PENNSTATE

Eva J. Pell
Vice President for Research
Dean of the Graduate School
Steimer Professor of Agricultural Sciences

The Pennsylvania State University
304 Old Main
University Park, PA 16802-1504

814-863-9580
Fax: 814-863-9659
E-mail: ejp@psu.edu

December 6, 2005

US Nuclear Regulatory Commission
Attn: Marvin Mendonca
Mail Stop 012-G13
One White Flint North
11555 Rockville North
Rockville, MD 20852-2738

Subj: License Renewal Application for the Penn State Breazeale Reactor (PSBR)
NRC License R-2, Docket 50-005

Dear Sir:

The Pennsylvania State University herewith submits an application for a twenty (20) year renewal of NRC License R-2, a class 104 license, to provide for continued operation of the Penn State Breazeale Reactor (PSBR). Enclosed are three (3) paper copies and one electronic copy (cd). The current NRC License R-2 expires on January 27, 2006.

The staff of the PSBR has prepared this renewal application. The Penn State Reactor Safeguards Committee has reviewed this application.

Documents previously approved by the NRC are referenced with this application but are not being submitted. These include the Physical Security Plan (serial 92-28 dated April 21, 1992), the Emergency Preparedness Plan (September 21, 2000, revision 4), the Operator Requalification Plan (TAC No. M99008, July 9, 1997), and the Technical Specifications (through Amendment #37).

The primary changes in this application from the existing license relate to the demographic, geological, and climatological updates and to reformatting of material to better correspond to NUREG 1537, February 1996.

Chapter 7, Instrumentation and Control Systems information, was reconfigured into the format of NUREG-1537 with a few minor wording changes. Section 7.6.1.1, Computers, reflects updates to console computers in August of 2005.

Chapter 13, Accident Analysis, is unchanged (other than typographic corrections and reconfiguring and renumbering to meet the NUREG format) from the Safety Evaluation last updated on January 14 and April 24 of 1997. Updates to MWhr/wk information were made to

A020
A035
A053

section 13.1.1, Maximum Hypothetical Accident, and to section 13.1.3, Loss of Coolant. No observations of operations or other information warranted changes or updates. The analysis was extremely conservative when submitted and remains so today.

There is one proposed change to the Technical Specifications. In definition 1.1.34.g., the work "containment" should be changed to "confinement". The present wording is that from ANSI/ANS-15.1.-1990. The proposed change reflects the fact that the PSBR has a confinement but not a containment.

It is requested that NRC License R-2 be renewed utilizing the enclosed documents and that these documents supersede the corresponding previous submittals made to the NRC for that license.

This submittal along with the safe and excellent operating history of the PSBR over the past 50 years supports the intent and desire of Penn State to continue the operation of the PSBR. In doing so, we will continue to support the Atoms for Peace initiative espoused by President Eisenhower in 1953. We are committed to "making life better" through the ongoing education, education, and service work of the PSBR.

Questions relating to the application should be directed to:

Dr. C.F. Sears, Director
Radiation Science and Engineering Center (RSEC)
Breazeale Nuclear Reactor
University Park, PA 16802-2301

This license renewal application is full, complete, and correct to the best of our knowledge.

Sincerely,

Eva J. Pell

Eva J. Pell
Vice President for Research and
Dean of the Graduate School

pc: L.C. Burton (w/o)
C.F. Sears (w/3)
T.A. Litzinger (w/o)

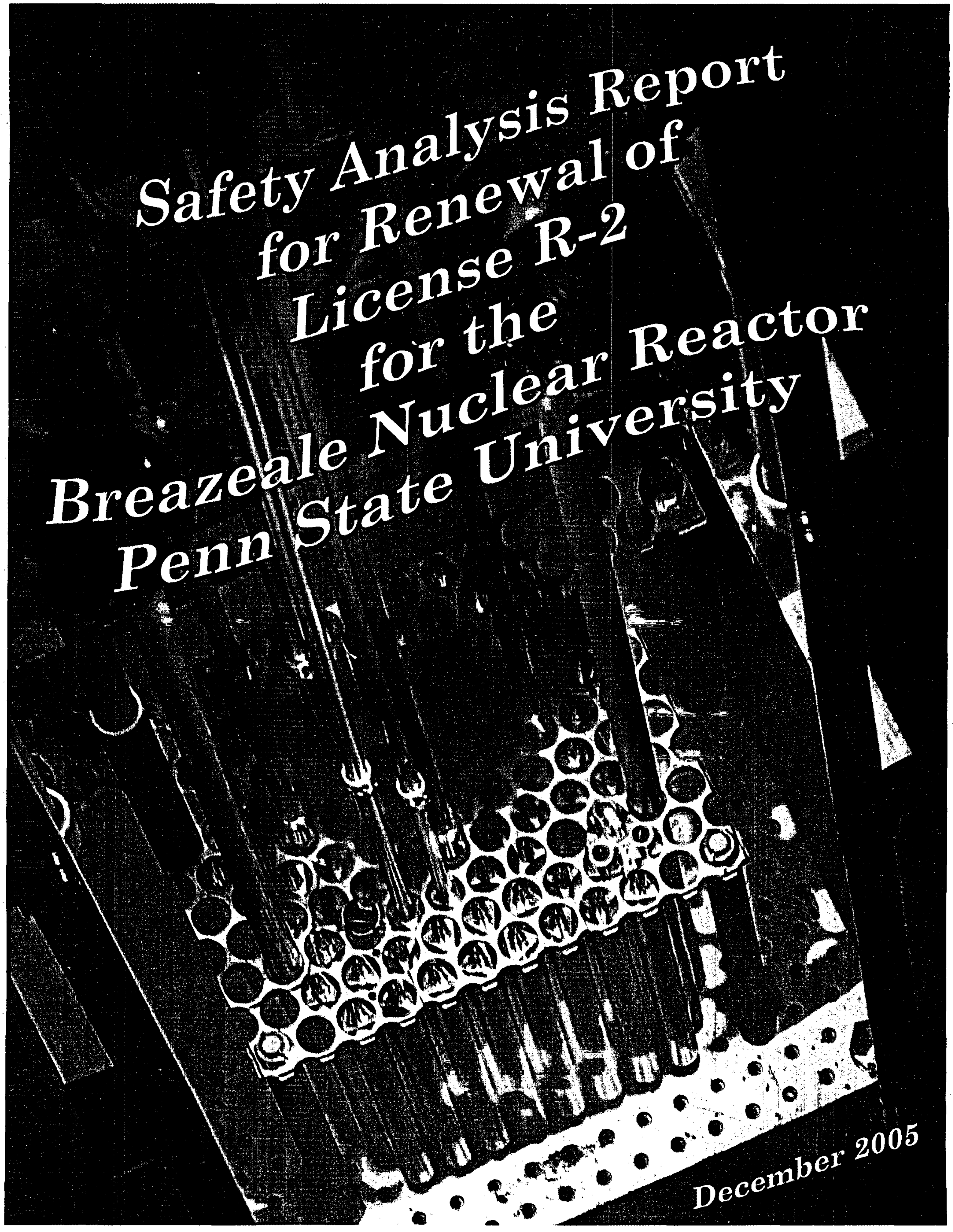
Subscribed and sworn before me on the 6th day of December 2005, Notary Public in and for Center County, Pennsylvania.

Lorri L. Somsky

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Lorri L. Somsky, Notary Public
State College Boro, Centre County
My Commission Expires Nov. 15, 2008
Member, Pennsylvania Association of Notaries

Safety Analysis Report for Renewal of License R-2 for the Breazeale Nuclear Reactor Penn State University

December 2005



SAFETY ANALYSIS REPORT

FOR RENEWAL OF LICENSE R-2

FOR THE

BREAZEALE NUCLEAR REACTOR

PENN STATE UNIVERSITY

DECEMBER 2005

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1. THE FACILITY

1.1 Introduction

The present R-2 operating license for the Penn State Breazeale Nuclear Reactor (PSBR) expires 27 January 2006. This document is being submitted to the Nuclear Regulatory Commission (NRC) to request a renewal of the R-2 operating license (docket 50-005) for another 20 years. The license is a Class 104 as defined in 10 CFR 50.22. The facility is considered a Class II for routine NRC inspection purposes. The fuel enrichment of slightly less than 20%, places the reactor in the low enriched uranium safeguards category.

The reactor is located on the main campus of The Pennsylvania State University, University Park, Centre County, Pennsylvania. The University Park Campus is adjacent to the town of State College, Pennsylvania.

Administratively, the Radiation Science and Engineering Center (RSEC) of which the PSBR is a part, answers to the Dean of the College of Engineering. For Nuclear Regulatory Commission licensing matters, the reporting chain continues up to the Vice President for Research and Dean of the Graduate School, who has legal responsibility for the R-2 license.

The reactor features a General Atomics TRIGA reactor core with the inherent safety of a large prompt negative temperature coefficient. The reactor is licensed for 1 MW(th) steady state operation and is capable of pulsing operation to ~2000 MW(th). The reactor safety system (RSS) and the reactor protection, control and monitoring system (PCMS) is by Atomic Energy of Canada Ltd. (AECL)/Gamma-Metrics. The reactor is located in a large 71,000 gallon (2.68×10^5 ℓ) concrete pool with a partial pool divider; a movable gate provides for isolation of the reactor in either end of the pool. The reactor bridge provides the unique characteristic of reactor core rotation and the bridge is movable north-south and east-west to various experimental locations in the reactor pool.

Chapter-13, Accident Analyses, of this document demonstrates that the design and operation of this reactor presents an extremely low probability for the release of any significant amounts of radioactive fission products. In all cases, any releases would be within the limits of 10 CFR 20 for any exposures to the public.

The Penn State reactor has been used since 1955 for a myriad of educational and research activities and continues presently to be used extensively to

- educate students in nuclear engineering and other fields
- conduct research in a wide variety of scientific and engineering fields, including neutron activation analysis, neutron transmission measurements of borated materials, neutron radiography, live-time neutron radioscopy, and neutron depth profile measurements
- educate the public regarding the peaceful uses of nuclear technology
- provide irradiation services to university, industrial, and government entities

1.2 Summary and Conclusions on Principal Safety Considerations

The "inherent safety" that distinguishes TRIGA reactors is made possible by the fuel itself, which has a unique built-in power regulator. In the TRIGA, a zirconium-hydride moderator is used with the uranium fuel to form a solid metal alloy fuel. Since the two components are alloyed, any sudden increase in power heats the fuel and moderator simultaneously. The higher temperature instantly makes the moderator less effective thus contributing a negative temperature coefficient. Since the feedback can occur within a few thousandths of a second, the effect is known as a "prompt negative temperature coefficient" and allows for an operation called pulsing the reactor. During a pulse, a control rod is ejected from a critical core (<1 kW(th)) allowing the reactor power to go safely to powers up to ~2000 MW(th), although for only milliseconds of time because of the prompt negative temperature coefficient.

Three accidents are examined in Chapter 13, Accident Analyses (other than typographic corrections and reconfiguring to meet NUREG-1537 format, the analyses are unchanged from the last updates on January 14 and April 24 of 1997). They are a Loss of Coolant Accident (LOCA), a Maximum Hypothetical Accident (MHA), and a Reactivity Accident. The results of the analyses for these three accident scenarios demonstrated that the reactor can be operated safely within the bounds of the safety analysis and regulatory limits. No observations of operations or other information during the present license period warranted changes or updates. The analyses were extremely conservative when submitted and remain so today.

The most conceivable LOCA is [redacted]

It takes only three minutes for the fuel to get within 20°C of the water temperature following a shutdown from 1 MW(th). [redacted] The analysis assumes the reactor was operating continuously for 168 hours when the LOCA occurs. The highest temperature postulated [redacted] is 468°C. [redacted], the temperature would peak at 480°C, [redacted] after the LOCA begins. In either case, the temperatures reached are well below the 950°C temperature where clad failure could occur for an air cooled element when the fuel and cladding are the same temperature.

The MHA assumes a single air cooled TRIGA fuel element ruptures. The mechanism for this to happen is not described. Prior to the rupture, it is assumed the reactor core has been operated continuously at 1 MW(th) throughout its life. Other assumptions are a maximum sustained measured steady state fuel element temperature of 650°C and a Maximum Element Power Density (MEPD) of 24.7 kW(th) (both of these values are the TS limits). A free bay volume of 1900 cubic meters is assumed, also a TS minimum value. The fraction of the fission product inventory in the fuel element that escapes, is called the release fraction. For the accident, the release fraction is [redacted] The only fission products that escape are those that have diffused into the gap between the fuel and cladding during the operation of the reactor. Therefore, the release fraction in accident conditions is characteristic of the sustained operating temperature and not the temperature at the time of the accident. There is an insignificant effect on the environment or the health and safety of the public from this accident. The emergency exhaust system (EES) would reduce any exposures to the public [redacted] of the Total Effective Dose Equivalent (TEDE) limit of 100 mrem for the public (with neither internal or external exposure exceeding 50 mRem). Even if the EES fails to work properly, the public exposures are still [redacted] of the 100 mRem limit allowed to the public.

The Reactivity Accident assumes the reactor is taken to 1.15 MW(th) with the transient rod inserted. Then the transient rod is used to make a $2.25 insertion. It is assumed that $4.75 of the $7.00 allowed excess is used to get to 1.15 MW(th), so only $2.25 remains for the accident insertion. The accident requires a breakdown in the adherence to PSBR operating procedures, and failures of the overpower scrams and interlocks. Should a $2.25 pulse occur while the reactor is at 1.15 MW(th), the measured fuel temperature will rise from 650°C to 1030°C (based on temperatures for core loading 47). The temperature increase from the pulse is added on top of the steady state fuel temperature. The 1030°C is below the 1150 °C Safety Limit.

1.3 General Description

The reactor is housed in the Breazeale Nuclear Reactor Building on the University Park Campus (Main Campus) of The Pennsylvania State University (see **Chapter-2 maps**). The original portion of the building (highlighted in red in Figures 1-1 and 1-2), which contains the reactor, reactor pool, the reactor pool water handling systems, and neutron beam laboratory was constructed in 1954 (at that time the control console was on the reactor bridge). In 1962, an addition to the building added offices, a future control room, laboratories, a classroom, and two hot cells. A 1967 addition added a cobalt-60 pool irradiator and adjoining laboratory space. A machine shop expansion was completed in 1999 and a lobby expansion was completed in 2004. The building is constructed of concrete blocks, bricks, insulated steel and aluminum panels, structural steel, and re-enforced concrete and is in general fireproof in nature.

The entire building is constructed on two levels with the neutron beam laboratory [redacted] and the mechanical equipment room [redacted] being a lower level than the remainder of the ground floor. The reactor is housed in the pool in the reactor bay [redacted] located in the central portion of the building. The reactor control room is located in [redacted] adjacent to the reactor bay. Major components of the pool recirculation system and heat exchanger system (demineralizer, cuno filters, heat exchangers, and associated pumps) are located in [redacted] The pool water transfer equipment (pump and valves) [redacted] Except for the hot cells [redacted] and the cobalt-60 facility [redacted]), the remainder of the building is utilized as research laboratories, teaching laboratories, and faculty and staff offices.

The 1 MW(th) steady-state (~ 2000 MW(th) pulse) TRIGA has the "inherent safety" that distinguishes TRIGA reactors, made possible by the prompt negative temperature coefficient of the reactor fuel. The reactor confinement is by a reactor bay that except for ventilation design, is an ordinary industrial building. During normal operations, a negative pressure is maintained in the reactor bay by operating one or both of the reactor bay facility exhaust system (FES) fans. For any condition when a building evacuation alarm sounds, the FES is secured and an emergency exhaust system (EES) comes on and ventilates the bay through roughing, absolute, and charcoal filters.

The reactor fuel grid plates are supported by a suspension tower that can be moved east-west on the reactor bridge. The reactor tower can also be rotated. The reactor bridge is movable north and south on rails mounted atop the pool walls. The reactor core typically has 90 to 110 TRIGA fuel elements and four reactor control rods. Three control rods are held by electromagnets and the fourth is held by compressed air. Interrupting magnet current and the air supply provides quick reactor shutdown due to gravity. The facility has two air compressors available to supply air to the control rod and for other facility needs.

The analog portion (RSS) of the AECL/Gamma-Metrics analog-digital control system meets the TS interlock and scram requirements. The digital portion (PCMS) of the system provides for reactor control. Safety parameter monitoring inputs to the control system are by a 10-decade range fission detector, a gamma ionization detector, and an instrumented fuel element to monitor fuel temperature.

Radiation protection is by area radiation monitors, air particulate monitors, and pool level monitors. [redacted] these alarms are monitored 24 hours a day [redacted]. The building is also served by a smoke and fire alarm system and emergency lighting system.

The reactor control system and radiation protection systems operate off of an Uninterruptable Power Supply (UPS). However, there is no safety related equipment that is required to be operable by the TS, when the reactor is "secured" (TS definition).

The 71,000 gallons (2.68×10^5 ℓ) of reactor pool water are filtered and demineralized by way of the pool recirculation loop. Low pool water conductivity and slightly acidic pH provide conditions favorable to the stainless steel fuel and aluminum reactor components. The high quality water minimizes the gross radioactivity of the water. The 18 feet (5.49 m) of water above the reactor core serves as a biological shield and heat sink. Since the Zr-H in the reactor fuel provides neutron moderation, the water's moderation of neutrons is of secondary significance, but the water is an effective neutron reflector.

The 1 MW(th) reactor is cooled by natural convection in the 71,000 gallon (2.68×10^5ℓ) reactor pool. An underwater diffuser pump, called the N-16 pump, diffuses the normal convective flow to the pool surface, thus mitigating radiation levels at the pool surface due to the Nitrogen-16 produced from the reactor's operation.

Full power operation can increase the bulk pool temperature by several °C in an hour if no attempt is made to cool the pool water. To limit pool evaporative losses and protect the demineralizer resins from temperature damage, a heat exchanger is available. The heat exchanger 's primary loop pumps pool water through the baffled shell side of two heat exchangers connected in series. The secondary loop passes water from a nearby spring fed pond through the tube bundle side of the two heat exchangers. Secondary pressure is always maintained higher than primary pressure for environmental protection.

Most of the radioactivity produced by the operation of the reactor is the fission products maintained by the integrity of the fuel cladding. In addition, demineralizer spent resins and normal radioactive laboratory waste, are handled by the university's Radiation Protection Office (RPO) as part of the normal campus radioactive waste stream. An evaporator system is available for treatment of liquid radioactive waste, if needed. An ALARA, As-Low-As-Reasonably-Achievable, program is in effect at the reactor.

Major experimental facilities consist of a central thimble, a pneumatic transfer system, two fast neutron irradiators, dry irradiation tubes, and neutron beam port facilities. The central thimble is a hollow tube, at the core center, whose outside diameter is the same as a fuel element. The central thimble is the point of maximum neutron fluence in the reactor. The pneumatic transfer system allows samples to be sent from a laboratory directly into the reactor core. Two fast neutron irradiators in the north end of the reactor pool are designed to minimize gamma and thermal neutron contribution. Various size vertical dry irradiation tubes are available. Seven beam ports penetrate the reactor pool wall on its south side. Currently, two of the beam ports can be coupled to the reactor through collimator tubes and a heavy water tank.

1.4 Shared Facilities and Equipment

The RSEC includes the reactor bay and control room, laboratories and classrooms, hot cells, and a cobalt-60 pool irradiator (see **Figures 1-1 and 1-2**). The building is located inside a chain-link fence and all activities conducted within this site are under the control of the facility Director (see **Figure 2-3**). Activities in addition to those under the R-2 license, are conducted in the reactor bay and parts of the reactor facility adjacent to the reactor bay under other university licenses with the NRC.

The reactor has its own dedicated water purification system and cooling system. Some water handling facilities within the building are shared; for example, a storage tank can be used for either the reactor pool water or the cobalt-60 facility pool water. However, all procedures for associated facilities have the same review and approval administrative chain as for the reactor facility, so it is assured that activities conducted under other licenses do not compromise the reactor R-2 license.

The reactor bay has its own dedicated heating and ventilation system. The ventilation is by either the FES or the EES. These two exhaust systems minimize the possibility of the movement of radioactive material from the reactor bay into any adjacent areas within the reactor building.

Building access control is by key locks and administrative procedures, so that persons having general access to the building do not automatically have access to areas covered by the R-2 or other licenses.

The building water, steam, and electricity are furnished through the campus wide systems. Some of the effluent from the secondary side of the reactor heat exchanger is used by the Combustion Engineering Laboratory in an adjacent building but that use has no safety impact on the reactor.

C01

[redacted]

C02

[redacted]

1.5 Comparison with Similar Facilities

The first TRIGA (Training-Research-Isotopes-General Atomic) reactor went into operation at the General Atomic (GA) Laboratories in San Diego, California on May 3, 1958. Since that time over 60 TRIGA reactors have been built and safely operated in over 20 countries. The Penn State installation in1965, was the first MTR to TRIGA conversion by GA.

Unlike most other TRIGA reactors that use 8.5 wt% fuel, the Penn State core is a mixture of 8.5 wt% and 12 wt% fuel. The Penn State core has a hexagonal fuel arrangement rather than the more common circular TRIGA core. The location in a large pool is different than the more common location in a tank. The flexibility of core rotation and lateral movements in four directions is a feature unique to the PSBR. The 1965 vintage standard General Atomics TRIGA control system was replaced with a AECL/Gamma-Metrics analog and digital control system in 1991 (with a further upgrade in 2004 but with no significant changes in functional performance of the systems).

1.6 Summary of Operations

The reactor normally operates on a 40 hour a week schedule with occasional extended evening and weekend hours. The trend in recent years has been toward a higher level of use of the reactor and associated facilities. It is anticipated that the trend will continue. **Table 1-1** below gives some statistical information. However, the assumptions for the MHA in section 13.1.1 (Accident Analysis) are not challenged by increased operations, since the MHA assumes continuous reactor operation prior to the postulated event.

