UNIVERSITY OF TEXAS AT AUSTIN
NUCLEAR ENGINEERING TEACHING LABORATORY
TRIGA RESEARCH REACTOR
LICENSE NO. R-129
DOCKET NO. 50-602

UPDATED SAFETY ANALYSIS REPORT
IN SUPPORT OF THE
LICENSE RENEWAL APPLICATION
AUGUST 4, 2023

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

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Safety Analysis Report

Supporting the Renewal of the Facility Operating License for the University of Texas at Austin TRIGA Nuclear Reactor

License No. R-129

Docket No. 5-602

August 2023

CONTENTS

1. THE FACILITY

This report describes the research reactor operated by the University of Texas at Austin (UT-Austin). This report provides the basis for a safety evaluation demonstrating that 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

UT-Austin operates a 1.1 MW TRIGA II research reactor (with pulsing to a maximum permitted reactivity addition of 2.2% $\Delta k/k$) at the J.J. Pickle Research Campus (PRC), approximately ten 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 are reasonable assurance operations for an additional 20-year term that does 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 organized research unit of the UT-Austin Walker Department of Mechanical Engineering in the Cockrell School of Engineering. The NETL serves a multipurpose role, with the primary function as a “user facility” for faculty, staff, and students from 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 including Historically Black College and Universities as well as other Minority Serving Institutions. 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 I on the UT-Austin main campus. Space considerations on the main campus and a well-established infrastructure at the PRC campus led to facility sitting.

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 that provides a large margin to thermal limits and complex emergency planning.

Heat generation in TRIGA fuel produces less than half of the 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 fuel inherently reduces the potential for thermal fission as fuel temperature increases. Therefore, temperature increases from 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. Buildup of airborne radioactive contamination in the facility is controlled by a dynamic confinement and an argon purge system. Stack effluent from the dynamic confinement system is limited to maintain receptor doses to less than 10CFR20 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.

1.2. GENERAL DESCRIPTION OF THE FACILITY

1.2.1. Site

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

The PRC is a multidisciplinary research campus on 1.87 square kilometers. The site consists of two approximately equal areas, east and west. The NETL building is located in an area of about nine thousand square meters on the east tract. 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 Texas Advanced Computing Center (TACC), illustrating the diverse research activities on the campus. The 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.

1.2.2. NETL Building

The NETL building is a 1950 sq meter (21,000 sq ft), facility with laboratory and office spaces. Building areas consist of two primary laboratories of 330 sq m (3600 sq ft) and eighty sq m (900 sq ft), eight support laboratories (217 sq m, 2340 sq ft), and six supplemental areas (130 sq m, 1430 sq ft). Conference and office space are allocated to twelve rooms totaling 244 sq m (2570 sq ft). 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.

1.2.3. Reactor

The largest room in the NETL building is a vault-type enclosure 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 material science.

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 (Figure 1.1). The water serves as a coolant, neutron moderator, and radiation shield. The reactor core is surrounded by a graphite cylinder acting as a neutron reflector. Details of the reactor are provided in Chapter 4, Reactor.

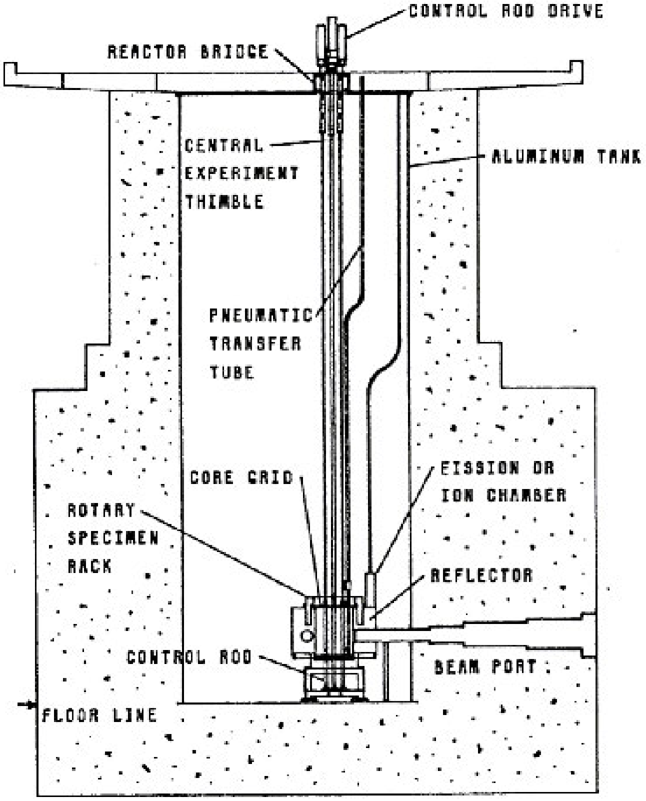

Figure 1.1, UT TRIGA Mark II Nuclear Research Reactor

1.2.3.a. 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.

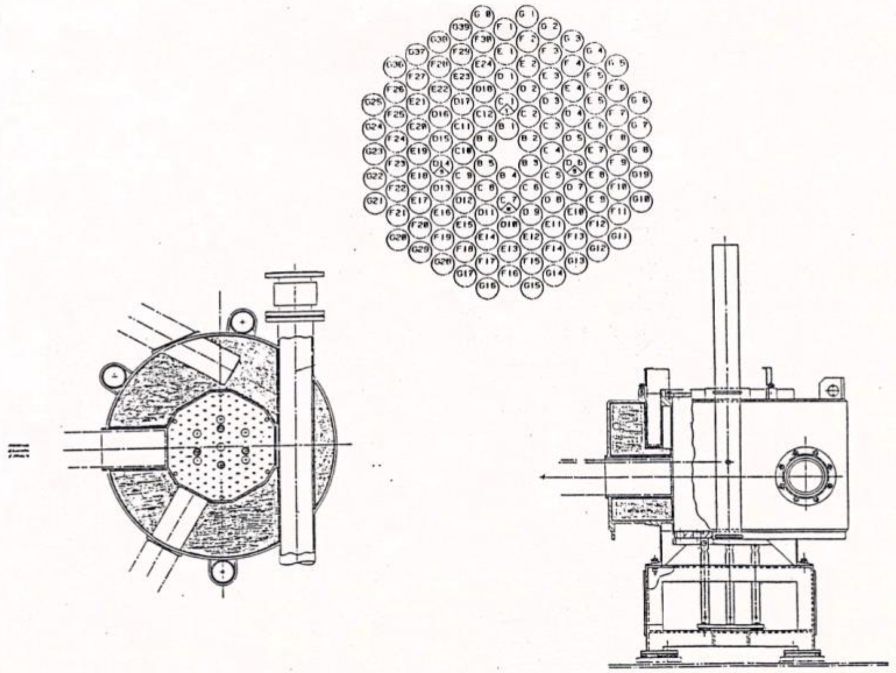

Figure 1.2, Core and Support Structure Details

1.2.3.b. Reactor Reflector

The reflector is a graphite cylinder in an aluminum-canister. A ten inch 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 modifications, including a modification to the upper grid plate for more flexible experiment facilities.

1.2.3.c. Reactor Control

The UT-TRIGA research reactor can operate continuously at steady-state powers up to 1.1 MW, or in the pulsing mode with maximum power levels in the GW range for periods of up to ten milli-seconds. 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 boron-carbide contained in stainless steel tubes. The transient rod is a solid cylinder of boron-carbide clad in aluminum. Removal of the rods from the core allows the rate of neutron induced fission (power) in the uranium-zirconium-hydride (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 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.

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.

1.2.4. 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; more complete descriptions are provided in Chapter 10 as well as Chapter 4.

1.2.4.a. 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.

1.2.4.b. 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.

1.2.4.c. 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.

1.2.4.d. 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.

1.2.4.e. Beam Port Facilities

Five neutron beam ports penetrate the concrete biological shield and reactor water tank at core level, as shown in Figure1.3. 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.

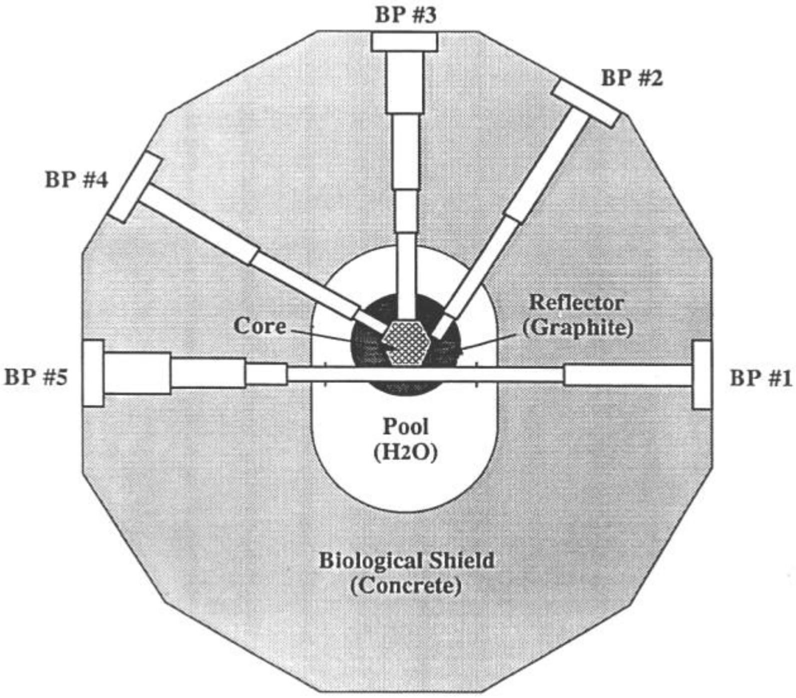

Figure 1.3, Beam Ports

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).

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 for neutron ad/or gamma irradiation. Equipment supporting testing of electronic components and systems has been designed and used in BP-1. Alternately BP-1 can provide beams of thermal neutrons with relatively low fast-neutron and gamma-ray contamination for external neutron irradiation.

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 has supported neutron depth profiling applications and prompt-gamma neutron activation analysis and is currently being used for irradiation of cryogenically cooled gaseous material.

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.

Cryogenically Cooled Gas Irradiation. Small quantities of gas are condensed to a solid phase on a cold-head in BP-2. The irradiation provides research quantities of high specific-activity neutron activated radioisotopes.

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 an eighty milliliters 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 neutron beam. 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.

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.

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.

1.3. 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.

1.4. 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.

1.5. 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 worldwide, and thirty-one 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. 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.

Table 1.1, Shutdown or Decommissioned U.S. TRIGA Reactors

	Thermal Power (kW)	Type	Initial Critical
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	250	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 (kW)	Type	Initial Critical
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 CONV	8/15/1955
AFRRI TRIGA	1,000	TRIGA MARK F	1/1/1962
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

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 from 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.

1.6. SUMMARY OF OPERATIONS

The UT TRIGA reactor has operated routinely since 1991 except for the 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 Figure 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.

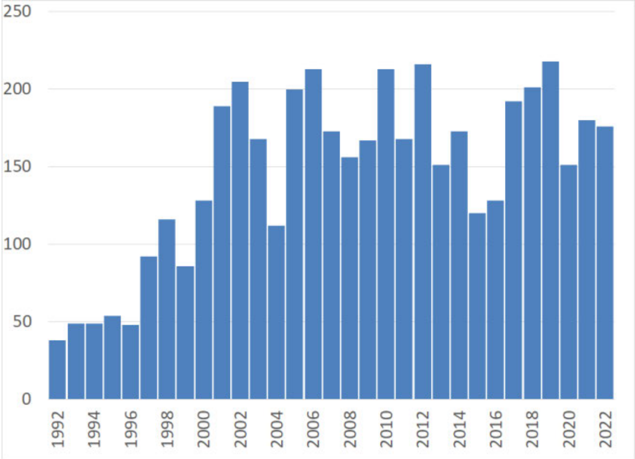

Figure 1.4A, Days of Operation per Year

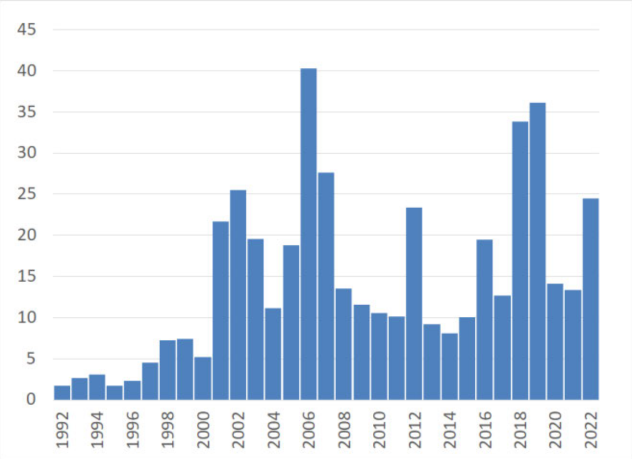

Figure 1.4B, Energy Generation (MWD) per Year

1.7. 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.

1.8. FACILITY HISTORY & MODIFICATIONS

The Department of Mechanical Engineering of the Cockrell School of Engineering at UT-Austin 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 ten 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 in 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. SITE DESCRIPTION

The site for the TRIGA reactor facility is located in the east tract of the J.J. Pickle Research Campus, an area owned and operated by The University of Texas at Austin. 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. Figures 2.1 to 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 J.J. 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.

The TRIGA reactor is located in the northeast region of the research center east tract. The site location is adjacent to the north boundary of the research center near to the eastern boundary. The location is near the intersection of Braker Lane and Burnet Road. Figure 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 access to other development projects north of the general site area.

Figure 2.1, State of Texas Counties

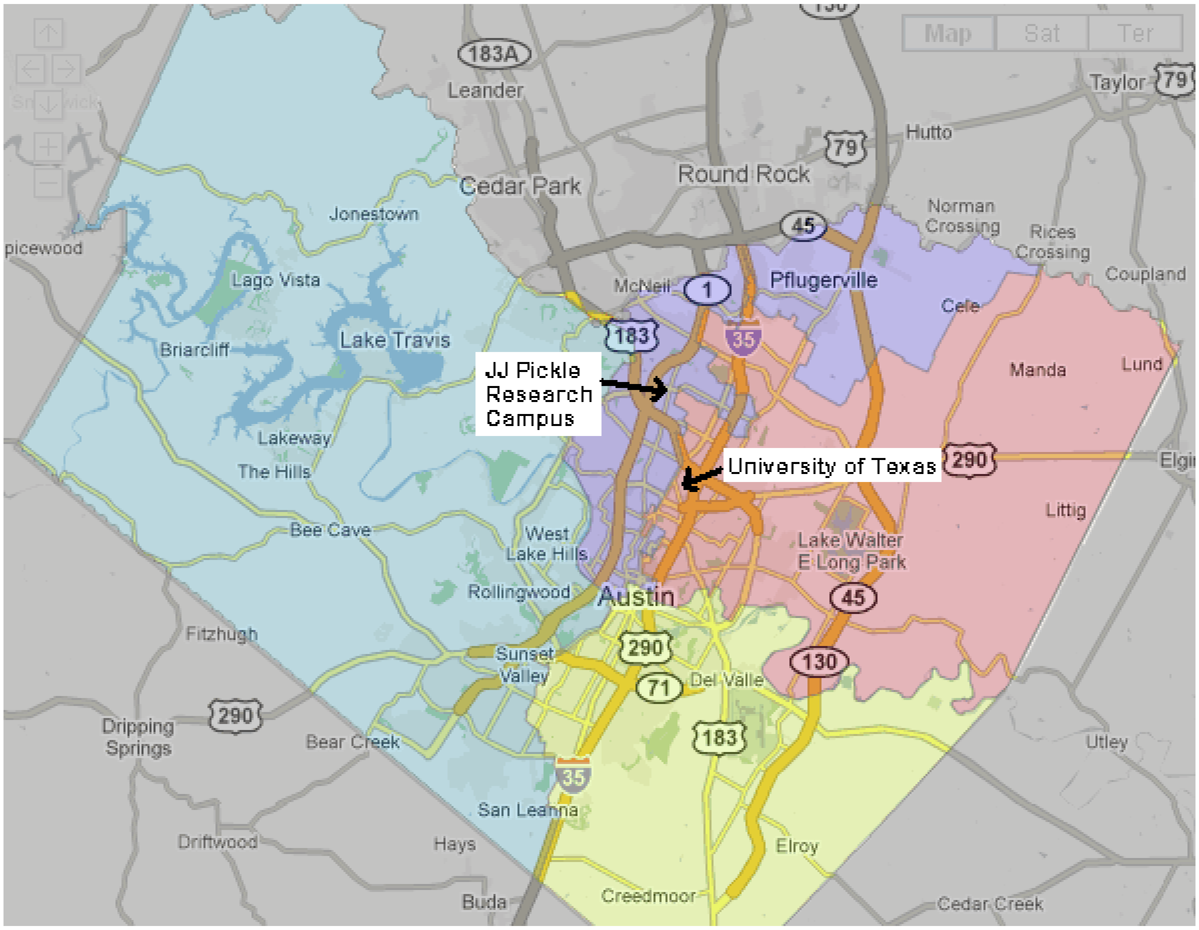

Figure 2.2, Travis County

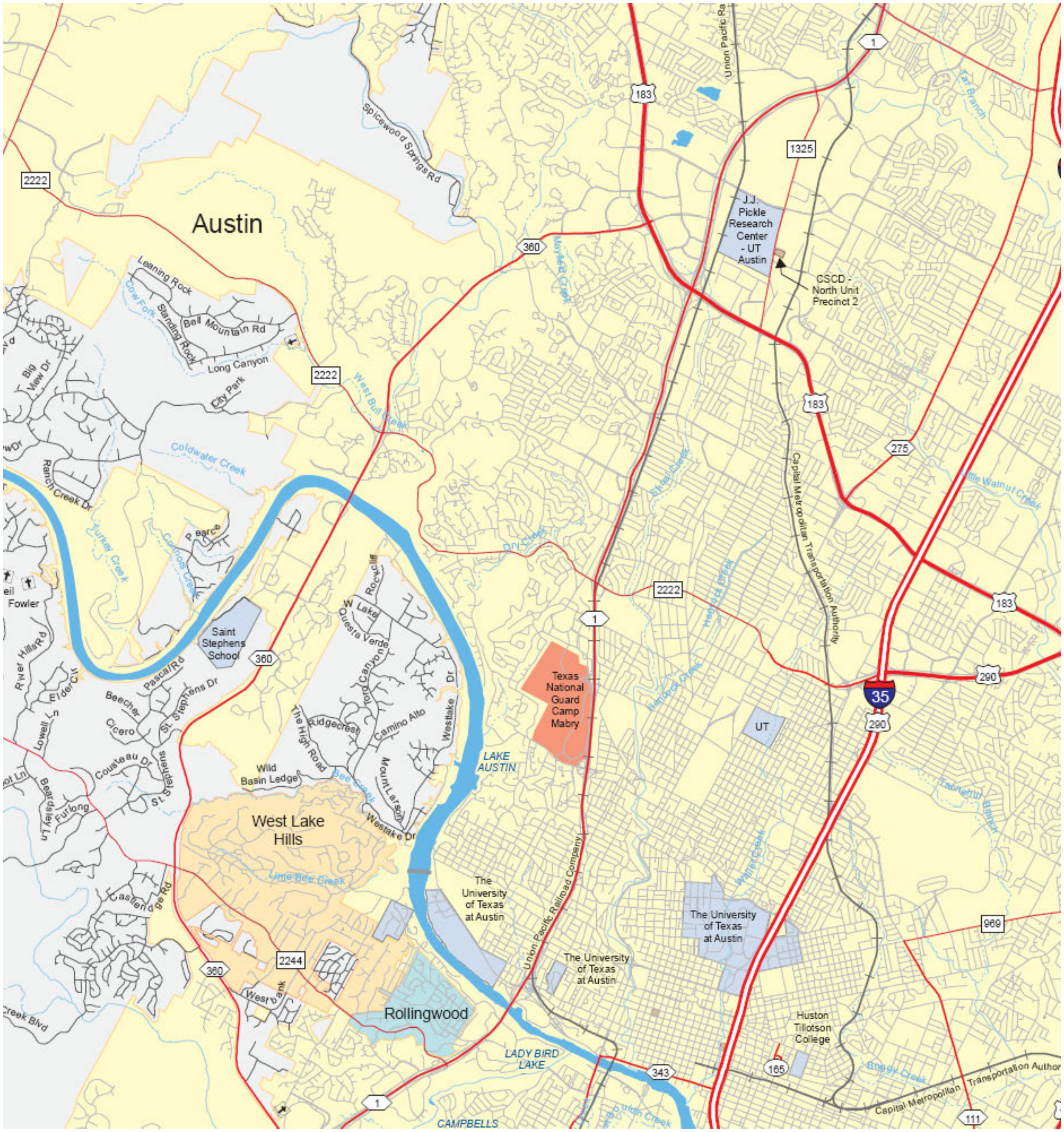

Figure 2.3, City of Austin

Figure 2.4, J.J. Pickle Research Campus (NETL is labeled as NEL on expanded map to the right).

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 Figure 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 the 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 974,447 in 2022, an over 100% increase. The growth rate slowed down from 2000-2004, but it has steadily increased from then. According to the 2020 census and predictive data, the growth rate will decrease over the next decade; however, the 2025 predicted population is 1,022,602 in Austin and 1,538,624 in Travis County. The annual growth rate in 2022 was 1.50% for Austin and 2.75% for Travis County.

Land usage of the area around JJ Pickle Research Campus is shown in Figure 2.5. The campus is surrounded by commercial mixed-use buildings, including multiple shopping centers. There is 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 Figure 2.6. Population density in the area containing NETL, zip code 78758, has an average of 5575 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 3153 people per square mile.

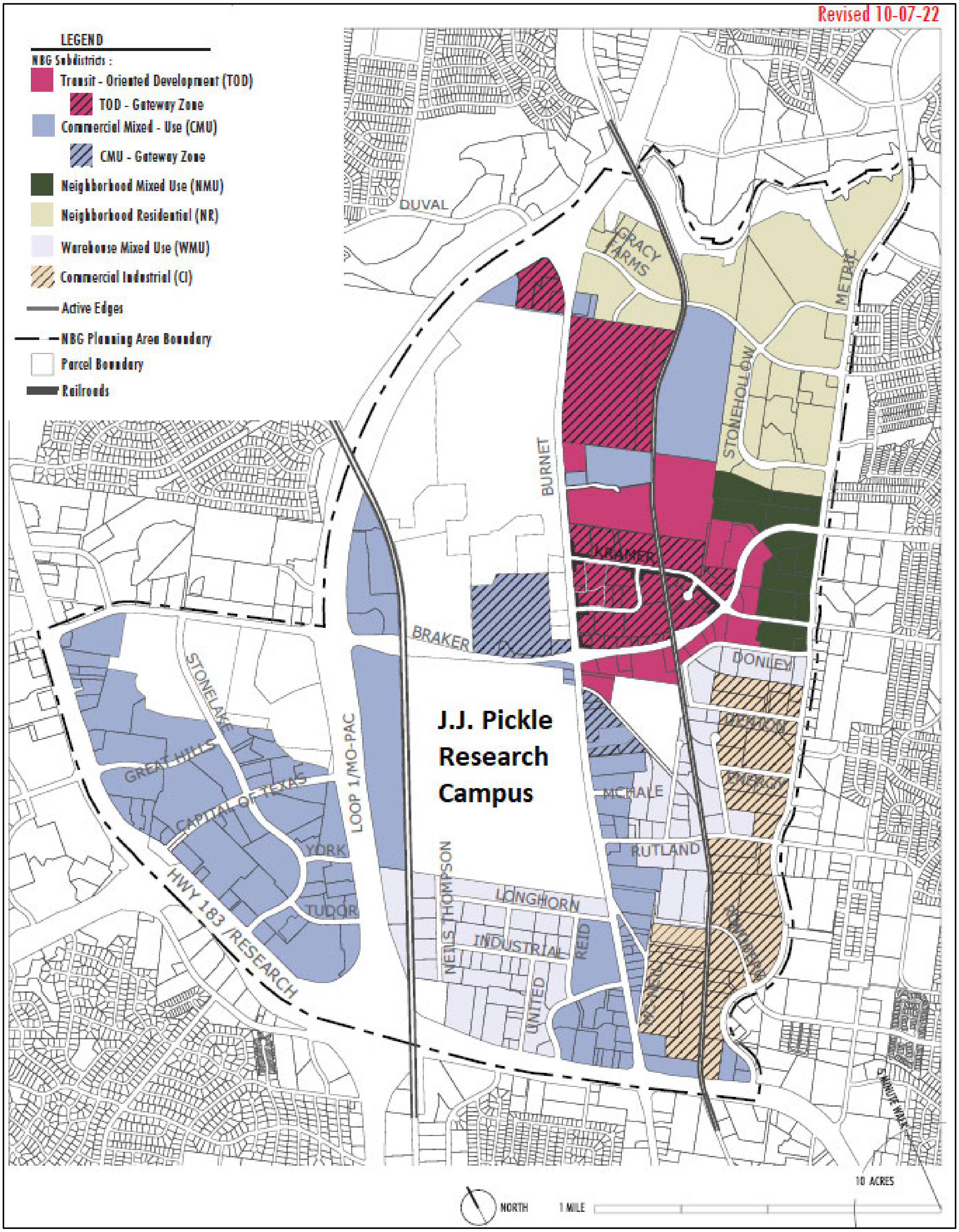

Figure 2.5, Land Use and Usage around JJ Pickle Research Campus, 2022.

Table 2.1, Austin and Travis County Population Trends[1]

Year	City of Austin Total Area Population	Annual Growth Rate	City of Austin Full Purpose Population	City of Austin Limited Purpose Population	Travis County	Annual Growth Rate	Five County MSA	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	807,536	2.17%	799,578	12,447	1,050,858	2.6%	1,763,487	2.75%
2012	824,682	2.12%	813,776	12,459	1,077,450	2.53%	1,811,983	2.75%
2013	841,828	2.08%	828,223	12,472	1,104,042	2.47%	1,861,812	2.75%
2014	858,974	2.04%	845,024	12,484	1,130,634	2.41%	1,917,667	3.00%
2015	876,120	2.00%	860,018	12,497	1,157,226	2.35%	1,975,197	3.00%
2016	893,266	1.96%	875,274	12,509	1,183,818	2.3%	2,034,453	3.00%
2017	910,412	1.92%	890,798	12,522	1,210,410	2.25%	2,100,572	3.25%
2018	927,558	1.88%	906,594	12,534	1,237,002	2.2%	2,168,841	3.25%
2019	944,704	1.85%	922,666	12,547	1,263,594	2.15%	2,239,328	3.25%
2020	963,121	1.95%	936,682	12,559	1,290,188	2.1%	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	1,819,686	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%

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[1] Population figures are as of April 1 of each year.

Table 2.2, Travis County 2023 Austin Population Density Distribution by Zip Code

Zip Code	**Population Density**
	(pop/square mile)
78705	15685.1
78751	6559
78752	6339
78701	5967.5
78741	5758.6
78704	5613.2
78758	5575.3
78756	5568.6
78753	5176.8
78757	5153
78723	5109.9
78702	4777.2
78745	4528.7
78722	4378.5
78727	3856.7
78748	3780.7
78703	3748.6
78749	3707.2
78728	3262.2
78759	3152.6
78729	3129.3
78721	3004

Zip Code	**Population Density**
	(pop/square mile)
78731	2959.7
78717	2594.1
78750	2502.1
78744	2144
78754	1958.5
78739	1715.2
78732	1251.5
78746	1180.2
78726	1145.5
78724	962.5
78735	947.4
78734	931.2
78747	930
78733	830.1
78737	679.4
78738	635.4
78730	610.4
78725	494.4
78736	424.8
78712	361.3
78742	178.9
78719	93.8

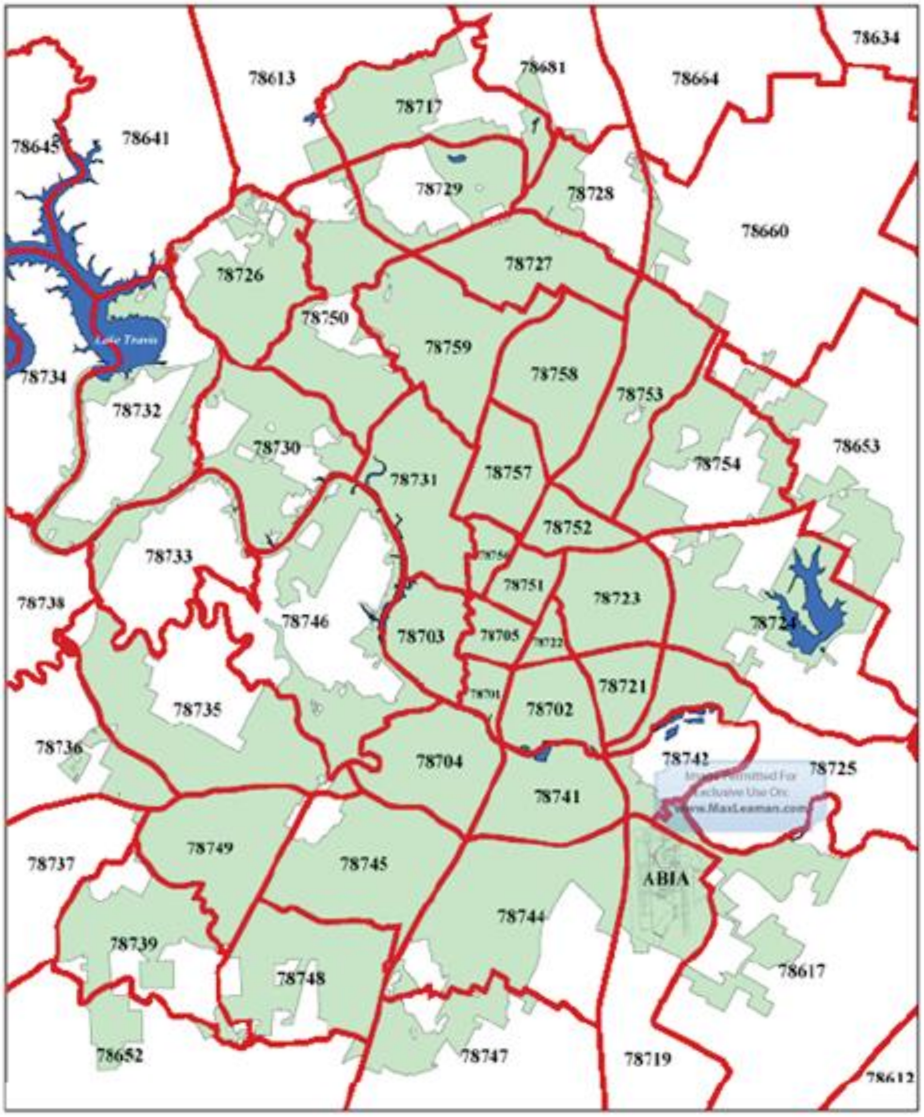

Figure 2.6, 2023 Zip Code Boundaries (the J.J. Pickle Research Campus is located in 78758 and adjacent to 78759)

Research activities at the J.J. 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[2]. It is difficult to put a number on how many people work at J.J. 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 workday during a semester, there are upwards of 1500 people on the Pickle Research Campus.

Immediately adjacent to the NETL building is the geology building (see Figure 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.

2.3. CLIMATOLOGY

Austin is located in central Texas at the junction of the Colorado River and the Balcones escarpment, separating the Texas Hill Country from the prairies to the east. Elevations within the city limits vary from 400 feet in the east/southeast to just above 1000 feet above sea level on the northwest side as you begin to enter into the Hill Country. Given these large changes in elevation, weather conditions at any one time can sometimes differ between various sectors of the city and metro area.

Austin belongs to the Humid Subtropical Climate under the Koppen Climate Classification. This climate is characterized by long, hot summers and short, mild winters, with warm spring and fall transitional periods. Austin averages around 35.5 inches of rainfall per year, with May, October, and June being the wettest months of the year, in that order. Austin has two automated surface observation system (ASOS) sites.

Winter in Austin is typically characterized by relatively mild temperatures and a general lack of precipitation. During winter, the area is alternately influenced by a continental air mass regime, with winds from the north and west and drier air, and by a modified maritime air mass regime, with south and southeast winds and moist air from the Gulf of Mexico. Mild weather prevails during most of the winter. January is the coldest month of the year, with normal highs in the low 60s and normal lows in the low 40s. Very strong arctic fronts will occasionally usher in frigid conditions to central Texas. The coldest low in recorded history was -2 on January 31, 1949. Significant wintry precipitation, in the form of freezing rain, sleet, or snow, impacts the Austin area on average about once every two years (significant meaning enough to cause large impacts to travel, etc.). The largest snowstorm on record occurred on November 22-23, 1937, in which 11 inches of snow was recorded. The most recent snowstorm occurred on February 14-15, 2021, in which 6.4 inches of snow fell.

Normal winter (DJF) precipitation is 7.25 inches, which comprises about 20% of the yearly precipitation. It is not particularly uncommon for there to be very warm days in winter in Austin. The hottest winter day on record was February 21, 1996, in which Austin reached 99 degrees. Late

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

winter is also typically the peak of fire weather season across the area. Very dry air and gusty northerly winds that filter into the region behind passing cold fronts, as well as the generally dry conditions, create favorable conditions for wildfires. Summers in Austin are long and hot. Normal highs reach 90 degrees by May 26 and remain above 90 until September 23. Temperatures reach their peak in the first half of August, with normal highs in the upper 90s and lows in the mid-70s. The hottest temperature on record is 112 degrees, which was reached on September 5, 2000, and again on August 28, 2011. Normal summer overnight lows range from the low to mid 70s. Southeast winds transporting moisture from the Gulf of Mexico can increase humidity values, taking heat indices up above 110 degrees on occasion. The hottest summer and second hottest year on record occurred in 2011, in which there were 90 days with temperatures reaching or exceeding 100 degrees.

June is now the third wettest month of the year, with an average of 3.68 inches of rain. July and August tend to be relatively dry. Normal summer (JJA) precipitation is 8.38 inches, comprising about 23% of the yearly precipitation. Precipitation is relatively evenly distributed throughout the year with heaviest amounts occurring in May, October, and June, in that order. Precipitation in the spring and summer usually results from thunderstorms. Thunderstorms in Austin can be very efficient rainmakers, with large amounts of rain falling within short periods of time. Rainfall amounts have exceeded 5 inches in several hours. The record for two-day rainfall occurred on September 9-10, 1921, in which 19.03" of rain fell. Austin has a history of devastating flash floods. Rainfall in the late summer and fall is typically controlled largely by any land-falling tropical weather systems. Average yearly rainfall is 36.25 inches. Extremes vary from 11.42 inches in 1954 to 65.31 inches in 1919.

Prevailing winds are typically southerly; however, in winter, northerly winds are about as frequent as those from the south, depending on the frequency of passing cold fronts through the region. Average sunshine varies from about 50 percent in the winter to nearly 75 percent in the summer. Low stratus clouds frequently develop at night and in the early morning hours during all seasons with south and southeast winds, as Gulf moisture is lifted from the coastal plains to the higher terrain over the Balcones Escarpment. On some days, these clouds do not dissipate, persisting all day. In the winter, these stratus clouds may be accompanied by fog and drizzle, as south and southeast winds brings Gulf moisture over the top of a shallow layer of cold air at the surface. Aggregate wind data is provided in Figure 2.6 as a wind rose.

The average occurrence of the first freeze is November 29 and the average occurrence of the last freeze is February 25 for the 120+ year period of record. Over just the 30 years from 1991-2020, the average first freeze is on December 1st and the average last freeze is much earlier, on February 15th. The earliest first freeze on record was October 26, 1924 and the latest last freeze was on April 9, 1914. The average occurrence of the first 100 degree day is July 9th and the average occurrence of the last 100 degree day is August 21st, although over the last 30 years this average is August 30th. The earliest 100 degree day on record was May 4, 1984 and the latest 100 degree day on record was on October 2, 1938.

The severe weather season in Austin is primarily March through May. The majority of severe weather comes in the form of large hail and strong winds. Tornadoes are not particularly common but do occur on occasion. The vast majority of these tornadoes are relatively weak, ranging from EF-0 to EF-1 on the Enhanced Fujita Scale. However, incidentally the last F/EF-5 tornado to occur in the state of Texas occurred in Jarrell, TX, just north of Austin in Williamson County, on May 27, 1997. There have been 16 (F1) tornados in Austin between 2001 and 2015 (Table 2.3).

Tropical storms impact Austin on rare occasions (Figure 2.7 and Figure 2.8). The primary threat to the Austin region from tropical storms is heavy rain causing flooding. The most recent tropical storm to impact Austin was Hurricane/Tropical Storm Harvey in late August 2017. Austin Camp Mabry received 7.94 inches of rain from Harvey while Austin Bergstrom received 10.07 inches.

Table 2.3, F1 Tornados in Austin, TX

Date	Number
5/25/2015	7
1/25/2012	1
4/27/2009	2
3/25/2005	1
12/23/2002	1
11/15/2001	4

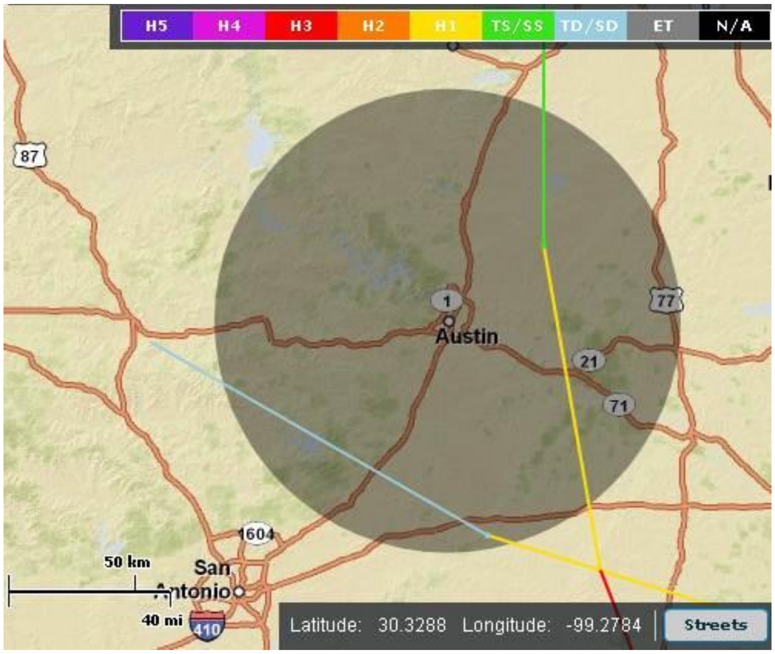

Figure 2.7, Tropical Storm Paths within 50 Nautical Miles of Austin Texas (All Recorded Hurricanes Rated H1 and UP)

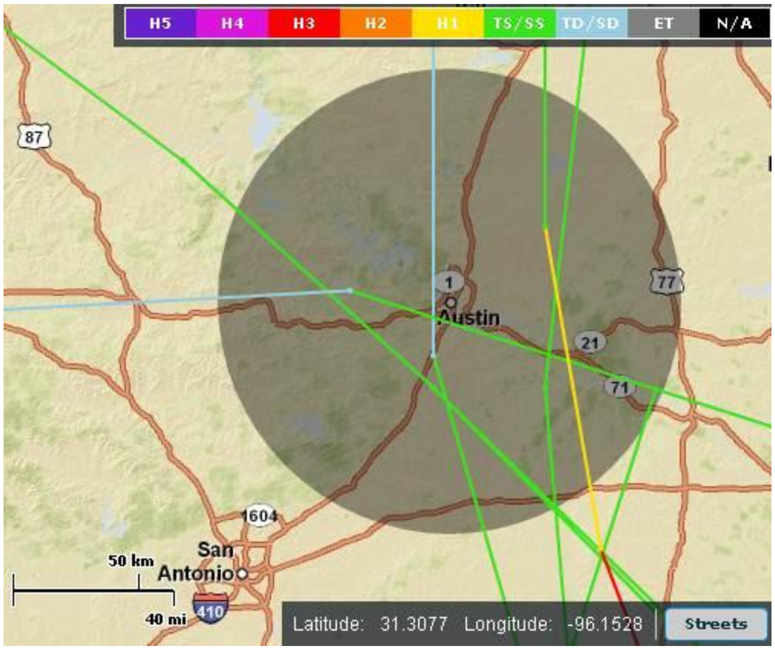

Figure 2.8, Tropical Storm Paths within 50 Nautical Miles of Austin Texas (All Recorded Storms Rated TROP or SUBTROP)

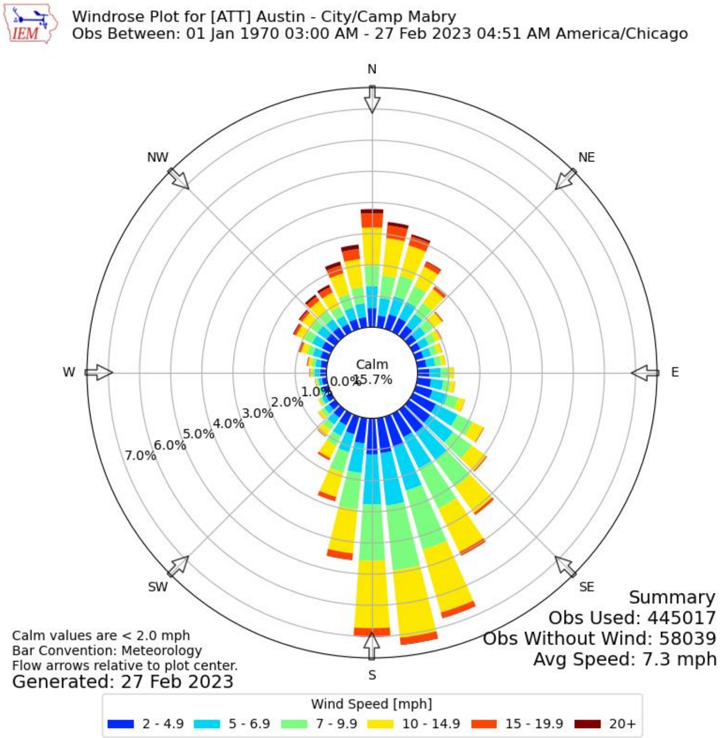

Figure 2.9, Austin Wind Rose Data[3]

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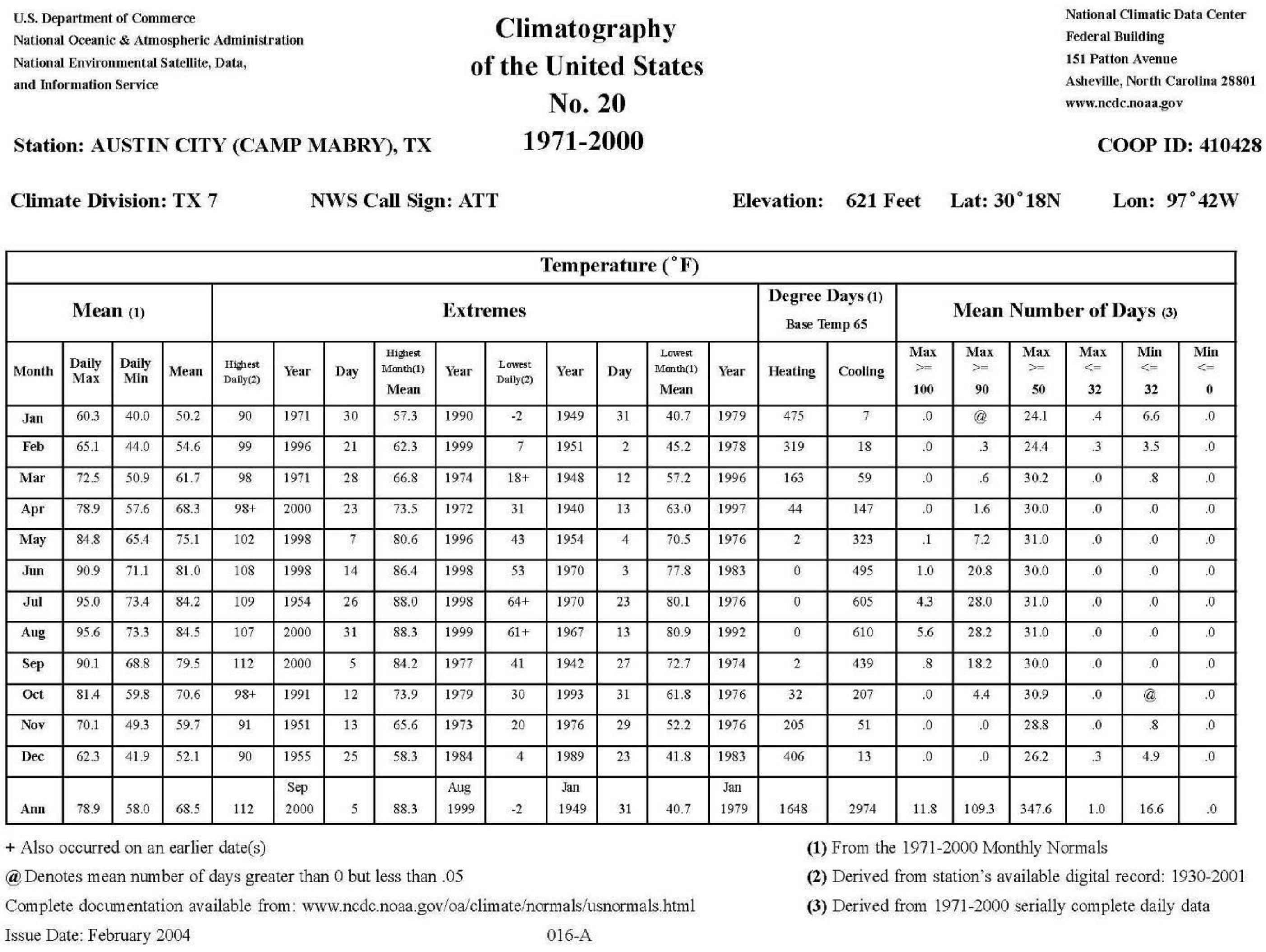
[3] Iowa State University, Iowa Environment Mesonet, https://mesonet.agron.iastate.edu/ (accessed on August 2, 2023).

The hottest year on record in Austin occurred in 2017, with an average temperature of 72.1 degrees. The coldest year on record in Austin occurred in 1899, with an average temperature of 65.8 degrees. Tables 2.4 through Table 2.8 provide historical Austin meteorological data.



U.S. Department of Commerce
National Oceanic & Atmospheric Administration
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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**

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	1.0	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

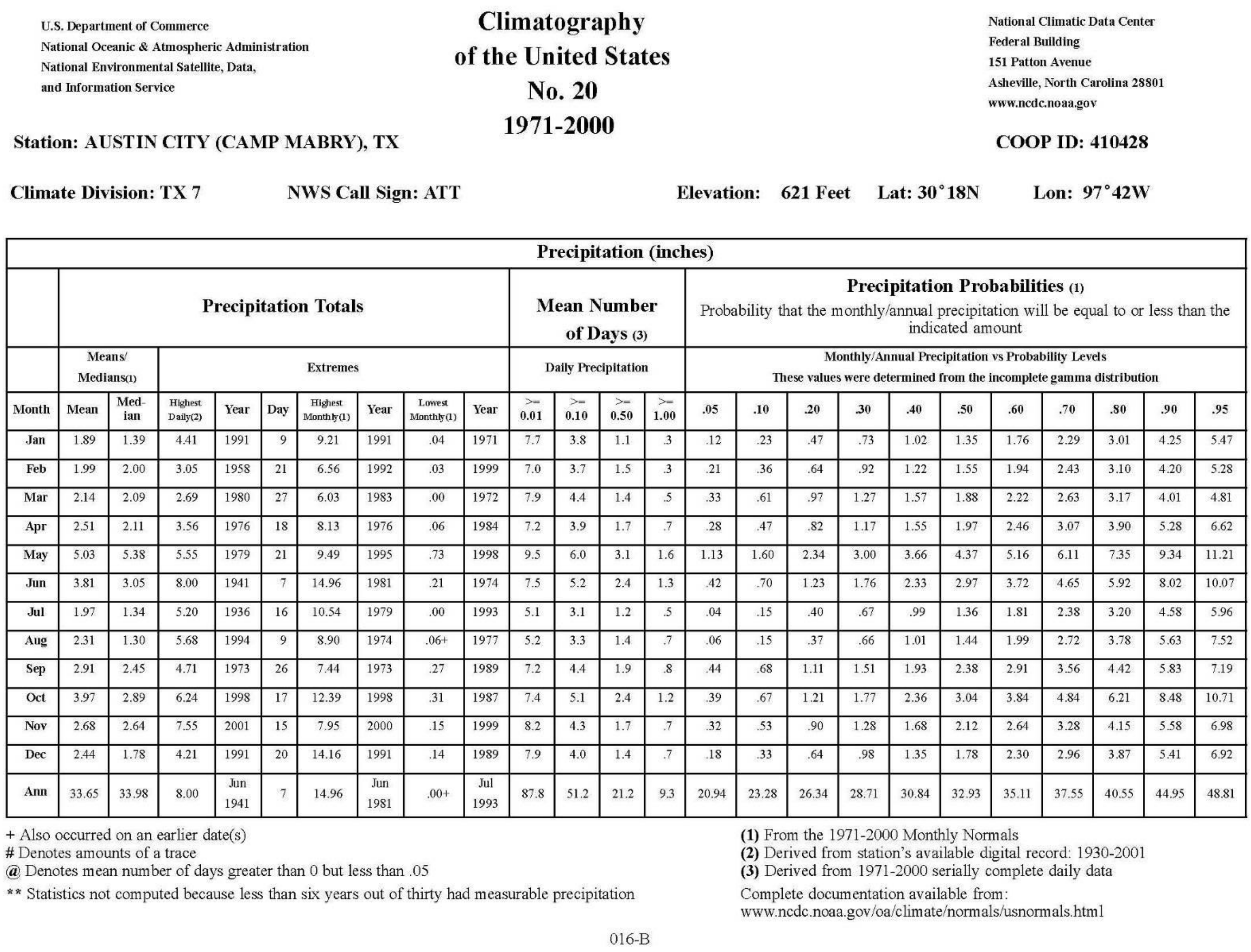
016-A

(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

Table 2.4, Historical Meteorological Data for Austin Texas



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

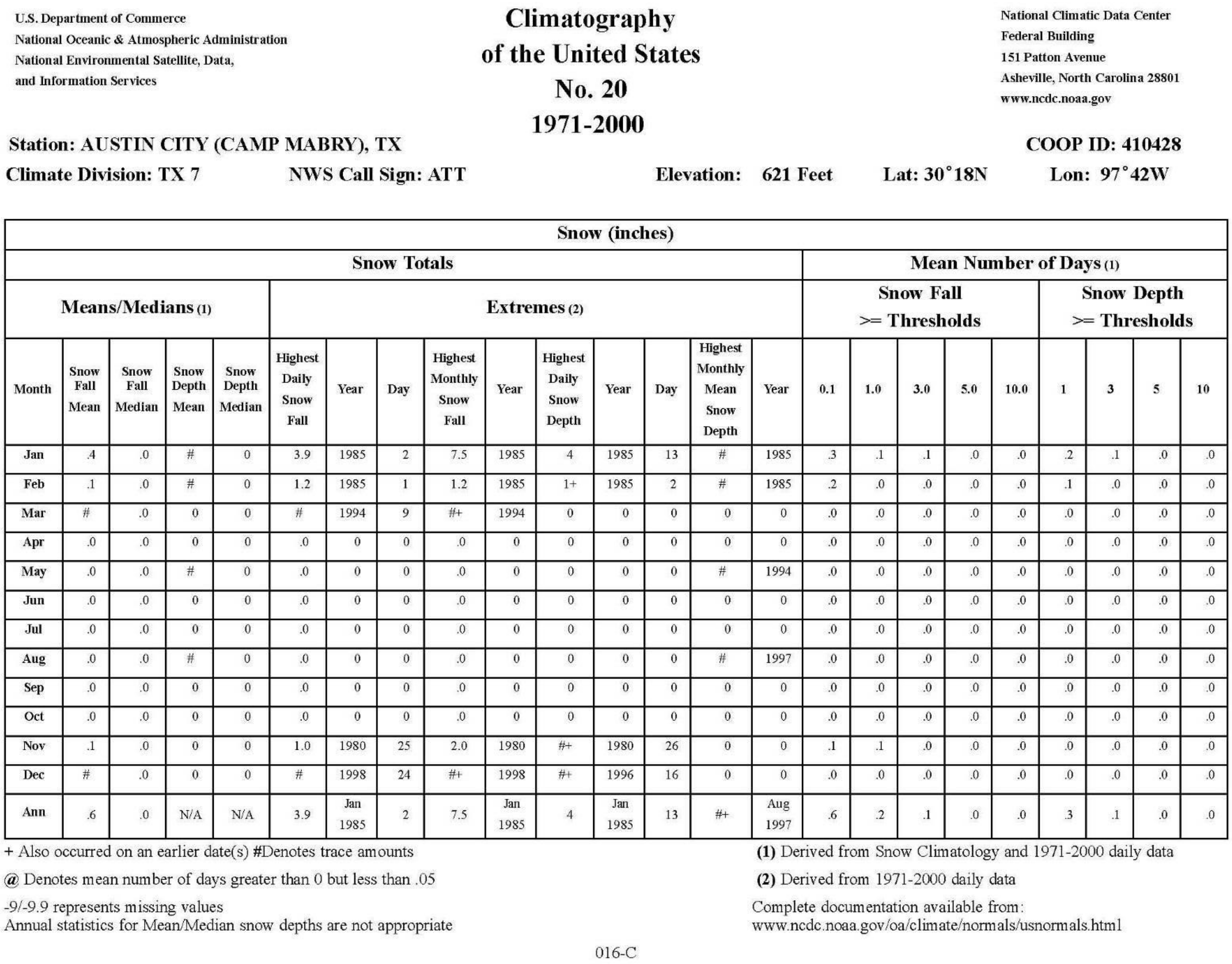
Precipitation (inches)																								
	Precipitation Totals									Mean Number of Days (3)				Precipitation Probabilities (1) Probability that the monthly/annual precipitation will be equal to or less than the indicated amount										
	Means/ Medians(1)		Extremes							Daily Precipitation				Monthly/Annual Precipitation vs Probability Levels These values were determined from the incomplete gamma distribution										
Month	Mean	Med-ian	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

Table 2.5, Historical Meteorological Data for Austin Texas



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**

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

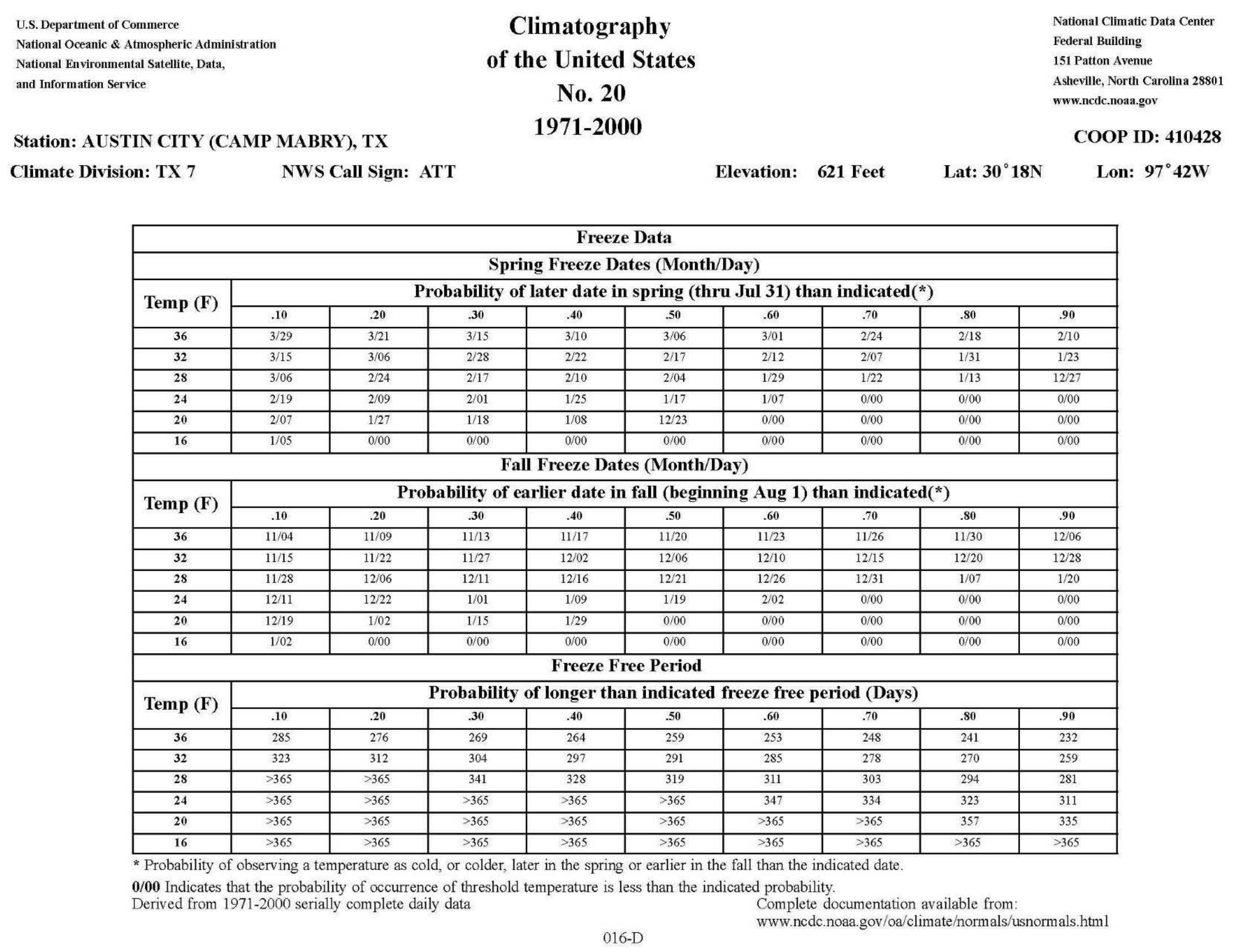
(1) Derived from Snow Climatology and 1971-2000 daily data

(2) Derived from 1971-2000 daily data

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

016-C

Table 2.6, Historical Meteorological Data for Austin Texas



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Climatography of the United States

No. 20

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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.

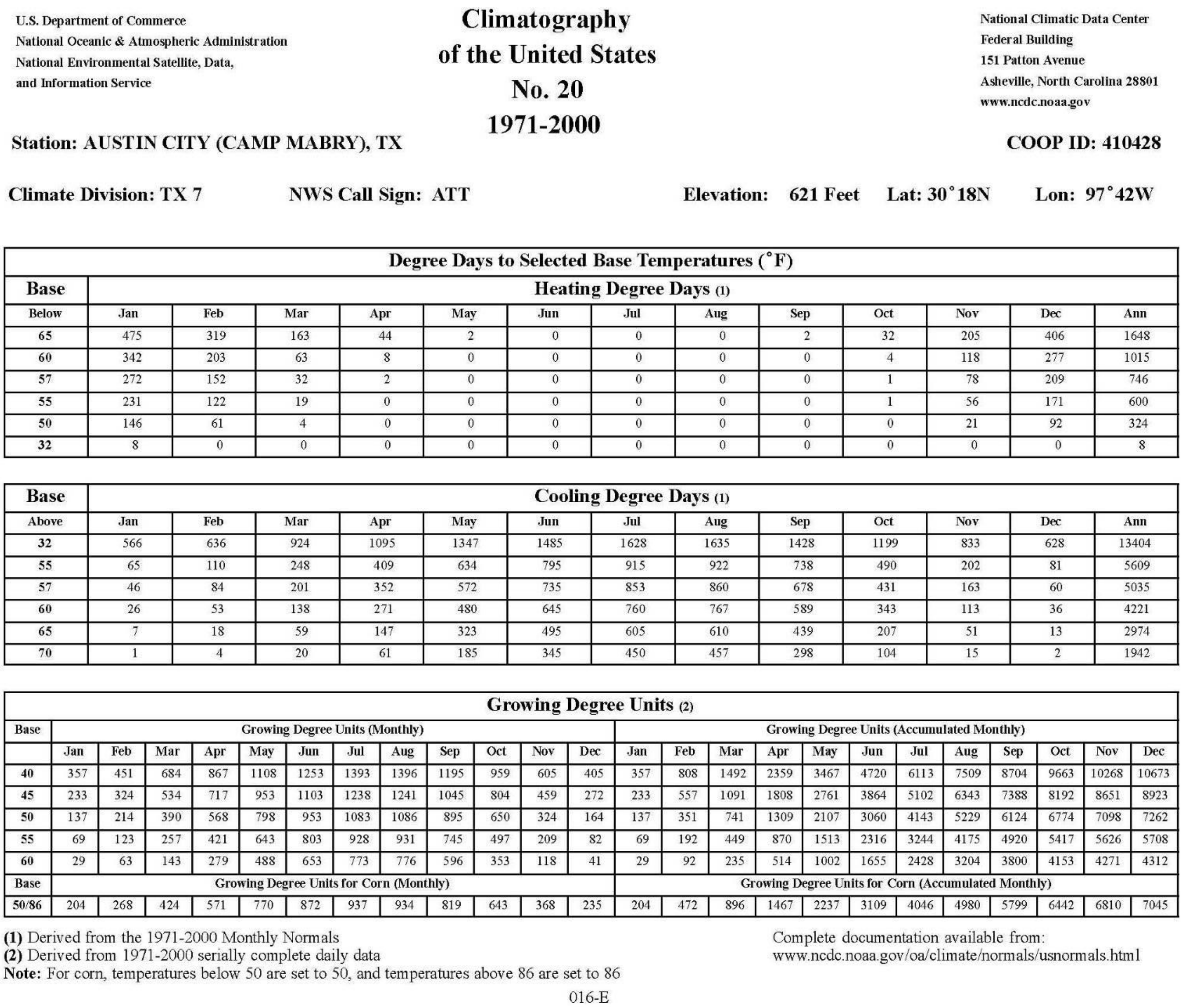
0/00 Indicates that the probability of occurrence of threshold temperature is less than the indicated probability.

Derived from 1971-2000 serially complete daily data

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

016-D

Table 2.7, Historical Meteorological Data for Austin Texas



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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, Historical Meteorological Data for Austin Texas

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 wooded 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. Subsurface Exploration Logs for the NETL building site are provided in Appendix 2.1. 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 downside 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 movements 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[4]. The location of the Balcones Fault zone and formations in the Austin area are depicted in Figure 2.10.

