any of the institutions but not received pursuant to the settlement or, in the case of a claim against the U. T. System, the total settlement amount to be paid by the U. T. System.

Group Purchasing Program – for purposes of this Rule, a purchasing program established by (1) a state agency that is authorized by law to procure goods and services for other state agencies, such as the Texas Procurement and Support Services Division of the Texas Comptroller of Public Accounts and the Texas Department of Information Resources, or any successor agencies, respectively; or (2) a group purchasing organization in which the institution participates, such as Novation, Premier, Western States Contracting Alliance, and U.S. Communities Government Purchasing Alliance.

4. Relevant Federal and State Statutes

Texas Education Code Section 51.928(b) – Written Contracts or Agreements Between Certain Institutions

Texas Education Code Section 51.948– Restrictions on Contracts with Administrators

Texas Education Code Section 65.31(g) – Delegation by the Board

Texas Government Code Section 618.001 – Uniform Facsimile Signature of Public Officials Act

Texas Government Code Sections 669.001 - 669.004 – Restrictions on Certain Actions Involving Executive Head of State Agency

Texas Insurance Code, Chapter 1601 – Uniform Insurance Benefits Act for Employees of The University of Texas System and The Texas A&M University System

5. Relevant System Policies, Procedures, and Forms

The University of Texas System Administration Policy UTS166, *Cash Management and Cash Handling Policy*

The University of Texas System Administration Policy UTS167, *Banking Services Policy*

Regents' *Rules and Regulations*, Rule 20204 – Determining and Documenting the Reasonableness of Compensation

Regents' *Rules and Regulations*, Rule 60101 – Acceptance and Administration of Gifts

Regents' *Rules and Regulations*, Rule 60103 – Guidelines for Acceptance of Gifts of Real Property

Regents' *Rules and Regulations*, Rule 70101 – Authority to Accept and Manage Assets

Regents' *Rules and Regulations*, Rule 70201 – Investment Policies

Regents' *Rules and Regulations*, Rule 70202 – Interest Rate Swap Policy

Regents' *Rules and Regulations*, Rule 70401 – Oversight Responsibilities for UTIMCO

Litigation Approval Request Form

Special Procedure Contracts

6. Who Should Know

Administrators

7. System Administration Office(s) Responsible for Rule

Office of the Board of Regents

8. Dates Approved or Amended

February 5, 2010
November 12, 2009
August 20, 2009

Editorial amendment to add Subsection 4.17 (Group Employee Benefits) back into the Rules made August 6, 2009
Editorial amendment to Number 4 made January 5, 2009
November 13, 2008
May 15, 2008
Editorial amendment to Sec. 3.3 made March 17, 2008
Editorial amendment to Number 3 made January 28, 2008
May 10, 2007
February 8, 2007
May 12, 2005
December 10, 2004

9. Contact Information

Questions or comments regarding this rule should be directed to:

- bor@utsystem.edu

Rule 20201 Presidents (last amended 8/23/07)

1. Title

Presidents

2. Rule and Regulation

Sec. 4 Duties and Responsibilities. Within the policies and regulations of the Board of Regents and under the supervision and direction of the appropriate Executive Vice Chancellor, the president has general authority and responsibility for the administration of that institution. Specifically, the president is expected, with the appropriate participation of the staff, to:

4.1 Develop and administer plans and policies for the program, organization, and operation of the institution.

4.2 Interpret the System policy to the staff, and interpret the institution's programs and needs to the System Administration and to the public.

4.3 Develop and administer policies relating to students, and where applicable, to the proper management of services to patients.

4.4 Recommend appropriate operating budgets and supervise expenditures under approved budgets.

4.5 Appoint all members of the faculty and staff, except as provided in Rule 31007, concerning the award of tenure, and maintain efficient personnel programs.

4.6 Ensure efficient management of business affairs and physical property; and recommend additions and alterations to the physical plant.

4.7 Serve as presiding officer at official meetings of faculty and staff of the institution, and as ex officio member of each college or school faculty (if any) within the institution.

4.8 Appoint, or establish procedures for the appointment of, all faculty, staff, and student committees.

4.9 Cause to be prepared and submitted to the appropriate Executive Vice Chancellor and the Vice Chancellor and General Counsel for approval, the rules and regulations for the governance of the institution and any related amendments. Such rules and regulations shall constitute the *Handbook of Operating Procedures* for that institution. Any rule or regulation in the institutional *Handbook of Operating Procedures* that is in conflict with any rule or regulation in the Regents' *Rules and Regulations* is null and void and has no effect.

(a) Input from the faculty, staff, and student governance bodies for the institution will be sought for all significant changes to an institution's *Handbook of Operating Procedures.* The institutional *Handbook of Operating Procedures* will include a policy for obtaining this input that is in accordance with a model policy developed by the Office of General Counsel.

(b) Sections of the *Handbook of Operating Procedures* that pertain to the areas of faculty responsibility as defined in Regents' *Rules and Regulations*, Rule 40101 titled Faculty Role in Educational Policy Formulation will be explicitly designated in the *Handbook of Operating Procedures.* The president, with the faculty governance body of the campus, shall develop procedures to assure formal review by the faculty governance body before such sections are submitted for approval. The formal review should be done within a reasonable timeframe (60 days or less).

4.10 Assume initiative in developing long-range plans for the program and physical facilities of the institution.

4.11 Assume active leadership in developing private fund support for the institution in accordance with policies and procedures established in the Regents' *Rules and Regulations*.

4.12 Develop and implement plans and policies to ensure that the institution remains in compliance with any accreditation requirements appropriate to the institution or its programs, including, for the health institutions and those academic institutions with student health services, the accreditation of hospitals, clinics, and patient-care facilities.

4.13 The president of each general academic institution of The University of Texas System that engages in intercollegiate athletic activities shall ensure that necessary rules and regulations are made so as to comply with the current *General Appropriations Act*.

3. Definitions

None

4. Relevant Federal and State Statutes

Current *General Appropriations Act*

5. Relevant System Policies, Procedures, and Forms

Model Policy – Handbook of Operating Procedures (HOP) Amendment Approval Process

6. Who Should Know

Administrators
Faculty
Staff
Students

7. System Administration Office(s) Responsible for Rule

Office of Academic Affairs
Office of Health Affairs

8. Dates Approved or Amended

August 23, 2007
August 10, 2006
May 11, 2006

March 10, 2005
December 10, 2004

9. Contact Information

Questions or comments regarding this rule should be directed to:

- bor@utsystem.edu

FIVE-YEAR OPERATING COST ESTIMATE

Initial year expenses in relevant categories are summarized from monthly expense records. Projected expenses are based on an average 3% rate of inflation.

NOTE[1]: Return on UT investment portfolio, consequently fluctuates

OFFICE OF THE VICE PRESIDENT AND CHIEF FINANCIAL OFFICER

THE UNIVERSITY OF TEXAS AT AUSTIN

P.O. Box 8179 • Austin, Texas 78713-8179
(512)471-1422 • (512)471-7742

December 1, 2011

Mr. A. Jason Lising
Project Manager
Division of Policy and Rule Making
Research and Test Reactor Licensing Branch
Washington, DC

RE: License R-129
Docket 50-602

Dear Mr. Lising:

This concerns the ultimate decommissioning of the University of Texas TRIGA II nuclear research reactor, currently licensed for operation by the University until January 17, 2012. Pursuant to the Code of Federal regulations, title 10, Part 50, this is to assure that the University an entity of the State of Texas will obtain funds for decommissioning when it is necessary.

It is our intention to propose renewal of the current facility operating license. Nevertheless, whenever a decision to decommission the facility is made, the University will request legislative appropriation of funds sufficiently in advance of decommissioning to prevent delay of required activities.

As Chief Financial Officer for the University, I have the authority to sign this statement of intent.

Sincerely,

Kevin P. Hegarty
Vice President and Chief Financial Officer

c: Dr. Juan M. Sanchez, UT Austin, VP for Research
Mr. Paul Michael Whaley, UT Austin, NETL

DECOMMISSIONING COST ESTIMATE

NUREG/CR-1576 analyzes data from decommissioning of a 0.1 MW university reactor (OSU/AGN-201), a 0.01 university facility (NCSUR-3), a 0.2 MW (1 MW forced flow) commercial facility (B&W, LPR), a 250 kW Army facility (DORF), and a 5 MW heavy water moderated DOE facility (ALRR).

Table 15.x, Summary of NUREG/CR-1576 Values

FACILITY	POWER	OWNER	BASE YEAR	COST ($1000)
OSU/AGN-2011[1]	0.1 W	Oregon State University	1980	10
NCSUR	10 kW	NC State University	--	33/part
LPR	200 kW/1 MW	Babcock & Wilcox	1982	86
DORF	250 kW	A.S. Army	1980	336
ALRR	5 MW	Department of Energy	1981	4,292

The ALRR was a more complex installation than the UT TRIGA, and would not be expected to have the comparable labor demands in decommissioning. The cost for decommissioning the UT reactor is therefore expected to be biased more towards the LPR and DORF; DORF decommissioning costs are therefore used for comparison of total costs, distributed according to NUREG recommended disposal cost estimation:

$$C_{1981,adjusted} = (X)\cdot\{(L)\cdot(L_a)+(R)\cdot(R_a)+(O)\cdot(O_a)\}$$

Where:

$C_{1981,adjusted}$ is the current value based on the 19981 values
L is the labor cost as a fraction of total decommissioning costs
L_a is the adjustment of labor costs from 1981 values
R is the radwaste burial costs as a fraction of the total decommissioning costs
R_a is the adjustment to account for changes between 1981 and the current year
O is the factor of all other coasts as a fraction of the total decommissioning costs
O_a is the adjustment to account for changes between 1981 and the current year

The average cost of labor is 44.72% of the total cost. There are two outliers in the data, 64% for a very low power reactor (where the remainder of the costs were disproportionally low), and a university reactor that minimized costs with student labor. With these outliers removed, the average value is 43.9% with a deviation of 1.9% from the aggregate average indicating the average value may be representative of the 1.1 MW UT TRIGA.

The average of the unspecified ("other") costs is 50.7% of the total cost. The influence of the outliers adds some bias but the average excluding the outliers is 52.0% (a deviation of about 2% from the aggregate), indicating the average value may be representative of the 1.1 MW UT TRIGA.

The cost of waste disposal ranges from 1% to 9.4%, probably because of the large variation in the volume of waste in the cases examined. The volume of waste ranges from 1157 m^3 for the largest facility to a negligible quantity for the smallest. The average fraction for waste disposal across all cases is 4.6%, with 4.2% excluding the outliers. The two highest power levels have fractions significantly different, 3.9% for the 5 MW kW facility and 1.6% for the 250 kW facility, suggesting the average may not be as representative of the 1.1 MW UT TRIGA; the 4.2% value is used.

The three individual fractions are normalized to get a valid distribution, so the fractions are (L) 44.8%, (O) 50.9% and (R) 4.2% for labor, non-specified and rad-waste disposal costs respectively.

The Consumer Price Index calculator (http://www.bls.gov/data/inflation_calculator.htm) indicates that the current value for the original $336,000 decommissioning cost is $836,936. Assuming an annual rate of 3% inflation, the decommissioning cost at the end of the new 20 year license will be $888,609.

STANDARD RESEARCH **SUBCONTRACT NO. 00078206** **"REACTOR FUEL ASSISTANCE AND FUEL ELEMENTS"**	Battelle Energy Alliance, LLC (BEA) 2525 Fremont Avenue P. O. Box 1625 Idaho Falls, ID 83415-3890
Subcontractor: The University of Texas at Austin P. O. Box 7726 Austin, TX 78713-7726 To: Susan Wyatt Sedwick PI: Sean O'Kelly **Period of Performance:** August 1, 2008 – August 31, 2013 **Award Amount:** $0.0	**Contractor's Procurement Representative** Lynda Keller Subcontract Administrator 208-526-5597 208-526-5780 Lynda.Keller@inl.gov

Introduction

This is a standard research subcontract for unclassified research and development work, not related to nuclear, chemical, biological, or radiological weapons of mass destruction or the production of special nuclear material for use in weapons of mass destruction. This Subcontract is between Battelle Energy Alliance, LLC (BEA) (Contractor) and University of Texas at Austin (Subcontractor). The Subcontract is issued under Prime Contract No. DE-AC07-05ID14517 between the Contractor and the United States Department of Energy (DOE) for the management and operation of the Idaho National Laboratory (INL).

Agreement

The parties agree to perform their respective obligations in accordance with the terms and conditions of the Schedule, General Provisions and other documents attached or incorporated by reference, which together constitute the entire Subcontract and supersedes all prior discussions, negotiations, representations, and agreements.

BATTELLE ENERGY ALLIANCE, LLC (BEA)

By: Lynda Keller
Name: Lynda Keller
Title: Subcontract Administrator
Date: 12/8/08

UNIVERSITY OF TEXAS AT AUSTIN

By: Jeanette Holmes
Name: Jeanette Holmes
Title: Associate Director
Date: Office of Sponsored Projects

DEC 1 1 2008

SCHEDULE OF ARTICLES

1. Statement of Work

The Subcontractor shall furnish the following services, in accordance with the requirements, terms and conditions specified or referenced in this Subcontract:

Provide for utilization of the reactor owned by the Subcontractor in a program of education and training of students in nuclear science and engineering, and for faculty and student research. The Subcontract provides for the continued possession and use of Department of Energy (DOE)-owned nuclear materials, including enriched uranium, in reactor fuel without incremental charge of use, burn-up, and reprocessing while used for research, education and training purposes.

The DOE-owned nuclear materials were originally provided to Subcontractor under Subcontract No. C85-110742-002. The nuclear materials will now reside with this Subcontract No. 00078206.

The Subcontractor's Principal Investigator assigned to this work is Sean O'Kelly. The Principal Investigator shall not be replaced or reassigned without the advance written approval of the Contractor's Subcontract Administrator.