Table 1-1 Reactor Operating Statistics

	04/05	3 yr Avg	5 yr Avg	10 yr Avg
Megawatt Hrs	657	617	594	451
Grams U-235 Consumed	34	32	31	23
Hours of Use	2152	2016	1971	1866
No of Pulses	152	131	124	95
Reactor Startups	1278	1164	1122	--
Reactor Power Changes	731	549	459	--

1.7 Compliance with the Nuclear Waste Policy Act of 1982

In accordance with a letter from the U. S. Department of Energy (R. L. Morgan) to the U. S. Nuclear Regulatory Commission (H. Denton) dated May 3, 1983, it has been determined that all universities operating non-power reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and DOE is obligated to take the spent fuel and/or high level waste for storage or reprocessing. Because the Pennsylvania State University has entered into such a contract with DOE, the applicable requirements of the Nuclear Policy Waste Act of 1982 have been satisfied. The contract between Penn State and Bechtel BWXT Idaho, LLC (BBWI) is number 00036822 (operating under U. S. Government Contract No. DE-AC07-99ID13727).

1.8 Facility Modifications and History

On 15 January 1954, The Pennsylvania State University's Board of Trustees announced that the construction of a nuclear reactor had been authorized. William M. Breazeale, who was a faculty member from 1953 to 1958 and the first Penn State Professor of Nuclear Engineering, designed the reactor facility. He was the facility's first director. On 8 July 1955, the US Atomic Energy Commission issued license number R-2 for the operation of the Penn State MTR Reactor at power levels up to 100 KW(th). Criticality of the reactor was reached on 15 August 1955.

Penn State University (PSU) (R-2) and North Carolina State University (NCSU) entered into a cooperative agreement with Argonne National Laboratory in April 1956 to establish the International School of Nuclear Science and Engineering. It was now possible for scientists from other countries to spend a semester at either PSU or NCSU in resident training and then go to Argonne for four more months of additional training. It was a natural extension to turn the material developed for this international training program into an academic program for nuclear engineers. On 16 June 1959, a Department of Nuclear Engineering was formed at Penn State.

Because of ever increasing experimental and educational demands, a maximum operating power level increase to 200 kW(th) was authorized on 13 May 1960 (license amendment #6). Also, in 1960, the General State Authority authorized funds for the construction of additions to the Nuclear Reactor Facility consisting of a Hot Laboratory containing two hot cells, additional research floor space around the reactor, added general office and classroom space, and a sizeable radiochemistry/chemical engineering wing. Full occupancy of the new facilities occurred in the Spring of 1964. By that time plans were also underway for the addition of a Co-60 pool irradiation facility. The Co-60 facility was occupied in January of 1967. A machine shop expansion was completed in 1999, and a lobby expansion was completed in 2004.

As heavy use of the reactor continued through the early 1960's, it became apparent that the 200 KW(th) MTR reactor was not adequate to handle the increased education and research demands. A General Atomics TRIGA reactor core and control system was installed in late 1965 (license amendment # 16, 30 December 1965) and the reactor was taken critical on 31 December 1965. The TRIGA is designed for 1 MW(th) steady state operation. Because of a very prompt negative temperature coefficient associated with the ZrH fuel-moderator, the reactor can be safely pulsed to ~2000 MW(th) (~15 to 20 msec full width at half maximum).

The original TRIGA core used 8.5 wt% fuel exclusively. Since 1972, the core has been a mixture of 8.5 wt% and 12 wt% fuel (change #5 to the TS on 14 April 1972 occurred between amendment #17 and #18). All of the fuel is enriched to slightly less than 20 % Uranium-235. The Penn State TRIGA core uses a hexagonal fuel arrangement rather than the normal circular TRIGA core. This was done to maintain a flat core front and back, for better coupling when operating against a graphite thermal column then in use. The graphite thermal column was replaced with a D_2O tank in 1971. A second generation D_2O tank was installed in 1997. In 1994 the reactor bridge was modified to allow for north-south, east-west, and rotational movement of the reactor core. Previous to that, only north-south movement was possible.

During August of 1991, the General Atomics control console was replaced with a new analog-digital control system developed by AECL as the primary contractor, with Gamma-Metrics a subcontractor for the hardware (license amendment #30, August 23, 1991). Two independent systems provide the safety, protection, control, and monitoring functions. The Reactor Safety System (RSS) is a hard-wired analog system that provides all of the SCRAM and operational interlock functions required by the TS. The Protection, Control, and Monitoring System (PCMS) is fully computerized. The PCMS provides scrams and interlocks redundant to the RSS, and provides live time and historical trending of operational parameters and data. The software and hardware were further upgraded in August of 2005 (there were no significant changes in functional performance of the systems and no license amendments were needed).

The Penn State Breazeale Reactor received an American Nuclear Society Historic Landmark Designation in 1991 and celebrated its fiftieth anniversary on August 15, 2005.

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The Penn State Breazeale Reactor (PSBR), a 1 MW(th) TRIGA reactor, is located on the University Park Campus of the Pennsylvania State University in central Pennsylvania in the county of Centre (see **Figure 2-1 insert**). The reactor is located at N40°48′15″and W77°51′15′. The campus is located in Nittany Valley among Bald Eagle Mountain, Nittany Mountain, and Tussey Mountain (see **Figure 2-2**). The Nittany Valley is located in an area called the Valley and Ridge Province. This province is characterized by broad limestone valleys, interrupted by steep, forested, sandstone ridges.

The campus is bordered by State College Borough, College Township, Ferguson Township, Patton Township, and Benner Township. Much of the campus is bordered by commercial and residential areas, with farmland to the north of campus (see **Figure 2-1**). Harris Township does not directly border the campus but a portion of it (see Boalsburg on **Figure 2-4**) falls within the 8 kilometer zone.

There are no prominent rivers or lakes in the vicinity of the reactor. Centre County streams are shown on **Figure 2-2.** Shingletown Reservoir is approximately 4 miles (6.4 km) southeast of the PSBR on Tussey Mountain (see **Figure 2-2**). This reservoir supplements the capacity of the State College Borough Water Authority, whose main water supply is from seven wellfields at various locations within the 5 mile (8 km) zone indicated in **Figure 2-4**. Penn State also has wellfields located within the 5 mile (8 km) zone.

Area highways are shown in **Figure 2-1**. All roads on the University Park Campus are under the control of the University Police. U.S. Routes 220 and 322 are located as close as 1.5 miles (2.41km) to the PSBR. Route 220 is built to interstate standards and will become I-99 when a missing ~10 mile (~16.1 km) section is completed (estimated by the end of 2007). I-99 will connect I-76 (Pennsylvania Turnpike) at Bedford to I-80 at Milesburg.

2.1.1.2 Boundary and Zone Area Maps

The operations boundary is the reactor building. The reactor site boundary is the chain-link fence surrounding the PSBR as shown in **Figure 2-3**.

Any airborne effluent from the reactor bay is exhausted ~24 feet above ground level. There are no buildings high enough or close enough that could affect diffusion and dispersion of airborne effluents (see **Figure 2-3**).

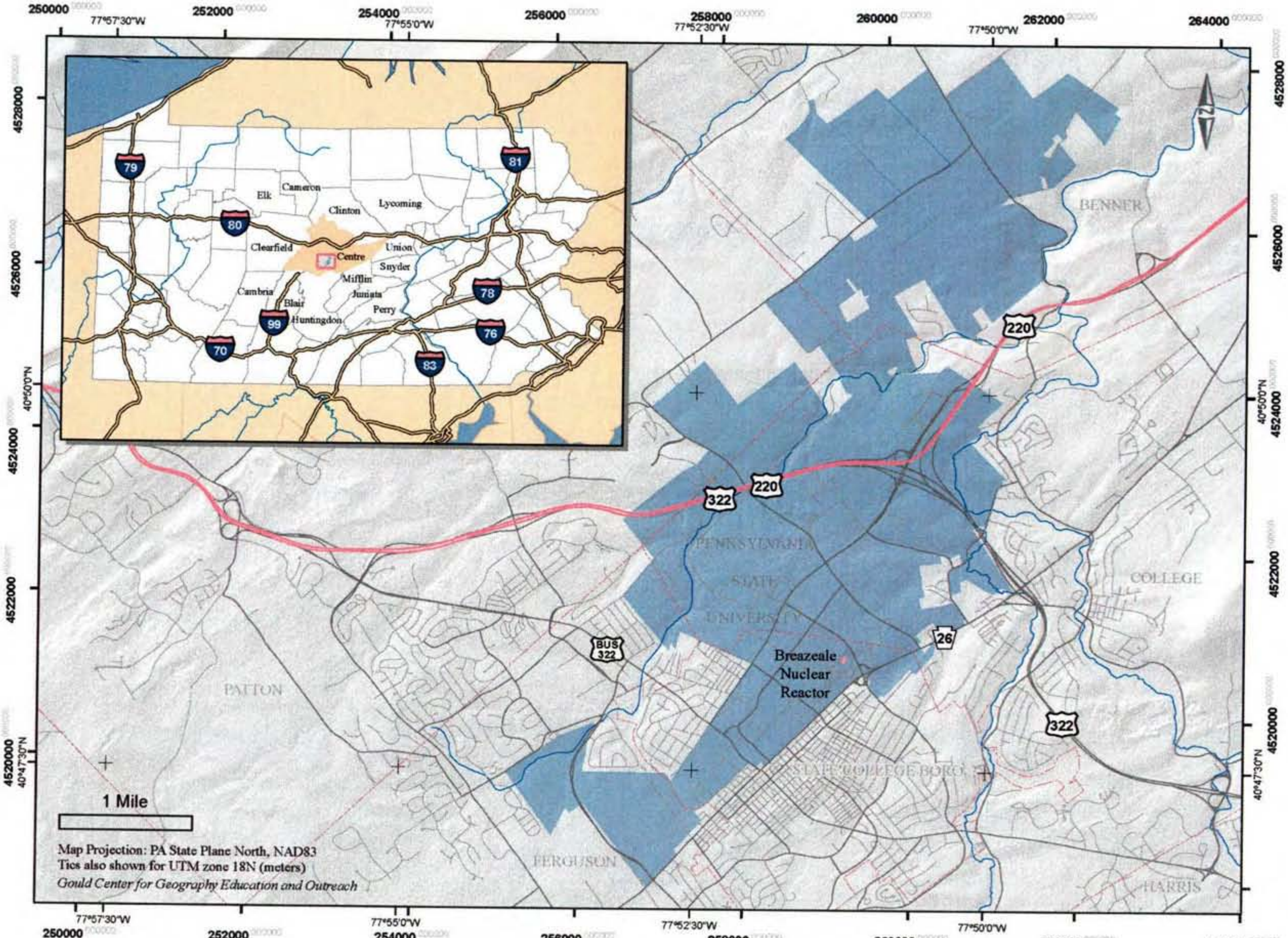

Figure 2-1 PSBR Site Location

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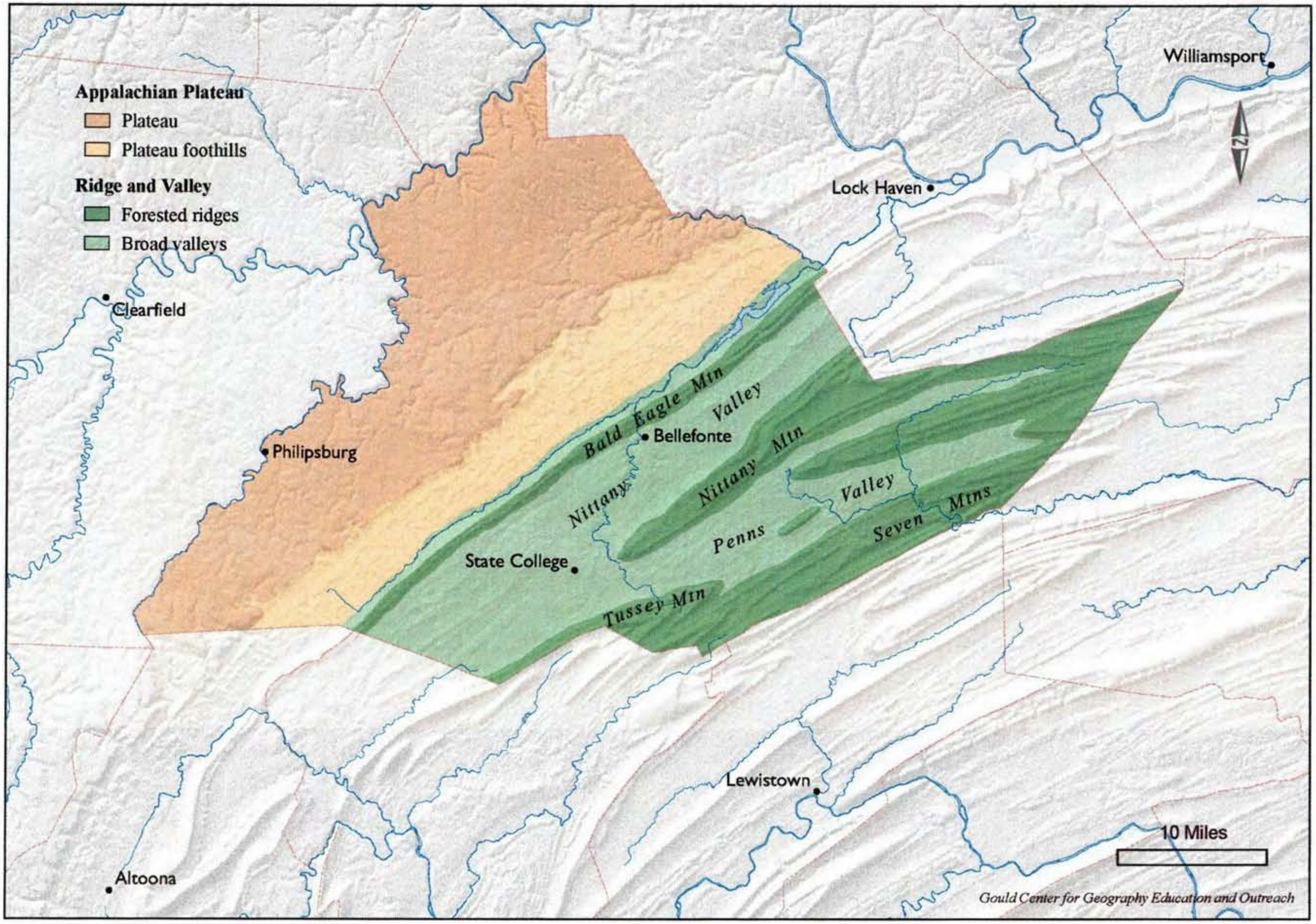

Figure 2-2 Physiographic Provinces, Centre County

C04

C05

[redacted]

The land adjacent to all sides of the reactor site boundary is owned and controlled by the University. The only entrance into the site is through the chain-link locked gate at the southwest end of the site.

2.1.2 Population Distribution

Centre County has a population of 135,758 (Census 2000). The region comprised of Patton Township, Ferguson Township, College Township, Harris Township, and State College Borough has Census 2000 populations of 11,420, 14,063, 8489, 4657 and 38,420 residents, respectively, for a total of 77,409. The Centre Region Planning Agency projects the 2030 population to be 99,106 for the above five areas, about a 28% increase over the 30 year period. Only a 7% population increase is projected for State College Borough since that area is almost completely developed, so most new population growth would be further from the PSBR. The university has stated it intends to keep the University Park campus student population fairly constant at 42,000 students (during the fall and spring semesters, with far fewer at other times).

The great majority of the population of the borough and the four townships mentioned above is within a 5 mile (8 km) radius of the reactor building. Harris Township doesn't directly border the campus, but a large portion of its population (Boalsburg) falls within the 5 mile (8km) zone. Benner Township (2000 census population of 5217) borders the University Park Campus and a small portion of it falls within the 5 mile (8km) zone, with most of its population further northeast toward the town of Bellefonte.

The density of population distribution for the campus, borough, and townships within the zones are given below and shown in **Figure 2-4**. A uniform population density calculation was used to determine population within zones. The source was tract data from Census 2000. Using the reactor as the center point, and concentric circles at distances of 1, 2, 4, 6, and 8 kilometers the following estimated populations are 16191, 30333, 56435, 68480, and 75142, respectively.

The nearest campus residential area (Eastview Terrace Dorms) is located approximately 140 yards (128 m) from the reactor building (see **Figure 2-3**). The nearest State College Borough residential area is located approximately 390 yards (357 m) from the reactor building. A child care center is located 170 yards (155 m) from the reactor building.

2.2 Nearby Industrial, Transportation, and Military Facilities

No major military facilities exist in the University Park or State College areas. A small Army Reserve center is located 1.5 miles (2.4 km) north-northwest from the facility.

The major employers in the area are Penn State University, state and local governments, school districts, health providers, light manufacturers, retail, and research and development firms. In general, it is not an area of heavy industry.

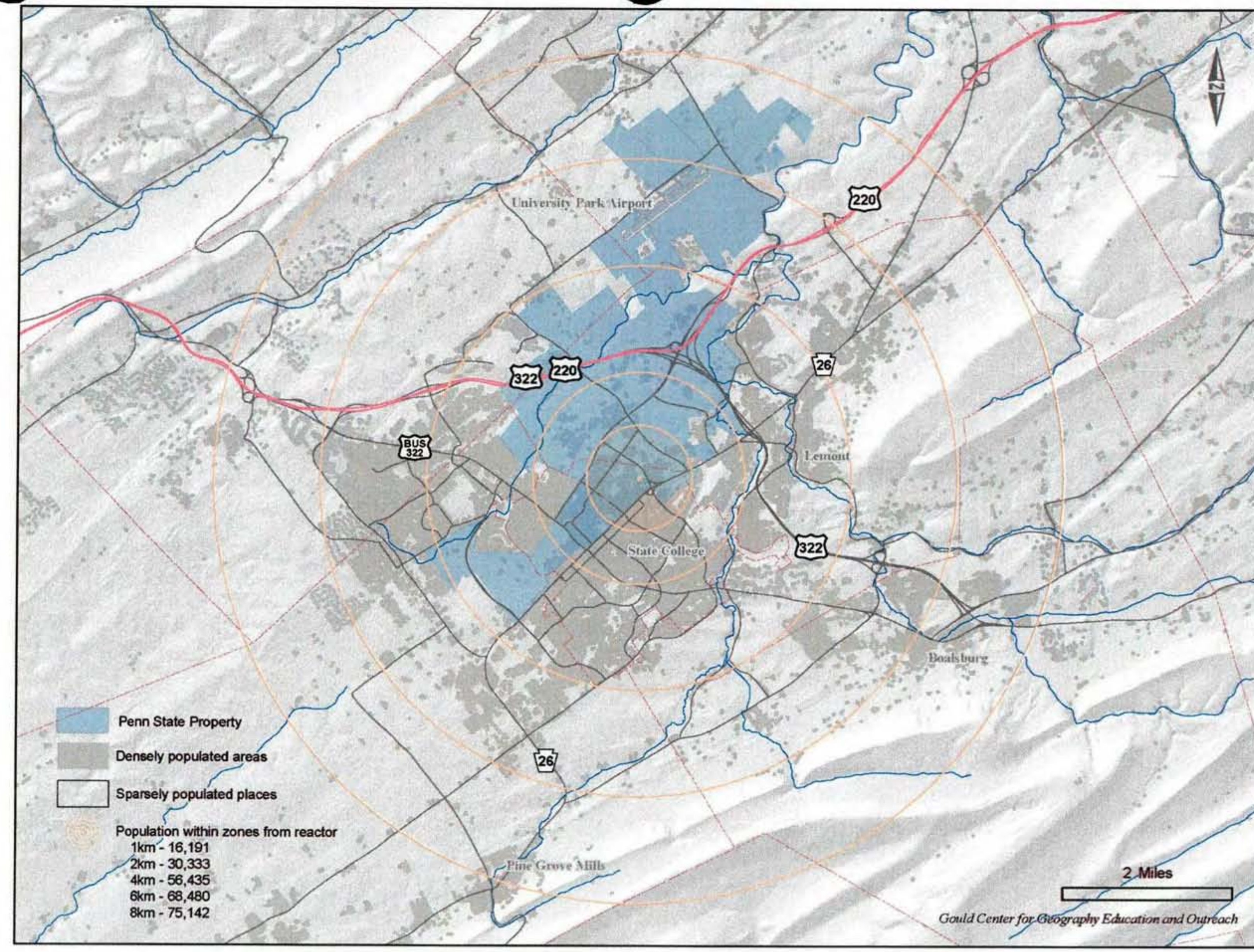

Figure 2-4 Population Distribution

c06

2.2.1 Locations and Routes

There is no rail service to University Park or State College. Area highways are discussed in section 2.1.1.1 and shown in **Figure 2-1**.

2.2.2 Air Traffic

University Park Airport is located about 2.8 miles (4.5 km) north of the reactor facility (see **Figure 2-4**). This airport serves private and commercial aircraft. The PSBR is not in the normal flight path for the airport.

2.2.3 Analysis of Potential Accidents at Facilities

There are no identified industrial, transportation, or military activities that could impact the safe operation of the PSBR.

2.3 Meteorology

2.3.1 General and Local Climate

The Pennsylvania State University is located in central Pennsylvania about 250 miles (402 kilometers) west of New York City. Severe thunderstorms can occasionally occur in the spring and summer, but at a much lower frequency than in the lower Midwest, Gulf region and Southeast. Snowfall is common in the winter, since Penn State lies in the path of many storm systems that come from a western quadrant. Systems that move north along the East Coast also have the potential to produce snow across central Pennsylvania. Lake effect snow can occur when winds turn from the northwest, although lake effect accumulations tend to be minor because of the distance from Lake Erie (~200 miles or 322 kilometers). Most snowstorms produce light to moderate accumulations, (2-6 in or ~ 5-15 cm) although heavy snowfall can occur in series. Rainfall is heaviest during the spring and summer and flooding is sometimes a problem but not in the vicinity of the reactor. Freezing rain, sleet, and ice storms also have the potential for occurring from November to March.