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

Figure 2.10, Balcones Fault Zone

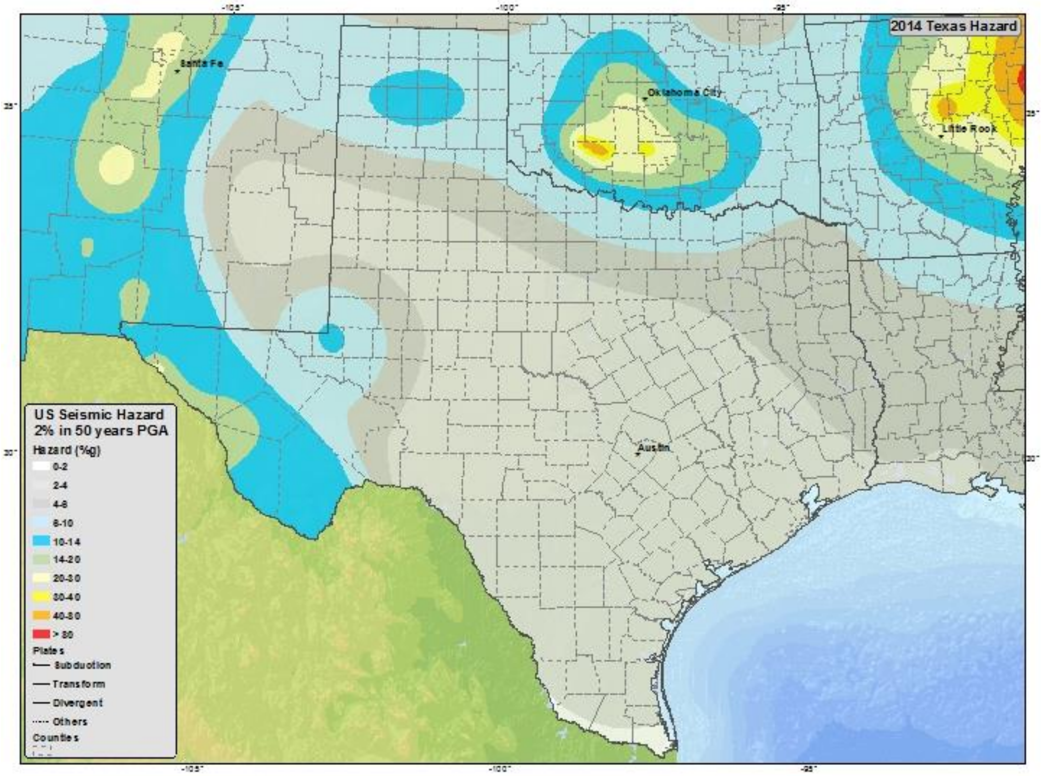

Figure 2.11, 2014 Seismic Hazard Map - Texas

2.5. SEISMOLOGY

Figure 2.11 is a map of probability for ground acceleration from a seismic event in a 50-year period. Thirty-three earthquakes of intensity IV or greater have had epicenters in Texas between 1873[5] and 2011. 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. Two earthquakes have occurred since 1900 at 93 miles (May 2018) and 158 miles (May 2015)[6] from Austin. No damage has ever occurred to local buildings in the Austin area from seismic activity.

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

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

[6] https://earthquakelist.org/usa/texas/austin/.

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 supply 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. Figure 2.12 shows the location of the ground water aquifers.

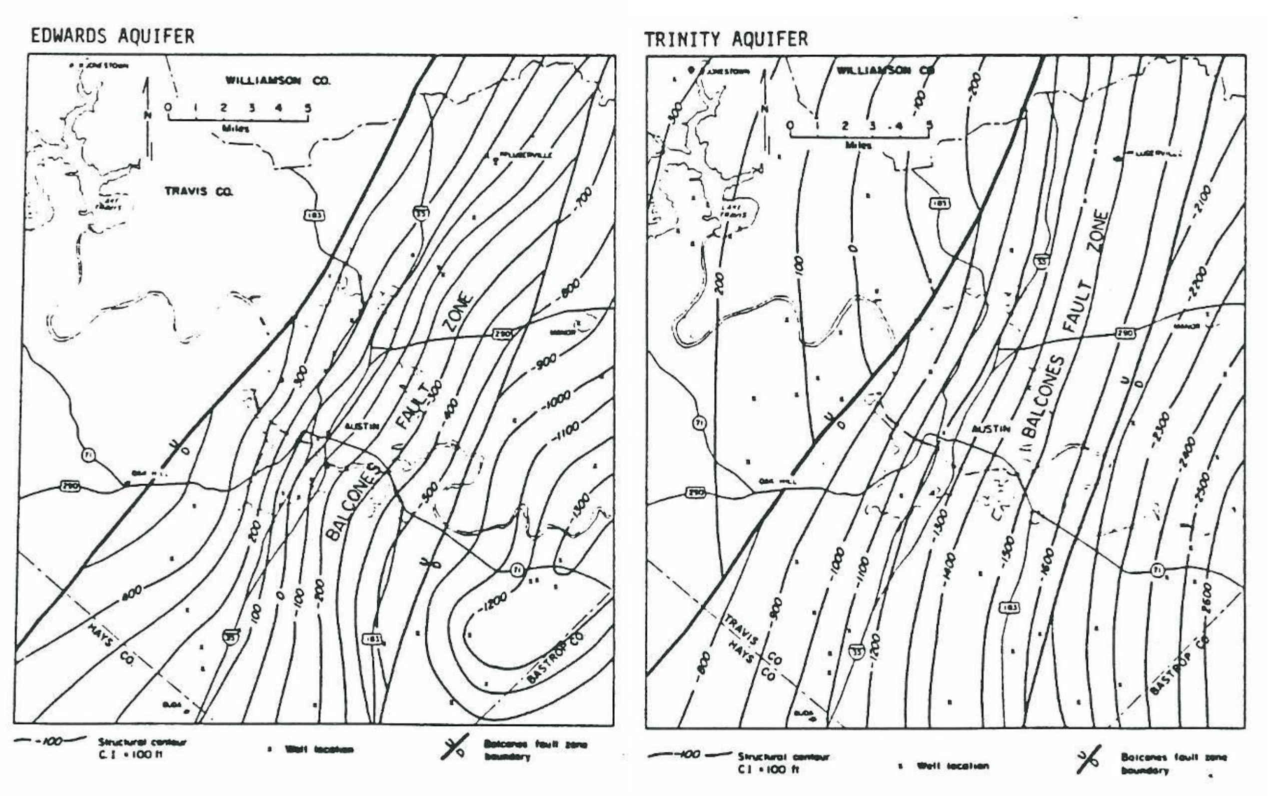


Figure 2.12, Local Water Aquifers

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.

Table 2.9, Texas Water Development Board (TWDB) Groundwater Database (GWDB) Well Information Report.

State Well Number	Sample Description	Sample Date	alpha	beta
58-35-721	Pickle Research Campus	10/4/2022	2	5.2
58-50-215	Balcones Fault Zone Aquifer (5 years)	5/18/2023	3	N/A

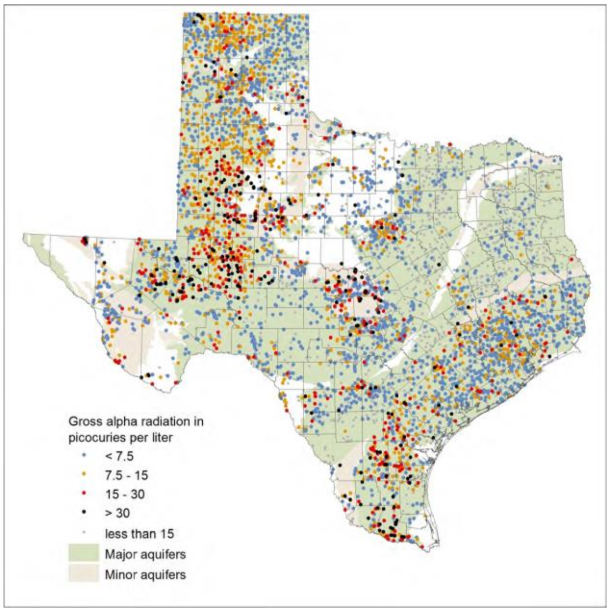


Figure 2.13a, Texas Groundwater Gross Alpha Radiation

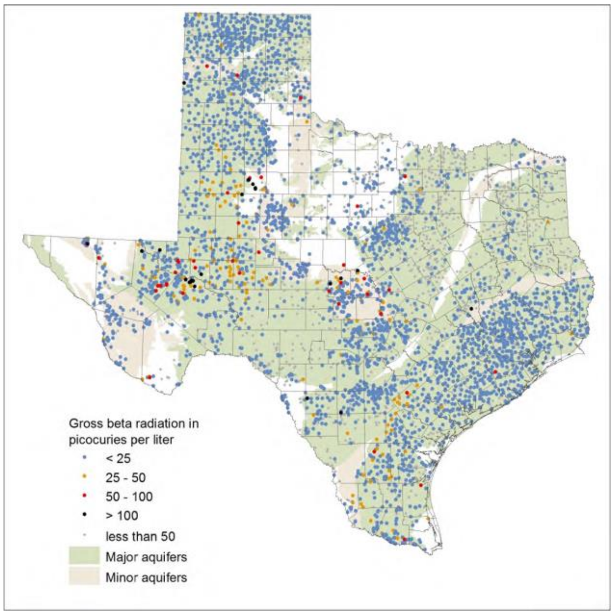


Figure 2.13b, Texas Groundwater Gross Beta Radiation

Ground water contamination by radioisotopes emitting alpha and beta radiation is monitored by the state for potential noncompliance with Environmental Protection Agency Standards related to maximum contamination levels for drinking water (Figures 2.13a and Figure 2.13b)[7] and data from local wells in the Texas Water Development Board (TWDB) Groundwater Database (GWDB) Well Information Report are provided in Table 2.9.

---

[7] R. C. Reedy, B. R. Scanion, S. Walden, G. Strasberg, “Naturally Occurring Groundwater Contamination in Texas,” Bureau of Economic Geology, The University of Teas at Austin, 2011.

2.7. HISTORICAL

Relocation of the UT TRIGA reactor and related facilities to the J.J. 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[8].

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 accommodation was 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[9] 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 Figure 2.14 and 2.16.

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[10].

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.

---

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

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

[10] "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/>

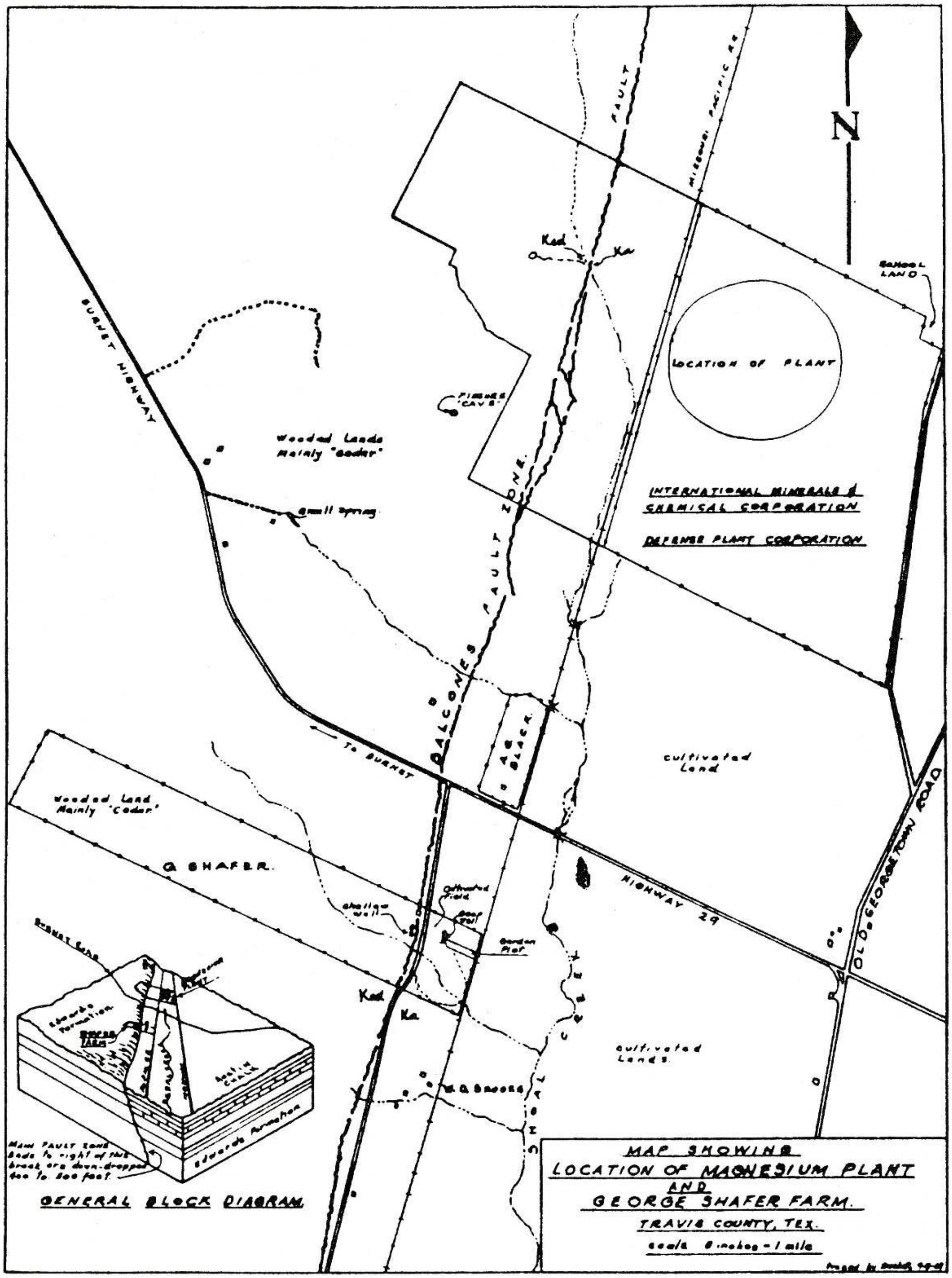


Figure 2.14, Research Campus Area 1940

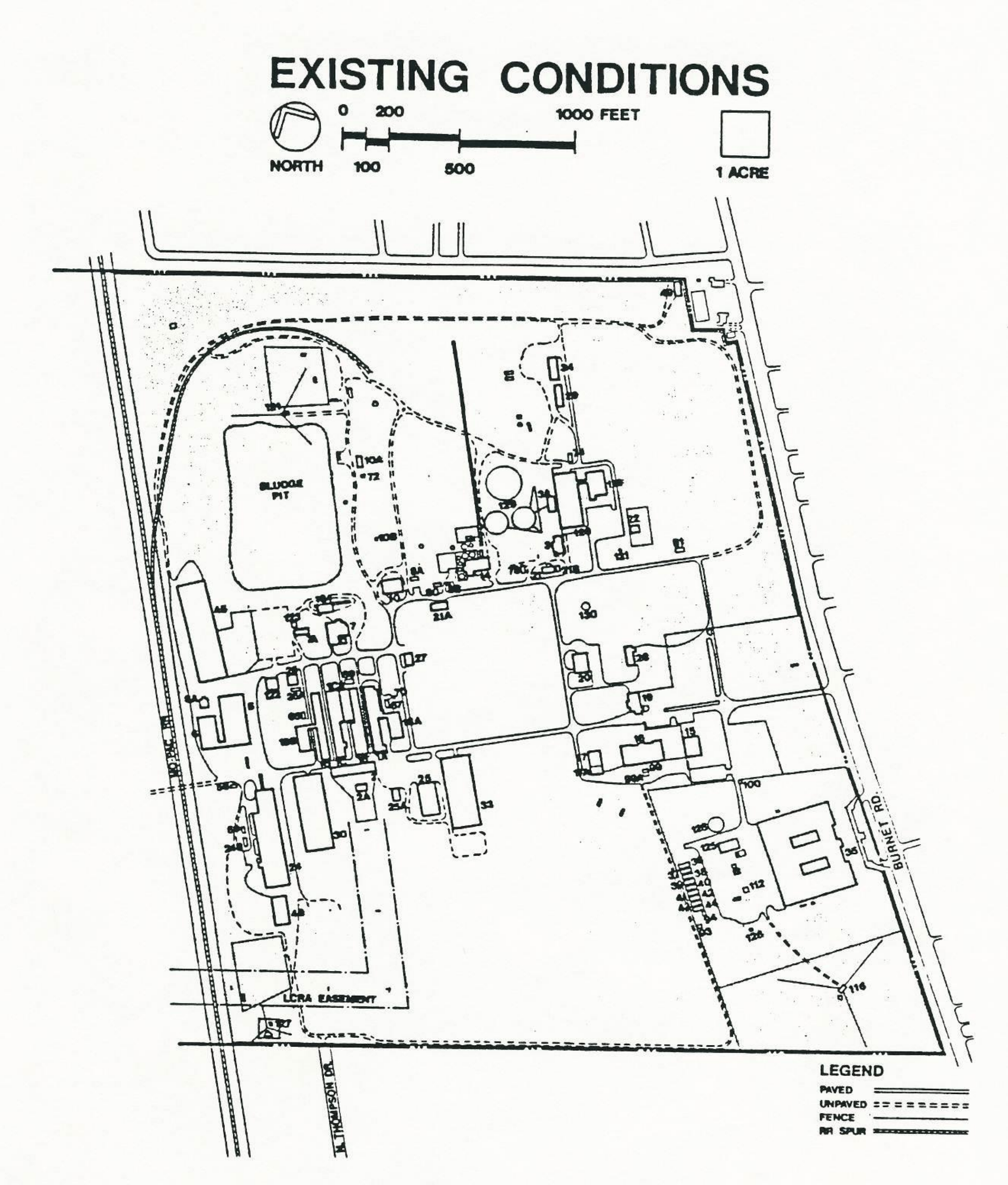


Figure 2.15, Pickle Research Campus 1960

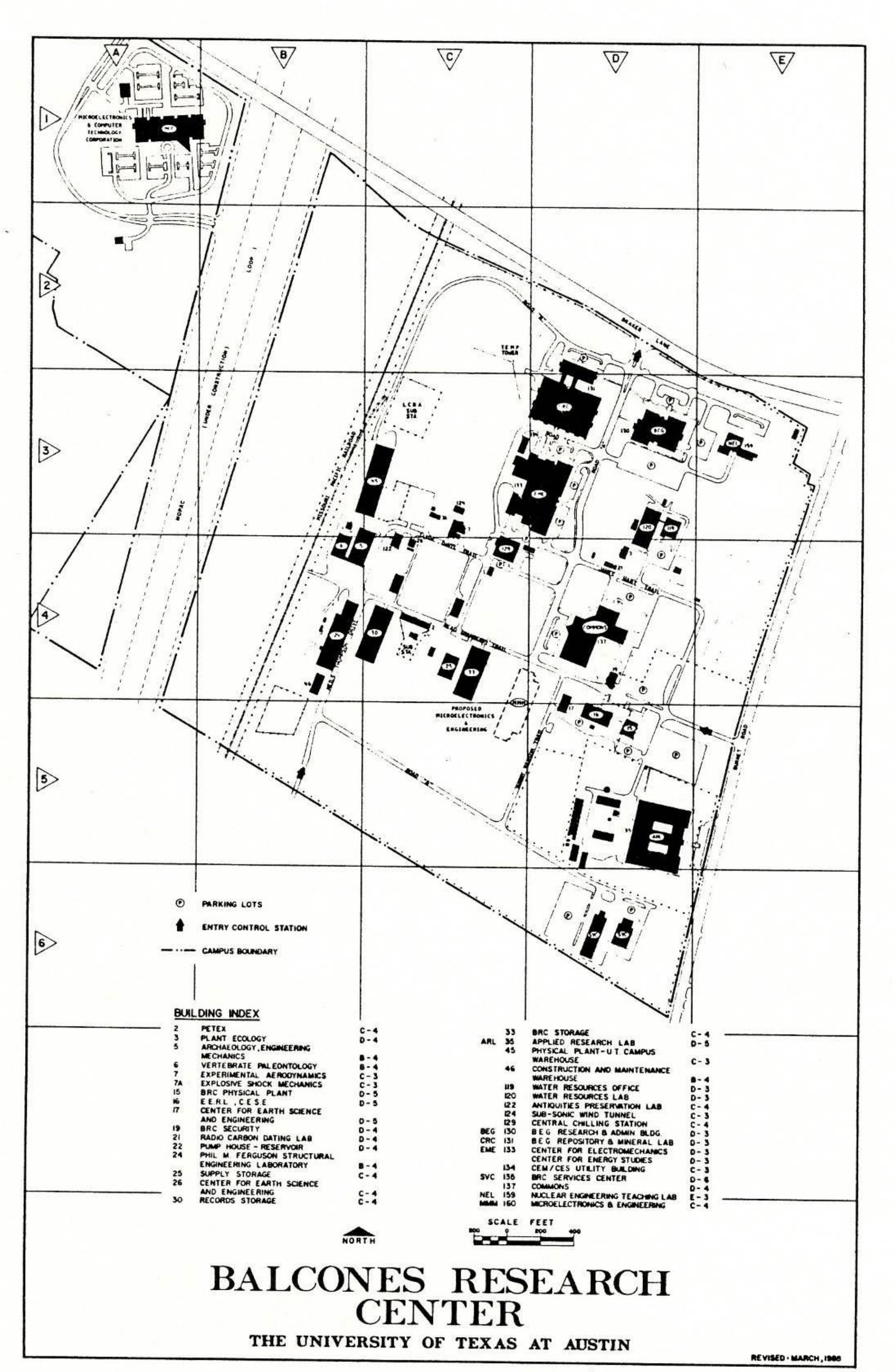
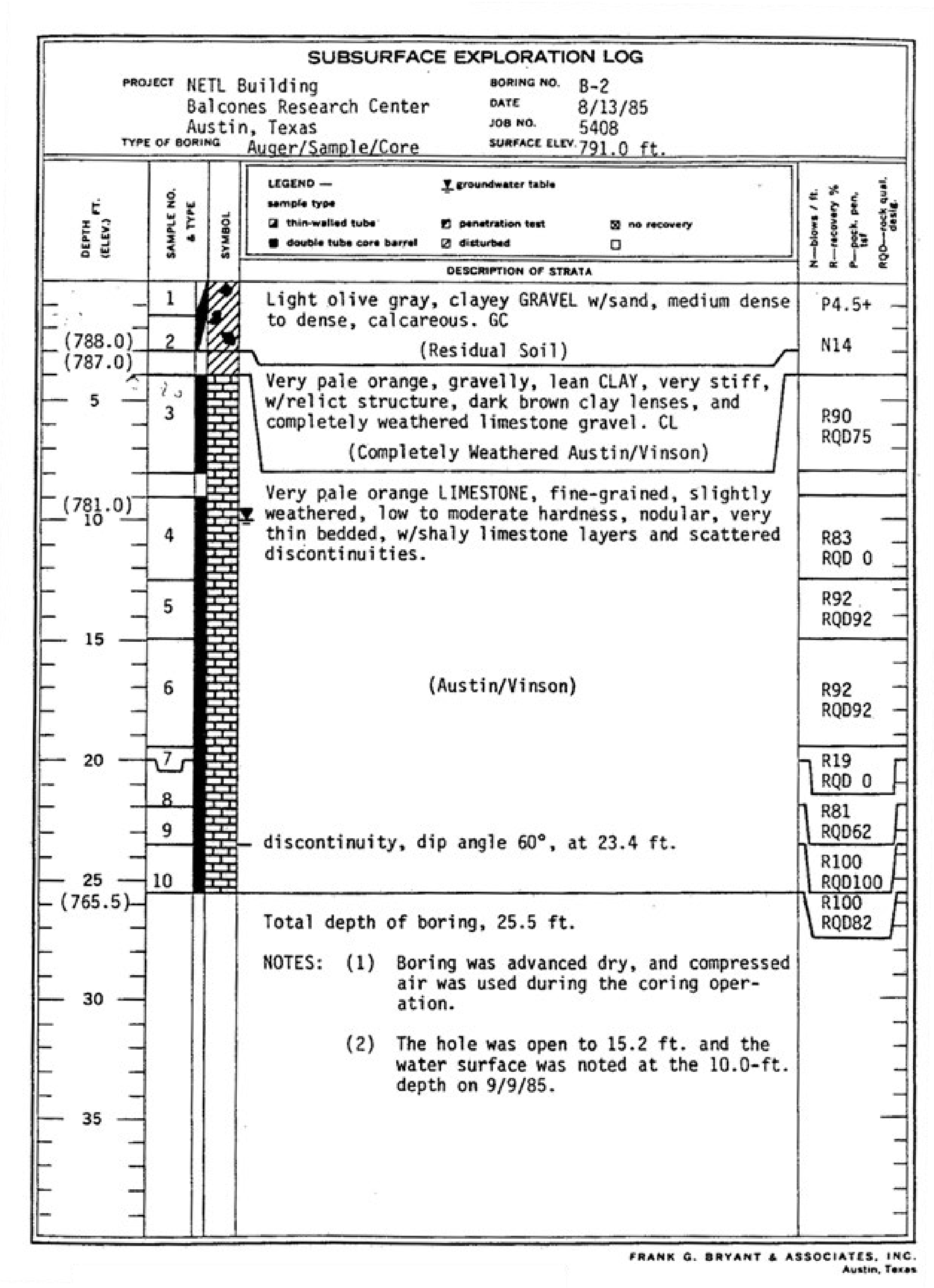


Figure 2.16, Balcones Research Center 1990

APPENDIX 2.1 - SUBSURFACE EXPLORATION LOGS



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.

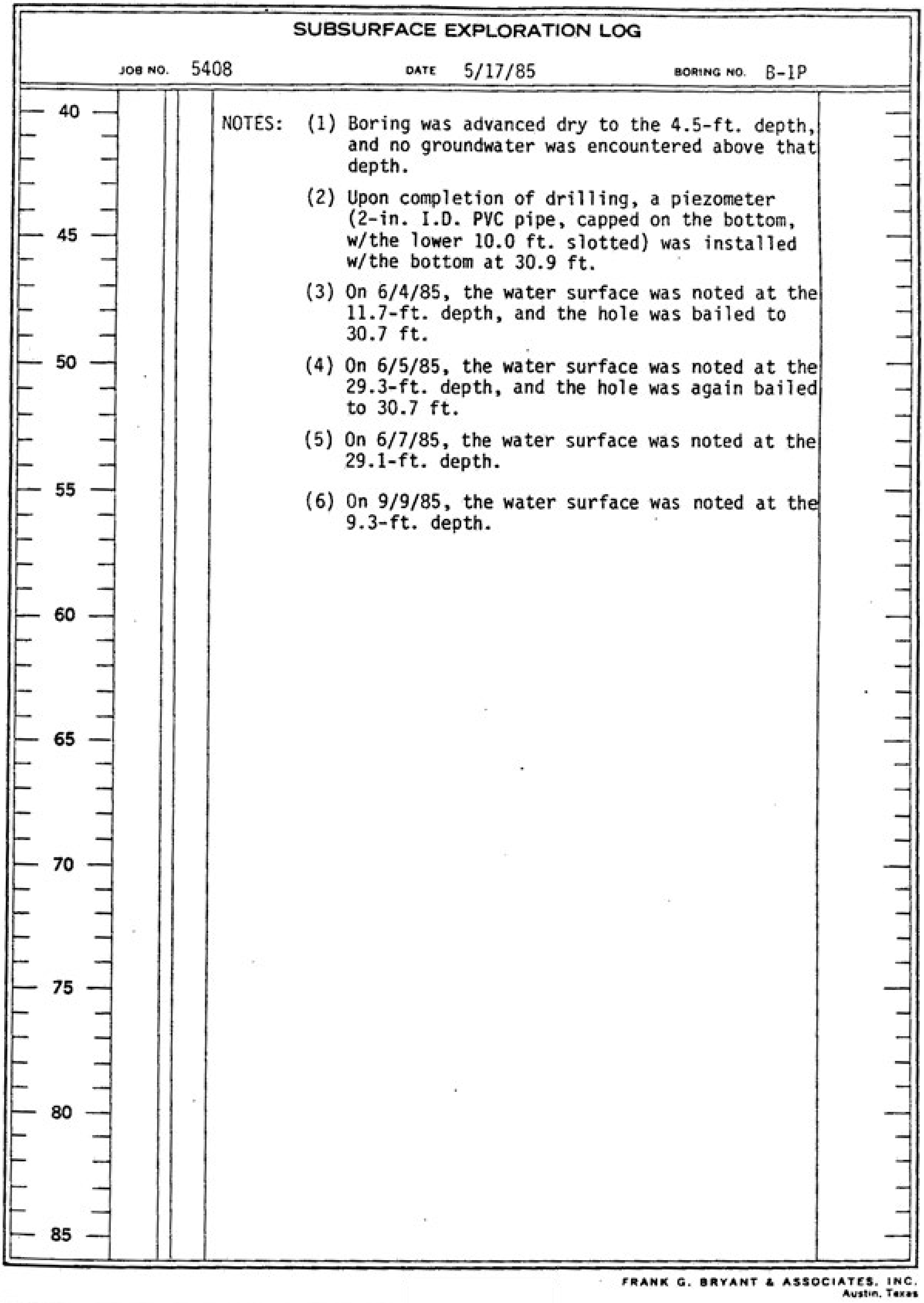
LEGEND — ▼ groundwater table
sample type
◪ thin-walled tube
◩ penetration test
☒ no recovery
■ double tube core barrel
▨ disturbed
☐

DEPTH FT. (ELEV.)	SAMPLE NO. & TYPE	SYMBOL	DESCRIPTION OF STRATA	N—blows / ft. R—recovery % P—pock. pen, tsf RQD—rock qual. desig.
	1		Light olive gray, clayey GRAVEL w/sand, medium dense to dense, calcareous. GC	P4.5+
(788.0)	2		(Residual Soil)	N14
(787.0)				
5	3		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)	R90 RQD75
(781.0) 10	4		Very pale orange LIMESTONE, fine-grained, slightly weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.	R83 RQD 0
	5			R92 RQD92
15	6		(Austin/Vinson)	R92 RQD92
20	7			R19 RQD 0
	8			R81 RQD62
	9		discontinuity, dip angle 60°, at 23.4 ft.	R100 RQD100
25	10			R100 RQD82
(765.5)			Total depth of boring, 25.5 ft.	
30				
35				

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.

FRANK G. BRYANT & ASSOCIATES, INC.
Austin, Texas



SUBSURFACE EXPLORATION LOG

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

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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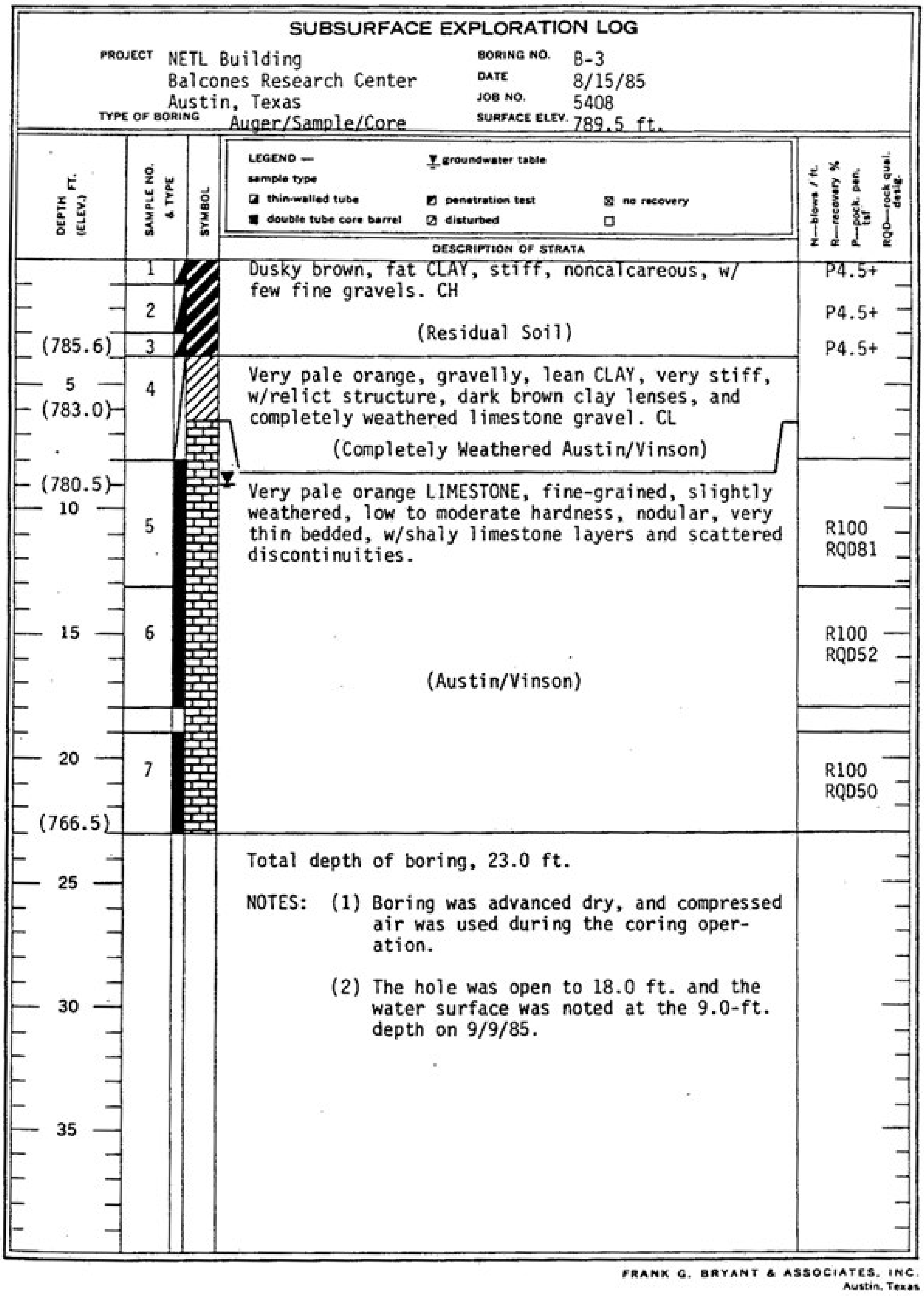
Austin, Texas



SUBSURFACE EXPLORATION LOG

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

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

LEGEND —
sample type
◪ thin-walled tube
■ double tube core barrel
▨ penetration test
▨ disturbed
☒ no recovery
☐
groundwater table

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	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 weathered, low to moderate hardness, nodular, very thin bedded, w/shaly limestone layers and scattered discontinuities.	
10	5		R100 RQD81
15	6	(Austin/Vinson)	R100 RQD52
20	7		R100 RQD50
(766.5)			

Total depth of boring, 23.0 ft.

NOTES: (1) Boring was advanced dry, and compressed air was used during the coring operation.

(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.

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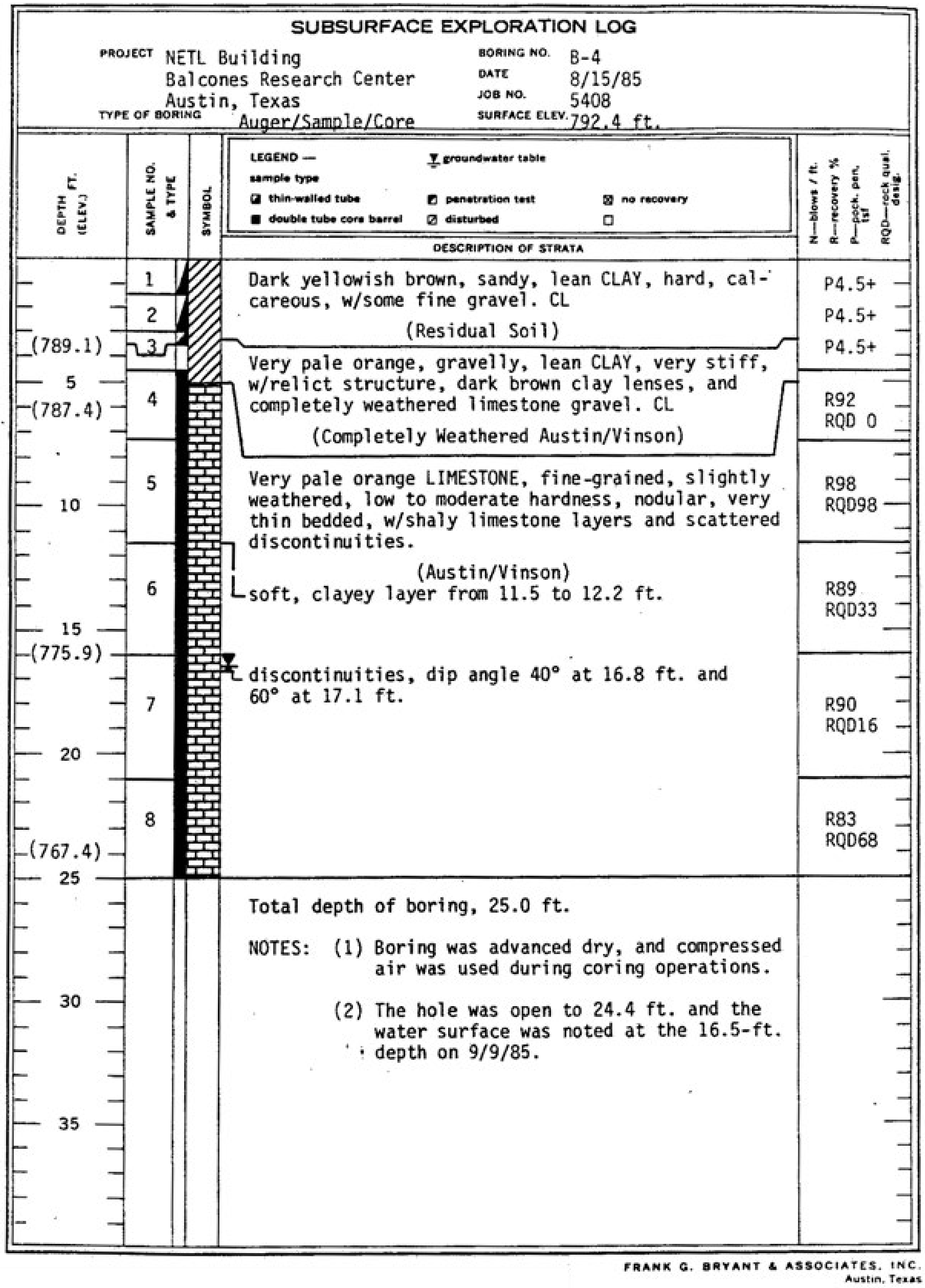
Austin, Texas



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 — sample type
▪ thin-walled tube
■ double tube core barrel
▪ penetration test
▨ disturbed
☒ no recovery
☐
groundwater table

DEPTH FT. (ELEV.)	SAMPLE NO. & TYPE	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 (Residual Soil)	P4.5+
	2		P4.5+
(789.1)	3		P4.5+
5 (787.4)	4	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)	R92 RQD 0
10	5	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)	R98 RQD98
	6	soft, clayey layer from 11.5 to 12.2 ft.	R89 RQD33
15 (775.9)	7	discontinuities, dip angle 40° at 16.8 ft. and 60° at 17.1 ft.	R90 RQD16
20 (767.4) 25	8		R83 RQD68
30 35		Total depth of boring, 25.0 ft.	

NOTES: (1) Boring was advanced dry, and compressed air was used during coring operations.

(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.

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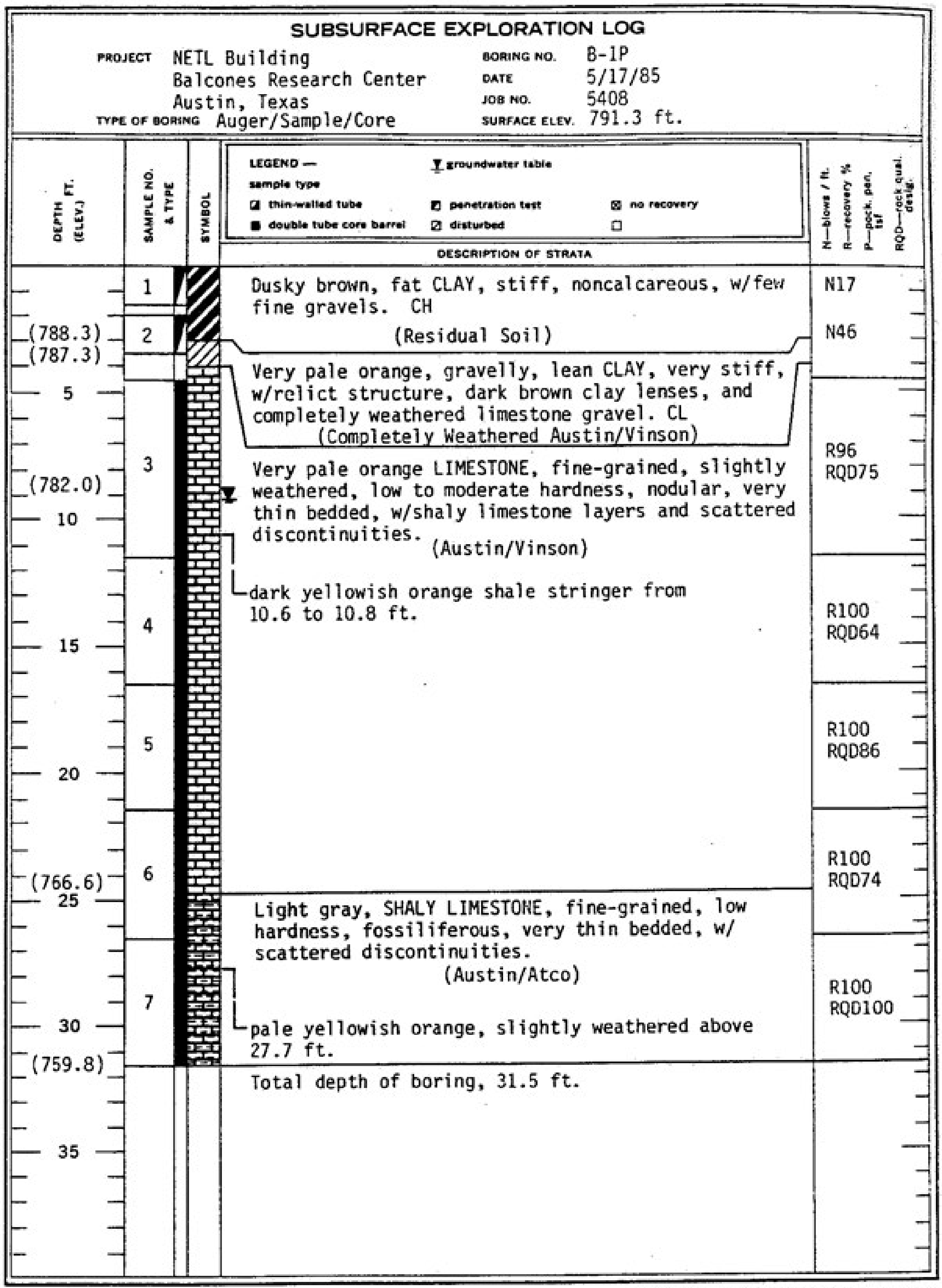

SUBSURFACE EXPLORATION LOG

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

BORING NO. B-1P
DATE 5/17/85
JOB NO. 5408
SURFACE ELEV. 791.3 ft.

LEGEND — sample type
◪ thin-walled tube
■ double tube core barrel
◩ penetration test
▨ disturbed
☒ no recovery
☐
▼ groundwater table

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 (Completely Weathered Austin/Vinson)	
5	3	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)	R96 RQD75
(782.0) ▼			
10		dark yellowish orange shale stringer from 10.6 to 10.8 ft.	
15	4		R100 RQD64
20	5		R100 RQD86
(766.6)	6		R100 RQD74
25		Light gray, SHALY LIMESTONE, fine-grained, low hardness, fossiliferous, very thin bedded, w/ scattered discontinuities. (Austin/Atco)	
30	7	pale yellowish orange, slightly weathered above 27.7 ft.	R100 RQD100
(759.8)		Total depth of boring, 31.5 ft.	
35			

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Austin, Texas

3. 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)[11] includes:

- Excess reactivity addition. Because of the negative temperature feedback associated 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 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.

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

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 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, auxiliary purge)	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 institutions 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[12], 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.

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

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

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[13]. Provisions of the Uniform Building Code[14] 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). The 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 a rock subsurface of limestone. Soil tests of the subsurface set the load capacity at 1690 kg/m^2 (2.4 psi). Concrete piers and footings provide building foundations. 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 specifications 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 thickness of 4” (minimum) under the reactor and neutron generator rooms, with 6” (minimum) under the base footing/slab of the reactor. Subdrainage systems were fabricated using American Society of Testing and Materials (ASTM) D-2665-78 (Poly Vinyl Chloride, PVC, Plastic Drain, Waste and Vent Pipe and Fittings, D-2729-78 (Poly Vinyl Chloride, PVC, Sewer Pipe and Fittings) and appropriate standards for joining (D-2564-78a, D1855-78).

3.1.6.a. Building

The architectural design of the building will develop two separate functional sections, the reactor bay wing and an academic and laboratory wing. The 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 7 feet (2.1 meters) below the mean grade, while the academic wing entry level is 7 feet (2.1 meters) above the mean grade.

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[13] [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)

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

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. The floor is a slab and beam design of reinforced concrete on compacted fill material. All building columns and first level walls are concrete, cast in place with steel enforcement. The reactor bay has a floor-to-roof level of 56.5 feet (17.2 meters). 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 4.25 feet (1.3 meters) thick shield walls 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 4.25 feet 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.

3.1.6.b. Reactor Bay

The 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 meters 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 provided 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 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 principal 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 use 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.

3.4.1. 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.

3.4.2. 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.

3.4.3. 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. REACTOR

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

4.1. SUMMARY DESCRIPTION

The nuclear research reactor at the Nuclear Engineering Teaching Laboratory (NETL) at UT-Austin is a 1.1 MW steady-state TRIGA II reactor with pulsed reactivity insertion permitted up to 2.2% Δk/k. The NETL reactor replaced a TRIGA Mark I reactor that operated for 25 years (1963-1988) in Taylor Hall on the main UT-Austin campus. The NETL reactor achieved initial criticality in 1992. The fuel is standard TRIGA fuel (including instrumented fuel elements, IFEs) and fuel followers attached to control elements. All of the fuel in the initial NETL core except for IFEs and fuel followers had previous operation in the original TRIGA I reactor, with a fraction previously utilized at other TRIGA reactors. A summary of the properties of the current version TRIGA fuel element is provided in Table 4.1 with developments in fabrication over time described in 4.2.1.

Table 4.1, TRIGA Fuel Properties

Property	Mark III Fuel Element
Fuel Element Dimensions	
Outside diameter, $D_o = 2r_o$	1.475 in. (3.7338 cm)
Inside diameter, $D_i = 2r_i$	1.435 in. (3.6322 cm)
Overall length	28.4 in. (72.136 cm)
Length of fuel zone, L	15 in. (38.10 cm)
Length of graphite axial reflectors	3.44 in (8.738 cm)
End fixtures and cladding	304 stainless-steel
Cladding thickness	0.020 in. (0.0508 cm)
Burnable poisons	None
Uranium content	
Weight percent U	8.5
^{235}U enrichment	≤20%
^{235}U content	38 g
Physical properties of fuel matrix	
H/Zr atomic ratio	1.6
Thermal conductivity (W cm^{-1} K^{-1})	0.18
Heat capacity [T ≥0°C] (J cm^{-3} K^{-1})	2.04 + 0.00417T
Mechanical properties of delta phase U-ZrH	
Elastic modulus at 20°C	9.1×10^6 psi
Elastic modulus at 650°C	6.0×10^6 psi
Ultimate tensile strength (to 650°C)	24,000 psi
Compressive strength (20°C)	60,000 psi
Compressive yield (20°C)	35,000 psi

The reactor pool is in an 11,000-gallon aluminum tank surrounded by concrete. [redacted]

Heat generated by reactor operation is removed from the fuel elements to the reactor pool by natural convection. Pool water is cooled in a heat exchanger using a campus chilled water loop. Pool water conductivity is maintained by a purification system with filtration and ion exchange resin. Pool water is principally a heat sink and radiation shield but also supports neutron moderation.

Experimental facilities include:

- Central thimble,
- One facility filling 7 contiguous fuel elements positions,
- Two facilities filling 3 contiguous fuel elements positions,
- One facility filling a fuel element position with a pneumatic sample transfer system position,
- Motorized rotary specimen rack on the upper core periphery,
- Tangential beam port terminating at the edge of the reflector,
- Two Tangential beam ports exiting opposite sides of the concrete shield,
- Radial beam port terminating at the edge of the reflector, and
- Radial beam port terminating at a reflector penetration.

4.2. REACTOR CORE

The reactor core consists of components inserted in grid plate positions and fixed components. The pitch of the grid plate positions is hexagonal, indexed in rings from A to G (Figure 4.1). Components inserted in grid plate positions include reactor fuel, dummy elements, control rods, neutron sources, the central thimble, and experiment facilities that displace fuel elements. Fixed components include core support structure, safety plate, graphite reflector, upper and lower grid plates, central thimble, and components inserted in gird plate positions.

4.2.1. Fuel Elements

The TRIGA fuel is uranium homogeneously distributed in zirconium-hydride. The zirconium-hydride is manufactured with a hydrogen to zirconium nominal ratio of 1.6. The uranium in the zirconium-hydride is 8.5% by weight, the uranium enriched to less than 20%. The hydrogen in the zirconium-hydride supplies a significant fraction of neutron moderation and the fuel with the zirconium-hydride is therefore labeled fuel-moderator material. Standard Fuel Elements (SFEs) consist of annular fuel-moderator material 1.435 in. with a central 0.25 in. diameter hole.[15] The fuel-moderator material is axially centered in a cladding tube with a graphite reflector at each end,

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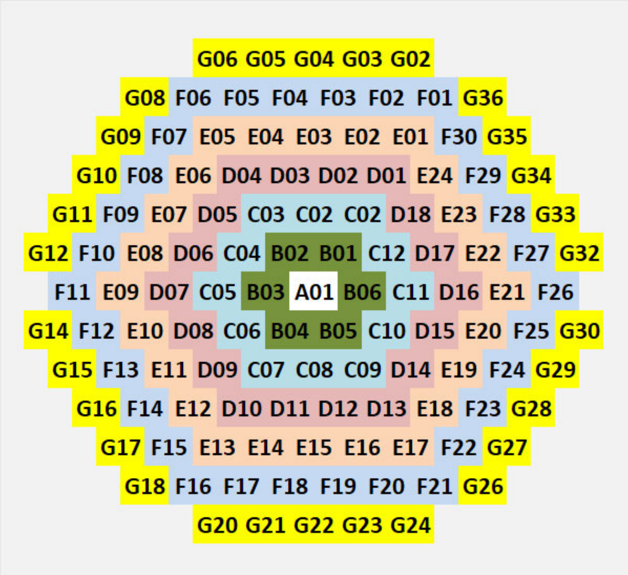
[15] General Atomics Schematic TOS210D213

a gas gap of at least 0.5 inches above the upper reflector and above the lower reflector a molybdenum disc 1.431 inches in diameter and 0.031 inches thick[16]. The molybdenum prevents damage to the lower reflector that could cause disturbing the position of fuel-moderator material. A zirconium rod 0.225 in. in diameter fills the 0.25 in. diameter central cavity. Fuel followers attached to control rods do not have axial graphite reflectors and the fuel-moderator material has a diameter of 1.311 in.[1]



G06 G05 G04 G03 G02
G08 F06 F05 F04 F03 F02 F01 G36
G09 F07 E05 E04 E03 E02 E01 F30 G35
G10 F08 E06 D04 D03 D02 D01 E24 F29 G34
G11 F09 E07 D05 C03 C02 C02 D18 E23 F28 G33
G12 F10 E08 D06 C04 B02 B01 C12 D17 E22 F27 G32
F11 E09 D07 C05 B03 A01 B06 C11 D16 E21 F26
G14 F12 E10 D08 C06 B04 B05 C10 D15 E20 F25 G30
G15 F13 E11 D09 C07 C08 C09 D14 E19 F24 G29
G16 F14 E12 D10 D11 D12 D13 E18 F23 G28
G17 F15 E13 E14 E15 E16 E17 F22 G27
G18 F16 F17 F18 F19 F20 F21 G26
G20 G21 G22 G23 G24

Figure 4.1, Grid Plate Positions

The fuel-moderator is encapsulated in a stainless-steel structure that acts as a barrier to fission product release. End fixtures in standard fuel elements and IFEs are heliarc welded to the top and bottom ends of a 0.020-inch-thick stainless steel cladding tube. The end fittings have been manufactured in three distinct designs (Figure 4.2): original, integral, and streamlined. The original TRIGA reactor design supported the fuel on the lower grid plate with a pin extruding from the bottom end fixture; holes were strategically drilled around depressions which fixed the element locations. The UT reactor is a later version with the fuel elements supported by fins attached to the bottom end fixtures positioned on large (1.25-inch) chamfered holes for cooling. The first generation TRIGA fuel was not fabricated with the support fins; an adapter is required to use the early TRIGA fuel in the UT reactor.

The top end fixtures have structures to center the fuel in the upper grid plate and provide a path for cooling flow through the 1.505-inch holes in the top grid plate. The upper end fixture has a pin fabricated with a rounded top and groove for fuel handling with a ball latch mechanism. For elements with thermocouples embedded in the fuel matrix (Instrumented Fuel Elements or IFEs), the upper end fixture has a passage for lead wires. Axial graphite reflectors in the streamlined

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[16] General Atomics Schematic TOS210B229

version are 1.353 inches in diameter, 2.56 inches long in the upper reflector and 3.72 inches long in the lower reflector.

The three standard fuel element versions and the associated IFEs are illustrated on the right of Figure 4.2 (left). An instrumented fuel element (IFE) and a fuel follower control rod are shown on the right of Figure 4.1. The overall length of the original TRIGA fuel element is 28.37 inches. A second version was developed with integral fins at the bottom end fixture and larger exit channel flow areas by curving the triangular surfaces of the end fixtures. The graphite reflectors in the integral-fin version are 1.43 inches in diameter and 3.420 inches long. Manufacturing of standard fuel elements was modified in 1985 with tapered surfaces added to the end fixture flat surfaces to reduce turbulence of the cooling flow. The overall length of these streamlined elements is 29.68 in. There is a gas gap above the fuel-moderator material in the fuel follower but no other structure or end fixtures comparable to the standard fuel elements and IFEs. Handling, transport, and storage of TRIGA fuel elements at the NETL, fresh and irradiated, are described in Chapter 9, Auxiliary Systems.

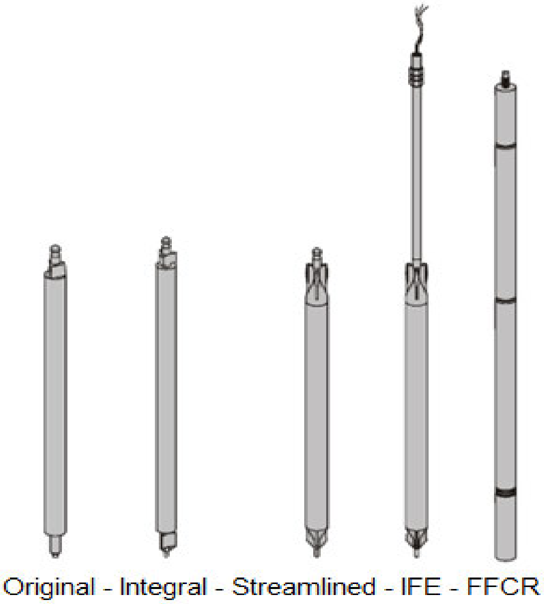

Figure 4.2, Variations in TRIGA Fuel Used at the UT Reactor

4.2.1.a. Fuel Properties

The fuel-moderator material is a solid metal alloy. Increases in reactor power therefore heat the fuel and moderator simultaneously. This increase in temperature reduces the effective ^{235}U cross section, increases the Doppler coefficient of the ^{238}U, and decreases neutron moderation by increasing the energy of hydrogen atoms in the zirconium hydride matrix. Spectrum hardening in the fuel causes an increase in neutron path length in the fuel element making capture in the fuel-moderator material less likely while rapid thermalization in the water encourages capture.

For hydrogen ratios greater than 1.5 the zirconium-hydride matrix is single phase and does not exhibit phase separation with thermal cycling. There is no potential for deformation, swelling, or cracking associated with phase changes. Thermal diffusion of hydrogen is limited at high-hydrogen ratios, limiting the potential for deformation from evolution of hydrogen gas in the matrix.

Fission recoils and products in uranium zirconium alloys causes swelling. Since the uranium is present as a fine dispersion (about 1 μm diameter) recoil damage is limited to about 10 μm - the range of fission recoil. Solid fission products cause swelling growth of about 3%ΔV/V per metal atom % burnup and is relatively insensitive to temperature. Potential fission gas evolution in voids generated by fission recoils does not occur at normal TRIGA operating temperatures. Burnup tests performed by General Atomics in excess of 50% of ^{235}U did not show significant fuel degradation.

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 at GA Technologies show that the zirconium-hydride systems have a relatively low chemical reactivity with respect to water and air[17]. 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. Acceptable[18] 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[19] indicate a value of 1.5×10^{-5} for noble gases, which is in SARs for other reactor facilities[20].

4.2.1.b. Fuel Element Cladding Design

Free hydrogen in the space within the fuel element is heated during reactor operations and pressurizes the interior of the cladding. 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. The classic equation for hoop stress created by internal pressure is:

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

where:

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[17] NUREG/CR-2387 Credible Accidents for TRIGA and TRIGA Fueled Reactors, prepared by S. C. Hawley. and R. L., Pacific Northwest Laboratory, PNL-4208 (1982)

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

[19] Simnad, M.T., F. C. Faushee, and G.B. West, "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology Vol. 28, pp. 31-56 (1976).

[20] NUREG-1390, "Safety Evaluation Report Relating to the Renewal, of the Operating License for the TRIGA Training and Research Reactor at the University of Arizona," Report NUREG-1390, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1990.

σ_θ is the hoop stress,

P is internal pressure,

r is inside radius, and

t is the wall thickness.

For 1.5 in. diameter and 0.02 in. wall thickness, stress is 36.7 times the internal pressure. Figure 4.3a provides temperature dependent ultimate strength and the 0.2% yield, and Figure 4.3b 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; 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 condition. 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.

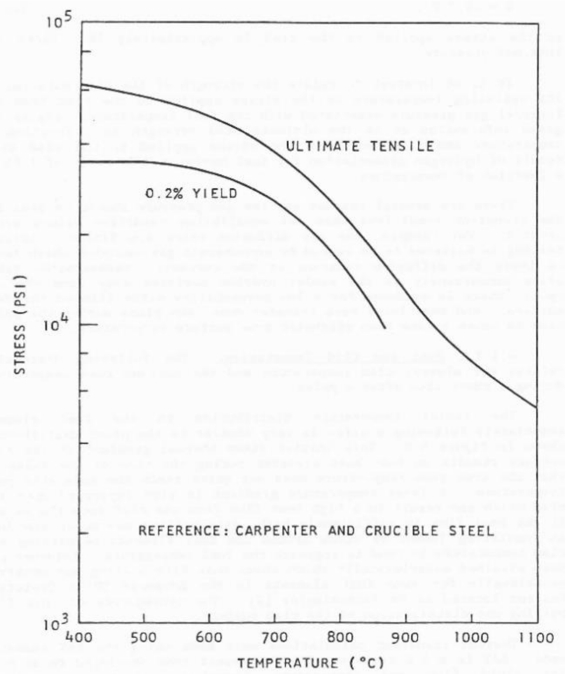

Figure 4.3a, Temperature, Cladding Strength, and Stress

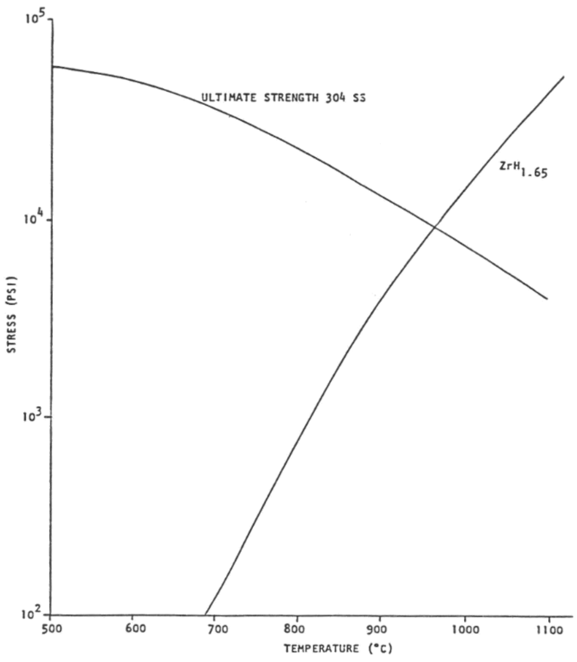

Figure 4.3b, Temperature, Cladding Strength for 0.2% Yield

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[21]. Based on a fuel failure of TRIGA fuel in a conversion

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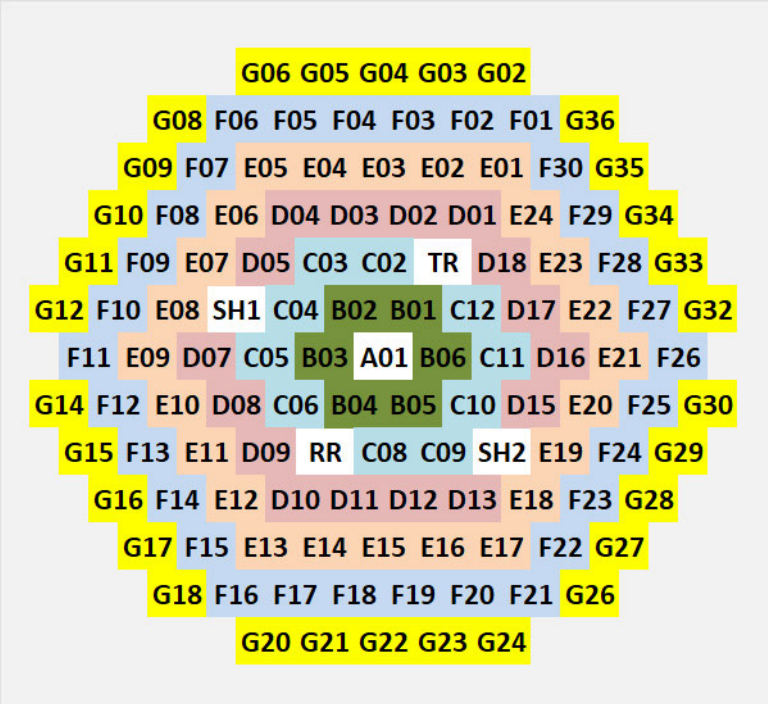
[21] "Annual Core Pulse Reactor," General Dynamics, General Atomics Division report GACD 6977 (Supplement 2), Dee. J. B., et. al. (1966)..

core during pulsing operations, General Atomics has recommended[22] a pulsing temperature limit of 830°C.

4.2.2. Control Rods and Control Rod Drive Mechanisms

The control rods and drive mechanisms consist of control rods, standard (or stepper) control rod drives, transient rod drives, and control functions. The UT TRIGA reactor currently has 4 control rods, three standard rods magnetically coupled to the control rod drive (regulating rod labeled RR, Shim1 labeled S1, and Shim 2 labeled S2 in Figure 4.4), and one pulse rod (labeled TR in Figure 4.4) pneumatically coupled to the control rod drive.