2. Reports and Data Requirements

a. Progress Reports

1. Distribution of the DOE/NRC Form 741, Nuclear Material Transaction Report, shall include JSG/MM. Copies of DOE/NRC Forms 742, Material Balance Report, and 742C, Physical Inventory Listing, shall be sent to the Contractor point-of-contact for nuclear material management and accountability.
2. Annually, in conjunction with submittal of the Material Balance Report and Physical Inventory Listing reports, the Subcontractor is required to submit information listed below so that the Contractor can meet DOE requirements for annual reporting contained in DOE Order 5660.1B, Management of Nuclear Materials. The Subcontractor is required to notify the Contractor of the following:

2.1. Fuel usage in grams Uranium 235 and number of fuel elements.

2.2. Current inventory of unirradiated fuel elements in storage.

2.3. Current inventory of fuel elements in core.

2.4. Current inventory of useable irradiated fuel elements outside of core.

2.5. Current inventory of spent fuel elements awaiting shipment.

2.6. Projected fuel needs for the next five years.

2.7. Current inventory of all other nuclear material items under Idaho Field Office (DOE-ID) assigned project identification number; i.e., those project numbers beginning with the character "J".

2.8. Current Subcontractor point-of-contact for nuclear material accountability.

b. Final Report:

The Subcontractor shall furnish within 6 months after the shipment of all remaining material under this Subcontract, a report indicating the amount of material returned and whether additional material requests are planned.

3. **Period of Performance**

The work described in the Statement of Work is effective August 1, 2008, and shall be completed on or before August 31, 2013.

4. **Reactor Fuel Special Provisions**

a. Title to all special nuclear materials loaned to the Subcontractor under this Subcontract shall at all times be and remain the United States Government.

b. The Contractor will not charge the Subcontractor for materials (1) consumed in the operation of the facility until expiration of this Subcontract, and (2) not recovered in reprocessing subsequent to the ultimate return of the special nuclear material.

c. As a Nuclear Regulatory Commission (NRC) Licensee, the Subcontractor shall, in addition to complying with 10CFR 73.37 and 73.72, be responsible for performing (or contracting others to perform) the actions necessary for compliance with the Order for Safeguards and Security Compensatory Measures on the Transportation of Spent Nuclear Fuel greater than 100 grams, as modified by the NRC from time to time. If required, arrangements for armed escorts are the responsibility of the Subcontractor.

d. If the Subcontractor desires to return material provided under this Subcontract, the Subcontractor shall submit a request to the Contractor, preferably within 18 months, but no later than 6 months, from the time which the Subcontractor desires to return the materials to the DOE, indicating the characteristic and amount of material the Subcontractor desires to return. The Contractor will provide requirements for documentation and instructions for returning the material. At the Contractor's option, the Contractor will provide a shipping container and provide funds directly to a Carrier, or under a Separate Purchase Order (subject to negotiated cost limitations), the Contractor will reimburse the Subcontractor for commercial shipping container rental, use of a Carrier, and other costs for activities incident to the shipment of the material. The Subcontractor has no responsibility for receipt at a DOE facility, storage nor processing of such material. The Subcontractor's obligation is to return material in the form defined, as affected by the activities listed above in Article 1.

e. Except as otherwise provided herein, the Subcontractor is responsible for and will pay the Contractor any charges imposed by the Contractor for material delivered to the Subcontractor and not ultimately returned to the Contractor.

g. Notwithstanding any other provision of this Subcontract, the Contractor or the Government shall not be responsible for or have any obligation to the Subcontractor for decontamination or decommissioning (D&D) of any of the Subcontractor's facilities.

h. The Subcontractor is responsible for the management, accountability and control of DOE-owned nuclear material in its possession. Nuclear material supplied under this Subcontract by the DOE shall comply with the following requirements:

1. Nuclear material is accounted for with a 10-digit alphanumeric, budget and reporting project identification number, which is assigned and controlled by Idaho Operations (NE-ID). The Subcontractor is not allowed to make changes to this number.
2. The project identification number must be recorded in the Project Number field on the DOE/NRC Form 741, "Nuclear Material Transaction Report", involving any activity, e.g., receipts, removal and adjustments (Reference NUREG BR-0006, "Instructions for Completing Nuclear Material Transaction Reports"); and DOE/NRC Form 742C, "Physical Inventory Listing" (Reference NUREG BR-0007, "Instructions for the Preparation and Distribution of Material Status Reports").

i. In the event the terms and conditions of this Subcontract are not in agreement with NRC rules and regulations, the NRC requirements will take precedence.

5. Subcontract Administration

a. The Contractor's Subcontract Administrator for this Subcontract is Lynda Keller. The Subcontract Administrator is the only person authorized to make changes in the requirements of this Subcontract or make modifications to this Subcontract, including changes or modifications to the Statement of Work and the Schedule. The Subcontractor shall direct all notices and requests for approval required by this Subcontract to the Subcontract Administrator.

Any notices and approvals required by this Subcontract from the Contractor to the Subcontractor shall be issued by the Subcontract Administrator.

b. The Contractor's Technical Representative for this Subcontract is D. Morrell. The Technical Representative is the person designated to monitor the Subcontract work and to interpret and clarify the technical requirements of the Statement of Work. The Technical Representative is not authorized to make changes to the work or modify this Subcontract.

c. The Contractor's Materials Management and Accountability representative for this Subcontract is M. Wilkinson. Progress reports as specified in Section 2.a. shall be provided to the representative according to the timeliness established by DOE and NRC directives.

d. The Subcontractor's Subcontract Administrator for this Subcontract is Dr. Susan Wyatt Sedwick.

6. Supplier Performance Evaluation System (SPES)

Contractor evaluates subcontractor performance in accordance with the SPES. The Subcontractor shall be formally evaluated no less than quarterly as applicable, and upon completion of the work. A minimum score of 80 points out of 100 is required to maintain approved status. Information concerning the SPES is available for review at: http://www.inl.gov/procurement/forms.shtml. Select INL Supplier Management Program.

7. Lower-tier Subcontractors

Subcontractor shall not subcontract performance of any portion of the work being performed at the INL without the advanced written approval of Contractor, (excluding material deliveries). Lower-tier

subcontracts and purchase orders must include provisions to secure all rights and remedies of Contractor and the Government provided under this Subcontract, and must impose upon the lower-tier subcontractor all of the general duties and obligations required to fulfill this Subcontract. Subcontractor is responsible for the performance and oversight of all lower-tier subcontractors

8. Order of Precedence

In the event of any inconsistency between provisions of this Subcontract, the inconsistency shall be resolved by giving precedence as follows: (a) Subcontract Change documents, if any, (b) Subcontract, (c) Specifications or Statement of Work, (d) General Provisions, and (e) other provisions of this Subcontract, whether incorporated by reference or otherwise. However, Subcontractor shall notify Contractor prior to performing work based on resolution of any inconsistency by he order of precedence set forth herein.

9. Applicable Documents

The following documents are applicable to Subcontract:

a. 10 CFR 73.37 and 73.72.

b. Order for Safeguards and Security Compensatory Measurements on the Transportation of Spent Nuclear Fuel.

c. DOE/NRC Form 741, Nuclear Material Transaction Report.

d. DOE-NRC form 742, Material Balance Report.

e. DOE/NRC Form 742C, Physical inventory Listing.

f. NUREG BR-0006, Instructions for Completing Nuclear Material Transaction reports.

g. NUREG BR-0007, Instructions for the Preparation and Distribution of Material Status Reports.

h. DOE Order 5660.1B, Management of Nuclear Materials.

GENERAL PROVISIONS

CLAUSE 1 - PUBLICATIONS

A. The Subcontractor shall closely coordinate with the Contractor's Technical Representative regarding any proposed scientific, technical or professional publication of the results of the work performed or any data developed under this Subcontract. The Subcontractor shall provide the Contractor an opportunity to review any proposed manuscripts describing, in whole or in part, the results of the work performed or any data developed under this Subcontract at least forty-five (45) days prior to their submission for publication. The Contractor will review the proposed publication and provide comments. A response shall be provided to the Subcontractor within forty-five (45) days; otherwise, the Subcontractor may assume that the Contractor has no comments. Subject to the requirements of Clause 9, the Subcontractor agrees to address any concerns or issues identified by the Contractor prior to submission for publication.

B. Subcontractor may acknowledge the Contractor and Government sponsorship of the work as appropriate.

CLAUSE 2 – NOTICES

A. The Subcontractor shall immediately notify the Contractor's Subcontract Administrator in writing of: (1) any action, including any proceeding before an administrative agency, filed against the Subcontractor arising out of the performance of this Subcontract; and (2) any claim against the Subcontractor, the cost and expense of which is allowable under the terms of this Subcontract.

B. If, at any time during the performance of this Subcontract, the Subcontractor becomes aware of any circumstances which may jeopardize its performance of all or any portion of the Subcontract, it shall immediately notify the Contractor's Subcontract Administrator in writing of such circumstances, and the Subcontractor shall take whatever action is necessary to cure such defect within the shortest possible time.

CLAUSE 3 – ASSIGNMENTS

The Contractor may assign this Subcontract to the Government or its designee(s). Except as to assignment of payment due, the Subcontractor shall have no right to assign or mortgage this Subcontract or any part of it without the prior written approval of the Contractor's Subcontract Administrator, except for subcontracts already identified in the Subcontractor's proposal.

CLAUSE 4 – DISPUTES

A. Informal Resolution

1. The parties to a dispute shall attempt to resolve it in good faith, by direct, informal negotiations. All negotiations shall be confidential. Pending resolution of the dispute, the Subcontractor shall proceed diligently with the performance of this Subcontract, in accordance with its terms and conditions.
2. The parties, upon mutual agreement, may seek the assistance of a neutral third party at any time, but they must seek such assistance no later than 120 days after the date of the Contractor's receipt of a claim. The requirement to seek the assistance of a neutral third party may be waived or modified only with the consent of all parties. The parties may

request the assistance of an established Ombudsman Program, where available, or hire a mutually agreeable mediator, or ask the DOE Office of Dispute Resolution to assist them in selecting a mutually agreeable mediator. The cost of mediation shall be shared equally by both parties. If requested by both parties, the neutral third party may offer a non-binding opinion as to a possible settlement. All discussions with the neutral third party shall be confidential.

3. In the event the parties are unable to resolve the dispute by using a neutral third party or waive the requirement to seek such assistance, the Contractor will issue a written decision on the claim.

B. Formal Resolution

1. If a dispute has not been resolved by informal resolution, it may be submitted to binding arbitration upon agreement of both parties, by and in accordance with the Commercial Arbitration Rules of the American Arbitration Association (AAA). If arbitration is agreed to by both parties, such decision is irrevocable and the outcome of the arbitration shall be binding on all parties.

2. Each party to the arbitration shall pay its pro rata share of the arbitration fees, not including counsel fees or witness fees or other expenses incurred by the party for its own benefit.

3. Judgment on the award rendered by the arbitrator may be entered in any court having jurisdiction.

C. Litigation

If arbitration is declined for such disputes, the parties may pursue litigation in any court of competent jurisdiction.

D. Governing Law

This Subcontract shall be interpreted and governed in accordance with all applicable federal and state laws and all applicable federal rules and regulations.

CLAUSE 5 - RESPONSIBILITY FOR TECHNOLOGY EXPORT CONTROL

The parties understand that materials and information resulting from the performance of this Subcontract may be subject to export control laws and that each party is responsible for its own compliance with such laws.

CLAUSE 6 – COST ACCOUNTING STANDARDS (CAS) LIABILITY

[Applicable to Subcontracts exceeding $500,000]

Clause 10 below incorporates into these GENERAL PROVISIONS clauses entitled, "*COST ACCOUNTING STANDARDS*" *and* "*ADMINISTRATION OF COST ACCOUNTING STANDARDS.*" Notwithstanding the provisions of these clauses, or of any other provision of the Subcontract, the Subcontractor shall be liable to the Government for any increased costs, or interest thereon, resulting from any failure of the Subcontractor, with respect to activities carried on at the site of the work, or of a subcontractor, to comply with applicable cost accounting standards or to follow any practices disclosed pursuant to the requirements of such clause.

CLAUSE 7 - DISCLOSURE AND USE RESTRICTIONS FOR LIMITED RIGHTS DATA

Generally, delivery of Limited Rights Data (or Restricted Computer Software) should not be necessary. However, only if Limited Rights Data will be used in meeting the delivery requirements of the subcontract, the following disclosure and use restrictions shall apply to and shall be inserted in, any FAR 52.227-14 Limited Rights Notice on any Limited Rights Data furnished or delivered by the Subcontractor or a lower-tier subcontractor:

A. These "Limited Rights Data" may be disclosed for evaluation purposes under the restriction that the "Limited Rights Data" be retained in confidence and not be further disclosed;

B. These "Limited Rights Data" may be disclosed to other contractors participating in the Government's program of which this Subcontract is a part for information or use in connection with the work performed under their contracts and under the restriction that the "Limited Rights Data" be retained in confidence and not be further disclosed; and

C. These "Limited Rights Data" may be used by the Government or others on its behalf for emergency repair or overhaul work under the restriction that the "Limited Rights Data" be retained in confidence and not be further disclosed.

CLAUSE 8 - ORDER OF PRECEDENCE

Any inconsistencies in the documents comprising this Subcontract shall be resolved by giving precedence in the following order: (a) the SCHEDULE OF ARTICLES and this Subcontract Signature Page; (b) these GENERAL PROVISIONS; (c) other referenced documents, exhibits, and attachments; and (d) any referenced specification or *Statement of Work*.

CLAUSE 9 – SECURITY REQUIREMENTS

A. This Subcontract is intended for unclassified, publicly releasable research or development work. The Contractor does not expect that results of the research project will involve classified information or Unclassified Controlled Nuclear Information (UCNI) (See 10 CFR part 1017). However, the Contractor may review the research work generated under this Subcontract at any time to determine if it requires classification or control as UCNI.

B. If, subsequent to the date of this Subcontract, a review of the information reveals that classified information or UCNI is being generated under this Subcontract, then the security requirements of this Subcontract must be changed. If such changes cause an increase or decrease in costs or otherwise affect any other term or condition of this Subcontract, the Subcontract shall be subject to an equitable adjustment as if the changes were directed under the Changes clause of this Subcontract.