Table 2.3-1 lists mean and extreme rainfall data for Penn State (PSU Weather Station). Between 1888 and 2004, the maximum 24-hour rainfall at Penn State was 5.05 inches (12.83 cm) on September 17-18 of 2004, and the maximum monthly rainfall was 12.82 inches (32.56 cm) in June of 1972. Between 1926 and 2004, the mean annual rainfall was 38.35 inches (97.41 cm).

Table 2.3-1. Mean and Extreme Rainfall Values for Penn State

Maximum rainfall in 24 hours	Maximum rainfall in 1 month	Annual Mean Rainfall
5.05 in, 12.83 cm	12.82 in, 32.56 cm	38.35 in, 97.41 cm
Sept 17-18 of 2004	June of 1972	1926-2004

Table 2.3-2 lists the mean and extreme snowfall data for Penn State (PSU Weather Station). Between 1888 and 2004, the maximum 24-hour snowfall was 27.5 inches (69.9 cm) on two occasions, the maximum one-month snowfall was 47.5 inches (120.6 cm), and the maximum snowfall in one winter was 102.6 inches (260.6 cm). Between 1926 and 2004, the mean snow accumulation was 46.9 inches (119.1 cm).

Table 2.3-2. Mean and Extreme Snowfall Values for Penn State

Maximum snowfall in 24 hours	Maximum snowfall in 1 month	Maximum snowfall in 1 winter	Annual Mean Snowfall
27.5 in, 69.9 cm	47.5 in, 120.6 cm	102.6 in, 260.6 cm	46.9 in, 119.1 cm
March 13 of 1993 March 3 of 1994	March of 1942	1994-1995	1926-2004

Table 2.3-3 shows the maximum effect of snow loading at the Pennsylvania State University. The snow load is calculated using the maximum winter rainfall in 24 hours and the maximum snowfall in 1 month (PSU Weather Station). The Canadian housing construction safety board approximates the maximum snow load by adding the maximum winter rainfall in 24 hours to the maximum snowfall in one month.

Table 2.3-3. Maximum Snow-Loading at Penn State

Maximum winter rainfall in 24 hours	Maximum snowfall in 1 month	Maximum snow-load.
3.25 in, 8.26 cm	47.5 in, 120.6 cm	50.75 lbs/ft^2

Table 2.3-4 shows wind direction frequency at the PSU Weather Station. The most common wind direction is from the southwest, which occurs 38% of the time. The least frequent wind direction is from the southeast, occurring 10% of the time.

Table 2.3-4. Wind Direction Frequency at the Pennsylvania State University

Variable	Northeast	Southeast	Southwest	Northwest
18%	13%	10%	38%	21%

Table 2.3-5 lists extreme wind reports. According to the American National Standards Institute, the 50-year mean recurrence interval is approximately 70 miles per hour (31.3 m/s) in central Pennsylvania. The strongest wind gust ever recorded at University Park was 95 miles per hour (42.4 m/s), occurring on July 23, 1991. When wind gusted above 57 miles per hour (25.4 m/s), there were 142 reports of damage from 1951 to 2000 in the centre region (PSU Weather Station).

Table 2.3-5. Extreme Wind Data for Penn State University

50 Year Mean Recurrence Interval	Number of times wind gusted over 57 miles per hour (25.4 m/s) and caused damage in Centre County from 1951-2000	Strongest wind gust ever recorded at Penn State
70 mph (31.3 m/s)	142	95 mph (42.4 m/s)
		July 23, 1991

Table 2.3-6 lists ice, sleet, and freezing rain data. Between 1982 and 1990, there were approximately 120 hours of freezing rain at University Park (Cortinas et al). For central Pennsylvania, the 50-year return period for uniform radial ice thickness and concurrent gust speed is .75 inches (1.90 cm) and 40 miles per hour (17.8 m/s), respectively (Cortinas et al).

Table 2.3-6. Mean and Extreme Ice Storm Data for Penn State University

Hours of Freezing Rain at University Park from 1982 to 1990	50 Year Return Period for Uniform Radial Ice Thickness (in and cm) and Concurrent Gust Speed (mph and m/s)
120	.75 in (1.90 cm); 40 mph (17.8 m/s)

Table 2.3-7 lists thunderstorm data for central Pennsylvania. There are approximately 40 thunderstorms per year in central Pennsylvania, with occasional severe storms that cause property damage from hail, wind, lightning, and local flash flooding. From 1950-2004, there were only 2 reported instances of injuries due to lighting in Centre County (PSU Weather Station).

Table 2.3-7. Thunderstorm Data for Central Pennsylvania

Mean Number of Thunderstorms Annually	Total Number of Lightning Injury Reports from 1950-2004
40	2

Pennsylvania is not in the normal hurricane path. The most recent storm to be classified as a hurricane was Hazel, which entered Pennsylvania in 1954. This storm was less severe in the University Park area than normally occurring thunderstorms. During the last decade, however, three hurricanes, after weakening to tropical storms, passed through the area. Tropical storms Fran in 1996, Isabel in 2003, and Ivan in 2004, all affected the Centre Region.

Table 2.3-8 shows tornado data for the entire state of Pennsylvania from 1950 to 2004. There was a mean of 15 tornadoes per year, and a mean of 3 F2 or greater tornadoes per year in the state (PSU Weather Station).

Table 2.3-8. Mean Tornado Data for Pennsylvania

Mean Number of Tornadoes Annually	Mean Number of F2 or Greater Tornadoes (>113 mph)
15	3

Table 2.3-9 shows tornado data in Centre County from 1950 to 2004. There were 9 total tornadoes in Centre County: 4 were F0, 3 were F1, 1 was F2, and 1 was an F4 tornado.

Table 2.3-9. Centre County Tornado Data

Total Number of Tornados from 1950 to 1995	Number of F0 Tornadoes (< 72 mph)	Number of F1 Tornadoes (73-112 mph)	Number of F2 Tornadoes (113-157 mph)	Number of F3 Tornadoes (158-206 mph)	Number of F4 Tornadoes (207-260 mph)
9	4	3	1	0	1
	Apr 9 of 1991 Nov 8 of 1996 May 29 of 1998 Jun 2 of 1998	Jun 5 of 1975 Jul 29 of 1976 Feb 16 of 1990	Jun 5 of 1975		May 31 of 1985

2.3.2 Site Meteorology

Assumptions for dispersion analyses of airborne releases from the facility, are given in section 13.1.1, in the discussion of the MHA.

2.4 Hydrology

The Penn State Breazeale Reactor (PSBR) is located on the flank of a broad syncline in the Valley and Ridge Province in Pennsylvania. The reactor foundation rests on the Nittany Formation, a massive Ordivician dolomite; adjacent geologic formations are limestones and dolomites as elaborated in Sections 2.5.1 and 2.5.2. The PSBR site is in the Spring Creek Drainage Basin which has an area of approximately 175 sq. mi (490 sq. km) as shown in **Figure 2.4-1.** The Basin is underlain primarily by limestone and dolomite formations. The drainage basin's outlet is located at Milesburg Gap (**Figure 2.4-1**) approximately 13 mi (22 km) North of the site.

The hydrology of the site area is typical of limestone terrains. Rainfall and runoff reach the underlying aquifer via fracture traces, sinkholes, caves and percolation through stream bottoms and soil covers. The facility's foundation rests directly on a massive dolomite (no soil condition) and there is no evidence of sinkholes in the near vicinity of the PSBR site.

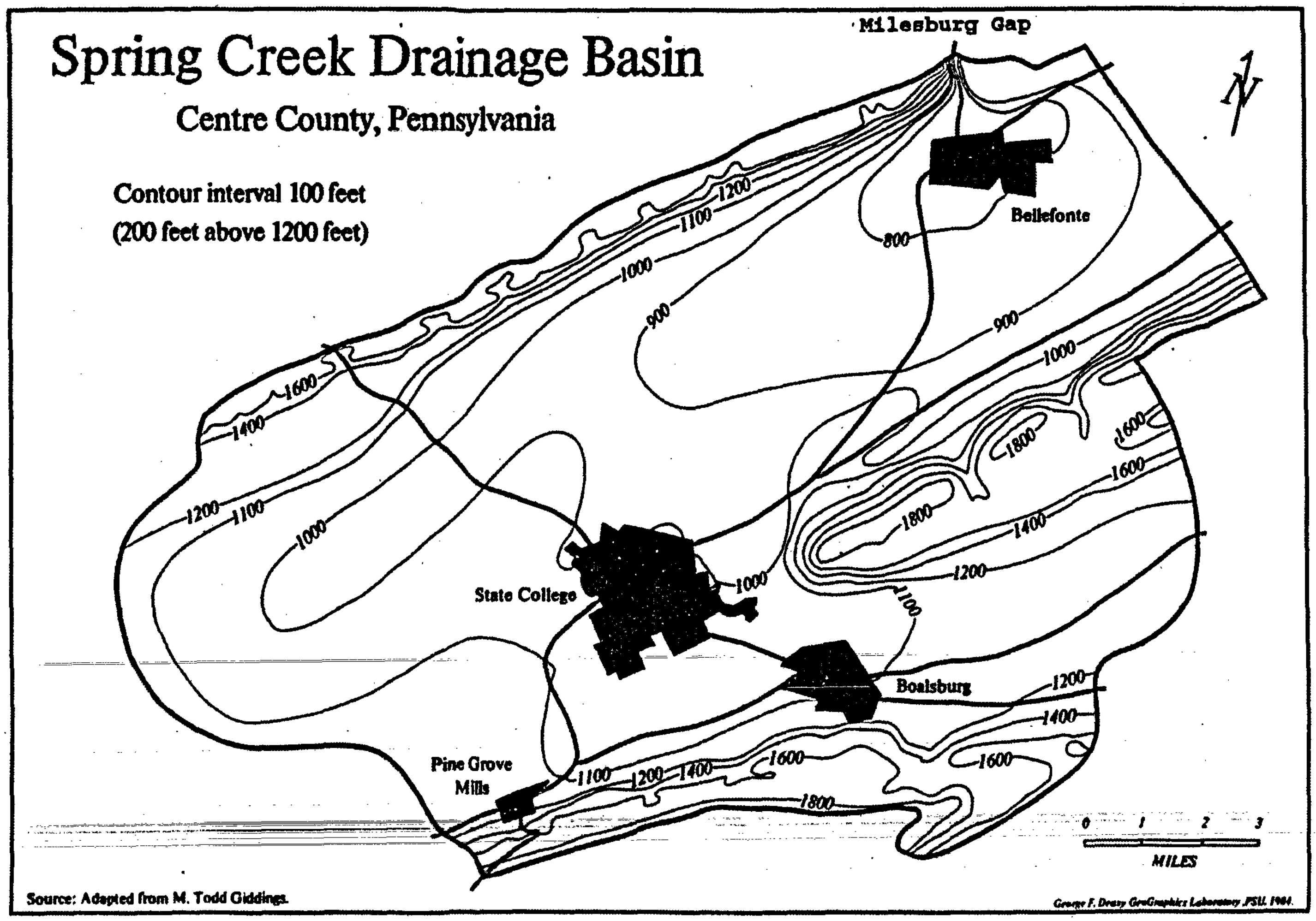

Figure 2.4-1 Spring Creek Drainage Basin in which the PSBR site is located (State College). Flow exits the basin at Milesburg Gap approximately 13 miles NNE of the site.

The water table is well below the ground surface at the site. **Figure 2.4-2** shows the elevation of the site and surrounding area along with streams, water wells and depths in them (ft.) to the water table. Given these water depth observations there is no likelihood that the water table will ever reach the foundation level of the PSBR. The site's elevation is approximately 47 m above the nearest stream and there is no history of flooding in the area that has ever reached to or near the site, so the likelihood of flooding reaching the facility is exceedingly small, posing no threat. Likewise there are no large dams upgradient in the drainage basin.

Therefore, there are no threats to the PSBR from hydrological phenomena at the site or in the surrounding area.

2.5 Geology, Seismology and Geotechnical Engineering

2.5.1 Regional Geology

The PSBR site is located on the flank of a syncline in the Valley and Ridge Province which trends SW-NE across Pennsylvania as shown on the geologic map in **Figure 2.5.1-1.** The solid star is the location of the PSBR site and the triangle is the location of the Susquehanna nuclear power plant site, both in the Valley and Ridge Province. **Figure 2.5.1-2** shows the tectonic map of the Valley and Ridge Province. The province is characterized by NE-SW trending folded and thrust-faulted Paleozoic sedimentary rocks that were last deformed during the Permian Alleghenian Orogeny (Gwinn, 1964). **Figure 2.5.1-3** shows a NW-SE geologic and tectonic cross-section along the blue line in **Figure 2.5.1-2** that passes near the PSBR site whose location projected onto the cross-section is approximately at letter N. This cross-section shows the major folds and imbricated thrust faults and their relation to the basal Waynesboro decollement. Based on the latest estimates by Alexander et al. (2005) of the depth to the crystalline basement rocks shown in **Figure 2.5.1-4**, the thickness of the sedimentary rocks beneath the PSBR site is approximately 7.3 km.

As discussed in more detail in Section 2.5.2, the near-surface geologic units beneath the site area consist primarily of Cambro-Ordovician limestones and dolomites.

2.5.2 Site Geology

The PSBR is located at (40.8042 N, 77.854 W) on the Nittany Formation, an Ordovician dolomite situated on the NW flank of a broad SW-NE trending syncline as shown in **Figure 2.5.2-1.** The adjacent formations are all Ordovician limestones and dolomites. The mapped surface faults in the site area are shown in red. There are no known faults at the site nor any that project to the site from the surrounding area. None of these faults is presently active; rather they are old faults most likely associated with the Permian Allegenian Orogeny (Gwinn, 1964).

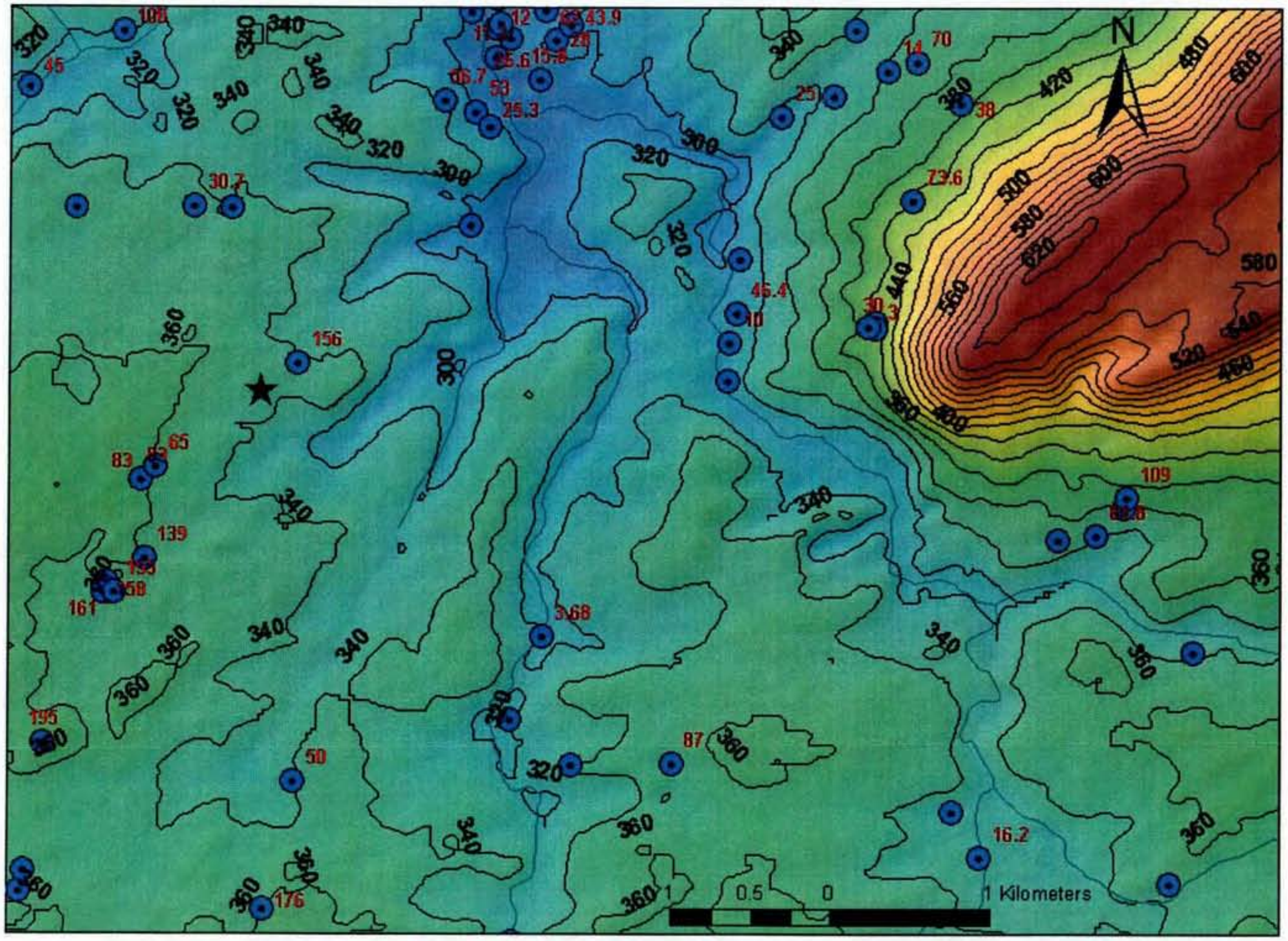

Figure 2.4-2 Map showing topography (m), streams and water wells (blue circles) with depth to water table (red) in ft. The star is the location of the PSBR.

C07

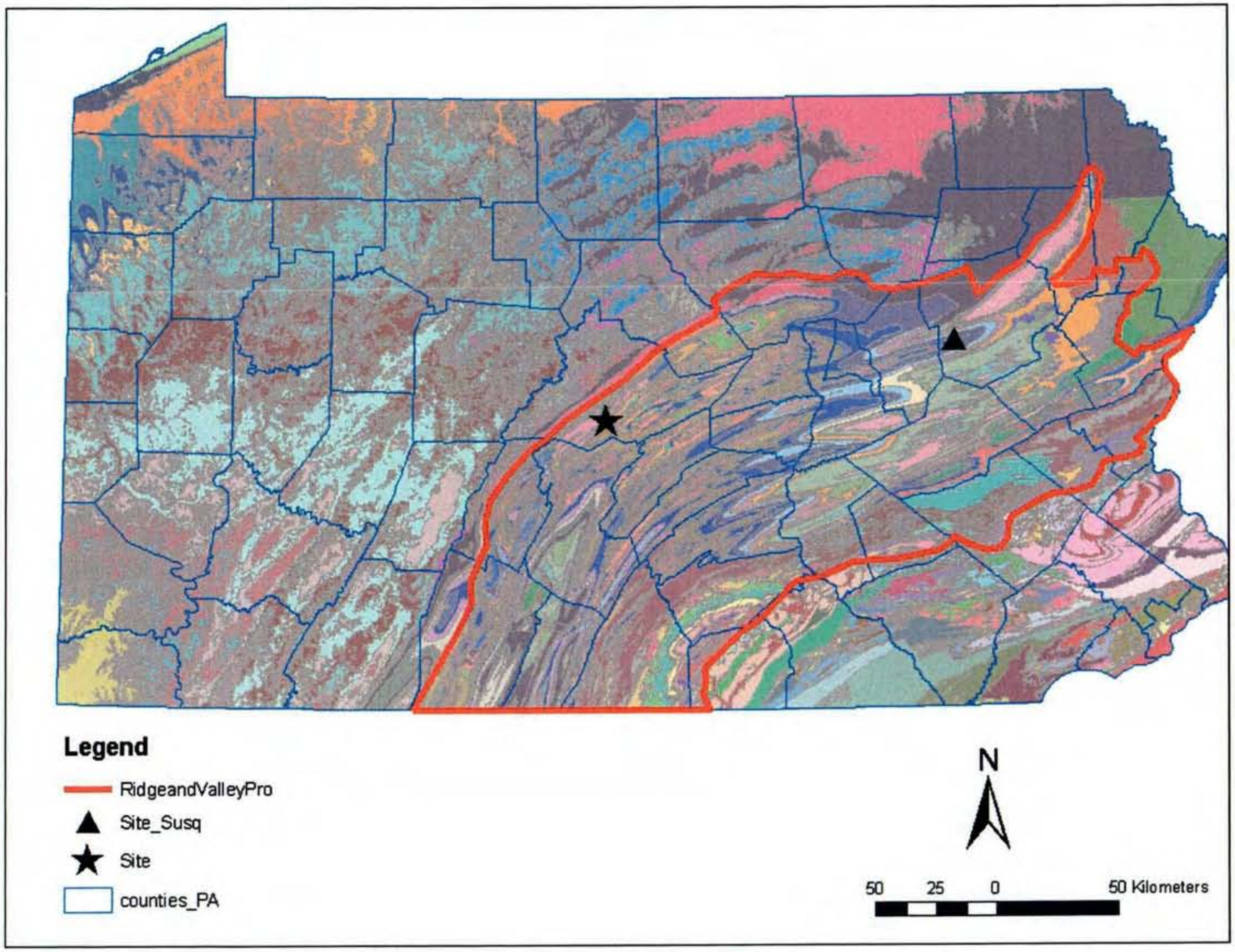

Figure 2.5.1-1 Geologic map of Pennsylvania showing the Valley and Ridge Province (red boundary), the PSBR site (star) and the Susquehanna nuclear power plant (triangle).