G06 G05 G04 G03 G02
G08 F06 F05 F04 F03 F02 F01 G36
G09 F07 E05 E04 E03 E02 E01 F30 G35
G10 F08 E06 D04 D03 D02 D01 E24 F29 G34
G11 F09 E07 D05 C03 C02 TR D18 E23 F28 G33
G12 F10 E08 SH1 C04 B02 B01 C12 D17 E22 F27 G32
F11 E09 D07 C05 B03 A01 B06 C11 D16 E21 F26
G14 F12 E10 D08 C06 B04 B05 C10 D15 E20 F25 G30
G15 F13 E11 D09 RR C08 C09 SH2 E19 F24 G29
G16 F14 E12 D10 D11 D12 D13 E18 F23 G28
G17 F15 E13 E14 E15 E16 E17 F22 G27
G18 F16 F17 F18 F19 F20 F21 G26
G20 G21 G22 G23 G24

Figure 4.4, Control Rod Positions

4.2.2.a. Control Rods

The standard/stepper 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 with the uppermost 6.5 in. (16. 5 cm) section an air void, followed by 15 in. (38.1 cm) of solid boron carbide, the neutron poison. Standard control rods have a fuel follower attached below the poison section so that as the control rod is withdrawn from the core the water channel is filled with a fuel element as illustrated in Figure 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. The transient (safety-transient or pulse) rod is a sealed, 36.75 in. (93.35 cm) long

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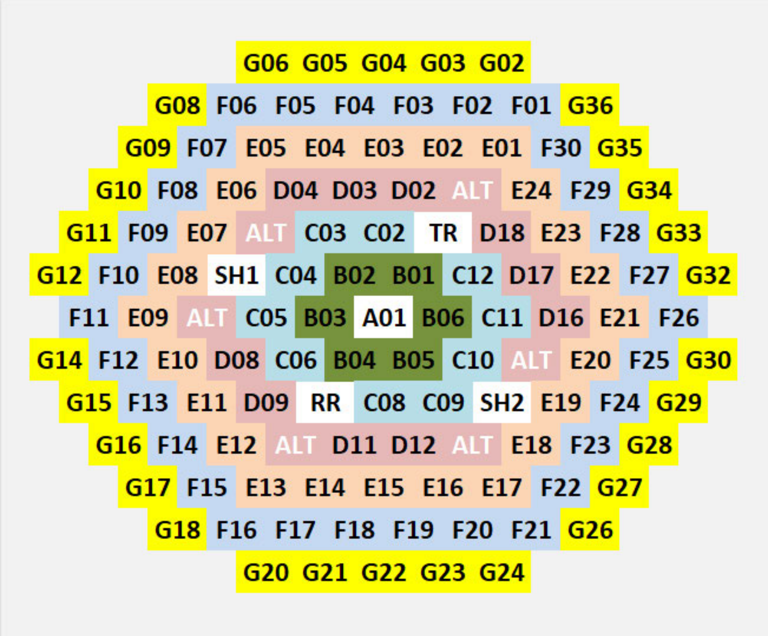
[22] General Atomics-ESI, “Pulsing Temperature Limit for TRIGA LEU Fuel,” Argonne National Laboratory TRD 070.01006.05 Rev A. (April 2008).

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 (Figure 4.6a). 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. Supported by the safety plate, a locking device is built into the lower end of the assembly. The guide tube receives lateral positioning from the upper and lower grid plates and extends approximately 10 in. (25.4 cm) above the top grid plate. Water passage through the tube is provided by holes distributed evenly over its length. Sections of all control rod are separated and secured by 1-inch magneform fittings.

While there are no current plans to change the control rod inventory or configuration there are viable options for alternate control rod locations as indicated in Figure 4.5.



G06 G05 G04 G03 G02
G08 F06 F05 F04 F03 F02 F01 G36
G09 F07 E05 E04 E03 E02 E01 F30 G35
G10 F08 E06 D04 D03 D02 ALT E24 F29 G34
G11 F09 E07 ALT C03 C02 TR D18 E23 F28 G33
G12 F10 E08 SH1 C04 B02 B01 C12 D17 E22 F27 G32
F11 E09 ALT C05 B03 A01 B06 C11 D16 E21 F26
G14 F12 E10 D08 C06 B04 B05 C10 ALT E20 F25 G30
G15 F13 E11 D09 RR C08 C09 SH2 E19 F24 G29
G16 F14 E12 ALT D11 D12 ALT E18 F23 G28
G17 F15 E13 E14 E15 E16 E17 F22 G27
G18 F16 F17 F18 F19 F20 F21 G26
G20 G21 G22 G23 G24

Figure 4.5, Lower Gird Plate Control Rod Positions

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 4.2.

Control rod worth is a function of neutron flux at the control rod, control rod materials and dimensions, and control rod burnup. The estimated control rod worths from the 1991 preliminary safety report are listed in Table 4.3, along with the worth of each control rod as measured in June 2011.

Table 4.2, 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-ZrH$_{.16}$	
OD	1.25 in.	3.18 cm	1.31 in	3.34 cm
Length	20.88 in.	53.02 cm	NA	NA

Table 4.3, Control Rod Information

Rod	Location	Diameter		Estimated (1991)		Measured (2011)
		In.	cm.	% $\Delta k/k$	$	$
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

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 as shown in Figure 4.6b. All fuel element position penetrations in the upper grid plate are identical. The lower grid plate has a set of 11 penetrations in the C and D rings (black label, white background in Figure 4.5) representing the current control rod positions with alternate positions (all in the D ring, labeled with a white ALT) with the same diameter as the upper grid plate. One of these penetrations is 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 so that the control rod cannot exit the core region in the downward direction.

A threaded fitting at the top of each control rod connects to a series of extensions that link to the control rod drive mechanisms mounted on the bridge spanning the reactor pool. The top section of the control rod extension rests on the bottom of a tube supported by the control rod drive housing so that the bottom of the control rod does not impact the safety plate. Slots in the tube provide a hydraulic cushion for the rod during a scram.

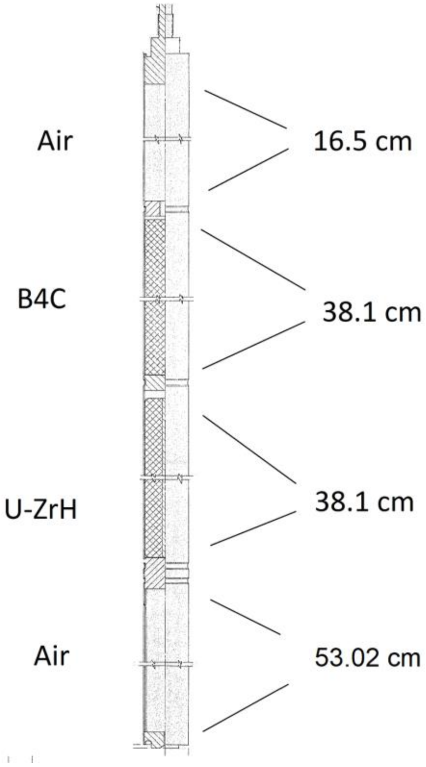


Figure 4.6a, Transient Rod

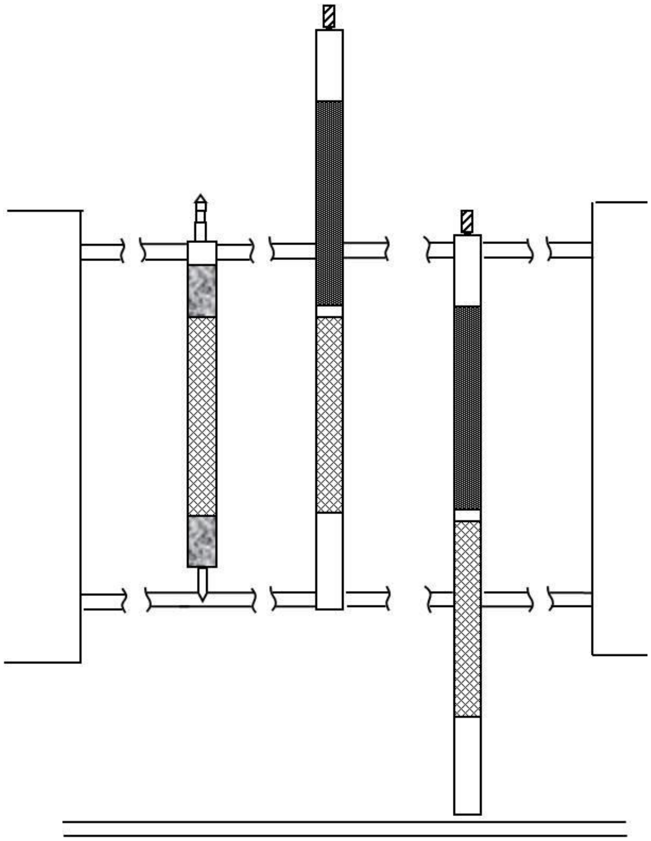


Figure 4.6b, Standard Control Rod Configuration

The shaft is secured to a cylinder that rests on the bottom of the housing when the rod is fully inserted (see Figure 4.7). 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 (see Figure 4.8), 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.

4.2.2.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.

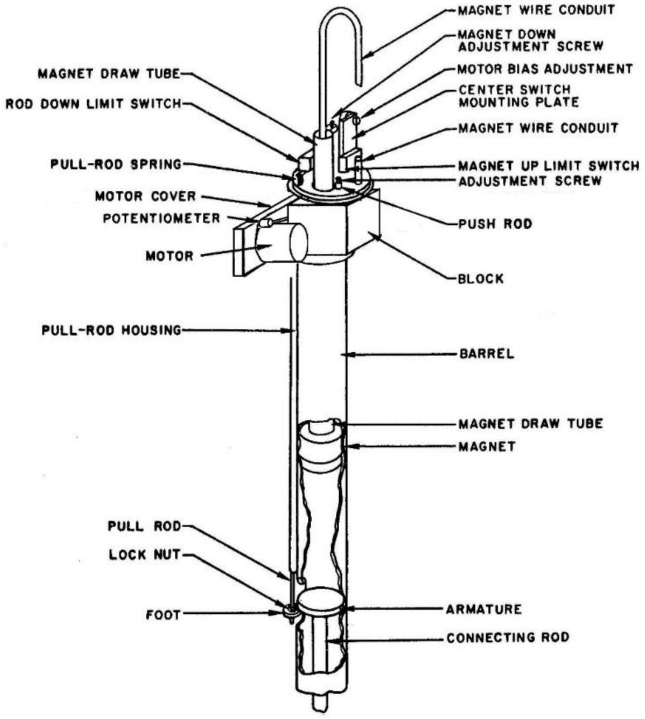


Figure 4.7, Standard/Stepper Motor Control Rod Drive

The pinion gear engages a rack attached to the magnet draw tube. An electromagnet, attached to the lower end of the draw tube provides force to hold 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 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 armature, connecting rod, and the control rod will then drop to reinsert the control rod in the core.

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 indicates a possible travel rate of 33 inches pe minute (1.40 cm/s). Measurements of the actual rate provide a speed of 27 inches per minute (1.14 cm/s).

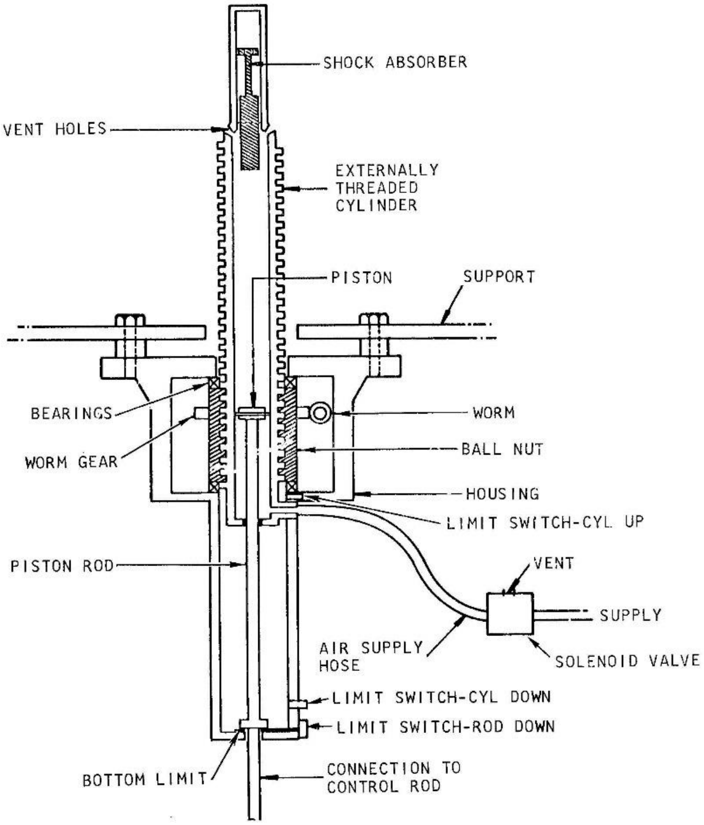


Figure 4.8, Transient Rod Drive

4.2.2.c. Transient Control Rod Drive

The transient rod is a scrammable rod operated from the reactor control console 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 and released during a scram. The reactor control console prevents firing of the transient rod drive for a pulse unless conditions for pulsing are met.

The transient rod drive uses a single-acting pneumatic cylinder with a piston connected 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.

Rod withdrawal speed is about 28 inches per minute (1.19 cm/s). The extent of transient rod withdrawal from the core during a pulse is controlled with the distance the piston travels when air is applied by raising or lowering the de-coupled cylinder. The cylinder has external threads that 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.

Attached to and extending downward from the transient rod drive housing is the rod guide support, which serves multiple 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 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 provides a signal that the rod is in the core. A scram signal de-energizes the solenoid valve which vents the air required to hold the rod in a withdrawn position. The transient rod drops into the core from the full out position in less than 1 second.

4.2.2.d. Control Functions

Instrumentation and controls provide protective actions through the control rod system, as described in Table 4.4. A trip signal from the reactor protection system or the reactor control systems will de-energize the electromagnets and the pulse rod air solenoid valve initiating automatic insertion of the control rods.

Table 4.4, Protective Actions

Measuring Channel	Required Trip Setpoint Steady State	 Pulse
Maximum thermal power	1100 kW	2000 MW
Power Channel High power	110%	110%
Detector High Voltage	80%	80%
High Fuel Temperature	550°C	550°C
Initiating Channel	Condition (All Modes)	
Magnet Current	Loss of Current	
Watchdog Timer	Loss of Communication	
Manual Scram	Operator Action	

The reactor control system (described in Chapter 7) has interlocks to prevent various conditions from developing. Table 4.5 is a summary of the functions.

Table 4.5, 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 greater than 10 kW

These settings are conservative; the consequence of normal or abnormal operation that causes a scram would result in fuel and cladding temperatures well below the safety limits of the reactor design bases.

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.

4.2.2.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 control rod speed or reactivity addition rates for control rods to ensure 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 only used to control the power level in steady state or pulsing operation and for intentional shutdown of the reactor. 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

The UT TRIGA core is supported within a reflector assembly shown in Figure 4.9a/b. The reflector assembly supports an upper grid plate, core barrel and reflector, and lower grid plate. 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.

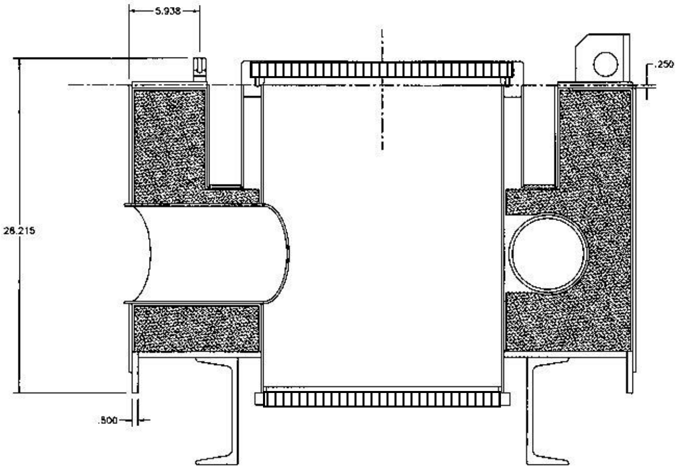


Figure 4.9a, UT TRIGA Core

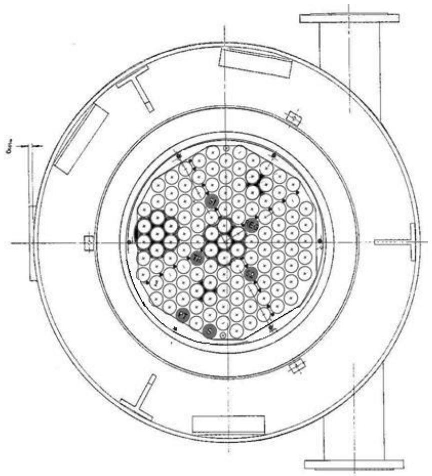


Figure 4.9b, Core Top View

4.2.3.a. *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 (Figure 4.10a), 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 (Figure 4.10b), and aluminum 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 4.10a, Reflector Top Assembly

Figure 4.10b, Reflector Bottom Assembly

Figure 4.11a, Graphite Reflector, Through Port

Figure 4.11b, Graphite Reflector Through Port Detail

Figure 4.11c, Graphite Reflector, Radial & Piercing-Beam Ports

Figure 4.12a, Tangential Beam Port Insert

Figure 4.12b, Radial Beam Port insert

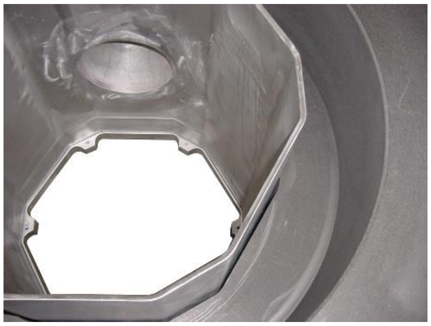


Figure 4.12c, Inner Shroud Surface

The lower radial reflector is constructed of graphite contained in a welded aluminum canister. The graphite is machined to accommodate two beam ports oriented radially from the center of the reactor core, with one “through port” (Figure 4.10b) and a 10 in. (25.3 cm.) cylinder cut from the inner surface to allow the well in the graphite to be used as an experimental facility.

The through port has a rectangular air-filled cut out between the core shroud and the beam port penetration (Figure 4.11b). Aluminum canisters that mate with the beam ports nest in the reflector in two of the beam ports, one radial and one tangential (Figure 4.12a, Figure 4.12b). The third beam port (radial) penetrates the core shroud (Figure 4.12c).

4.2.3.b. 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 (22 in.) and are clad with aluminum.

4.2.3.c. Axial Reflector

Graphite cylinders are integral to fuel elements as previously discussed.

4.2.4. Upper and Lower Grid Plates

The upper and lower grid plates establish positions for in-core components.

4.2.4.a. Upper Grid Plate

The upper grid plate provides alignment for fuel elements and control rods, and (in conjunction with the top fuel assembly end/fixtures) 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 inches (55.245 cm) diameter, anodized to resist wear and corrosion. The top of the upper grid plate is 59 inches (150 cm.) above the bottom of the pool. Fuel positions, each 1.505 inches (3.823 cm) diameter are machined on a triangular pitch of 1.714 inches (4.35 cm) forming 6 hexagonal rings around the center position. 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 inches (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^{o}) to 0.31 inches (11 mm). The center fuel element position is reserved as an experimental facility. Upper grid plate penetrations are summarized in Table 4.6.

Table 4.6, Upper Grid Plate Penetrations

Penetration	Diameter	
	in	cm
Fuel Elements	1.505	3.8227
3-element	2.4	6.098
6/7-Element	4.4	11.176
Upper grid plate alignment	3/8	0.9525
Flux probes	0.203	0.5156

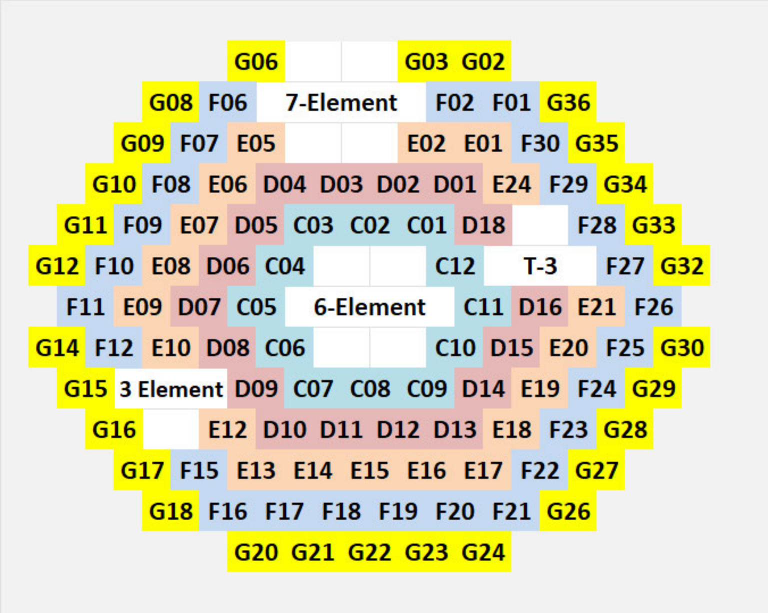


Figure 4.13: Location of Experiment Facilities that Displace Fuel

The upper 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 inch (0.953 cm) holes centered along each of the hexagonal faces of the G ring fuel positions. Circular cutouts to replace fuel element positions are fabricated in the upper grid plate with 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) as illustrated in Figure 4.13. These positions may be filled with fuel, water filled, or filled with irradiation canisters. Fuel element positions in these facilities are established using 0.62 inch (1.575 cm) thick inserts (referred to as "spiders", Figure 4.14a/b). The inserts have standard 1.505-inch penetrations to position fuel elements when tabs on the inserts fit into slots milled in the circular grid plate cutouts. Removing the inserts allows using the space previously occupied by fuel elements as experiment facilities. There are two locations fabricated for each design. The 6/7 element facilities permit specimens as large as 4.4 inches (11.8 cm) and the 3 element facilities permit specimens as large as 2.4 inches (6.1 cm).

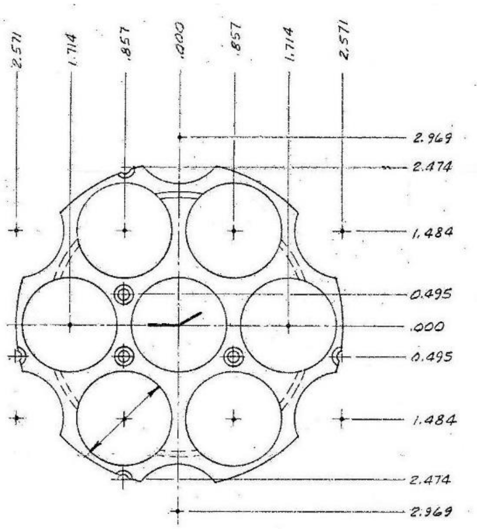


Figure 4.14a, 6/7-Element Facility Grid

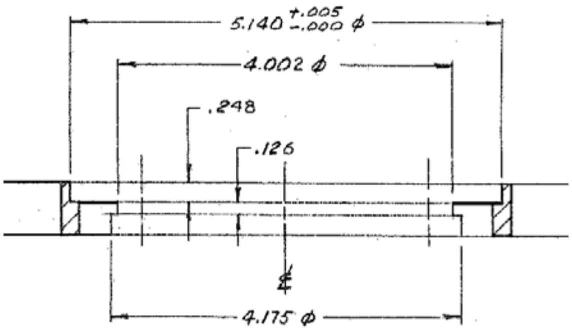


Figure 4.14b, Upper Grid Plate Cut-out for 6/7-Element Grid

In addition to the four positions for control rods and the positions for experiment facilities that replace fuel positions, the current core configuration uses one position for a neutron source and one position for a pneumatic facility. The source and pneumatic tube positions are usually in (but not restricted to) the G ring. Fuel element positions not occupied may be left as water voids.

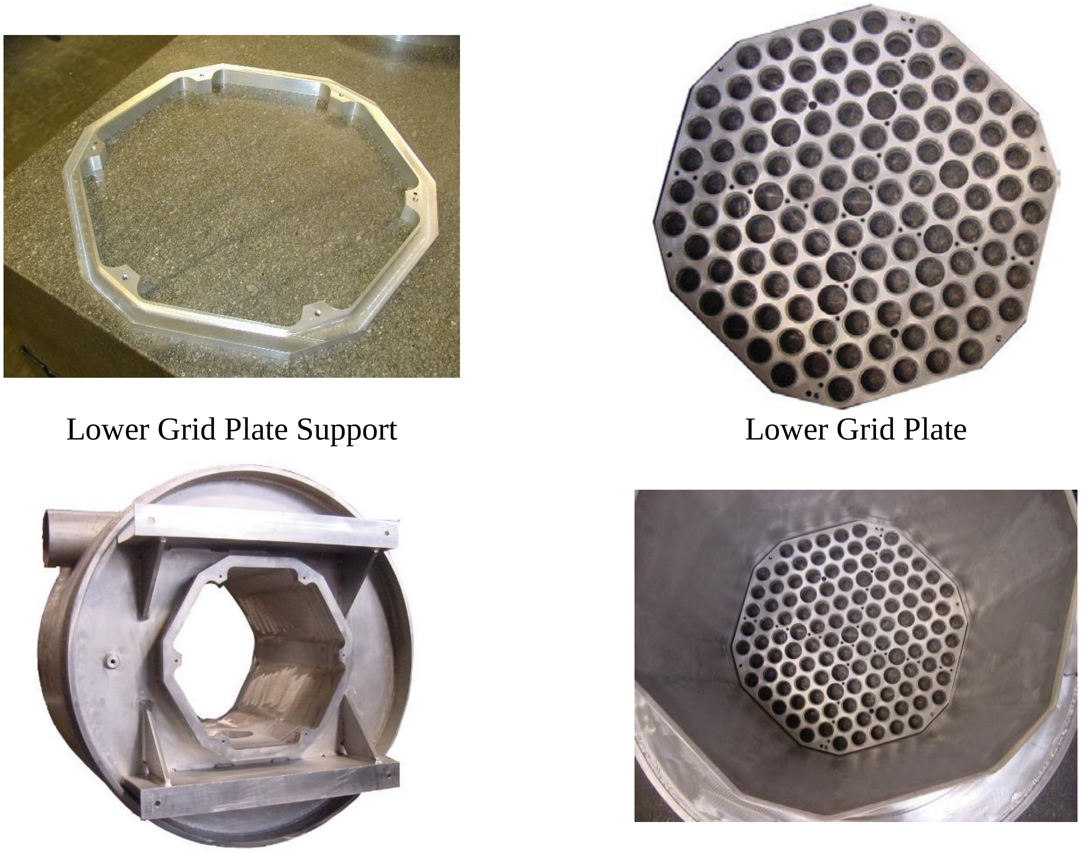
Control rod worth and power level instrument calibration are a function of neutron flux distribution. A change in the 6-element, 7-element, or T-3 facilities require calibration of the control rod worths and power level instrumentation, but the 3-element facility has been demonstrated to have minimal effect on flux distribution.

4.2.4.b. Lower Grid Plate

The lower grid plate (Figure 4.15) 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.

Table 4.7, Lower Grid Plate Penetrations

Penetration	Diameter	
	In.	cm
Central thimble	1.505	3.227
Control Rod	1.505	3.227
Fuel Position	1.250	3.175
Flux Hole Probes	1/3	0.80
Lower Grid Plate Alignment	3/8	0.9525



Lower Grid Plate Support

Lower Grid Plate

Reflector Canister Bottom View

Grid Plate in Core Shroud

Figure 4.15, Reflector Component and Assembly Views

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 4.7). 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 elements.

Eleven 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 (Figure 4.16).

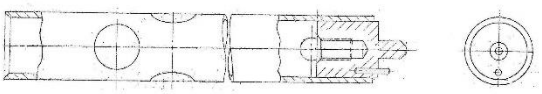


Figure 4.16, Fuel Element Adapter

4.2.5. Neutron Startup Source

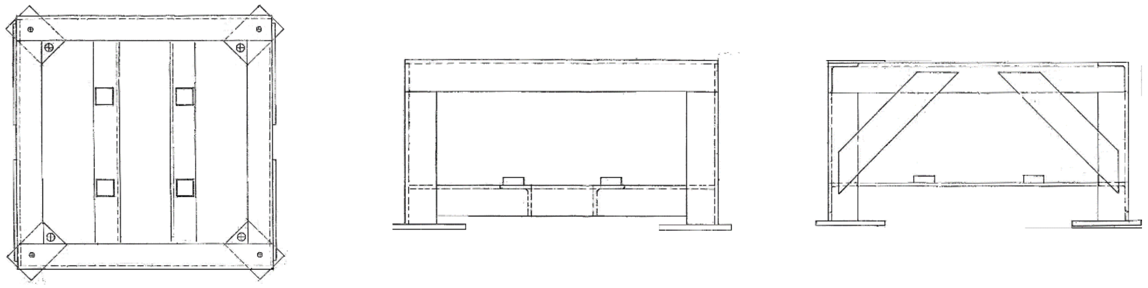
The reactor license permits the use of sealed neutron sources, including a 6 Ci polonium-beryllium (PuBe) and a 2 Ci curie americium-beryllium (AmBe) neutron source. The 2 Ci AmBe source currently in use at the UT TRIGA (serial 618AM369) is a standard sealed neutron source, encapsulated in stainless 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. The source is positioned in a cylindrical cavity 0.981 inches (2.492 cm.) in diameter and approximately 3 inches (7.62 cm.) deep. 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. Recent reactivity measurements indicate the source assembly has a worth of about 3 cents.

4.2.6. Core support structure

The core support structure includes a platform supporting the reflector and core structure, and a “safety plate” that prevents the control rods from falling out of the core if the control rods should be disconnected from the extension assembly.

4.2.6.a. Core Support Platform

The reflector assembly rests on a platform (Figure 4.17) 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. Aluminum 6061-T651 plate is used for safety plate support pads (¾ inch, 1.905 cm), cross braces (⅜ inch, 0.953 cm.), and platform support pads (½ inch, 1.27 cm.). Angle aluminum is inserted 9 inches (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 4.17, 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).

4.2.6.b. Safety Plate

The safety plate (Figures 4.18 and 4.19) limits the distance that a control rod can fall to less than 17.44 inches (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).

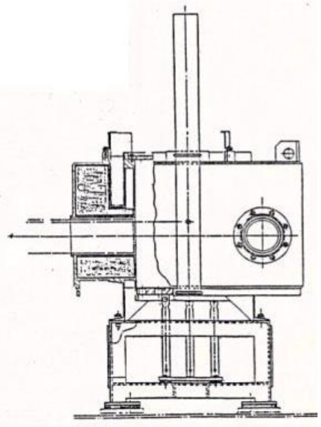


Figure 4.18, Core and Support Structure

The top of the safety plate is 16 inches (40.6 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.

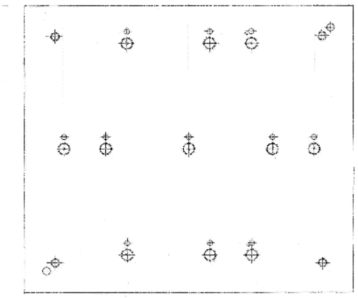


Figure 4.19, Safety Plate

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. 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. Table 4.8 summarizes reactor coolant system design.

Table 4.8, 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	1000 kW
	Flow Rate (shell)	250 gpm (15.8 lps)
	Flow Rate (tubes)	400 gpm (25.2 lps)
Typical Heat Exchanger Operating Parameters	Tube Inlet	100 °F 42 psia
	Tube Outlet	69 °F 27 psia
	Shell Inlet (Shell Inlet)	48 °F 55 psia
	Shell Outlet	67 °F 48 psia

The pool tank (Figures 4.20a/b/c) is fabricated from sheets of 0.25 inch (0.635 cm) 6061 aluminum in 4 vertical sections welded to a ½ inch thick horizontal 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 ¼ inch (0.635) thick 2x2 in. (5x5 cm) square aluminum channel was rolled and welded to the upper edge of the tank.

[redacted]

Figure 4.20a, Pool Figure 4.20b, Side View Figure 4.20c, Top View

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 sections 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 beam port sections are [redacted] in diameter, with a 0.25 inch (0.635 cm) wall thickness. The in-pool sections for BP 1 and 5 are extended into the pool 6 inches (15 cm), while the remaining in-pool beam port sections are much longer. Supports (2x2x¼ in., 5x5x0.635 cm, aluminum angle brackets) 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 16.94 in. (43 cm) into the pool with the support 5 in. (12.7 cm) from the in-core end. BP 3 extends 28.75 in. (73 cm) into the pool with the support 14.813 in. (37.62 cm) from the in-pool end. BP 4 extends 16.9 in. (43 cm) into the pool with the support 3 in. (7.62 cm) from the in-pool end. Beam ports 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. Beam ports 1 and 5 are equipped with bellows to accommodate thermal expansion of the 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 13.5 in. (34.29 cm) 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. [redacted] 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 [redacted] outside of the area defined 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. (17.145 cm) 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. The welded and bolted structural support assures stability in the unlikely occurrence of an earthquake event.

Detection of potential pool leakage could occur in multiple 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. Changes in makeup requirements or increases in makeup water that do not correspond to the history of power level operation may indicate a primary pool-leak.
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

The pool water and shield structure (Figure 4.21) 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. Design of the reactor pool was of ½ inch (1.27 cm) base plate and ¼ inch (0.635 cm) wall plate of 6061 aluminum alloy. 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.

[redacted]

Figure 4.21, Biological Shielding, Base Dimensions

A [redacted] thick foundation pad supports the reactor pool and shield structure. Standard weight concrete, 150 lb/ ft^3 (2.33 g/cm^3), comprises the foundation pad. High density concrete, 180 lb/ft^3 (2.89 g/cm^3), with a magnetite aggregate is the shield material of the first level of the shield structure. The core is shielded radially by a minimum of [redacted] of concrete with a density of 2.88 g/cc, 1.5 feet (45 cm) of water, and 10.2 inches (25.9 cm.) of graphite reflector. A transition from high density to standard density concrete is present about 4.5 feet (1.4 m) above the mid-level platform of the shield. The top part of the shield stem and the top-level platform are standard density concrete. The total shield weight is 2.03 x 10^6 lbs. (920 metric tons). Approximately 24,400 lbs. (11,100 kg) of structural steel, 56 conduits for signal and electrical lines with diameters of ½ to 3 inches, three central junction boxes and numerous local junction boxes are part of the shield system. Five beam tubes at the level of the reactor provide experimental access to reactor neutron and gamma radiation. 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 4.9.

Table 4.9, Significant Shielding and Pool Levels

Parameter of Interest	Level (meters)	Notes
Concrete Pad	[redacted]	[redacted]
Pool Floor	[redacted]	[redacted]
Safety Plate	[redacted]	[redacted]
Grid Plate	[redacted]	[redacted]
Core Bottom	[redacted]	[redacted]
Beam Port CL	[redacted]	[redacted]
Core CL	[redacted]	[redacted]
Core Top	[redacted]	[redacted]
Grid Plate	[redacted]	[redacted]
Main Lower Shielding	[redacted]	[redacted]
Transitional Concrete Step	[redacted]	[redacted]
Top of High-Dens. Concrete	[redacted]	[redacted]
Min, Core Level (TS)	6.5	5.25 ft over the core
Vacuum Breakers	6.7	
Low Pool Level Scram	7.8	
Low Pool Level	8.05	Min. required by operating procedures
Low Pool Level Alarm	8.07	Low level alarm
Normal Pool Level	8.1	Nominal operating level
High Pool Level	8.15	Maximum operating level
High Pool Level Alarm	8.17	High level alarm
Top of 24 ft. Concrete	8.534	

4.5. NUCLEAR DESIGN

Nuclear performance of the NETL TRIGA reactor is documented in *Analysis of the Neutronic Behavior of the Nuclear Engineering Teaching Laboratory at the University of Texas* (Appendix 1 of Chapter 4).

4.5.1. Normal Operating Conditions

Core excess reactivity is managed by a combination of fuel, water voids, graphite rods, and experiment facilities. The fuel in the core inventory and configuration is adjusted or augmented (by lightly burned or unirradiated fuel elements) when excess reactivity is inadequate to support experiment needs. The fuel element positions in each ring are not radially symmetric, but calculations for the worth of nominal average of fuel elements for each ring are listed in Table 4.10.

Table 4.10, Nominal Fuel Element Worth Compared to Water Void

Position	Reactivity Worth		No.
Ring	δk	$	Positions
B	1.07%	$1.53	6
C	0.85%	$1.21	12
D	0.54%	$0.77	18
E	0.36%	$0.51	24
F	0.25%	$0.36	30
G	0.19%	$0.27	36

Measurements recorded in experiment authorizations for the pneumatic tubes in a G ring location, the T-3, 3-element, and 7-element facilities are provided in Table 4.11.

Table 4.11, Experiment Facility Worth Measurements

Pneumatic Sample Transit (PNT) Facilities	
Cd Lined	-$0.30
Bare	Negligible
T-3 Position	
Water Filed Void	-$1.95
Cd Lined	-$1.08
Bare	
3 Element Position	
Cd lined	-$0.59
Bare	Negligible
7 Element Facility	
Water Filled Void	-$0.26
Lead Sleeve	+$0.08
Sleeve & Canister	-$0.33

Information on the number, types, and locations of all core components including fuel, control rods, neutron reflectors, moderators, and in-core experimental components are described in Chapter 4.2 and Chapter 10. The design requirements (including reactivity limits) and dynamic features of the control rod are described in Chapter 4.2.2. Experiments are evaluated for effects on core reactivity before approval. The basic parameter which allows the TRIGA reactor system to operate safely with large step insertions of reactivity is the prompt negative temperature coefficient associated with the TRIGA fuel and core design. This temperature coefficient allows significant freedom in steady-state operation as the effect of incidental reactivity changes occurring from the experimental devices in the core is greatly reduced.

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 4.12). Control rod worth is influenced by the experiment configuration, with significant impact from the large in-core irradiation sites. Table 4.12 also provides the worth of the control rods in the current configuration (3 element facility in E11, F13, and F14). Changes 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 4.12, Control Rod Worth

Control Rod	Reference			Current (June 2023)	
	Position	Worth		Position	Worth
Transient rod	C ring	2.1% Δk/k	\$3.00	C-1	\$3.16
Regulating rod	C ring	2.6% Δk/k	\$3.71	C-7	\$2.72
Shim 1	D ring	2.0% Δk/k	\$2.86	D-14	\$2.83
Shim 2	D ring	2.0% Δk/k	\$2.86	D-6	\$3.36

Initial criticality was accomplished in 1992. The NETL is a very active facility with 18 discrete core configurations between initial criticality and 2018, when a model was developed using MCNP-6 for core analysis. The MCNP model was used to simulate eighty critical core configurations (calculating expected criticality with the control rod models at their measured critical position). The deviation between the calculated criticality of the core and the recorded critical condition was used to evaluate mode bias with results shown in Figure 4.2. The results for the last 40 configurations show reasonable agreement between the model and recorded data; however, some of the data from the first forty configurations contained significant statistical outliers (Figure 4.22). All fuel used in the initial core except for fuel followers had been used previously in anywhere from one to three other reactor facilities before use at NETL. The exact progeny of those fuel elements was not well known and so as the fuel ha been operated at NETL the MCN-6 model and the measured data has come closer into alignment.

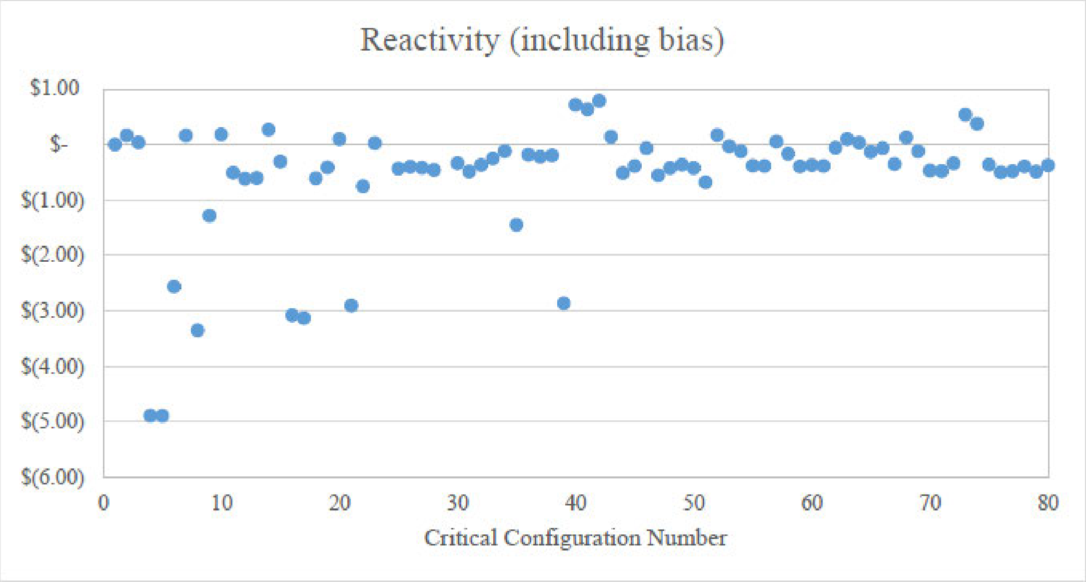


Figure 4.22, MCNP-6 Model Bias Calculations

The MCNP-6 burn capability was used with the model to simulate operation from initial criticality to 2018. The model with materials burned was used to calculate control rod worths and excess reactivity, with the results shown in Table 4.13. Nuclear characteristics of the 2018 core were calculated in the Neutronics Report as provided in Table 4.14.

Table 4.13, Comparison of MCNP and 2018 Experimental Data

Control Rod	MCNP Calculations			Experimental	Difference
	Rod in	Rod out	Worth	Worth	
Transient	1.00035	1.02354	\$3.24	\$3.44	-\$0.20
Regulating	0.99978	1.02214	\$3.13	\$3.18	-\$0.05
Shim1	1.00078	1.02248	\$3.03	\$3.09	-\$0.06
Shim2	1.00014	1.0211	\$2.93	\$2.94	-\$0.01
Core Excess	-	1.04118	\$6.75	\$6.06	\$0.69

Table 4.14: 2018 Core Properties

Nuclear Characteristics	Value
Shutdown Margin	\$2.34
Reactivity Coefficients:	
- Fuel Temp	-\$0.013/°C
- Moderator Temp	\$0.06/°C
- Void coefficient	\$0.10/%V[23]
Full Power Defect	\$3.81
β_{eff}	0.0071
Prompt Neutron Lifetime	47 μs

4.5.2. Limiting Core Configuration

Generally, core power distribution is a function of the number of fuel elements and the shape of the neutron flux, but these factors are not independent. Average power (total core power divided by the number of elements) decreases as the number of fuel elements increases. As the number of elements in the core increases the peaking factor increases but the power produced in the hot channel (the maximum power of a single element in the core) may be lower.

Calculating the fission density for MCNP with a varying number of fresh fuel elements shows that although the flux peaks more strongly in the center as more fuel elements are added, the decrease in the average fission density has a larger effect. Therefore, the core configuration with the smallest number of TRIGA fuel elements that can support full power operations has the fuel element with the highest power density. An 84-element core of fresh fuel is shown in the Neutronics Report to have an excess reactivity of \$6.93, slightly less than the \$7.00 limit. In this configuration a core

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[23] Ranging approximately from -\$0.1at 4% voids to \$1.35 at 96% voids

licensed to 1.1 MW with a potential 10% error would have the maximum fuel element generating 24.35 kW.

4.5.2.a. LCC Reactor Core Parameters

Nuclear characteristics of the Limiting Core Configuration (LCC) were calculated in the Neutronics Report and the results are provided in Table 4.16. The properties of the fuel element in the LCC producing the most energy (hot channel) are provided in Table 4.17. The axial and radial power distributions in the hot channel fuel matrix are represented in Figures 4.23 and 4.24.

Table 4.16, LCC Nuclear Characteristics

Parameter	Value
Core Excess	\$6.93
Shutdown Margin	\$3.60
Fuel Temp Coefficient	-\$0.01/°C
Moderator Temp coefficient	~0/°C
Void Coefficient	-0.25/%Voids
Power Defect	\$3.11
β_{eff}	0.00735
Prompt Neutron Lifetime	42.79 μs

MCNP calculations were performed for each of the experiment facilities accommodated by the upper grid plate, based on the fuel materials as used in the LCC. The number of fuel elements that produced an excess reactivity within the maximum limit was determined, where the addition of a single fuel element exceeds the limit.

Table 4.17, Hot Channel Properties

Element Location	B06
Thermal Power	24.35
Peaking Factors	
Core Radial	1.691
Element Radial	1.296
Element Axial	1.017
Effective	2.229

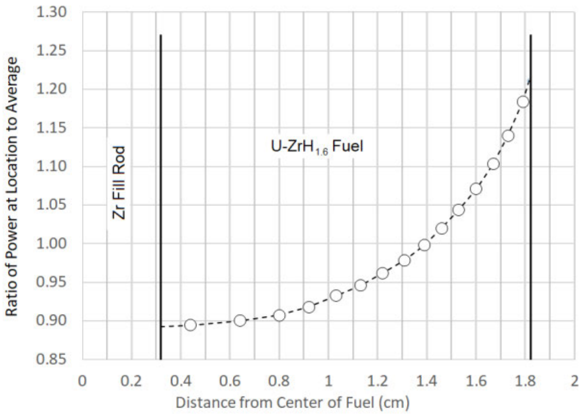

Figure 4.23, Radial Power Profile

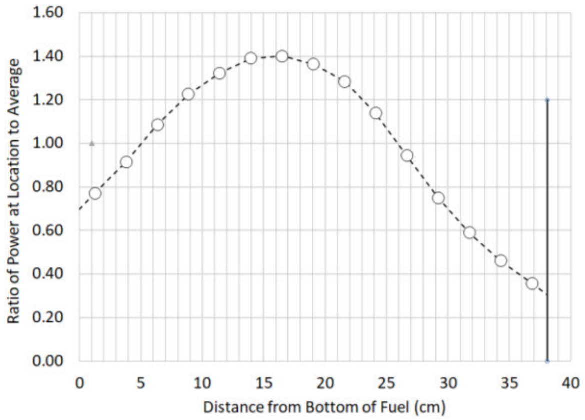

Figure 4.24, Axial Power Profile

4.5.2.b. The Effects of Burnup

Comparing the LCC physics data to the 2018 core physics data, the neutron lifetime and the delayed neutron fraction both decrease with burnup. The shutdown margin and the fuel temperature coefficients are slightly lower, related to the lower amount of zirconium-hydride in the core and the lower control rod worths that occur as the core burns.

4.5.2.c. The Effects of Experiment Facilities

MCNP calculations were performed for each of the experiment facilities accommodated by the upper grid plate, based on the fuel materials as used in the LCC. The number of fuel elements that produced excess reactivity within the maximum limit was determined, where the addition of a single fuel element would exceed the limit. The change in excess reactivity and the shutdown margin with the most reactive fully with drawn with specified experiment facilities installed was calculated. The excess reactivity that results from removing all of the B ring elements is too low to compensate for the full power temperature deficit, but the core is designed with a facility that displaces the central thimble and the B ring. The hot channel location, peaking factor, and power level with specified experiment facilities installed was calculated. The results are shown in Table 4.18. In all cases, the installation of the experiment facility has adequate shutdown margin and a hot channel element power level more conservative than the LCC configuration without the experiment facility installed.

Table 4.18, Experiment Facilities in the LCC

Parameter	T3	3EL	6EL	7EL	3EL & 7EL
Excess Δ/k	$5.66	$5.70	$1.18	$5.78	$5.84
Tech Spec Shutdown Margin[24]	-$5.07	-$5.70	-$7.37	-$4.86	-$4.14
No. Elements	99	95	111	98	104
Hot Element	B03	B01	C06	B04	B05
Hot Channel Peaking Factor	1.70	1.66	1.58	1.71	1.73
Hot Channel Power (kW)	20.66	21.14	17.19	21.15	20.10

4.5.3. Operating Limits

The limit on excess reactivity is 4.9% Δk/k. ($7.00) The neutronics report[25] identified a full power defect for the LCC of $3.11 (2.18% Δk/k), with 2.77% Δk/k ($3.89) to compensate for xenon poisoning and burnup.

The limit on shutdown margin is 0.2% Δk/k with the most reactive control rod fully withdrawn, all moveable experiments in their most reactive state, ambient temperature, and xenon less than $0.30. All analyzed conditions in the neutronics report and analysis of the impact of experiment facilities were well within the limit, assuring that the reactor can be shutdown safely by a margin sufficient to compensate for failure of a control rod or movement of an experiment. The reactivity worth of all experiment facilities with the most reactive rod fully withdrawn is bounded by the LCC.

The insertion of pulsed reactivity is limited to 2.2% Δk/k ($3.14). Thermal hydraulic analysis of pulsing (Chapter 4.6) indicates a large margin to temperature limits that could occur at the maximum pulse.

Thermal hydraulic analysis (Chapter 4.6) demonstrates that a continuous reactivity addition less than 0.2% (Δk/k)/s does not have potential to cause fuel temperature to exceed limits, and that a maximum scram setpoint for reactor power at 1.1 MW (with an instrument error of 10%) is adequate to prevent exceeding the fuel temperature safety limit.

The control rod system as described (Chapter 4.2.2) consists of four independent and fail-safe mechanisms, including electromagnet and air driven coupling. Limits on the minimum available shutdown reactivity assume a single control rod is inoperable and secured in the fully withdrawn position.

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[24] Most Reactive Rod Fully Withdrawn

[25] Analysis of the neutronic behavior of the Nuclear Engineering Teaching Laboratory at The University of Texas, February 2023

4.6. THERMAL-HYDRAULIC DESIGN

The UT TRIGA reactor is cooled by natural convection discharging to the reactor pool described in Chapter 4.3. Thermal hydraulic hot-channel analysis[26] was performed using TRACE (TRAC/RELAP Advanced Computational Engine) code. The hot channel was modeled as a hexagonal structure (Figure 4.25) with the principle thermal hydraulic values as indicated in Table 4.19. The nominal and limiting values for conditions affecting hydrostatic pressure in the core are listed in Table 4.20.

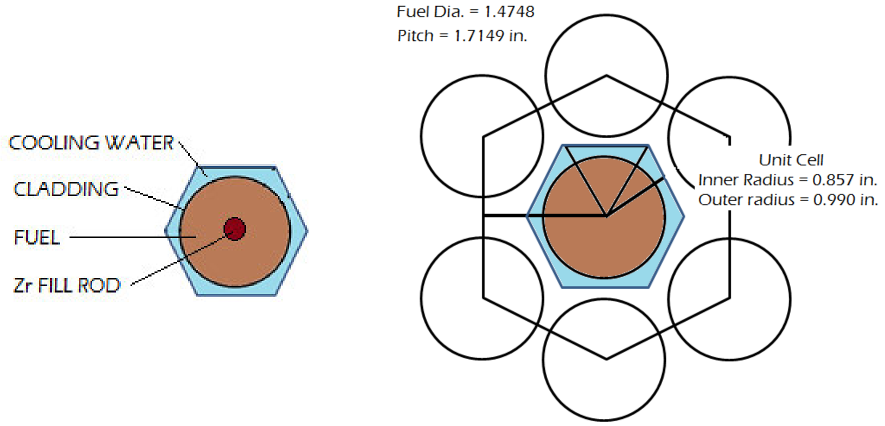

Figure 4.25, Flow Channel for UT TRIGA Fuel Elements

Table 4.19, Summary of Principle Thermal Hydraulic Values

Description	Var.	Value							
Fuel Element Pitch	*P*	1.7149	In	0.1428	ft	4.3535	cm	0.04353	m
Fuel Element Diameter	*Dfuel*	1.4784	In	0.1232	ft	3.7551	cm	0.03755	m
Wetted Perimeter	P_W	4.6445	In	0.387	ft	11.7971	cm	0.1179	m
Fuel Cross Section/Area	A_{FC}	1.7166	in^2	0.01192	ft^2	11.0749	cm^2	0.001107	m^2
Unit Cell Area	*ACell*	2.5442	in^2	0.01766	ft^2	16.4142	cm^2	0.001641	m^2
Flow Channel Area	*AFC*	0.8275	in^2	0.005747	ft^2	5.3392	cm^2	0.000534	m^2
Hydraulic Diameter	D_h	0.7127	In	0.05939	ft.	1.8103	cm	0.0181	M

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[26] Appendix 4.2, Thermal Hydraulic Analysis of the University of Texas (UT) TRIGA Reactor (Whaley, Charlton) Feb 2023

Table 4.20, Pressure Boundary Condition

Condition	Temperature °C	Density $kg\cdot m^{-3}$	Height m	Hydrostatic Pressure kPa	Pressure kPa	Pressure Psia
Limiting	49	988.488	5.25	50.9	146.9	21.3
Nominal	25	997.048	7.25	70.8	166.8	24.2
	27	996.516	7.25	70.8	166.8	24.2

The TRACE internal materials-library for stainless-steel and water were material properties for zirconium and uranium-zirconium hydride were taken from eFunda (engineering fundamentals, material properties) and “The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuels”[27].

The Fuel Temperature Coefficient of Reactivity (FTC) was developed from the Neutronics Report’s MCNP model and cross-sections at temperatures taken from the distributed libraries for scattering data and isotopes for comparison to the FTC developed by General Atomics. An auxiliary program distributed with MCNP (MAXSF) was used to provide additional temperature-based cross sections for more granular cross section data. The moderator temperature coefficient (MTC) and the FTC for the full range of operation were developed.

Data from the TRACE output files supported critical heat flux ratio calculation with the Bernath correlation as reported in ANL/RERTR/TM-07-01.

Model validation was accomplished by comparing data from steady state and pulsing operations to TRACE calculations. The temperatures of the thermocouple locations used in the measuring channel were calculated using the TRACE model for comparison to data from steady-state operation, with maximum deviation less than 8%. Simulation of pulsing with the TRACE model were compared to maximum fuel temperatures and peak power from historical records (Figures 4.26 and 4.27 with data calculated by TRACE in the large circles, historical data in small squares).

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[27] E-117-833 - The U-ZrHx Alloy: It’s Properties and Use in TRIGA Fuels, M.T. Simnad, GA Project No. 4314 (1980)

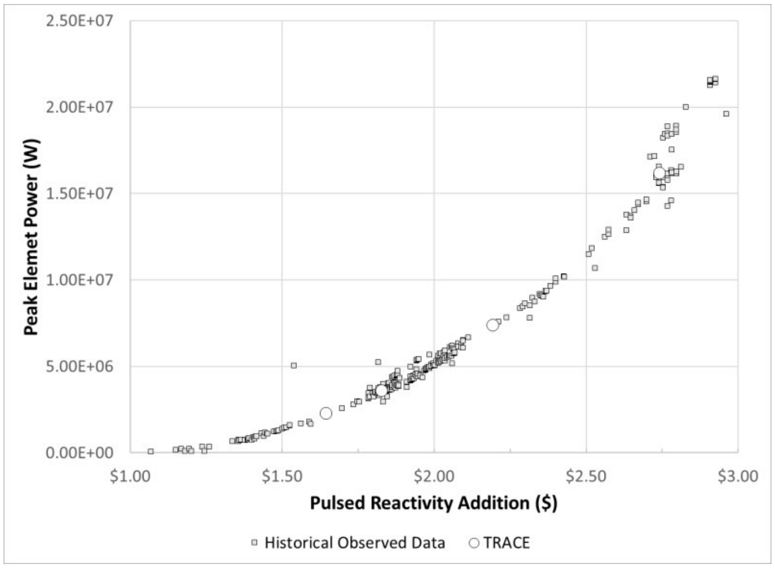

Figure 4.26, Peak Element Power Level versus Pulse Reactivity Addition from UT TRACE Calculation Compared to Observed Historical Data

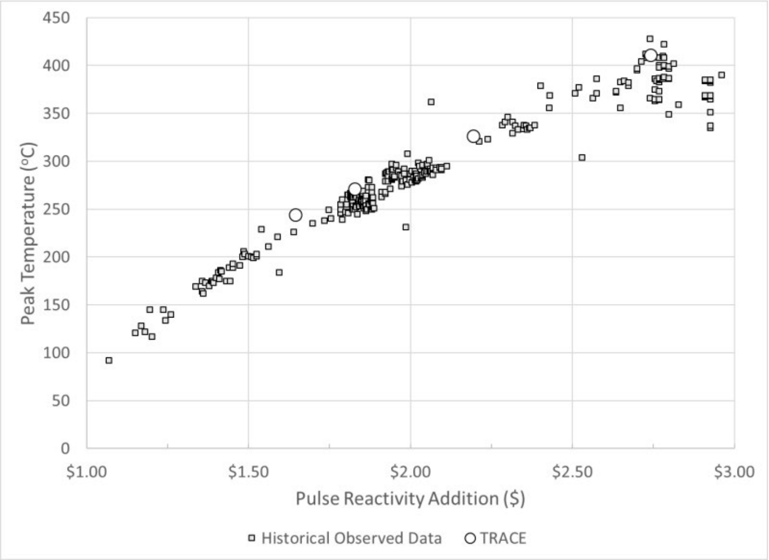

Figure 4.27, Peak Fuel Temperature versus Pulse Reactivity Addition from UT TRACE Calculation Compared to Historical Data

The comparison demonstrates that the TRACE model predicts thermal hydraulic performance of the UT TRIGA reactor with reasonable accuracy. The model was used to calculate maximum fuel temperatures and develop CHFR for steady state element power levels. Resulting channel flow rates are shown in Figure 4.28.

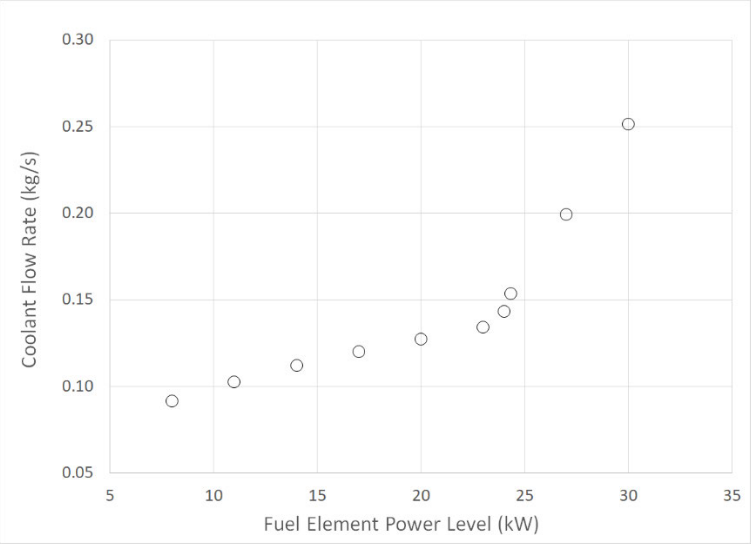

Figure 4.28, Coolant Flow Rate at Element Power

A hot channel power level of 24.34 kW results in a CHFR of 2.39 (Figure 4.29) and a maximum fuel temperature of 559°C (Figures 4.30 and 4.31). Therefore, a limiting safety system setting of 550°C is adequate to assure reactor power remains below the limiting safety system setting.

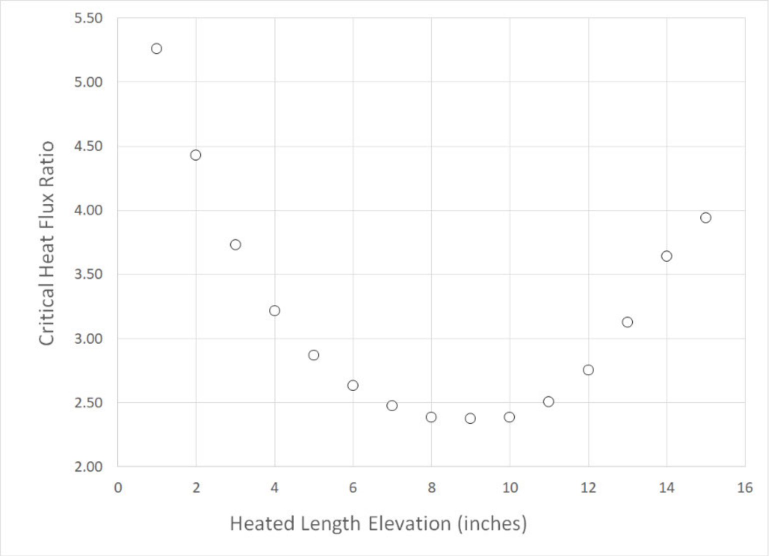

Figure 4.29, Critical Heat Flux, Bernath Correlation

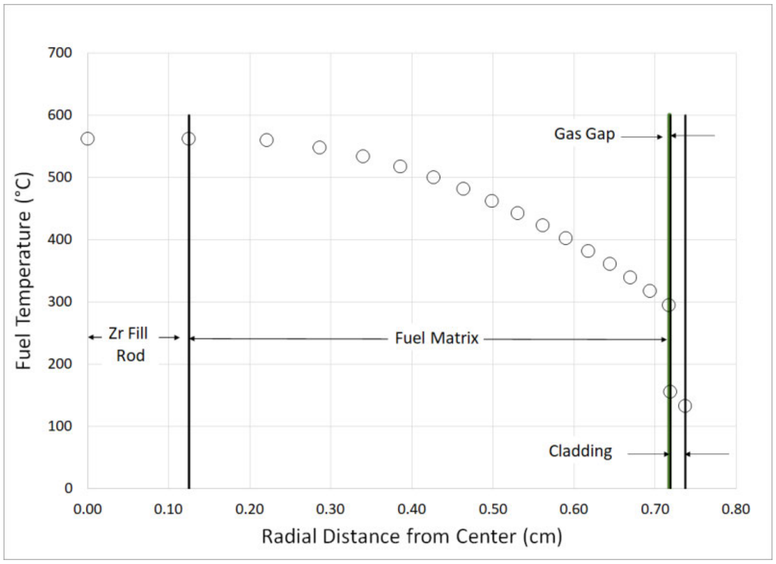

Figure 4.30, Hot Channel Radial Temperature Distribution

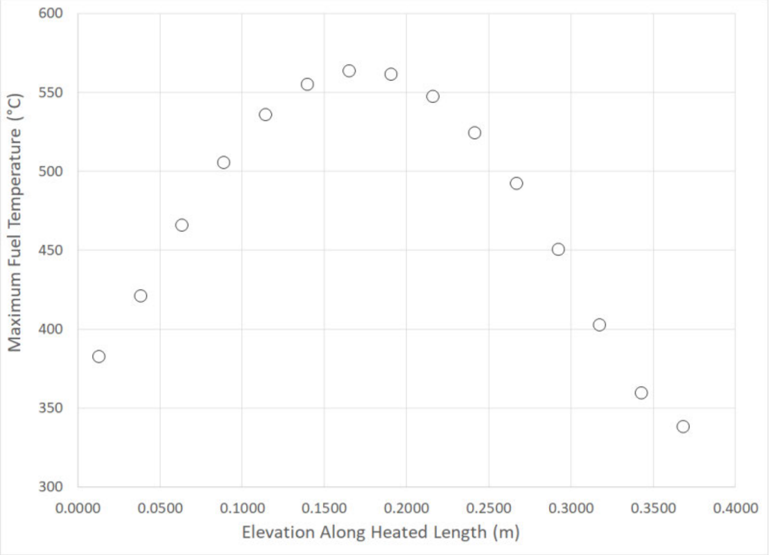

Figure 4.31, Hot Channel Axial Temperature Distribution

Power and temperature response to pulsed reactivity insertion from $1.00 to $4.40 was calculated from low power initial conditions (Figures 4.32 and 4.33). At $4.40 the maximum fuel temperature was 824°C. Therefore, a maximum pulse limit of 2.2% Δk/k ($3.14) is adequate to ensure the pulsing safety limit is met.