C. If the security requirements are changed, the Subcontractor shall exert every reasonable effort compatible with its established policies to continue the performance of work under the Subcontract in compliance with the change in the security requirements. If the Subcontractor determines that continuation of the work under this Subcontract is not practicable because of the change in security requirements, the Subcontractor shall notify the Contractor's Procurement Representative in writing. Until the Contractor's Procurement Representative provides direction, the Subcontractor shall protect the material as directed by the Contractor.

D. After receiving the written notification, the Contractor's Procurement Representative shall explore the circumstances surrounding the proposed change in security requirements and shall

endeavor to work out a mutually satisfactory method to allow the Subcontractor to continue performance of work under this Subcontract.

E. Within 15 days of receiving the written notification of the Subcontractor's stated inability to proceed, the Contractor's Procurement Representative must determine whether (1) these security requirements do not apply to this contract or (2) a mutually satisfactory method for continuing performance of work under this Subcontract can be agreed upon. If this determination is not made, the Subcontractor may request the Contractor's Procurement Representative to terminate the Subcontract in whole or in part. The Contractor's Procurement Representative shall terminate the Subcontract in whole or in part, as may be appropriate, and the termination shall be deemed a termination under the terms of the Termination for the Convenience of the Government clause.

CLAUSE I0 - CLAUSES INCORPORATED BY REFERENCE

The FEDERAL ACQUISITION REGULATION (FAR) and the U.S. DEPARTMENT OF ENERGY ACQUISITION REGULATION (DEAR) clauses listed below, which are located in Chapters 1 and 9, respectively, of Title 48 of the Code of Federal Regulations, are incorporated by this reference as a part of these GENERAL PROVISIONS with the same force and effect as if they were given in full text, as prescribed below.

The full text of the clauses may be accessed electronically at http://www.arnet.gov/far/ (FAR) and http://professionals.pr.doe.gov/ma5/MA-5Web.nsf/Procurement/Acquisition/Regulation (DEAR).

As used in the clauses, the term "contract" shall mean this Subcontract; the term "Contractor" shall mean the Subcontractor; the term "subcontractor" shall mean the Subcontractor's subcontractor, and the terms "Government" and "Contracting Officer" shall mean the Contractor, except in FAR clause 52.227-14, and DEAR clauses 970.5227-4, 952.227-11, 970.5232-3 and 52.245-5 Alternate I, in which clauses "Government" shall mean the United States Government and "Contracting Officer" shall mean the DOE/NNSA Contracting Officer for Prime Contract DE-AC07-05ID14517 with the Contractor. As used in DEAR clauses 952.204-72 and 952.227-9, the term "DOE" shall mean DOE/NNSA or the Contractor.

The modifications of these clause terms are intended to appropriately identify the parties and establish their contractual and administrative reporting relationship, and shall not apply to the extent they would affect the U.S. Government's rights. The Subcontractor shall include the listed clauses in its subcontracts at any tier, to the extent applicable.

APPLICABLE TO ALL SUBCONTRACTS UNLESS OTHERWISE INDICATED BELOW:

DEAR 952.204-71	SENSITIVE FOREIGN NATIONS CONTROLS (APR 1994). Applies if the Subcontract is for unclassified research involving nuclear technology.
FAR 52.216-7	ALLOWABLE COST AND PAYMENT (DEC 2002). Substitute 31.3 in subcontracts with educational institutions for 31.2 in paragraph (a).
FAR 52.216-15	PREDETERMINED INDIRECT COSTS RATES APR 1998).
FAR 52.222-21	PROHIBITION OF SEGREGATED FACILITIES (FEB 1999).
FAR 52.222-26	EQUAL OPPORTUNITY (APR 2002).
FAR 52.223-3	HAZARDOUS MATERIAL IDENTIFICATION AND MATERIAL SAFETY DATA SHEETS (JAN 1997) AND ALTERNATE I. Applies only if Subcontract involves delivery of hazardous materials.

FAR 52.225-13	RESTRICTIONS ON CERTAIN FOREIGN PURCHASES (DEC 2003).
DEAR 970.5227-4	AUTHORIZATION AND CONSENT (AUG 2002), Paragraph (a).
DEAR 952.227-9	REFUND OF ROYALTIES (FEB 1995). Applies if "royalties" of more than $250 are paid by a subcontractor at any tier.
DEAR 952.227-11	PATENT RIGHTS - RETENTION BY THE CONTRACTOR (SHORT FORM) (FEB 1995). (Applies only if Subcontractor is a nonprofit organization as set forth in 48 CFR 27.301. If Subcontractor does not qualify in accordance with 48 CFR 27.301, it may request a patent waiver pursuant to 10 CFR 784.)
FAR 52.227-14	[Check provision below that applies OR include only applicable provision]. _____ RIGHTS IN DATA-GENERAL (JUN 1987) with ALTERNATE V and DEAR 927.409 Paragraphs (a) and (d)(3). Applies if the Subcontract is for development work, or for basic and applied research where computer software is specified as a Deliverable in the Statement of Work or other special circumstances apply as specified in the agreement. _X_ RIGHTS IN DATA-GENERAL (JUN 1987) with ALTERNATE IV, subparagraph (c)(1) and DEAR 927.409, subparagraph (a) Definitions. Applies if the Subcontract is for basic or applied research and computer software is not specified as a Deliverable in the Statement of Work, and no other special circumstances apply per DEAR 927.409.
FAR 52.227-23	RIGHTS TO PROPOSAL DATA (TECHNICAL) (JUNE 1987). Applies if the Subcontract is based upon a technical proposal.
FAR 52.229-10	STATE OF NEW MEXICO GROSS RECEIPTS AND COMPENSATING TAX (APR 2003). Applies if any part of this Subcontract is to be performed in the State of New Mexico.
FAR 52.232-20	LIMITATION OF COST (APR 1984). Applies if the Subcontract is fully funded.
FAR 52.232-22	LIMITATION OF FUNDS (APR 1984). Applies if the Subcontract is incrementally funded.
FAR 52.242-15	STOP-WORK ORDER (AUG 1989) with ALTERNATE I (APR 1984).
FAR 52.243-2	CHANGES – COST-REIMBURSEMENT (AUG 1987), WITH ALTERNATE V
FAR 52.244-2	SUBCONTRACTS (AUG 1998) . Insert in Paragraph (e): "Any subcontract or purchase order for other than "commercial items" exceeding the simplified acquisition threshold. ("Commercial item" has the meaning contained in FAR 52.202-1, Definitions".) Applies only if there are subcontracts under this Contract.
DEAR 970.5245-1	PROPERTY (DEC 2000).
FAR 52.246-9	INSPECTION OF RESEARCH AND DEVELOPMENT (SHORT FORM) (APR 1984).
FAR 52.247-63	PREFERENCE FOR U. S. FLAG AIR CARRIERS (JUNE 2003). Applies if the Subcontract involves international air transportation.

FAR 52.247-64	PREFERENCE FOR PRIVATELY OWNED U.S.-FLAG COMMERCIAL VESSELS (APR 2003).
DEAR 952.247-70	FOREIGN TRAVEL (DEC 2000).
FAR 52.249-5	TERMINATION FOR CONVENIENCE OF THE GOVERNMENT (EDUCATIONAL AND OTHER NONPROFIT INSTITUTIONS) (SEP 1996).
DEAR 952.217-70	ACQUISITION OF REAL PROPERTY (APR 1984). Applies if the Subcontract involves leased space that is reimbursed.
DEAR 970.5232-3	ACCOUNTS, RECORDS, AND INSPECTION (DEC 2000)

APPLICABLE IF THE SUBCONTRACT IS FOR $10,000 OR MORE:

FAR 52.222-35	EQUAL OPPORTUNITY FOR SPECIAL DISABLED VETERANS, VETERANS OF THE VIETNAM ERA AND OTHER ELIGIBLE VETERANS (DEC 2001).
FAR 52.222-36	AFFIRMATIVE ACTION FOR WORKERS WITH DISABILITIES (JUNE 1998).
FAR 52.222-37	EMPLOYMENT REPORTS ON SPECIAL DISABLED VETERANS, VETERANS OF THE VIETNAM ERA AND OTHER ELIGIBLE VETERANS (DEC 2001).

APPLICABLE IF THE SUBCONTRACT EXCEEDS $100,000:

FAR 52.203-5	COVENANT AGAINST CONTINGENT FEES (APR 1984)
FAR 52.203-6	RESTRICTIONS ON SUBCONTRACTOR SALES TO THE GOVERNMENT (JULY 1995).
FAR 52.203-7	ANTI-KICKBACK PROCEDURES (JULY 1995), excluding Paragraph (c)(1).
FAR 52.203-10	PRICE OR FEE ADJUSTMENT FOR ILLEGAL OR IMPROPER ACTIVITY (JAN 1997).
FAR 52.203-12	LIMITATION ON PAYMENTS TO INFLUENCE CERTAIN FEDERAL TRANSACTIONS (JUNE 2003).
FAR 52.219-8	UTILIZATION OF SMALL BUSINESS CONCERNS (MAY 2004).
FAR 52.222-04	CONTRACT WORK HOURS AND SAFETY STANDARDS ACT - OVERTIME COMPENSATION (SEP 2000).
DEAR 970.5227-5	NOTICE AND ASSISTANCE REGARDING PATENT AND COPYRIGHT INFRINGEMENT (AUG 2002).

APPLICABLE IF THE SUBCONTRACT EXCEEDS $500,000:

FAR 52.215-10	PRICE REDUCTION FOR DEFECTIVE COST OR PRICING DATA (OCT 1997) if subcontract exceeds $550,000.

FAR 52.215-11	PRICE REDUCTION FOR DEFECTIVE COST OR PRICING DATA-MODIFICATIONS (OCT 1997) not used when 52.215-10 is included. In subcontracts greater than $550,000.
FAR 52.215-12	SUBCONTRACTOR COST OR PRICING DATA (OCT 1997). Applies if 52.215-10 applies.
FAR 52.215-13	SUBCONTRACTOR COST OR PRICING DATA-MODIFICATIONS (OCT 1997). Applies if 52.215-11 applies.
FAR 52.219-9	SMALL BUSINESS SUBCONTRACTING PLAN (JAN 2002). Applies unless there are no subcontracting possibilities.
FAR 52.227-16	ADDITIONAL DATA REQUIREMENTS (JUNE 1987).
FAR 52.230-2	COST ACCOUNTING STANDARDS (APR 1998), excluding paragraph (b). Applies to nonprofit organizations if they are subject to full CAS coverage as set forth in 48 CFR Chapter 99, Subpart 9903.201-2 (FAR Appendix B).
FAR 52.230-3	DISCLOSURE AND CONSISTENCY OF COST ACCOUNTING PRACTICES (APR 1998), excluding paragraph (b). Applies to nonprofit organizations if they are subject to modified CAS coverage as set forth in 48 CFR Chapter 99, Subpart 9903.201-2 (FAR Appendix B).
FAR 52.230-5	COST ACCOUNTING STANDARDS – EDUCATIONAL INSTITUTION (APR 1998), excluding paragraph (b).
FAR 52.230-6	ADMINISTRATION OF COST ACCOUNTING STANDARDS (NOV 1999).

(END OF GENERAL PROVISIONS)

Technical Specifications

The University of Texas at Austin
Nuclear Engineering Teaching Laboratory
TRIGA Mark II Nuclear Research Reactor

License R-129
Docket 50-602
12 December 2011

The University of Texas at Austin
Nuclear Engineering Teaching Laboratory
10100 Burnet Rd, Bldg 159
Austin, TX 78758

Table of Contents

1. DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications. Capitalization is used in the body of the Technical Specifications to identify defined terms.

ACTION Actions are steps to be accomplished in the event a required condition identified in a "Specification" section is not met, as stated in the "Condition" column of "Actions."

In using Action Statements, the following guidance applies:

- Where multiple conditions exist in an LCO, actions are linked to the failure to meet a "Specification" "Condition" by letters and number.
- Where multiple action steps are required to address a condition, COMPLETION TIME for each action is linked to the action by letter and number.
- AND in an Action Statement means all linked steps need to be performed to complete the action; OR indicates options and alternatives, only one item needs to be performed to complete the action.
- If a "Condition" exists, the "Action" consists of completing all steps associated with the selected option (if applicable) unless the "Condition" is corrected prior to completion of the steps

ANNUAL 12 months, not to exceed 15 months

BIENNIAL Every two years, not to exceed a 30 month interval

CHANNEL CALIBRATION A channel calibration is an adjustment of the channel so that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures.

CHANNEL CHECK A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with expected values, other independent channels, or other methods of measuring the same variable where possible.

CHANNEL TEST A channel test is the introduction of an input signal into a channel

	to verify that it is operable. A functional test of operability is a channel test.
CONFINEMENT	The enclosure which controls the movement into and out of the reactor bay
CONFINEMENT ISOLATION	Condition for reactor bay ventilation where: (1) dampers controlling confinement ventilation are closed, and (2) confinement ventilation fans are secured (3) the reactor bay fume/sort hood fans are secured (4) the reactor bay fume/sort hood dampers are closed The purge system may be operated in manual override
CONTROL ROD (STANDARD)	A standard control rod is one having an electric induction or stepper motor drive coupled to the control rod by an electromagnet, with scram capability.
CONTROL ROD (TRANSIENT)	A transient rod is one that is pneumatically coupled to the control rod drive, is capable of initiating a power pulse, is operated by a motor drive, and/or air pressure operated and has scram capability.
DAILY	Prior to initial operation each day (when the reactor is operated), or before an operation extending more than 1 day
ENSURE	Verify existence of specified condition or (if condition does not meet criteria) take action necessary to meet condition
EXCESS REACTIVITY	That amount of reactivity above the critical condition which would exist if all the control rods were moved to the maximum positive reactivity condition
EXPERIMENT	An EXPERIMENT is (1) any apparatus, device, or material placed in the reactor core region (in an EXPERIMENTAL FACILITY associated with the reactor, or in line with a beam of radiation emanating from the reactor) or (2) any in-core operation designed to measure reactor characteristics.
EXPERIMENTAL FACILITY	Experimental facilities are the beamports, pneumatic transfer systems, central thimble, rotary specimen rack, and displacement of fuel element positions used for EXPERIMENTS (single-element positions and the multiple element positions fabricated in the upper grid plate displacing 3, 6 or 7 elements).
IMMEDIATE	Without delay, and not exceeding one hour.