C09

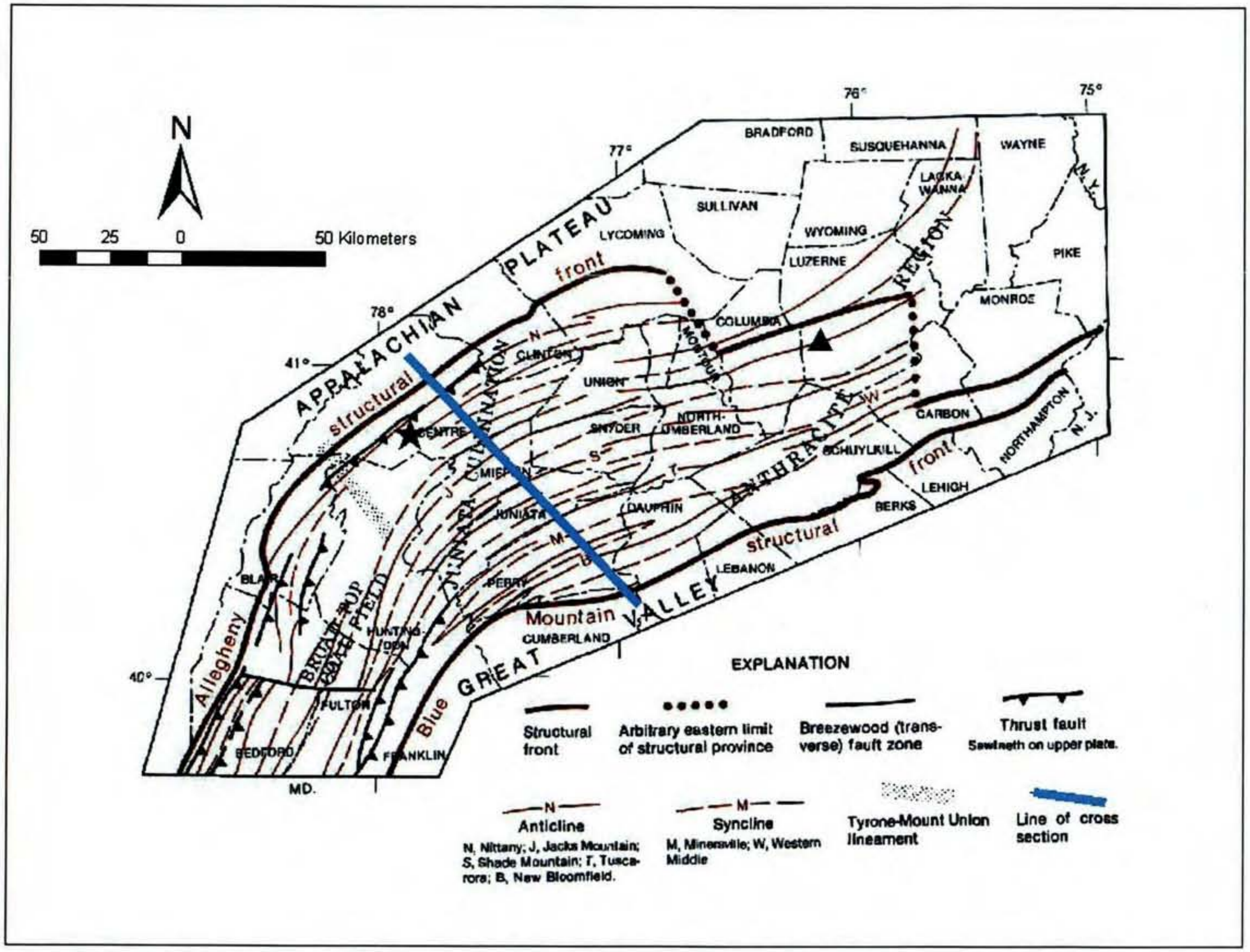

Figure 2.5.1-2 Tectonic map of the Valley and Ridge Province in Pennsylvania. The PSRB site (star) and Susquhanna nuclear power plant (triangle) are shown along with a line of cross-section (blue). (Modified from Faill and Nickelsen (1999))

C09

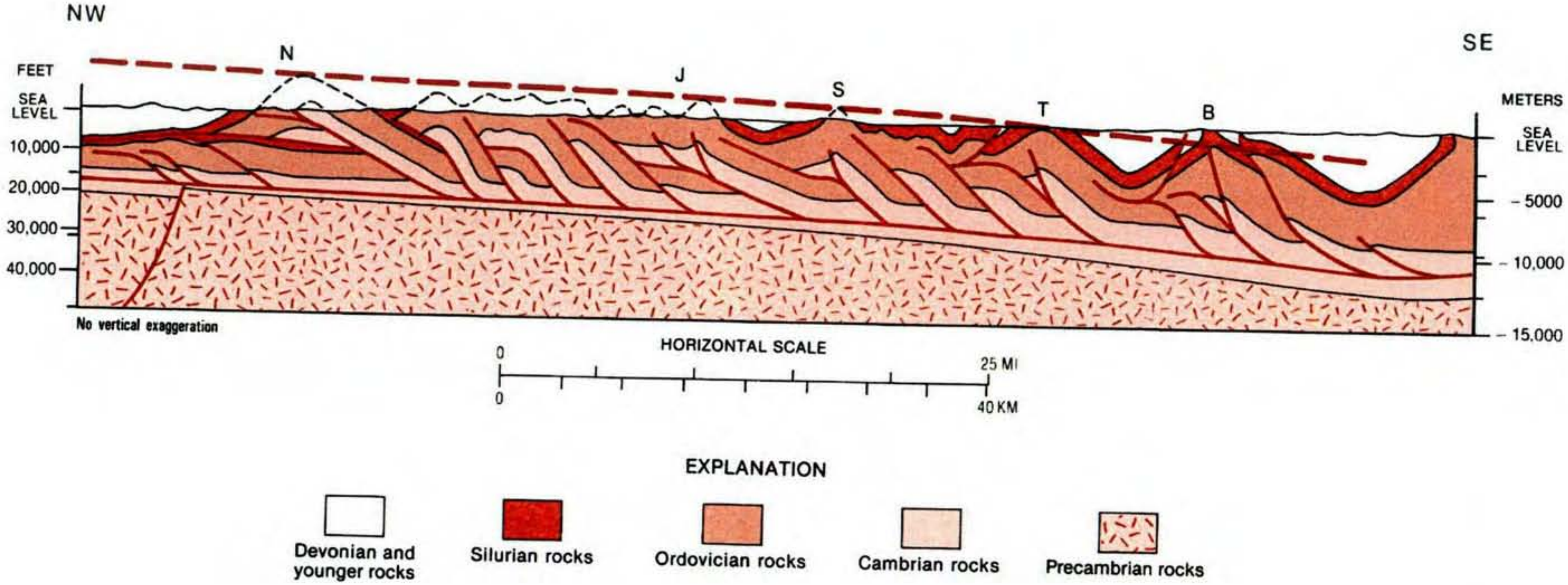

Figure 2.5.1-3 Cross-section along the blue line in Figure 2.5.1-2 across the Valley and Ridge structural province in Pennsylvania. This cross-section shows the major folds and faults and their relation to the basal Waynesboro decollement. The projection of the PSBR onto this cross-section is located approximately at the letter **N**. (Modified from Faill and Nickelsen (1999))

C10

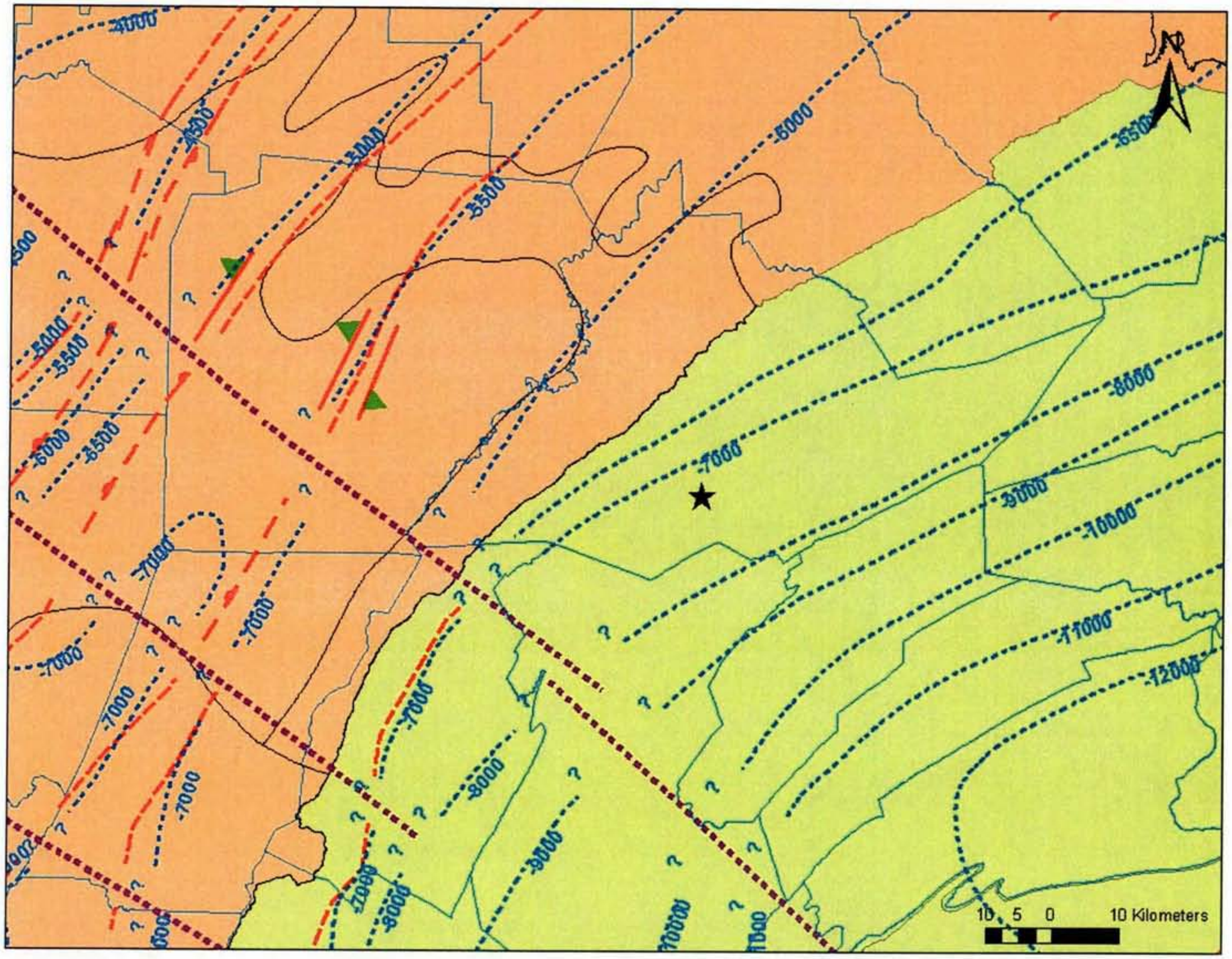

Figure 2.5.1-4 Map showing depth to basement (blue contours) and basement faults and major lineaments in the vicinity of the PSBR site (star). (Modified from Alexander et al. 2005)

C11

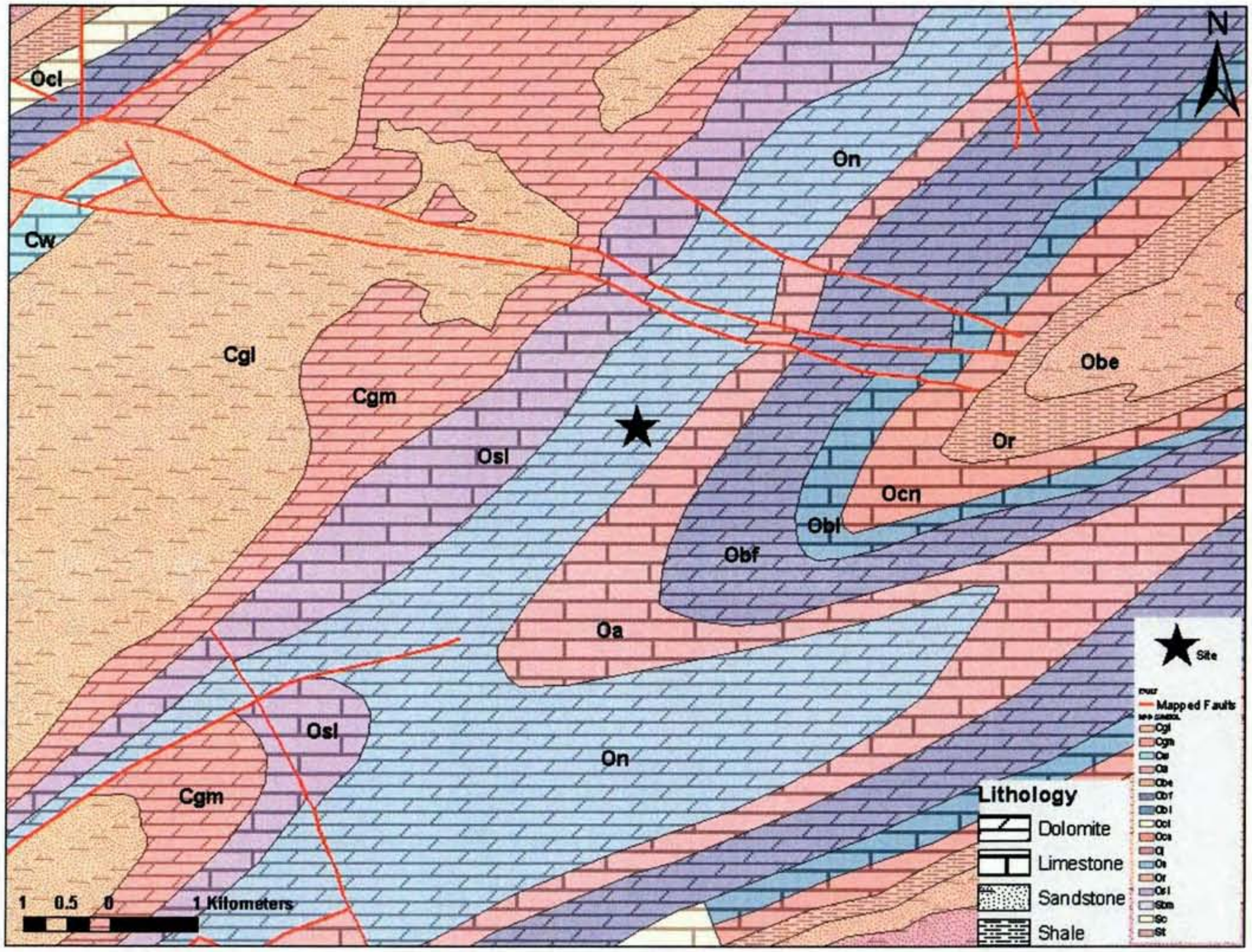

Figure 2.5.2-1 Geologic map of the PSBR site (star) showing geologic formations and mapped faults. The PSBR is located on the Nittany Formation, an Ordovician dolomite; adjacent formations are limestones and dolomites.

C12

Figure 2.5.2-2 shows a NW-SE geologic cross-section near the PSBR site. The approximate location of the site is shown by the solid star. There are three imbricated thrust faults beneath the site area, two emanating from the basal decollment as shown. These features are similar to those on the longer cross-section across the Valley and Ridge Province shown in **Figure 2.5.1-3.**

The PSBR site area is typical of the Valley and Ridge Province. It is located on the flanks of one of the many synclines in the province, which along with adjacent anticlines comprise the SW-NE trending folded Appalachians. Likewise the mapped surface faults in the site area are typical of the many old, inactive faults found throughout the Valley and Ridge Province. As discussed in Section 2.5.3, the entire province is characterized by a very low level of seismicity with infrequent small (M < 3.6) earthquakes.

2.5.3 Seismicity

The site area and the Valley and Ridge Province in which it is located has a very low level of seismicity compared to the surrounding, more-active areas in the northeastern U.S. All historic and instrumentally-recorded earthquakes within 300 km of the PSBR site contained in the United States Geologic Survey (USGS) seismicity database are listed in Appendix 1 and Appendix 2. Earthquakes in Appendix 1 occurred from 1724-1972 and their size is based primarily on Modified Mercalli Intensity (MMI) observations, some with inferred magnitude values. Earthquakes in Appendix 2 are those that have occurred from 1973-2005 within 300 km of the site; most are based on instrumental recordings of ground motion. Appendix 3 is a recent revised catalog of all earthquakes in Pennsylvania and immediately surrounding areas compiled by Faill (2003). More information is provided on the sources of the observations than in the USGS compilation. Although there are minor differences compared to the USGS database, the two are very nearly the same.

Figure 2.5.3-1 shows the locations of all the USGS events from 1724-2005 within 300 km of the site (star) superimposed on a map of Pennsylvania tectonic provinces. The green squares represent events based only on Modified Mercalli Intensity (MMI) observations and the green circles represent events with measured magnitudes from instrumental recordings. Also shown are basement faults and major lineaments from a recent study by Alexander et al. (2005). The solid triangle is the site of the Susquehanna nuclear power plant, located in the same Valley and Ridge tectonic province (blue) as the PSBR site (solid star); available probabilistic seismic hazard estimates for that analog site will be presented in Section 2.5.5.

Figure 2.5.3 -2 shows the seismicity from the revised catalog by Faill (2003) (Appendix 3) for Pennsylvania and immediately surrounding areas. There are only minor differences compared to the USGS seismicity database (**Figure 2.5.3-1**).

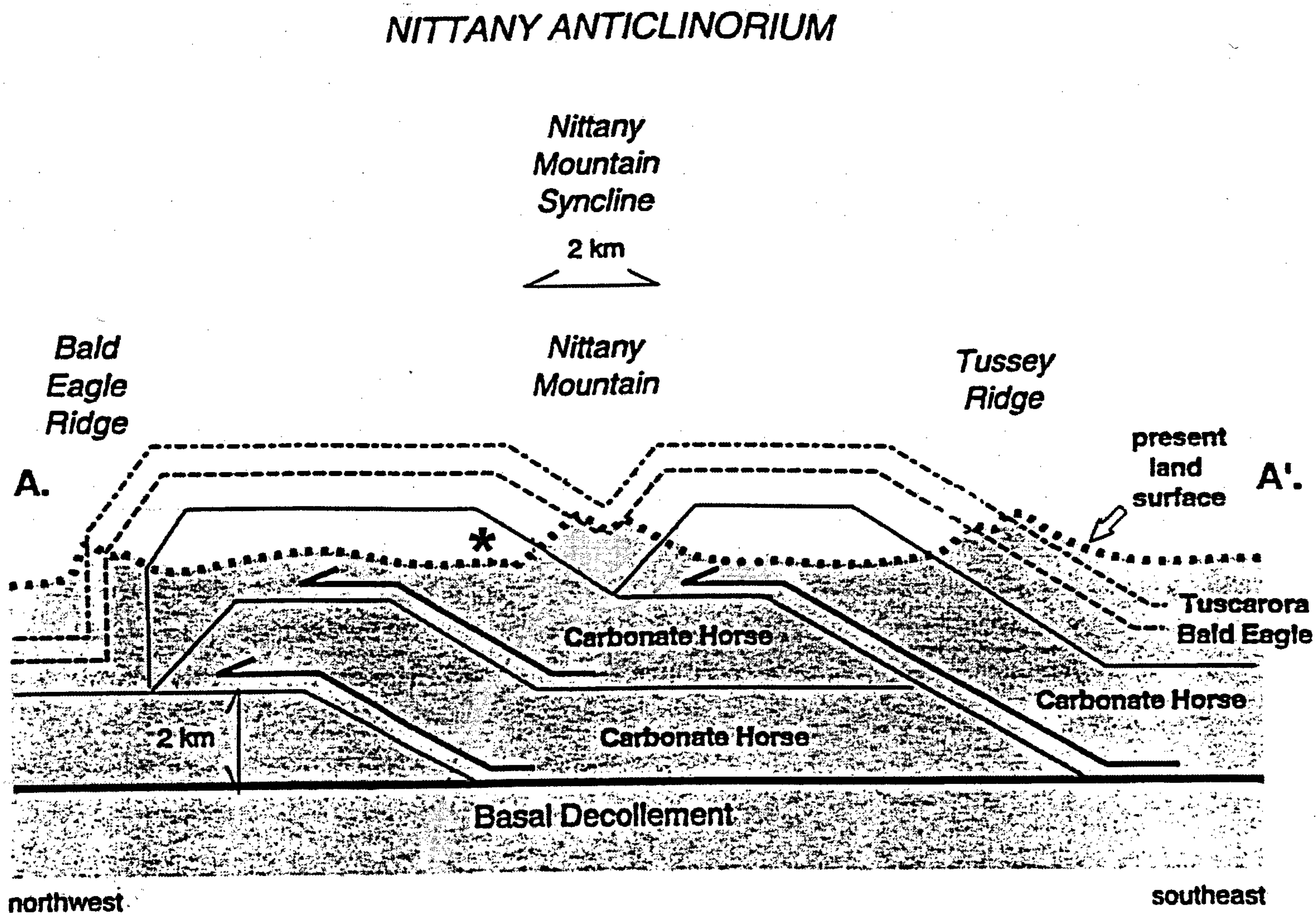

Figure 2.5.2-2 NW-SE geologic cross-section near the PSBR site (star) from Engelder, 2005 (pers. comm.)