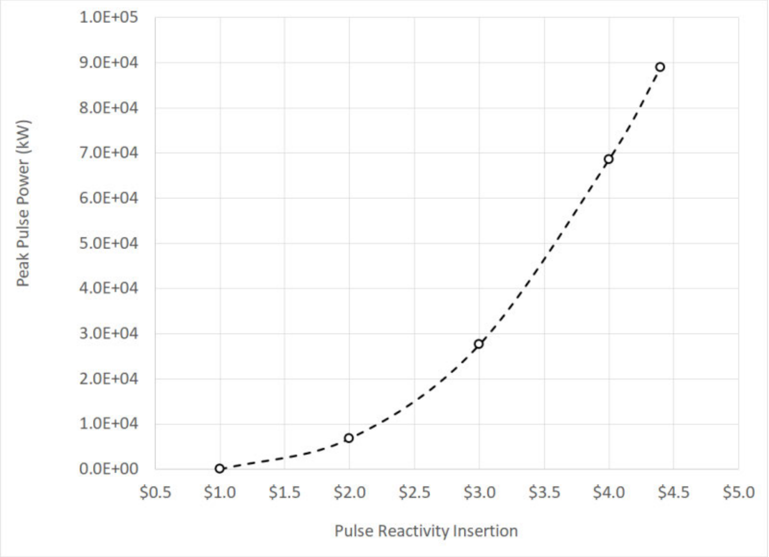

Figure 4.32, Hot Channel LCC Peak Power Level
Versus Pulsed Reactivity Insertion of $1, $2, $3, $4 and $4.40

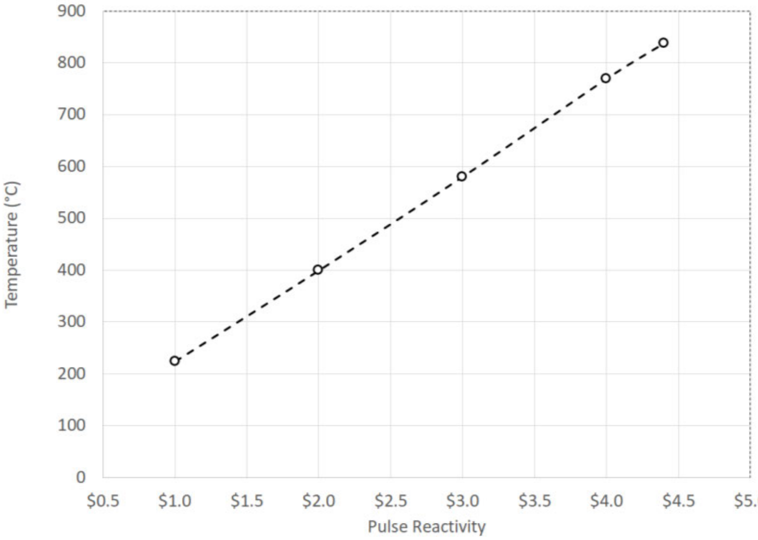

Figure 4.33, LCC Peak Fuel Temperature for Varying Reactivity Insertions

Pulsing from power results in increased temperature compared to pulsing from low power. Calculations were performed to determine the maximum hot channel fuel temperature during the pulse and the final power level following the pulse if allowed to come to equilibrium with the results in Table 4.21. The final power level approaches the maximum LCC power during steady-state operations for $3.00 pulses from 111 kW. The final steady state power level reaches the scram setpoint (with maximum instrument error) for steady state operations for $3.00 pulses from 124 kW. With an initial power level of 174 kW the fuel temperature approaches the safety limit for pulsing. Therefore, an interlock to prevent pulsing from power levels at or above 1 kW is adequate to ensure the temperature safety limit is met.

Table 4.21, Pulsing from Power Summary

Init Core Power (kW)	111 kW	124 kW	174 kW
Initial Ave. Element Power (kW)	1.33 kW	1.47 kW	2.08 kW
Final Element Power (kW)	24.02 kW	26.92 kW	28.80 kW
Final Core Power (kW)	1193 kW	1216 kW	1337 kW
Max Hot Channel Temperature	724 °C	747 °C	826 °C

An analysis of continuous reactivity addition from power was performed. The model was simulated as operating at steady state power level followed by initiation of reactivity addition. The minimum for reactivity addition rate and the maximum time from initiation of scram to an all-rods-down condition were based on the 1992 Technical Specification limits.

The maximum hot channel temperature using reactivity addition rates from 0.2% per second to 0.7% per second at delays between reaching the power level scram setpoint and control rod full insertion were calculated (Table 4.22). Full insertion delays of 1 to 3 seconds do not cause the steady state limit to reach the steady state fuel temperature safety limit for cladding temperature less than 500°C up to 0.7% per second. Therefore, a control rod drop time (full-out to full-in following initiation of a scram) is adequate to assure fuel temperature safety limit is not challenged during a continuous reactivity insertion of 0.2% per second.

Table 4.22, Peak Temperature Following Rod Full-Insertion Intervals

Reactivity addition rate	0.2%/s $0.29/s	0.4%/s $0.57/s	0.5%/s $0.71/s	0.6%/s $0.86/s	0.7%/s $1.00/s
Delay (seconds)	T_{max} (°C)				
1	573	589	608	627	651
2	585	639	679	726	778
3	609	709	773	863	993
4	630	772	878	1050	1448
5	634	800	992	N/A	N/A

Analysis of fuel temperature response to a loss of cooling accident (LOCA) was performed. A TRACE calculation was performed to establish initial conditions operating at steady state. The method of ANSI/ANS-5.1-2014 (Decay Heat Power in Light Water Reactors) was employed to establish fission product power as a function of time. Four cases were developed, with delays before loss of water cooling of 1 second, 1 minute, 10 minutes and 20 minutes. A loss of cooling accident with a 1 second transition to fully air-cooled modality was simulated. A maximum power steady state power level of 1.1 MW with a maximum instrument error of 10% was shown to be adequate to assure cladding integrity is not challenged in a loss of coolant accident.

Table 6.6, Loss Of Water-Cooling Analysis

Delay For Air Cooling (s)	Maximum Temperature (°C)
1	787
60	780
600	753
1200	733

5. 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 a 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 26 foot, 11.5 in. (8.2169 m) tall tank formed by the union of two half-cylinders with a radius of 6 ½ ft. (1.9812 m) with 6 ½ feet separating the half-cylinders as shown in Figure 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. The pool vertical cross section is shown in Figure 5.1B. Basic reactor coolant system data is provided in Table 5.1.

[redacted]

Figure 5.1A, Pool Fabrication Figure 5.1B, Cross Section Figure 5.C, Beam Orientation

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	1000 kW
	Flow Rate (shell)	400 gpm (25.2 lps)
	Flow Rate (tubes)	250 gpm (15.8 lps)
Typical Heat Exchanger Operating Parameters	Tube Inlet	100 °F 42 psia
	Tube Outlet	69 °F 27 psia
	Shell Inlet	48 °F 55 psia
	Shell Outlet	67 °F 48 psia

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 provides ambient cooling adequate to control the pool water temperature at low power operations and for decay heat removal. At 1 MW operation, the reactor is capable of heating up the pool under the 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. Beam port configurations are shown in Figure 5.1C.

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 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. [redacted]. 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 [redacted] outside of the area define by the pool walls. A stainless-steel type 304 ring is machined for a slip fit over the [redacted] 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.

5.3. POOL COOLING SYSTEM

The pool cooling system is shown in Figure 5.2.

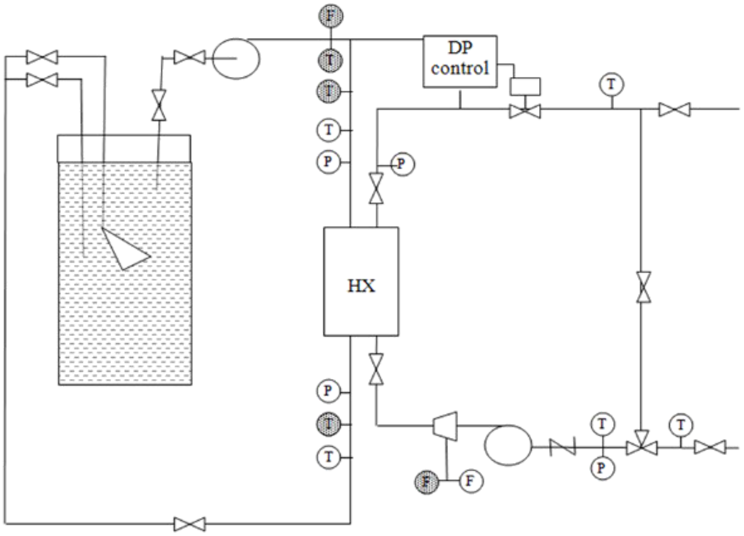

Figure 5.2, Pool Cooling System (F=Flow, T=Temperature, and P=Pressure)

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, and 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:

$$q = U \cdot A \cdot \delta T \tag{1}$$

where:

U is the overall heat transfer coefficient (watt/m^2 -°C)
A is the surface area for heat transfer (m^2)
δT_m is the 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[28]:

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

where:

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 the 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:

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

$$U_f = \frac{1}{R_f + \frac{1}{U_c}} \qquad (3)$$

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

$$h = \frac{N_u \cdot k}{d} \qquad (4)$$

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[29] recommend the following relation for fully developed turbulent flow in tubes:

$$Nu_t = 0.023 \cdot Re^{0.8} \cdot Pr^n \qquad (5)$$

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[30] as follows:

$$Nu_t = 0.33 \cdot Re^{0.6} \cdot Pr^{0.33} \qquad (6)$$

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, δT_m, is related to the heat exchanger type by a correction factor, F, and a log mean temperature difference, LMTD[31].

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[29] University of California (Berkeley) Pub. Eng, Dittus, F. W and Boelter, L. M. K., Vol 2, pp 443 (1930)

[30] 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)

[31] Heat Tansfer, White, op. cit.

The correlation relates a simple single pass heat exchanger with more complex multiple pass baffled units. A relation is defined by

$$\delta T_m = F \cdot LMTD \tag{7}$$

where

F is the correction factor[32,33],

$$LMTD = \frac{T_a - T_b}{ln\frac{T_a}{T_b}} \tag{8}$$

For a counter flow heat exchanger

$$T_a = T_{hot\_fluid\_in} - T_{cold\_fluid\_out} \tag{9}$$
$$T_b = T_{hot\_fluid\_out} - T_{cold\_fluid\_in} \tag{10}$$

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

$$q = C \cdot (T_{in} - T_{out}) \tag{11}$$

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[34,35] is used. The dimensionless parameter called the heat exchanger effectiveness ε is defined as

$$\varepsilon = \frac{Actual\_HX}{Max\_HX} \tag{12}$$

where the maximum possible heat transfer is

$$Max_{HX} = c_{min} \cdot \left(T_{hot\_in} - T_{cold\_in}\right) \tag{13}$$

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

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[32] Mean Temperature Difference in Design, Bowman, R. A., Mueller, A. C., and Nagle, W. M., Trans. ASME, Vol 62 (1940) pp283-294

[33] Standards, TEMA 3rd Ed., Tubular Heat Exchanger Manufacturers Association New York (1952)

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

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

$$\varepsilon = \frac{c_{hot} \cdot (T_{hot\_in} - T_{cold\_in})}{c_{min} \cdot (T_{hot\_in} - T_{cold\_in})} \tag{14}$$

for the hot fluid and

$$\varepsilon = \frac{c_{cold} \cdot (T_{cold\_in} - T_{cold\_in})}{c_{min} \cdot (T_{hot\_in} - T_{cold\_in})} \tag{15}$$

for the cold fluid. The heat exchange effectiveness determined by[36] for a shell and tube heat exchanger with one shell pan and any multiple of tube passes is given by

$$\varepsilon = z \cdot \left[1 + r + B \cdot \frac{1 + \mathrm{e}^{-N \cdot B}}{1 - \mathrm{e}^{-N \cdot B}}\right]^{-1} \tag{16}$$

where

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

Once the effectiveness is calculated, then (14) and (15) above used to determine T_{hot_out} and T_{cold_out}. These may then be used in (11) to determine the capacity of the heat exchanger. The parameters used and results from these calculations are given in Table 5.2.

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.

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[36] Compact Heat Exchangers op. cit.

Table 5.2, Heat Exchanger, Heat Transfer and Hydraulic Parameters

Component/Parameter	Specification	Value	Units
Tubes	Outside Diameter	0.750 (1.91)	in. (cm)
	Wall Thickness	0.049 (0.124)	in. (cm)
	Thermal Conductivity	8.21	Btu/hr-ft-°F
Flow Area	Tube Side	8.1 (52.3	in^2 (cm^2)
	Shell Side	33.8 (218)	in^2 (cm^2)
Heat transfer Surfaces	Na	346 (32.1)	ft^2 (m^2)
Average Prandtl No.	Tube	5.38	na
	Shell	8.41	na
Average Kinematic Viscosity	Tube	8.63E-6 (8.02E-7)	ft^2/s (m2/^{s})
	Shell	1.28E-5 (1.19E-6)	ft^2/s (m2/^{s})
Reynolds No.	Tube	6.19E5	na
	Shell	2.02E4	na
Corrective Heat Transfer Coefficients	Tube	1710 (9701)	Btu/hr-ft^2-°F (W/m^2-°C)
	Shell	1395 (7922)	Btu/hr-ft^2-°F (W/m^2-°C)
Overall Heat Transfer Coefficient	Tube	520 (2953)	Btu/hr-ft^2-°F (W/m^2-°C)
	Shell	430 (2442)	Btu/hr-ft^2-°F (W/m^2-°C)
Effectiveness	Clean	0.6	na
	Fouled	0.56	na
LMTD	Na	26.1 (14.5)	° F (°C)
Corrective Factor F	Na	0.83	na
Capacity	Clean	1140	kW
	Fouled	1070	kW

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 through a heat exchanger. The chilled water system is operated by the University for cooling of the 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 TACC) 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. Pool Instrumentation and Controls

5.3.4.a. Level gage

A metric scale is used to provide local indication of pool level. With the 23.5 cm mark secured to the tank equipment monitoring ring, 8 cm corresponds to 8.10 meters above the floor of the pool. Pool level is normally maintained at 8.10 ± 0.05 m, corresponding to an indicated 8 ± 5 cm.

5.3.4.b. Level Control Functions

Pool level is monitored by 5 float switches using magnetically actuated read switches on two separate assemblies in each hollow stainless-steel float. Two level-switches are adjusted to alarm if water level is high (8.15 m above the pool floor) or low (8.07 m above the pool floor. Two level switches provide a reactor trip initiation if water level falls to 7.6 m above the pool floor (low level scram). One level switch initiates an alarm at a remote 24-hour response station at 7.6 m.

5.3.4.c. Temperature Control

Chill water flow is normally about 500 gpm. Chill water flow through the heat exchanger is regulated to control pool temperature by a pneumatic control system. If temperature is lower than the control setpoint, an air operated three-way valve line diverts chill water flow around the heat exchanger. Conversely, if temperature rises above the setpoint, the bypass flow is reduced so that chill water flow rate through the heat exchanger is higher.

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, monitored by a pitot tube and differential transmitter with local and console . 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.

5.3.4.d. Nitrogen Diffuser

A fraction of pool water-return (from the heat exchanger) is routed to a triangular nozzle. The return flow from the nozzle is directed to disturb the convection flow of water heated in the core and lengthen the path of water containing nitrogen 16, reducing potential personnel exposure.

5.4. PRIMARY CLEANUP SYSTEMINDICATIONS

The primary cleanup system (Figure 5.3) is designed to use filtration and ion exchange to control water quality for corrosion control. Ion exchanger inlet and outlet conductivity is continuously monitored to assure water quality and assess resin performance.

The purification skid is located in room 1.104b at about the same level as the reactor core. The skid consists of a pump, flowmeter, filter, resin bed, and instrumentation. The surface of the resin bed tank is instrumented with a radiation monitored that provides local and control room indications. 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.

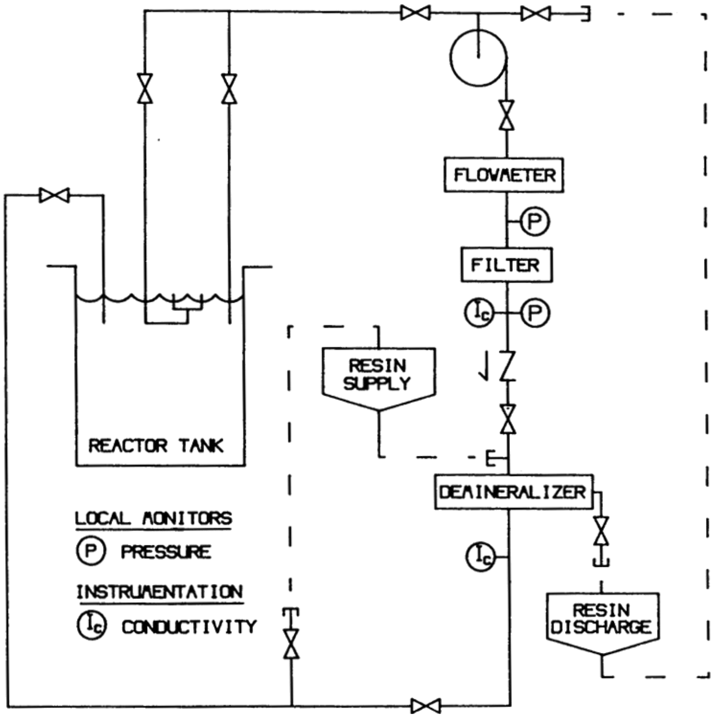

Figure 5.3, Pool Cleanup System

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 Figure 5.2 and 5.3. The parameters that are considered part of the water system instrumentation system are presented in Figure 5.4.

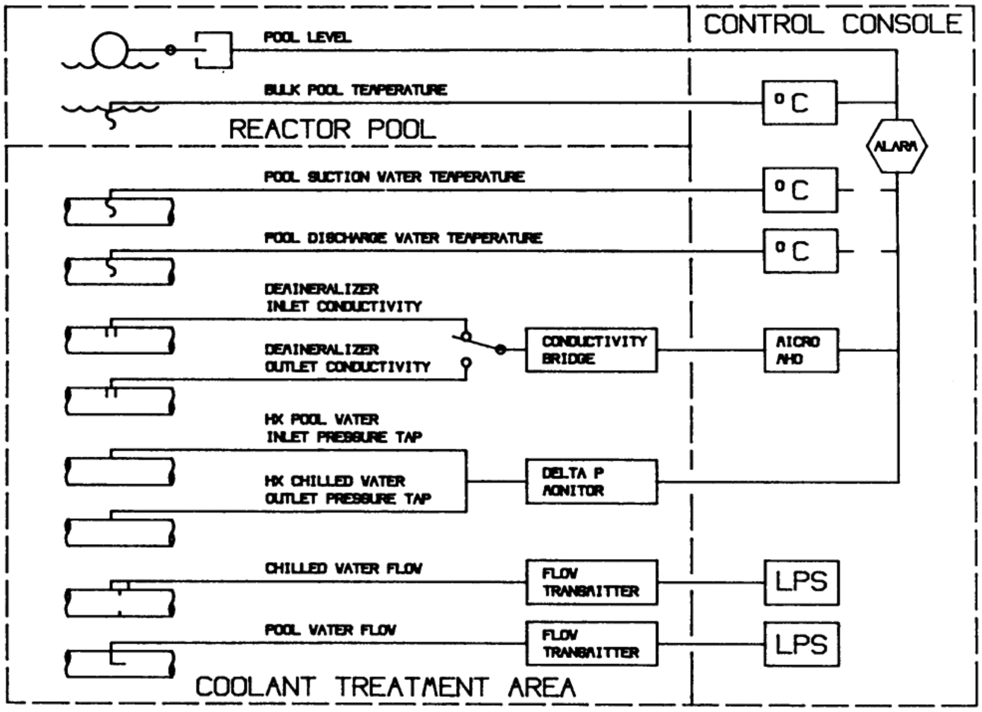

Figure 5.4, Cooling and Cleanup Instrumentation

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. ENGINEERED SAFEGUARD FEATURES

As discussed in Chapters 5 and 13 of the UT-Austin SAR from 1991[37], identified in previous analysis in NUREG-1135[38], and identified from experience at other TRIGA reactors such as in NUREG-1282[39], 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.

While confinement isolation is described in Chapter 9 (Section 9.2) as designed to limit the abnormal release of radioactive materials with a setpoint based on fission products (Section 9.2.4), the consequences of a maximum hypothetical accident in Chapter 13 do not credit automatic confinement isolation. Therefore, confinement isolation is not considered an engineered safeguards feature.

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[37] "Safety Analysis Report," TRIGA Reactor Facility," Nuclear Engineering Teaching Laboratory, The University of Texas at Austin, Submitted May 1991.

[38] NUREG-1135, "Safety Evaluation Report Related to the Construction Permit and Operating License for the Research Reactor at the University of Texas," U.S. Nuclear Regulatory Commission, Docket 50-602, May 1985.

[39] 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. INSTRUMENTATION AND CONTROL SYSTEM

The digital General Atomics instrumentation and control system design used at UT-Austin was intended to replace analog reactor consoles TRIGA reactor facilities. The manufacturer performed initial verification and testing of the control system (Factory Acceptance Testing) prior to installation at UT-Austin. The University performed instrument and control console evaluation (Site Acceptance Testing) for the TRIGA in conjunction with 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, and (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 at Austin 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)[40,41]. 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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[40] "Criteria for the Reactor Safety Systems of Research Reactors", American Nuclear Society, American National Standard, ANSI/ANS-15.15-1978.

[41] "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[42] 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 Figure 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 Figure 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.

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

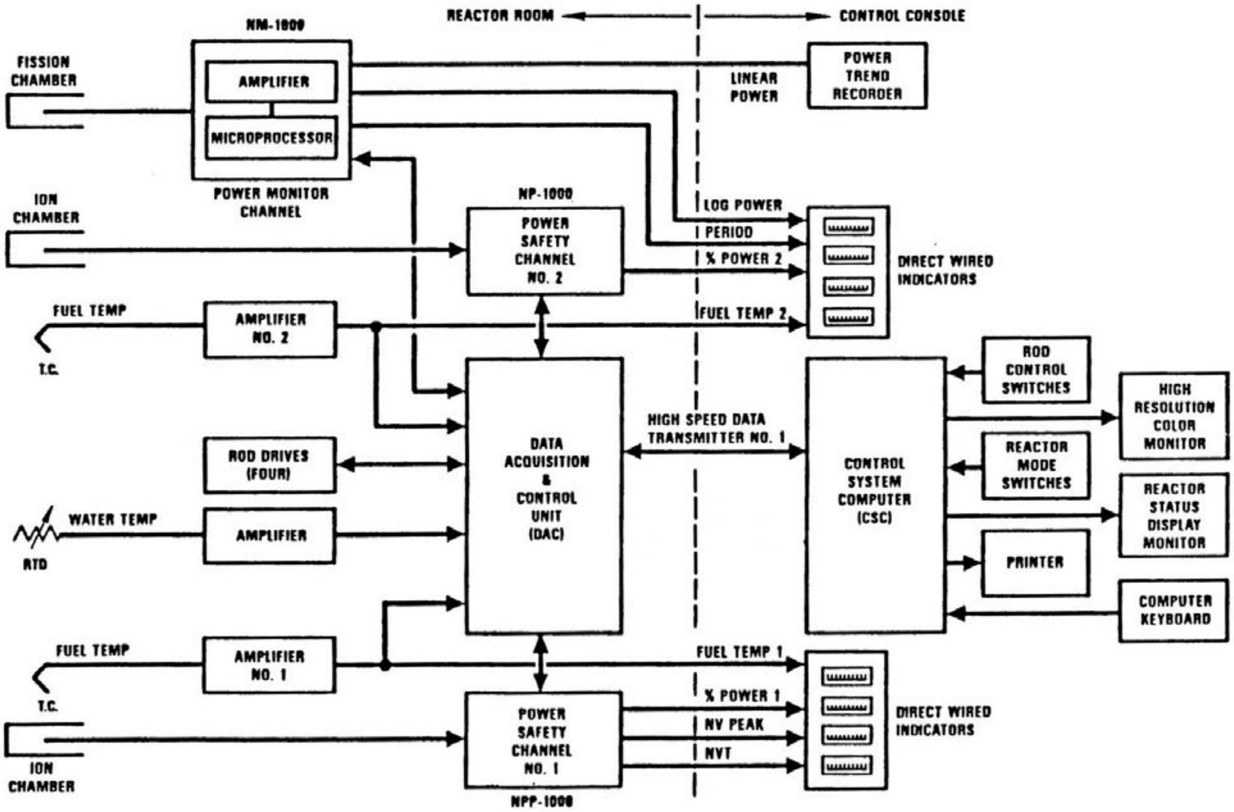


Figure 7.1, Control System Block Diagram

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.

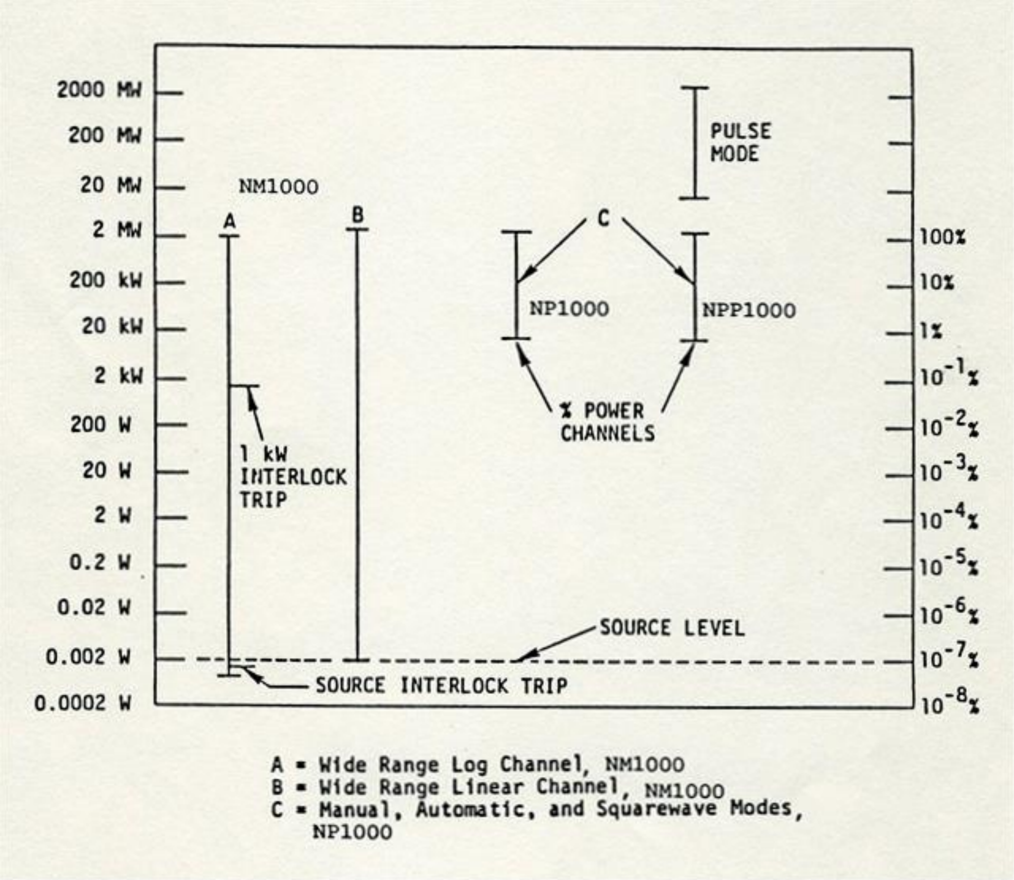
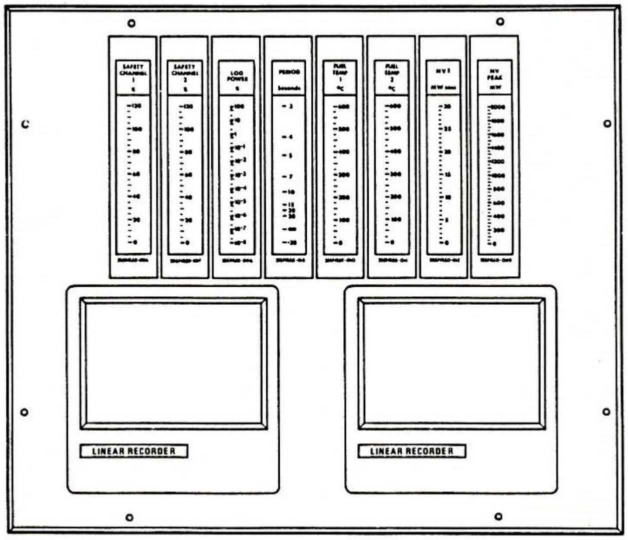


Figure 7.2, Neutron Channel Operating Ranges

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.



Analog Display Panel

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 with good sensitivity.

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 and hardwired to a bar graph indicator.

A modified version of the safety channel, the NPP-1000, adds functionality to measure peak pulse power, total pulse energy and manage automatic gain change and related trip points for pulsing. The control system automatically selects proper gain setting for steady-state or pulse mode when the operator shifts the operating mode to pulse mode. Peak pulse power and total pulse energy circuits are also enabled in the pulse mode.

The NP-1000 and the NPP-1000 are identical except for the peak power and total pulse energy circuits. The detectors for both safety channels are uncompensated ionization chambers.

7.1.3. Reactor Control Console

The layout of the control console is shown in Figure 7.4 with selected details in Figure 7.5. 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. Monitors (High Resolution Operations Monitor and Reactor Information Monitor).
d. Power and temperature bar graphs.
e. Computer hardware (CSC, expansion chassis, etc.).
f. Storage and printer.

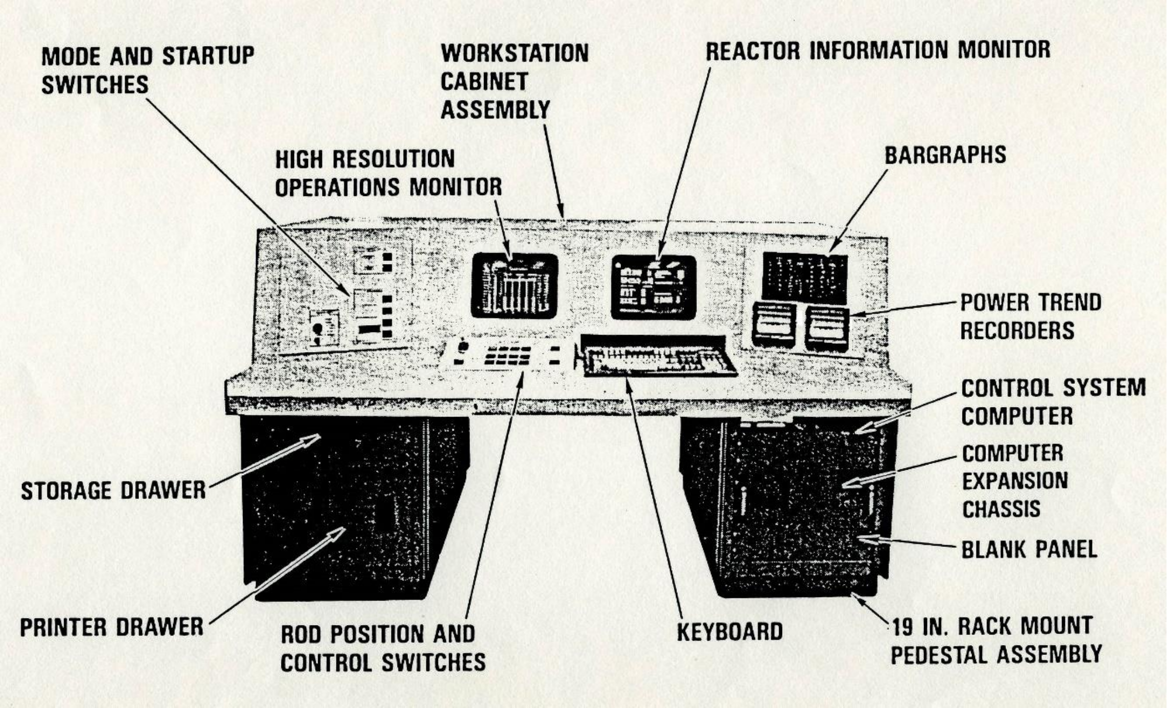
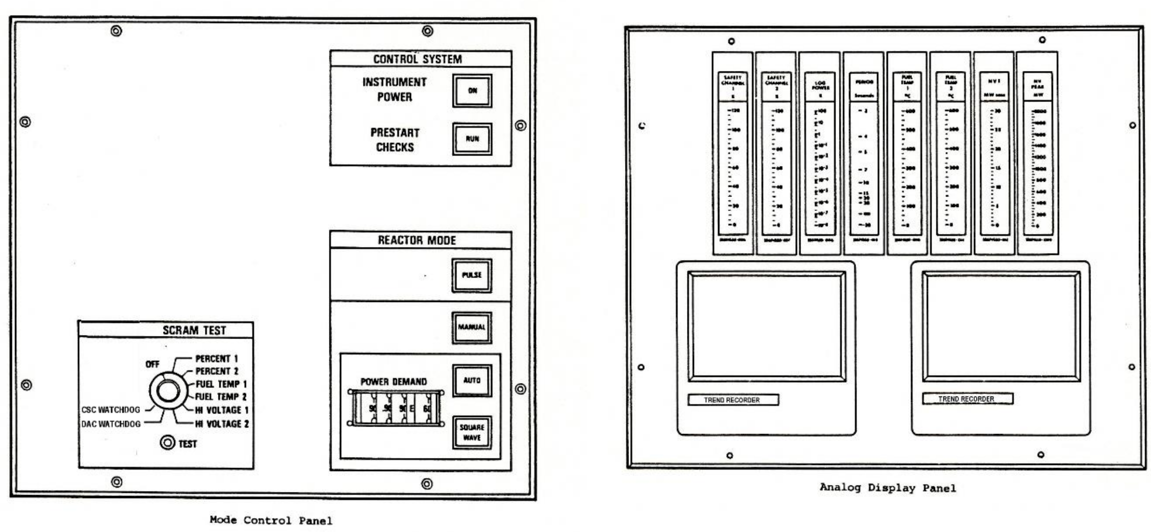


Figure 7.4, Layout of the Reactor Control Console



Mode Control Panel

Analog Display Panel

Figure 7.5, 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 Figure 7.6 and 7.7.

Functions of the rod control panel are represented in Figure 7.8, and are presented as:

a. Key switch for rod magnet power.
b. Rod control switches and annunciators.
c. SCRAM-switch for safety function.
d. Annunciator panel and scram reset (audio channel and visual indicators when the condition is cleared).

The CSC provides all 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 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 use of computer programming allows great versatility and flexibility in operationally related activities. The information collected in the CSC provides digital support for various administrative functions such as monitoring reactor usage, storing pulse data, reactor operating history and logging operator usage.

The original installation used an oscilloscope in the digital display. Display of original hardware obsolescence was mitigated by installing an additional computer to emulate the oscilloscope.

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. Pulse and square-wave modes establish required conditions to control the pulse rod drive. The pulse mode generates high-power levels for very short periods of time. The square-wave operation allows the power level to be raised quickly to a desired power level. Manual and automatic reactor modes control reactor operation from source level to 100% power. Manual and automatic mods modes are used to perform reactor startup, changing power level, and steady-state operation.

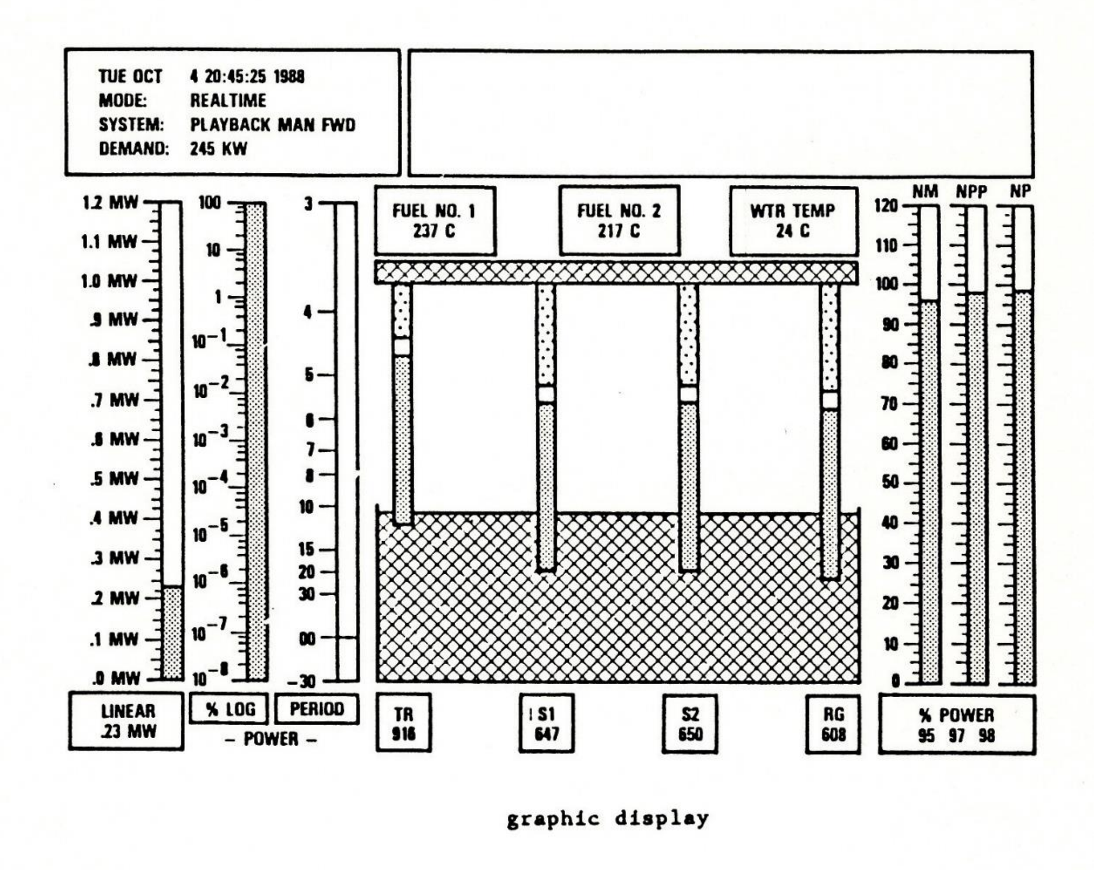
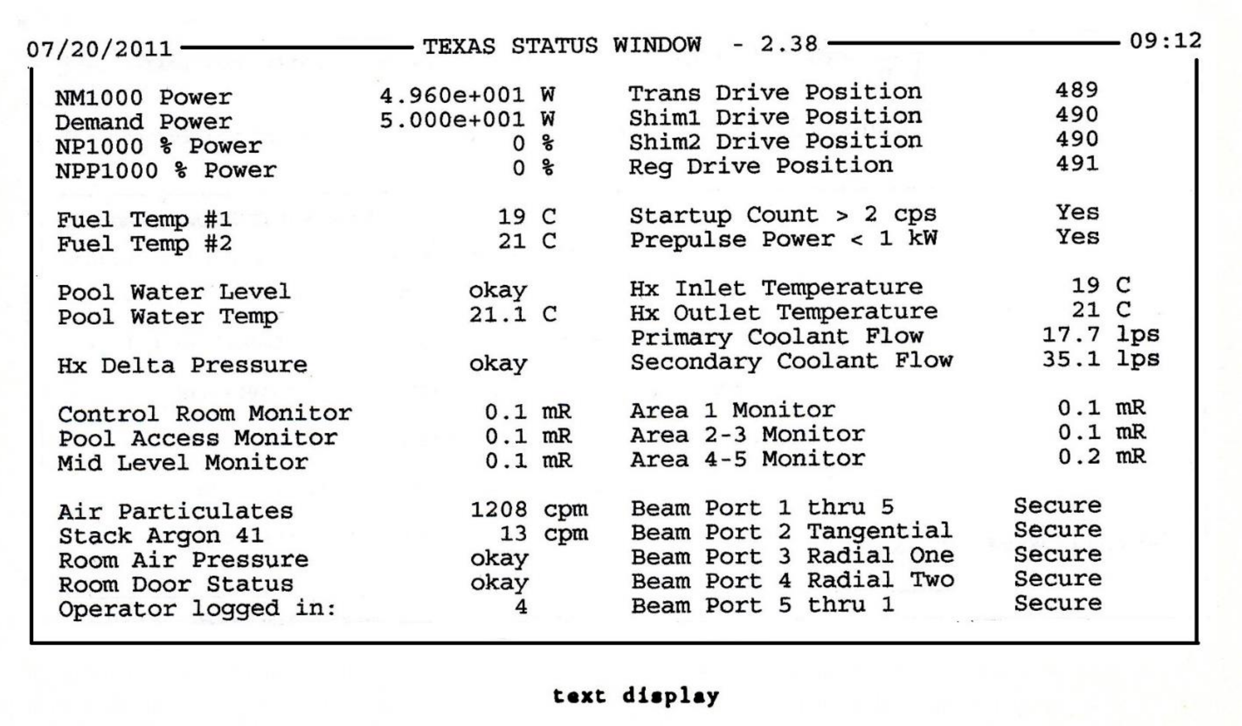


Figure 7.6, Typical Console Control Panel Display



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 NM1000 Power          4.960e+001 W   Trans Drive Position       489
 Demand Power          5.000e+001 W   Shim1 Drive Position       490
 NP1000 % Power                 0 %   Shim2 Drive Position       490
 NPP1000 % Power                0 %   Reg Drive Position         491

 Fuel Temp #1                  19 C   Startup Count > 2 cps      Yes
 Fuel Temp #2                  21 C   Prepulse Power < 1 kW      Yes

 Pool Water Level          okay       Hx Inlet Temperature        19 C
 Pool Water Temp           21.1 C     Hx Outlet Temperature       21 C
                                      Primary Coolant Flow      17.7 lps
 Hx Delta Pressure         okay       Secondary Coolant Flow    35.1 lps

 Control Room Monitor        0.1 mR   Area 1 Monitor             0.1 mR
 Pool Access Monitor         0.1 mR   Area 2-3 Monitor           0.1 mR
 Mid Level Monitor           0.1 mR   Area 4-5 Monitor           0.2 mR

 Air Particulates          1208 cpm   Beam Port 1 thru 5       Secure
 Stack Argon 41              13 cpm   Beam Port 2 Tangential   Secure
 Room Air Pressure         okay       Beam Port 3 Radial One   Secure
 Room Door Status          okay       Beam Port 4 Radial Two   Secure
 Operator logged in:          4       Beam Port 5 thru 1       Secure

                          text display
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Figure 7.7, Typical Video Display Data

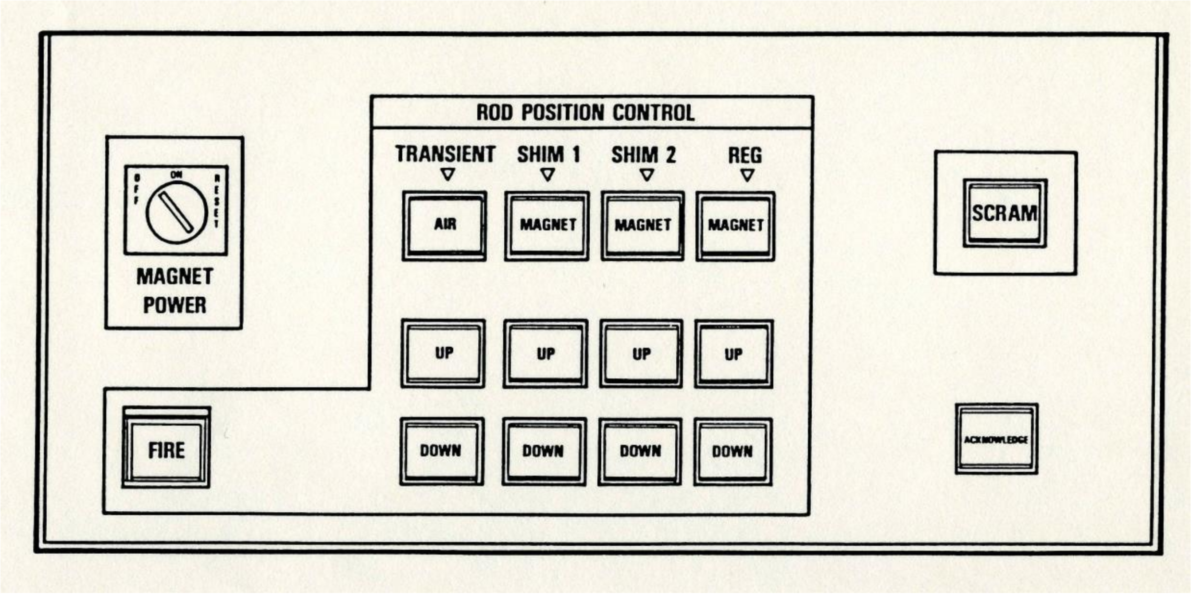

Figure 7.8, Rod Control Panel

Manual rod control is accomplished by back-lighted lighted push buttons on the rod control panel. The top row of pushbuttons labeled MAGNET indicate magnet contact with the armature and magnet current when illuminated. The pushbutton labeled AIR indicates the solenoid valve that supplies air to the pulse rod is energized. Depressing any one of the AIR-MAGNET push buttons will interrupt the current to that magnet (or the pulse rod solenoid) and extinguish the 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 (automatically for shim 1, shim 2, and the regulating rod, manually for the pulse rod). Current will be restored automatically to the shim and regulating rods when the rods are fully inserted, and for the pulse rod when the fire button is depressed if (1) the pulse rod is fully inserted in the manual or automatic mode or (2) any position in the pulse or square-wave modes. 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. Interlocks prevent the movement of the rods in the up direction (no interlock inhibits down movement of control rods) under the conditions where:

a. Scrams are not reset.
b. Magnet is not coupled to armature.
c. The source level below minimum count.
d. Two UP switches are depressed at the same time.
e. The mode switch is in one of the pulse positions.

Automatic (servo) power control can be obtained by switching from manual mode to automatic mode. In automatic mode the regulating rod is controlled automatically in response to a target power level, measured 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 operation is proportional, integral-differential (PID) control 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 begins by establishing criticality at a flux level below 1 kW in the manual mode using the motor-driven control rods with the transient rod either fully or partially inserted. The pulse mode selector switch is then depressed. The selection automatically connects the NPP-1000 chamber to monitor and record peak flux (nv) and energy release (nvt) and prevents the NP-1000 and NM-1000 scram trips. 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 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 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[43] interrupts the magnet current and results in the immediate insertion of all rods under any of the following:

a. High neutron fluxes from NP-1000, NPP1000, or NM-1000.
b. High-voltage failure on the NM-1000, NP-1000, or NP1000.
c. High fuel temperature on one of two channels.
d. Manual scram.
e. Peak neutron flux or energy (pulse mode).
f. External safety switches (for experiments).
g. Loss of electrical power to the control console
h. Watchdog circuits for each computer to monitor computer status by updating timers.
i. CSC based scram.

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.

Abnormal 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.

7.1.6. Logic Functions

A simplified control system logic diagram is shown in Figure 7.9. The two separate flux monitoring safety channels ensure safe operation of the reactor by monitoring the power level and acting independently to shut the reactor down if a potential safety challenge 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 writing the information in data base. The database is transmitted to the CSC on request and maintained for historical purposes.

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

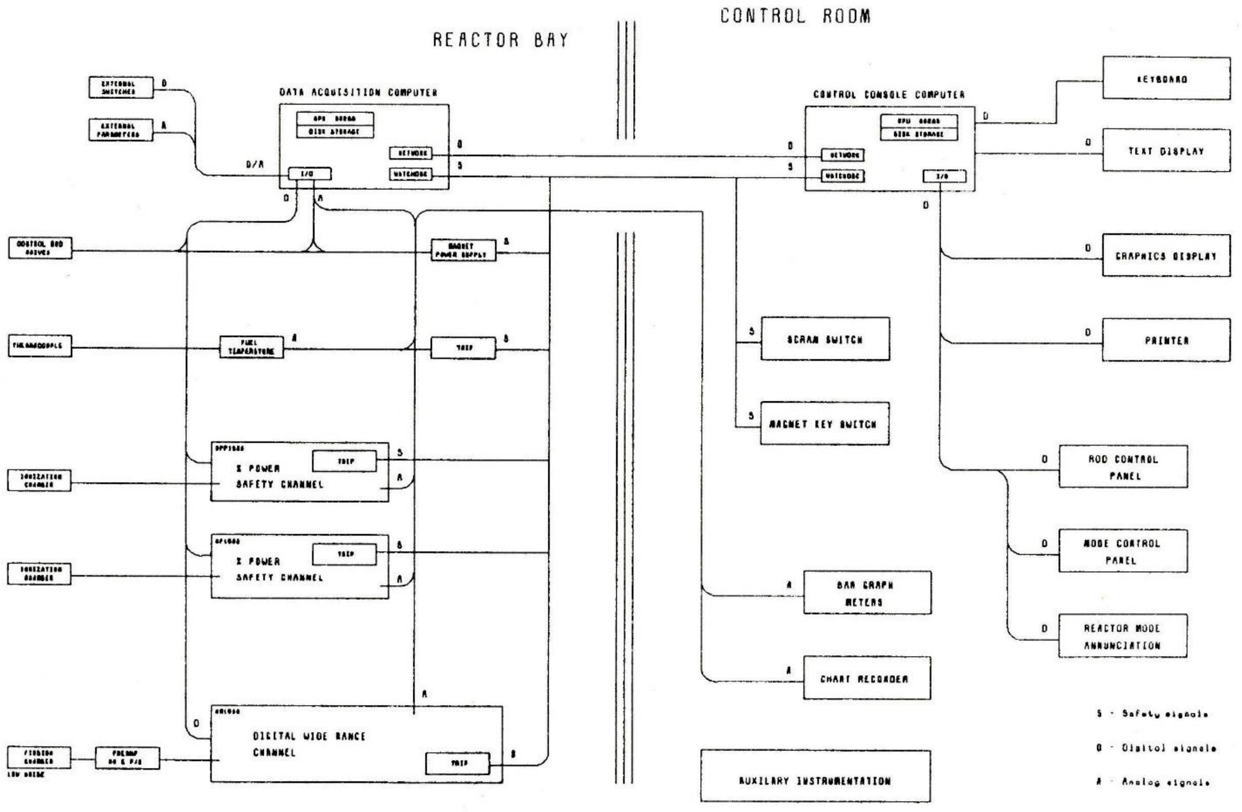

Figure 7.9, Logic Diagram for Control System

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.).

7.1.7. Mechanical Hardware

The reactor installation is 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.

NEMA 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.

NEMA 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.

The electronic enclosure cabinet 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.

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.).

7.2. DESIGN EVALUATION

The TRIGA reactor console[44,45,46] [6,7,8] has developed through the successful operation of many installed facilities throughout the world. The ICS unit design 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 for effective and efficient operation. A multiphase design, development and installation program by the system manufacturer provided the initial demonstration of the system acceptance by analysis and review.

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

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

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

8. 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.

The 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 shut down. The underground distribution system prevents the potential for most external events affecting the power supply, with exceptions that damage the distribution station.

Distributed uninterruptible power supplies backup power to emergency lights, area radiation monitors, and intrusion alarm systems.

9. 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 precast concrete tilt panels, approximately 11 in. (0.2794 m) thick 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. The roof is concrete on steel, supported by structural steel. The roof concrete is approximately 6 in. (0.1524 m) in the center, tapering to about 4 in. (0.1016 m) at the edges to support drainage. 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 (including 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. A fume/sorting hood is installed in the reactor bay, using a separate exhaust fan and isolation damper that discharges into a separate roof stack.

Water temperatures of the heating and cooling coils in the air handling units are controlled by a set of on-site and off-site systems. The on-site heating system is a 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 Figure 9.1. Motor controllers, motors, and dampers are located in a room on the 4th floor. Manual operation controls for both main and purge air systems are in the reactor control room.

An exhaust stack on the roof combines the ventilation exhausts from both the main and the purge systems. As illustrated in Figure 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).

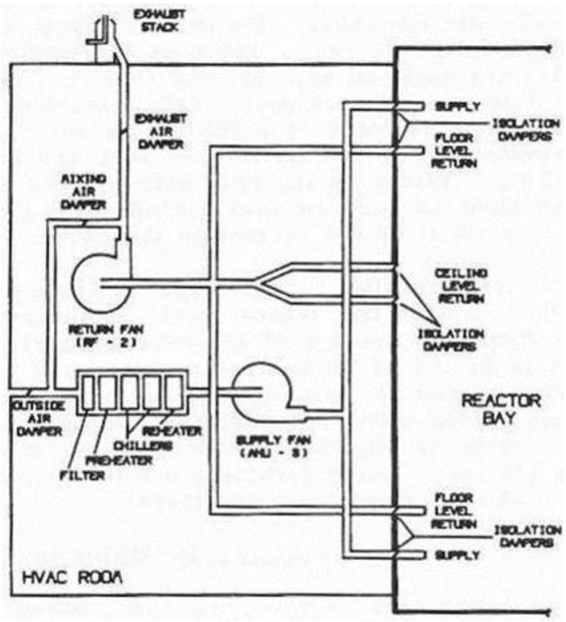

Figure 9.1, Conceptual Diagram of the Reactor Bay HVAC System.

9.2.1. Design Basis

Confinement system ventilation has three modes of operation, reactor run mode, quiescent mode, and confinement isolation. The design goal for HVAC system in the reactor run mode 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 managing temperature and humidity for personnel comfort with two (2) air changes of reactor bay air per hour. The differential pressures are nominally 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. The design goal of the confinement system ventilation during quiescent mode is to minimize energy utilization during periods when the reactor is not operated while maintaining pressure in the reactor bay 0.06 in. below atmospheric pressure. 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. 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 confinement isolation secures fans and dampers in the confinement HVAC, fume/sorting hood, and auxiliary purge system. Provisions are made to allow subsequent operation of the auxiliary purge system with the remaining HVAC confinement in isolation. Release criterion is based on Title 10 Chapter 20 of the U.S. Code of Federal Regulations.

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.

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

The 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 and vents running down the wall and ending near the floor), enhancing mixing and preventing stratification. Air is discharged from the bay through 4 return grills, two parallel ducts to grills near the floor, and two grills near the ceiling. 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 supply and return fans.

Confinement system ventilation discharge is through a stack on the reactor building roof. HVAC components and configuration for each of the modes of operation are described in Table 9.2. 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 Figure 9.2A and B.

Table 9.2, Reactor Ventilation System Modes

SYSTEM	COMPONENT	MODE		
		REACTOR RUN	QUIESCENT	ISOLATION
Confinement HVAC	Control damper	CLOSED	OPEN	CLOSED
	Supply & exhaust control dampers	OPEN	MINIMUM	MINIMUM
	Supply Fan	Controlled for Stack velocity	Constant Speed	OFF
	Exhaust fan	Controlled for bay dp	Controlled for bay dp	OFF
Auxiliary Purge System	Supply & exhaust control dampers	OPEN	CLOSED[1]	CLOSED OPEN[2]
	Exhaust fan	ON	OFF[1]	OFF OFF or ON[2]
Fume/Sorting Hood	Supply & exhaust control dampers	OPEN or CLOSED[3]	CLOSED[1]	CLOSED
	Exhaust fan	ON or OFF[3]	OFF[1]	OFF

NOTE [1]: Mode is set manually

NOTE [2]: Provisions have been made to permit operation of auxiliary purge system in conferment isolation

NOTE [3]: Fume hood is operated on manually, as required and not correlated to reactor operation

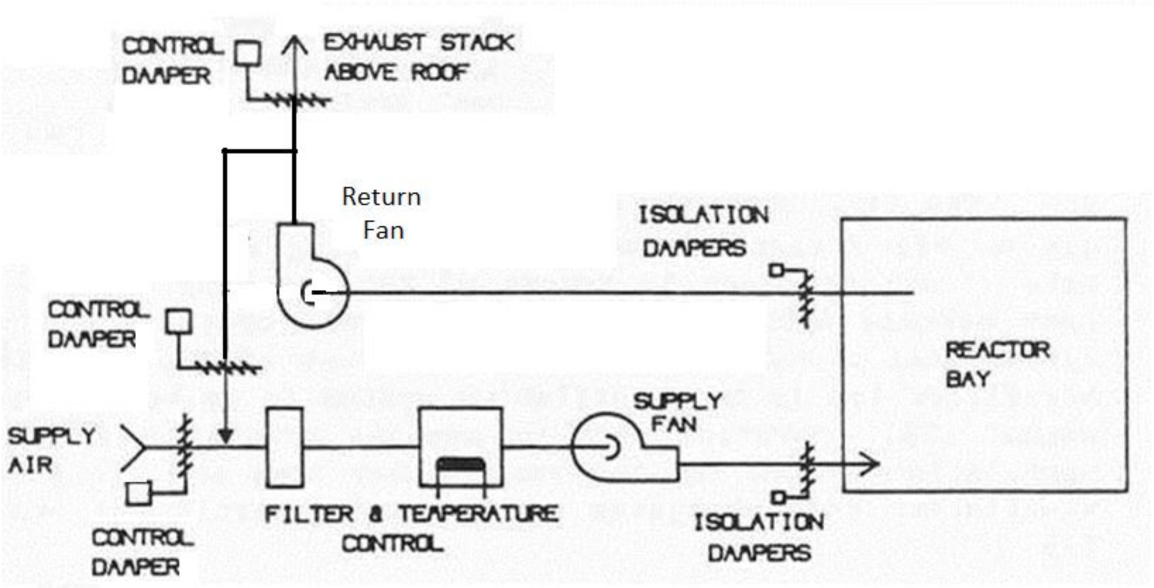

Figure 9.2A, Main Reactor Bay HVAC System.

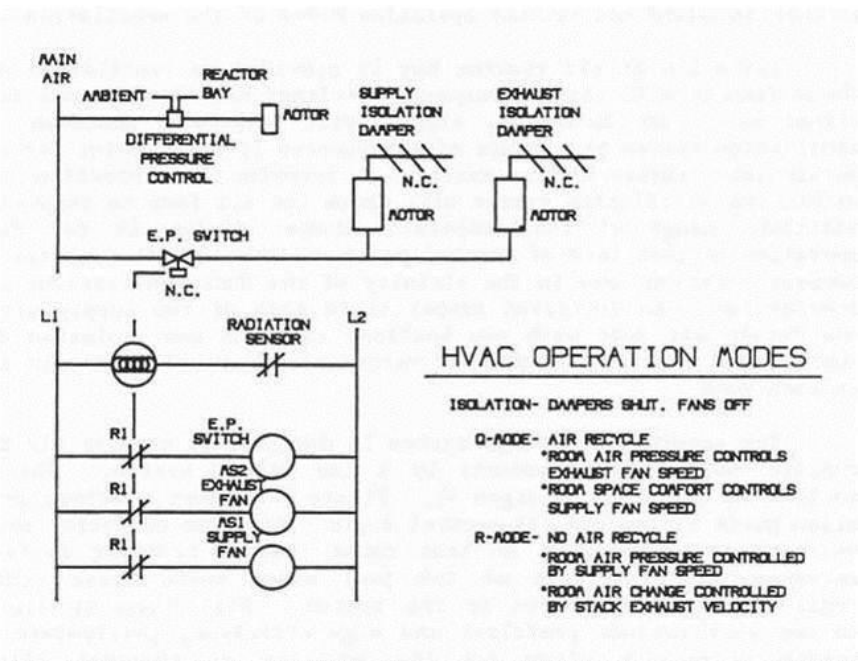

Figure 9.2B, Main Reactor Bay HVAC Control System Control

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 controlled stack velocity, and regulate negative pressure in the reactor bay. In the reactor run mode, the confinement HVAC supply fan is controlled to maintain the reactor bay at nominal minimum 0.06 in. water.

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

In the confinement isolation mode the confinement HVAC and the reactor bay fume/sorting hood are secured; the auxiliary purge system is secured when isolation occurs, but may be manually configured to operate. 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. Separate controls allow the confinement HVAC and the reactor bay fume/sorting hood to be isolated while the auxiliary purge system can be operated.

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 height (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} m 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 Figure 9.1, as well as the fume/soring hood in the reactor bay auxiliary purge system. 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.

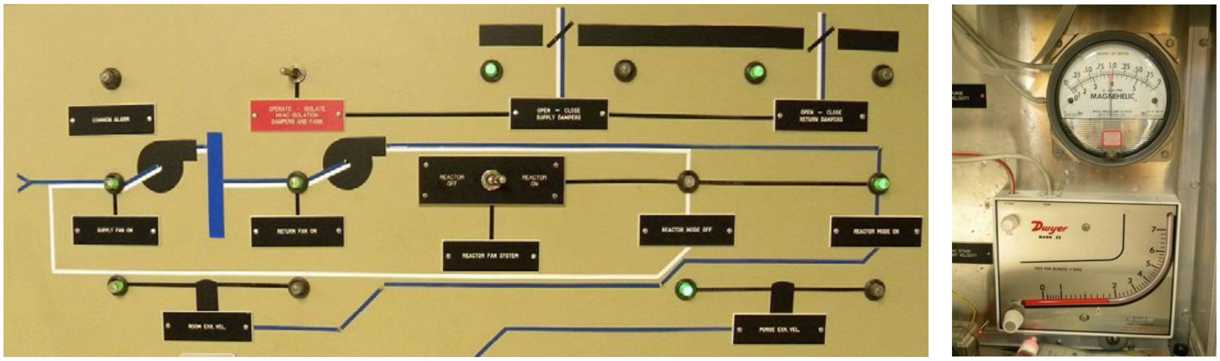

Figure 9.3, Confinement System Ventilation Controls

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.

Confinement HVAC mode is controlled manually by a toggle switch labeled “Reactor Off/ Reactor On.” Reactor Off establishes quiescent mode described above. Reactor On establishes the reactor run mode described above.

A toggle switch labeled “Operate – Isolate” near the top of the panel in the “Isolate” position manually initiates confinement isolation for the confinement HVAC, fume/sort hood and the auxiliary purge system. A remote confinement isolation switch is located on the ground floor of the reactor bay near the equipment access door. A pushbutton switch inside the confinement HVAC control panel is used to reset the confinement isolation relays when the signal has cleared.

Inside the confinement HVAC control panel a switch labeled “HVAC Control” with “ON” and “OFF/ISOLATE” positions allows normal operation in the ON position, and confinement isolation for the confinement HVAC system (fans and dampers) in the OFF/ISOLATE with the option to operate the auxiliary purge system still being available. The HVAC fan mode fans are always operating and dampers are always open except during confinement isolation; the HVAC Control switch secures confinement HVAC independent of a confinement isolation trip signal from the particulate cam.

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 concentrations at the occupational values of 10CFR20 for 70% of fission products for isotopes 84-105 and 129-149. A second alarm setpoint occurs at the occupational values for any single fission product radionuclide in the ranges of 84-105 and 129-149. An alarm and automatic confinement isolation initiates at a count rate of 10,000 cpm (calculated to require an accumulation time of two hours[47]).

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 ensure 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 of argon 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 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 to allow controlled venting of the reactor bay air.

9.3.2. System Description

Piping embedded in the biological shield connect beam port internal space and the rotary specimen rack to a manifold through plug valves. The manifold is connected to piping which also connects to the air-cavity above the pool and an open line to the reactor bay atmosphere. The valves for beam port in use are open to reduce radioactive argon in the reactor bay during operation, the rotary specimen rack can be vented if samples are removed (or loaded) while the radioactive argon is present in the RSR. The ventilation of the air cavity above the pool is normally on to reduce potential buildup of radioactive argon, and the open line reduce the moisture content to prevent damaging the HEPA filters in the discharge to the environment. The discharge is continuously monitored to allow assessing compliance with limits on argon-41 effluent.