NOTE:
IMMEDIATE permits activities to restore required conditions for up to one hour; this does not permit or imply deferring or postponing action

INITIAL STARTUP	A reactor startup and approach to power following: 1 Fuel element or control rod relocations or installations within the reactor core region 2 Relocation or installation of any experiment in the core region with a reactivity worth of greater than one dollar 3 Recovery from an unscheduled (a) shutdown or (b) significant power reductions, or 4 modifications to reactor safety or control rod drive systems.
LIMITING CONDITION FOR OPERATION (LCO)	The lowest functional capability or performance levels of equipment required for safe operation of the facility.
LIMITING SAFETY SYSTEM SETTING (LSSS)	Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit placed, the setting shall be chosen so that the automatic protective action will correct the abnormal situation before a safety limit is exceeded.
MEASURED VALUE	The measured value of a parameter is the value as indicated at the output of a MEASURING CHANNEL.
MEASURING CHANNEL	A MEASURING CHANNEL is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.
MOVABLE EXPERIMENT	A MOVABLE EXPERIMENT is one the EXPERIMENT may be moved into, out-of or near the reactor while the reactor is OPERATING.
OPERABLE	A system or component is OPERABLE when it is capable of performing its intended function in a normal manner
OPERATING	A system or component is OPERATING when it is performing its intended function in a normal manner.
PULSE MODE	The reactor is in the PULSE MODE when the key switch is in the "on" position, the reactor mode selection switch is in the pulse position and the reactor display indicates pulse mode.

NOTE:
In the PULSE MODE, reactor power may be increased on a period of much less than I second by motion of the transient control rod.

REACTOR SAFETY SYSTEM	The REACTOR SAFETY SYSTEM is that combination of MEASURING CHANNELS and associated circuitry that is designed to initiate a reactor scram or that provides information that requires manual protective action to be initiated.
REACTOR SHUTDOWN	The reactor is shutdown if it is subcritical by at least the minimum required amount of reactivity (shutdown margin) in the REFERENCE CORE CONDITION with the reactivity worth of all experiments included.
REFERENCE CORE CONDITION	The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$0.30)
SAFETY CHANNEL	A safety channel is a MEASURING CHANNEL in the REACTOR SAFETY SYSTEM.
SAFETY LIMITS	Limits on important process variables which are found to be necessary to protect reasonably the integrity of the principal barriers (i.e., fuel element cladding) which guard against the uncontrolled release of radioactivity.
SECURED EXPERIMENT	A secured EXPERIMENT is an EXPERIMENT held firmly in place by a mechanical device or by gravity providing that the weight of the EXPERIMENT is such that it cannot be moved by forces (1) normal to the operating environment of the experiment or (2) that might result from credible failures.
SHALL (SHALL NOT)	Indicates specified action is required/(or required not to be performed)
SEMIANNUAL	Every six months, with intervals not greater than 7 ½ months
SHUTDOWN MARGIN	The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical without further operator action
STANDARD FUEL ELEMENT	A standard fuel element is a single TRIGA element of standard type, U-ZrH clad in stainless steel with nominal hydrogen to zirconium ratio of 1.6.

INTSTRUMENTED FUEL ELEMENT	An instrumented fuel element (IFE) is a stainless steel clad fuel element containing three sheathed thermocouples embedded in the fuel element.
STEADY-STATE MODE	The reactor is in the steady-state mode when the key switch is in the "on" position, the reactor mode selector pushbutton switch has requested either the manual, automatic, or square wave position and the reactor display indicates manual, automatic, or square wave.
TECHNICAL SPECIFICATION VIOLATION	(1) A violation of a Safety Limit occurs when the Safety Limit value is exceeded. (2) A violation of a Limiting Safety System Setting or Limiting Condition for Operation) occurs when a "Condition" exists which does not meet a "Specification" and the corresponding "Action" has not been met within the required "Completion Time." A violation has not occurred if the "Action" statement of (1) an LSSS or LCO is completed or (2) the "Specification" is restored within the prescribed "Completion Time," *NOTE* *"Condition," "Specification," "Action," and "Completion Time" refer to applicable titles of sections in individual Technical Specifications*

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Fuel Element Temperature Safety Limit

2.1.1 Applicability

This specification applies when the reactor in STEADY STATE MODE and the PULSE MODE.

2.1.2 Objective

This SAFETY LIMIT ensures fuel element cladding integrity

2.1.3 Specification

A	Stainless steel clad, high-hydride fuel element temperature SHALL NOT exceed 1150°C.
B	Steady state fuel temperature shall not exceed 750°C.

2.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Stainless steel clad, high-hydride fuel element temperature exceeds 1150°C.	A.1 ENSURE SHUTDOWN condition	A.1 IMMEDIATE
OR	AND	
B. Fuel temperature exceeds 750°C in steady state conditions	A.2 Report per Section 6.8	A.2 Within 24 hours

2.1.5 Bases

Safety Analysis Report Chapter 4 (4.2.1 B) identifies design and operating constraints for TRIGA fuel that will ensure cladding integrity is not challenged.

NUREG 1282 identifies the safety limit for the high-hydride ($ZrH_{1.6}$) fuel elements with stainless steel cladding based on the stress in the cladding (resulting from the hydrogen pressure from the dissociation of the zirconium hydride). This stress will remain below the yield strength of the stainless steel cladding with fuel temperatures below 1150°C. A change in yield strength occurs for stainless steel cladding temperatures of 500°C, but

there is no scenario for fuel cladding to achieve 500°C while submerged or in air; consequently the safety limit during reactor operations is 1150°C.

Therefore, the important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification, and it is readily measured. During operation, fission product gases and dissociation of the hydrogen and zirconium builds up gas inventory in internal components and spaces of the fuel elements. Fuel temperature acting on these gases controls fuel element internal pressure. Limiting the maximum temperature prevents excessive internal pressures that could be generated by heating these gases.

Fuel growth and deformation can occur during normal operations, as described in Chapter 4 (4.2.1 Z). Damage mechanisms include fission recoils and fission gases, strongly influenced by thermal gradients. Limiting steady state operating fuel temperature to less than 750°C limits potential fuel growth.

2.2 Limiting Safety System Settings (LSSS)

2.2.1 Applicability

This specification applies when the reactor in STEADY STATE MODE

2.2.2 Objective

The objective of this specification is to ensure the safety limit is not exceeded.

2.2.3 Specifications

A	Power level SHALL NOT exceed 1100 kW (th) in STEADY STATE MODE of operation
B	Instrumented elements in the B or C ring SHALL indicate less than 550°C

2.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Steady state power level exceeds 1100 kW (th)	A.1 Reduce power to less than 1100 kW (th) OR A.2. ENSURE REACTOR SHUTDOWN condition	A.1 IMMEDIATE A.2. IMMEDIATE
B. An INTSTRUMENTED FUEL ELEMENT in the B or C ring indicates greater than 550°C	B.1. ENSURE REACTOR SHUTDOWN condition OR B.2 VERIFY the MEASURED VALUE is not correct	B.2. IMMEDIATE B.2 IMMEDIATE

2.2.5 Bases

Analysis in SAR Chapter 4 (4.6 B) demonstrates that if operating thermal (th) power is 1100 kW, the maximum steady state fuel temperature is less than the safety limit for steady state operations by a large margin. For normal pool temperature, calculations in Chapter 4 demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore,

steady state operations at a maximum of 1100 kW meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin. Steady state operation of 1100 kW was assumed in analyzing the loss of cooling and maximum hypothetical accidents. The analysis assumptions are protected by assuring that the maximum steady state operating power level is 1100 kW.

The actual safety system setting will be chosen to ensure that a scram will occur at a level that does not exceed 1,100 kW.

Instrumented fuel element temperatures less than 550°C ensures the SAFETY LIMIT on fuel temperature is met.

3. LIMITING CONDITIONS FOR OPERATION (LCO)

3.1 Core Reactivity

3.1.1 Applicability

These specifications are required prior to entering STEADY STATE MODE or PULSING MODE in OPERATING conditions; reactivity limits on experiments are specified in Section 3.8.

3.1.2 Objective

This LCO ensures the reactivity control system is OPERABLE, and that an accidental or inadvertent pulse does not result in exceeding the safety limit.

3.1.3 Specification

A	The maximum available core reactivity (EXCESS REACTIVITY) with all control rods fully withdrawn does not exceed 4.9% Δkk ($7.00) when: 1. REFERENCE CORE CONDITIONS exists 2. No MOVEABLE EXPERIMENTS with net-negative reactivity worth are in place
B	The reactor is capable of being made subcritical by a SHUTDOWN MARGIN more than 0.002 Δk/k ($0.29) under REFERENCE CORE CONDITIONS and the following conditions: 1. The highest worth control rod is fully withdrawn 2. The highest worth NONSECURED EXPERIMENT is in its most positive reactive state, and each SECURED EXPERIMENT is in its most reactive state.

3.1.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactivity with all control rods fully withdrawn exceeds 4.9% Δk/k ($7.00)	A.1 ENSURE REACTOR SHUTDOWN AND A.2 Configure reactor to meet LCO	A.1 IMMEDIATE A.2 Prior to continued operations

B. The reactor is not subcritical by more than 0.002 Δk/k ($0.29) under specified conditions	B.1.a ENSURE operable control rods are fully inserted AND B.1.b Secure electrical power to the control rod circuits (magnet or motor power) AND B.1.c Secure all work on in-core experiments or installed control rod drives AND B.2 Configure reactor to meet LCO	B.1 IMMEDIATE B.2 Prior to continued operations

3.1.5 Bases

The stated value for excess reactivity was used in establishing core conditions for calculations in Chapter 13 (13.4) to demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis. Since the fundamental protection for the UT reactor is the maximum power level and fuel temperature that can be achieved with the available positive core reactivity, experiments with positive reactivity are included in determining excess reactivity. Since experiments with negative reactivity will increase available reactivity if they are removed during operation, they are not credited in determining excess reactivity.

Analysis shows that at the limiting pool water temperature and zero power, fuel temperature approaches 950°C with a reactivity addition of $5.94, and 1050°C with a reactivity addition of $5.66, while a $4.00 reactivity addition results in peak fuel temperature of about 770°C. If the pulse occurs with the reactor operating at 880 MW, a $4.00 reactivity insertion results in peak fuel temperature of 930°C; this is only 3% below the safety limit for cladding with temperature greater than 500°C, but is well below the safety limit when cladding temperature is less than 500°C. Since the cladding temperature is shown to be less than 500°C with the reactor operating in Chapter 4, worst-case steady state operation at 880 kW leads to a maximum fuel temperature well below the safety limit.

The limiting SHUTDOWN MARGIN is necessary so that the reactor can be shut down from any operating condition, and will remain shutdown after cool down and xenon decay, even if one control rod (including the transient control rod) should remain in the fully withdrawn position. Analysis in Chapter 4 (4.5.1) demonstrates the capability of the control rods to meet this requirement.

3.2 PULSED MODE Operations

3.2.1 Applicability

These specifications apply to operation of the reactor in the PULSE MODE.

3.2.2 Objective

This Limiting Condition for Operation prevents fuel temperature safety limit from being exceeded during PULSE MODE operation.

3.2.3 Specification

A	The transient rod drive is positioned for reactivity insertion (upon withdrawal) less than or equal to 2.8% δk ($4.00)

3.2.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With all stainless steel clad fuel elements, the worth of the pulse rod in the transient rod drive position is greater than $4.00 in the PULSE MODE	A.1 Position the transient rod drive for pulse rod worth less than or equal to $4.00 OR A.2 Place reactor in STEADY STATE MODE	A.1 IMMEDIATE OR A.2 IMMEDIATE

3.2.5 Bases

The value for pulsed reactivity with all stainless steel elements in the core was used in establishing core conditions for calculations in Chapter 13 (13.4) that demonstrate fuel temperature limits are met during potential accident scenarios under extremely conservative conditions of analysis.

3.3 MEASURING CHANNELS

3.3.1 Applicability

This specification applies to the reactor MEASURING CHANNELS during STEADY STATE MODE and PULSE MODE operations.

3.3.2 Objective

The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor

3.3.3 Specifications

A	The MEASURING CHANNELS specified in TABLE 1 SHALL be OPERATING
B	The neutron count rate on the startup channel is greater $2x10^{-7}$ %
C	The particulate continuous air monitor SHALL be operating and capable of initiating CONFINEMENT ISOLATION

TABLE 1: MINIMUM MEASURING CHANNEL COMPLEMENT

MEASURING CHANNEL	Minimum Number Operable	
	STEADY STATE MODE	PULSE MODE
Reactor power level[1]	2	1
Primary Pool Water Temperature	1	1
Fuel Temperature	1	1
Pool area radiation monitor[2]	1	1
Lower or middle level area monitor[2]	1	1
Effluent air radiation monitor	1	1
Particulate air radiation monitor	1	1

NOTE[1]: One "Startup Channel" required to have range that indicates <10 W
NOTE[2]: High-level alarms audible in the control room may be used

3.3.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.1 Reactor power channels not OPERATING (min 2 for STEADY STATE, 1 PULSE MODE)	A.1.1 Restore channel to operation OR A.1.2 ENSURE reactor is SHUTDOWN	A.1.1 IMMEDIATE A.1.2 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.2 High voltage to reactor safety channel (power level) detector less than 80% of required operating value	A.2.1 Establish REACTOR SHUTDOWN condition AND A.2.2 Enter REACTOR SECURED mode	A.2. IMMEDIATE
B. Primary water temperature, reactor bay differential pressure or fuel temperature CHANNEL not operable	B.1 Restore channel to operation OR B.2 Monitor pool water temperature OR B.3 ENSURE reactor is SHUTDOWN	B.1 IMMEDIATE B.2 IMMEDIATE AND At least once per hour B.3 IMMEDIATE
C. Pool Area Radiation Monitor is not OPERATING	C.1 Restore MEASURING CHANNEL OR C.2 ENSURE reactor is shutdown OR C.3 ENSURE personnel are not on the upper level OR C.4 ENSURE personnel on upper level are using portable survey meters to monitor dose rates	C.1 IMMEDIATE C.2 IMMEDIATE C.3 IMMEDIATE C.4 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Lower or middle level area monitor is not OPERATING	D.1 Restore MEASURING CHANNEL OR D.2 ENSURE reactor is shutdown OR D.3 ENSURE personnel are not in the reactor bay OR D.4 ENSURE personnel entering reactor bay are using portable survey meters to monitor dose rates	D.1 IMMEDIATE D.2 IMMEDIATE D.3 IMMEDIATE D.4 IMMEDIATE
E. Continuous particulate air radiation monitor is not OPERATING	E.1 Restore MEASURING CHANNEL OR E.2 ENSURE reactor is shutdown OR E.3.a ENSURE Argon 41 monitor radiation monitor is OPERATING AND E.3.b Restore MEASURING CHANNEL	E.1 IMMEDIATE E.2. IMMEDIATE E.3.a. IMMEDIATE E.3.b Within 30 working days

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Argon monitor is not OPERATING	F.1 Restore MEASURING CHANNEL OR F.2 ENSURE reactor is shutdown OR F.3.a ENSURE continuous air radiation monitor is OPERATING AND F.3.b Restore MEASURING CHANNEL	F.1 IMMEDIATE F.2. IMMEDIATE F.3.a. IMMEDIATE F.3.b Within 30 working days
G. The neutron count rate on the startup channel is not greater than $2x10^{-7}$ %	G.1 Do not perform a reactor startup OR G.2 Perform a neutron-source check on the startup channel prior to startup	G.1 IMMEDIATE G.2 IMMEDIATE

3.3.5 Bases

Maximum steady state power level is 1100 kW; neutron detectors measure reactor power level. Chapter 4 and 13 discuss normal and accident heat removal capabilities. Chapter 7 discusses radiation detection and monitoring systems, and neutron and power level detection systems.