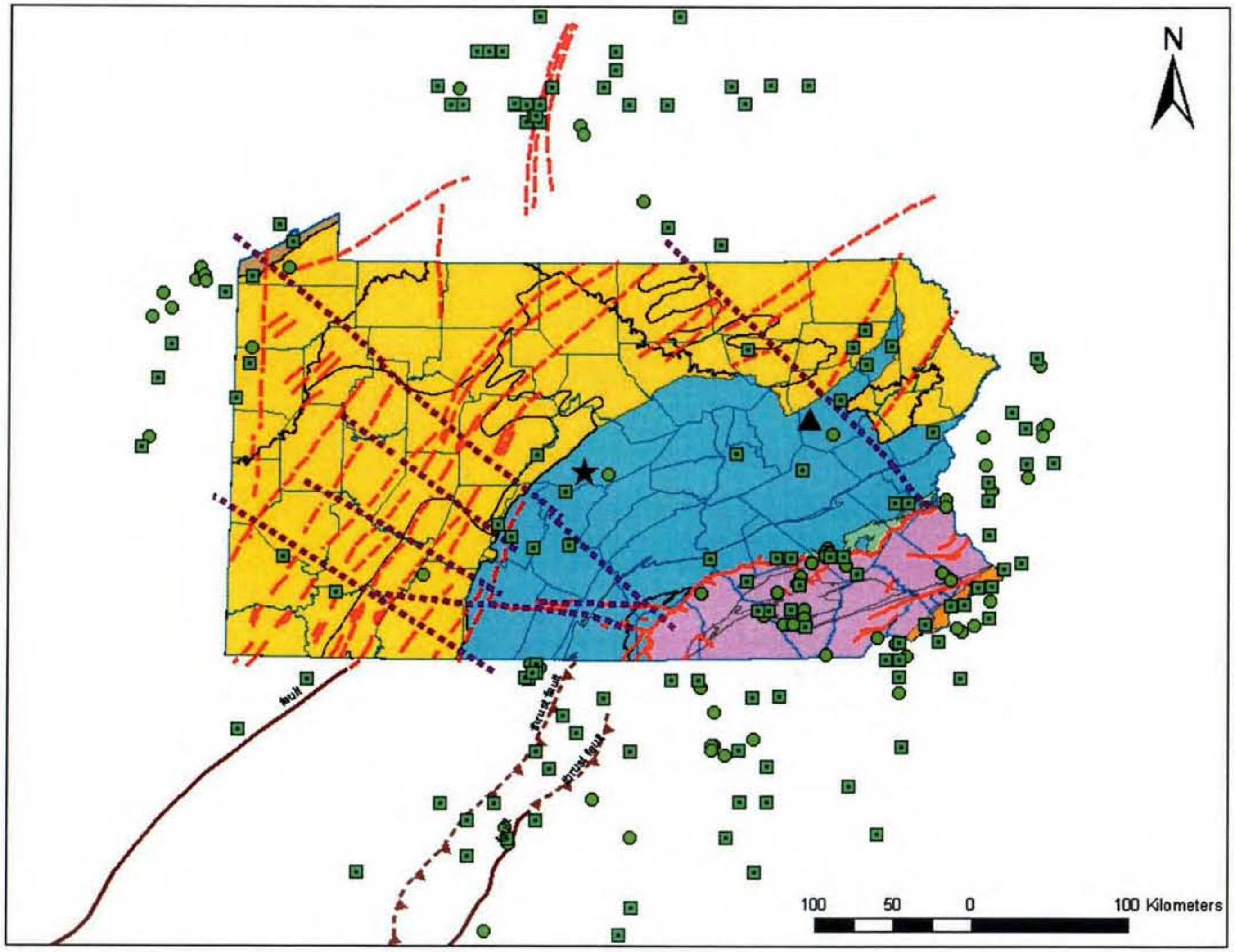

Figure 2.5.3-1 Map showing all seismicity within 300 km of the PSBR site (star) from 1724 to 2005 from the USGS (NEIC) database (see Appendices 1 and 2). Also shown are basement faults and major lineaments (Alexander et al., 2005) and tectonic provinces in Pennsylvania.

C13

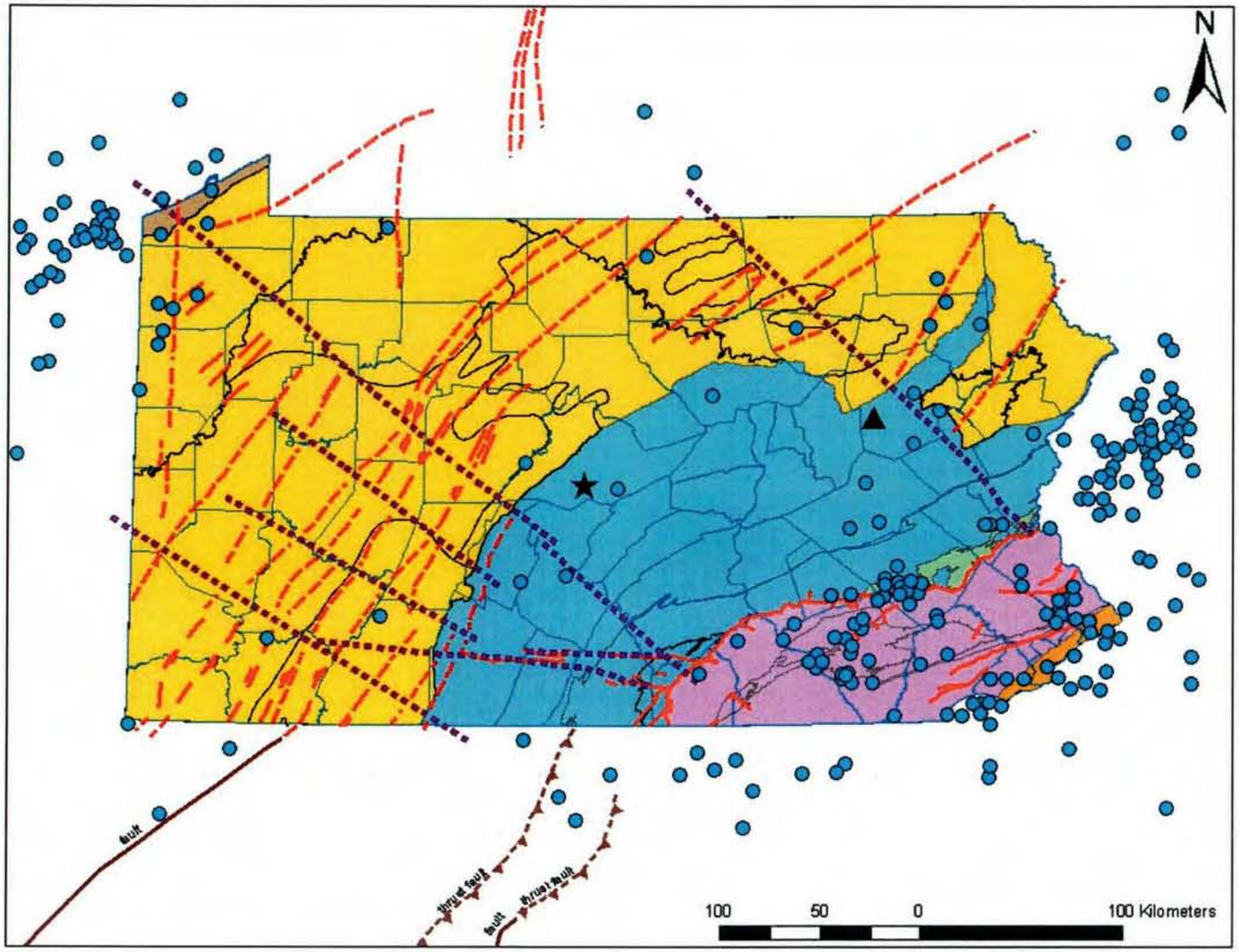

Figure 2.5.3-2 Map showing all seismicity in Pennsylvania and immediately adjacent areas from 1724 to 2003 compiled by Faill (2003) and listed Appendix 3. Also shown are the PSBR site (star), basement faults and major lineaments (Alexander et al., 2005) and tectonic provinces in Pennsylvania.

C14

2.5.4 Maximum Earthquake Potential

Figure 2.5.4-1 shows all the earthquakes greater than magnitude 3.0 within 300 km of the PSBR site (star) from 1724-2005 listed in the USGS seismicity database in Appendices 1 and 2. The Valley and Ridge Province is outlined in yellow. **Figure 2.5.4 - 2** shows all the earthquakes with MMI values greater than 4 that have occurred from 1724-2005 that are listed in the USGS seismicity database in Appendices 1 and 2. Note that prior to 1973 only MMI observations are available, except for a few inferred magnitudes, whereas most of the events since 1973 have instrumentally determined magnitudes but no MMI estimates.

Figure 2.5.4-3 shows all the earthquakes with magnitude greater than 3.0 in Pennsylvania and immediately surrounding areas from 1724-2003 compiled by Faill (2003) and included in Appendix 3. The Valley and Ridge Province is outlined in yellow. **Figure 2.5.4-4** shows all earthquakes with intensities greater than MMI IV in Pennsylvania and immediately surrounding areas from 1724-2003 from Faill's compilation in Appendix 3. As with the USGS seismicity database, instrumentally determined magnitudes have been available only since about 1973 so most events since 1724 have only intensity or magnitude estimates, although some have both.

From these seismicity maps it is clear that the Valley and Ridge Province has a very low level of seismicity compared to more-active areas in the surrounding regions. The largest known earthquake in the Valley and Ridge Province during approximately the past 300-year period of observations is magnitude 3.7 and MMI = 6 (VI) (see **Figures 2.5.4-1** and **2.5.4-2** (USGS) and **Figures 2.5.4-3** and **2.5.4-4** (Faill)). Note that the MMI intensity 7(VII) and 6(VI) events near the Susquehanna nuclear power plant site were the result of a mine collapse and are not tectonic earthquakes; consequently the largest MMI for earthquakes in the Valley and Ridge Province is 6(VI).

Earthquakes with a maximum magnitude of 3.7 or MMI of 6(VI) will not cause damage to well engineered structures such as the PSBR or Susquehanna nuclear power plants. In fact worldwide there is no documented case of damage to well-engineered structures caused by earthquakes of magnitude 5 or smaller. Therefore the maximum earthquake potential for the PSBR site is well below the level that would cause damage to the facility.

2.5.5 Vibratory Ground Motion

As discussed in the previous sections, the PSBR site is located in the Valley and Ridge Province in an area typical of the province as a whole. Therefore, the maximum magnitude or intensity of past earthquakes in the Valley and Ridge Province should guide the seismic hazard for the site. The maximum magnitude of 3.7 and maximum MMI of 6(VI) of earthquakes that have occurred in this province during the past 300 years indicate a very low probability of an earthquake large enough to threaten the PSBR facility.

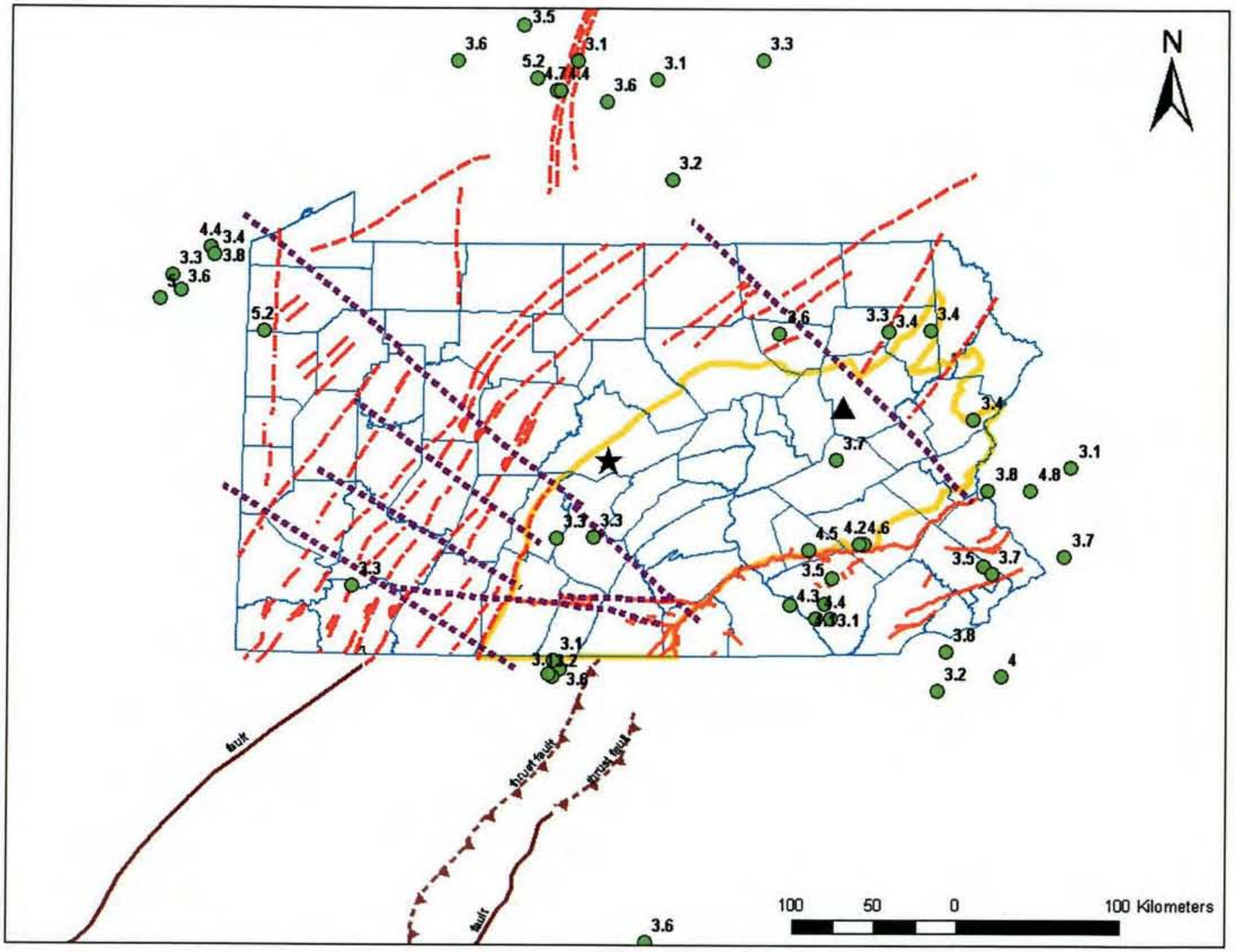

Figure 2.5.4-1 Earthquakes greater than magnitude 3 within 300 km of the PSBR site (star) from 1724 to 2005 subset from the USGS (NEIC) database (see Appendix 4). Also shown are basement faults and major lineaments (Alexander et al., 2005) and the Valley and Ridge Province in Pennsylvania (yellow boundary).

C15

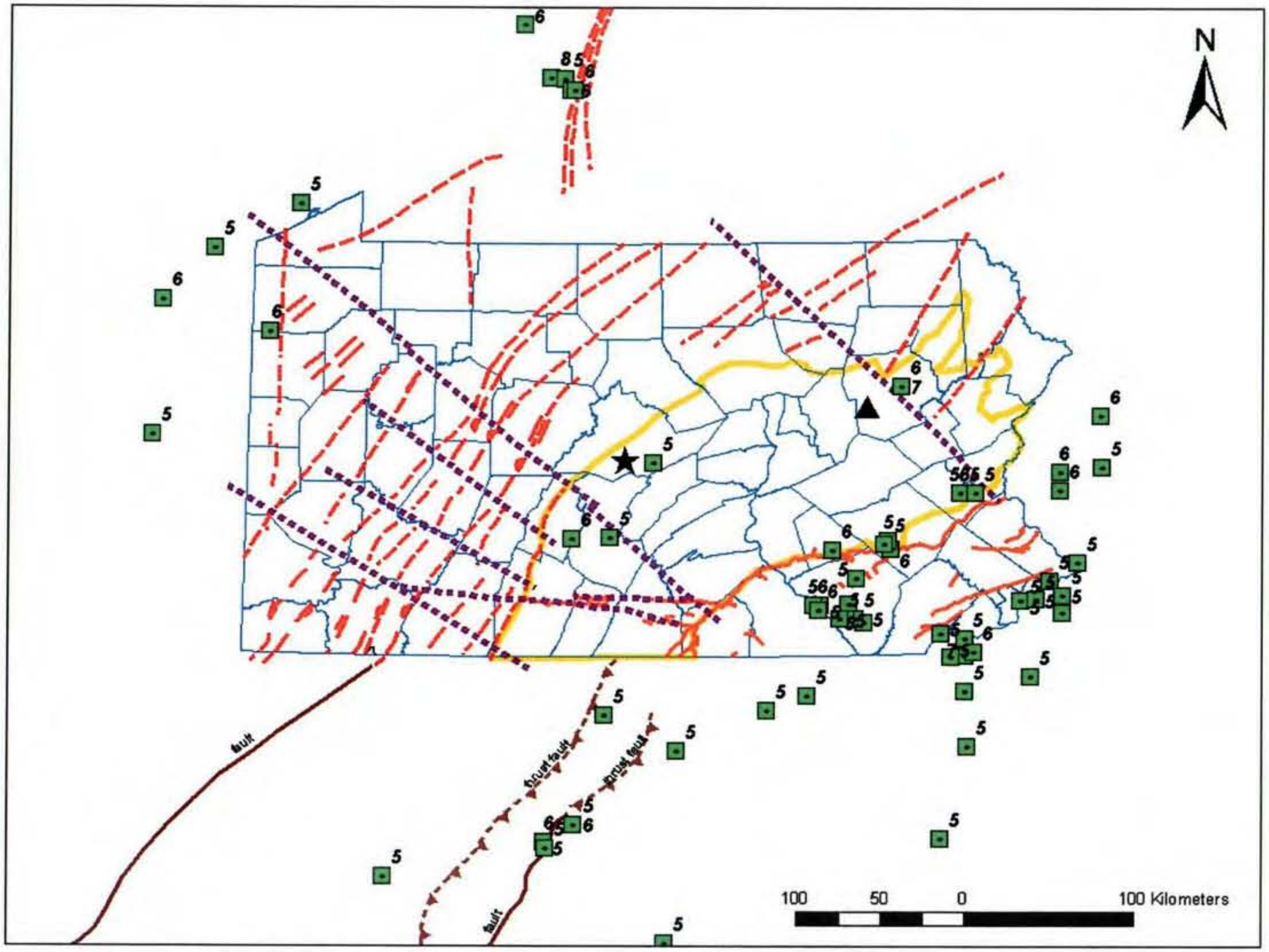

Figure 2.5.4-2 Earthquakes with intensities greater than MMI 4 within 300 km of the PSBR site (star) from 1724 to 2005, subset from the USGS (NEIC) database (see Appendix 5). Also shown are basement faults and major lineaments (Alexander et al., 2005) and the Valley and Ridge Province in Pennsylvania (yellow boundary). The events with intensities 6 and 7 near the Susquehanna nuclear plant site (triangle) were caused by a mine collapse.

C16

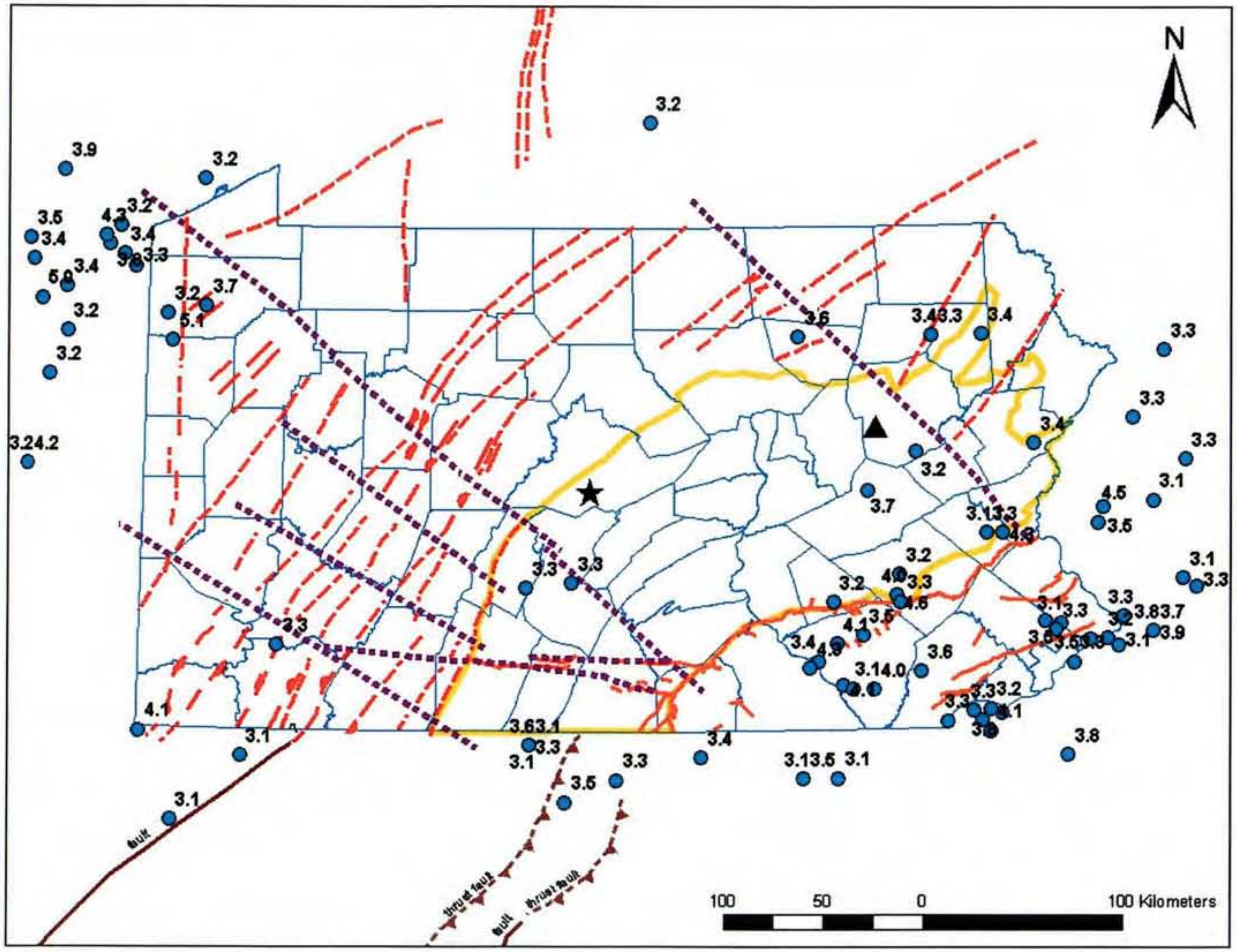

Figure 2.5.4-3 Earthquakes with magnitude greater than 3 in Pennsylvania and immediately adjacent areas from 1724 to 2003 compiled by Faill (2003) and subset from Appendix 3. Also shown are the PSBR site (star), basement faults and major lineaments (Alexander et al., 2005) and the Valley and Ridge Province in Pennsylvania (yellow boundary).