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[47] TRIGA Reactor Facility Safety, Nuclear Engineering Teaching Laboratory, The University of Texas at Austin, Safety Analysis Report (May 1991) Section 9.5

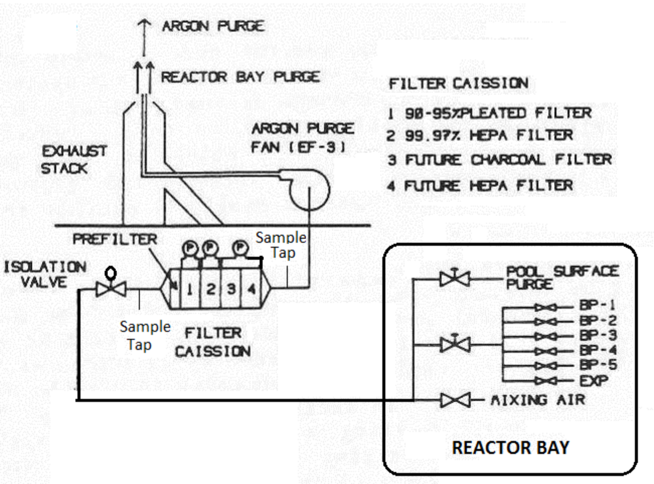

Figure 9.4A, Purge Air System

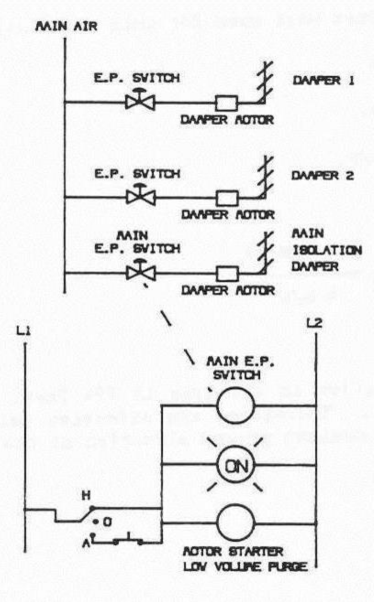

Figure 9.4B, Purge Air Controls

9.3.3. Operational Analysis and Safety Function

The primary nuclide of interest is argon-41. Figures 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.

Purge flow is nominally adjusted for continuous operations with approximately 525 cfm from the pool and a similar dilution flow rate from the reactor bay environment. The dilution flow controls effluent humidity from the reactor pool area to limit possible degradation of the purge system HEPA filters. A purge flow of approximately 4 cfm is drawn from the beam port interior when a beam port is used. The beam guide prevents closure of the outer shutter door, and beam port three is normally purged. The rotary specimen rack is purged prior to loading or unloading for about 10 minutes to control personnel exposure and also to remove hydrogen that may evolve from the polyethylene capsules during irradiation.

The auxiliary purge system may be operated with the confinement HVAC system secured. Since the confinement HVAC operates continuously except during isolation, confinement HVAC can be secured using the HVAC Control toggle switch (inside the HVAC control panel, described previously). Since the auxiliary purge system is equipped with HEPA filters and has the capability for using charcoal filters, operation of the auxiliary purge system could reduce elevated airborne radionuclide contamination in the reactor bay and contain a large fraction of the radionuclides in filtration. Operation to reduce airborne contamination using the purge system requires confinement

HVAC be secured to prevent unfiltered releases, and then bypassing the confinement isolation trip signal.

9.3.4. Instruments and Controls

The auxiliary purge system is controlled from the same panel as the confinement ventilation system. Toggle switches on the control room confinement HVAC control panel open dampers to allow the pool surface purge flow. A manual valve manifold accessible on the ground level of the rector bay independently manages flow from experimental facilities. A separate manually operated valve in the same general area controls the amount of dilution flow to the purge 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 has a Continuous Air Monitor (CAM) to monitor airborne radioactivity in the purge system effluent. The CAM is calibrated to record the ^{41}Ar equivalent concentration (the detector is sensitive to the energies of some fission product gasses). The CAM has local and control room indications and alarms. The alarm setpoint corresponds to 2E-5 μCi/ml (50 times the ^{41}Ar effluent limit in 10CFR20, Appendix B)[47].

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. [redacted]

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

A standard TRIGA fuel handling tool is used to remotely grapple irradiated fuel elements. Most routine fuel storage is intended to be inside the reactor pool. 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 fuel storage racks in the reactor pool. Other locations not in the pool will also provide storage for radioactive non-fuel materials.

Space inside the reactor pool is adequate for a large number of fuel racks. Designated pool racks store fuel that has been disqualified for use in the UT TRIGA reactor. The racks are aluminum, suspended from the pool edge by connecting rods. Elements are stored six per rack in a linear array. Each rack is 24 inches long (60.96 cm.) by 12 inches wide (30.48 cm.) by 3 inches (7.62 cm.) deep and is generally located below more than 8 feet of water shielding. 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. If a storage requirement of 80% of the core grid capacity is specified, then 16 racks with a total of 96 positions would be necessary.

Special storage wells are located in the floor [redacted]. Nineteen elements may be stored in specially designed racks in each well, with a total capacity for two racks (Figure 9.5). The wells are pits [redacted] Water may be added for shielding or cooling. If more than 700 grams of contained ^{235}U are stored in a fuel storage well a criticality alarm system is required unless the wells are configured as pools (i.e., the stored elements are underwater). Spaces in the rack provide a storage array for the fuel equivalent to the innermost three rings of the reactor core, including one element in the center.

[redacted]

Figure 9.5a, Storage Well

Figure 9.5b, Fuel Storage Closure

A fuel transfer cask (Figure 9.6) has been developed that allows transfer of 4 irradiated fuel elements within the rector bay. The transfer cask has a bottom loading port and top access with a removable lead shield. The shielding in the cylindrical cavity is lead shot, covered on the top with the polymeric material used for waterproofing the shielding at the top of the pool tank.

[redacted]

Figure 9.6, Fuel Transfer Cask

[redacted]

[redacted]

9.4.3. Operational Analysis and Safety Function

Benchmark 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 TRIGA reactors, with good performance history although the first generation tool occasionally released an element if pressure was not maintained on the tool operator.

MCNP calculations verified acceptable values for k_{eff} and dose rates for the fuel transfer cask. The fuel transfer cask was designed to be less than 5500 lbs, well within the crane capacity and allowing movement by a pallet jack.

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 [redacted] 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 most areas of the facility, 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[48,49]. 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.

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.

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[48] Code of Federal Regulations, Chapter 10 part 20, U.S. Government Printing Office, 1982.

[49] Dorsey, N.E., Properties of Ordinary Water-Substance, Reinhold Publishing Corp., New York p. 537.

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 all areas of the building except the reactor bay, 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 ventilation system has smoke detectors that provide a warning of problems within the reactor bay.

Manual protection equipment includes a wet standpipe system in each building stairwell. Portable extinguishers such as CO_2, 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.

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. A UT campus wide communications system provides information for fire events with an interface to the NETL public address system.

9.5.5. Technical Specifications, Bases, Testing and Surveillances

There are no Technical Specifications associated with fire protection.

9.6. 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.

9.6.1. Design Basis

Communications is required to support routine and emergency operations.

9.6.2. 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 and voice over internet protocol. 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. 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.

9.6.3. 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.

9.6.4. Instruments and Controls

As specified above.

9.6.5. 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.

9.7. CONTROL, STORAGE, USE OF BYPRODUCT MATERIAL (INCLUDING LABS)

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

9.7.1. Design Basis

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

9.7.2. System Description

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.

9.7.3. Operational Analysis and Safety Function

Engineered controls permit safe handling of radioactive materials.

9.7.4. Instruments and Controls

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

9.7.5. 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.

9.8. 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 (Chapter 10).

9.8.1. Design Basis

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

9.8.2. 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

9.8.3. Operational Analysis and Safety Function

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

9.8.4. Instruments and Controls

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

9.8.5. 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.

9.9. 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.

9.9.1. Design Basis

Service air is provided to support laboratory and service operations with high-capacity applications (including the transient rod). Instrument air is intended to support HVAC and chill water controls for reactor operations.

9.9.2. System Description

The laboratory dual compressor system provides oil free compressed air for laboratories and services. The instrument air compressor provides air to HAVC pneumatic controls, pool cooling flow controls. The laboratory air compressor provides air to shops and to the transient rod drive system.

9.9.3. Operational Analysis and Safety Function

The two systems are connected through a manual shut off valve, providing maximum flexibility in the event of a system (or associated air dryer or filtration) failure.

Failure of the laboratory 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 and HVAC dampers to shut, stopping pool cooling and securing HVAC. Securing chill water flow is a fail-safe condition that prevents potential leakage from the pool to the chill water system. Securing HVAC is a fail-safe condition, assuring that there is no potential for inadvertent release of radioactive material into the environment.

9.9.4. Instruments and Controls

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

9.9.5. Technical Specifications, Bases, Testing and Surveillances

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

10. 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, specialized experiments, and medical isotope research. In-core experimental facilities are used mostly for activations for neutron activation analysis and for detector calibration related isotope production. Beam port facilities largely utilize neutrons for prompt gamma analysis, fission product analysis, imaging, or testing electronic components. Laboratory facilities are utilized for radiation detection and measurement or radiochemistry. The neutron generator facility contains a D-T neutron generator capable of high energy neutron activation studies. The UT-Austin TRIGA does not have thermal columns or irradiation rooms associated with the reactor.

The following is a list of experimental facilities at NETL:

1. In-core facilities
    a. Central Thimble
    b. Fuel element positions
    c. Pneumatic transfer systems
    d. Three-element Facility
    e. Six- and Seven-Element Facilities
2. In-reflector facilities/Rotary Specimen Rack
3. Automatic sample transfer facilities
    a. Manual
    b. Automatic
4. Beam ports
5. Cold neutron source
6. Non-reactor experiment facilities
    a. Neutron generator room
    b. Laboratories
        i. General purpose laboratories
        ii. Neutron activation analysis laboratory
        iii. Radiation detection laboratory
        iv. Sample preparation laboratory

The facility runs experiments in three basic categories: in-core irradiations, beam port experiments, and non-reactor experiments. The in-core experiments include irradiations for neutron activation analysis, generation of radioactive isotopes, and neutron damage studies. Beam port experiments utilize neutrons for various nuclear analytical techniques and neutron damage studies. Non-reactor experiments including utilization of the D-T neutron generator or other radiation sources.

Experimenters work with licensed reactor operators for experiment planning, facility access, and facility utilization. Experiments that are designed to remove or insert while the reactor is operated are analyzed to assure the maximum reactivity worth of the experiment is less than $1.00. Experiments held firmly in place by a mechanical device or by gravity with weight 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 are analyzed to assure the maximum reactivity worth is less than $2.50. The total inventory of experiments in the core is managed to assure that the total reactivity worth of all experiments is less than $3.00. Experiment design and analysis ensures 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. Experiment design will not cause bulk boiling of core water. Experiments shall be designed so that they do not mechanically interfere with control rods and do not cause control rod shadowing or shadowing of power level instrumentation. Experimental design will minimize the potential for industrial hazards, such as fire or the release of hazardous and toxic materials. If the failure of an experiments (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 a decay time of five-minutes following irradiation may be used in radioactive inventory calculations to account for processing prior to potential exposure). 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.

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 is assumed to escape. 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 are assumed to escape. If a material boiling point is above 130ºF and vapors formed by boiling this material could escape only through an undisturbed column of water above the core, at least 10% of these vapors are assumed to escape.

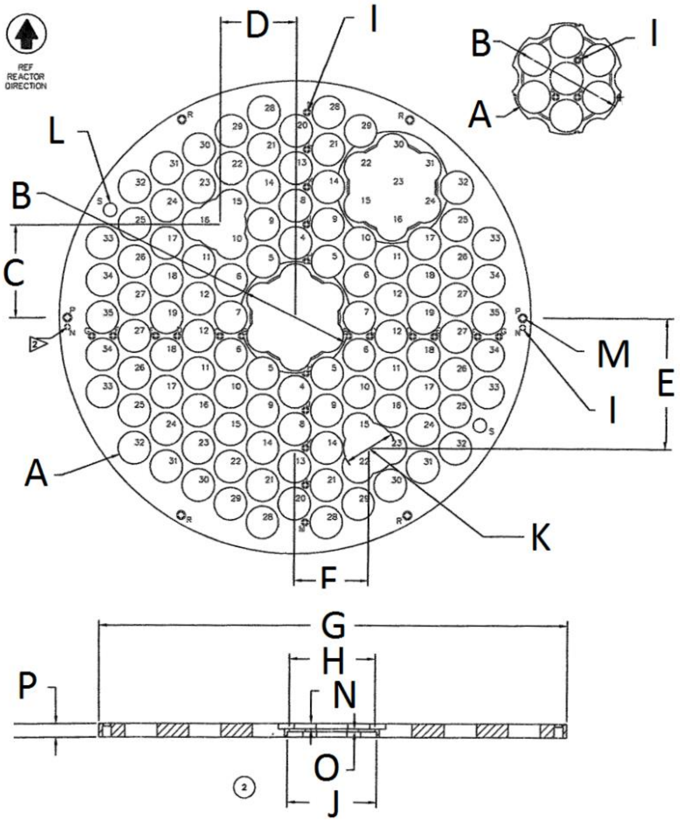
Use of explosive solid or liquid material with a National Fire Protection Association Reactivity (Stability) index of 2, 3, or 4 in the reactor pool or biological shielding. Explosives are not allowed above an equivalent of 25-milligrams of TNT. Explosives will be irradiated in a container capable of withstanding twice the pressure the experiment can potentially produce.

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 argon purge system draws air from the cavity above pool water, beam ports, and the rotary specimen rack. A continuous air monitor is installed to measure the ^{41}Ar in the argon purge system. The status of access controls to beam port facilities is displayed on the reactor console.

The Reactor Oversight Committee (ROC) reviews reactor-based experiments and other experiments utilizing radiation sources such as the D-T neutron generator. A safety analysis is written by the experimenter and often presented in an oral format to the ROC. A ROC subcommittee may be used to review the 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, a pneumatic sample transfer system that displaces one fuel element, 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. The grid plate and cutouts are illustrated in Figure 10.1.



LABEL	Dimension (inches)	Dimension (m)
A	1.505	0.038227
B	5.140	0.130556
C	4.285	0.108839
D	3.464	0.087986
E	5.999	0.152375
F	3.464	0.087986
G	21.750	0.552450
H	4.002	0.101651

LABEL	Dimension (inches)	Dimension (m)
I	0.203	0.005156
J	4.175	0.106045
K	2.370	0.060198
L		
M	3/8-16 UNC-28	
N	0.248	0.006299
O	0.126	0.003200
P	0.620	0.015748

Figure 10.1, Core Grid Plate Design and Dimensions

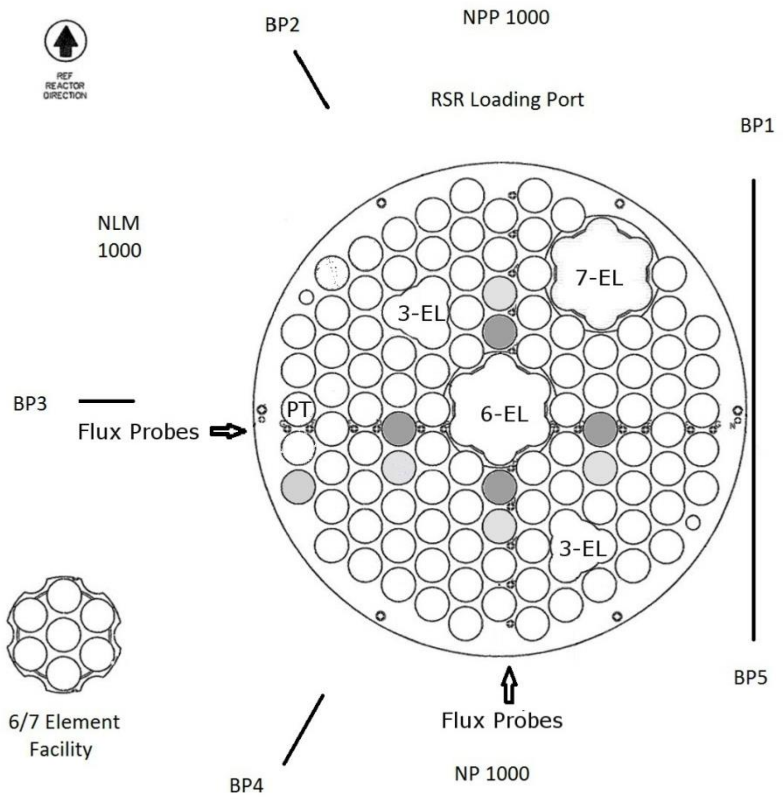


Figure 10.2, Reactor Core Diagram

10.2.1. Central Thimble (In-Core Facility)

10.2.1.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. 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.

Experimental objectives for normal operations maximize activation, gamma irradiation, or reactivity. The central thimble is used to generate the maximum activation or radiation damage in the core for isotope production, neutron activation analysis, or 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.

10.2.1.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 inch (0.19 m) below the lower grid plate and 8.7 inch (0.22 m) above the safety plate. The central thimble tube outer diameter is 1.5 inch (3.81 cm.), with 1.33 inch (3.38 cm.) inner diameter. There are four ¼ inch (0.00635 m) holes in the central thimble approximately three inch (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.

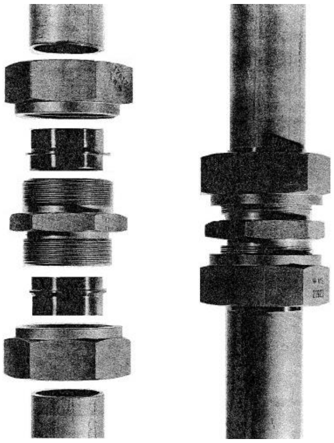


Figure 10.3, Central Thimble Union Assembly

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 alloy has excellent joining characteristics, and good acceptance of applied coatings. The alloy combines relatively high strength, good workability, and high resistance to corrosion. The central thimble tubing is anodized to further control potential corrosion.

10.2.1.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% $\Delta k/k$ ($1.29), and that replacing the central thimble with a void would result in a reactivity change of -0.15% $\Delta k/k$ (-$0.21). As noted above, voiding of the portion in the core region of the central thimble (-for development of a radiation beam from the central thimble) is prevented passively by design.

10.2.1.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 consequence 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 fifteen inch (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.

10.2.1.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.

10.2.1.f. Physical Restraints, Shields, or Beam Catchers

The central thimble is bolted to the structure of the bridge over the reactor pool. The central thimble facility is shielded by the same materials that shield the reactor core (water and concrete).

10.2.1.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.

10.2.1.h. Safety Assessment

The central thimble facility does not carry significant risk during reactor operation. Reactivity changes could occur as a result of sample introductions. Negative reactivity changes may occur due to sample introduction into the facility or for water introduction into the facility when it is voided. Potential reactivity changes from sample insertion are assessed as part of the experiment review process. Sample reactivity is operationally assessed on an individual basis 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 three 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.a. Pneumatic Sample Transit System

10.2.2.a.1. 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.

10.2.2.a.2. 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 two 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 (in.)	1.25	0.875
Terminal point ID (in.)	1.085	0.685
Terminal point tube Thickness (in.)	0.0825	0.095
Transport tube OD (aluminum) (in.)	1.25	0.875
Transport tube ID (aluminum) (in.)	1.12	0.745
Transport tube Thickness (in.)	0.065	0.065
Transport bends OD (polyethylene) (in.)	1.5	1
Transport bends ID (polyethylene) (in.)	1.25	0.75
Transport bends Thickness (polyethylene) (in.)	0.125	0.125
Polyethylene transport capsule (in.)	0.985 OD X 4.75 L	0.650 OD x 2.15 L
Total transport tube length (feet)	90	90
Transit time (seconds)	6	6

Terminals are fabricated from aluminum 6061 alloy with the associated radioactive nuclides. One 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 weather head) 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 (3.175 cm) diameter (OD) aluminum (6061) tube. Transport bends are 1.5-inch (3.81 cm) 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). 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 diameter (OD) aluminum tube for straight sections and 2.25-inch (5.715 cm) diameter (OD) 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 capsules without any reuse to transport the sample capsule.

10.2.2.a.3. 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% $\Delta k/k$ (-\$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.

10.2.2.a.4. Radiological Assessment

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 (7.7x10^{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 (0.508 cm) thick sheets. They line the interior of the irradiation terminal that has in inner diameter of 0.685 inches (1.7399 cm) and a height of twenty inches (50.8 cm). 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.

10.2.2.a.5. Instrumentation

Instrumentation supporting the pneumatic transit system includes two control systems (located in both the control room and in the laboratory associated with the system) and a radiation monitor in the laboratory hood near the end of one ex-core terminal. One control system has a timer with preset values for ejection from the core terminating in a laboratory hood. Each sample is inserted by an operator in the laboratory with manual action for each insertion. The other control system allows automatic insertion and removal of up to thirty samples to a counting system station in the laboratory using PLC-based system with programmed logic for start and stop signals. Both systems are enabled in the control room and have the capability to force ejection from the core.

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 high activity samples are measured.

10.2.2.a.6. Physical Restraints, 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.

10.2.2.a.7. 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 five hundred 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

10.2.2.A.8. 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 unexpectedly 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.B. Three-Element Irradiator

10.2.2.B.1. Description

The three-element facility is typically used to generate radioisotopes for research or neutron activation analysis.

The three-element experiment facility displaces three 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 predominantly for thermal neutron irradiation. A cadmium liner is used when reduced thermal neutron flux and enhanced epithermal irradiation is desired.

10.2.2.B.2. Design & Specifications

Upper and Lower Grid Plate

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-inch (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.

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 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 (1.27 cm) 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 (5.08 cm) 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 so 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.

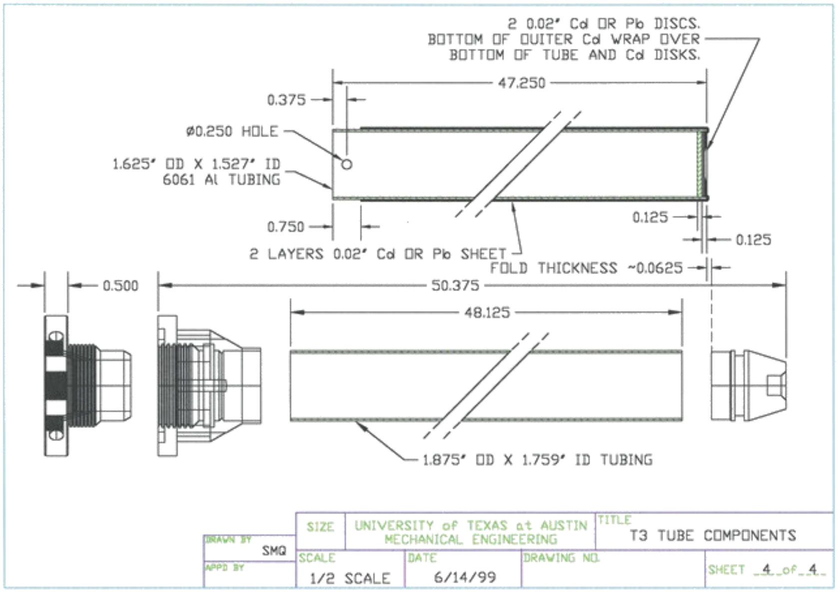
Three-element Facility Canister

The facility uses a sealed canister with a usable space 1.527 inch (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 inch (0.047625 m). The canister wall is 0.1 inch (0.000254 m). The inner liner is 1.625 inch (0.041275 m) outer diameter, and 1.527 inch (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 inch (1.279525 m) with the length of the usable volume 48.125 inch (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 (609.6 cm) 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 inch (0.01588 cm) 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 (118.745 cm), 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.



Fig, 10.4, Three-element Irradiator

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 lead nor cadmium has potential for significant chemical activity in contact with aluminum.

10.2.2.b.3. 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 movable experiment is $1.00. The reactivity limit for a single fixed experiment is $2.50. 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.

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 final 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. Finally, the most plausible accident condition (flooding of the irradiation canister volume with water) was evaluated.

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 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 for a movable 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 $1.00.

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. The 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.

10.2.2.b.4. 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 2×10^{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 4.37×10^{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 µR h^{-1} m^{-3} resulting from ^{41}Ar.

Based on 50.374-nch (127.95 cm) length of the air volume in the canister at 1.527-inch (.879 cm) 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.

10.2.2.b.5. Instrumentation

Instrumentation is not typically used 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.

10.2.2.b.6. 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.

10.2.2.b.7. 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.

10.2.2.b.8. Safety Assessment

A. 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. The 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.

B. Temperature

The physical design of the cylindrical irradiation canister with internal aluminum cylindrical insert provides a 0.072-inch (0.183 cm) 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.

C. Pressure

Air pressure relief for excessive pressure buildup in the canister is a design feature to protect the canister from rupture.

The yield stress for the T-6 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 T-6 to T-0. The design of the top fitting controls the pressure with a double O-ring seal and two one-eighth-inch (0.3175 cm) 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 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 released 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.

D. LOCA potential

The canister is completely submerged during irradiation and does not offer any leakage path for pool water.

10.2.2.c. Six- and Seven-Element Irradiator

10.2.2.c.1. <u>Description</u>

The six and seven element irradiators are large in-core facilities to perform neutron irradiations. The seven-element irradiator is located in a 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. The seven-element irradiator is placed in the location that overlays part of the outer three fuel rings. The seven-element irradiator has largely been utilized for irradiation of circuit boards and irradiation of samples for neutron activation analysis.

10.2.2.c.2. <u>Design and Specifications</u>

The irradiation can for the six- and seven element facilities is composed of 6061-T6 aluminum and contains a 0.08 in (2 mm) thick borated aluminum (B) liner. The inner diameter of the irradiation can is 2.25-inch (5.715 cm). The boron concentration is 4.5% by weight in the 1100 series aluminum alloy. The boron is enriched to greater than 95% ^{10}B. 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 (132.08 cm). 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 three-element 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 centers 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-feet (91.4 cm) 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.

10.2.2.c.3. 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.

10.2.2.c.4. Radiological Assessment

From a radiological perspective, the seven-element irradiator is similar to a three-element irradiator. However, the seven-element irradiator has a boron (95% ^{10}B) liner instead of the three-element canister cadmium liner. The primary ^{10}B absorption 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 performed in the seven-element irradiator require analysis on an individual basis.

10.2.2.c.5. Instrumentation

Electrical testing is performed at 1 kW and a junction box for testing is elevated to a position where neutron flux is low enough that activation is not significant. No other instrumentation is typically 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.

10.2.2.c.6. 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.

10.2.2.c.7. 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 watertight tube going to the pool surface. The facility allows for direct monitoring of electronics during irradiation.

10.2.2.c.8. Safety Assessment

A. 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.

B. Temperature (Lead)

Calculations were performed to ensure that the lead in the sleeve would not approach melting temperatures for lead (325 °C) 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}. A collision heating (+F6) tally was utilized with the MCNPX model to determine energy deposition in the lead sleeve. Since the temperature increase was minor, thermal expansion of the lead and aluminum clad are neglected. Additionally, a 1/16-inch (0.1588 cm) 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.

C. Pressure (irradiation Can)

Through the aluminum tube and Tygon tubing, the irradiation can is open to atmosphere. Therefore, no internal pressurization will occur.

D. Pressure (Lead Sleeve)

The lead sleeve consists of two aluminum tubes attached together by two end caps. The lead was not poured within the two tubes completely to the top of 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.

E. 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.

F. Structural

To minimize the risk of dropping the sleeve and can, they 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 and the irradiation canister are stored in the pool when not in use, secured to the top of the pool.

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 has 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

10.2.3.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 distribute the neutron fluence received by each sample.

10.2.3.b. Design Specifications

The RSR housing is a cylindrical canister with an internal diameter of 22 inches (55.88 cm), and an outer diameter of 27 2/7 inches (69.31 cm). Specimen positions are 1.23 inch (3.18 cm.) in diameter by 10.80 inch (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 samples and an enclosed drive shaft for rotating the samples.

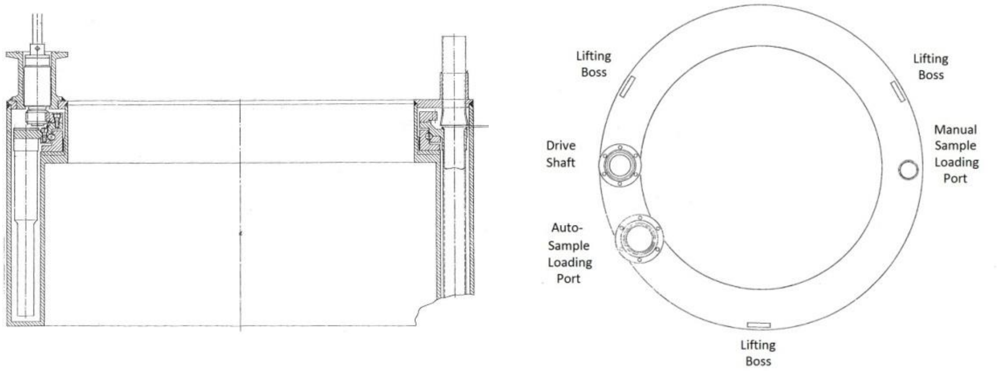


Figure 10.5, Rotary Specimen Rack Diagram

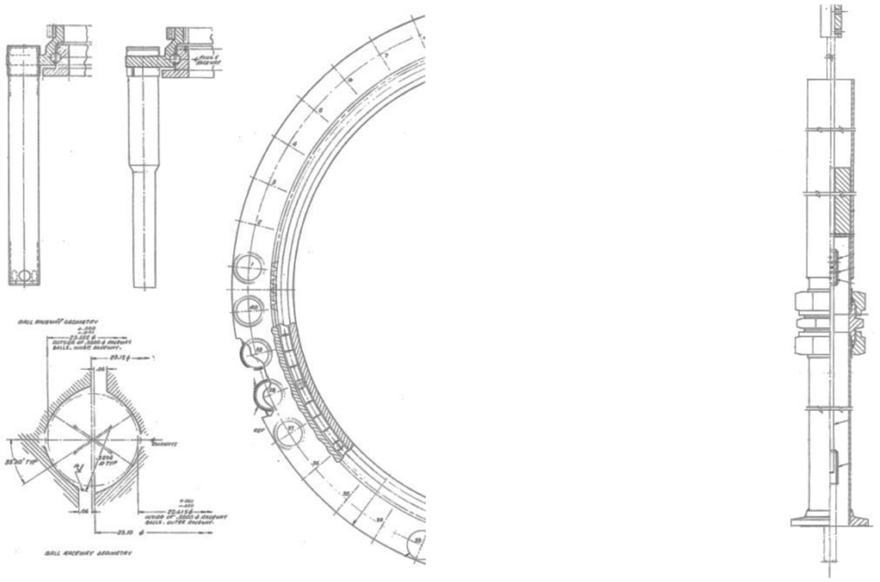


Figure 10.6, Rotary Specimen Rack Raceway Geometry

The RSR contains raceways manufactured from titanium forgings supporting sample rotation as illustrated in Figure 10.6. 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 (62.23 cm). 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 (2.845 cm) tall by 0.56-inch (2.422) wide ring (23.12 OD, 22 in ID). Four cylindrical titanium separators space 0.045-inch (0.114 cm) radius ball bearings; 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 (4.75 m) tall by 5 5/8 inch (13.208 cm) 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 (3.502 cm) in diameter equally spaced on a 26.312-inch diameter (66.832 cm) circle. The upper section is formed from a ring 2 5/8 inches (6.6675 cm) thick (24 1/8-inch OD by 21 ½ inch ID) to accept spur gear. A spur gear is secured to the top of the outer raceway.

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.

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

The overall length of all specimen tubes is 11.44 inch (29.058 cm), with wall thicknesses of 0.058 inches (017 cm). The top of the tubes is flared 45° to-1.62-inch 94.064 cm) OD. Position 1 is in two sections. The top section is 5.5-inch (13.97 cm) tall, with 1 3/8-inch (3.493) OD. The bottom section OD is 1 inch (2.54 cm). Positions 2 through 40 have an OD of 1 3/8 inches. The bottom of the cylinder is penetrated by 3 ½ inch (8.89 cm) holes in the wall at 120° intervals. The bottom of tube is terminated with a ring 0.06 inches (0.152 cm) thick that has a ¾ inch (.905 cm) 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.

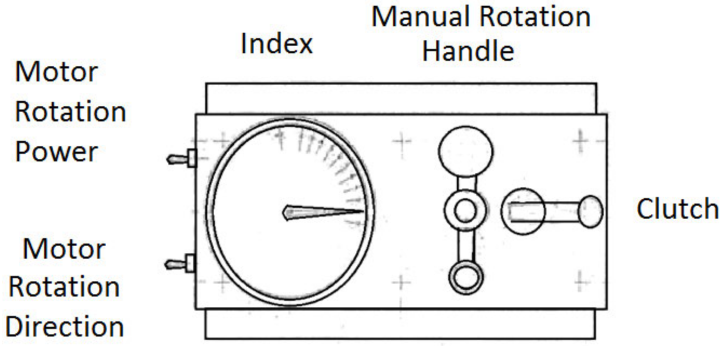


Figure 10.7, Rotary Specimen Rack Rotation Control Box

10.2.3.B.1. 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.

10.2.3.b.2. Radiological Assessment

The neutron flux at full reactor power within the RSR facility is $1x10^{12}$ 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.

10.2.3.b.3. 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.

10.2.3.b.4. 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.

10.2.3.b.5. 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 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.

10.2.3.b.6. Safety Assessment

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

10.3.1. 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 radially 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 [redacted] The five beam ports are divided into two categories: tangential beam ports and radial beam ports.

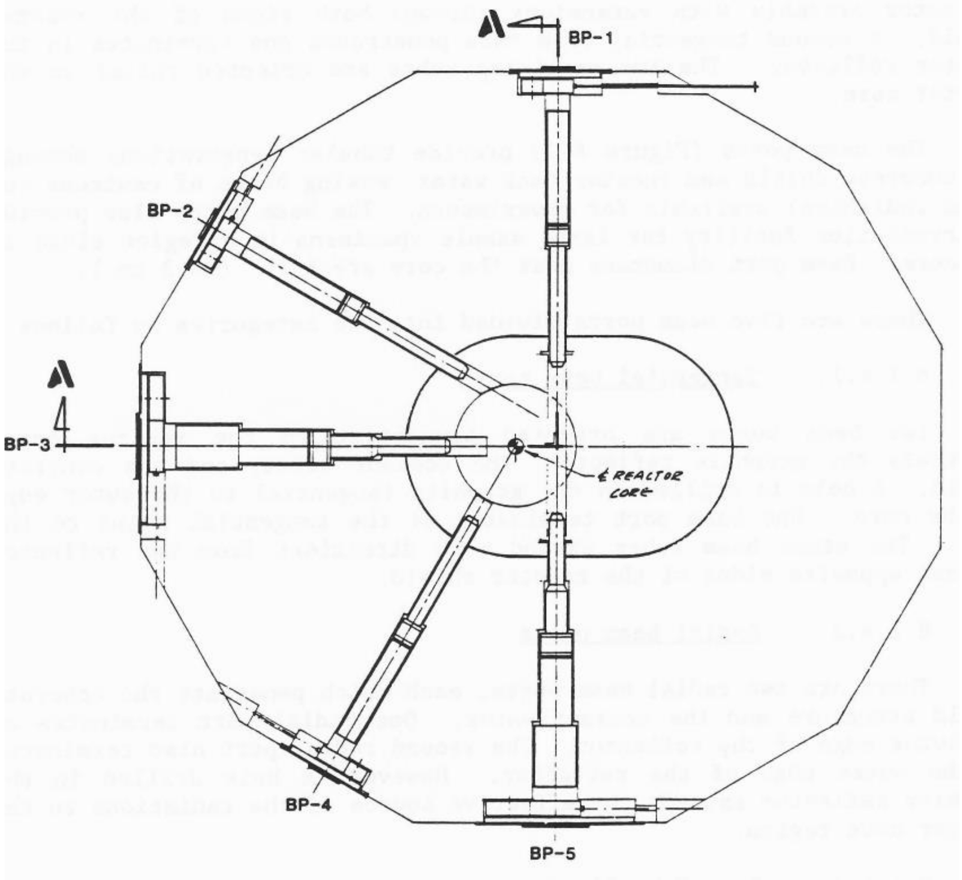


Figure 10.8, Beam Port Layout

10.3.2. 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 radiation to the reactor core region.

Experiments using the beam ports may be passive or active. Passive experiments have no connections from the experiment inside the beam tube to external supplies such as electrical power, temperature controls, fluids, etc. Active beam port experiments have samples or apparatus with external connection outside the beam tube for providing signal measurement, electrical power, heating, and/or cooling while being irradiated.

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.

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 [redacted] in diameter. Beam ports 3 and 5 have the same configuration as the other beam ports, except that the graphite portion is [redacted] in diameter, with a change to 8 inch (20.3 cm) in diameter to provide graphite shielding in the [redacted] 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 [redacted] in diameter and 42 inches (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 inches (1.22 m) and diameter of 15 inches (38.1 cm) for the outer portion of the tube. A handle on the outer end of this plug is provided for manual handling.

10.3.3. Reactivity

Core reactivity changes occur when neutrons that are normally lost by going into a beam port are scattered back into the core by objects placed in the beam port. Experiments utilizing beam port facilities require analysis on an individual basis. If a collimator and/or filter assembly placed within a beam port has no portion closer than 2 feet from the outer edge of the core (fuel), reactivity changes due to insertion of 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. MCNP analysis was performed to evaluate the reactivity effects of filling the beam port with cadmium resulting in –6.23 cents. Similarly, if the beam port were to be filled with graphite, the reactivity effects result in +24.31 cents. Both measures were significantly lower than the technical specifications requirement of $1.00 for movable experiments. Thus, any experiment that does not include fissionable material will not exceed the reactivity requirements.

Insertion or removal of beam port samples requires prior approval by a Senior Reactor Operator.

10.3.4. 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.

Slots in adjacent shield plugs to allow cabling or piping may be offset or fabricated with non-linear configurations to prevent streaming paths. The beam port lead shutter may be partially closed to mitigate streaming. External shielding may be used to control dose rates.

Radiation monitors around the reactor bay area provide information during beam port operation. External neutron beam fluxes are controlled by temporary shielding or restricting access where radiation levels are elevated in a beam port experiment.

10.3.5. Instrumentation

A position switch mounted in the front of the inner plug and with electrical connector at the rear of the plug. The circuitry can therefore indicate on the console when the plug is removed. An alternate configuration has been developed that monitors access to the area around the beam port where access controls are used.

Instrumentation that might be used as part of an experiment program will be evaluated as part of the experiment review and approval process. Experiments that require interaction such as measurement, electrical power, or gas/fluid flow require a path for cabling and piping.

10.3.6. Physical Restraints, Shields, or Beam Catchers

While in use, each beam port has external shielding and may have controlled access through concrete walls with 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.

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 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 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.

10.3.7. 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 hours up to an entire day of operation. Experiments on multiple beam port facilities may be run simultaneously.

10.3.8. 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. Placement of sharp objects, 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.

Passive experiments 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 from passive experiments.

Heating loads to the beam ports from passive experiments with collimators, neutron filters, or other materials inserted at a distance no closer than 2 feet (60.96 cm) 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.

While an experiment placed in a beam port will not change the cooling channel configuration of the UT-TRIGA reactor core, an active experiment could affect the thermal hydraulic properties of the core if sufficient heat were generated and transferred to the water near the core. A non-reactor experiment was performed to test the heat generated in a 6-inch diameter pipe with 0.25-inch walls. The experiment utilized a 100-Watt heater and temperature measuring device to evaluate the heat imparted to the pipe over time. The experiment concluded that the pipe would heat (in air) to approximately 100 deg C. While inefficient, heat evolved from an experiment could transfer to the beam port structures and then to the water. Due to this relatively inefficient transfer of heat, negligible heating of the water near the core will occur provided the temperature of the experiment is no more than 100 deg C if the active experiment does not generate heat greater than 100 Watts.

Mechanical stresses resulting from the weight of collimators, filter pieces, or equipment 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

10.4.1. 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.

The curved neutron guide, the converging neutron guide, and the prompt gamma activation analysis system are currently being used independent of the cold neutron cooling system.

10.4.2. 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 cold head 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 of neon consists of condensation and subsequent vaporization.

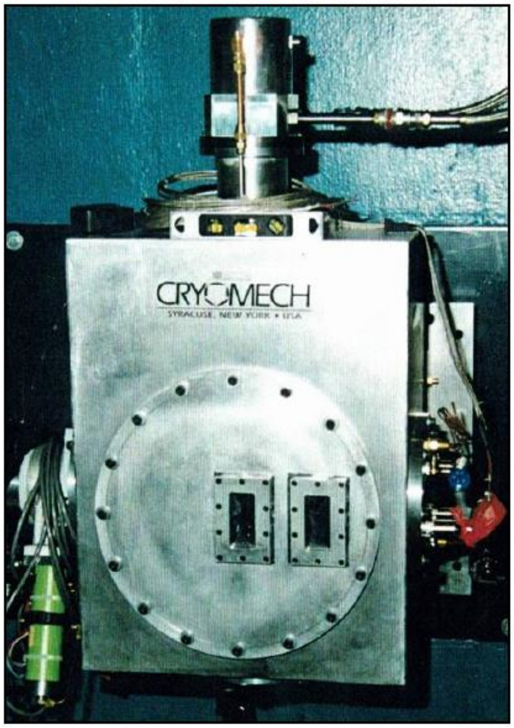


Figure 10.10, Cryomech Cold-Head and Vacuum Box

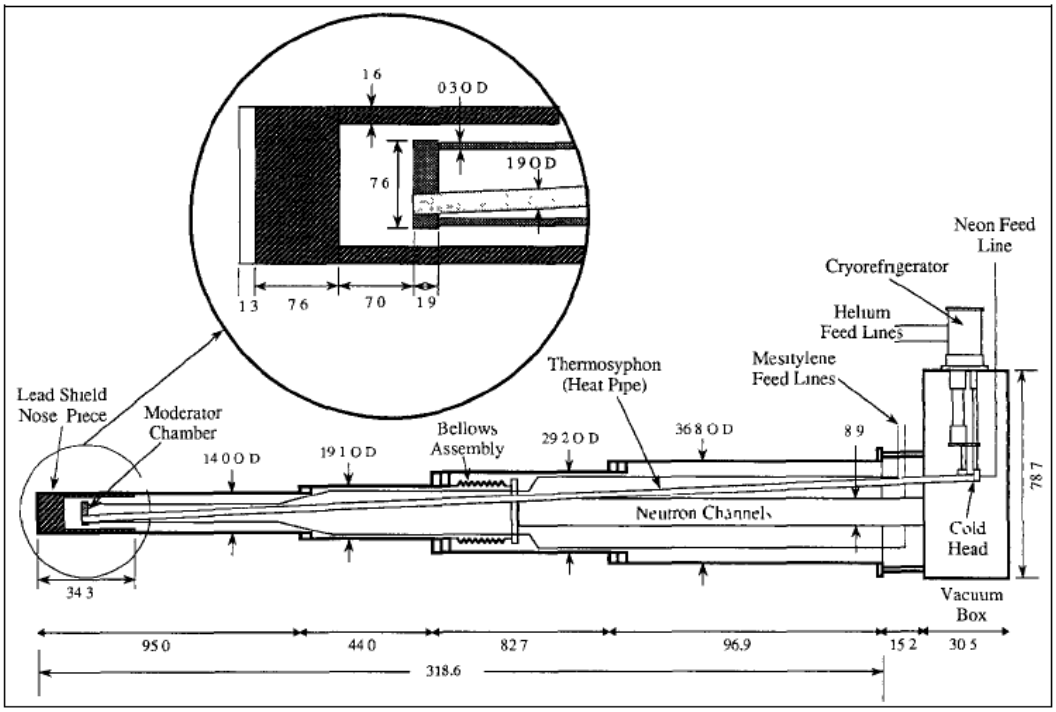


Figure 10.11, TCNS Vacuum Jacket and Other Instruments (units in cm)

10.4.2.a. 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.

10.4.2.b. Radiological

At the end of the TCNS beam line the thermal equivalent neutron flux was measured at 1 x 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. 1 mrem/hr. The neutron beam line can be turned on and off when via the remote controlled boral shutter.

10.4.2.c. 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.

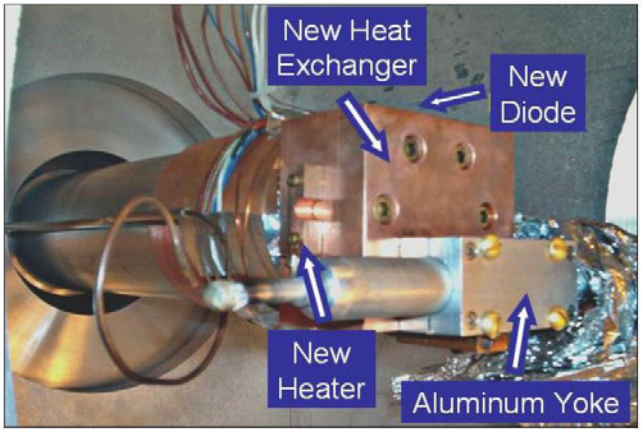


Figure 10.12, Silicone Diode and Heater Relative to Cold-Head

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 can be switched to 10 μA. The heater provides 25 W of heating (25 V @ 1 A) and connects to the computer through an RS-232C serial port.

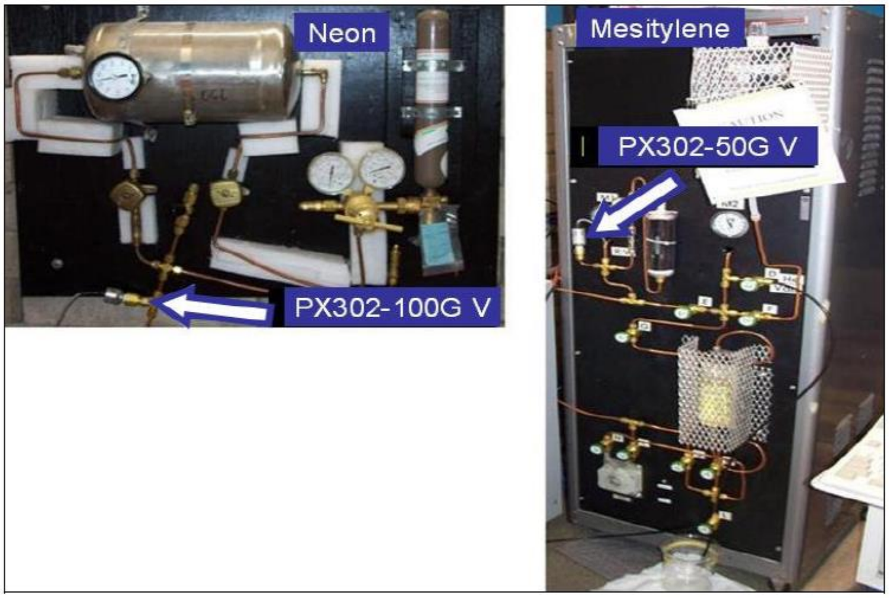


Figure 10.13, Neon and Mesitylene Handling System with Pressure Transducers

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.

10.4.2.d. Physical Restraints, Shields, or Beam Catchers

The TCNS system has an array of restraints, shields, and beam catchers. Figure 10.14 shows this 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.

10.4.2.e. 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.

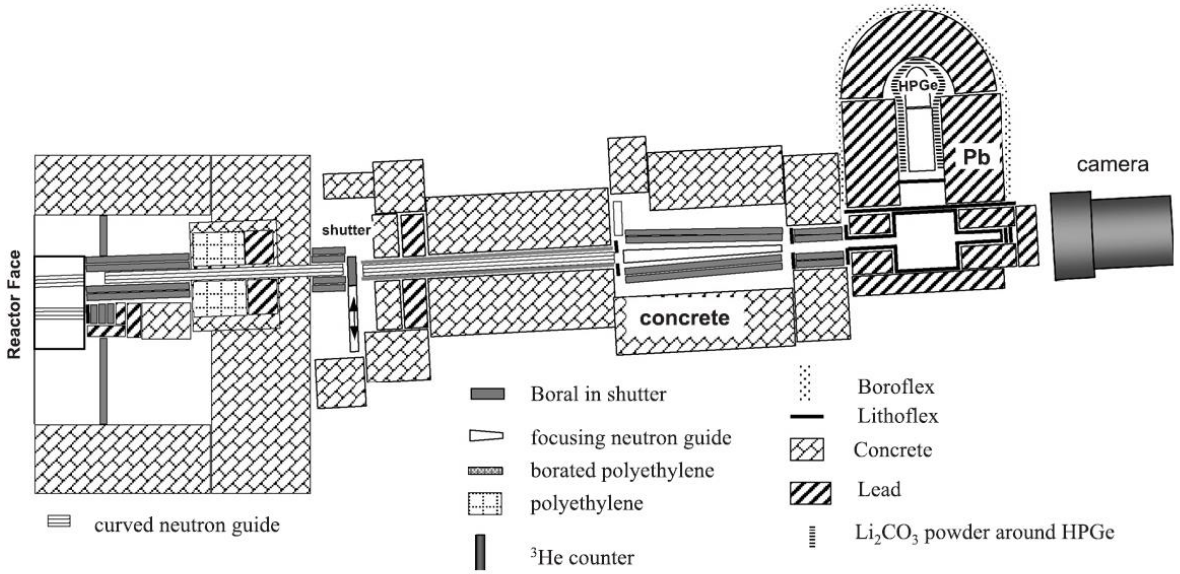


Figure 10.14, Shielding around TCNS Facility

10.4.2.f. 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 condensates 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. 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 non-condensable gas also raises 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, occurring. Each of the above-mentioned failures falls within the limits and capabilities previously evaluated.

The curved neutron guide, the converging neutron guide, and the prompt gamma activation analysis system are currently being used independent of the cold neutron cooling system.

10.5. CRYOGENIC IRRADIATION FACILITY

10.5.1. Description

A cryogenic irradiation facility has been developed for Beam Port 2. The facility inside the beam port consists of a cold head in a vacuum canister and condenser. The vacuum canister and condenser are connected to a 14-foot flexible liquid helium transfer line extending the length of the beam port and out into the reactor bay where the heat sink and vacuum pump are located.

10.5.2. Design and Specifications

BP2 runs tangential to the reactor core and ends at the outer edge of the graphite reflector (Figure 10.15). The condenser is placed as close to the reflector as possible, but never actually within the reactor, minimizing the chance that a catastrophic failure would damage the reactor. A void in the graphite reflector extends the effective source of neutrons into the reflector, providing a thermal neutron beam with minimal fast neutron and gamma-ray backgrounds. Beam port 2 will house a condenser attached to a copper cold head that is surrounded by an outer shell of steel to make up the irradiation canister (Figure 10.16). The canister itself is 13.375 in. long while the condenser and copper cold head are just 4.820 in. and are surrounded by vacuum. This canister will also be attached to a 14-foot-long helium flex line leading outside of the reactor structure to a compressor unit. The facility will be equipped with a gas transfer system in which two aluminum pipes will be attached via VCR connections to the ends of the canister.

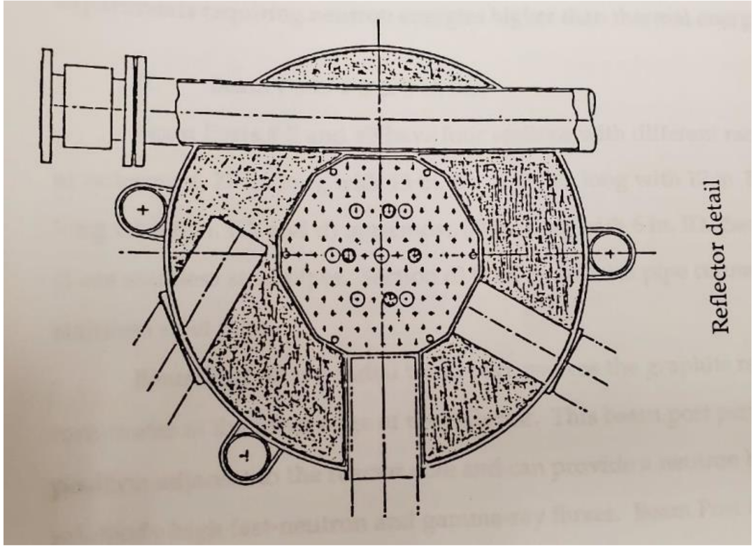


Figure 10.15, NETL Beam Port Configuration (BP2 is the first tangential port, located on the left).

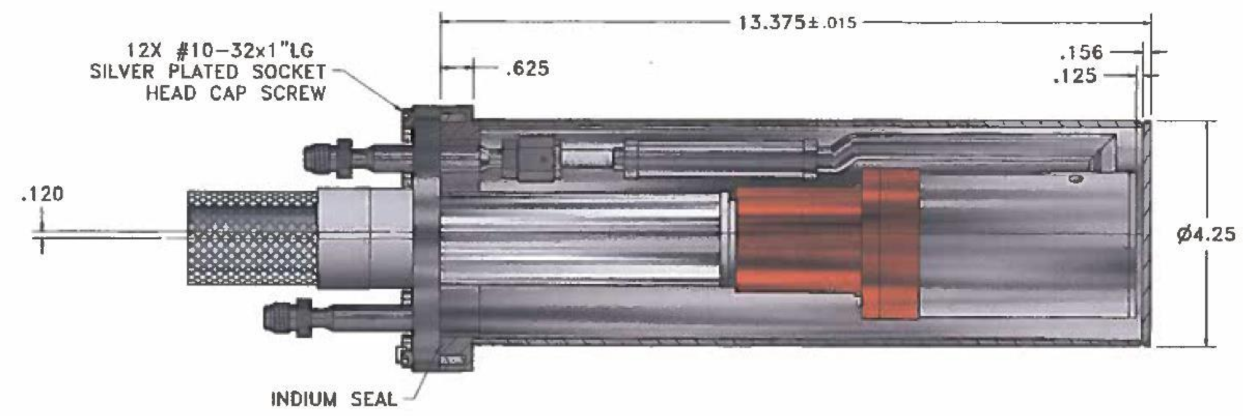


Figure 10.16, Cold Finger Design

Outside of the canister are three outlets, two to the condenser, and one to the volume between irradiation volume and canister exterior that will be held at vacuum. The outlet from the vacuum section will be isolated with a valve, while the other two outlets to the condenser will function as the gas transfer system. One line being an input /output, while the other will function as a safety release to an expansion tank just outside the reactor structure.

The design of the inside of the condenser is shown in Figure 10.17. The condenser was designed to house part of the copper to create the coldest surface possible for the gas to condense onto. The second part of the design was to incorporate a fin section protruding from the copper to maximize the surface area on the cold finger.

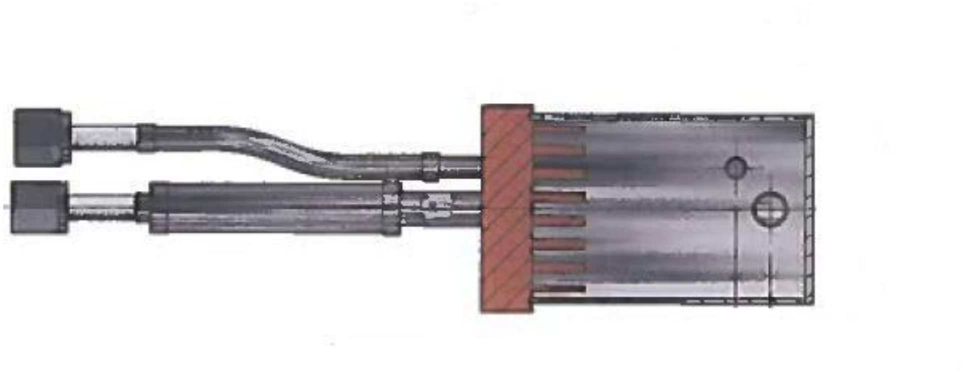


Figure 10.17, Cold Finger Fins

When the facility is in use, the canister will have been inserted the full 11-feet down the beam port and the support structure slid along the length of the pipes to its designated position. After this, the outside pipes will be connected to their respective manifolds for loading and unloading gases.

10.5.2.a. Reactivity

The experiment is outside the reflector void in Beam Port 2 and the cryogenic facility does not have the potential for significant reactivity effects. Calculations specific to an experiment irradiating a highly absorbent material will be required.

10.5.2.b. Radiological Assessment

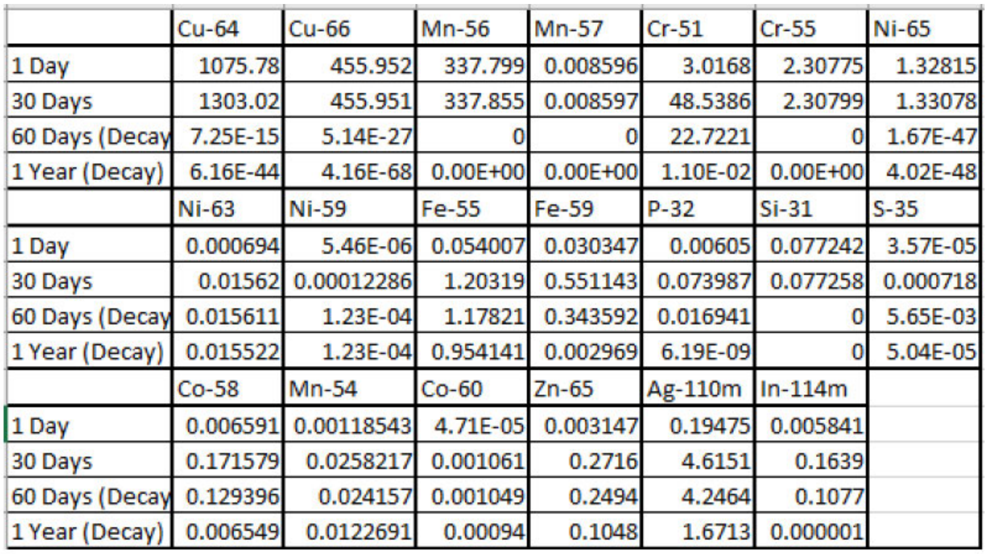
No experiment in the cryogenic irradiation facility shall be conducted that includes sample activity levels that exceed 365 DAC so that an inadvertent release will not exceed radiation worked does limits. A limit of 50 DAC in experiment planning is chosen to assure the DAC limits are met.

The canister is manufactured from SS-304 and 101 OFE copper. A SCALE code was used to produce a list of activation products, for example irradiation. The SCALE code operates under the typical reactor schedule at NETL, which assumes each day consists of an 8-hour irradiation period while the reactor is on, followed by a 16-hour decay period while the reactor is off. As reported by Cryomech, the condenser consists of SS-304 and 101 OFE copper, and the products in Table 10.8 below reflect the makeup of the condenser. Most isotopes rapidly decay away to negligible amounts following the end of irradiation, but a few stand out as concerns. The half-life of Fe-55 is 2.77 years, and as such an activity of around 1 Ci will still be present even a year later. Of primary concern is the presence of Cr-51, which after a month of decay will still be at almost 23 Ci. Should the condenser remain in the reactor permanently as is intended, this is unlikely to be an issue. Should a situation arise where the condenser needs to be removed before the Cr-51 has sufficient time to decay significantly (Cr-51 has a 27.7-day half-life), proper radiation protection measures need to be implemented to ensure a safe removal of the condenser without jeopardizing worker health.

Additionally, the condenser contains an indium gasket and silver-plated bolts. The half-life of the activated indium, In-113, is 72 seconds, and the half-life of the activated silver Ag-107 and Ag-

109, is 2.37 minutes and 24.60 s, respectively. Such short half-lives ensure that these activation products will decay away to nothing during the reactor's normal operation cycle of 8 hours of irradiation followed by a 16-hour decay period. However, both silver and indium possess troublesome metastable states, Ag-110m and In-114m, which have a 250 day and 50-day half-life, respectively. A brazing process was used to join the SS-304 with the 101 OFE copper, and the process utilized contained silver, copper, and zinc used in the braze wire. Of these additional materials, Zn-65 has one of the longest half-lives at 243.9 days and provides another potential activation hazard. The average weight of a braze wire varies, but seems to be around 1.5-2 ounces, approximately 50 grams, and 25% of the wire is made up of zinc. This is likely an overestimation of the amount of zinc present, but better to be safe than sorry. Using a weight of 12.5 grams and the Wise Uranium neutron activation calculator, Zn-65 has an activity of around 0.25 Ci after 30 days irradiation/30 days decay and an activity of around 0.1 Ci after 1 year of decay. The previous technique was also used for determining the potentially significant activity of the metastable states from Ag and In, using an estimated weight of 250g and 12.5g, respectively. The presence of Ag-110m provides another potentially significant activation hazard, furthering the need for proper radiation protection.

Table 10.4, Activation Products for 1-Day and 30-Day Irradiation and Decay of 60-Days and 1-Year (units are Curies)



	Cu-64	Cu-66	Mn-56	Mn-57	Cr-51	Cr-55	Ni-65
1 Day	1075.78	455.952	337.799	0.008596	3.0168	2.30775	1.32815
30 Days	1303.02	455.951	337.855	0.008597	48.5386	2.30799	1.33078
60 Days (Decay	7.25E-15	5.14E-27	0	0	22.7221	0	1.67E-47
1 Year (Decay)	6.16E-44	4.16E-68	0.00E+00	0.00E+00	1.10E-02	0.00E+00	4.02E-48
	Ni-63	Ni-59	Fe-55	Fe-59	P-32	Si-31	S-35
1 Day	0.000694	5.46E-06	0.054007	0.030347	0.00605	0.077242	3.57E-05
30 Days	0.01562	0.00012286	1.20319	0.551143	0.073987	0.077258	0.000718
60 Days (Decay	0.015611	1.23E-04	1.17821	0.343592	0.016941	0	5.65E-03
1 Year (Decay)	0.015522	1.23E-04	0.954141	0.002969	6.19E-09	0	5.04E-05
	Co-58	Mn-54	Co-60	Zn-65	Ag-110m	In-114m	
1 Day	0.006591	0.00118543	4.71E-05	0.003147	0.19475	0.005841	
30 Days	0.171579	0.0258217	0.001061	0.2716	4.6151	0.1639	
60 Days (Decay	0.129396	0.024157	0.001049	0.2494	4.2464	0.1077	
1 Year (Decay)	0.006549	0.0122691	0.00094	0.1048	1.6713	0.000001	

The Cryogenic Irradiation Facility is designed to operate normally at a steady state full reactor power of 950 kW. The materials used in the irradiation canister (copper and steel) do not pose concern from heating or radiation damage.

10.5.2.c. Instrumentation

The helium compressor is instrumented with a temperature and pressure controller. A pressure sensor on the VCR fitting emits an alarm if there is an abnormal pressure condition.

10.5.2.d. Physical Restraints, Shields, or Beam Catchers

Concrete blocks are used around the exit of the beam port to provide external shielding.

10.5.2.e. Operating Characteristics

Experiments were run externally to the reactor to test pressurization. The condenser was loaded with nitrogen to atmospheric pressure, cooled, and then slowly warmed. Figures 5 and 6 below show the temperature and pressure within the condenser over several hours with the cryogenic system first cooling to the minimum temperature, then heated to 70 K, and then turned off. The first test examined how the helium compressor and temperature controller operated together, with the second a temperature and pressure test.

The first scenario examined how the helium compressor and temperature controller operated together, with the second test applying temperature and pressure.