According to General Atomics, detector voltages less than 80% of required operating value do not provide reliable, accurate nuclear instrumentation. Therefore, if operating voltage falls below the minimum value the power level channel is inoperable.

Pool water temperature indication is required to assure water temperature limits are met, protecting primary cleanup resin integrity. Analysis in Chapter 4 and 13 assume a maximum fuel temperature based on protection of resin integrity. Fuel temperature indication provides a means of observing that the safety limits are met.

The upper and lower level area radiation monitors provide information about radiation hazards in the reactor bay. A loss of reactor pool water (Chapter 13), changes in

shielding effectiveness (Chapter 11), and releases of radioactive material to the restricted area (Chapter 11) that could cause changes in radiation levels within the reactor bay detectable by these monitors. Portable survey instruments will detect changes in radiation levels.

The air monitors (continuous particulate air- and argon radiation-monitor) provide indication of airborne contaminants in the reactor bay. These channels provide evidence of fuel element failure on independent channels; the particulate air monitor gas has maximum sensitivity to iodine and particulate activity, while the argon channel detects noble gas.

Permitting operation using a single channel of atmospheric monitoring will reduce unnecessary shutdowns while maintaining the ability to detect abnormal conditions as they develop. Relative indications ensure discharges are routine; abnormal indications trigger investigation or action to prevent the release of radioactive material to the surrounding environment. Ensuring the alternate airborne contamination monitor is functioning during outages of one system provides the contamination monitoring required for detecting abnormal conditions. Limiting the outage for a single unit to a maximum of 30 days ensures radioactive atmospheric contaminants are monitored while permitting maintenance and repair outages on the other system.

Chapter 13 discusses inventories and releases of radioactive material from fuel element failure into the reactor bay, and to the environment. Particulate and noble gas channels monitor more routine discharges. Chapter 11 discusses routine discharges of radioactive gasses generated from normal operations into the reactor bay and into the environment. Chapters 3 and 9 identifies design bases for the confinement and ventilation system. Chapter 7 discusses air-monitoring systems. The 30 day interval is selected as adequate to accomplish complex repairs, and limited enough that with one system functional there is no significant chance that the system will fail during a period that requires detection of airborne radioactivity.

Experience has shown that subcritical multiplication with the neutron source used in the reactor does not provide enough neutron flux to correspond to an indicated power level of $2x10^{-7}$ %. Therefore an indicated power of $2x10^{-7}$ %. or more indicates operating in a potential critical condition, and at least one neutron channel is required with sensitivity at a neutron flux level corresponding to reactor power levels less than $2x10^{-7}$ % ("Startup Channel"). If the indicated neutron level is less than the minimum sensitivity for the channel, a neutron source will be used to determine that the channels is responding to neutrons to ensure that the channel is functioning prior to startup.

3.4 Safety Channel and Control Rod Operability

3.4.1 Applicability

This specification applies to the reactor MEASURING Channels during STEADY STATE MODE and PULSE MODE operations.

3.4.2 Objective

The objectives are to require the minimum number of REACTOR SAFETY SYSTEM channels that must be OPERABLE in order to ensure that the fuel temperature safety limit is not exceeded, and to ensure prompt shutdown in the event of a scram signal.

3.4.3 Specifications

A	The SAFETY SYSTEM CHANNELS specified in TABLE 2 are OPERABLE
B	CONTROL RODS (STANDARD) are capable of full insertion from the fully withdrawn position in less than 1 sec.

TABLE 2: REQUIRED SAFETY SYSTEM CHANNELS

Safety System Channel or Interlock	Minimum Number Operable	Function	Required OPERATING Mode	
			STEADY STATE MODE	PULSE MODE
Reactor power level	2	Scram	YES	NA
Manual scram bar	1	Scram	YES	YES
CONTROL ROD (STANDARD) position interlock	1	Prevent withdrawal of standard rods in the PULSE MODE	NA	YES
Pulse rod interlock[1]	1	Prevent inadvertent pulsing while in STEADY STATE MODE	YES	NA

NOTE [1]: The pulse rod interlock prevents air from being applied to the pulse rod unless the transient rod is fully inserted except during pulse mode or square wave operations.

3.4.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any required SAFETY SYSTEM CHANNEL or interlock function is not OPERABLE	A.1 Restore channel or interlock to operation OR A.2 ENSURE reactor is SHUTDOWN	A1. IMMEDIATE A2. IMMEDIATE

3.4.5 Bases

The power level scram is provided to ensure that reactor operation stays within the licensed limits of 1,100 kW, preventing abnormally high fuel temperature. The power level scram is not credited in analysis, but provides defense in depth to assure that the reactor is not operated in conditions beyond the assumptions used in analysis (Chapter 4 and 13).

The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

The CONTROL ROD (STANDARD) interlock function is to prevent withdrawing control rods (other than the pulse rod) when the reactor is in the PULSE MODE. This will ensure the reactivity addition rate during a pulse is limited to the reactivity added by the pulse rod.

The pulse rod interlock function prevents air from being applied to the transient rod drive when it is withdrawn while disconnected from the control rod to prevent inadvertent pulses during STEADY STATE MODE operations. The control rod interlock prevents inadvertent pulses which would be likely to exceed the maximum range of the power level instruments configured for steady state operations.

3.5 Gaseous Effluent Control

3.5.1 Applicability

This specification applies to gaseous effluent in STEADY STATE MODE and PULSE MODE.

3.5.2 Objective

The objective is to ensure that exposures to the public resulting from gaseous effluents released during normal operations and accident conditions are within limits and ALARA.

3.5.3 Specification

A	The reactor bay HVAC confinement system SHALL provide ventilation to the reactor bay when particulate continuous air monitor indicates less than 10,000 cpm
B	The reactor bay confinement system will enter CONFINEMENT ISOLATION if the particulate continuous air monitor is in-service and indicates greater than 10,000 cpm
C	Auxiliary purge system SHALL exhaust from reactor bay pool and in-use experiment areas
D	Releases of Ar-41 from the reactor bay to an unrestricted environment SHALL NOT exceed 100 Ci per year.

3.5.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The reactor bay HVAC confinement ventilation system is not OPERABLE	A.1 ENSURE reactor is SHUTDOWN OR A.2.1 ENSURE auxiliary air purge system is OPEATING AND A.3.b SECURE EXPERIMENT operations if failure could result in significant release of rad. gases or aerosols. A.3.c ENSURE no irradiated fuel handing	A.1 IMMEDIATE A.2.1 IMMEDIATE A.3.a IMMEDIATE A.3.b IMMEDIATE A.3.c IMMEDIATE
B The particulate continuous air monitor is in service and indicates greater than 10,000 cpm, and the reactor bay confinement system is not in CONFINEMENT ISOLATION	B.1 ENSURE reactor is SHUTDOWN AND B.2 SECURE reactor bay ventilation AND B.3 SECURE the fume/sorting hood	B.1 IMMEDIATE B.1 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. The auxiliary purge system is not OPERABLE	C.1 ENSURE reactor bay HVAC confinement ventilation system is OPERATING OR	C.1 IMMEDIATE
	C.2.a ENSURE reactor is SHUTDOWN	C.2.a IMMEDIATE
	C.2.b Secure EXPERIMENT operations for EXPERIMENT with failure modes that could result in the release of radioactive gases or aerosols	C.2.b IMMEDIATE
	C.2.c ENSURE no irradiated fuel handing	C.2.c IMMEDIATE
D Calculated releases of Ar-41 from the reactor bay exhaust plenum exceed 100 Ci per year.	D. Do not operate.	D. IMMEDIATE

3.5.5 Bases

The confinement and ventilation system is described in Chapter 9. Routine operations produce radioactive gas, principally Argon 41, in the reactor bay. If the confinement system is not functioning and the purge system is not operating, radioactive gasses will buildup in the reactor bay. During this interval, experiment activities that might cause airborne radionuclide levels to be elevated are prohibited.

Chapter 13 addresses the maximum hypothetical fission product inventory release. Using unrealistically conservative assumptions, concentrations for a few nuclides of iodine would be in excess of occupational derived air concentrations for a matter of hours or days. ^{90}Sr activity available for release from fuel rods previously used at other facilities is estimated to be at most about 4 times the ALI. In either case (radio-iodine or -Sr), there is no credible scenario for accidental inhalation or ingestion of the undiluted nuclides that might be released from a damaged fuel element. Finally, fuel element failure during a fuel handling accident is likely to be observed and mitigated immediately.

The CAP-88 (Clean Air Act Assessment Package-1988) computer model is a set of computer programs, databases and associated utility programs for estimation of dose and risk from radionuclide emissions to air. CAP-88 is composed of modified versions of AIRDOS-EPA (Mo79) and DARTAB (ORNL5692). CAP-88 was used to analyze argon 41 effluents from the UT TRIGA reactor. Analysis shows 100 Ci per year results in a maximum does to individuals in the effluent plume of 0.142 mrem in a year, well within the 10CFR20 limit of 10 mrem/year for stack effluents.

3.6 Limitations on Experiments

3.6.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.6.2 Objectives

These Limiting Conditions for Operation prevent reactivity excursions that might cause the fuel temperature to exceed the safety limit (with possible resultant damage to the reactor), and the excessive release of radioactive materials in the event of an EXPERIMENT failure

3.6.3 Specifications

A	The reactivity worth of any individual MOVEABLE EXPERIMENT SHALL NOT exceed $1.00
B	The reactivity worth of any individual SECURED EXPERIMENT SHALL NOT exceed $2.50
C	The total reactivity worth of all EXPERIMENTS shall not exceed $3.00

3.6.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MOVEABLE EXPERIMENT worth is greater than $1.00	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Remove the experiment	A.1 IMMEDIATE A.2 Prior to continued operations
B. SECURED EXPERIMENT worth is greater than $2.50	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Remove the experiment	B.1 IMMEDIATE B.2 Prior to continued operations

C. Total EXPERIMENT worth is greater than $3.00	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Remove the experiment	C.1 IMMEDIATE C.2 Prior to continued operations

3.6.5 Bases

Chapter 13 demonstrates that pulsed reactivity worth less than 2.8% Δk/k ($4.00) will not challenge fuel integrity. These limits provide assurance that experiments do not exceed the reactivity analyzed; experiment limits are established lower than analysis limits is used to assure margin for experimental error.

3.7 Fuel Integrity

3.7.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.7.2 Objective

The objective is to prevent the use of damaged fuel in the UT TRIGA reactor.

3.7.3 Specifications

A	Fuel elements in the reactor core SHALL NOT be (1) elongated more than 1/10 in. over manufactured length OR (2) laterally bent more than 1/16 in.
B	Fuel elements SHALL NOT have visual indications of cladding integrity failure.
C	Fuel elements in the core SHALL NOT release fission products.

3.7.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any fuel element is elongated greater than 1/10 in. over manufactured length, or bent laterally greater than 1/16 in.	Do not re-insert the fuel element into the upper core grid plate.	IMMEDIATE
B. Fuel elements have visual indication of cladding integrity failure	Do insert or not re-insert the fuel element into the upper core grid plate.	IMMEDIATE
C. Fission products are determined to be leaking from fuel elements in the core	C.1 SECURE PULSE MODE operations C.2.a Operate in STEADY STATE MODE only to identify the failed element AND C.2.b Remove the failed element from service	C.1 IMMEDIATE C.2.a IMMEDIATE C.2.b When the element is identified

3.7.5 Bases

The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at TRIGA installations. Fuel cladding integrity is important since it represents the only process barrier for fission product release from the TRIGA reactor.

Lateral bend less than 1/16 in. in adjacent fuel elements assures that there is adequate clearance to prevent element contact during operation.

Limiting the use of fuel elements where cladding has been challenged as specified limits release of fission products to the minimum required for assessing fuel elements.

3.8 Reactor Pool Water

3.8.1 Applicability

This specification applies to operations in STEADY STATE MODE, PULSE MODE, and SECURED MODE.

3.8.2 Objective

The objective is to set acceptable limits on the water quality, temperature, conductivity, and level in the reactor pool.