C17

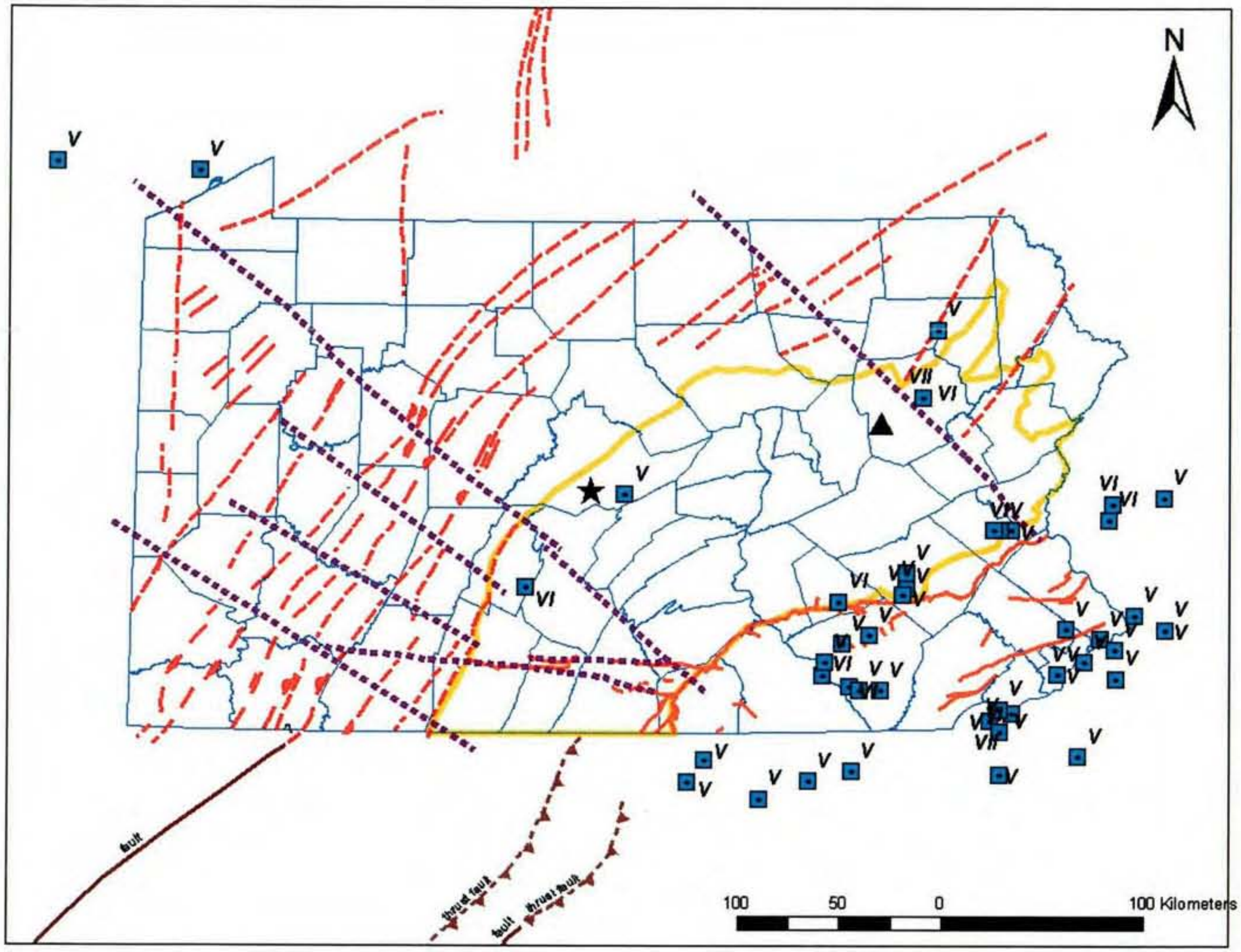

Figure 2.5.4-4 Earthquakes with intensities greater than MMI IV in Pennsylvania and immediately adjacent areas from 1724 to 2003 compiled by Faill (2003) and subset from Appendix 3. Also shown are the PSBR site (star), basement faults and major lineaments (Alexander et al., 2005) and the Valley and Ridge Province in Pennsylvania (yellow boundary).

C18

This assertion is backed up by probabilistic estimates of earthquake hazard in the Central and Eastern United States and at the nearby Susquehanna nuclear power plant site. **Figure 2.5.5-1** shows the earthquake hazard map of peak ground acceleration (%g) with 2 % probability of exceedance in 50 years for the Central and Eastern United States prepared by the USGS. The PSBR site (solid star) is located in a very low risk area compared to the surrounding more-active areas. **Figure 2.5.5-2** shows the portion of this hazard map for Pennsylvania and immediately surrounding areas along with the epicenters of earthquakes that have occurred from 1724-2005. From this hazard map the PSBR site has a 2% probability in 50 years that a peak ground acceleration of 0.07 g will occur, a very low hazard value.

Another useful perspective on earthquake hazard at the site is the estimated return time of an earthquake of magnitude greater than 4.75 at distance less than 50 km generated by the USGS and shown in **Figure 2.5.5-3**. The return time at the PSBR site (solid star) is approximately 5000 years. The return time of an earthquake with a magnitude greater than 5.5 at a distance less than 50 km is shown in **Figure 2.5.5-4**. The return time at the PSBR site (solid star) is 10000 years or more, representing an exceeding low probability of an event of this magnitude or greater occurring at the site.

Finally, the probabilistic seismic hazard assessment for the Susquehanna nuclear power plant site carried out by EPRI (1989) provides a very good upper bound on the probabilistic seismic hazard at the PSBR site, because the seismic hazard at the Susquehanna site is greater as seen in **Figure 2.5.5-2**, for example. The EPRI approach was to include past seismicity around a site without regard to its association with a tectonic province, all geologic or tectonic features in the surrounding region with some probability of future seismic activity with magnitudes of 5 or greater under current tectonic stress conditions, and attenuation of ground motion with distance from the source; this methodology which has been formally accepted by the US Nuclear Regulatory Commission for seismic hazard assessment is described in detail in EPRI (1986).

Table 2.5.5-1 gives EPRI's estimated annual probability of exceedance for peak ground acceleration at the Susquehanna nuclear power plant site. **Table 2.5.5-2** gives the spectral velocities (cm/sec) for various exceedance probabilities at the Susquehanna site. **Figure 2.5.5-5** shows the annual probability of exceedance of peak ground acceleration at the Susquehanna site together with the estimated uncertainty bounds. **Figure 2.5.5-6** shows the spectral velocity for three different small probability of exceedance values for the Susquehanna site. This site has a very low seismic hazard compared to other nuclear power plant sites in the eastern United States. For the reasons stated earlier the PSBR site should have a still lower probabilistic seismic hazard than Susquehanna. This result reinforces the conclusion that the damage potential from vibratory ground motion from earthquakes in the region is exceedingly low.

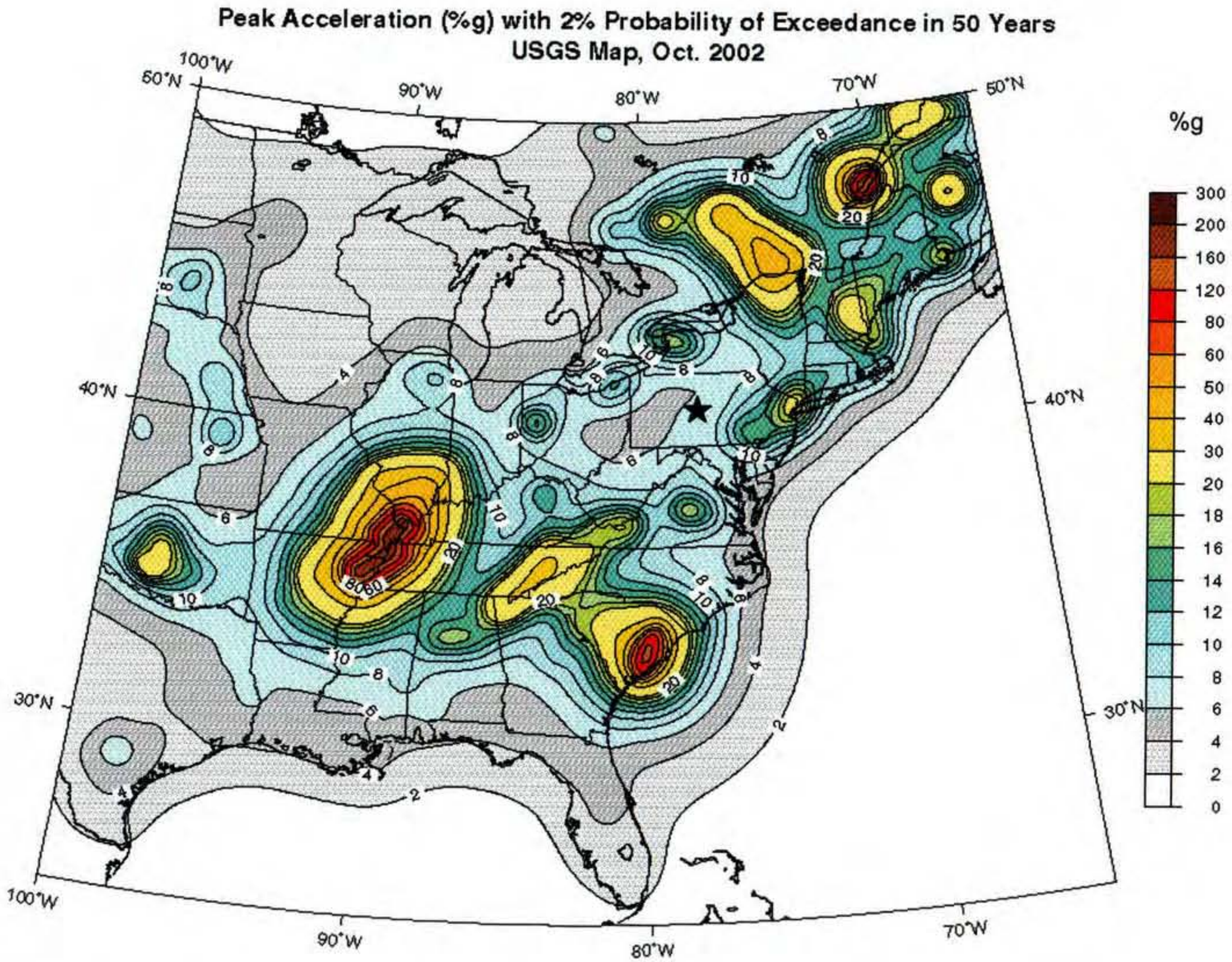

Figure 2.5.5-1 Earthquake hazard map for the Central and Eastern United States, prepared by the USGS. The PSBR site (star) is located in the very low risk area compared to surrounding more seismically active areas.

c19

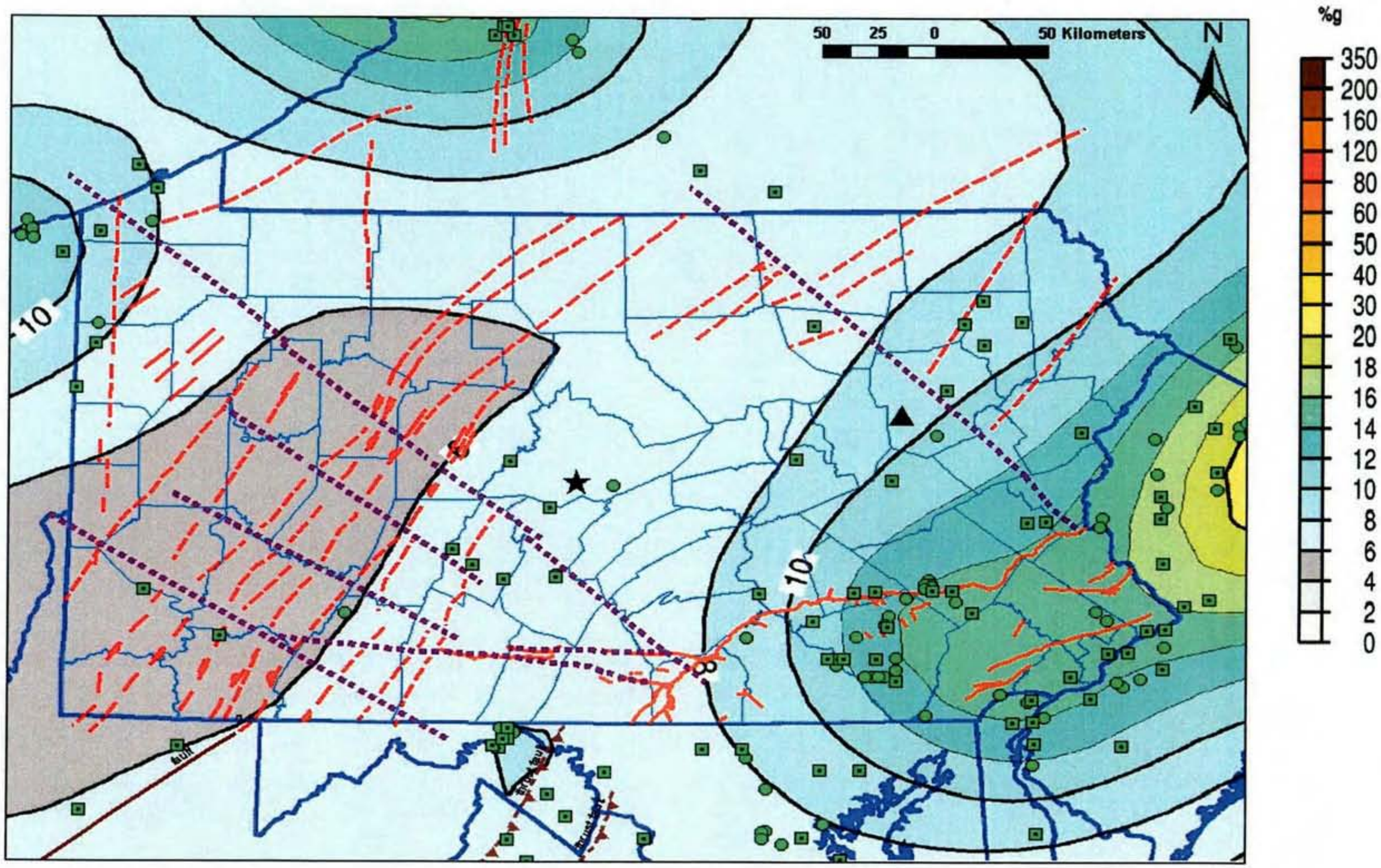

Figure 2.5.5-2 Earthquake hazard map (peak acceleration (%g) with 2% probability of exceedance in 50 years) for Pennsylvania, prepared by the USGS. The PSBR site (star) is located in the very low risk area compared to surrounding more seismically active areas. Also shown is the seismicity from 1724 to 2005 (green circles and squares) and basement faults (red) and major lineaments (purple).

C20

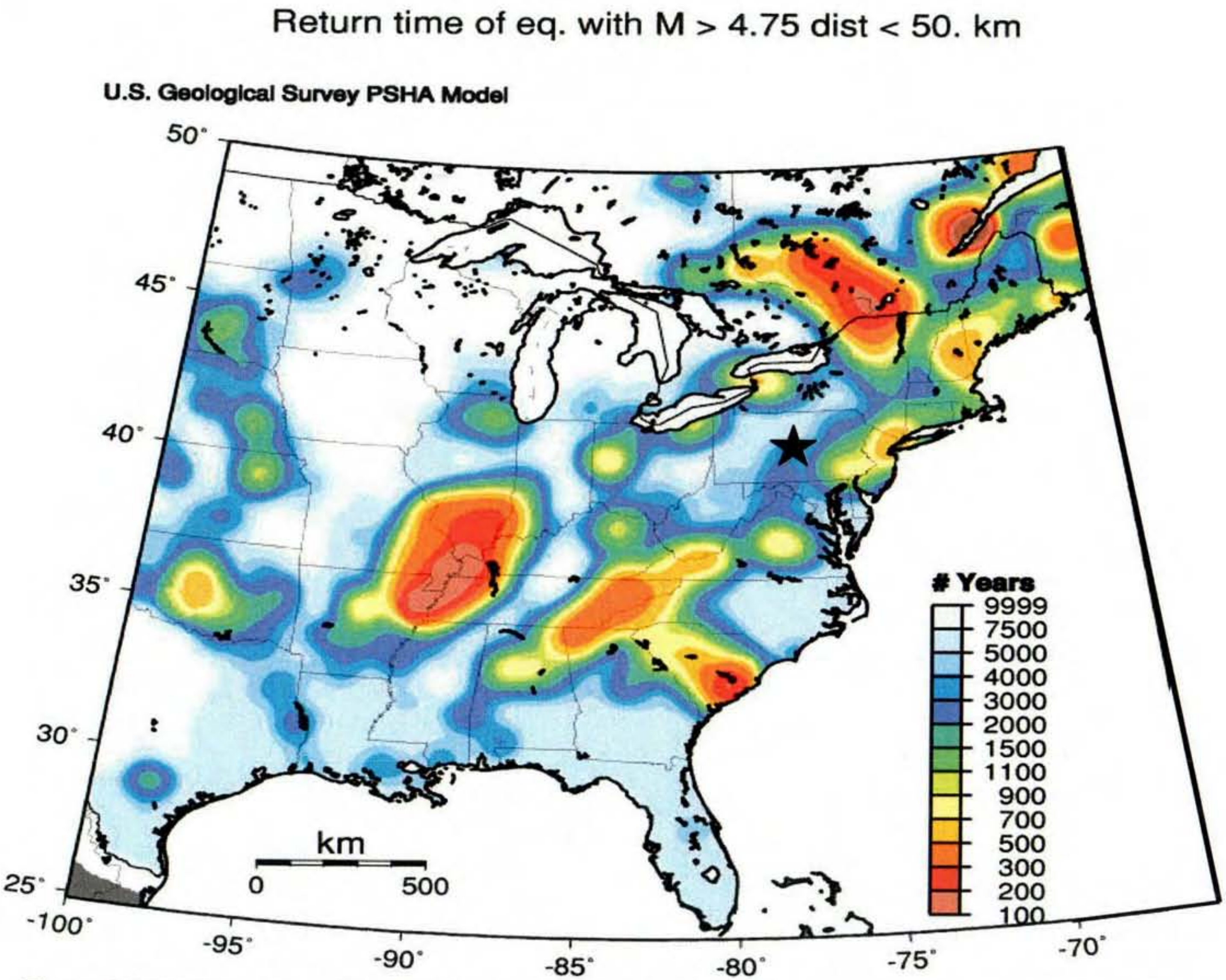

Figure 2.5.5-3 Return time of an earthquake of magnitude greater than 4.75 at a distance less than 50 km. The return period at the PSBR site (star) is approximately 5000 years.

C21

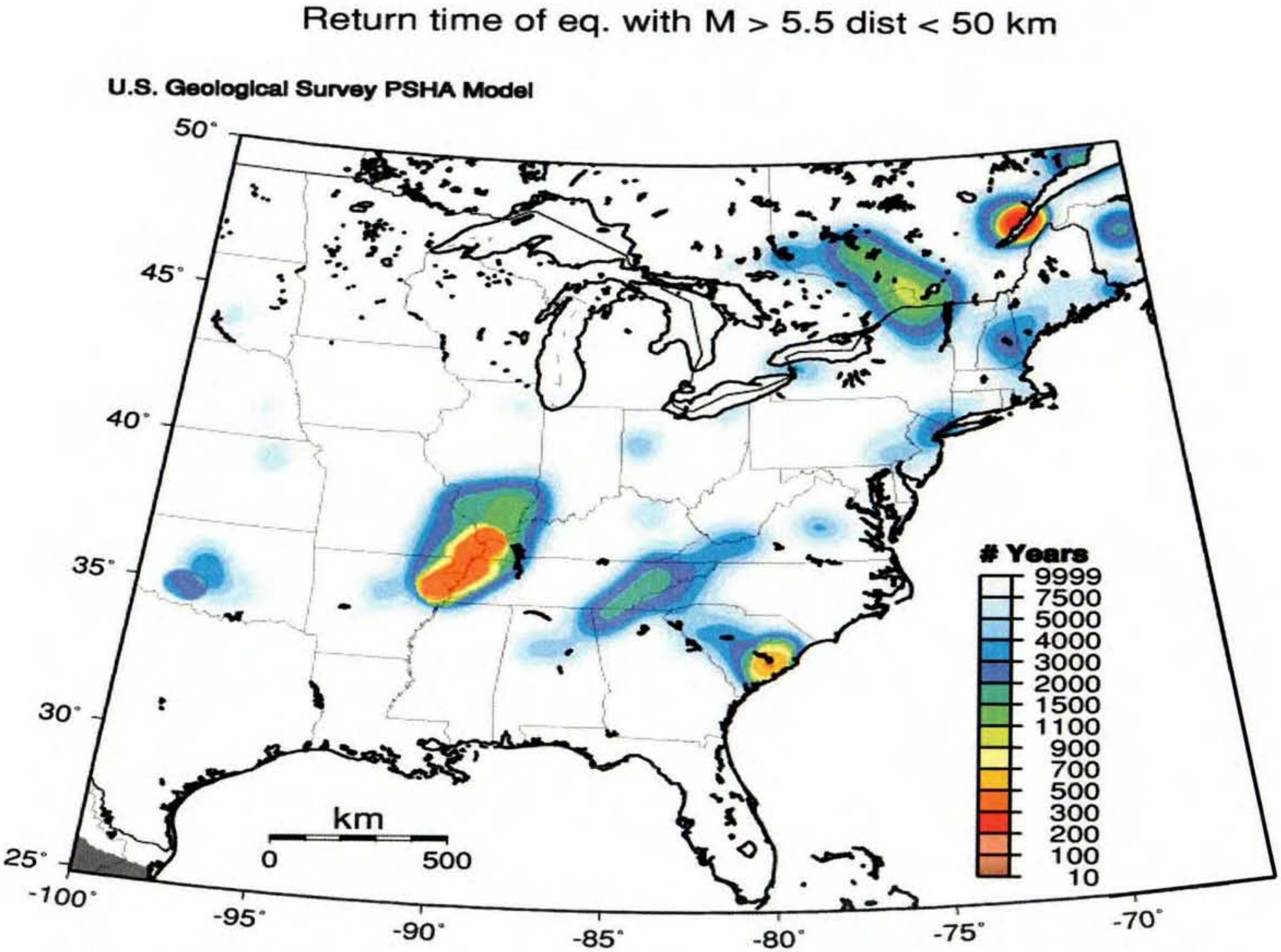

Figure 2.5.5-4 Return time of an earthquake of magnitude greater than 5.5 at a distance less than 50 km. The return period at the PSBR site (star) is 10000 years or more.