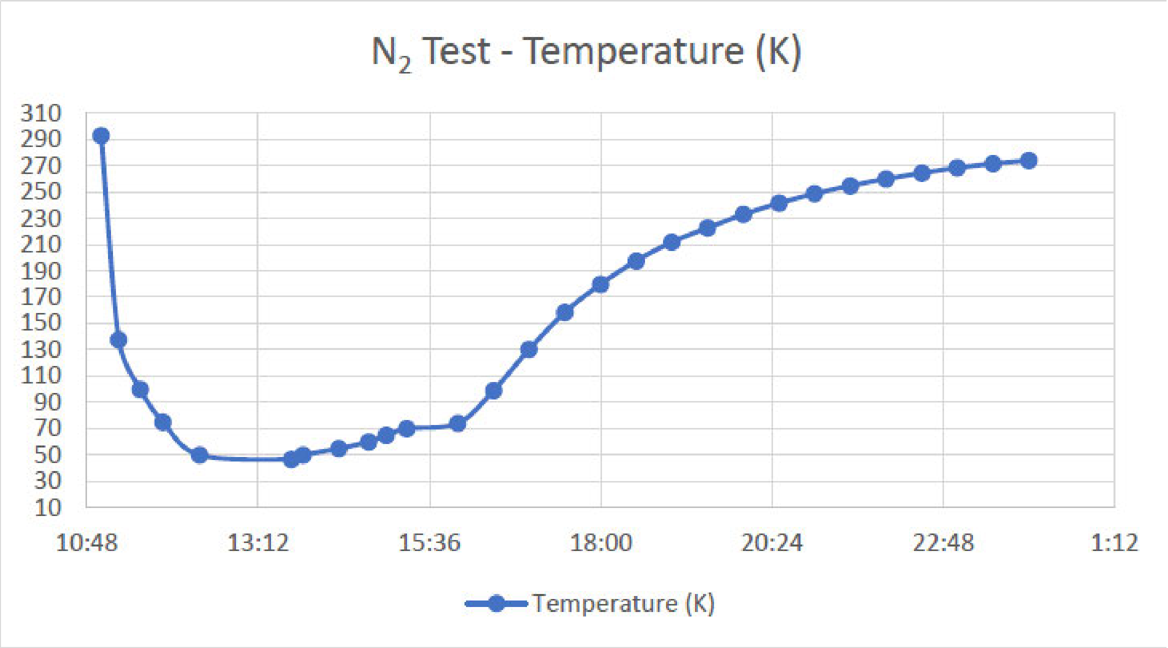


Figure 10.18, Nitrogen Freeze Test 2 Temperature

From the graphs we can see that the gas will cool down to a minimum temperature and pressure of ~47 K and ~15 torr, respectively, over the course of 1-2 hours. With the cryocooler turned off, the gas slowly warmed and rose in pressure over the next several hours, alleviating concerns of a potential gas flash.

The next scenario used stable xenon as the test material. The results are shown in Figures 10.19 and 10.20. Figure 10.20 shows the xenon pressure in the 1-4 torr range when properly sealed at any pressure.

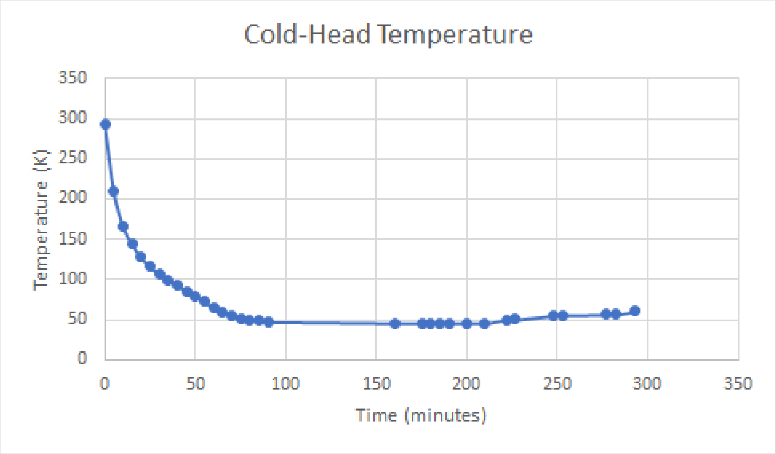


Figure 10.19, Scenario 3 Xenon Loading Temperature

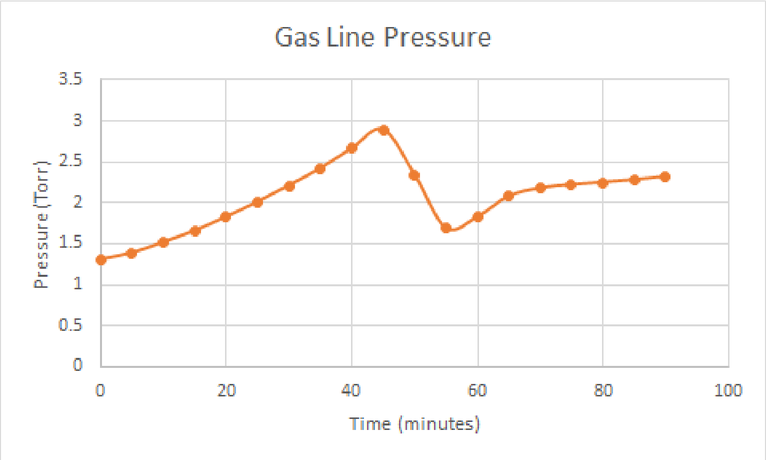


Figure 10.20, Scenario 3 Xenon Loading Gas Line Pressure

10.5.2.f. Safety Assessment

10.5.2.f.1. Thermal Hydraulic and Experiment Temperature/Pressure

The possibility of cryogenic material within the irradiation canister flashing from solid to gaseous form was evaluated. The pressure sensor used employs a thermal conductivity (TC) measurement and calibrated for nitrogen and is retained for safety. If the pressure readout on the TC gauge shows that the pressure has increased to nitrogen levels (~14 torr at 50 K), then it is likely an air leak has occurred, and air has infiltrated the system. The pressure is monitored continuously and instrumented with an alarm to alert the potential for freezing large volumes of air in the system. The alarm sends an audible notification to the phone of an operator within seconds should a leak occur. Due to the limitations of the TC gauge, a Piezo absolute pressure transducer will also be used to get an accurate readout of the Xe and Ar pressure within the system. The Piezo operates in tandem with the pressure gauge and serves as an alarm for the presence of air in the system.

An MCNP model of the reactor developed previously was utilized to assure gamma heating of the canister was used to determine that with coolant there would be no major changes in the temperature profile with cooling, and without coolant gamma heating would be unable to induce a gas flash.

10.5.2.f.2. Mechanical Stress

Stress testing of the pressure vessel was done using ANSYS to simulate the response to a gas flash. The gas line containing the helium coolant was designed specifically for liquid helium, and stress concerns for the helium are as such a non-issue. Three scenarios were analyzed and scaled using a safety factor (ratio of the strength of the material to the maximum stress on the part) from 0 to 15. A safety factor below 1 indicates a failure somewhere in the condenser. The condenser is made up of 304 SS (UNS S30400) and is 0.125” thick on the top and sides of the cylinder but has a mated copper face on the bottom side through a brazing process. The pressure vessel is in a sealed containment of stainless steel. Damage to the pressure vessel that would put pressure on surrounding structures or inhibit removal of the apparatus from the reactor is not likely if deformation occurs to the cold finger apparatus.

The first analysis (Figure 10.21 and 10.22) assumed a flash of the maximum volume (1-liter at STP) of Ar/Xe in the vessel, 3.091 liters at STP.

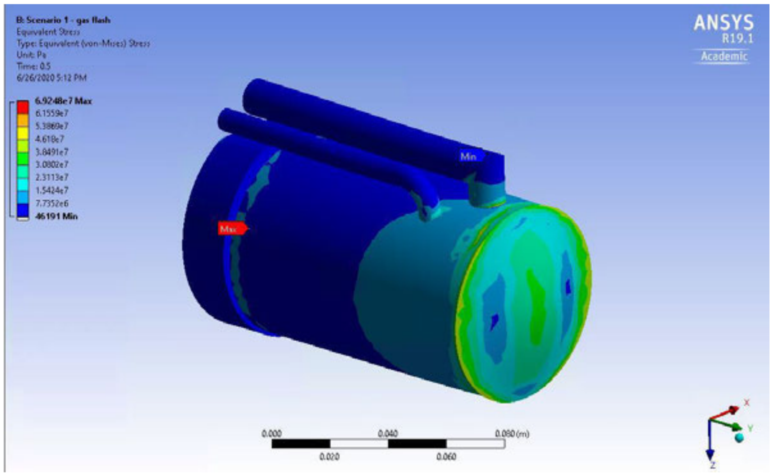


Figure 10.21, Equivalent Stress – Value 6.93e7 Pa<2.15e8 Pa Tensile Yield Strength – No Fracture

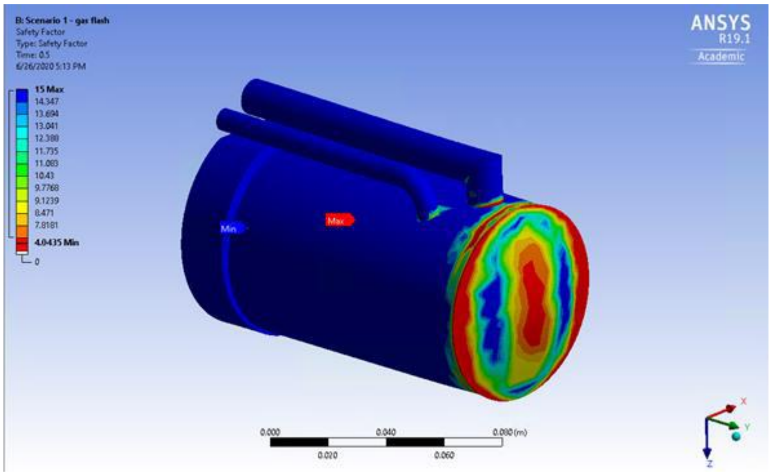


Figure 10.22, Safety Factor – Minimum value of 4.04 > 1 – No Failure of Condenser

The second analysis (Figures 10.23 and 10.24) assumed a crack in the gas line allowing outside air to leak into the system, resulting in 6-liters of gas at STP in the condenser (a volume based on the total volume of air needed to create the Argon pressure wave in both directions for the first scenario). Deformation results in the condenser but does not cause a fracture failure. This section of the equipment is contained in an outer steel container and any deformation should not cause the equipment to become secured in the beam port.

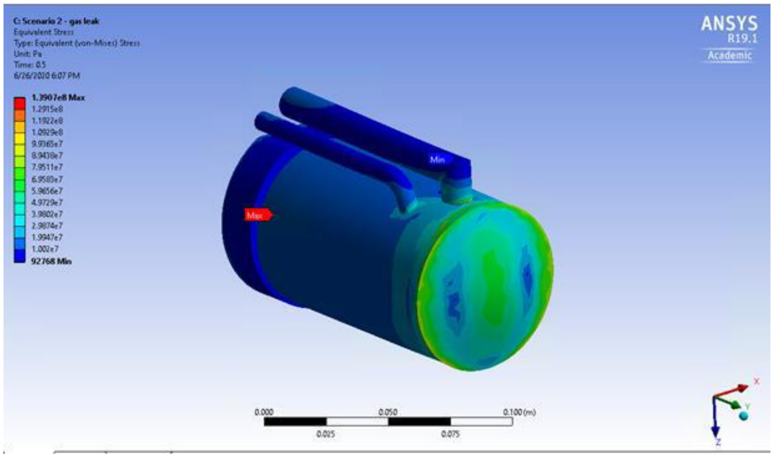


Figure 10.23, Equivalent Stress – Value of 1.39e8 Pa < 2.15e8 Pa
Tensile Yield Strength – No Fracture

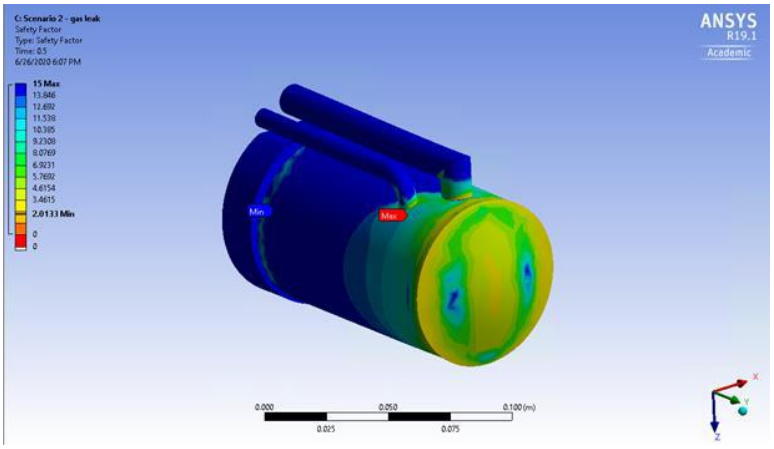


Figure 10.24, Safety Factor – Minimum Value of 2.01 > 1
– No Failure of Condenser

The third analysis (Figures 10.25 and 10.26) identified the failure point, or the volume that would be sufficient to induce a fracture in the condenser. Permanent deformation/fracture will occur at 4.64e6 Pa, or a total volume of 13.7 liters of gas at STP in the condenser.

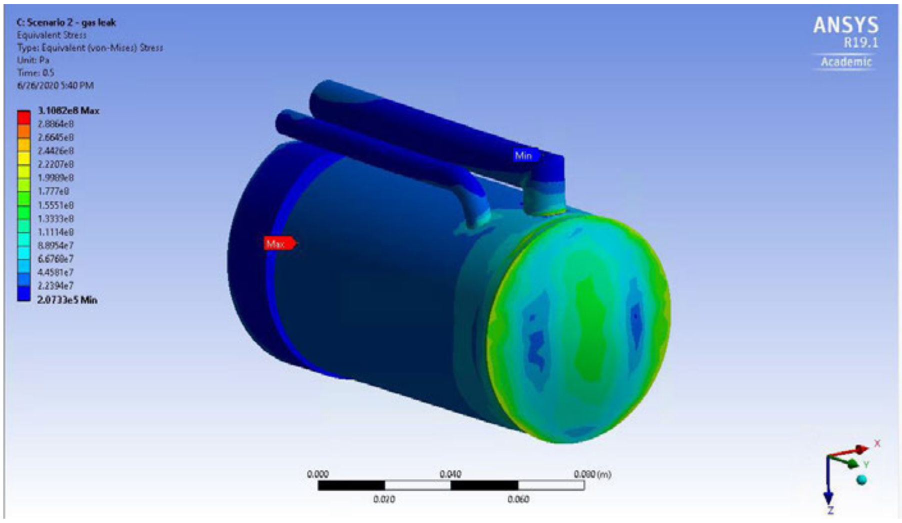


Figure 10.25, Equivalent Stress – Values over 2.15e8 are Fractures within SS304; the Larger Values (3e8) are within the Copper Region of the condenser

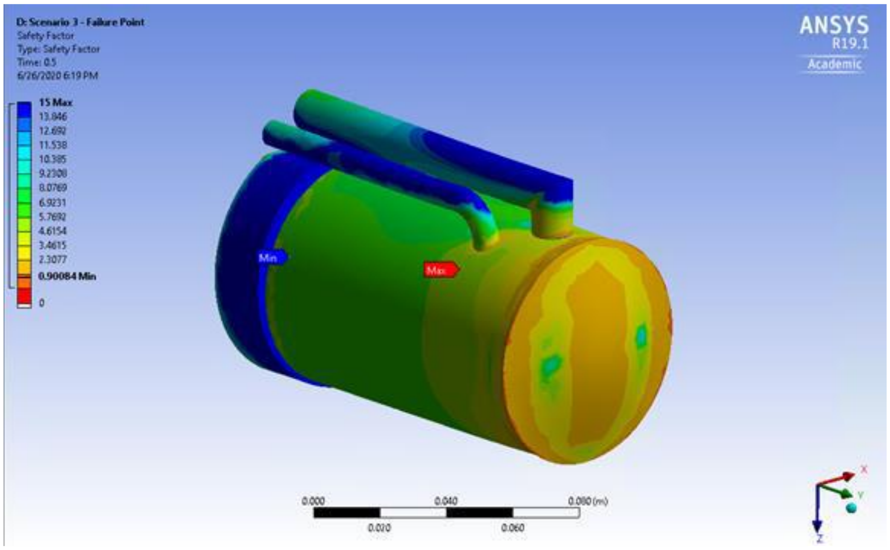


Figure 10.26, Safety Factor – Values Less than 1 Indicate Failure of Condenser End Cap

A permanent deformation and fracture in the condenser the volume in the canister would need to reach or exceed 13.7 liters at STP, an additional 10 liters of air at STP would need to leak into the system and for the system to undergo a rapid loss of coolant resulting in cryogenic material flashing and causing a pressure wave. Neither the loading nor the leakage is likely.

10.5.2.f.3. Material Evaluation

The potential for instantaneous release of 50 DAC was considered. The material is assumed to disperse into the reactor bay open-air volume of 4,120 m^3. A release of 50 DAC of Xe-127 is (50x1e-5 µCi/ml, 500-µCi/m3) is 14% of the allowable amount by the facility Technical Specifications and provides a very conservative margin. The DAC multiplied by the open-air volume of the reactor bay provides a limit of 2.06 Ci. More than seven complete experiments to vent within one year would be required to generate a value near the Technical Specifications Limit.

The effluent concentrations at the closest receptor site to NETL was evaluated. The release is assumed to be diluted by the 4, 120 m^3 reactor bay air volume. The concentration at the nearest receptor to NETL as calculated via CAP-88[50] reduced by (about) 10^4 to 10^5. The 50 DAC value diluted by a factor of 10^4 is well within the 10 CFR 20 Appendix B effluent concentration. However, maximum effluent concentrations will be evaluated for each operations request. For an accidental release to the public, a calculation is performed using CAP88 with a release of 50 DAC as the release rate/year at NETL. The resulting dose to an individual is insignificant, 1.53e-4 mrem effective dose equivalent.

Potential exposure to personnel from the activated cold head is evaluated. The cold head will produce different dose dependent on the time after irradiation it is removed, as illustrated in the

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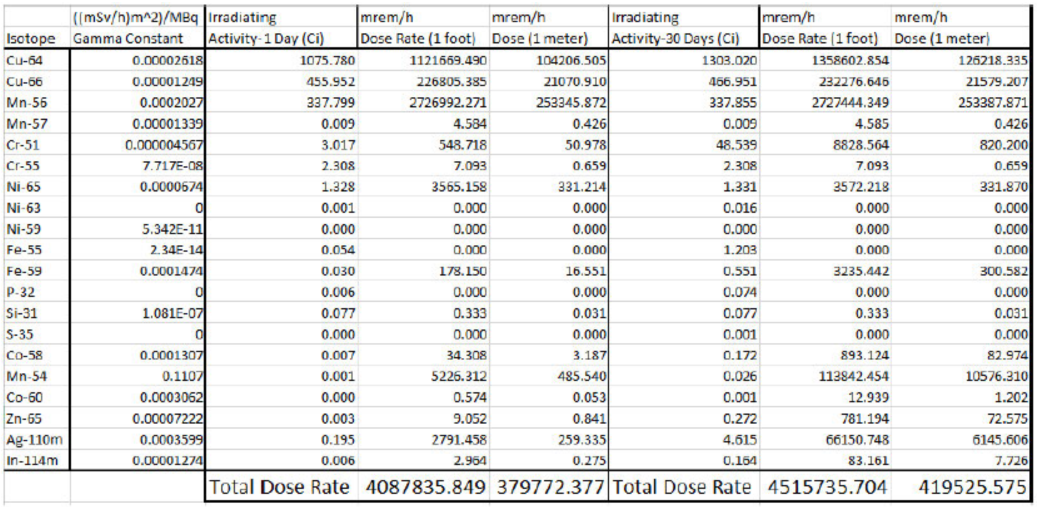
[50] Clean Air Act Assessment Package – 1988 (CAP-88), http://www.epa.gov/radiation/docs/cap88/cap88mf_guide.html

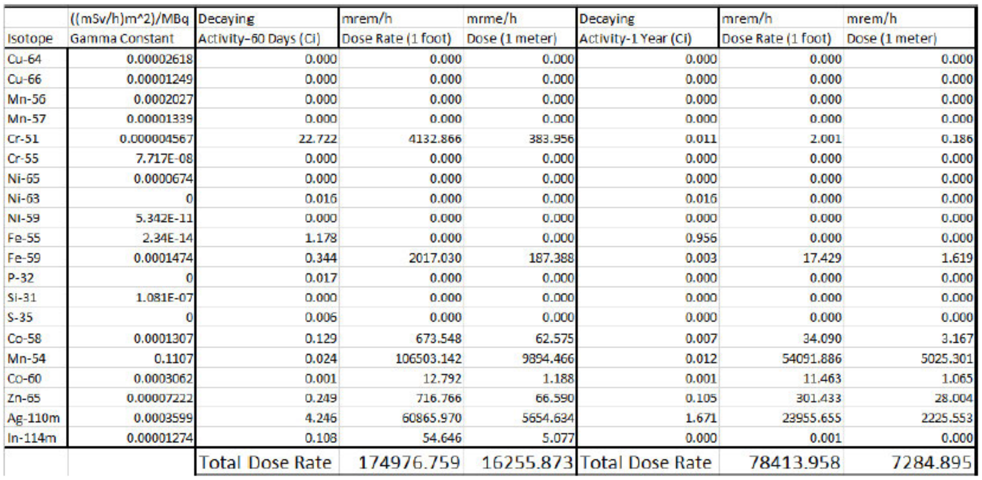
chart below. Gamma constants obtained from Table 10.5 of the Specific Gamma-Ray Dose Constants with current emission data from ORNL[2].

Table 10.5a, Effective dose, mrem/h, for Activated Condenser During Irradiation at 1-Foot and 1-Meter[51]



	((mSv/h)m^2)/MBq	Irradiating	mrem/h	mrem/h	Irradiating	mrem/h	mrem/h
Isotope	Gamma Constant	Activity-1 Day (Ci)	Dose Rate (1 foot)	Dose (1 meter)	Activity-30 Days (Ci)	Dose Rate (1 foot)	Dose (1 meter)
Cu-64	0.00002618	1075.780	1121669.490	104206.505	1303.020	1358602.854	126218.335
Cu-66	0.00001249	455.952	226805.385	21070.910	466.951	232276.646	21579.207
Mn-56	0.0002027	337.799	2726992.271	253345.872	337.855	2727444.349	253387.871
Mn-57	0.00001339	0.009	4.584	0.426	0.009	4.585	0.426
Cr-51	0.000004567	3.017	548.718	50.978	48.539	8828.564	820.200
Cr-55	7.717E-08	2.308	7.093	0.659	2.308	7.093	0.659
Ni-65	0.0000674	1.328	3565.158	331.214	1.331	3572.218	331.870
Ni-63	0	0.001	0.000	0.000	0.016	0.000	0.000
Ni-59	5.342E-11	0.000	0.000	0.000	0.000	0.000	0.000
Fe-55	2.34E-14	0.054	0.000	0.000	1.203	0.000	0.000
Fe-59	0.0001474	0.030	178.150	16.551	0.551	3235.442	300.582
P-32	0	0.006	0.000	0.000	0.074	0.000	0.000
Si-31	1.081E-07	0.077	0.333	0.031	0.077	0.333	0.031
S-35	0	0.000	0.000	0.000	0.001	0.000	0.000
Co-58	0.0001307	0.007	34.308	3.187	0.172	893.124	82.974
Mn-54	0.1107	0.001	5226.312	485.540	0.026	113842.454	10576.310
Co-60	0.0003062	0.000	0.574	0.053	0.001	12.939	1.202
Zn-65	0.00007222	0.003	9.052	0.841	0.272	781.194	72.575
Ag-110m	0.0003599	0.195	2791.458	259.335	4.615	66150.748	6145.606
In-114m	0.00001274	0.006	2.964	0.275	0.164	83.161	7.726
		Total Dose Rate	4087835.849	379772.377	Total Dose Rate	4515735.704	419525.575

Table 10.5b, Effective dose in mrem/h for Activated Condenser Following Decay at 1-Foot and 1-Meter



	((mSv/h)m^2)/MBq	Decaying	mrem/h	mrme/h	Decaying	mrem/h	mrem/h
Isotope	Gamma Constant	Activity-60 Days (Ci)	Dose Rate (1 foot)	Dose (1 meter)	Activity-1 Year (Ci)	Dose Rate (1 foot)	Dose (1 meter)
Cu-64	0.00002618	0.000	0.000	0.000	0.000	0.000	0.000
Cu-66	0.00001249	0.000	0.000	0.000	0.000	0.000	0.000
Mn-56	0.0002027	0.000	0.000	0.000	0.000	0.000	0.000
Mn-57	0.00001339	0.000	0.000	0.000	0.000	0.000	0.000
Cr-51	0.000004567	22.722	4132.866	383.956	0.011	2.001	0.186
Cr-55	7.717E-08	0.000	0.000	0.000	0.000	0.000	0.000
Ni-65	0.0000674	0.000	0.000	0.000	0.000	0.000	0.000
Ni-63	0	0.016	0.000	0.000	0.016	0.000	0.000
Ni-59	5.342E-11	0.000	0.000	0.000	0.000	0.000	0.000
Fe-55	2.34E-14	1.178	0.000	0.000	0.956	0.000	0.000
Fe-59	0.0001474	0.344	2017.030	187.388	0.003	17.429	1.619
P-32	0	0.017	0.000	0.000	0.000	0.000	0.000
Si-31	1.081E-07	0.000	0.000	0.000	0.000	0.000	0.000
S-35	0	0.006	0.000	0.000	0.000	0.000	0.000
Co-58	0.0001307	0.129	673.548	62.575	0.007	34.090	3.167
Mn-54	0.1107	0.024	106503.142	9894.466	0.012	54091.886	5025.301
Co-60	0.0003062	0.001	12.792	1.188	0.001	11.463	1.065
Zn-65	0.00007222	0.249	716.766	66.590	0.105	301.433	28.004
Ag-110m	0.0003599	4.246	60865.970	5654.634	1.671	23955.655	2225.553
In-114m	0.00001274	0.108	54.646	5.077	0.000	0.001	0.000
		Total Dose Rate	174976.759	16255.873	Total Dose Rate	78413.958	7284.895

Effective dose to personnel can be significant if condenser removed from reactor soon after irradiation. Shielding and distance between personnel and condenser will be necessary. Should an accident occur, it will be possible to partially withdraw the canister away from the reflector and

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[51] Times consistent with Table 10.4

allow it to decay, shielded by multiple sections of borated poly along the pipe path leading to the condenser as well as exterior shielding in the beam port. Further neutron activation minimized while the canister decays to safe levels for removal.

10.5.2.f.4. Material Hazards

A. Trace Element Impurities Which May Represent a Significant Radiological Hazard

No trace element impurities will represent a significant radiological hazard in this experiment. Each gas container is verified by the supplier for its elemental composition.
High Cross-Section Elements

The location of the cold head in BP2 is close enough to the core to be a minor reactivity concern if high cross-section elements are in the sample. Calculations would be performed to assure the effect meets Technical Specifications limits. The reactivity worth of experiments is measured during reactor start-up.

B. Flammable, Volatile, or Liquid Materials

Liquid helium fed into the cold head creates a surface to cool target gases down to a liquid and/or solid state. The liquid helium will always be separated from the condensing chamber. Target materials may be one or more liquids in the irradiation canister. If any new materials are part of an experiment, a new experiment authorization based on safety analysis report for the new material is required.

C. Explosive Chemicals

There are no explosive chemicals in this experiment. If explosive materials are part of a proposed experiment, a new or revised experiment authorization for the new material is required based on a new safety analysis.

D. Corrosive Chemicals

No corrosive chemicals are used in this experiment. If explosive materials are part of a proposed experiment, a new or revised experiment authorization for the new material is required based on a new safety analysis.

E. Radiation Sensitive Materials Which When Exposed to Radiation Exhibit Degradation of Mechanical Properties, Decomposition, Chemical Changes, or Gas Evolution

Materials evaluated within the irradiation canister become activated to significant levels of radioactivity when exposed to neutron flux at the level in the core. These material isotopes include Copper-64 and 66 found in the cold head of the canister, as well as Mn-56, found in the steel. The activity levels will vary based upon the amount of time irradiated and will have to be evaluated and monitored consistently to make sure the correct precautions are taken to minimize the dose to workers.

If new materials are part of a proposed experiment, a new or revised experiment authorization for the new material is required based on a new safety analysis.

F. Toxic Compounds

Toxic compounds are not used in this experiment. If toxic materials are part of an experiment, a new or revised experiment authorization for the new material is required based on a new safety analysis.

G. Cryogenic Liquids

This experiment will include the use of Argon, Xenon, and Helium, three commonly used industrial gasses that can be operated in the liquid state at cryogenic temperatures. For this experiment liquid helium will be used as a coolant while the Argon and Xenon represent a target gas to be cooled to a liquid and even solid state. While these are the three main gases that are intended to be used, other gases may also be proposed as part of an experiment, a new or revised experiment authorization for the new material is required based on a new safety analysis.

10.6. NON-REACTOR EXPERIMENT FACILITIES

The NETL maintains several facilities related to nuclear radiation and detection. These facilities are utilized for teaching, research, and service work.

10.6.1. Neutron Generator Room

The NETL houses a neutron generator room with 3-foot-thick concrete walls, floor, and ceiling. Experiments with a neutron generator, neutron sources, calibrations of radiation detectors, and hot cell manipulations of radioisotopes are performed in the room.

The Thermo Scientific MP 320 D-T neutron generator Figure 10.27) is a compact neutron generator designed for portability. The MP 320 has a flux of 1 x 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 10.27, Thermo MP 320 Neutron Generator at NETL

Hot cells to support medical isotope research have been installed in the neutron generator room.

10.6.2. Laboratories

10.6.2.a. 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.

10.6.2.b. 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.

10.6.2.c. 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.

10.6.2.d. 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.

10.6.2.e. 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.

10.7. 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 10.6.

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 10.6, 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.

TOPIC	**DESCRIPTION**
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 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 gases upon being irradiated. Also address possible degradation of sample containment during irradiation.

TOPIC	DESCRIPTION
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 the activity of 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. 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 diverse types of facility radiation protection programs for monitoring and controlling 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.a. Airborne Radiation Sources

During normal operation of the NETL reactor, airborne radioactivity is almost exclusively Ar-41.

11.1.1.a.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 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 pool water 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.a.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 (3.0x10^{-4} µCi cm^{-3}) by the volume release rate of the stack (3.9x10^{6} cm^{3} s^{-1}). Thus Q = 1170 µCi s^{-1} and X(0,0,0) = 9.1 µCi m^{-3} = 9.1x10^{-6} µCi cm^{-3}. While this concentration is about 900 times the effluent concentration limit of $1x10^{-8}$ 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 $1.6x10^{-3}$ µCi m^{-3} = $1.6x10^{-9}$ 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 conducted 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.b. 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 representative list includes up to 0.1 mCi each of Mn-54, Co-57, Co-60, Zn-65, Sr-85, Sr-90, Y-88, Cd-109, Sn-113, Ba-133, Cs-137, Ce-139, Ce-141, Pm-147, Eu-152, Eu-154, Eu-155, Hg-203, Bi-207, Po-209, Np-237, Am-241, Cm-244 and 0.5 mCi of Pb-210. In addition, neutron activation of 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.b.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 $3.3x10^{22}$ 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. 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.b.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 considering 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.c. 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.

Table 11.1, Representative Solid Radioactive Sources

Source Description	Radionuclide(s)	Nominal Activity (Ci)	Physical Characteristics	Wt% Uranium	Approximate Original Grams	
					U-235	Total U
114 TRIGA Fuel Elements	Enriched Uranium		In Core	8.5	[redacted]	[redacted]
17 TRIGA Fuel Elements	Enriched Uranium		Storage (New)	8.5	[redacted]	[redacted]
69 TRIGA Fuel Elements	Enriched Uranium		Storage (Used)	8.5	[redacted]	[redacted]
Pu Source	Pu-238	0.1	Sealed Source			
2PuBe Sources	Pu-238	45	Sealed Sources			
4 PuBe Sources	Pu-239	9	Sealed Sources			
Calibration Source	Cs-137	2.5	Sealed Source			
Calibration Source	Am-241	1	Sealed Source			
Calibration Sources	Co-60	0.5	Sealed Sources			
Small Instrument Calibration and Check Sources	Cf-252, Cm-244, Eu-152, Ra-226, Na-22	≤0.005 each	Sealed Sources			
Small Instrument Calibration and Check Sources	C-14, Cl-36, Fe-55, Fe-59, Co-57, Mn-54, Ni-63, Zn-65, Sr-90, Tc-99, Cd-109, Sn-113, Sb-125, I-129, Pm-147, Eu-155, Tl-204, Bi-207, Pb-210, Po-210, Bi-210, Ba-133, Th-230, Np-237	≤0.0001 each	Sealed Sources			
Irradiated Items and Materials	Mixed Activation Products	0.0001 to 1.0	Unsealed items irradiated in pneumatic transfer system and other in-core irradiation facilities			
Solid Waste	Mixed Activation Products	0.001	Annual solid waste volume typically <2 ft^3			

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.c.1. Shielding Logic

Although not a solid source of radioactivity itself, shielding is important 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.a. Management and Administration

11.1.2.a.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 (CSE) manages eight departments (including the Walker Department of Mechanical Engineering) with individual degree programs. The Nuclear Engineering Teaching Laboratory (NETL) is one of several education and research functions within the CSE and is administratively located within the Walker Department of Mechanical Engineering.

A. 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.

B. 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 Chair of the Mechanical Engineering Department.

C. 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.a.2. <u>Level 2 Personnel</u>

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.

A. Chair, Walker Department of Mechanical Engineering

The Chair reports to the Dean of the Cockrell School of Engineering. The Department manages 8 areas of study, including Nuclear and Radiation Engineering.

B. 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.

C. Associate Director, Nuclear Engineering Teaching Laboratory

The Associate Director is responsible for the 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.

D. Safety Oversight

Safety oversight is provided for radiation protection and facility safety functions. The 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 the Reactor Oversight Committee.

E. 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.

F. 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. The 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 ensures independence of the Health Physicist through the Radiation Safety Officer while maintaining close interaction with NETL line management.

G. 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.a.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.

A. 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.

B. 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 Health Physicist 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.a.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.a.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.b. Health Physics Procedures and Document Control

Operation of the radiation protection program functions 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, and 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.c. 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.d. 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.e. 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 (or designee) 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 on the total effective dose equivalent being equal to 1 rem (10 mSv),

2. The annual dose limits to the general public being equal to 0.05 rem (0.5 mSv)

These dose limits may only be exceeded in special circumstances and by written permission of the NETL director who in consultation with the health physicist will assign a new local dose limit.

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 performed: 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.a. 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.b. Radiation Monitoring Equipment

The 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	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
Ludlum	23	N/A	Personnel dosimetry
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
HIDEX	300SL	N/A	Liquid Scintillation Counter
Canberra	CAM100G	N/A	Ar-41 CAM
Ludlum	333-2	N/A	Particulate CAM
Ludlum	375	0.1–1000 mR/hr	Area Radiation Monitor

11.1.4.c. 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 distinct 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.a. 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 to determine the appropriate control measures for new configurations, as necessary.

11.1.5.b. 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.c. 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 to have 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.d. 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.e. 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 record keeping 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

Year	Numbers of persons in annual-dose categories			
	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.e.1. 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 diverse 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 conducted 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 of 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 managed, 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 innovative 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.a. 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.b. 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.c. 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 25 cubic feet. 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 of 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 designated 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.d. 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, the generation of mixed waste cannot be completely avoided. 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 in storage 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.e. 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 reactor operators subject to 10CFR50.55 requirements.

12.1.1. Structure

12.1.1.a. University Administration

Figure 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 Figure 12.1[52]. These responsibilities include safeguarding the public and staff from undue radiation exposure 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.a.1. NETL Facility Administration

The facility administrative structure is shown in Figure 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 people 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.

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[52] "Standard for Administrative Controls" ANSI/ANS - 15.18 1979

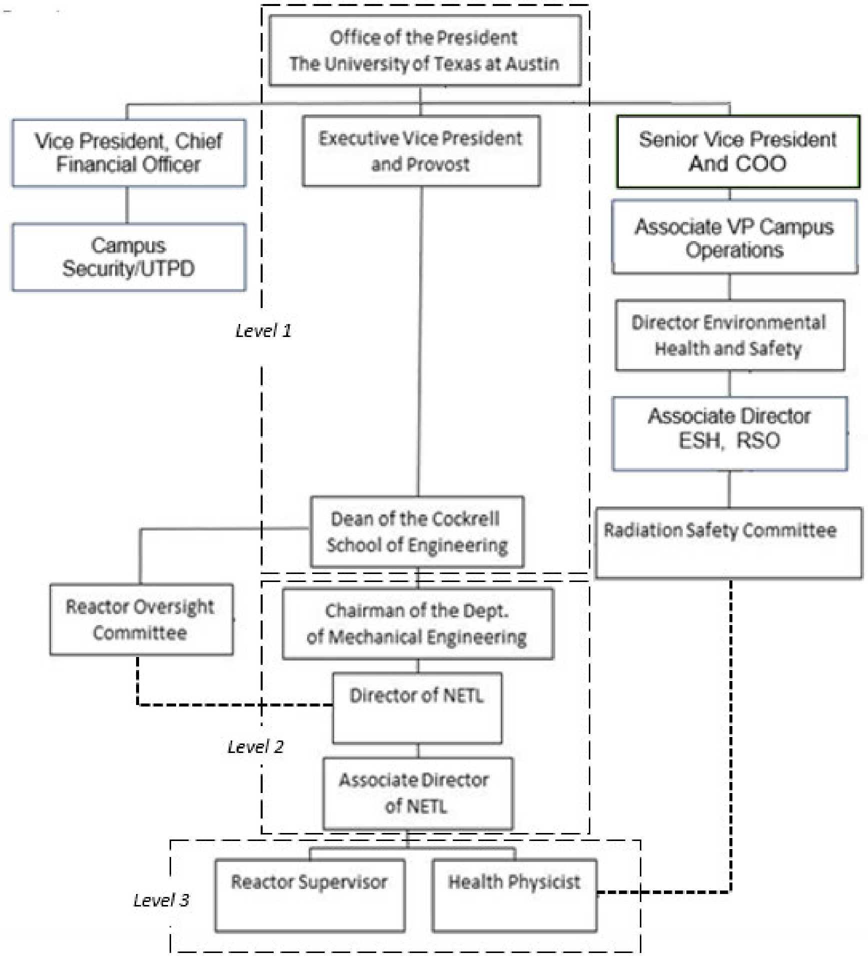

Figure 12.1, University Administration

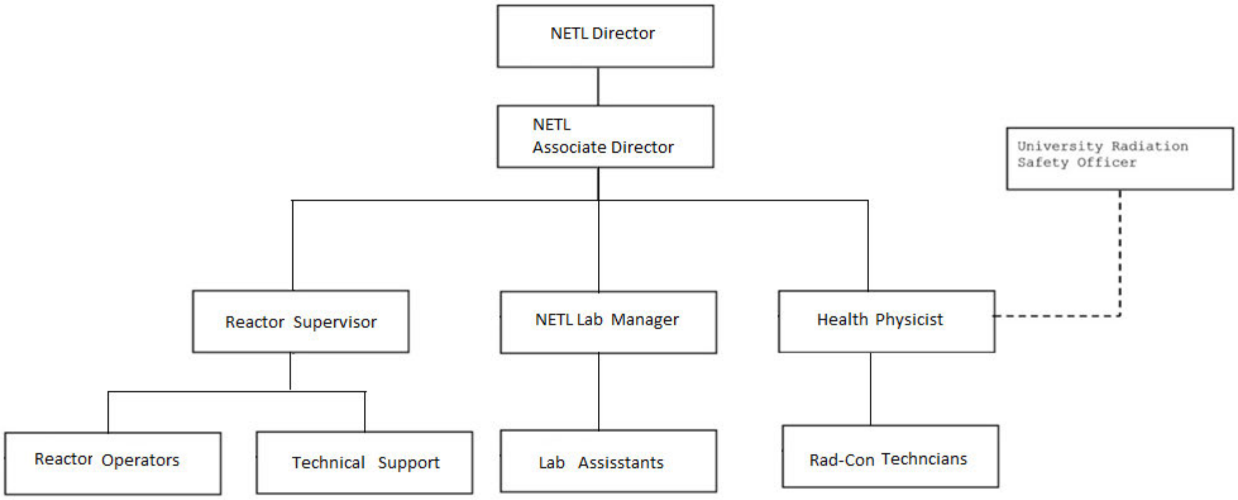

Figure 12.2, NETL Facility Administration

12.1.2. Responsibility

12.1.2.a. 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 Chair of the Mechanical Engineering Department.

12.1.2.b. Vice President and Chief Financial Officer

The Vice President and Chief Financial Officer is the financial steward of the university's resources, the chief executive for the Financial and Administrative Services (FAS) portfolio, and manager for campus security and the University Police Department.

12.1.2.c. Senior Vice President and Chief Operating Officer

University operations activities are administered through the Office of the Senior Vice President and Chief Operating Officer. This office is responsible for multiple operational functions of the University including Environment Health and Safety programs, university support programs, human resources, campus real estate, and campus planning and facilities management.

12.1.2.d. 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.e. 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 conducting research utilizing the reactor and other specialized equipment at the Nuclear Engineering Teaching Laboratory.

12.1.2.f. 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.g. 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 Radiological Health Manual.

12.1.2.h. 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 source of ionizing radiation at The University of Texas at Austin. The Committee will meet at least once each regular semester on a called basis, or as required, to formally approve applications to use radioactive material. Four members shall constitute a quorum. The Committee shall be consulted by the Office of Environmental Health and Safety concerning any unusual or exceptional action that affects the administration of the Radiation Safety Program.

12.1.2.i. Reactor Supervisor

The Reactor Supervisor is responsible for scheduling and supervising daily operations and maintenance The Reactor Supervisor is responsible for facility operating and maintenance procedures and records. The Reactor Supervisor is responsible for requalification program training. The Reactor Supervisor is responsible for implementing the emergency plan and the security plan.

12.1.2.j. 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.

The 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.k. Laboratory Manager

Laboratory operations and research support is provided by a designated Laboratory Manager. The function is typically combined with the Health Physicist position.

12.1.2.l. Reactor Operators

Reactor operators (and senior reactor operators) are licensed by the USNRC to operate the UT TRIGA II nuclear research reactor. University faculty, staff, and/or students may be licensed as reactor operators.

12.1.2.m. Technical Support

Staff positions supporting various aspects of facility operations are assigned as required.

12.1.2.n. 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.o. 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[53].

Whenever the reactor is not secured, the reactor shall be under the direction of a (USNRC licensed) Senior Operator who is designated as SSRO. The SSRO may be on call if capable of arriving at the facility within thirty minutes and being aware of reactor operations. The SSRO 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.

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[53] "Selection and Training of Personnel for Research Reactors", ANSI/ANS -15.4 - 1970 (N380)

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 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. The 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.a. Qualifications

Personnel associated with the research reactor facility[54] 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.

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[54] ANS/ANSI-15.4, op. cit.

12.1.4.b. 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 specific 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.b.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.

12.1.4.b.2. Associate Director

A combination of academic training and nuclear experience will fulfill the qualifications for the individual identified as the associate 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.b.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.b.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 a similar field of study and some experience is preferred. Certification is not a qualification, but working towards certification should be considered a requirement.

12.1.4.b.5. Laboratory Manager

Laboratory operations and research support is provided by a designated Laboratory Manager. The function is typically combined with the Health Physicist position.

12.1.4.b.6. Reactor Operators

Reactor operators (and senior reactor operators) are licensed by the USNRC to operate the UT TRIGA II nuclear research reactor. Training and requalification requirements are indicated below.

12.1.4.b.7. 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.

12.1.4.b.8. Radiological Controls Technicians

Radiological Controls Technicians training is provided in the Radiation Protection Program.

12.1.4.b.9. 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 Environmental Health and 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[55] [56].

Safety programs are operated as a function of the Associate Vice President for Campus Safety and include a radiation safety organization as presented in Figure 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. The person responsible for radiation protection at the reactor facility has 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

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[55] "Radiological Control at Research Reactor Facilities", ANSI/ANS-15.11 1977(N628)

[56] "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents", ANSI/ANS - 15.12 1977(N647)

The review and audit process are 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 radiation 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.

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 Oversight 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, which 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 Oversight 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 and other staff as appropriate. 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[57], 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.

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 shut down, 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:
    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 follow up reports
f. The report and any follow-up 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 the 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.

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[57] ANSI/ANS 15.6, op. cit.

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 include, 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 in effluents. 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 (Figure 12.1) involving Level 1 (Office of the President, Executive Vice President and Provost, Dean of the Cockerell School of Engineering) or Level 2 (Chairman of the Dept. of Mechanical Engineering, Director of NETL, or Associate Director of NETL).
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[58]. 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 regulations.

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[58] "Records and Reports for Research Reactors", ANSI/ANS - 15.3-1974 (N399).

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 post 9/11 security posture changes. The Plan and portions of the procedures are classified as Safeguards Information. Procedures reviewed and approved by the Reactor Oversight Committee are established to manage implementation of emergency response.

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 and follows the guidelines of Reg. Guide 2.5[59] [60]. 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.

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.

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[59] "Quality Assurance Requirements for Research Reactors", Nuclear Regulatory Guide 2.5 (77/05).

[60] "Quality Assurance Program Requirements for Research Reactors," ANSI/ANS - 15.8 - 1976 (N402).

13. ACCIDENT ANALYSES

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 for credible accidents. Reference values for physical properties and values used in analysis are provided in Section 13.1. An overview of accident scenarios is provided in Section 13.2, followed by detailed analyses.

13.1. INTRODUCTION

13.1.1. Computer Codes

Three principle computer codes (SCALE, MCNP and TRACE) were used to perform simulations supporting accident analysis.

SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design that was 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. The initial version of SCALE was distributed in 1980 to support analysis for NRC licensing activities. SCALE integrates various codes in sequences for specific applications. The T-6 depletion sequence was used to generate radioisotope inventory for the NETL TRIGA fuel in the maximum hypothetical accident: complete loss of fuel cladding following steady-state operation. The sequence uses KENO (a Monte Carlo code), to calculate neutron transport in the core, coupled to ORIGEN (Oak Ridge Isotope Generation code), to calculate isotopic inventory of the fuel.

The Monte Carlo N-Particle Transport code (MCNP) is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies and is developed by Los Alamos National Laboratory. MCNP was used to calculate reactor physics parameters, reactivity coefficients, and core power distribution.

A fuel temperature coefficient of reactivity was developed from MCNP criticality calculations at varying fuel temperatures. The data derived from MCNP above about 200°C agreed well with the data described in GA-7882[61], for water-reflected stainless-steel clad 8.5% enriched TRIGA fuel temperature coefficient, although MCNP-derived values diverge below 200°C. MCNP does not model optical scattering properties, and recent changes in scattering data for ZrH cross sections at the lower temperatures suggests the cross sections are somewhat uncertain. Since the temperatures of interest for thermal hydraulic calculations occur at relatively high temperatures where the MCNP-derived values agree with the data in GA-7882 and the changes in the MCNP-derived

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[61] GA-7882, “Kinetic Behavior of TRIGA Reactors”, General Atomics (1967)

values below 200°C exhibit suspect large changes with temperature, data from GA-7882 (Figure 13.1) was used in this chapter.

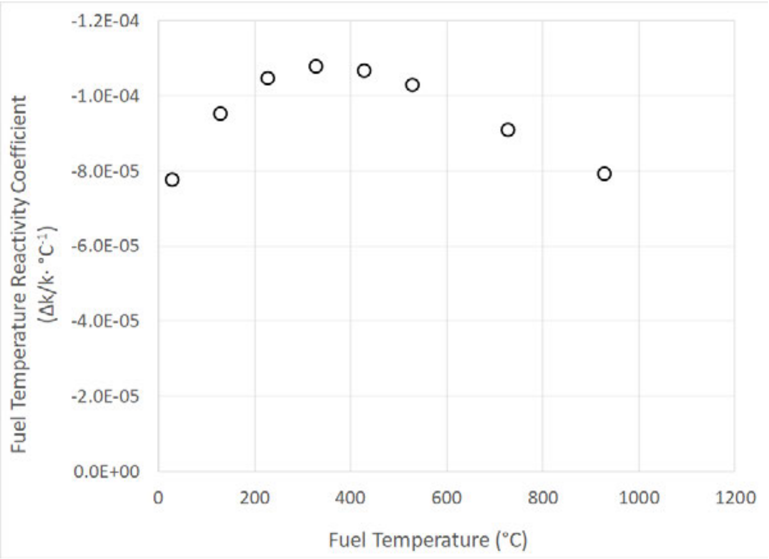

Figure 13.1, Fuel Temperature Coefficient of Reactivity for TRIGA Fuel (from GA-7882)

The TRAC/RELAP Advanced Computational Engine code (TRACE) is a thermal-hydraulics code designed to consolidate and extend the capabilities of NRC's 3 legacy safety codes - TRAC-P, TRAC-B and RELAP. TRAC analyzes large/small break LOCAs and system transients in both pressurized- and boiling-water reactors (PWRs and BWRs). TRAC models thermal hydraulic phenomena in both one-dimensional (1-D) and three-dimensional (3-D) space. TRAC is the NRC's flagship thermal-hydraulics analysis tool.

13.1.2. Accident Scenarios

Three accident scenarios were identified in the initial licensing of the University of Texas TRIGA reactor in 1992:

1. maximum hypothetical accident (fuel element failure in air),
2. insertion of excess reactivity, and
3. loss of coolant.

The current accident analysis substantially reprises the original, with updates to the methodology based on current standards.

NUREG/CR-2387[62] 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.

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[62] NUREG/CR-2387, "Credible Accidents for TRIGA and TRIGA Fueled Reactors," prepared by S. C. Hawley and R.L. Kathren, Pacific Northwest National Laboratory PNL-4208 (1982).

- 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 content as well as a standard review plan for the spectrum of accidents identified including:

- Maximum Hypothetical Accident (addressed in Section 13.2)

  The consequences of a fuel handling accident with a complete loss of cladding in air are analyzed. Under extremely conservative assumptions the consequences of a fuel handling accident are acceptable.

- Insertion of Excess Reactivity (addressed in Section 13.3)

  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.

- Pulsing (addressed in Section 13.3.1)

  Pulsed insertion of excess reactivity is considered for two initial conditions: (1) low power or shutdown and (2) pulsed reactivity insertions from power. The TRIGA instrument and controls includes an interlock to prevent pulsing above a setpoint power level. Analysis was performed to assure temperature at the setpoint does not exceed fuel temperature limits during pulsing, and the margin to the power level where the temperature limits are challenged.

  An administrative limit for experiments of $3.00 assures that the consequence of rapid removal of all experiments from the core is bounded by the analysis.

- Continuous Reactivity Addition (addressed in Section 13.3.2)

  Reactor response to a continuous reactivity addition from high-power operations was analyzed. The maximum fuel temperature does not challenge the safety limit.

- Beam Port Flooding (Section 13.3.3)

  Beam port flooding has a potential impact on core reactivity. The reactivity addition associated with beam port flooding could be essentially a pulse or a continuous reactivity addition to the maximum reactivity of the flooded beam port. Authorization for beam port utilization includes MCNP analysis to calculate the maximum reactivity possible from beam port flooding. The magnitude of reactivity is much less than the reactivity added in the continuous reactivity accident bounded by the analysis.

- Loss of Coolant (Section 13.4)

  A loss of coolant accident is analyzed to determine doses from scattered radiation from the uncovered core. The radiation levels from the uncovered core are high, but manageable. The maximum fuel temperature of the air-cooled element is analyzed to determine that the fuel temperature safety limit is not challenged.

- Loss of Coolant Flow (addressed in Section 13.5)

  The system response to loss of coolant flow is considered and is acceptable.

- Mishandling or Malfunction of Fuel (addressed in Section 13.6)

  The maximum hypothetical accident bounds the consequences of fuel malfunction, although transport of fission products released in the pool would affect the release. The onsite doses are adequately controlled through the Radiation Protection Program, off site doses are calculated to be within the limits of 10CFR20.

- Loss of Normal Electric Power (addressed in Section 13.7)

  A loss of normal electric power would cause a reactor shutdown. Shutdown cooling is not required.

- External Events (addressed in Section 13.8)

  External events are considered with respect to potential mechanical rearrangement of the core (specified in NUREG/CR-2387). The core support structure described in Chapter 4 is secure.

- Mishandling or Malfunction of Experiment (addressed 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.

13.2. 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 (suspended in air) resulting in the release of airborne fission products to the reactor bay and ultimately the environment. Postulated fuel failures could result from a fuel-handling accident or from a failure during operation in the core following a loss of coolant accident. The source term from fuel failure in water would be reduced by migration of particulate activity and the release of gaseous of activity would be delayed. Therefore, this section addresses potential consequences, should a failure occur in air.

Calculations show limits on derived air concentration and effluent are met. The accident has the potential to release an amount that exceeds annual limit on intake because of radioactive strontium generated in the fission process and released by the event, but there is no realistic way that the total inventory could be collected for a single individual uptake.

13.2.1. Assumptions

- Burnup calculations are performed per metric ton of uranium in fuel. Radionuclide concentrations (in Ci/MTU) are then used with actual fuel mass for one element to determine fission product inventory for a single element.

- Fuel is irradiated in continuous steady-state operation at specified power levels to a burnup of ten grams ^{235}U, assumed end of useful fuel life. At the end of useful life, one week of regular operations (8 hours per day) is assumed.

- Calculations are performed assuming a 5-minute decay time for radioisotope 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 released is assumed to be 1.0×10^{-4}. This is a very conservative value prescribed in NUREG/CR-2387[63] compared to the value of 1.5×10^{-5} measured at General Atomics[64] which is used in SARs for other reactor facilities[65].

- 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 NUREG/CR-2387.

- The reactor bay free air volume is 4120 m^3. Ten percent (10%) of this volume is not credited in dilution calculations.

- Radioisotopes specified in NUREG/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 10CFR20 Appendix B is provided in Table 13.1. *(The 10CFR20 Appendix B Annual Limit on Intake, ALI, and*

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[63] NUREG/CR-2387, "Credible Accidents for TRIGA and TRIGA Fueled Reactors," prepared by S. C. Hawley and R.L. Kathren, Pacific Northwest National Laboratory PNL-4208 (1982).
[64] Simnad, M.T., F. C. Faushee, and G.B. West, "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology Vol. 28, pp. 31-56 (1976).
[65] NUREG-1390, "Safety Evaluation Report Relating to the Renewal, of the Operating License for the TRIGA Training and Research Reactor at the University of Arizona," Report NUREG-1390, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1990.

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 and therefore daughters are not calculated or considered separately.)

- While the licensed power level for the NETL reactor is 1.1 MW, to be conservative calculations were performed with the reactor operating at 2 MW and with a hot channel core peaking factor of 1.7. This results in a single fuel element power of 43.2 kW for an 81 element core.

Table 13.1, 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	2.0E5	8.0E-5	3.0E-7	Cs131	3.0E4	1.0E-5	4.0E-8
Br80m	2.0E4	6.0E-6	2.0E-8	Cs132	4.0E3	2.0E-6	6.0E-9
Br82	4.0E3	2.0E-6	5.0E-9	Cs134m	1.0E5	6.0E-5	2.0E-7
Br83	6.0E4	3.0E-5	9.0E-8	Cs135	1.0E3	5.0E-7	2.0E-9
Br84	6.0E4	2.0E-5	8.0E-8	Cs135m	2.0E5	8.0E-5	3.0E-7
I125	6.0E1	3.0E-8	3.0E-10	Cs136	7.0E2	3.0E-7	9.0E-10
I128	1.0E5	5.0E-5	8.0E-4	Cs137	2.0E2	6.0E-8	2.0E-10
I129	9.0E0	4.0E-9	4.0E-11	Cs138	6.0E4	2.0E-5	8.0E-8
I130	7.0E2	3.0E-7	2.0E-10	Sr85	2.0E3	6.0E-7	2.0E-9
I131	5.0E1	2.0E-8	2.0E-10	Sr85m	6.05	3.0E-4	9.0E-7
I132	8.0E3	3.0E-6	2.0E-8	Sr87m	1.0E5	5.0E-5	2.0E-7
I133	3.0E2	1.0E-7	1.0E-9	Sr89	1.0E2	6.0E-8	2.0E-10
I134	5.0E4	2.0E-5	6.0E-8	Sr90	4.0E0	2.0E-9	6.0E-12
I135	2.0E3	7.0E-7	6.0E-9	Sr91	4.0E3	1.0E-6	5.0E-9
Kr79		2.0E-5	7.0E-8	Sr92	7.0E3	3.0E-6	9.0E-9
Kr81		7.0E-4	3.0E-6				
Kr83m		1.0E-2	5.0E-5				
Kr85		1.0E-4	7.0E-7				
Kr85m		2.0E-5	1.0E-7				
Kr87		5.0E-6	2.0E-8				
Kr88		2.0E-6	9.0E-9				
Xe125		2.0E-5	7.0E-8				
Xe127		1.0E-5	6.0E-8				
Xe129m		2.0E-4	9.0E-7				
Xe131m		4.0E-4	2.0E-6				
Xe133		1.0E-4	5.0E-7				
Xe133m		1.0E-4	6.0E-7				
Xe135		1.0E-5	7.0E-8				
Xe135m		9.0E-6	4.0E-8				
Xe138		4.0E-6	2.0E-8				

13.2.2. Analysis

Analysis of the maximum hypothetical accident is based on fission products generated in the reactor, with methods and strategy as described for calculating the UT TRIGA fission product inventory and the 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, Annual Limit on Intake, Derived Air Concentration, and Effluent Limits. Based on the results, measures prescribed by the Radiation Protection Program would be required for worker protection in the worst-case scenario, and effluent limits are met.

13.2.2.a. Radionuclide Inventory Buildup and Decay

TRIGA radionuclide inventory is a function of the mass of uranium and transuranic material that fission to yield thermal power P (kW) in neutron interactions. The fission rate is related to the thermal power by the factor $k = 3.12 \times 10^{13}$ fissions per second per kW. Given a fission product radionuclide produced with yield Y, and which decays with rate constant λ, equilibrium activity A_∞ (Bq) of the fission product exists when the rate of creation by fission is equal to the rate of loss by decay, i.e., $A=\lambda \cdot N$ (assuming power is 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:

$$A(t) = A_\infty \cdot \left(1 - e^{-\lambda \cdot t}\right) \qquad \text{Equation 13.1}$$

For times much greater than the half-life of the radionuclide, $A(t) = A_\infty$. 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:

$$A(t) = A_\infty \cdot \left(1 - e^{-\lambda \cdot t_1}\right) \cdot e^{-\lambda \cdot (t - t_1)} \qquad \text{Equation 13.2}$$

The fission product ^{131}I has a half-life of 8.04 days ($\lambda = 0.00359\ h^{-1}$) and a chain (cumulative) fission product yield of about 0.031 atoms/fission. At a thermal power of 1 kW, the equilibrium activity is $A_\infty = 9.67 \times 10^{11}$ Bq (26.1 Ci). After four hours of operation the activity is about 0.37 Ci. For equilibrium operation of the core at 3.5 kW, distributed over 81 fuel elements, the average activity per element would be (26.1 Ci/kW) × (3.5 kW) ÷ (81 fuel elements) = 1.13 Ci per fuel element. Conservatively, the worst-case element would contain about twice this activity (assuming a power peaking factor of 2.0). With a release fraction of 1.0×10^{-4}, the activity available for release would be about (1.13 Ci per fuel element) × (2 kW hottest element/kW average element) × (1.0×10^{-4} activity released/activity produced) = 2.26×10^{-4} Ci released. 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 can account for depletion of ^{235}U and ingrowth of ^{239}Pu.

13.2.2.b. Fission Product Inventory Calculations

General Atomics has demonstrated that TRIGA fuel can be operated safely up to 50% burnup. However, the reactivity contribution to criticality at full power fuel temperature is negligible or negative when burnup for TRIGA fuel containing 8.5% uranium enriched to 19.5% reaches about six grams ^{235}U. At that point, the element is removed from service. Thus, the end of TRIGA 8.5% fuel element life is approximately six grams burnup, but a 10-gram burnup is used in calculations to maximize potential fission product inventory and be conservative.

SCALE depletion sequence T-6 was used to generate inventories of radioactive fission products for operation at steady state power until target burnup was achieved. The sequence uses KENO VI to develop a reactor specific (geometry and materials) flux. SCALE integrates calculations of flux averaged cross sections by modules in sequences 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. Since ORIGEN calculates the burnup of the whole core the fission product inventory values are core averages. ORIGEN defaults to one metric ton of heavy metal (i.e., uranium) for calculations. Thus, the results are reported as per 1 MTU. ORIGEN ARP (a code in the T-6 depletion sequences) was used to determine the fission product inventory following specified decay intervals.

The inventories were then scaled to reflect the license power limit, potential instrument error, and peaking factor of the fuel element producing the most power level. 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 from hours to days are therefore not well represented. A schedule for one 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 to obtain fission product inventories, gaseous and particulate (Tables 13.2a and 13.2b), that are more representative of NETL. The hot channel fission product inventory was calculated from the average values reported by ORIGEN using the peaking factor of the hot channel.