3.8.3 Specifications

A	Water temperature at the exit of the reactor pool SHALL NOT exceed 110°F (48.9°C)
B	Water conductivity SHALL be less than or equal to 5 µmho/cm averaged over 1 month
C	Water level above the core SHALL be at least 6.5 m from bottom of the pool
D	The pressure difference between chilled water outlet from the pool heat exchanger and pool water inlet SHALL not exceed 7 kPa (1 psig)

3.8.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Water temperature at the exit of the reactor pool exceeds 110°F (48.9°C)	A.1 ENSURE the reactor is SHUTDOWN AND A.2 Secure flow through the demineralizer AND A.3 Initiate action to reduce water temperature to less than 110°F	A.1 IMMEDIATE A.2 IMMEDIATE A.3 IMMEDIATE

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Water conductivity is greater than 5 µmho/cm	B.1 ENSURE the reactor is SHUTDOWN AND B.2 Restore conductivity to less than 5 µmho/cm	B.1 IMMEDIATE B.2 Within 1 month
C. Water level above the core SHALL be at least 6.5 m from the bottom of the pool for all operating conditions	C.1 ENSURE the reactor is SHUTDOWN AND C.2 Restore water level	C.1 IMMEDIATE C.2 IMMEDIATE
D. The pressure difference between chilled water outlet from the pool heat exchanger and pool water inlet exceeds 7 kPa (1 psig)	D.1 ENSURE the reactor is SHUTDOWN OR D.2 Verify TRUE VALUE is less than 7 kPa (1 psig) OR D.3 RESTORE pressure difference to less than 7 kPa (1 psig) OR D.4 Isolate chill water	D.1 IMMEDIATE D.2 IMMEDIATE D.3 IMMEDIATE D.4 IMMEDIATE

3.8.5 Bases

The resin used in the mixed bed deionizer limits the water temperature of the reactor pool. Resin in use (as described in Section 5.4) maintains mechanical and chemical integrity at temperatures below 110°F (48.9°C). Therefore, thermal hydraulic analysis was conducted to a maximum pool temperature of 48.9°C, and limiting pool temperature ensures analysis conditions are met.

Maintaining low water conductivity over a prolonged period prevents possible corrosion, deionizer degradation, or slow leakage of fission products from degraded cladding. Although fuel degradation does not occur over short time intervals, long-term

integrity of the fuel is important, and a 4-week interval was selected as an appropriate maximum time for averaging conductivity values.

For normal pool temperature, calculations in Chapter 4 assuming 8.1 and 6.5 m above the bottom of the pool demonstrate that the heat flux of the hottest area of the fuel rod generating the highest power level in the core during operations is less than the critical heat flux by a large margin up to the maximum permitted cooling temperatures; margin remains even at temperatures approaching bulk boiling for atmospheric conditions. Therefore, pool levels greater than 6.5 m above the pool floor meet requirements for safe operation with respect to maximum fuel temperature and thermal hydraulics by a wide margin.

The principle contributor to radiation dose rates at the pool surface is Nitrogen 16 generated in the reactor core and dispersed in the pool. Pool surface radiation dose rates from Nitrogen 16 with 6.5 m of water above the core are acceptable.

Therefore, a minimum pool level of 6.5 feet above the core is adequate to support the core cooling and provide shielding.

The specified pressure difference assures that any postulated heat exchanger leakage will not release potentially contaminated water to the chill water system.

3.9 Retest Requirements

3.9.1 Applicability

This specification applies to operations in STEADY STATE MODE and PULSE MODE.

3.9.2 Objective

The objective is to ensure Technical Specification requirements are met following maintenance or operational activities that occur within surveillance test intervals.

3.9.3 Specifications

Maintenance or operational activities SHALL NOT change, defeat or alter equipment or systems in a way that prevents the systems or equipment from being OPERABLE or otherwise prevent the systems or equipment from fulfilling the safety basis

3.9.4 Actions

CONDITION	REQUIRED ACTION	COMPLETION TIME
Maintenance or an operational activity is performed that has the potential to change a setpoint, calibration, flow rate, or other parameter that is measured or verified in meeting a surveillance or operability requirement	Perform surveillance OR Operate only to perform retest	Prior to continued, normal operation in STEADY STATE MODE or PULSE MODE

3.9.5 Bases

Operation of the UT TRIGA reactor will comply with the requirements of Technical Specifications. This specification ensures that if maintenance might challenge a Technical Specifications requirement, the requirement is verified prior to resumption of normal operations.

4. Surveillance Requirements

4.1 Core Reactivity

4.1.1 Objective

This surveillance ensures that the minimum SHUTDOWN MARGIN requirements and maximum excess reactivity limits of section 3.1 are met.

4.1.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SHUTDOWN MARGIN Determination	ANNUAL
EXCESS REACTIVITY Determination	ANNUAL Following Insertion of experiments with measurable positive reactivity
Control Rod Reactivity Worth determination	BIENNIAL

4.1.3 Basis

Experience has shown verification of the minimum allowed SHUTDOWN MARGIN at the specified frequency is adequate to assure that the limiting safety system setting is met

When core reactivity parameters are affected by operations or maintenance, additional activity is required to ensure changes are incorporated in reactivity evaluations.

Reactivity limits are verified by comparing critical control rod positions to reference values. The reference values change with burnup and core configuration. Biennial evaluation of control rod position is adequate, although other activities may result in control rod worth determination through retest requirements.

4.2 PULSE MODE

4.2.1 Objectives

The verification that the pulse rod position does not exceed a reactivity value corresponding to $4.00 assures that the limiting condition for operation is met.

4.2.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
ENSURE Transient Pulse Rod position corresponds to reactivity not greater than $4.00	Prior to pulsing operations

4.2.3 Basis

Verifying pulse rod position corresponds to less than or equal to $4.00 ensures that the maximum pulsed reactivity meets the limiting condition for operation.

4.3 MEASURING CHANNELS

4.3.1 Objectives

Surveillances on MEASURING CHANNELS at specified frequencies ensure instrument problems are identified and corrected before they can affect operations.

4.3.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Reactor power level MEASURING CHANNEL	
CHANNEL TEST	DAILY
Calorimetric calibration	ANNUAL
CHANNEL CHECK loss of high voltage to required power level instruments	DAILY
CALIBRATION high voltage to required power level instruments	ANNUAL
Primary pool water temperature CHANNEL CALIBRATION	ANNUAL
Fuel temperature CHANNEL CALIBRATION	ANNUAL
Upper level Area radiation monitor	
CHANNEL CHECK	MONTHLY
CHANNEL CALIBRATION	ANNUAL
Lower or middle level Area Radiation Monitor	
CHANNEL CHECK	MONTHLY
CHANNEL CALIBRATION	ANNUAL
Continuous Air Radiation Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION	ANNUAL
Argon Monitor	
CHANNEL CHECK	DAILY
CHANNEL CALIBRATION (Electronic)	SEMIANNUAL
Continuous Particulate Air Monitor	
CHANNEL CHECK	MONTHLY
CHANNEL CALIBRATION	ANNUAL

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Startup Count Rate	DAILY

4.3.3 Basis

The DAILY CHANNEL CHECKS will ensure that the SAFETY SYSTEM and MEASURING CHANNELS are operable. The required periodic calibrations and verifications will permit any long-term drift of the channels to be corrected.

4.4 Safety Channel and Control Rod Operability

4.4.1 Objective

The objectives of these surveillance requirements are to ensure the REACTOR SAFETY SYSTEM will function as required. Surveillances related to safety system MEASURING CHANNELS ensure appropriate signals are reliably transmitted to the shutdown system; the surveillances in this section ensure the control rod system is capable of providing the necessary actions to respond to these signals.

4.4.2 Specifications

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Manual scram SHALL be tested by releasing partially withdrawn CONTROL RODS (STANDARD)	DAILY
CONTROL ROD (STANDARD) drop times SHALL be measured to have a drop time from the fully withdrawn position of less than 1 sec.	ANNUAL
The control rods SHALL be visually inspected for corrosion and mechanical damage at intervals	BIENNIAL
CONTROL ROD (STANDARD) position interlock functional test	SEMIANNUAL
Pulse rod interlock functional test	SEMIANNUAL
The CONTROL ROD (TRANSIENT) rod drive cylinder and the associated air supply system SHALL be inspected, cleaned, and lubricated, as necessary.	ANNUAL

4.4.3 Basis

Manual and automatic scrams are not credited in accident analysis, although the systems function to assure long-term safe shutdown conditions. The manual scram and control rod drop timing surveillances are intended to monitor for potential degradation that might interfere with the operation of the control rod systems. The functional test of loss of high voltage to the power level monitoring channels assures that the safety channels will function on demand.

The control rod inspections (visual inspections and transient drive system inspections) are similarly intended to identify potential degradation that lead to control rod degradation or inoperability.

A test of the interlock that prevents the pulse rod from coupling to the drive in the state mode unless the drive is fully down or square wave mode is being used assures that pulses will not unintentionally occur. In particular, instrumentation alignment for the pulsing mode causes safety channels to be capable of monitoring pulse power; if pulsing occurs while the instruments are set to normal, steady state operations, they will not be capable of monitoring peak power.

A test of the interlock that prevents standard control rod motion while in the pulse mode assures that the interlock will function as required.

The functional checks of the control rod drive system assure the control rod drive system operates as intended for any pulsing operations. The inspection of the pulse rod mechanism will assure degradation of the pulse rod drive will be detected prior to malfunctions.

4.5 Gaseous Effluent Control

4.5.1 Objectives

These surveillances ensure that routine releases are normal, and (in conjunction with MEASURING CHANNEL surveillances) that instruments will alert the facility if conditions indicate abnormal releases.

4.5.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
CONFINEMENT ISOLATION functional test	MONTHLY
CONFINEMENT ISOLATION damper inspection	ANNUALLY
Calculate Ar41 discharge	SEMIANNUALLY

4.5.3 Basis

Confinement isolation functional test frequency is adequate to ensure potential failures are detected prior to system demand.

The annual test is adequate to detect degradation of sealing surfaces.

Semiannual calculation of Argon 41 is adequate to ensure that discharge limits are met.

4.6 Limitations on Experiments

4.6.1 Objectives

This surveillance ensures that experiments do not have significant negative impact on safety of the public, personnel or the facility.

4.6.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Experiments SHALL be evaluated and approved prior to implementation.	Prior to inserting a new experiment for purposes other than determination of reactivity worth
Measure and record experiment worth of the EXPERIMENT (where the absolute value of the estimated worth is greater than $0.50).	Initial insertion of a new experiment where absolute value of the estimated worth is greater than $0.50

4.6.3 Basis

These surveillances support determination that the limits of 3.6 are met.

Experiments with an absolute value of the estimated significant reactivity worth (greater than $0.40) will be measured to assure that maximum experiment reactivity worths are met. If an absolute value of the estimate indicates less than $0.50 reactivity worth, any error less than 100% will result in actual reactivity less than the assumptions used in analysis for inadvertent pulsing at low power operations in the Safety Analysis Report (13.2.3, Case I).

4.7 Fuel Integrity

4.7.1 Objective

The objective is to ensure that the dimensions of the fuel elements remain within acceptable limits.

4.7.2 Applicability

This specification applies to the surveillance requirements for the fuel elements in the reactor core.

4.7.3 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
The STANDARD FUEL ELEMENTS SHALL be visually inspected for corrosion and mechanical damage, and measured for length and bend	500 pulses of magnitude equal to or greater than a pulse insertion of $3.00 AND Following the exceeding of a limited safety system set point with potential for causing degradation
Approximately 1/4 of the core SHALL be visually inspected annually for corrosion and mechanical damage such that	BIENNIAL
Full core inspection complete	4, not to exceed 5, years

4.7.4 Basis

The most severe stresses induced in the fuel elements result from pulse operation of the reactor, when fuel to cladding differential expansion occurs and gas pressure. The magnitude of $3.00 pulses warrants inspection following a sufficient number of cycles.

Visual inspection of fuel elements at the specified intervals combined with measurements at intervals determined by pulsing as described is considered adequate to identify potential degradation of fuel prior to catastrophic fuel element failure.

4.8 Reactor Pool Water

This specification applies to the water contained in the UT TRIGA reactor pool.

4.8.1 Objective

The objective is to provide surveillance of reactor primary coolant water quality, pool level, temperature and (in conjunction with MEASURING CHANNEL surveillances), and conductivity.

4.8.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify reactor pool water level above the inlet line vacuum breaker	DAILY
Verify reactor pool water temperature channel operable	DAILY
Measure reactor Pool water conductivity	WEEKLY
	At least every 30 days

4.9.3 Bases

Surveillance of the reactor pool will ensure that the water level is adequate before reactor operation. Evaporation occurs over longer periods of time, and daily checks are adequate to identify the need for water replacement. Pool water level status (not high, not low) is indicated on the control console.

Water temperature must be monitored to ensure that the temperature limit related to resin will not be exceeded, and that the conditions for analysis are maintained.. A daily check on the instrument prior to reactor operation is adequate to ensure the instrument is operable when it will be needed.

Water conductivity must be checked to ensure that the pool cleanup system is performing properly and to detect any increase in water impurities. A weekly check is adequate to verify water quality is appropriate and also to provide data useful in trend analysis. If the reactor is not operated for long periods of time, the requirement for checks at least every 30 days ensures water quality is maintained in a manner that does not permit fuel degradation.

4.9 Retest Requirements

4.9.1 Objective

The objective is to ensure that a system is OPERABLE within specified limits before being used after maintenance or operational activities has been performed.

4.9.2 Specification

SURVIELLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Evaluate potential for maintenance or operational activities to affect operability and function of equipment required by Technical Specifications; for standard procedures, this evaluation is incorporated in instructions.	Following maintenance or operational activities for systems of equipment required by Technical Specifications
Perform surveillance to assure affected function meets requirements	Prior to resumption of normal operations

4.9.3 Bases

This specification ensures that work on systems or components has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

5. Design Features

5.1 Reactor Fuel

5.1.1 Applicability

This specification applies to the fuel elements used in the reactor core.

5.1.2 Objective

The objective is to ensure 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 mechanical integrity.

5.1.3 Specification

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

(2) For the fuel loading process, elements shall be loaded in a close packed array except for experimental facilities or for single positions occupied by control rods and a neutron startup source.

5.1.4 Bases

These types of fuel elements have a long history of successful use in TRIGA reactors.

5.2 Reactor Fuel and Fueled Devices in Storage

5.2.1 Applicability

This specification applies to reactor fuel elements in storage

5.2.2 Objective

The objective is to ensure fuel elements or fueled devices in storage are maintained Subcritical in a safe condition.