C22

Table 2.5.5-1

ANNUAL PROBABILITY OF EXCEEDANCE FOR PEAK GROUND ACCELERATION: SUSQUEHANNA SITE

Acceleration		Percentiles		
(cm/sec²)	Mean	15	50	85
5	6.8E−03	1.8E−03	5.4E−03	1.2E−02
50	3.9E−04	5.1E−05	2.6E−04	8.3E−04
100	1.0E−04	1.0E−05	6.1E−05	2.2E−04
250	1.1E−05	6.6E−07	6.3E−06	2.1E−05
500	1.3E−06	2.5E−08	4.9E−07	3.2E−06
700	3.9E−07	2.6E−09	1.1E−07	7.8E−07
1000	9.3E−08	5.6E−10	1.7E−08	1.6E−07

Table 2.5.5-2

SPECTRAL VELOCITIES (cm/sec) FOR VARIOUS EXCEEDANCE PROBABILITIES: SUSQUEHANNA SITE

		Frequency (Hz)				
		25	10	5	2.5	1
		Period (sec)				
Exceedance Probability	Percentile	0.04	0.1	0.2	0.4	1.0
1.E−03	15	0.12	0.18	0.23	0.26	0.18
	50	0.25	0.46	0.72	0.75	0.49
	85	0.37	1.19	1.58	1.71	1.77
2.E−04	15	0.31	0.53	0.78	0.95	0.56
	50	0.70	1.36	1.77	1.83	1.39
	85	1.02	2.67	4.06	4.49	4.61
1.E−04	15	0.49	0.86	1.15	1.28	0.90
	50	1.00	1.90	2.44	2.56	1.92
	85	1.44	3.56	5.77	6.24	6.45
1.E−05	15	1.31	2.39	2.84	3.18	2.22
	50	2.40	5.00	6.43	6.77	5.43
	85	3.90	8.08	13.80	15.70	16.20

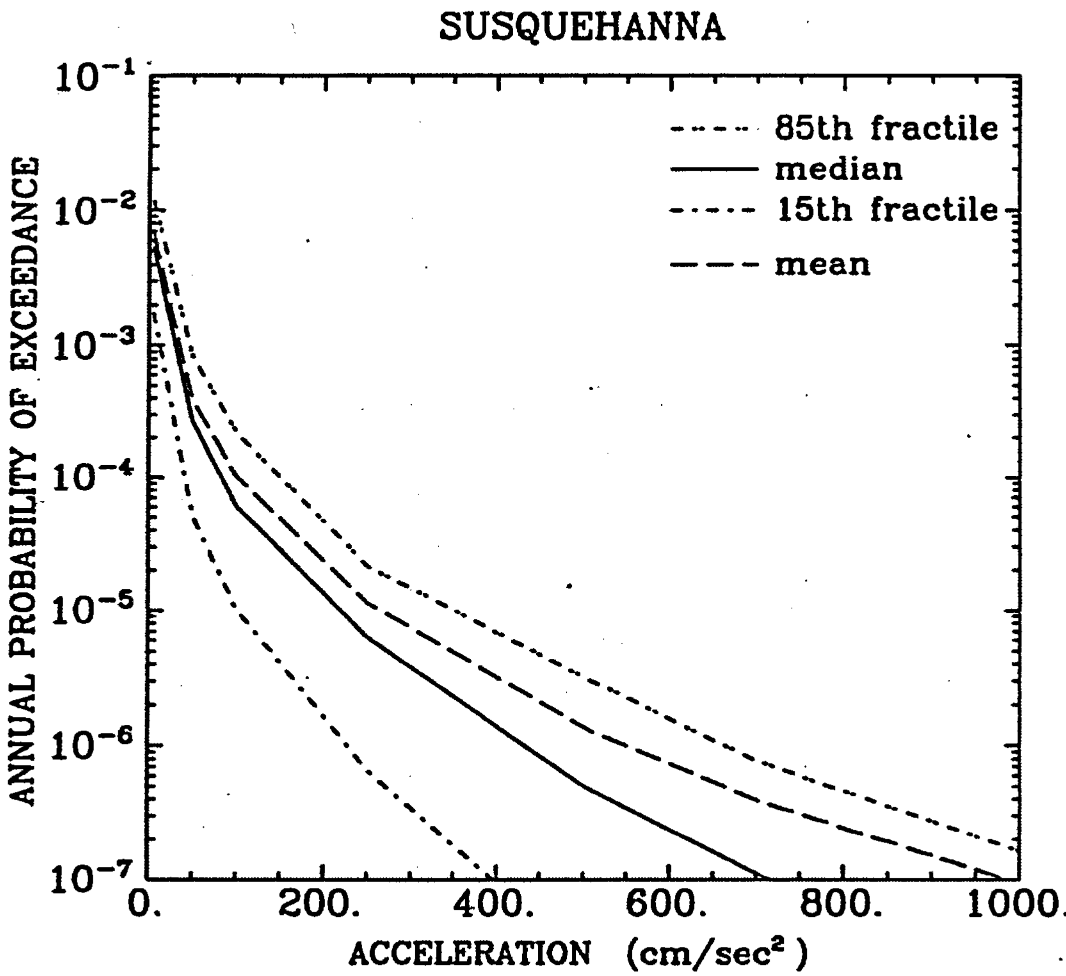

Figure 2.5.5-5 Annual probability of exceedance of peak ground acceleration at the Susquehanna nuclear power plant site.

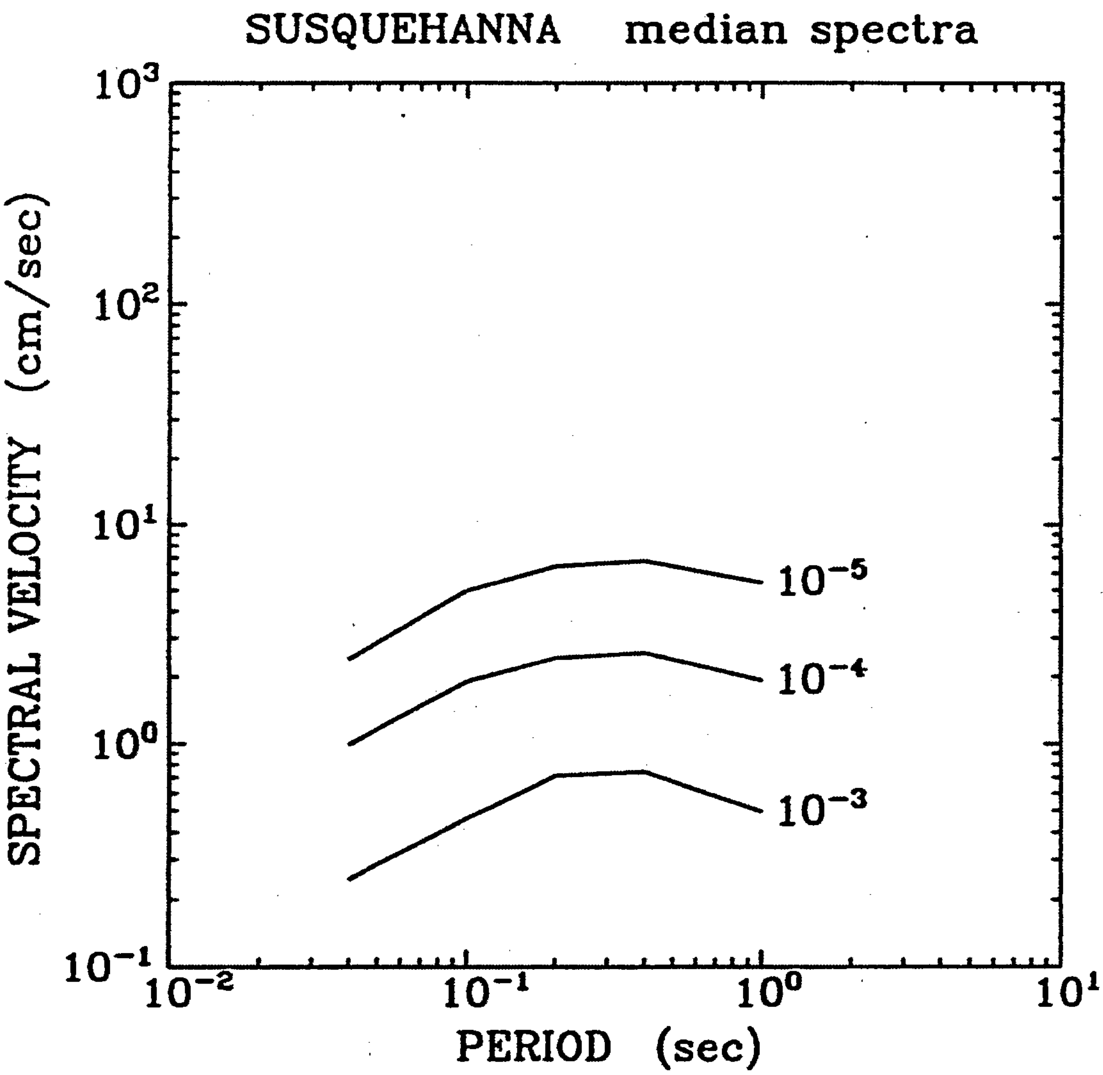

Figure 2.5.5-6 Median uniform hazard spectra for the 10^{-3}, 10^{-4} and 10^{-5} probability of exceedance at the Susquehanna nuclear power plant site.

2.5.6 Surface Faulting

Although there are numerous old, inactive surface fault in the site area (e.g. **Figure 2.5.2-1**), none is known to exist at the PSBR site. Surface faulting in the site area is typical of the surrounding region in the Valley and Ridge Province but these are inactive, old faults that were likely active during the Permian Allegenian Orogeny (Gwinn, 1964). In addition, there are no reported instances of surface ruptures caused by earthquakes in the eastern United States, including the largest ones (e.g. Zamani, 2003; Zamani and Alexander, 2005).

The closest known earthquake in the site area is the magnitude 3.0 event on August 15, 2001 located approximately 12 km ESE of the PSBR site as shown in **Figure 2.5.6-1**, This event was investigated in detail by Clouser (1992). It may be spatially associated with a nearby NNE-striking mapped fault shown on the Figure, but there is no evidence of surface displacement on this fault or in the vicinity of the epicenter, even though the event was shallow (about 1 km) according to Clauser (1992). This earthquake had a strike-slip focal mechanism as shown on **Figure 2.5.6-1** with the most-likely fault plane strike almost identical to the nearby mapped fault. Neither fault plane projects towards the PSBR site. The event's focal mechanism indicates a ENE-WSW horizontal maximum compressive tectonic stress direction, consistent with the prevailing direction throughout the eastern United States.

Figure 2.5.6-2 shows the relocated epicenter and felt area of the August 15, 1991 magnitude 3 earthquake near Centre Hall, PA as reported by Clauser (1992). The maximum intensity reported for this event was MMI V, but there is evidence that it may have been somewhat smaller. There was no reported damage caused by this earthquake.

Therefore, since there are no known faults at the site or others that project to the site and since earthquakes in the eastern United States do not cause surface ruptures, there is no potential for surface faulting at the PSBR site.

2.5.7 Liquefaction Potential

As noted earlier, the foundation of the PSBR reactor rests on a massive dolomite (hard rock) formation with no soil layer. Therefore, there is no potential for liquefaction at the site.

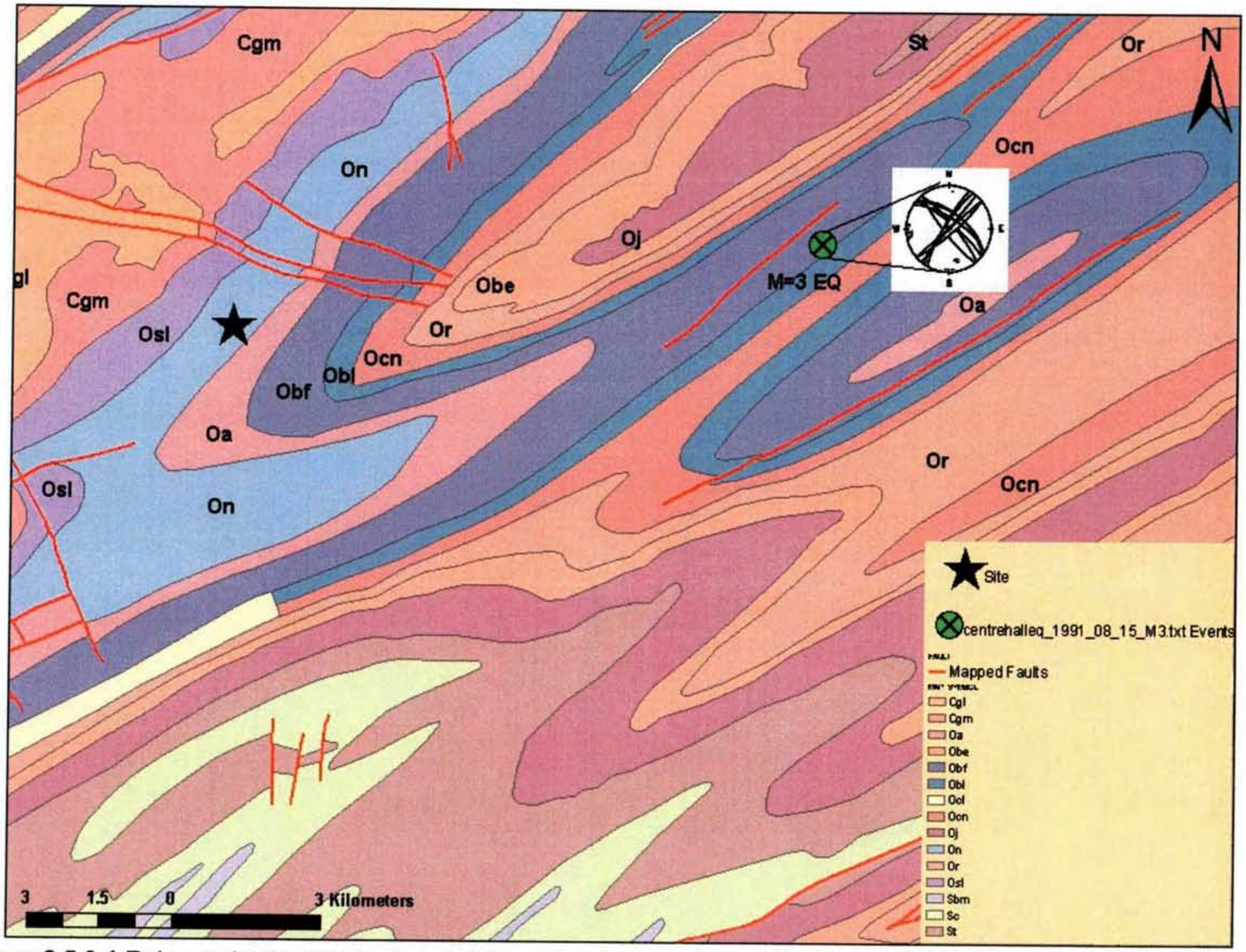

Figure 2.5.6-1 Relocated epicenter (green circle) and focal mechanism of the August 15, 1991 magnitude 3 earthquake near Center Hall, Pennsylvania, approximately 12 km ENE from the PSBR site (star) determined by Clouser (1992).

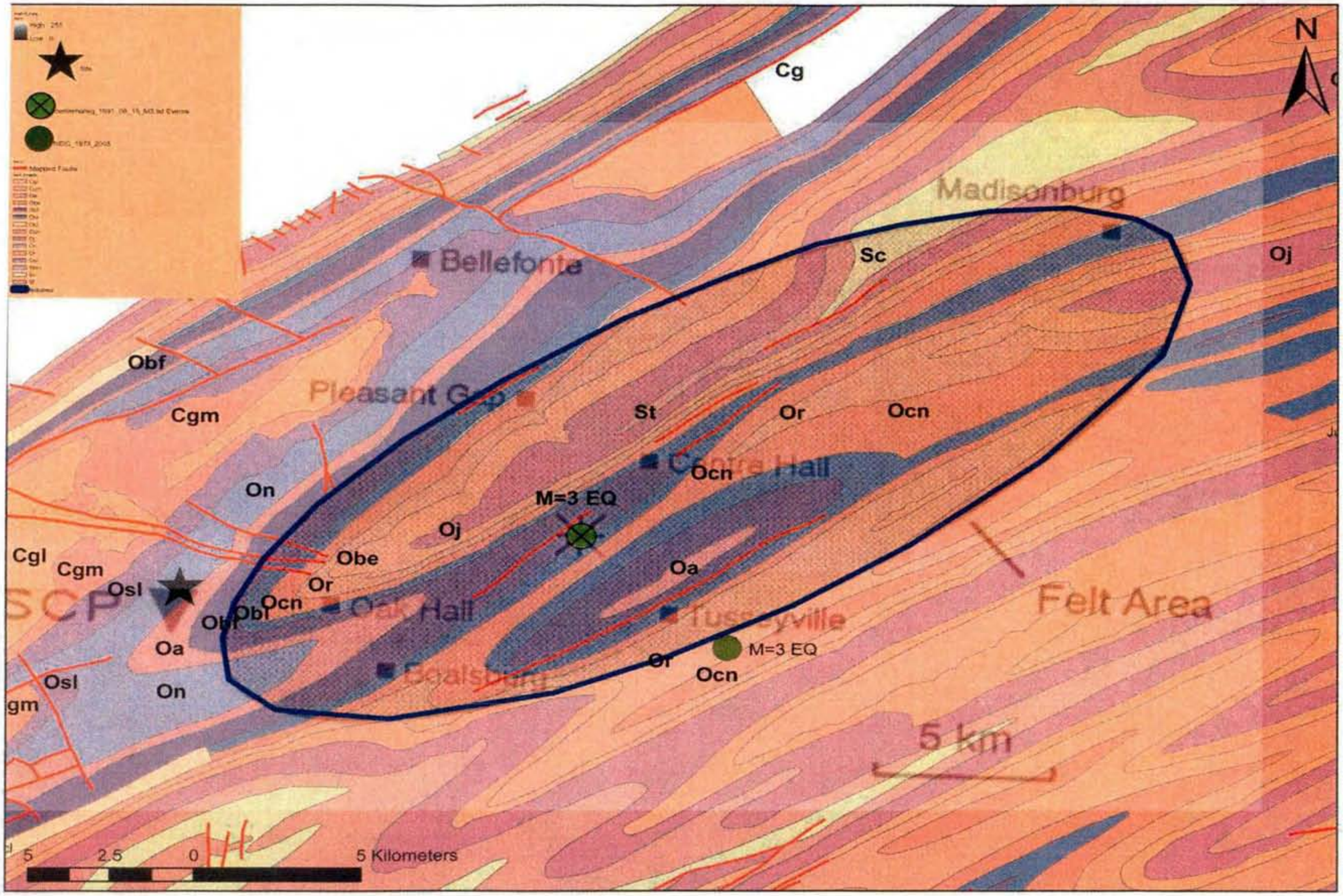

Figure 2.5.6-2 The relocated epicenter (green circle) and felt area determined by Clouser (1992).for the August 15, 1991 magnitude 3 earthquake near Center Hall, Pennsylvania, approximately 12 km ENE from the PSBR site (star).

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2.6 References

Alexander, S. S., Cakir, R., Doden, A. G., Gold, D. P., and Root, S. I., 2005, Basement Depth and Related Geospatial Database for Pennsylvania, Version 1.0, Department of Geosciences, Pennsylvania State University, Pennsylvania Geological Survey, 4th ser., Open-File General Geology Report (OFGG) 0501.0,www.dcnr.state.pa.us/topogeo/openfile, http://www.dcnr.state.pa.us/topogeo/openfile/basementmap.aspx

Clauser, R. H., 1992, Estimating the Hypocenter and Mechanism of the August 15, 1991 Centre Hall, Pennsylvania Earthquake Using Single-Station Data, Seis. Res. Ltrs. Vol. 63, No. 4, 541-555

EPRI, 1986, Probabilistic Seismic Hazard Methodology for the Central and Eastern United States, Electric Power Research Institute, Technical Report EPRI NP 4726-A, Vol. 1 Part 1: Methodology, Vol. 1 Part 2: Theory, Vol. 2: EQHAZARD Programmers Manual, Vol. 3: EQHAZARD Users Manual, Vol. 4: Applications, Vols. 5-20: Tectonic Interpretations, Vol. 11: Nuclear Regulatory Commission Safety Review,

EPRI,1989, Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States; Resolution of the Charleston Earthquake Issue, EPRI Report NP-6395-D, Special Report, 1989.