Table 13.2a, Gaseous Fission Product Inventory (Ci/MTU) for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
br80	7.7E-3	5.4E-3	4.5E-3	1.2E-3	9.8E-5	3.5E-16	0.0	0.0	0.0	0.0
br80m	4.6E-3	4.3E-3	3.9E-3	1.1E-3	9.1E-5	3.3E-16	0.0	0.0	0.0	0.0
br82	8.9	8.9	8.8	7.5	5.5	2.0E-1	3.9E-6	2.0E-18	0.0	0.0
br83	1.6E4	1.5E4	1.4E4	1.4E3	1.4E1	1.2E-20	0.0	0.0	0.0	0.0
br84	2.8E4	1.6E4	8.5E3	2.4E-1	2.0E-10	0.0	0.0	0.0	0.0	0.0
i125	5.5E-9	5.5E-9	5.5E-9	5.4E-9	5.4E-9	5.0E-9	3.8E-9	1.9E-9	6.6E-10	7.6E-11
i128	1.4E1	6.0	2.6	4.3E-6	1.2E-17	0.0	0.0	0.0	0.0	0.0
i129	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3	7.2E-3
i130	7.7E1	7.5E1	7.3E1	4.7E1	1.9E1	1.5E-3	5.5E-17	0.0	0.0	0.0
i131	8.6E4	8.6E4	8.6E4	8.4E4	8.0E4	4.4E4	6.0E3	3.4E1	1.4E-2	1.6E-9
i132	1.3E5	1.3E5	1.3E5	1.2E5	1.1E5	2.3E4	1.6E2	3.7E-4	1.3E-12	0.0
i133	2.0E5	2.0E5	2.0E5	1.5E5	8.9E4	3.3E2	3.4E-6	4.9E-27	0.0	0.0
i134	2.3E5	2.1E5	1.7E5	7.3E2	2.6E-3	0.0	0.0	0.0	0.0	0.0
i135	1.9E5	1.8E5	1.7E5	7.2E4	1.3E4	2.7E-4	1.4E-29	0.0	0.0	0.0
kr79	9.9E-9	9.8E-9	9.7E-9	8.3E-9	6.0E-9	2.2E-10	3.9E-15	1.7E-27	0.0	0.0
kr81	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9	2.9E-9
kr83m	1.6E4	1.6E4	1.6E4	3.7E3	5.2E1	2.2E-5	1.8E-5	1.1E-5	5.4E-6	1.2E-6
kr85	5.1E3	5.1E3	5.1E3	5.1E3	5.1E3	5.1E3	5.1E3	5.1E3	5.0E3	4.8E3
kr85m	3.9E4	3.7E4	3.4E4	9.9E3	8.4E2	4.3E-9	0.0	0.0	0.0	0.0
kr87	7.6E4	5.9E4	4.5E4	5.7E2	9.3E-2	0.0	0.0	0.0	0.0	0.0
kr88	1.0E5	9.2E4	8.1E4	1.1E4	2.3E2	3.6E-16	0.0	0.0	0.0	0.0
xe125	4.8E-11	4.7E-11	4.6E-11	3.3E-11	1.7E-11	1.8E-14	2.6E-24	0.0	0.0	0.0
xe127	1.9E-6	1.9E-6	1.9E-6	1.9E-6	1.9E-6	1.6E-6	1.0E-6	3.3E-7	6.0E-8	1.8E-9
xe129m	5.7E-4	5.7E-4	5.6E-4	5.5E-4	5.2E-4	3.0E-4	5.0E-5	4.6E-7	4.1E-10	2.2E-16
xe131m	1.0E3	1.0E3	1.0E3	1.0E3	1.0E3	9.1E2	3.7E2	1.5E1	7.9E-2	1.6E-6
xe133	2.0E5	2.0E5	2.0E5	2.0E5	1.9E5	8.3E4	4.0E3	1.4	9.7E-6	2.3E-16
xe133m	2.1E3	2.1E3	2.1E3	2.1E3	1.9E3	2.8E2	1.9E-1	1.1E-9	4.6E-22	0.0
xe135	1.8E5	1.8E5	1.8E5	1.5E5	6.5E4	2.9E-1	1.9E-19	0.0	0.0	0.0
xe135m	2.4E4	1.9E4	1.8E4	7.4E3	1.4E3	2.7E-5	0.0	0.0	0.0	0.0
xe138	1.9E5	4.3E4	9.7E3	5.3E-7	1.6E-27	0.0	0.0	0.0	0.0	0.0

Table 13.2a, Particulate Fission Product Inventory (Ci/MTU) for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
cs131	1.4E-6	1.4E-6	1.4E-6	1.4E-6	1.3E-6	7.9E-7	1.5E-7	2.1E-9	3.3E-12	6.0E-18
cs132	5.1E-2	5.1E-2	5.1E-2	4.9E-2	4.6E-2	2.2E-2	1.8E-3	3.0E-6	2.0E-10	5.1E-19
cs134m	1.7E2	1.5E2	1.4E2	2.0E1	4.5E-1	1.9E-18	0.0	0.0	0.0	0.0
cs135	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1	5.5E-1
cs135m	1.0E1	7.0	4.7	8.8E-3	3.1E-8	0.0	0.0	0.0	0.0	0.0
cs136	4.8E2	4.8E2	4.8E2	4.7E2	4.5E2	3.1E2	9.3E1	3.9	3.4E-2	2.0E-6
cs137	3.8E4	3.8E4	3.8E4	3.8E4	3.8E4	3.8E4	3.8E4	3.7E4	3.7E4	3.7E4
cs138	2.0E5	1.5E5	9.0E4	4.6	1.0E-8	0.0	0.0	0.0	0.0	0.0
sr85	7.4E-6	7.4E-6	7.4E-6	7.3E-6	7.3E-6	6.8E-6	5.3E-6	2.8E-6	1.1E-6	1.5E-7
sr85m	4.0E-6	2.9E-6	2.2E-6	1.6E-8	8.4E-13	0.0	0.0	0.0	0.0	0.0
sr87m	1.1E-2	1.0E-2	9.0E-3	1.2E-3	2.4E-5	2.6E-23	0.0	0.0	0.0	0.0
sr89	1.4E5	1.4E5	1.4E5	1.4E5	1.4E5	1.2E5	9.1E4	4.0E4	1.2E4	9.2E2
sr90	3.6E4	3.6E4	3.6E4	3.6E4	3.6E4	3.6E4	3.6E4	3.6E4	3.6E4	3.5E4
sr91	1.7E5	1.7E5	1.6E5	9.0E4	2.8E4	1.6E-1	8.8E-19	0.0	0.0	0.0
sr92	1.7E5	1.5E5	1.3E5	1.7E4	2.6E2	2.5E-17	0.0	0.0	0.0	0.0

13.2.2.c. Fission Product Release

Most of the fission products generated during operation are retrained in the fuel matrix with only a fraction of the inventory escaping. The fraction escaping is calculated using release fractions provided by NUREG/CR-2387 applied to each radionuclide. The inventory for a single fuel element was calculated from the concentrations in Tables 13.2a and 13.2b and using a release fraction of 10^{-4}. The Regulatory Guide 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 13.3a and 13.3b for particulate and gaseous fission products.

Table 13.3a. Gaseous Fission Product Release from Single Element (μCi), for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
br80	1.5E-04	1.0E-04	8.6E-05	2.3E-05	1.9E-06	6.7E-18	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br80m	8.7E-05	8.2E-05	7.4E-05	2.1E-05	1.7E-06	6.3E-18	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br82	1.7E-01	1.7E-01	1.7E-01	1.4E-01	1.0E-01	3.8E-03	7.4E-08	3.8E-20	0.0E+00	0.0E+00
br83	3.0E+02	2.9E+02	2.7E+02	2.7E+01	2.7E-01	2.3E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br84	5.3E+02	3.0E+02	1.6E+02	4.6E-03	3.8E-12	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i125	1.0E-10	1.0E-10	1.0E-10	1.0E-10	1.0E-10	9.5E-11	7.2E-11	3.6E-11	1.3E-11	1.4E-12
i128	2.7E-01	1.1E-01	4.9E-02	8.2E-08	2.3E-19	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i129	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04	1.4E-04
i130	1.5E+00	1.4E+00	1.4E+00	8.9E-01	3.6E-01	2.9E-05	1.0E-18	0.0E+00	0.0E+00	0.0E+00
i131	1.6E+03	1.6E+03	1.6E+03	1.6E+03	1.5E+03	8.4E+02	1.1E+02	6.5E-01	2.7E-04	3.0E-11
i132	2.5E+03	2.5E+03	2.5E+03	2.3E+03	2.1E+03	4.4E+02	3.0E+00	7.0E-06	2.5E-14	0.0E+00
i133	3.8E+03	3.8E+03	3.8E+03	2.9E+03	1.7E+03	6.3E+00	6.5E-08	9.3E-29	0.0E+00	0.0E+00
i134	4.4E+03	4.0E+03	3.2E+03	1.4E+01	4.9E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i135	3.6E+03	3.4E+03	3.2E+03	1.4E+03	2.5E+02	5.1E-06	2.7E-31	0.0E+00	0.0E+00	0.0E+00
kr79	1.9E-10	1.9E-10	1.8E-10	1.6E-10	1.1E-10	4.2E-12	7.4E-17	3.2E-29	0.0E+00	0.0E+00
kr81	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11	5.5E-11
kr83m	3.0E+02	3.0E+02	3.0E+02	7.0E+01	9.9E-01	4.2E-07	3.4E-07	2.1E-07	1.0E-07	2.3E-08
kr85	9.7E+01	9.7E+01	9.7E+01	9.7E+01	9.7E+01	9.7E+01	9.7E+01	9.7E+01	9.5E+01	9.1E+01
kr85m	7.4E+02	7.0E+02	6.5E+02	1.9E+02	1.6E+01	8.2E-11	0.0E+00	0.0E+00	0.0E+00	0.0E+00
kr87	1.4E+03	1.1E+03	8.6E+02	1.1E+01	1.8E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
kr88	1.9E+03	1.7E+03	1.5E+03	2.1E+02	4.4E+00	6.8E-18	0.0E+00	0.0E+00	0.0E+00	0.0E+00
xe125	9.1E-13	8.9E-13	8.7E-13	6.3E-13	3.2E-13	3.4E-16	4.9E-26	0.0E+00	0.0E+00	0.0E+00
xe127	3.6E-08	3.6E-08	3.6E-08	3.6E-08	3.6E-08	3.0E-08	1.9E-08	6.3E-09	1.1E-09	3.4E-11
xe129m	1.1E-05	1.1E-05	1.1E-05	1.0E-05	9.9E-06	5.7E-06	9.5E-07	8.7E-09	7.8E-12	4.2E-18
xe131m	1.9E+01	1.9E+01	1.9E+01	1.9E+01	1.9E+01	1.7E+01	7.0E+00	2.9E-01	1.5E-03	3.0E-08
xe133	3.8E+03	3.8E+03	3.8E+03	3.8E+03	3.6E+03	1.6E+03	7.6E+01	2.7E-02	1.8E-07	4.4E-18
xe133m	4.0E+01	4.0E+01	4.0E+01	4.0E+01	3.6E+01	5.3E+00	3.6E-03	2.1E-11	8.7E-24	0.0E+00
xe135	3.4E+03	3.4E+03	3.4E+03	2.9E+03	1.2E+03	5.5E-03	3.6E-21	0.0E+00	0.0E+00	0.0E+00
xe135m	4.6E+02	3.6E+02	3.4E+02	1.4E+02	2.7E+01	5.1E-07	0.0E+00	0.0E+00	0.0E+00	0.0E+00
xe138	3.6E+03	8.2E+02	1.8E+02	1.0E-08	3.0E-29	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table 13.3b, Particulate Fission Product Release from Single Element (μCi), for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
cs131	2.7E-10	2.7E-10	2.7E-10	2.7E-10	2.5E-10	1.5E-10	2.9E-11	4.0E-13	6.3E-16	1.1E-21
cs132	9.7E-06	9.7E-06	9.7E-06	9.3E-06	8.7E-06	4.2E-06	3.4E-07	5.7E-10	3.8E-14	9.7E-23
cs134m	3.2E-02	2.9E-02	2.7E-02	3.8E-03	8.6E-05	3.6E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs135	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04
cs135m	1.9E-03	1.3E-03	8.9E-04	1.7E-06	5.9E-12	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs136	9.1E-02	9.1E-02	9.1E-02	8.9E-02	8.6E-02	5.9E-02	1.8E-02	7.4E-04	6.5E-06	3.8E-10
cs137	7.2E+00	7.2E+00	7.2E+00	7.2E+00	7.2E+00	7.2E+00	7.2E+00	7.0E+00	7.0E+00	7.0E+00
cs138	3.8E+01	2.9E+01	1.7E+01	8.7E-04	1.9E-12	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr85	1.4E-09	1.4E-09	1.4E-09	1.4E-09	1.4E-09	1.3E-09	1.0E-09	5.3E-10	2.1E-10	2.9E-11
sr85m	7.6E-10	5.5E-10	4.2E-10	3.0E-12	1.6E-16	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr87m	2.1E-06	1.9E-06	1.7E-06	2.3E-07	4.6E-09	4.9E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr89	2.7E+01	2.7E+01	2.7E+01	2.7E+01	2.7E+01	2.3E+01	1.7E+01	7.6E+00	2.3E+00	1.7E-01
sr90	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.8E+00	6.7E+00
sr91	3.2E+01	3.2E+01	3.0E+01	1.7E+01	5.3E+00	3.0E-05	1.7E-22	0.0E+00	0.0E+00	0.0E+00
sr92	3.2E+01	2.9E+01	2.5E+01	3.2E+00	4.9E-02	4.8E-21	0.0E+00	0.0E+00	0.0E+00	0.0E+00

13.2.2.d. Consequence Analysis, Annual Limit on Intake

Regulatory Guideline 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses, provides methodology to determine potential doses 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 (Tables 13.4a and 13.4b) 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 sum greater than 1.0 exceeding the ALI value.

ALI values are exceeded for the 3.5 MW case; data from all cases is provided graphically in Figure 13.2 along with values scaled to the nominal 1 and 2 MW case. 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.

Table 13.4a, Fraction of Gaseous Fission Product Inventory to 10CFR20 ALI, for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
br80	7.3E-10	5.1E-10	4.3E-10	1.1E-10	9.3E-12	3.3E-23	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br80m	4.4E-09	4.1E-09	3.7E-09	1.0E-09	8.6E-11	3.1E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br82	4.2E-05	4.2E-05	4.2E-05	3.6E-05	2.6E-05	9.5E-07	1.9E-11	9.5E-24	0.0E+00	0.0E+00
br83	5.1E-03	4.8E-03	4.4E-03	4.4E-04	4.4E-06	3.8E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br84	8.9E-03	5.1E-03	2.7E-03	7.6E-08	6.3E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i125	1.7E-12	1.7E-12	1.7E-12	1.7E-12	1.7E-12	1.6E-12	1.2E-12	6.0E-13	2.1E-13	2.4E-14
i128	2.7E-06	1.1E-06	4.9E-07	8.2E-13	2.3E-24	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i129	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05
i130	2.1E-03	2.0E-03	2.0E-03	1.3E-03	5.2E-04	4.1E-08	1.5E-21	0.0E+00	0.0E+00	0.0E+00
i131	3.3E+01	3.3E+01	3.3E+01	3.2E+01	3.0E+01	1.7E+01	2.3E+00	1.3E-02	5.3E-06	6.1E-13
i132	3.1E-01	3.1E-01	3.1E-01	2.9E-01	2.6E-01	5.5E-02	3.8E-04	8.8E-10	3.1E-18	0.0E+00
i133	1.3E+01	1.3E+01	1.3E+01	9.5E+00	5.6E+00	2.1E-02	2.2E-10	3.1E-31	0.0E+00	0.0E+00
i134	8.7E-02	8.0E-02	6.5E-02	2.8E-04	9.9E-10	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i135	1.8E+00	1.7E+00	1.6E+00	6.8E-01	1.2E-01	2.6E-09	1.3E-34	0.0E+00	0.0E+00	0.0E+00
br80	7.3E-10	5.1E-10	4.3E-10	1.1E-10	9.3E-12	3.3E-23	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SUMS:	47.6	47.5	47.3	42.4	36.4	16.8	2.28	0.01	2.0E-05	1.5E-05

Table 13.4b, Fraction of Particulate Fission Product Inventory to 10CFR20 ALI, for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
cs131	8.9E-15	8.9E-15	8.9E-15	8.9E-15	8.2E-15	5.0E-15	9.5E-16	1.3E-17	2.1E-20	3.8E-26
cs132	2.4E-09	2.4E-09	2.4E-09	2.3E-09	2.2E-09	1.0E-09	8.6E-11	1.4E-13	9.5E-18	2.4E-26
cs134m	3.2E-07	2.9E-07	2.7E-07	3.8E-08	8.6E-10	3.6E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs135	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07	1.0E-07
cs135m	9.5E-09	6.7E-09	4.5E-09	8.4E-12	2.9E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs136	1.3E-04	1.3E-04	1.3E-04	1.3E-04	1.2E-04	8.4E-05	2.5E-05	1.1E-06	9.2E-09	5.4E-13
cs137	3.6E-02	3.6E-02	3.6E-02	3.6E-02	3.6E-02	3.6E-02	3.6E-02	3.5E-02	3.5E-02	3.5E-02
cs138	6.3E-04	4.8E-04	2.9E-04	1.5E-08	3.2E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr85	7.0E-13	7.0E-13	7.0E-13	6.9E-13	6.9E-13	6.5E-13	5.0E-13	2.7E-13	1.0E-13	1.4E-14
sr85m	1.3E-10	9.1E-11	6.9E-11	5.0E-13	2.6E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr87m	2.1E-11	1.9E-11	1.7E-11	2.3E-12	4.6E-14	4.9E-32	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr89	2.7E-01	2.7E-01	2.7E-01	2.7E-01	2.7E-01	2.3E-01	1.7E-01	7.6E-02	2.3E-02	1.7E-03
sr90	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00
sr91	8.1E-03	8.1E-03	7.6E-03	4.3E-03	1.3E-03	7.6E-09	4.2E-26	0.0E+00	0.0E+00	0.0E+00
sr92	4.6E-03	4.1E-03	3.5E-03	4.6E-04	7.1E-06	6.8E-25	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SUMS:	2.0	2.0	2.0	2.0	2.0	2.0	1.9	1.8	1.8	1.7

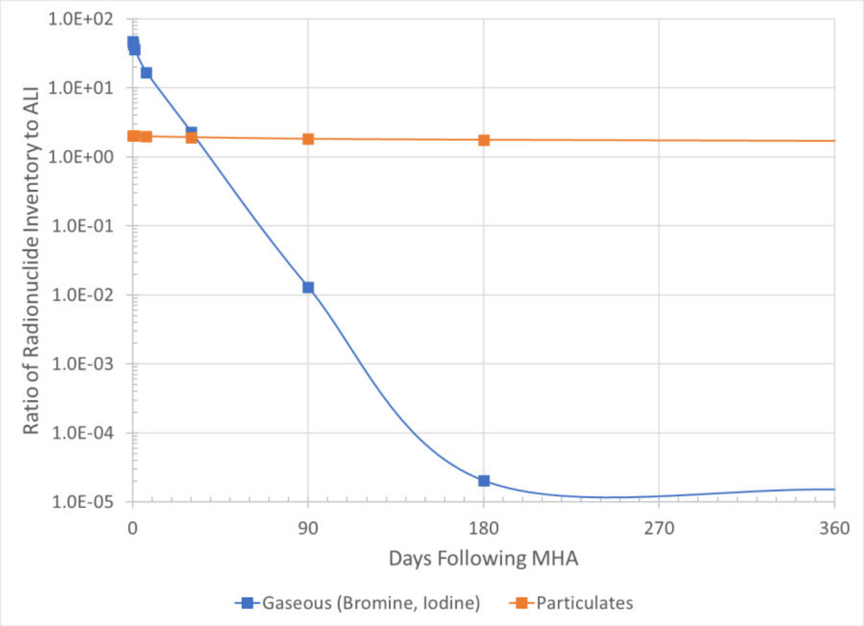

Figure 13.2, Ratio of Radionuclide Inventory to ALI for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

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, while a reduction in the amount of uptake to an individual reduces the ALI ratio.
- in assuming a burnup of ten grams ^{235}U and a continuous operating history: An assumption of 6-gram burnup reduces the inventory of the long-lived ^{90}Sr by approximately 60%, and a less aggressive operating schedule reduces shorter lived radionuclides considerably.
- in assuming that the radionuclide inventory is maintained in one location for the duration of the analysis, and not considering 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.
- in neglecting compensatory or mitigating actions that would respond 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.

The ALI values in the reactor bay from gaseous activity are exceeded in the reactor bay for about 40 days. The activity for ^{90}Sr activity causes the values to exceed ALI for greater than a year following release. Therefore, it is necessary to evaluate the impact on workers and the environment.

13.2.2.e. Consequence Analysis, Derived Air Concentration

The fission product inventory that escapes the fuel matrix is assumed to mix with reactor bay atmosphere. The 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. Similar to ALI analysis, consequences of exposure to a mixture of radionuclides are evaluated based on the derived air concentration in 10CFR20 Appendix B with the results shown in Tables 13.5a and 13.5b.

Table 13.5a, Fraction of Instantaneous Gaseous Fission Product Inventory to 10CFR20 DAC[1], for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
br80	4.9E-10	3.4E-10	2.9E-10	7.7E-11	6.3E-12	2.2E-23	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br80m	3.9E-09	3.7E-09	3.3E-09	9.4E-10	7.7E-11	2.8E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br82	2.3E-05	2.3E-05	2.2E-05	1.9E-05	1.4E-05	5.1E-07	1.0E-11	5.1E-24	0.0E+00	0.0E+00
br83	2.7E-03	2.6E-03	2.4E-03	2.4E-04	2.4E-06	2.0E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00
br84	7.2E-03	4.1E-03	2.2E-03	6.1E-08	5.1E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i125	9.4E-13	9.4E-13	9.4E-13	9.2E-13	9.2E-13	8.5E-13	6.5E-13	3.2E-13	1.1E-13	1.3E-14
i128	1.4E-06	6.1E-07	2.7E-07	4.4E-13	1.2E-24	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i129	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06
i130	1.3E-03	1.3E-03	1.2E-03	8.0E-04	3.2E-04	2.6E-08	9.4E-22	0.0E+00	0.0E+00	0.0E+00
i131	2.2E+01	2.2E+01	2.2E+01	2.1E+01	2.0E+01	1.1E+01	1.5E+00	8.7E-03	3.6E-06	4.1E-13
i132	2.2E-01	2.2E-01	2.2E-01	2.0E-01	1.9E-01	3.9E-02	2.7E-04	6.3E-10	2.2E-18	0.0E+00
i133	1.0E+01	1.0E+01	1.0E+01	7.7E+00	4.5E+00	1.7E-02	1.7E-10	2.5E-31	0.0E+00	0.0E+00
i134	5.9E-02	5.4E-02	4.3E-02	1.9E-04	6.6E-10	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
i135	1.4E+00	1.3E+00	1.2E+00	5.3E-01	9.5E-02	2.0E-09	1.0E-34	0.0E+00	0.0E+00	0.0E+00
kr79	2.5E-15	2.5E-15	2.5E-15	2.1E-15	1.5E-15	5.6E-17	1.0E-21	4.3E-34	0.0E+00	0.0E+00
kr81	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17	2.1E-17
kr83m	8.2E-06	8.2E-06	8.2E-06	1.9E-06	2.7E-08	1.1E-14	9.2E-15	5.6E-15	2.8E-15	6.1E-16
kr85	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.6E-04	2.5E-04
kr85m	1.0E-02	9.5E-03	8.7E-03	2.5E-03	2.1E-04	1.1E-15	0.0E+00	0.0E+00	0.0E+00	0.0E+00
kr87	7.8E-02	6.0E-02	4.6E-02	5.8E-04	9.5E-08	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
kr88	2.6E-01	2.4E-01	2.1E-01	2.8E-02	5.9E-04	9.2E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00
xe125	1.2E-17	1.2E-17	1.2E-17	8.4E-18	4.3E-18	4.6E-21	6.6E-31	0.0E+00	0.0E+00	0.0E+00
xe127	9.7E-13	9.7E-13	9.7E-13	9.7E-13	9.7E-13	8.2E-13	5.1E-13	1.7E-13	3.1E-14	9.2E-16
xe129m	1.5E-11	1.5E-11	1.4E-11	1.4E-11	1.3E-11	7.7E-12	1.3E-12	1.2E-14	1.0E-17	5.6E-24
xe131m	1.3E-05	1.3E-05	1.3E-05	1.3E-05	1.3E-05	1.2E-05	4.7E-06	1.9E-07	1.0E-09	2.0E-14
xe133	1.0E-02	1.0E-02	1.0E-02	1.0E-02	9.7E-03	4.2E-03	2.0E-04	7.2E-08	5.0E-13	1.2E-23
xe133m	1.1E-04	1.1E-04	1.1E-04	1.1E-04	9.7E-05	1.4E-05	9.7E-09	5.6E-17	2.4E-29	0.0E+00
xe135	9.2E-02	9.2E-02	9.2E-02	7.7E-02	3.3E-02	1.5E-07	9.7E-26	0.0E+00	0.0E+00	0.0E+00
xe135m	1.4E-02	1.1E-02	1.0E-02	4.2E-03	7.9E-04	1.5E-11	0.0E+00	0.0E+00	0.0E+00	0.0E+00
xe138	2.4E-01	5.5E-02	1.2E-02	6.8E-13	2.0E-33	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SUMS:	34.6	34.3	34.1	30.0	25.3	11.3	1.5	0.01	2.7E-04	2.5E-04

Table 13.5b, Fraction of Instantaneous Particulate Fission Product Inventory to 10CFR20 DAC [1], for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

	1 sec	30 min	1 hr	8 hr	1 d	7 d	30 d	90 d	180 d	365 d
cs131	7.2E-15	7.2E-15	7.2E-15	7.2E-15	6.6E-15	4.0E-15	7.7E-16	1.1E-17	1.7E-20	3.1E-26
cs132	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.2E-09	5.6E-10	4.6E-11	7.7E-14	5.1E-18	1.3E-26
cs134m	1.4E-07	1.3E-07	1.2E-07	1.7E-08	3.8E-10	1.6E-27	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs135	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08	5.6E-08
cs135m	6.4E-09	4.5E-09	3.0E-09	5.6E-12	2.0E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
cs136	8.2E-05	8.2E-05	8.2E-05	8.0E-05	7.7E-05	5.3E-05	1.6E-05	6.6E-07	5.8E-09	3.4E-13
cs137	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02	3.2E-02
cs138	5.1E-04	3.8E-04	2.3E-04	1.2E-08	2.6E-17	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr85	6.3E-13	6.3E-13	6.3E-13	6.2E-13	6.2E-13	5.8E-13	4.5E-13	2.4E-13	9.4E-14	1.3E-14
sr85m	6.8E-16	4.9E-16	3.7E-16	2.7E-18	1.4E-22	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr87m	1.1E-11	1.0E-11	9.2E-12	1.2E-12	2.5E-14	2.7E-32	0.0E+00	0.0E+00	0.0E+00	0.0E+00
sr89	1.2E-01	1.2E-01	1.2E-01	1.2E-01	1.2E-01	1.0E-01	7.7E-02	3.4E-02	1.0E-02	7.8E-04
sr90	9.2E-01	9.2E-01	9.2E-01	9.2E-01	9.2E-01	9.2E-01	9.2E-01	9.2E-01	9.2E-01	8.9E-01
sr91	8.7E-03	8.7E-03	8.2E-03	4.6E-03	1.4E-03	8.2E-09	4.5E-26	0.0E+00	0.0E+00	0.0E+00
sr92	2.9E-03	2.6E-03	2.2E-03	2.9E-04	4.4E-06	4.3E-25	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SUMS:	1.08	1.08	1.08	1.08	1.07	1.05	1.03	0.99	0.96	0.93

NOTE[1]: DAC limits are based on two thousand hours of exposure over a year; these tables compare the instantaneous value of airborne radionuclides not the 2000-hour exposure period. Integration of the instantaneous values over a year evaluates compliance with DAC limits.

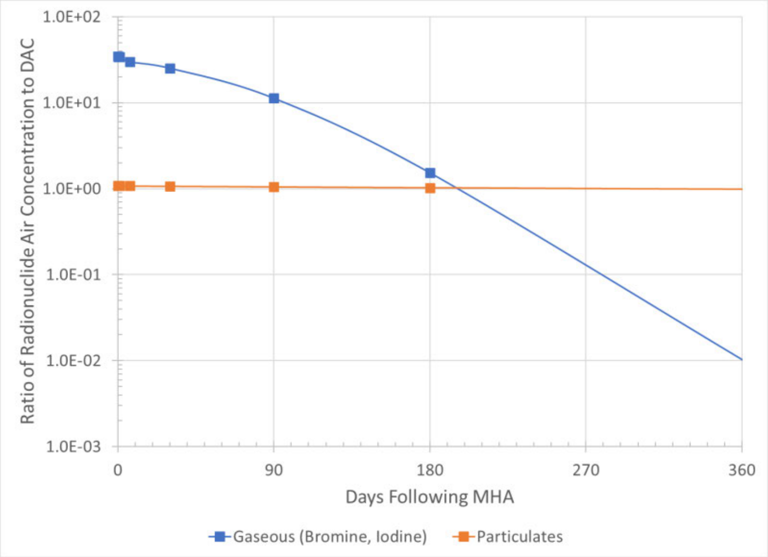

Figure 13.3, Ratio of Radionuclide Concentration to 10CFR 20 DAC Values, for Continuous Operation of a Fuel Element at 43kW and at Various Decay Times Following Shutdown

DAC values apply to continuous exposure over a year. Concentrations averaged using time interval weighting over a year following the event are provided in Table 13.6. The DAC values are exceeded for gaseous fission product concentration for about 40 days and particulate for about 90 days following the accident assuming continuous operation at 43 kW for the fuel rod. However, since the DAC is a limit on an annual exposure, continuous exposure over the year would result in less than 1 DAC for both gaseous and particulate for the maximum hypothetical power level and significantly less than 1 DAC for the licensed power level. A summary of results is provided in Table 13.6. DAC ratios are above one for about 40 days following the MHA.

Table 13.6, Summary of Radionuclide Concentrations to DAC Ratios

Time after MHA	3.5 MW	
	Gaseous	Particulate
1 sec	35	1.08
30 min	34	1.08
1 hr	34	1.08
8 hr	30	1.08
1 d	25	1.07
7 d	11	1.05
30 d	1.5	1.03
90 d	0.0090	0.99
180 d	2.7E-4	0.96
365 d	2.5E-4	0.93
Weighted Average	0.36	0.95

This analysis is conservative:

- as in the ALI analysis, in assuming a 10-gram ^{235}U burnup and a continuous operating history. A slightly more realistic assumption of 6-gram burnup would reduce the inventory of the long-lived ^{90}Sr by approximately 60%, interpolating between the particulate ratios indicates that the DAC ratio is 1.0 at or below 3.5 MW, and a less aggressive operating schedule reduces shorter lived radionuclides considerably.
- in assuming 10% of the volume is occupied by equipment. Increasing the reactor bay air volume decreases nuclide concentration.
- in assuming that the radionuclide inventory is maintained in one location for the duration of the analysis, neglecting 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.

- in not considering 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 maximum hypothetical accident under extremely conservative assumptions for approximately 90 days, access control to manage personnel exposure under the Radiation Protection Program is adequate to maintain personnel dose within limits of 10CFR20.

13.2.2.f. Consequence Analysis, Effluent Release Directly from Reactor Bay

The radionuclide concentration in the reactor bay atmosphere following the maximum hypothetical accident is compared to the effluent limit (Table 13.7), assuming the radionuclide inventory is not transported from confinement and is only removed through decay. The results demonstrate that release of reactor bay atmosphere to the environment requires credit for mitigating factors to meet limits. However, individuals are not directly exposed to effluent releases and transport of radionuclides with atmospheric dispersion reduces maximum exposure.

Table 13.7, Ratio of Reactor Bay Air Radionuclide Concentration Following MHA Compared to Effluent Limit

Time	Gaseous	Particulate
1 s	3583	355
30 m	3524	355
1 h	3494	354
8 h	3026	353
1 d	2545	352
7 d	1132	346
30 d	153	339
90 d	0.907	326
180 d	0.0038	319
365 d	0.0036	307

13.2.2.g. Consequence Analysis, Effluent Stack Release

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, 2^{nd} Ed., 1994) reports dispersion parameters determined experimentally for urban areas (see Table 13.8).

Table 13.8, 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 calculate 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 as shown in the following:

$$Q_0 \cdot \frac{\chi}{Q}_x = N_x \qquad \text{Equation 13.3}$$

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 concentration and the decay constant (λ) is the activity of the radionuclide. Therefore, where $\dot{w}$ is the volumetric flow rate (in units consistent with the nuclide concentration) Eqn. 13.3 becomes:

$$A_0 \cdot \frac{\chi}{Q}_x \cdot \dot{w} = N_x \cdot \lambda_x \qquad \text{Equation 13.4}$$

The analysis here considers two cases. 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 through the event. In the second case, the discharge is not through the stack, but through normal building aspiration processes.

13.2.2.g.1. Releases from Stacks

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, addresses releases from stacks. The ground-level relative concentration at the plume centerline for stack releases is:

$$\chi/Q = \frac{1}{\pi \cdot \bar{U}_h \cdot \sigma_y \cdot \sigma_z} \cdot e^{\frac{-h_e^2}{2 \cdot \sigma_z}} \qquad \text{Equation 13.5}$$

Where:

$\bar{U}_h$ is the wind speed applicable to the release height,
h_e is the effective stack height, and
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 calculated in Chapter 9 is 1.71/{wind velocity} m above the top of the stack at 63 feet (19.202 m),
- A high efficiency particulate filter in the auxiliary purge system is credited in analysis,
- The auxiliary purge system operates at a nominal 1100 cfm (0.52 m^3/s, 5.2E5 cm^3/s used in the dilution calculation), and
- Removal rate for air from the reactor bay in days is (Eqn. 13.6):

$$r = \frac{0.52 \frac{cm^3}{s} \cdot \frac{3600s}{1h} \cdot \frac{24h}{1d}}{4120m^3 \cdot \frac{1E6cm^3}{m^3}} \qquad \text{Equation 13.6}$$

For a limiting case where the wind speed is assumed to be 1, the χ/Q was calculated for each associated set of dispersion parameters (σ_y, σ_z), with the results provided in Table 13.9. The maximum χ/Q value that provides the least dispersion is 0.001416 (Class C, 0.02 km).

Gaseous fission product effluent limits are not met for about 1-day (Table 13.10). However, effluent limits are based on continuous discharge over a year; the total annual average is well within limits. In all cases where the auxiliary purge system is operating with the confinement ventilation system secured, the 10CFR20 effluent limits are met in the maximum hypothetical accident.

Table 13.9, Calculated χ/Q Values

Km	A-B	C	D	E-F
0.01	0.000541	0.000388	8.11E-5	6.54E-7
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

Table 13.10, Stack Exhaust Plume Following MHA Compared to Effluent Limits

Time	Gaseous	Particulate
1 s	2.71	8.11E-5
30 m	2.66	8.10E-5
1 h	2.63	8.10E-5
8 h	2.30	8.06E-5
1 d	1.92	8.04E-5
7 d	0.856	7.96E-5
30 d	0.117	7.75E-5
90 d	6.80E-4	7.42E-5
180 d	2.85E-5	7.20E-5
365 d	2.73E-5	7.04E-5

13.2.2.g.2. Consequence Analysis, Vent\Building Penetration Release

The reactor bay ventilation system as described in Chapter 9 is designed to provide at least two air changes per hour (2.29 m^3 s^{-1}, 2.29E6 cm^3 s^{-1}), and produces about five air changes per hour. Also described, a control system secures ventilation when the atmospheric contamination of the reactor bay reaches a fraction of a DAC. Effluent is then driven by building leakage.

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, addresses releases through vents or other building penetrations. Regulatory Guide 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 Regulatory Guide provides three different formulae (Eqns. 13. 7, 13.8 and 13.9) 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. The equation for plume meander 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}).

$$\chi/Q = \frac{1}{\bar{U}_{10} \cdot \left(\pi \cdot \sigma_y \cdot \sigma_z + \frac{A}{2} \right)} \qquad \text{Equation 13.7}$$

$$\chi/Q = \frac{1}{\bar{U}_{10} \cdot 3 \cdot \pi \cdot \sigma_y \cdot \sigma_z} \qquad \text{Equation 13.8}$$

$$\chi/Q = \frac{1}{\bar{U}_{10} \cdot \pi \cdot \Sigma_y \cdot \sigma_z} \qquad \text{Equation 13.9}$$

Plume meander and building wake effects (mixing effects) from the building at distances less than eight hundred meters from the release use a coefficient, Σ_y in Eqn. 13.9, which is the product of a correction factor (*M*) and the dispersion coefficient, σ_y (Table 13.1). The correction factor is graphically presented in the Regulatory Guide for stability classes D, E, F, and G. Each class has a constant value from the minimum wind speed to 2 m·s^{-1}, decreasing linearly from a maximum value at 2 m·s^{-1} to 0 at 6 m·s^{-1}. The Regulatory Guide allows calculated plume meander factors (*M*) greater than 3 where winds are less than 6 m s^{-1}, or alternately (for winds < 6 m s^{-1}, where $M > 3$):

$$\chi/Q = \frac{1}{\bar{U}_{10} \cdot M \cdot \sigma_y \cdot \sigma_x} \qquad \text{Equation 13.10}$$

Table 13.11, 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 13.12 and the lowest correction factor (M) for the applicable category are in Table 13.13. The χ/Q for each stability class calculated for each equation in Regulatory Guide 1.145, using the minimum values for σ_y, σ_z, and M, are in Table 13.12.

Table 13.12, 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 13.13, Minimum χ/Q by Stability Class

	A-B	C	D	E-F
RG 1.145 (1)	0.004164	0.004351	0.004484	0.004572
RG 1.145 (2)	0.013801	0.024114	0.047368	0.122098
RG 1.145 (3)	n/a	n/a	0.071051	0.122098

The limiting value for χ/Q is 0.122.

13.2.2.h. Source Term Release Rate

The reactor bay ventilation system as described in Chapter 9 is designed to provide at least two air changes per hour (2.29 $m^3\ s^{-1}$, 2.29E6 $cm^3\ s^{-1}$), and produces about five air changes per hour. Also described, a control system secures ventilation when the atmospheric contamination of the reactor bay reaches a fraction of a DAC. Effluent is then driven by building leakage.

Reactor bay doors open to closed spaces, so there is essentially no potential for differential pressure at the reactor bay openings from environmental conditions. Although leakage rates are not tested, all doors to the reactor bay are equipped with seals that are checked for degradation monthly. It is therefore reasonable to consider air leakage from the reactor bay to adjacent space to be significantly less than the limit specified by the International Energy Conservation Code for swinging doors, 0.5 cfm per square foot.

The equipment hatch has two hinged doors, 66 in. x 132.5 in. (60.73 ft^2) with a center seal. All three personnel doors are 36 inches in width, 84 inches in height (21 ft^2 each, 63 ft^2 total). The total surface area is therefore 124 ft^2. Leakage is conservatively assumed 62 cfm (0.0293 $m^3 \cdot s^{-1}$).

The reactor bay doors are not directly connected to the environment but are connected to a space with a personnel door and a rollup door for movement of large equipment. The volume of the space adjacent to the reactor bay doors is 8314 ft^3 (235 m^3). The ratio of the volume exiting the reactor bay to the adjacent space volume is 0.00157.

The most limiting atmospheric dispersion factor (0.122098) and the conservative estimate of the reduction in building leakage (from doors) factor (0.00157) provide a reduction in the airborne concentration of fission products released from the reactor bay in indicated in Table 13.15 reduced by a factor of 2.14E-4.

The ratio of the reactor bay fission product concentration reduced by the minimum atmospheric dispersion in transit to the effluent limits are in Table 13.14.

Table 13.14, Effluent Limit Ratio to Release Concentrations

Time	Gaseous	Particulate
1 sec	0.707	0.070
30 min	0.694	0.070
1 hr	0.686	0.070
8 hr	0.599	0.070
1 d	0.500	0.070
7 d	0.223	0.069
30 d	0.030	0.067
90 d	1.775E-4	0.064
180 d	7.483E-6	0.063
365 d	7.085E-6	0.061

In all cases for the maximum hypothetical accident with the HVAC system secured, the annual effluent concentration limit is met.

This is a conservative analysis:

- Meteorological conditions are assumed to maintain the lowest possible dilution factor for a year; any changes to meteorological conditions will increase dilution and reduce the concentration of the effluent.
- As in the ALI analysis, this analysis assumes a burnup of ten grams ^{235}U and a continuous operating history.
- This analysis assumes less than 100% dilution in the reactor bay volume, with 10% of the volume occupied by equipment. Increasing the volume decreases nuclide concentration.
- This analysis assumes 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.
- There are two doors in the release path from the reactor doors to the environment that are not considered, which will reduce the flow rate from the reactor bay to the environment.

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, effluent limits are met as the radionuclides dilute from the point of release to the receptor location.

13.2.3. Results and Conclusions

For the maximum hypothetical case of a fuel element failure in air following operation to equilibrium radioisotope inventory regulatory limits for DAC and effluents are met. The ALI values for Sr-90 are not met, but the isotope is distributed in the reactor bay and cannot physically be concentrated in a single individual, ventilation is configured to support controlled discharge of reactor bay atmosphere through appropriate filters, and measures to control exposure under the Radiation Protection Program are capable of controlling personnel exposure.

13.3. REACTIVITY INSERTIONS

Rapid compensation of a reactivity insertion is the distinguishing design feature of the TRIGA reactor. Characteristics of a slow (ramp) reactivity insertion are considered separately since the response is modified by the introduction of heat transfer during the transient. The fuel-integrity safety limit is:

> *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℃ 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℃. When clad temperatures can equal the fuel temperature, the fuel temperature limit is 950℃.*[66]

As noted in Chapter 4.2.1.b, a fuel failure in a TRIGA conversion reactor resulted in the imposition of a new pulsing temperature limit, 830°C.

Pulsed reactivity response is simulated using TRACE. Hot channel peaking factor, prompt neutron decay constants, and prompt neutron lifetime identified in Chapter 4 are modeled. The fuel temperature reactivity coefficient is modeled as described in Section 13.1.

Two pulsed reactivity addition scenarios are presented. The first is a reactivity insertion from low power, the second is a reactivity insertion from power with sensible heat. The continuous reactivity insertion analysis assumes the reactor is operating at full power with the reactivity insertion at specified rates after an assumed delay before a scram occurs. Beam Port-1 and -5 penetrate the reflector with an adjacent air-void in the reflector; potential for a flooded flight tube is considered. All analyses demonstrate the reactor can be operated safely, without challenging temperature limits.

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[66] Massoud T. Simnad, Fabian C. Foushee & Gordon B. West (1976) Fuel Elements for Pulsed TRIGA® Research Reactors, Nuclear Technology, 28:1, 31-56, DOI: 10.13182/NT76-A31537

13.3.1. Pulsing Analysis

13.3.1.a. 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 reactivity addition is modeled based on control rod worth curves and response time of the pulse rod.
- Pulses are terminated within 15 seconds.
- Continuous reactivity insertion from power assumes full power operation.

Analysis shows operating within established limits is adequate to assure these events do not challenge fuel temperature limits.

13.3.1.b. Pulsing from Low Power

Power and temperature response to pulsed reactivity insertion from $3.00 to $4.00 was calculated from low power initial conditions (Table 13.15). The pulsed reactivity was then varied to find the margin to the pulsed reactivity approaching the 830°C temperature limit. At $4.40 the maximum fuel temperature was 824°C. Therefore, a maximum pulse limit of 2.2% Δk/k ($3.14) that terminates in 15 seconds is adequate to ensure the pulsing safety limit is met.

Table 13.15, Pulse Response to 15 s

Pulsed δk $	Peak Power W	Peak Temp. °C
$3.00	3.43E+07	530
$3.50	5.51E7	610
$4.20	9.43E7	778
$4.30	1.01E8	795
$4.40	1.08E8	824

13.3.1.c. Pulsing from Power

Calculations were performed to determine the maximum hot channel fuel temperature during the pulse and the final power level following the pulse if allowed to come to equilibrium with the results in Table 13.16. The final power level approaches the maximum LCC power during steady-state operations for $3.00 pulses from 111 kW. The final steady state power level reaches the scram setpoint (with maximum instrument error) for steady state operations for $3.00 pulses from 124 kW. With an initial power level of 174 kW the fuel temperature approaches the safety limit for pulsing. Therefore, an interlock to prevent pulsing from power levels at or above 1 kW is adequate to ensure the temperature safety limit is met.

Table 13.16, Pulsing from Power Summary

Init Core Power (kW)	111 kW	124 kW	174 kW
Initial Ave. Element Power (kW)	1.33 kW	1.47 kW	2.08 kW
Final Element Power (kW)	24.02 kW	26.92 kW	28.80 kW
Final Core Power (kW)	1193 kW	1216 kW	1337 kW
Max Hot Channel Temperature	724 °C	747 °C	826 °C

An analysis of continuous reactivity addition from power was performed. The model was simulated as operating at steady state power level followed by initiation of reactivity addition. The minimum for reactivity addition rate and the maximum time from initiation of scram to an all-rods-down condition were based on the 1992 Technical Specification limits.

13.3.2. Continuous Reactivity Insertion at Power

The maximum hot channel temperature using reactivity addition rates from 0.2% per second to 0.7% per second at delays between reaching the power level scram setpoint and control rod full insertion were calculated (Table 13.17). Full insertion delays of 1 to 3 seconds do not cause the steady state limit to reach the steady state fuel temperature safety limit for cladding temperature less than 500°C up to 0.7% per second. Therefore, a one second control rod drop time (full-out to full-in following initiation of a scram) is adequate to assure fuel temperature safety limit is not challenged during a continuous reactivity insertion of 0.2% per second.

Table 13.17, Peak Temperature Following Rod Full-Insertion Intervals

Reactivity addition rate	0.2%/s $0.29/s	0.4%/s $0.57/s	0.5%/s $0.71/s	0.6%/s $0.86/s	0.7%/s $1.00/s
Delay (seconds)	T_{max} (°C)				
1	573	589	608	627	651
2	585	639	679	726	778
3	609	709	773	863	993
4	630	772	878	1050	1448
5	634	800	992	N/A	N/A

13.3.3. Beam Port Flooding

The beam port experiment authorization used the MCNP model from the Neutronics Report to evaluate the reactivity addition from flooding Beam Port 1/5 and shows a small reactivity change from a completely flooded tube ($0.06) and is well within the reactivity transient analyses.

13.4. LOSS OF REACTOR COOLANT ACCIDENT

There are two concerns in a loss of coolant accident, (1) dose rates from an uncovered core and (2) potential fuel element failure because of fission product decay heat. Reactor shutdown is assumed to occur with initiation of coolant loss, and the interval between initiation of the event and significant loss of water shielding allows some radioactive decay that reduces the source term and decay heat generation. The reactor is instrumented with a scram and notification of the University of Texas Police Department on a low pool water level.

The results of analysis indicate that a loss of coolant event can be managed safely, and that fuel temperature will remain within safe limits if uncovered.

13.5.1 Dose from Uncovered Core

Analysis of dose rates from an uncovered core was accomplished by calculating the fission product inventory of 100 fuel elements at equilibrium conditions using the SCALE depletion sequence (T-6) assuming 10 MWD per element, specifying a decay in the sequence over intervals, and using MCNP to transport the radiation and calculate the dose rates from the fission products at locations of interest.

13.4.1.a. Water Loss

Discharge flow rate from a tank at atmospheric pressure[67] is given by:

$$Q = a \cdot C \cdot \sqrt{2 \cdot g \cdot h} \qquad \text{Equation 13.11}$$

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, and
g is the acceleration of gravity.

Flow from a tank with a constant cross sectional area A is also characterized by:

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[67] Streeter, V. L., E. B. Wylie, and K. W. Bedford, 1998, Fluid Mechanics. McGraw-Hill, Inc. 9th ed, and Daugherty, R. L., J. B. Franzinin, and E. J. Finnemore, 1985; Fluid Mechanics with Engineering Applications. Mc Graw Hill, Inc. 8ed

$$Q = -A \cdot \frac{dh}{dt}$$ Equation 13.12

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 in:

$$t = \frac{A}{a \cdot C} \cdot \left(\sqrt{H_i} - \sqrt{H_f} \right) \cdot \sqrt{\frac{2}{g}}$$ Equation 13.13

As described in Chapter 4, the pool has a composite surface area of a circle with a radius of 39 in. (0.9906 m) and a 39 in. X 78 in. (0.9906 m by 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%). The discharge from the pool through a beam port is through about three meters with multiple abrupt changes in diameter so significant additional loss can be expected; for conservatism, a loss factor of 0.61 is assumed. If a beam port shears and the beam tube in the pool 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 level 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.

13.4.1.b. Consequence Analysis, 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-, 2-, and 3.5-MW 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 (Table 13.18). Radiation transport calculations used the MCNP code.

Modeling of the reactor core was performed using MCNP with the approximate geometry described in Tables 13.19, 13.20, and 13.21 and Figure 13.4. 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 inches) high.

Table 13.18, Gamma Source Term, Photons per Second

	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

Table 13.19, Height/Thickness Dimensions of Unit Cell

Zone	Thickness/Length			Channel	Fuel
1	Lower grid plate	3.27	cm	Al	air, SS
2	Lower element	12.70	cm		
	(a) End Cap (Lower)	5.09	cm	air	air, SS
	(b) Graphite	8.36	cm	air	graph., SS
3	Fuel	38.10	cm	air	fuel, SS
4	Upper element	11.58	cm		
	(a) Graphite	8.36	cm	air	graph., SS
	(b) End Cap (Upper)	3.22	cm	air	air, SS
5	Upper grid plate	1.59	cm	Al	air, SS

Table 13.20, Unit Cell Areas

Unit cell area	8.2071	cm^2
Fuel	4.7886	cm^2
Cladding	0.3397	cm^2
Channel void	3.0788	cm^3

Table 13.21, Material Characterization

Component	Value	Unit
U-$ZrH_{1.6}$ Fuel		
U^{235}	38.00	G
U^{238}	156.87	G
Zr	2052.38	G
H	45.36	G
SS 304 (8.03 g/cc)		
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

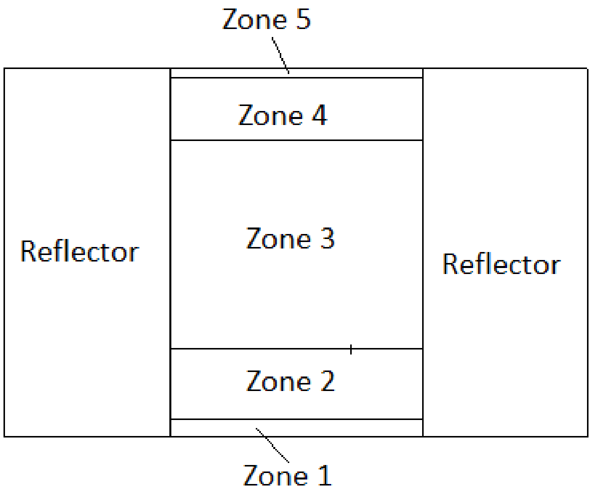


Figure 13.4, Core Model

Axial zones are defined above and below the fuel, and at the grid plate elevations. The zones are described in Table 13.20, 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).

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 within the area formed by connecting three fuel center points (Table 13.21, Unit cell Area). Materials within the unit cell are either fuel, graphite (assumed to have the same cross section as fuel), cladding, or void.

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 (Figure 13.5a and 13.5b), and the top deck neglected.

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 (6) directly over the core. Receptor locations for dose calculations outside the reactor bay were set one 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.

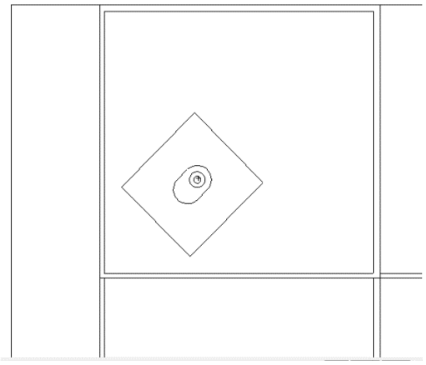


Figure 13.5a, Bay Model Top View

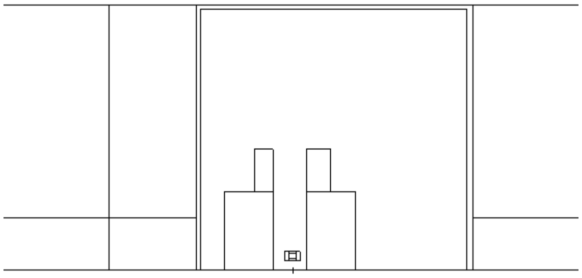


Figure 13.5b, Bay Model Cross Section

The building geometry (Figure 13.6a) is simplified to single thickness walls, and the floor structures are neglected. The colocation boundary (Figures 13.6b and 13.6c) extends four meters into the ground below the reactor bay, and spherically (in air) to approximately seven hundred meters. The results of the calculation are provided in Table 13.22.

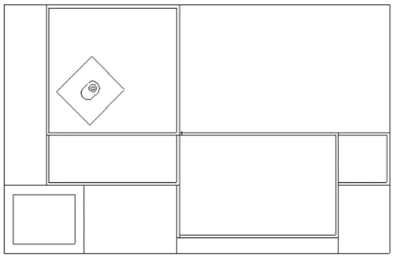


Figure 13.6a, Building Model

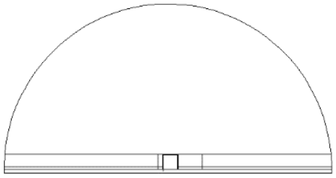


Figure 13.6b, MCNP Side View

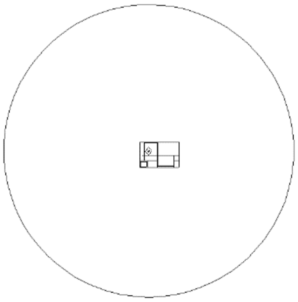


Figure 13.6c, Top View

Table 13.22, Post LOCA Dose Rates

	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 north 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

13.4.2. Maximum Fuel Temperature

The maximum fuel temperature during a LOCA was calculated using TRACE in a 2-step process. The first steps established initial conditions with operations simulated for an element power of 25 kW (slightly greater than the LCC hot channel power) concluding with a shutdown. Four cases were run with the reactor shutdown for 1, 60, 600, and 1200 seconds before a restart case was initiated with air cooling (air temperature assumed at 77°F). The method of ANSI/ANS-5.1-2014, Decay Heat Power in Light Water Reactors, was used to evaluate fission and fission product power decay in time. The maximum cladding temperature of 784°C occurred with 1 s delay before air cooling (Table 13.23). Therefore, on a loss of cooling event following steady-state operation at 25 kW per element, the maximum fuel temperature remains at acceptable levels.

Table 13.23, Loss of Water-Cooling Analysis

Delay For Air Cooling (s)	Maximum Temperature (°C)
1	787
60	780
600	753
1200	733

13.4.3. Results and Conclusions

Maximum 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. Maximum temperatures are within limits.

13.5. LOSS OF COOLANT FLOW

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 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 a low pool water level.

13.5.1. Accident Analysis and Determination of Consequences

If the UT TRIGA operated without coolant flow for an extended period, 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.

13.6. MISHANDLING OR MALFUNCTION OF FUEL

A fuel handling accident is considered to lead to the maximum hypothetical accident, with consequences analyzed in section 13.3.

13.6.1. 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.

13.6.2. Analysis

Releases in water delay or partially retain (because of gas solubility) gaseous fission products. Releases in water substantially retain particulate fission product. Cladding failure under water is therefore bounded by cladding failure in air, the maximum hypothetical accident.

13.7. EXPERIMENT MALFUNCTION

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.

13.7.1. Accident Initiating Events and Scenarios

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.

13.7.2. 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 limit potential experiment failure and assure that if failure does occur there is no reactor damage or radioactivity releases/radiation doses which exceed the limits of 10CFR20. For example, the detailed procedures call for the safety review and approval of all reactor experiments.

13.7.2.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. The successful use of the process for many years at two UT reactors has helped assure the safety of experiments placed in these reactors, demonstrating the process is an effective measure to assure experiment safety at the UT TRIGA reactor.

13.7.2.b. Reactivity Considerations

Technical Specifications limits the reactivity worth of any moveable experiment to a maximum of \$1.00. This limit is well below the maximum reactivity limit analyzed in the insertion of excess reactivity of \$13.4.

Technical Specifications limits the reactivity worth of any single secured experiment to \$2.50. Any inadvertent pulse by experiment manipulation while operating at power is therefore well below the maximum reactivity limit analyzed in the insertion of excess reactivity. A transient that occurs from removal while operating at power will be less severe, and reactor protective systems will terminate operations.

Technical Specifications limits the reactivity worth of all experiments during operation to a maximum of \$3.00. Therefore, removal of all experiments while operating is bounded by the positive reactivity addition analysis.

13.7.2.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 10CFR20. 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 given by:

$$F_x = \frac{{}^{A_x}\!/_{V}}{DAC_x} \qquad \text{Equation 13.14}$$

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 is given by:

$$F_x = \frac{{}^{A_E}\!/_{V} \cdot Y\%}{DAC_x} \qquad \text{Equation 13.15}$$

The total DAC fraction for the element is then:

$$F = \frac{A_E}{V} \cdot \sum_x \frac{Y\%}{DAC_x} \qquad \text{Equation 13.16}$$

For a target DAC fraction, activity is given by:

$$A_E = F \cdot \frac{V}{\sum_x \frac{Y\%}{DAC_x}} \qquad \text{Equation 13.17}$$

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 reactor shutdown 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 in Table 13.24.

Table 13.24, 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
		sum	8.0E6
		sum/vol	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
		sum	3.97E7
		sum/vol	9.35E1

For a 2-hour evacuation period, the DAC fraction is 1000. 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.

13.7.2.d. Explosives

Projected damage to the reactor from experiments involving explosives varies significantly depending on the quantity of explosives irradiated and explosive placement 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 twenty-five milligrams could be safely performed if the containment is properly chosen. A 25-milligram quantity of explosives, upon detonation, releases approximately twenty-five 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 twenty-five 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 13.25 lists the mechanical properties of these encapsulation materials.

Table 13.25, 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. Hoop stress is two times the longitudinal stress, and hence hoop stress is limiting. The maximum stress is calculated by:

$$\sigma_{max} = \frac{p \cdot d}{2 \cdot t}$$ Equation 13.18

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 calculated by:

$$\frac{p \cdot d}{2 \cdot t} \le \sigma_{yield}$$ Equation 13.19

Solving this equation for d/t allows evaluating an encapsulation material with:

$$\frac{d}{t} \leq \frac{2 \cdot \sigma_{yield}}{p} \qquad \text{Equation 13.20}$$

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 13.26. 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 aluminum and stainless steel make satisfactory containers. As a result of the 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 13.26, Container Diameter to Thickness Ratio

Material	***d/t***
Stainless Steel (304)	4.5
Aluminum (6061)	5.1
Polyethylene	0.22

13.8. LOSS OF NORMAL ELECTRIC POWER

13.8.1. 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.

13.8.2. 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 generator 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 shutdown be initiated by the operator on duty. The backup power supply will allow monitoring of the orderly shutdown and verification of the reactor's shutdown condition.

13.9. EXTERNAL EVENTS

13.9.1. 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.

As described in Chapter 4, the core support structure is secured to the floor, the core is surrounded by a reflector bolted to the core support structure, grid plates are bolted to the reflector, and fuel elements are positioned in the core by the grid plates. There is no credible scenario that would disturb the core lattice or structure while simultaneously retaining fuel elements in a critical geometry.

13.9.2. 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.

14. TECHNICAL SPECIFICATIONS

The Technical Specifications are contained in Appendix 14.1.

15. FINANCIAL QUALIFICATIONS

15.1. FINANCIAL ABILITY TO OPERATE A NUCLEAR RESEARCH REACTOR

The University of Texas at Austin is an agency of the State of Texas, as documented in Appendix 15.1. UT-Austin has operated a TRIGA nuclear research reactor since 1967. In 1998, UT-Austin decided to decommission a 250 kW TRIGA located on the main campus and construct a new 1.1 MW TRIGA on the J.J. Pickle Research Campus (PRC). The PRC facility has operated successfully and 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).

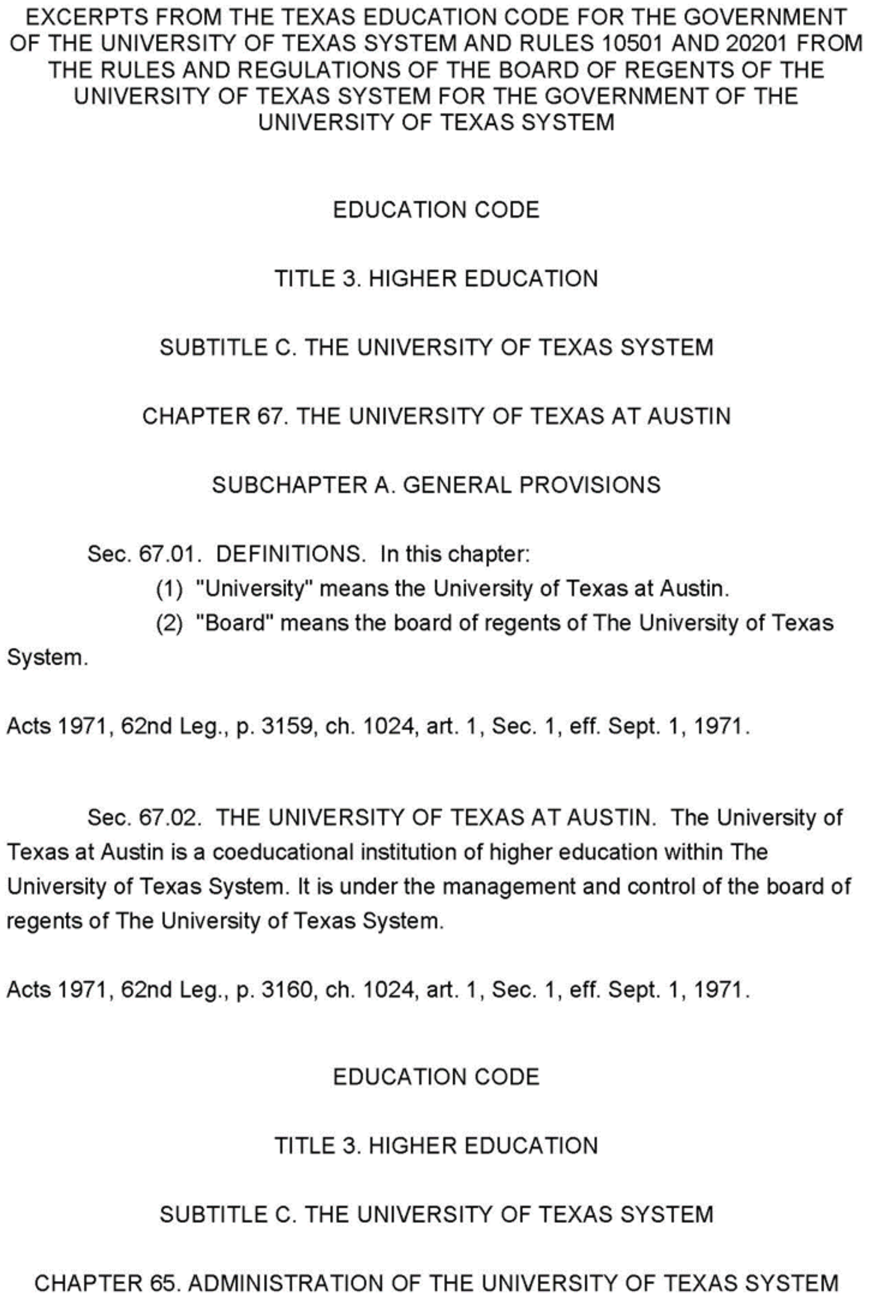
15.2. FINANCIAL ABILITY TO DECOMMISSION THE FACILITY

The University of Texas at Austin 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 15.3.

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.

APPENDIX 15.1 – STATUES AND EXERCPTS REGARDING UT-AUSTIN



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.