5.2.3 Specification

(1) All fuel elements or fueled devices shall be in a safe, stable geometry;

(2) The k_{eff} of all fuel elements or fueled devices in storage is less than 0.9;

(3) The k_{eff} of fuel elements or fueled devices in an approved shipping container will meet the applicable Certificate of Compliance specifications for k_{eff};

(4) Irradiated fuel elements or fueled devices will be stored in an array which will permit sufficient natural convection cooling by air or water such that the fuel element or fueled device will not exceed design values.

5.2.4 Bases

This specification is based on American Nuclear Society standard 15.1, section 5.4.

5.3 Reactor Building

5.3.1 Applicability

This specification applies to the building that houses the TRIGA reactor facility.

5.3.2 Objective

The objective is to ensure that provisions are made to restrict the amount of release of radioactivity into the environment.

5.3.3 Specification

(I) The reactor shall be housed in a closed room designed to restrict leakage when the reactor is in operation, with HVAC system designed to maintain negative differential pressure with respect to adjacent spaces and the environment.

(2) The minimum free volume of the reactor room shall be approximately 4120 m^3.

(3) The reactor bay HVAC confinement ventilation system is capable of exhausting air or other gases from the reactor room at a minimum of 60 ft. above ground level.

(4) Reactor bay HVAC confinement ventilation system operation is designed to provide a minimum of 2 changes of reactor bay air per hour.

5.3.4 Bases

To control the escape of gaseous effluent, the reactor room contains no windows that can be opened. The room air is exhausted through an independent exhaust system, and discharged above the roof to provide dilution.

5.4 Experiments

5.4.1 Applicability

This specification applies to the design of experiments.

5.4.2 Objective

The objective is to ensure that experiments are designed to meet criteria.

5.4.3 Specifications

(1) EXPERIMENTS with design reactivity worth greater than $1.00 SHALL be securely fastened (as defined in Section I, Secured Experiment).

(2) Design shall ensure that failure of an EXPERIMENT SHALL NOT lead to a direct failure of a fuel element or of other experiments that could result in a measurable increase in reactivity or a measurable release of radioactivity due to the associated failure.

(3) EXPERIMENTS SHALL be designed so that they do not cause bulk boiling of core water

(4) EXPERIMENT design SHALL ensure no interference with control rods or shadowing of reactor control instrumentation.

(5) EXPERIMENT design shall minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials.

(6) Where the possibility exists that the failure of an EXPERIMENT (except fueled EXPERIMENTS) could release radioactive gases or aerosols to the reactor bay or atmosphere, the quantity and type of material shall be limited such that

the airborne concentration of radioactivity is less than 1,000 times the Derived Air Concentration.

For in-core samples a decay time of five minutes following irradiation may be used in radioactive inventory calculations to account for processing prior to potential exposure.

(7) Each fueled experiment shall be limited such that the total inventory of (1) radioactive iodine isotopes 131 through 135 in the experiment is not greater than 9.32E5 μCi, and (2) radioactive strontium is not greater than 9.35E4 μCi.

Alternate calculations may be accomplished to demonstrate equivalent times for protective actions based on DAC limits for specific experiments, if desired.

These limits do not apply to TRIGA fuel elements used in experiments as maximum hypothetical accident analysis applies. For in-core samples a decay time of five minutes following irradiation to account may be used in calculations.

(8) The following assumptions shall be used in experiment design:

a. If effluents from an experimental facility exhaust through a hold-up tank which closes automatically at a high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

b. If effluents from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the aerosols produced will escape.

c. For materials whose boiling point is above 130ºF and where vapors formed by boiling this material could escape only through an undisturbed column of water above the core, at least 10% of these vapors will escape.

(9) Use of explosive solid or liquid material with a National Fire Protection Association Reactivity (Stability) index of 2, 3, or 4 in the reactor pool or biological shielding SHALL NOT exceed the equivalent of 25 milligrams of TNT without prior NRC approval.

5.4.4 Basis

Designing the experiment to reactivity and thermal-hydraulic conditions ensures that the experiment is not capable of breaching fission product barriers or interfering with the control systems (interferences from other - than reactivity - effects with the control and safety systems are also prohibited). Design constraints on industrial hazards

ensure personnel safety and continuity of operations. Design constraints limiting the release of radioactive gasses prevent unacceptable personnel exposure during off-normal experiment conditions.

A Derived Air Concentration assumes a 2000 hour per year exposure; if exposure is controlled to a specific time limit, such as time required for recognizing the situation and evacuating, limiting values for an experiment can be higher than a DAC.

Limits on radioiodine and radioactive strontium in fueled experiments permits a 1 hour evacuation time for releases of radioiodine and a 2-hour evacuation time for releases of radioactive strontium based on a TRIGA fuel distribution of the radioisotopes from fission of ^{235}U.

6. Administrative Controls

6.1 Organization and Responsibilities of Personnel

This chapter describes and discusses the Conduct of Operations at the University of Texas TRIGA. 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. License is used in Chapter 12 in reference to reactor operators and senior reactors subject to 10CFR50.55 requirements.

a) Structure

University Administration

Fig. 1 illustrates the organizational structure that is applied to the management and operation of the University of Texas and the reactor facility. Responsibility for the safe operation of the reactor facility is a function of the management structure of Fig. 1[1]. These responsibilities include safeguarding the public and staff from undue radiation exposures and adherence to license or other operation constraints. Functional organization separates the responsibilities of academic functions and business functions. The office of the President administers these activities and other activities through several vice presidents.

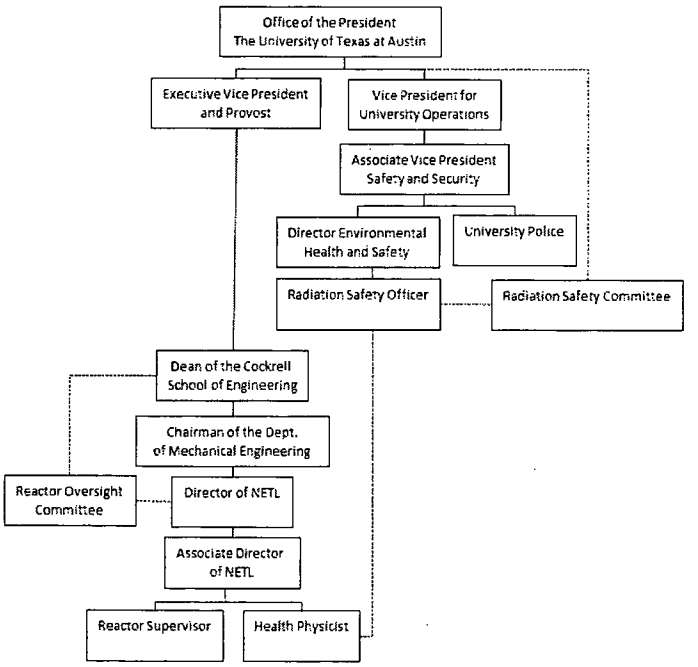


Figure 6.1, Organzational Structure

[1] "Standard for Administrative Controls" ANSI/ANS - 15.18 1979

NETL Facility Administration

The facility administrative structure is shown in Fig. 2. Facility operation staff is an organization of a director and at least four full time equivalent persons. This staff of four provides for basic operation requirements. Four typical staff positions consist of an associate director, a reactor supervisor, a reactor operator, and a health physicist. One or more of the listed positions may also include duties typical of a research scientist. The reactor supervisor, health physicist, and one other position are to be full time. One full time equivalent position may consist of several part-time persons such as assistants, technicians and secretaries. Faculty, students, and researchers supplement the organization. Titles for staff positions are descriptive and may vary from actual designations. Descriptions of key components of the organization follow.

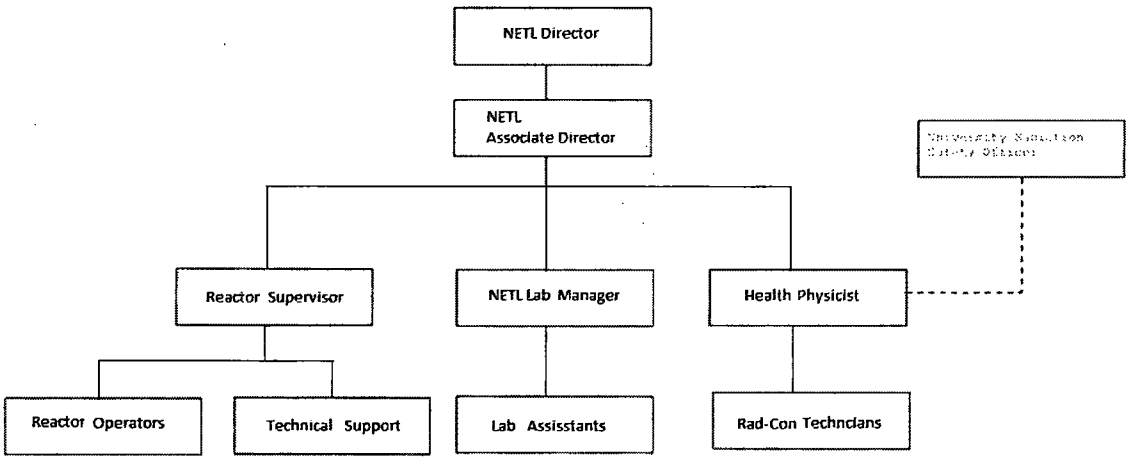


Figure 2, NETL Facility Administration

b) Functional Responsibility

Vice President and Provost

Research and academic educational programs are administered through the Office of the Executive Vice President and Provost. Separate officers assist with the administration of research activities and academic affairs with functions delegated to the Dean of the College of Engineering and Chairman of the Mechanical Engineering Department.

Vice President for University Operations

University operations activities are administered through the Office of the Vice President for Operations. This office is responsible for multiple operational functions of the University including university support programs, human resources, campus safety and security, campus real estate, and campus planning and facilities management.

Associate Vice President Campus Safety and Security

The associate vice president for campus safety and security oversees multiple aspects of safety and security on campus including environmental health and safety, campus police, parking and transportation, fire prevention, and emergency preparedness.

Director of Nuclear Engineering Teaching Laboratory

Nuclear Engineering Teaching Laboratory programs are directed by a senior classified staff member or faculty member. The director oversees strategic guidance of the Nuclear Engineering Teaching Laboratory including aspects of facility operations, research, and service work. The director must interact with senior University of Texas at Austin management regarding issues related to the Nuclear Engineering Teaching Laboratory.

Associate Director of Nuclear Engineering Laboratory

The Associate Director performs the day to day duties of directing the activities of the facility. The Associate Director is knowledgeable of regulatory requirements, license conditions, and standard operating practices. The associate director will also be involved in soliciting and carrying out research utilizing the reactor and other specialized equipment at the Nuclear Engineering Teaching Laboratory.

Reactor Oversight Committee

The Reactor Oversight Committee is established through the Office of the Dean of the College of Engineering of The University of Texas at Austin. Broad responsibilities of the committee include the evaluation, review, and approval of facility standards for safe operation.

The Dean shall appoint at least three members to the Committee that represent a broad spectrum of expertise appropriate to reactor technology. The committee will meet at least twice each calendar year or more frequently as circumstances warrant. The Reactor Oversight Committee shall be consulted by the Nuclear Engineering Teaching Laboratory concerning unusual or exceptional actions that affect administration of the reactor program.

Radiation Safety Officer

A Radiation Safety Officer acts as the delegated authority of the Radiation Safety Committee in the daily implementation of policies and practices regarding the safe use of radioisotopes and sources of radiation as determined by the Radiation Safety Committee. The Radiation Safety Program is administered through the University

Environmental Health and Safety division. The responsibilities of the Radiation Safety Officer are outlined in The University of Texas at Austin Manual of Radiation Safety.

Radiation Safety Committee

The Radiation Safety Committee is established through the Office of the President of The University of Texas at Austin. Responsibilities of the committee are broad and include all policies and practices regarding the license, purchase, shipment, use, monitoring, disposal, and transfer of radioisotopes or sources of ionizing radiation at The University of Texas at Austin.

The President shall appoint at least three members to the Committee and appoint one as Chairperson. The Committee will meet at least once each year on a called basis or as required to approve formally applications to use radioactive materials. The Radiation Safety Committee shall be consulted by the University Safety Office concerning any unusual or exceptional action that affects the administration of the Radiation Safety Program.

Reactor Supervisor

Reactor operation at the Nuclear Engineering Teaching Laboratory is directed by a Reactor Supervisor. Responsibilities of the Reactor Supervisor include control of license documentation, reactor operation, equipment maintenance, experiment operation, and instruction of persons with access to laboratory areas.

Activities of reactor operators with USNRC licenses will be subject to the direction of a person with a USNRC senior operator license. The Reactor Supervisor shall be qualified as a senior operator. This person is to be knowledgeable of regulatory requirements, license conditions, and standard operating practices.

Health Physicist

Radiological safety of the Nuclear Engineering Teaching Laboratory is monitored by a health physicist, who will be knowledgeable of the facility radiological hazards. Responsibilities of the health physicist will include calibration of radiation detection instruments, measurements of radiation levels, control of radioactive contamination, maintenance of radiation records, and assistance with other facility monitoring activities.

Activities of the health physicist will depend on two conditions. One condition will be the normal operation responsibilities determined by the director of the facility. A second condition will be communications specified by the radiation safety officer. This combination of responsibility and communication provides for safety program implementation by the director, but establishes independent review. The health

physicist's activities will meet the requirements of the director and the policies of an independent university safety organization.

Laboratory Manager

Laboratory operations and research support is provided by a designated Laboratory Manager. The function is typically combined with the Health Physicist position.

Reactor Operators

Reactor operators (and senior reactor operators) are licensed by the USNRC to operate the UT TREIGA II nuclear research reactor. University staff and/or students may be employed as reactor operators.

Technical Support

Staff positions supporting various aspects of facility operations are assigned as required.

Radiological Controls Technicians

Radiological Controls Technicians are supervised by the Health Physicist to perform radiological controls and monitoring functions. Radiological Controls Technicians are generally supported as Undergraduate Research Assistant positions.

Laboratory Assistants

Laboratory Assistants are supervised by the Laboratory Manager to perform laboratory operations and analysis. Laboratory Assistants are generally supported as Undergraduate Research Assistant positions.

c) Staffing

Operation of the reactor and activities associated with the reactor, control system, instrument system, radiation monitoring system, and engineered safety features will be the function of staff personnel with the appropriate training and certification[2].

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as Reactor Supervisor. The Supervisor may be on call if capable of arriving at the facility within thirty minutes and cognizant of reactor operations. The Reactor Supervisor shall directly supervise:

[2] "Selection and Training of Personnel for Research Reactors", ANSI/ANS -15.4 - 1970 (N380)

a. All fuel element or control rod relocations or installations within the reactor core region, and subsequent INITIAL STARTUP and approach to power.

b. Relocation or installation of any experiment in the core region with a reactivity worth of greater than one dollar, and subsequent INITIAL STARTUP and approach to power.

c. Recovery from an unscheduled shutdown or significant power reductions,

d. Any INITIAL STARTUP and approach to power following modifications to reactor safety or control rod drive systems.

Whenever the reactor is not secured, a (USNRC licensed) Reactor Operator (or Senior Reactor Operator) who meets requirements of the Operator Requalification Program shall be at the reactor control console, and directly responsible for control manipulations. All activities that require the presence of licensed operators will also require the presence in the facility complex of a second person capable of performing prescribed written instructions.

Only the Reactor Operator at the controls or personnel authorized by, and under direct supervision of, the Reactor Operator at the controls shall manipulate the controls. Whenever the reactor is not secured, operation of equipment that has the potential to affect reactivity or power level shall be manipulated only with the knowledge and consent of the Reactor Operator at the controls. The Reactor Operator at the controls may authorize persons to manipulate reactivity controls who are training either as (1) a student enrolled in academic or industry course making use of the reactor, (2) to qualify for an operator license, or (3) in accordance the approved Reactor Operator requalification program.

Whenever the reactor is not secured, a second person (i.e., in addition to the reactor operator at the control console) capable of initiating the Reactor Emergency Plan will be present in the NETL building. Unexpected absence of this second person for greater than two hours will be acceptable if immediate action is taken to obtain a replacement.

Staffing required for performing experiments with the reactor will be determined by a classification system specified for the experiments. Requirements will range from the presence of a certified operator for some routine experiments to the presence of a senior operator and the experimenter for other less routine experiments.

6.2 Review and Audit

The review and audit process is the responsibility of the Reactor Oversight Committee (ROC).

Composition and Qualifications

The ROC shall consist of at least three (3) members appointed by the Dean of the College of Engineering that are knowledgeable in fields which relate to nuclear safety. The university radiological safety officer shall be a member or an ex-officio member. The committee will perform the functions of review and audit or designate a knowledgeable person for audit functions.

Charter and Rules

The operations of the ROC shall be in accordance with an established charter, including provisions for:

a. Meeting frequency (at least twice each year, with approximately 4-8 month frequency).
b. Quorums (not less than one-half the membership where the operating staff does not contribute a majority).
c. Dissemination, review, and approval of minutes.
d. Use of subgroups.

Review Function

The responsibilities of the Reactor Safeguards Committee to shall include but are not limited to review of the following:

a. All new procedures (and major revisions of procedures) with safety significance
b. Proposed changes or modifications to reactor facility equipment, or systems having safety significance
c. Proposed new (or revised) experiments, or classes of experiments, that could affect reactivity or result in the release of radioactivity
d. Determination of whether items a) through c) involve unreviewed safety questions, changes in the facility as designed, or changes in Technical Specifications.
e. Violations of Technical Specifications or the facility operating licensee
f. Violations of internal procedures or instruction having safety significance
g. Reportable occurrences
h. Audit reports

Minor changes to procedures and experiments that do not change the intent and do not significantly increase the potential consequences may be accomplished following review and approval by a senior reactor operator and independently by one of the Reactor Supervisor, Associate Director or Director. These changes should be reviewed at the next scheduled meeting of the Reactor Oversight Committee.

Audit Function

The audit function shall be a selected examination of operating records, logs, or other documents. Audits will be by a Reactor Oversight Committee member or by an individual appointed by the committee to perform the audit. The audit should be by any individual not directly responsible for the records and may include discussions with cognizant personnel or observation of operations. The following items shall be audited and a report made within 3 months to the Director and Reactor Committee:

a. Conformance of facility operations with license and technical specifications at least once each calendar year.

b. Results of actions to correct deficiencies that may occur in reactor facility equipment, structures, systems, or methods of operation that affect safety at least once per calendar year.

c. Function of the retraining and requalification program for reactor operators at least once every other calendar year.

d. The reactor facility emergency plan and physical security plan, and implementing procedures at least once every other year.

6.3 Procedures

Written procedures shall govern many of the activities associated with reactor operation. Activities subject to written procedures will include:

a. Startup, operation, and shutdown of the reactor
b. Fuel loading, unloading, and movement within the reactor.
c. Control rod removal or replacement.
d. Routine maintenance, testing, and calibration of control rod drives and other systems that could have an effect on reactor safety.
e. Administrative controls for operations, maintenance, conduct of experiments, and conduct of tours of the Reactor Facility.
f. Implementing procedures for the Emergency Plan or Physical Security Plan.

Written procedures shall also govern:

a. Personnel radiation protection, in accordance with the Radiation Protection Program as indicated in Chapter 11
b. Administrative controls for operations and maintenance
c. Administrative controls for the conduct of irradiations and experiments that could affect core safety or reactivity

A master Procedure Control procedure specifies the process for creating, changing, editing, and distributing procedures. Preparation of the procedures and minor modifications of the procedures will be by certified operators. Substantive changes or major modifications to procedures, and new prepared procedures will be submitted to the Reactor Oversight Committee for review and approval. Temporary deviations from the procedures may be made by the reactor supervisor or designated senior operator provided changes of substance are reported for review and approval.

Proposed experiments will be submitted to the reactor oversight committee for review and approval of the experiment and its safety analysis[3], as indicated in Chapter 10. Substantive changes to approved experiments will require re-approval while insignificant changes that do not alter experiment safety may be approved by a senior operator and independently one of the following, Reactor Supervisor, Associate Director, or Director. Experiments will be approved first as proposed experiments for one time application, and subsequently, as approved experiments for repeated applications following a review of the results and experience of the initial experiment implementation.

6.4 Review of Proposals for Experiments

a) All proposals for new experiments involving the reactor shall be reviewed with respect to safety in accordance with the procedures in (b) below and on the basis of criteria in (c) below.

b) Procedures:

1. Proposed reactor operations by an experimenter are reviewed by the Reactor Supervisor, who may determine that the operation is described by a previously approved EXPERIMENT or procedure. If the Reactor Supervisor determines that the proposed operation has not been approved by the Reactor Oversight Committee, the experimenter shall describe the proposed EXPERIMENT in written form in sufficient detail for consideration of safety aspects. If potentially hazardous operations are involved, proposed procedures and safety measures including protective and monitoring equipment shall be described.

2. The scope of the EXPERIMENT and the procedures and safety measures as described in the approved proposal, Including any amendments or conditions added by those reviewing and approving it, shall be binding on the experimenter and the OPERATING personnel. Minor deviations shall be allowed only in the manner described in Section 6 above. Recorded affirmative votes on proposed new or revised experiments or procedures

[3] ANSI/ANS 15.6, op. cit.

must indicated that the Committee determines that the proposed actions do not involve changes in the facility as designed, changes in Technical Specifications, changes that under the guidance of 10 CFR 50.59 require prior approval of the NRC, and could be taken without endangering the health and safety of workers or the public or constituting a significant hazard to the integrity of the reactor core.

3. Transmission to the Reactor Supervisor for scheduling.

c) Criteria that shall be met before approval can be granted shall include:

1. The EXPERIMENT must meet the applicable Limiting Conditions for Operation and Design Description specifications.

2. It must not involve violation of any condition of the facility license or of Federal, State, University, or Facility regulations and procedures.

3. The conduct of tests or experiments not described in the safety analysis report (as updated) must be evaluated in accordance with 10 CFR 50.59 to determine if the test or experiment can be accomplished without obtaining prior NRC approval via license amendment pursuant to 10 CFR Sec. 50.90.

4. In the safety review the basic criterion is that there shall be no hazard to the reactor, personnel or public. The review SHALL determine that there is reasonable assurance that the experiment can be performed with no significant risk to the safety of the reactor, personnel or the public.

6.5 Operator Requalification

An NRC approved UT TRIGA Requalification Plan is in place to maintain training and qualification of reactor operators and senior reactor operators. License qualification by written and operating test, and license issuance or removal, are the responsibility of the U.S. Nuclear Regulatory Commission. No rights of the license may be assigned or otherwise transferred and the licensee is subject to and shall observe all rules, regulations and orders of the Commission. Requalification training maintains the skills and knowledge of operators and senior operators during the period of the license. Training also provides for the initial license qualification.

6.6 Emergency Plan and Procedures

An NRC approved Emergency Plan following the general guidance set forth in ANSI/ ANS15.16, Emergency Planning for Research Reactors is in place. The plan specifies two action levels, the first level being a locally defined Non-Reactor Specific Event, and the second level being the lowest level FEMA classification, a Notification of Unusual Event.

Procedures reviewed and approved by the Reactor Oversight Committee are established to manage implementation of emergency response.

6.7 Physical Security Plan

An NRC approved Security Plan Security Plan is in place. The plan incorporates compensatory measures implemented following security posture changes initiated post 9/11. The Plan and portions of the procedures are classified as Safeguards Information. Security procedures implementing the plan, approved by the Reactor Oversight Committee, are established.

6.8 Action To Be Taken In The Event A Safety Limit Is Exceeded

In the event that a Safety Limit is not met,

a. The reactor shall be shutdown and secured.
b. The Reactor Supervisor, Associate Director, and Director shall be notified
c. The safety limit violation shall be reported to the Nuclear Regulatory Commission within 24 hours by telephone, confirmed via written statement by email, fax or telegraph
d. A safety limit violation report shall be prepared within 14 days of the event to describe:
    1. Applicable circumstances leading to the violation including (where known) cause and contributing factors
    2. Effect of the violation on reactor facility components, systems, and structures
    3. Effect of the violation on the health and safety of the personnel and the public
    4. Corrective action taken to prevent recurrence
e. The Reactor Oversight Committee shall review the report and any followup reports
f. The report and any followup reports shall be submitted to the Nuclear Regulatory Commission.
g. Operations shall not resume until the USNRC approves resumption.

6.9 Action To Be Taken In The Event Of A Reportable Occurrence

a) A reportable occurrence is any of the following conditions:

1. Any actual safety system setting less conservative than specified in Section 2.2, Limiting Safety System Settings;

2. VIOLATION OF SL, LSSS OR LCO;

NOTES

Violation of an LSSS or LCO occurs through failure to comply with an "Action" statement when "Specification" is not met; failure to comply with the "Specification" is not by itself a violation.

Surveillance Requirements must be met for all equipment/components/conditions to be considered operable.

Failure to perform a surveillance within the required time interval or failure of a surveillance test shall result in the /component/condition being inoperable

3. Incidents or conditions that prevented or could have prevented the performance of the intended safety functions of an engineered safety feature or the REACTOR SAFETY SYSTEM;

4. Release of fission products from the fuel that cause airborne contamination levels in the reactor bay to exceed 10CFR20 limits for releases to unrestricted areas;

5. An uncontrolled or unanticipated change in reactivity greater than $1.00;

6. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy has caused the existence or development of an unsafe condition in connection with the operation of the reactor;

b) In the event of a reportable occurrence, as defined in the Technical Specifications, and in addition to the reporting requirements,

1. The Reactor Supervisor, the Associate Director and the Director shall be notified
2. If a reactor shutdown is required, resumption of normal operations shall be authorized by the Associate Director or Director
3. The event shall be reviewed by the Reactor Oversight Committee during a normally scheduled meeting

6.10 Plant Operating Records

Records of the following activities shall be maintained and retained for the periods specified below[4]. The records may be in the form of logs, data sheets, electronic files, or

[4] "Records and Reports for Research Reactors", ANSI/ANS - 15.3-1974 (N399).

other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

Lifetime Records

Lifetime records are records to be retained for the lifetime of the reactor facility. (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

a. Gaseous and liquid radioactive effluents released to the environs.

b. Offsite environmental monitoring surveys required by Technical Specifications.

c. Events that impact or effect decommissioning of the facility.

d. Radiation exposure for all personnel monitored.

e. Updated drawings of the reactor facility.

Five Year Period

Records to be retained for a period of at least five years or for the life of the component involved whichever is shorter.

a. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least one year).

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Reactor facility radiation and contamination surveys where required by applicable regulations.

f. Experiments performed with the reactor.

g. Fuel inventories, receipts, and shipments.

h. Approved changes in operating procedures.

i. Records of meeting and audit reports of the review and audit group.

One Training Cycle

Training records to be retained for at least one license cycle are the requalification records of licensed operations personnel. Records of the most recent complete cycle shall be maintained at all times the individual is employed.

6.11 Reporting Requirements

This section describes the reports required to NRC, including report content, timing of reports, and report format. Refer to section 12.4 above for the reporting requirements for safety limit violations, radioactivity releases above allowable limits, and reportable occurrences. All written reports shall be sent within prescribed intervals to the United States Nuclear Regulatory Commission, Washington, D.C., 20555, Attn: Document Control Desk.

Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to licensing authorities within three months following the end of each prescribed year. Each annual operating report shall include the following information:

a. A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.

b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests or experiments, or both, that are significantly different from those performed previously, including conclusions that no new or unanalyzed safety questions were identified.

e. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.

f. A summarized result of environmental surveys performed outside the facility.

g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

Other or Special Reports

There shall be a report not later than the following working day by telephone and confirmed in writing by facsimile or similar conveyance of any reportable occurrence identified in 6.9.

There shall be a written report describing the circumstances of any reportable occurrence identified in 6.9 within 14 days of occurrence.

There shall be a written report within 30 days of:

a. Permanent changes in the facility organization involving Director or Supervisor.

b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.