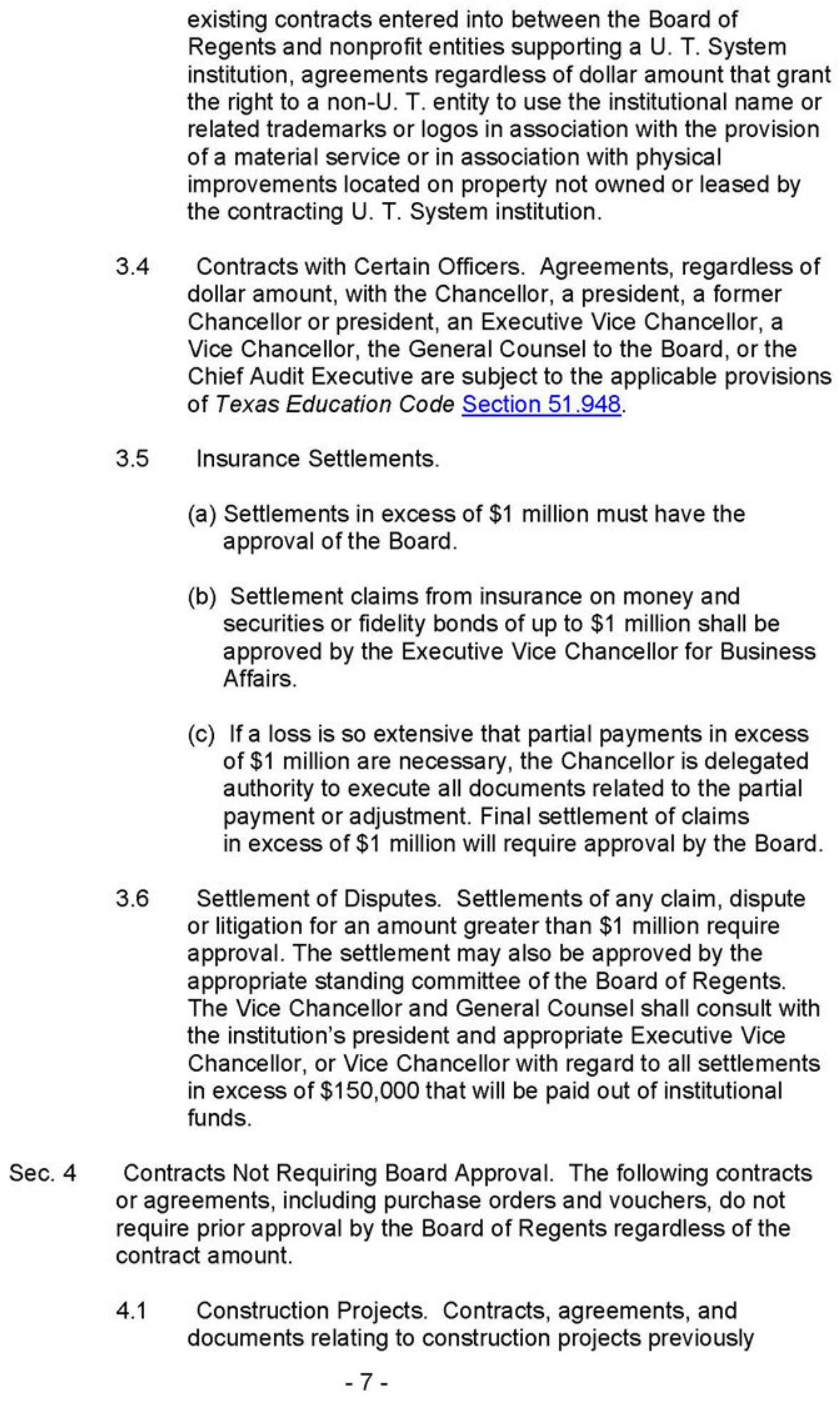
Sec. 2 Compliance with Special Instructions. All authority to execute and deliver contracts, agreements, and other documents is subject to these *Rules and Regulations* and compliance with all applicable laws and special instructions or guidelines issued by the Chancellor, an Executive Vice Chancellor, 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

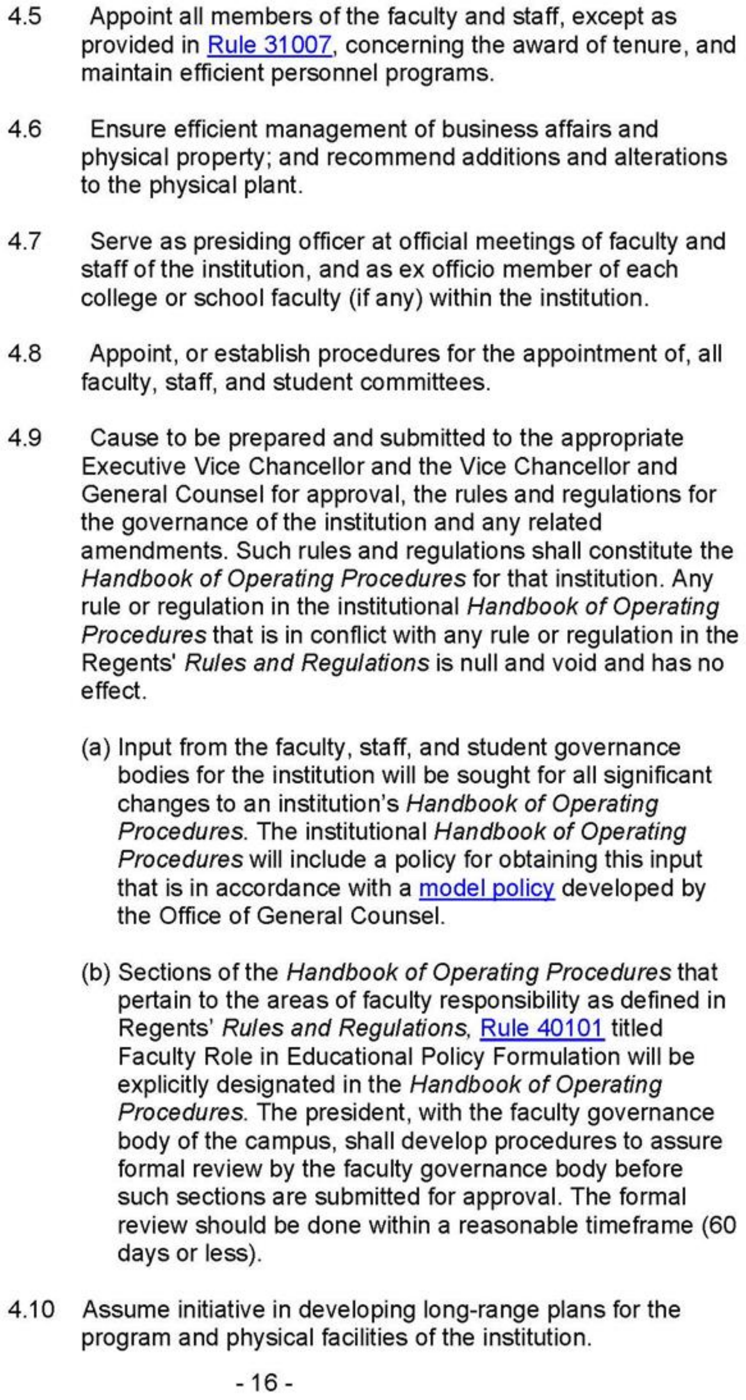
Sec. 4 Duties and Responsibilities. Within the policies and regulations of the Board of Regents and under the supervision and direction of the appropriate Executive Vice Chancellor, the president has general authority and responsibility for the administration of that institution. Specifically, the president 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.

- 16 -

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

APPENDIX 15.2 – 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.

EXPENSES	2024	2025	2026	2027	2028
FTE	$552,458	$569,032	$586,103	$603,686	$621,796
Student support	$14,700	$15,141	$15,595	$16,063	$16,545
Personnel monitoring	$8,000	$8,240	$8,487	$8,742	$9,004
Operator license expenses	$5,800	$5,974	$6,153	$6,338	$6,528
Operations and maintenance	$68,327	$70,377	$72,488	$74,663	$76,903
Communications & Security	$21,000	$21,630	$22,279	$22,947	$23,636
TOTAL	$670,285	$690,394	$711,105	$732,439	$754,412

INCOME	2024	2025	2026	2027	2028
State Budget	$249,296	$256,775	$264,478	$272,412	$280,585
Auxiliary University Fund[1]	$149,112	$153,585	$158,193	$162,939	$167,827
Overhead Return	$204,597	$210,735	$217,057	$223,569	$230,276
Research & service	$211,725	$218,077	$224,619	$231,358	$238,298
TOTAL	$814,730	$839,172	$864,347	$890,277	$916,986

NOTE[1]: Return on UT investment portfolio, consequently fluctuates

APPENDIX 15.3 – DECOMMISSIONING COST ESTIMATES

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 15.3.1, 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:

$$C_{1981,adjusted} = (X)\cdot\left\{(L)\cdot(L_a)+(R)\cdot(R_a)+(O)\cdot(O_a)\right\}$$

Where:

$C_{1981,adjusted}$ is the current value based on the 1981 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.

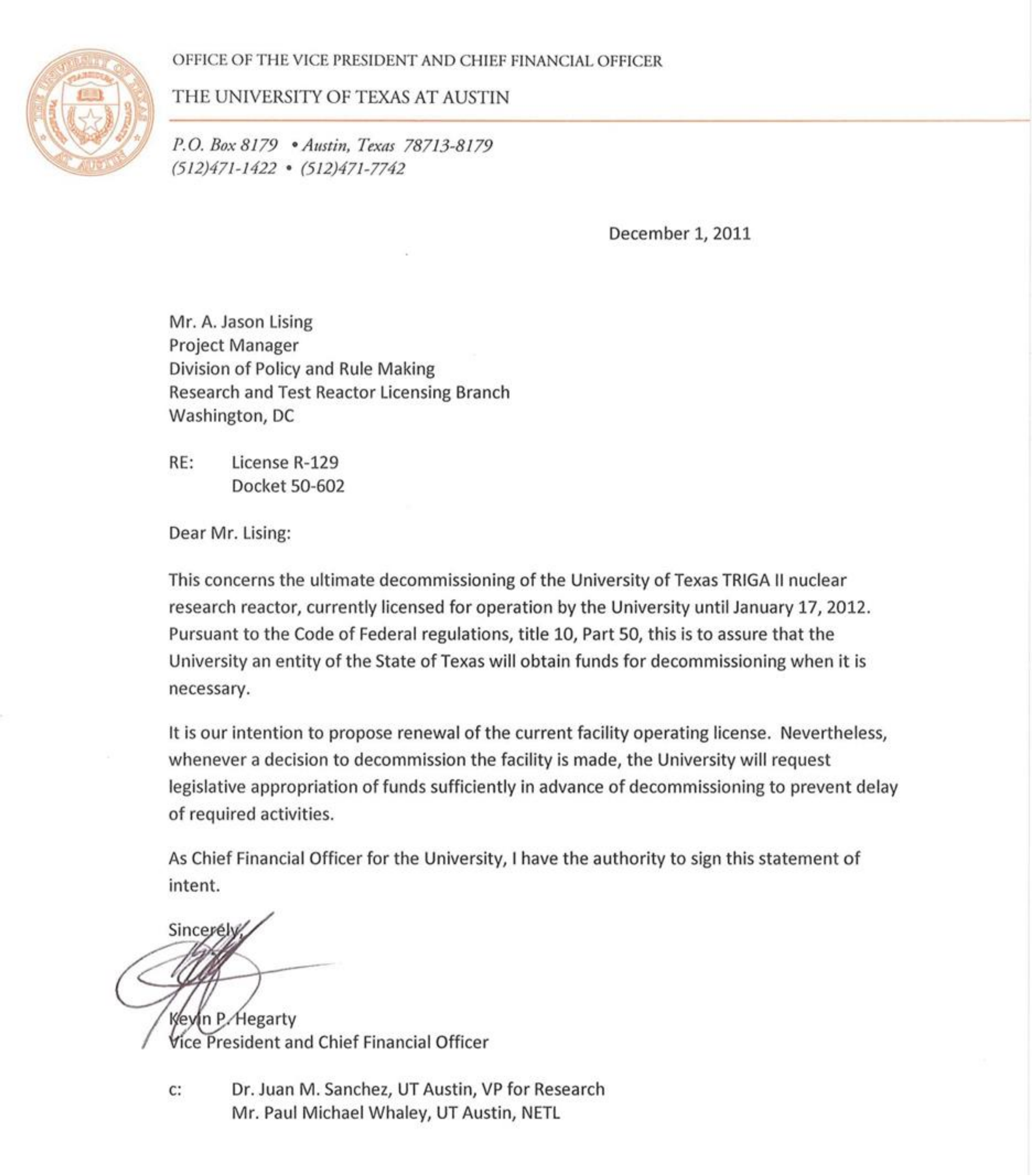
The average of the unspecified ("other") costs is 50.7% of the total cost. The influence of the outliers adds some bias but the average excluding the outliers is 52.0% (a deviation of about 2% from the aggregate), indicating the average value may be representative of the 1.1 MW UT TRIGA.

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 $1,135,078. Assuming an annual rate of 3% inflation, the decommissioning cost at the end of the new 20-year license will be $2,050,076.

Below is a letter signed by the UT administration specifying the University’s intent to acquire funds whenever decommissioning is needed.



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.

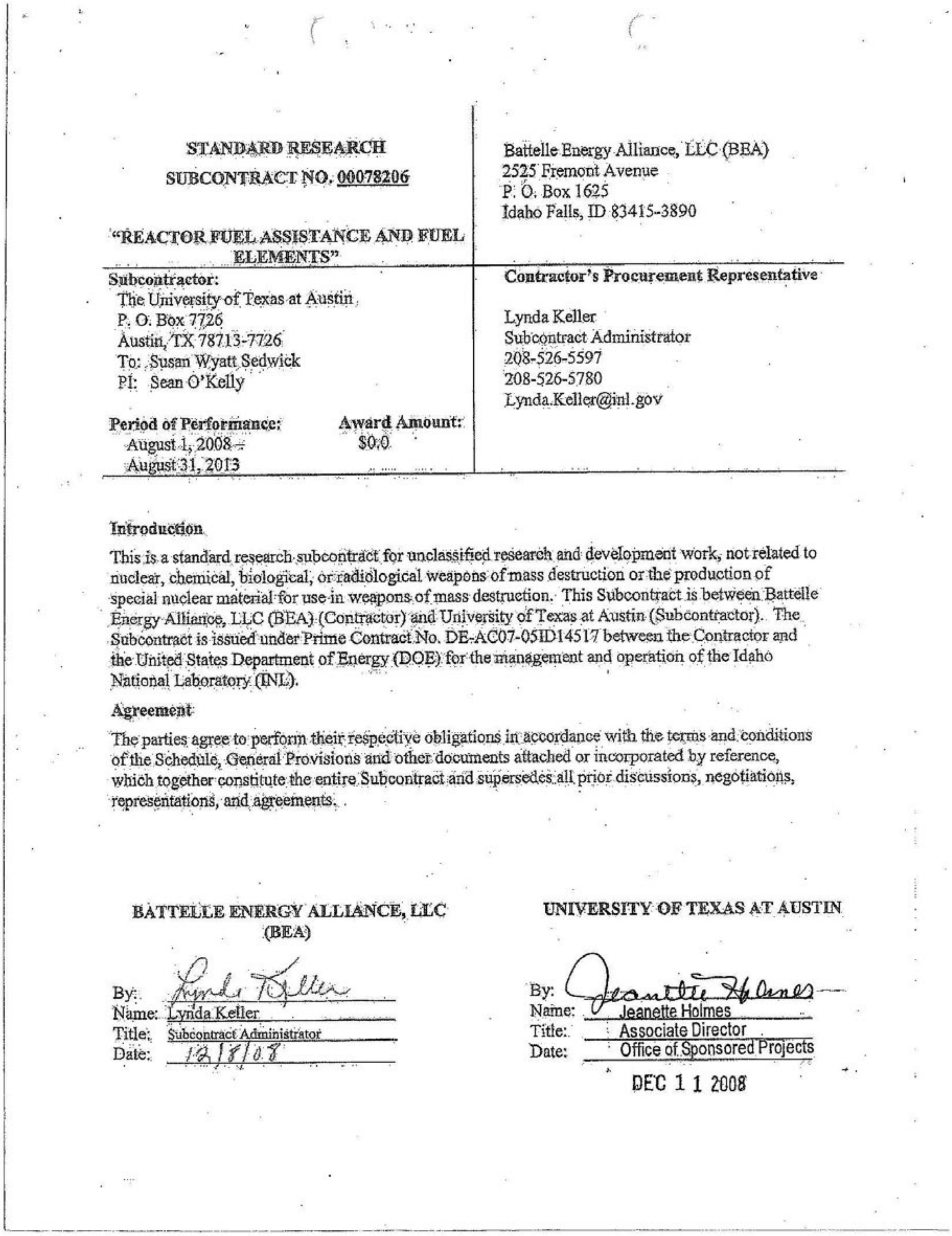
As Chief Financial Officer for the University, I have the authority to sign this statement of intent.

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

APPENDIX 15.4 – DOE FUELS ASSISTANCE CONTRACT



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

Contractor's Procurement Representative

Lynda Keller
Subcontract Administrator
208-526-5597
208-526-5780
Lynda.Keller@inl.gov

Period of Performance:
August 1, 2008 –
August 31, 2013

Award Amount:
$0.0

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.

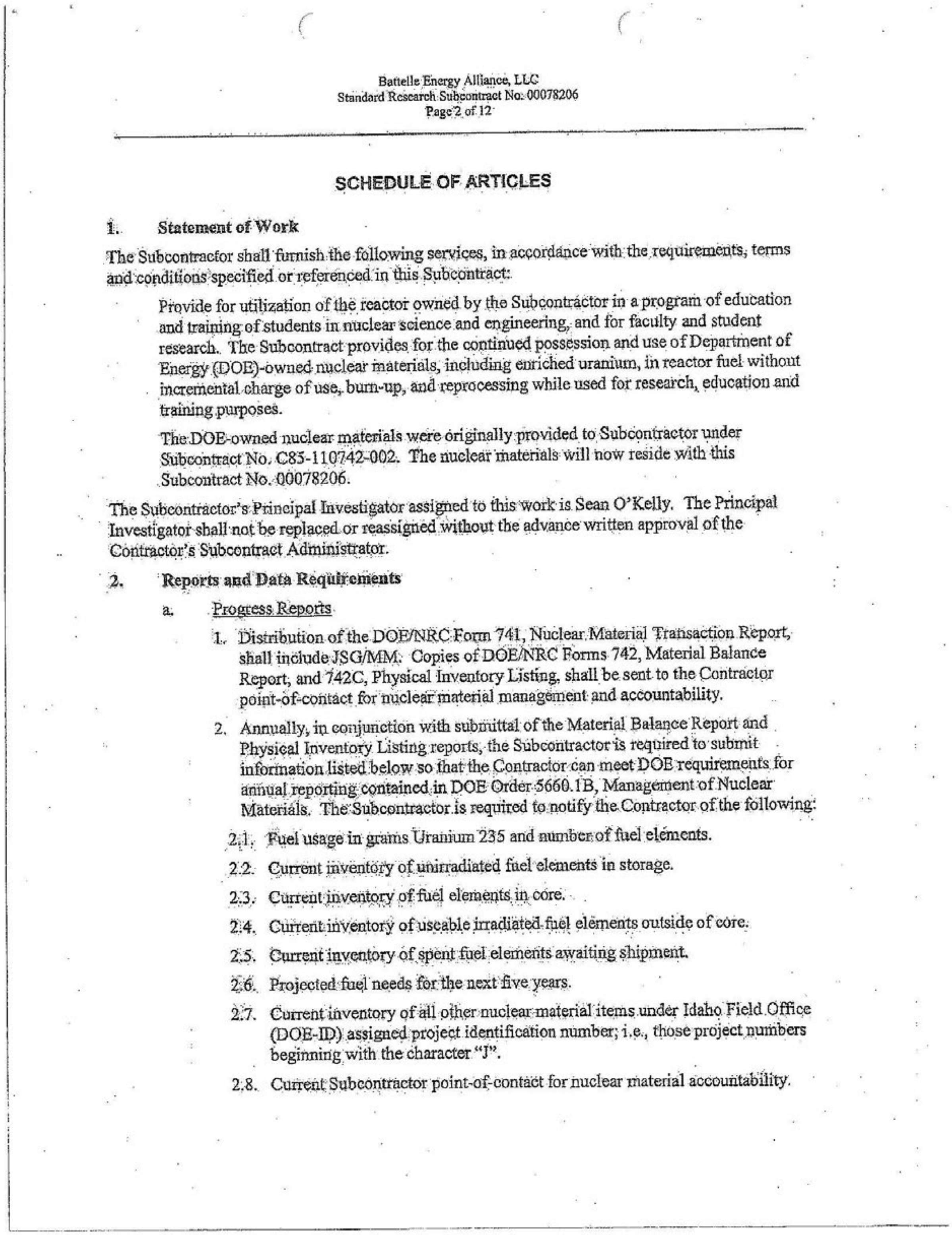
BATTELLE ENERGY ALLIANCE, LLC (BEA)

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.

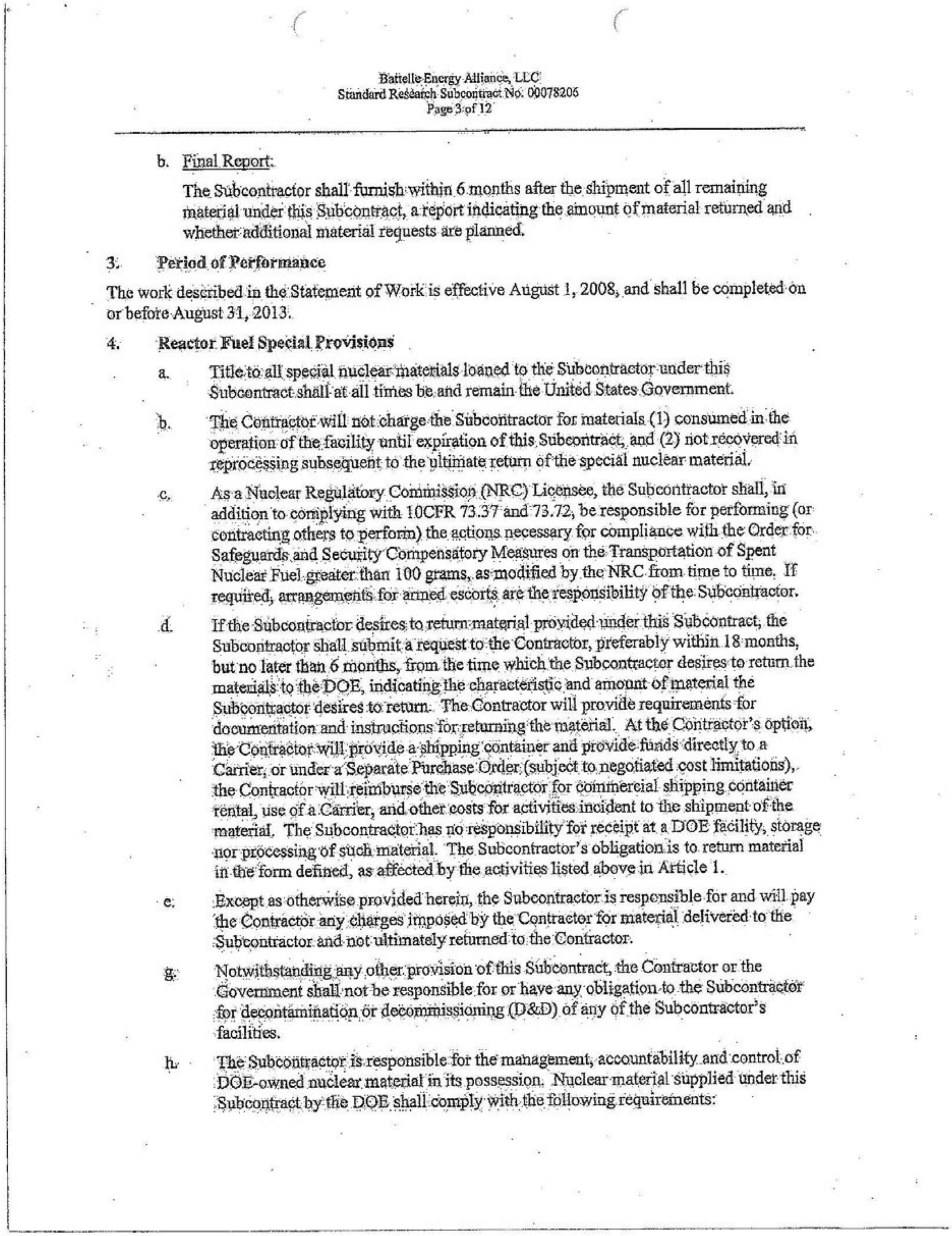
2.4. Current inventory of useable irradiated fuel elements outside of core.

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.

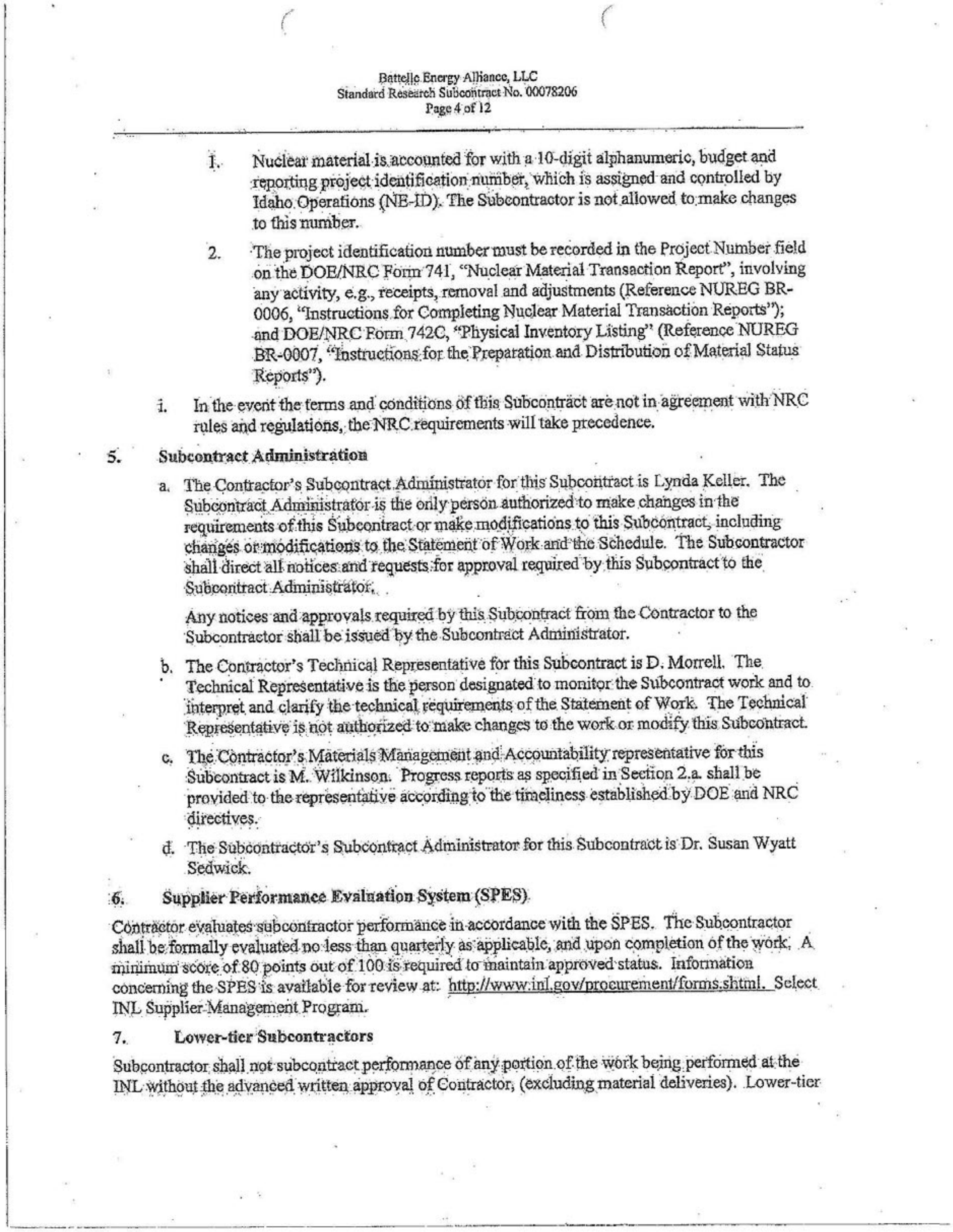
c. As a Nuclear Regulatory Commission (NRC) Licensee, the Subcontractor shall, in addition to complying with 10CFR 73.37 and 73.72, be responsible for performing (or contracting others to perform) the actions necessary for compliance with the Order for Safeguards and Security Compensatory Measures on 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.

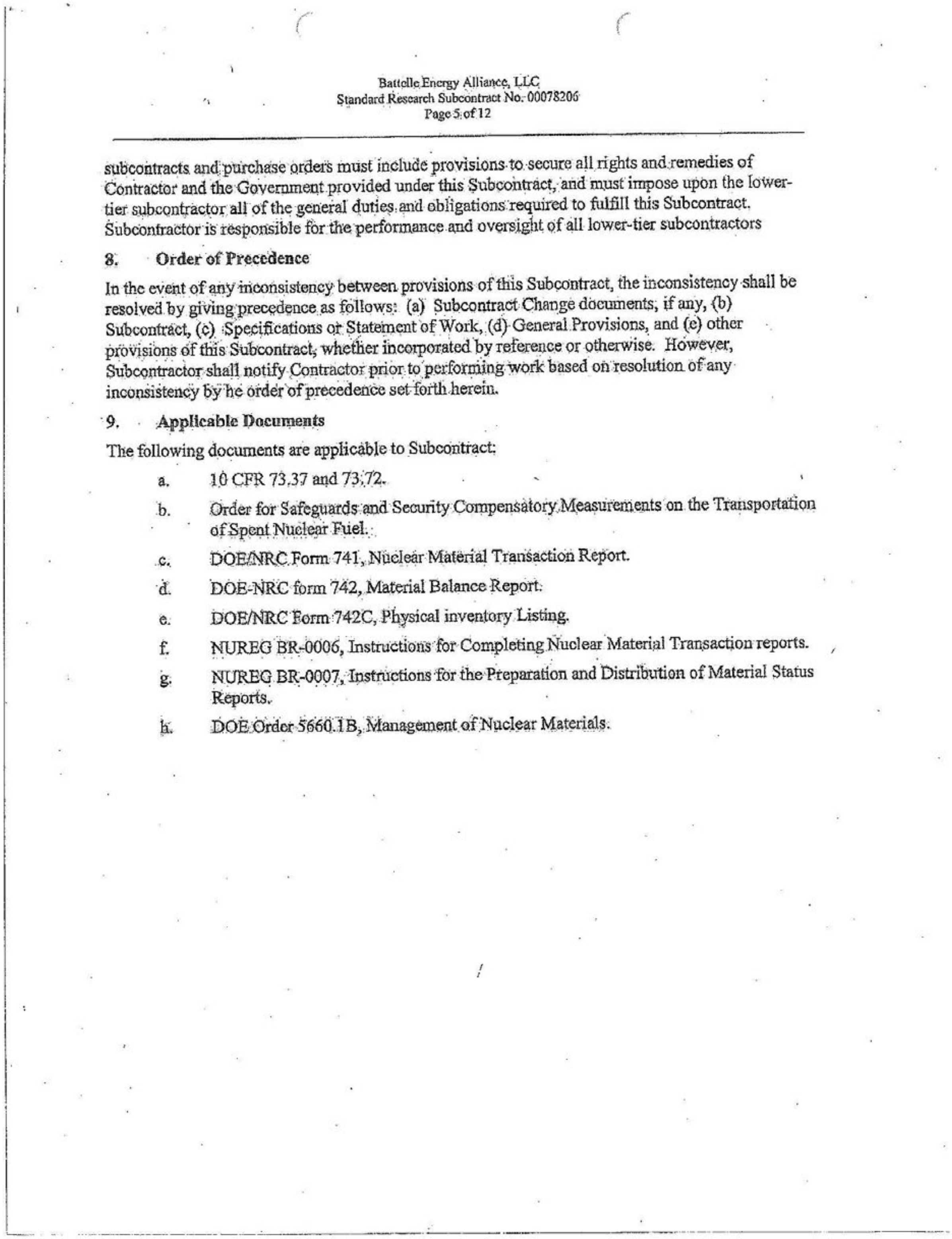
d. The Subcontractor's Subcontract Administrator for this Subcontract is Dr. Susan Wyatt Sedwick.

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

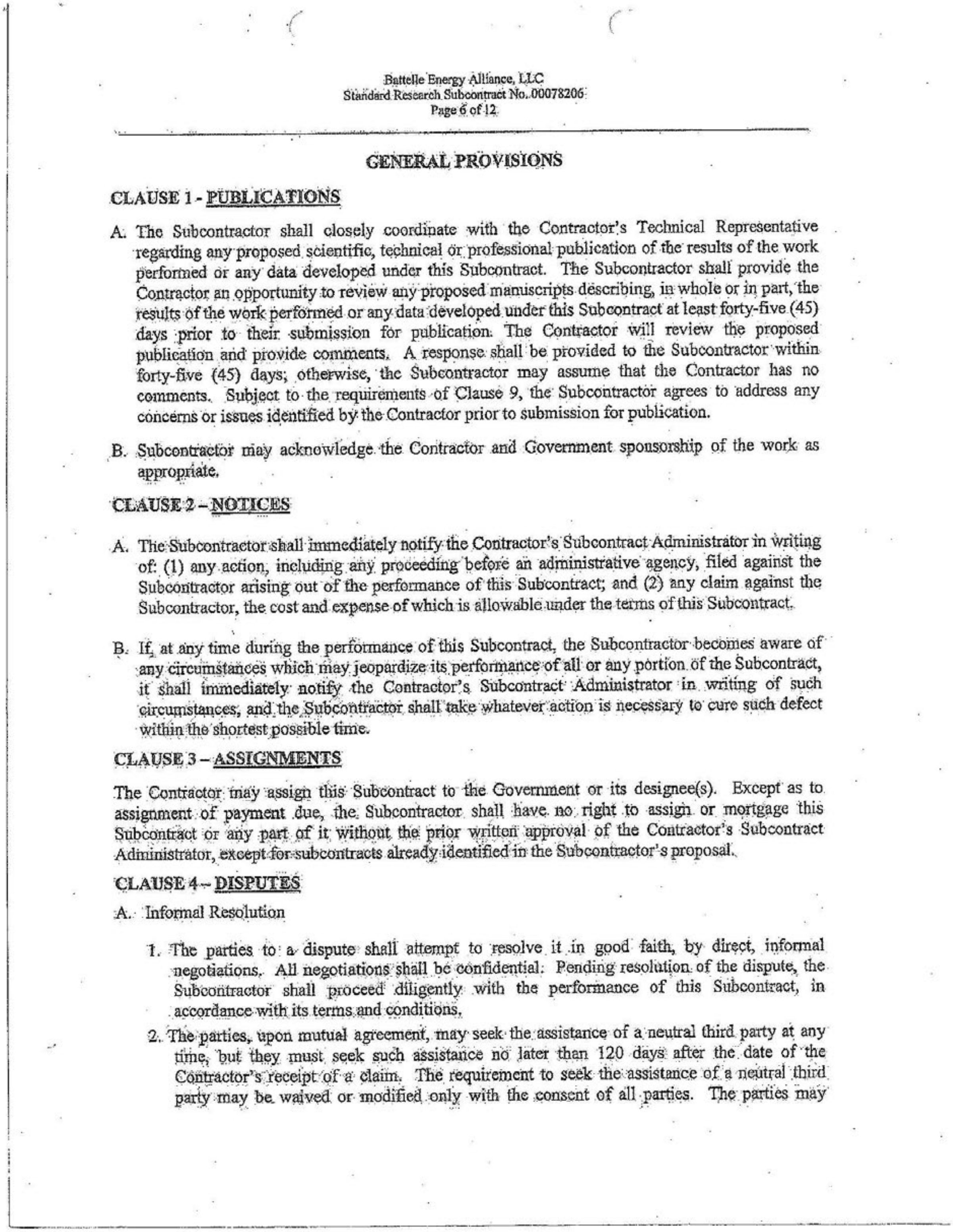
8. Order of Precedence

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.

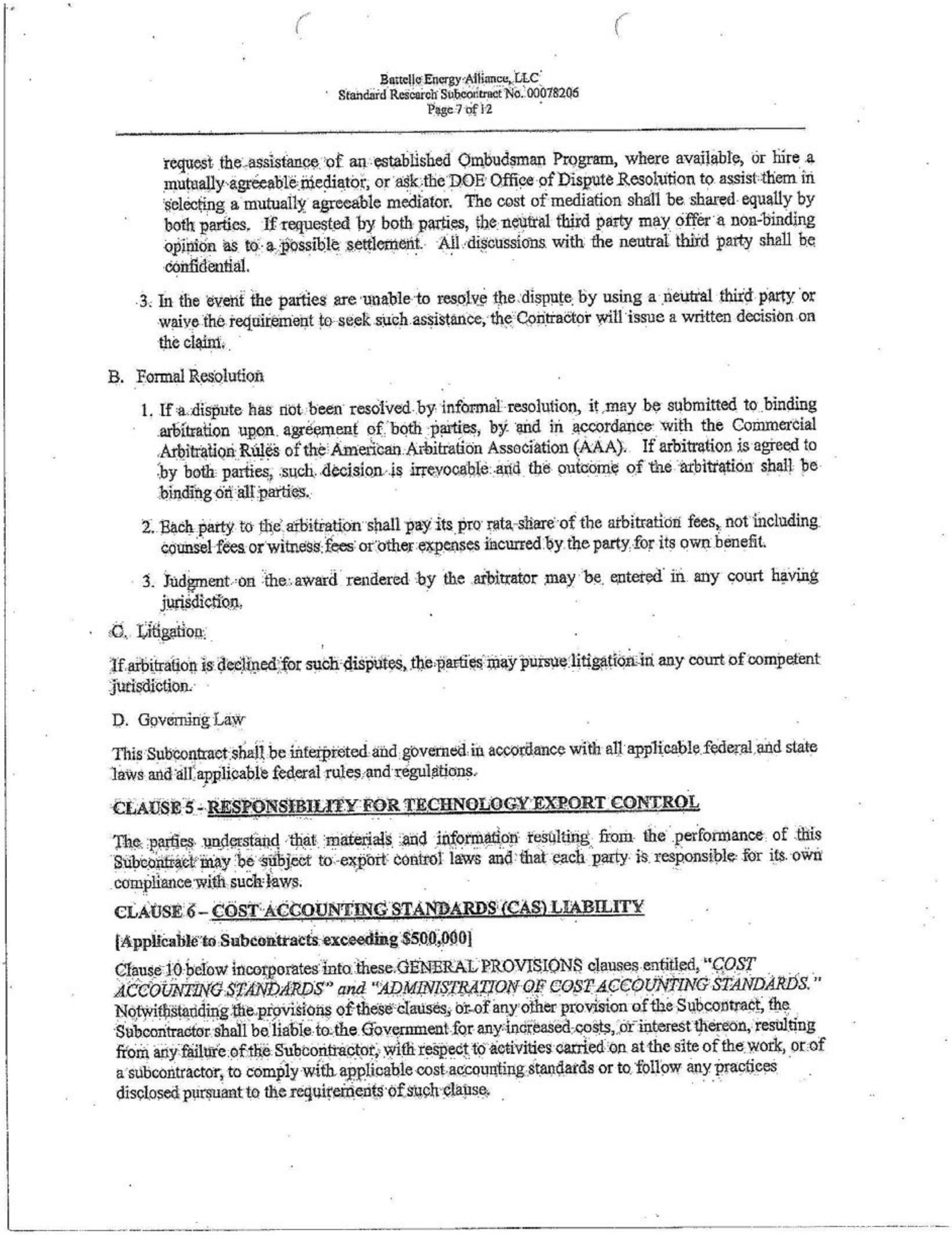
CLAUSE 3 – ASSIGNMENTS

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.

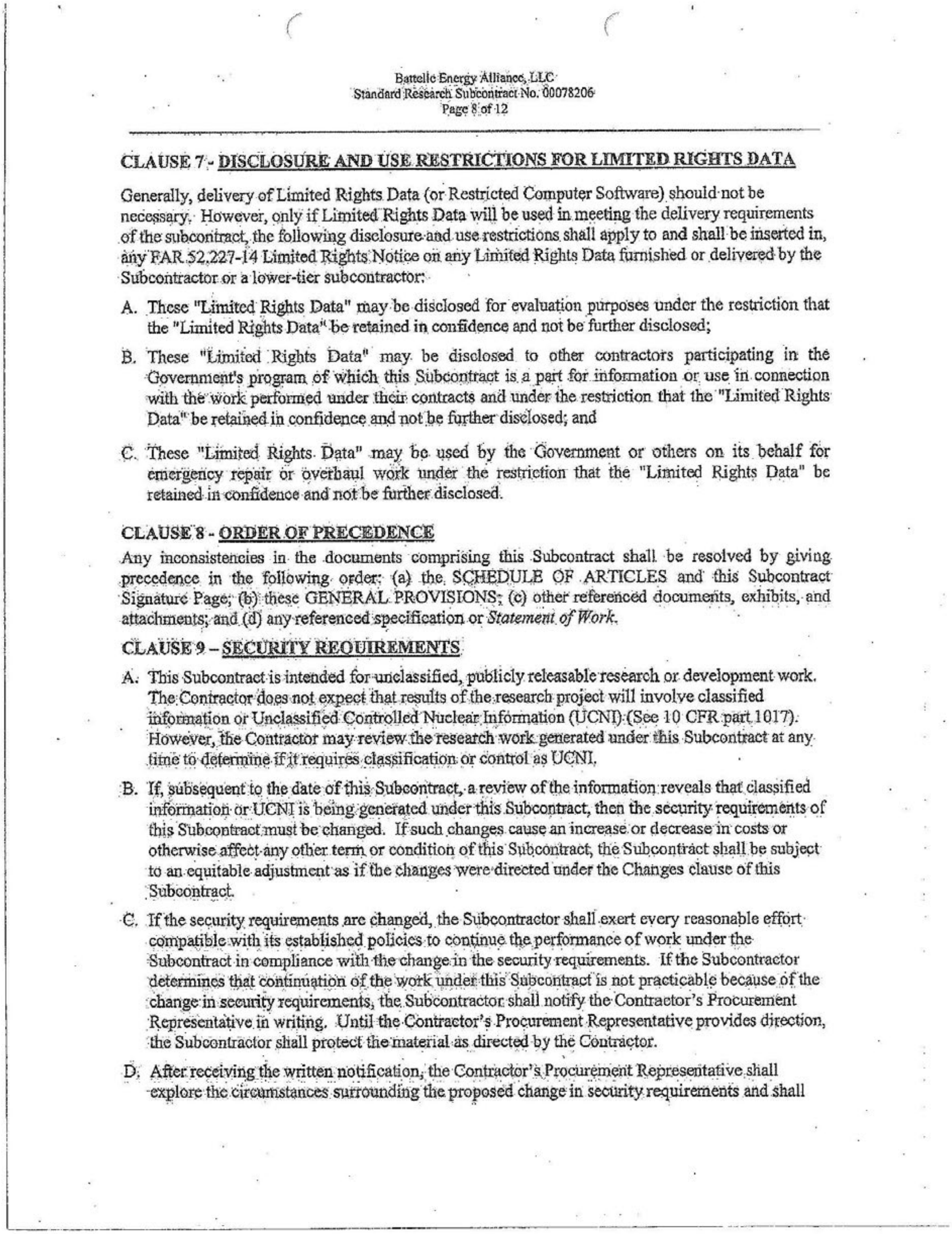
CLAUSE 5 - RESPONSIBILITY FOR TECHNOLOGY EXPORT CONTROL

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*.

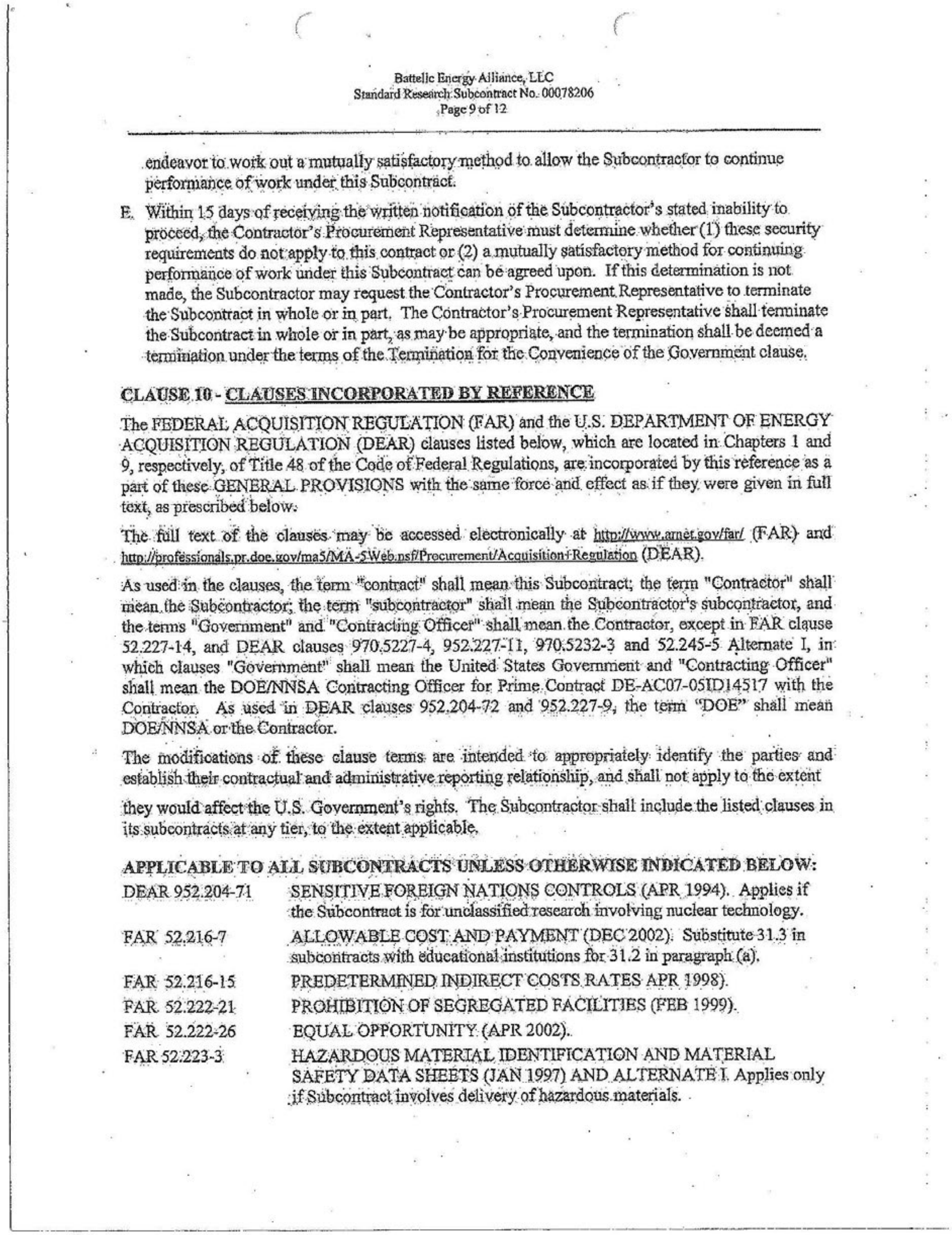
CLAUSE 9 – SECURITY REQUIREMENTS

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 10 - 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.

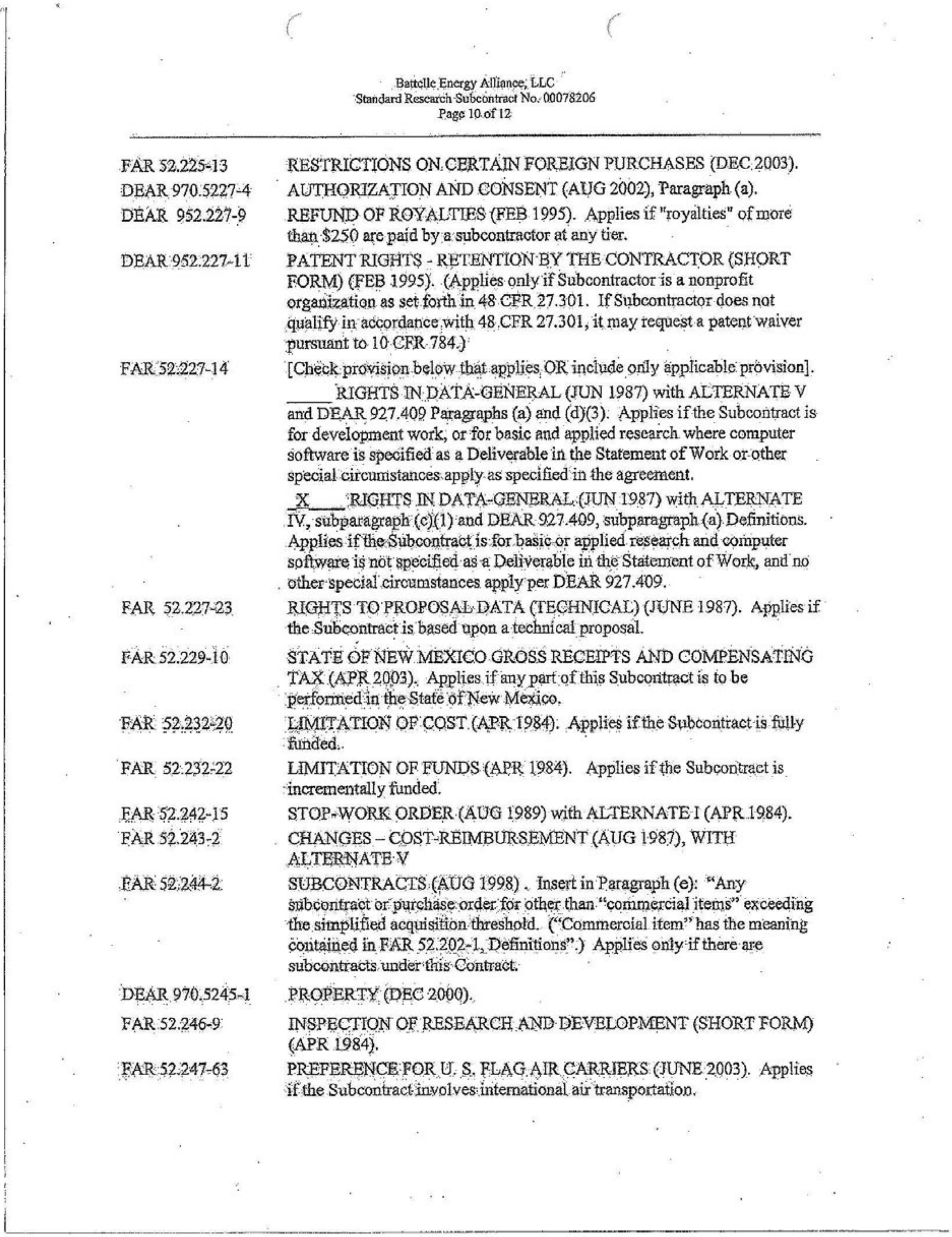
The full text of the clauses may be accessed electronically at http://www.arnet.gov/far/ (FAR) and http://professionals.pr.doe.gov/ma5/MA-5Web.nsf/Procurement/Acquisition+Regulation (DEAR).

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.

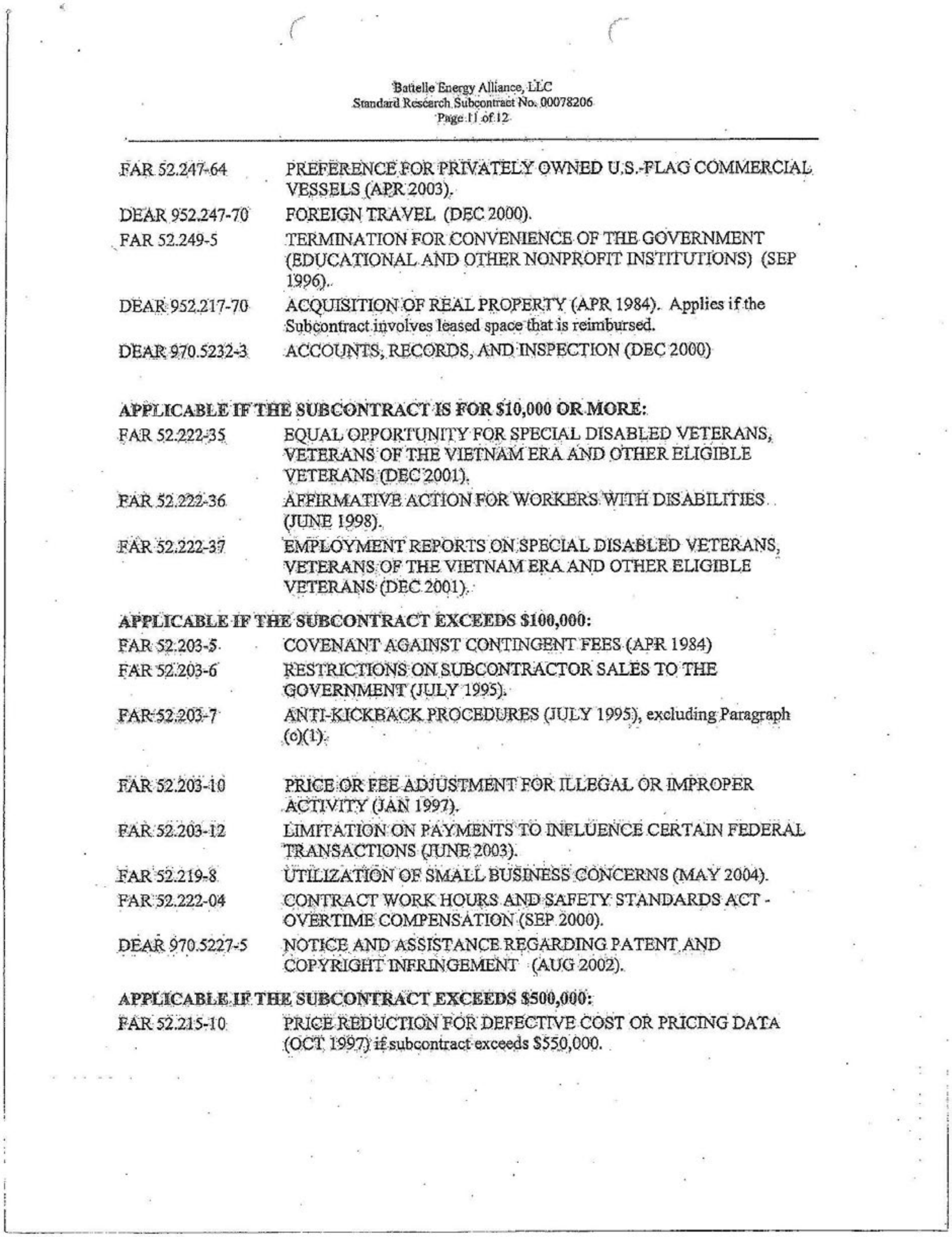
The modifications of these clause terms are intended to appropriately identify the parties and establish their contractual and administrative reporting relationship, and shall not apply to the extent they would affect the U.S. Government's rights. The Subcontractor shall include the listed clauses in 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:

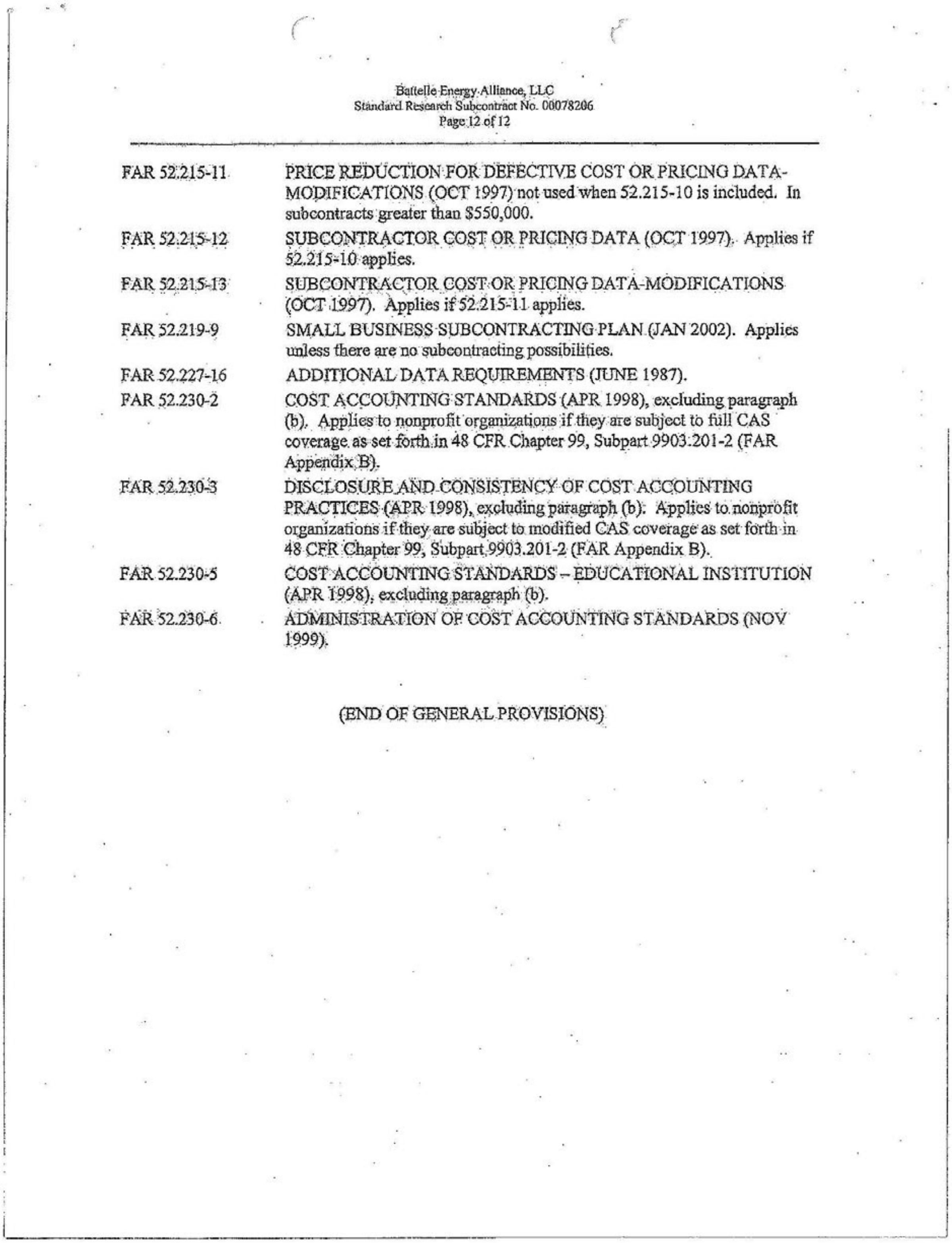
FAR 52.222-35	EQUAL OPPORTUNITY FOR SPECIAL DISABLED VETERANS, VETERANS OF THE VIETNAM ERA AND OTHER ELIGIBLE VETERANS (DEC 2001).
FAR 52.222-36	AFFIRMATIVE ACTION FOR WORKERS WITH DISABILITIES (JUNE 1998).
FAR 52.222-37	EMPLOYMENT REPORTS ON SPECIAL DISABLED VETERANS, VETERANS OF THE VIETNAM ERA AND 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).

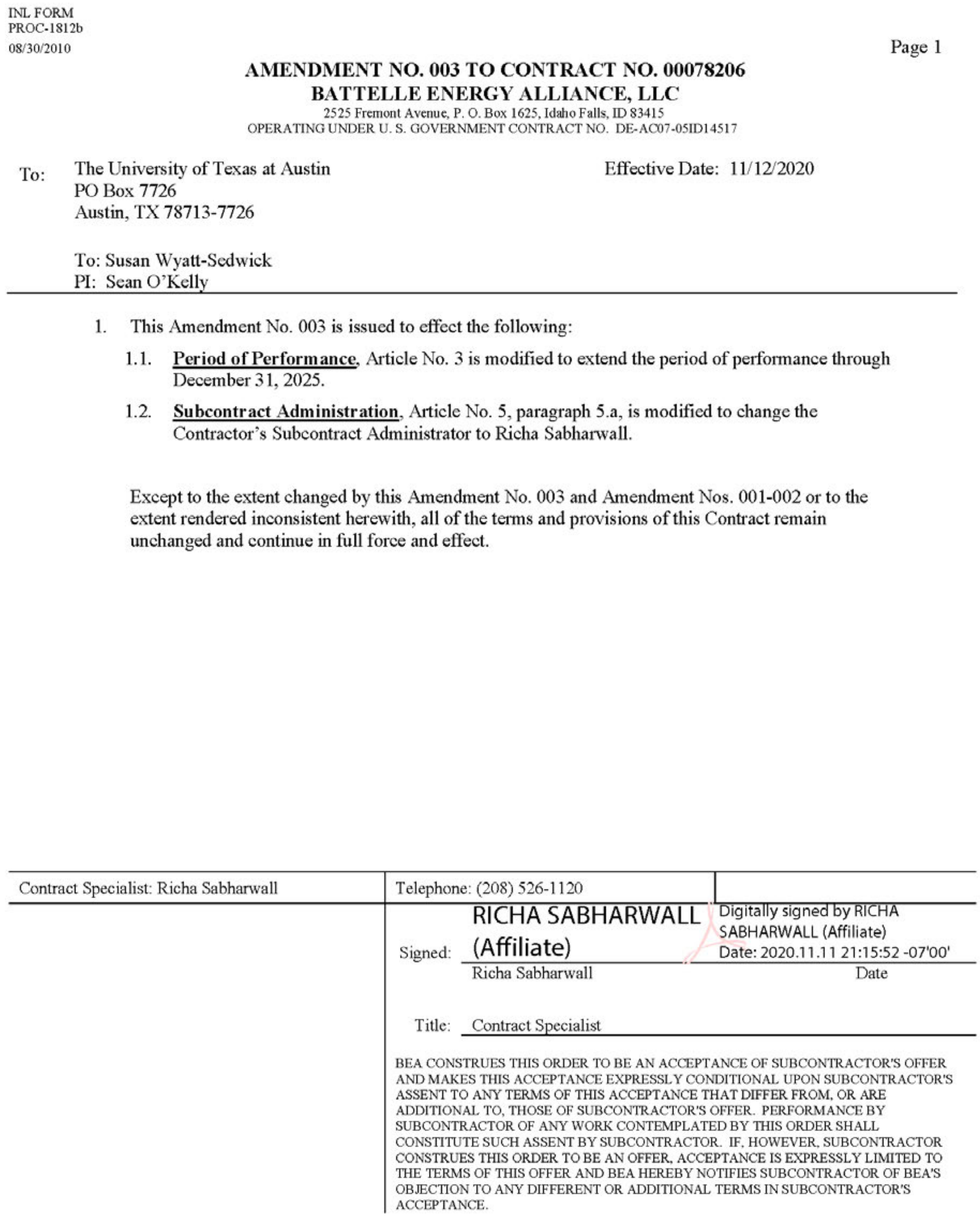
APPLICABLE IF THE SUBCONTRACT EXCEEDS $500,000:

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)



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AMENDMENT NO. 003 TO CONTRACT NO. 00078206
BATTELLE ENERGY ALLIANCE, LLC

2525 Fremont Avenue, P. O. Box 1625, Idaho Falls, ID 83415
OPERATING UNDER U. S. GOVERNMENT CONTRACT NO. DE-AC07-05ID14517

To: The University of Texas at Austin
PO Box 7726
Austin, TX 78713-7726

Effective Date: 11/12/2020

To: Susan Wyatt-Sedwick
PI: Sean O'Kelly

1. This Amendment No. 003 is issued to effect the following:
   1.1. **Period of Performance**, Article No. 3 is modified to extend the period of performance through December 31, 2025.
   1.2. **Subcontract Administration**, Article No. 5, paragraph 5.a, is modified to change the Contractor's Subcontract Administrator to Richa Sabharwall.

Except to the extent changed by this Amendment No. 003 and Amendment Nos. 001-002 or to the extent rendered inconsistent herewith, all of the terms and provisions of this Contract remain unchanged and continue in full force and effect.

Contract Specialist: Richa Sabharwall

Telephone: (208) 526-1120

Signed: RICHA SABHARWALL (Affiliate) Digitally signed by RICHA SABHARWALL (Affiliate) Date: 2020.11.11 21:15:52 -07'00'

Richa Sabharwall Date

Title: Contract Specialist

BEA CONSTRUES THIS ORDER TO BE AN ACCEPTANCE OF SUBCONTRACTOR'S OFFER AND MAKES THIS ACCEPTANCE EXPRESSLY CONDITIONAL UPON SUBCONTRACTOR'S ASSENT TO ANY TERMS OF THIS ACCEPTANCE THAT DIFFER FROM, OR ARE ADDITIONAL TO, THOSE OF SUBCONTRACTOR'S OFFER. PERFORMANCE BY SUBCONTRACTOR OF ANY WORK CONTEMPLATED BY THIS ORDER SHALL CONSTITUTE SUCH ASSENT BY SUBCONTRACTOR. IF, HOWEVER, SUBCONTRACTOR CONSTRUES THIS ORDER TO BE AN OFFER, ACCEPTANCE IS EXPRESSLY LIMITED TO THE TERMS OF THIS OFFER AND BEA HEREBY NOTIFIES SUBCONTRACTOR OF BEA'S OBJECTION TO ANY DIFFERENT OR ADDITIONAL TERMS IN SUBCONTRACTOR'S ACCEPTANCE.