Faill, R. T. and R. P. Nichelsen, 1999, Appalachian Mountain Section of the Ridge and Valley Province, in The Geology of Pennsylvania, ed. C. H. Shultz, Spec. Publ. 1, PA Geol. Survey and Pittsburgh Geol. Soc. 269-285.

Faill , R. T., 2004, Earthquake Catalog and Epicenter Map of Pennsylvania, Pa. Topo. and Geol. Survey, Dept. of Conserv. and Natural Res.www.dcnr.state.pa.us/topogeo/pub/recent.aspx

Gwinn, V. E., 1964, Thin-skinned tectonics in the Plateau and northwestern Valley and Ridge Province of the central Appalachians, Bull. Geol. Soc. Am., 75, 863-900

USGS-NEIC Database, 2003. U.S. Geological Survey National Earthquake Information Center. http://neic.usgs.gov/ (21 Feb. 2005).

USGS NEIC Earthquake Catalog http://neic.usgs.gov/neis/epic/epic_circ.html

Zamani, A., 1999, Investigation of the Dependence of Earthquake Source Depth on Magnitude in Eastern North America, M.S. Thesis in Geosciences, The Pennsylvania State University, 75 pp.

Zamani, A. and S.S. Alexander, 2005, The Dependence of Earthquake Source Depth on Magnitude in Eastern North America (submitted to Geophys. Journ. Int.)

2.7 Appendices (because of the large volume of data, the appendices data is not enclosed, but is available from the PSBR if requested). Pertinent data from the appendices is included as appropriate on the figures in section 2.5.3 through 2.5.7 of the SAR.

2.7.1 Appendix 1 - Seismicity catalog from 1724-1972 within 300 km of the PSBR site from the USGS (NEIC) database for the Central and Eastern United States

2.7.2 Appendix 2 - Seismicity catalog from 1973-2005 within 300 km of the PSBR site from the USGS (NEIC) database for the Central and Eastern United States

2.7.3 Appendix 3 - Seismicity catalog from 1724-2003 for Pennsylvania and immediately surrounding areas compiled by Faill (2003)

2.7.4 Appendix 4 - Seismicity catalog of events with magnitude greater than 3.0 from 1724-2005 within 300 km of the PSBR site, subset from the USGS (NEIC) database in Appendices 1 and 2.

2.7.5 Appendix 5 - Seismicity catalog of events with intensity MMI greater than 4 (IV) from 1724-2005 within 300 km of the PSBR site, subset from the USGS (NEIC) database in Appendices 1 and 2.

3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The major design feature criterion to protect the safety of the public is the physical mechanisms and characteristics of the stainless steel clad TRIGA fuel elements (see section 4.5). The facility confinement (see section 6.2.1) and facility emergency exhaust system (see section 13.1.1, Engineered Safety Features) are adequate to help mitigate any airborne environmental release and limit any significant hazard to the public.

The major design change since the last license renewal was the 1991 console upgrade as a license amendment (see sections 1.8 and 7.0). A reactor core support structure modification in 1994 allows reactor core rotation and east-west core movement in addition to the previous north-south only movement (see sections 1.8 and 4.2.5) and was done under a 50.59 review.

Structural experimental facility modifications are a D_20 tank and beam port extension upgrade in 1997 (see section 10.2.2) and a new Fast Neutron Irradiator in 1997 (see section 10.2.4). These changes were done under 50.59 reviews.

Building modifications (machine shop and lobby extensions) are discussed in section 1.3. These modifications had no effect on the portion of the building structure that houses the reactor.

3.2 Meteorological Damage

There are no design criteria for the protection of facility structures from meteorological conditions except that all facility structures were constructed to building codes applicable at the time of construction. The reactor bay portion of the reactor building has withstood ~50 years of Pennsylvania weather without sustaining any damage. See section 2.3.1. (tables 2.3-5, 2.3-8, and 2.3-9) for extreme wind data, mean tornado data, and Centre County Tornado data.

3.3 Water Damage

There are no design criteria for the protection of facility structures or systems from water damage. The reactor facility site elevation precludes flooding, and withstood a record monthly rainfall in 1972 and a maximum 24 hour rainfall in 2004 without consequence. See section 2.3.1, table 1, for mean and extreme rainfall values and section 2.4 for hydrology discussion.

3.4 Seismic Damage

There are no design criteria for the protection of facility structures from seismic damage except that all facility structures were constructed to building codes applicable at the time of construction. The probability of a seismic event in the vicinity of the reactor site is considered insignificant as discussed in section 2.5.3.

3.5 Systems and Components

Reliance on the safety of the TRIGA fuel precludes the need for reliance upon other systems, structures, and components to ensure the safety of the general public. The emergency exhaust system would mitigate the consequences of the MHA (section 13.1.1) but is not required to be operable when the reactor is secured (see section 7.5).

4. REACTOR DESCRIPTION

4.1 Summary Description

The PSBR is a steady-state 1 MW(th) TRIGA, with pulsing capabilities of ~ 2000 MW(th). The TRIGA fuel is enriched to slightly less than 20% Uranium-235. The PSBR core consists of a mixture of 8.5 wt% and 12 wt% fuel.

The reactor pool is concrete with earth fill on most of its sides (see **Figure 4-1**). There is no earth fill on the south side of the pool where beam ports penetrate through the wall, however, an approximately 3 feet (.91 m) thick biological shield envelopes the south pool wall.

The core features natural convection cooling by filtered and demineralized water. The water serves as a reflector, a secondary moderator, a heat sink, and biological shield. The primary moderator is the Zr-H that is a part of the homogeneous fuel-moderator mix.

Unique features of the reactor core support structure, enable the core to be positioned at various locations in the reactor pool. Major experimental facilities consist of a central thimble, dry irradiation tubes, a pneumatic transfer system, fast neutron irradiators, and neutron beam port facilities.

4.2 Reactor Core

4.2.1 Reactor Fuel

The PSBR utilizes fuel-moderator elements in which zirconium hydride moderator is homogeneously combined with partially enriched uranium fuel. The fueled section of these elements is [redacted], and contains uranium in one of two different weight percentages enriched to slightly less than 20% in U-235. Some elements are 8.5 weight % uranium in zirconium hydride and the remainder are 12 weight %. The hydrogen to zirconium atom ratio of the fuel-moderator material for the 8.5 weight % fuel is 1.7 to 1.0 and for the 12 weight % fuel is 1.65 to 1.0. To facilitate hydriding, a 0.18 inches (.46 cm) diameter hole was drilled through the center of the active fuel section. A zirconium rod was inserted in this hole after hydriding was completed. **Figure 4-2** shows a standard fuel element.

The weight of a fuel element is about [redacted] with the U-235 content for new elements approximately [redacted] in the 8.5 weight % elements and approximately [redacted] in the case of 12 weight % elements. Serial numbers scribed on the top end fixtures are used to identify individual fuel elements. Each element is clad with 0.02 inch (.05 cm) thick stainless steel.

[redacted]

[redacted]

To measure fuel temperature during reactor operation, instrumented fuel elements are fabricated similar to standard elements but with three thermocouples embedded in the fuel region. One thermocouple is at the vertical centerline of the element and the other two are located 1 inch (2.54 cm) above and 1 inch (2.54 cm) below center. All three thermocouples are located [redacted] from the center of the fuel element. **Figure 4-3** shows an instrumented fuel element. The thermocouple lead wire passes through a water tight seal [redacted]

A pin on the bottom of the element locates the element in the bottom grid plate. Triangular spacers or fins (depending on the vintage of the fuel elements) on top of the element position the element in the top grid plate and also allow a pathway for the natural convection coolant flow.

4.2.2 Control Rods

Three motor-drive standard control rods (one safety, one shim, and one regulating) and one electro-pneumatic transient rod, control reactor power during steady-state operation. The control rods are also part of the reactor safety system as they provide reactor shutdown. The drive systems are all independent and a malfunction in one would not affect insertion or withdrawal of any other. The control rods pass through and are guided by the top and bottom grid plates (see **Figure 4-4**). The rod locations are shown in **Figure 4-5**. The stainless steel clad control rods are 43 inches (109 cm) long and 1 3/8 inches (3.49 cm) in diameter. A standard control rod is shown in the withdrawn and inserted positions in **Figure 4-6**. All three of the standard control rods have a stroke of approximately 15 inches (38.1 cm).

The upper section of the three standard rods is graphite; the next 15 inches (38.1 cm) is graphite impregnated with powdered boron carbide which provides neutron absorption; the follower section consists of [redacted] of uranium zirconium hydride fuel (8.5 wt% Uranium with about [redacted]); the bottom section is 6 ½ inches (16.51 cm) of graphite.

The fourth control rod, shown in **Figure 4-7**, is the transient rod. It has two functions: (1) it acts as a safety and/or control rod in the steady-state mode of operation, and (2) it is pneumatically driven from the core for the square wave and pulse modes of operation. The aluminum clad transient rod is 37 inches (94 cm) long with a 1 ¼ inches (3.18 cm) diameter. The borated graphite section is 15 inches (38.1 cm) long. Unlike the standard control rods, the transient rod has an air-filled follower that is ~21 inches (~ 53 cm) long. The transient rod is guided vertically through the core by a perforated thin-walled aluminum guide tube that passes through the upper and lower grid plates and screws into the safety plate. The transient rod has a stroke of approximately 15 inches (38.1 cm).

Rack-and-pinion drives (see **Figures 4-8 and 4-9**) are used to position the shim rod, the regulating rod, and the safety rod. Each drive consists of a variable speed, reversible servo motor/resolver; a magnetic rod-coupler; and a rack-and-pinion gear system. The pinion gear

[redacted]

[redacted]

[redacted]

[redacted]

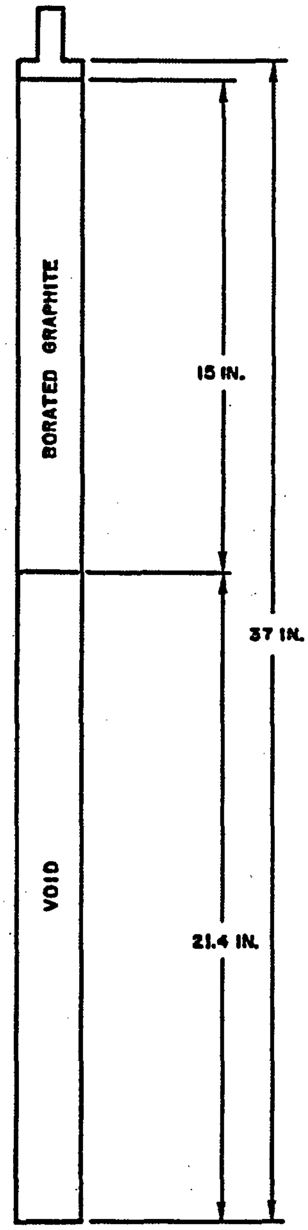

Figure 4-7 A Transient Control Rod

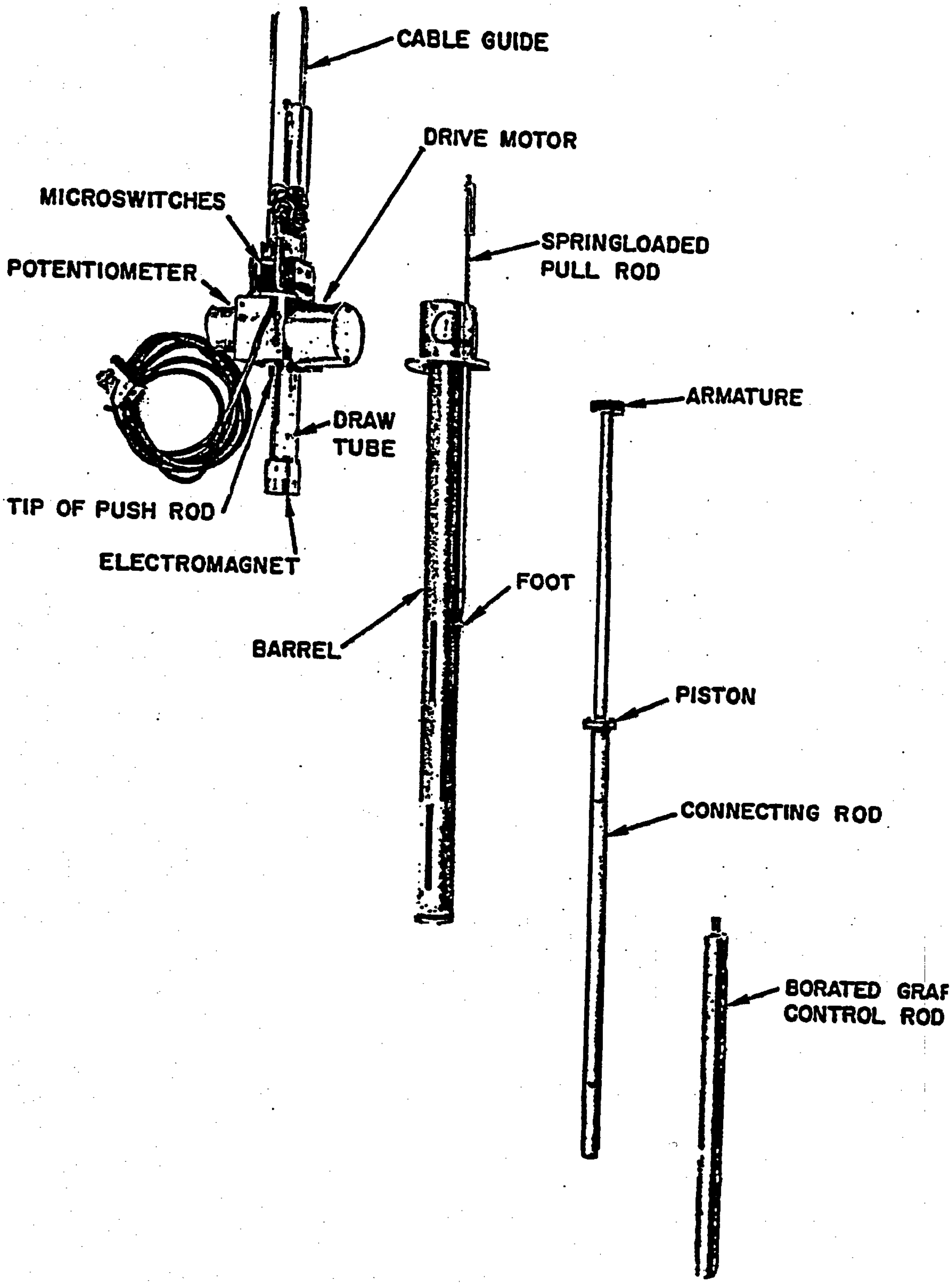

Figure 4-8 A Rack-and-Pinion Control Rod Drive

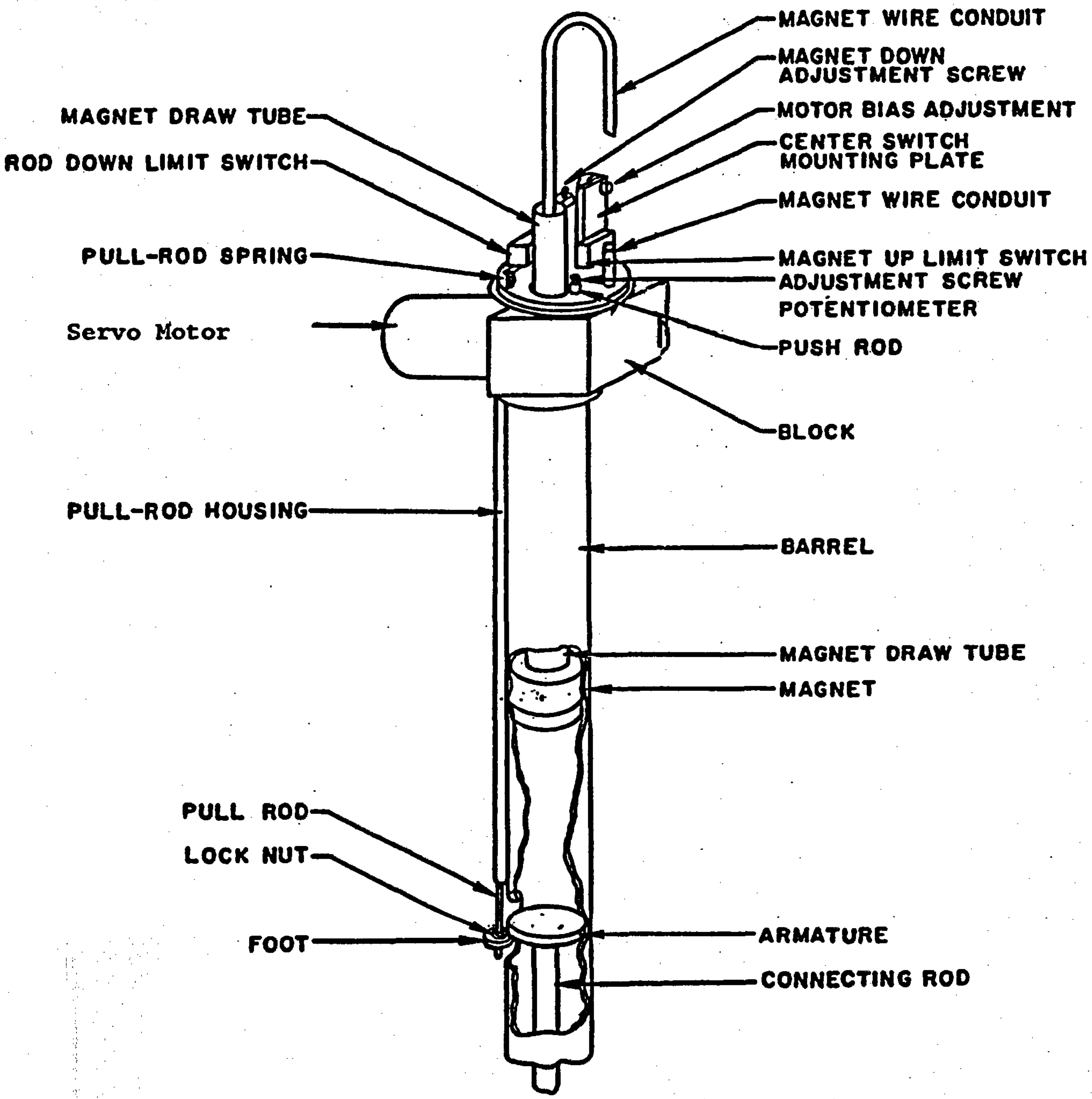

Figure 4-9 Rod Drive Mechanism, Showing Components and Adjustment Locations

engages a rack attached to a draw tube supporting an electromagnet. The magnet engages an iron armature attached above the water level to the end of a long connecting rod that terminates at the lower end in the control rod. The magnet, its draw tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the pool water level with the lower end of the barrel serving as a mechanical stop to limit the downward travel of the control rod assembly. Part way down the upper portion of the connecting rod, i.e., below the armature, there is a piston that travels within the barrel assembly. Because the upper portion of the barrel is well ventilated by large slotted openings, the piston moves freely in this range, but when the piston is within 2 inches (5.08 cm) of the bottom, its movement is restrained by the dashpot action provided by the graduated vents in the lower end of the barrel. This dashpot action reduces bottoming impact when the rods are dropped by a scramming action. The control rod is withdrawn from the core by the rotation of the motor shaft when the electromagnet is energized. When the reactor is scrammed, the electromagnet is de-energized and the control rod drops into the core by gravitational force.

The drive motors for all of the control rods are variable speed, reversible servo motor/resolver/controller systems. The speed of the motors is determined by the jumper positions within the controller and the analog velocity signal to the controller from the Protection Control and Monitoring System (PCMS) of the console (see section 7.6.1.3). The maximum speed of each drive is tunable such that the maximum reactivity addition rate of the TS is not exceeded.

The control rod position and the drive position are mimicked on the operator display of the PCMS. The rod position indication is highly reproducible for both steady-state and pulse operation. The positions are determined by the PCMS from the resolver signal from the motor, the state of the drive end of travel switches and the state of the control rod down switches. The rod drives may be "homed" under certain conditions to a known position.

To allow transient operation with the fourth control rod, use was made of a pneumatic-electro-mechanical drive system to allow ejection of a predetermined amount of the transient rod from the core (see **Figures 4-10, 4-10a and 4-11**). The pneumatic portion of the pneumatic-electro-mechanical drive, referred to herein as the "transient rod drive," is basically a single-acting pneumatic cylinder. A piston within the cylinder is attached to the transient rod by means of a connecting rod. This piston rod passes through an air seal at the lower end of the cylinder. Compressed air is admitted at the lower end of the cylinder to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke, the piston strikes the anvil of a shock absorber. This piston is thus decelerated at a controlled rate during its final inch of travel. This action minimizes rod vibration when the piston reaches its upper limit stop.

An accumulator tank mounted on the movable reactor bridge stores the compressed air that operates the pneumatic portion of the transient rod drive. A three-way solenoid valve, located in the piping between the accumulator tank and the cylinder, controls the air supplied to the pneumatic cylinder. De-energizing the solenoid valve interrupts the air supply and relieves the pressure in the cylinder so that the piston drops to its lower limit by gravity. With this operating feature, the transient rod is inserted in the core except when air is supplied to the cylinder.

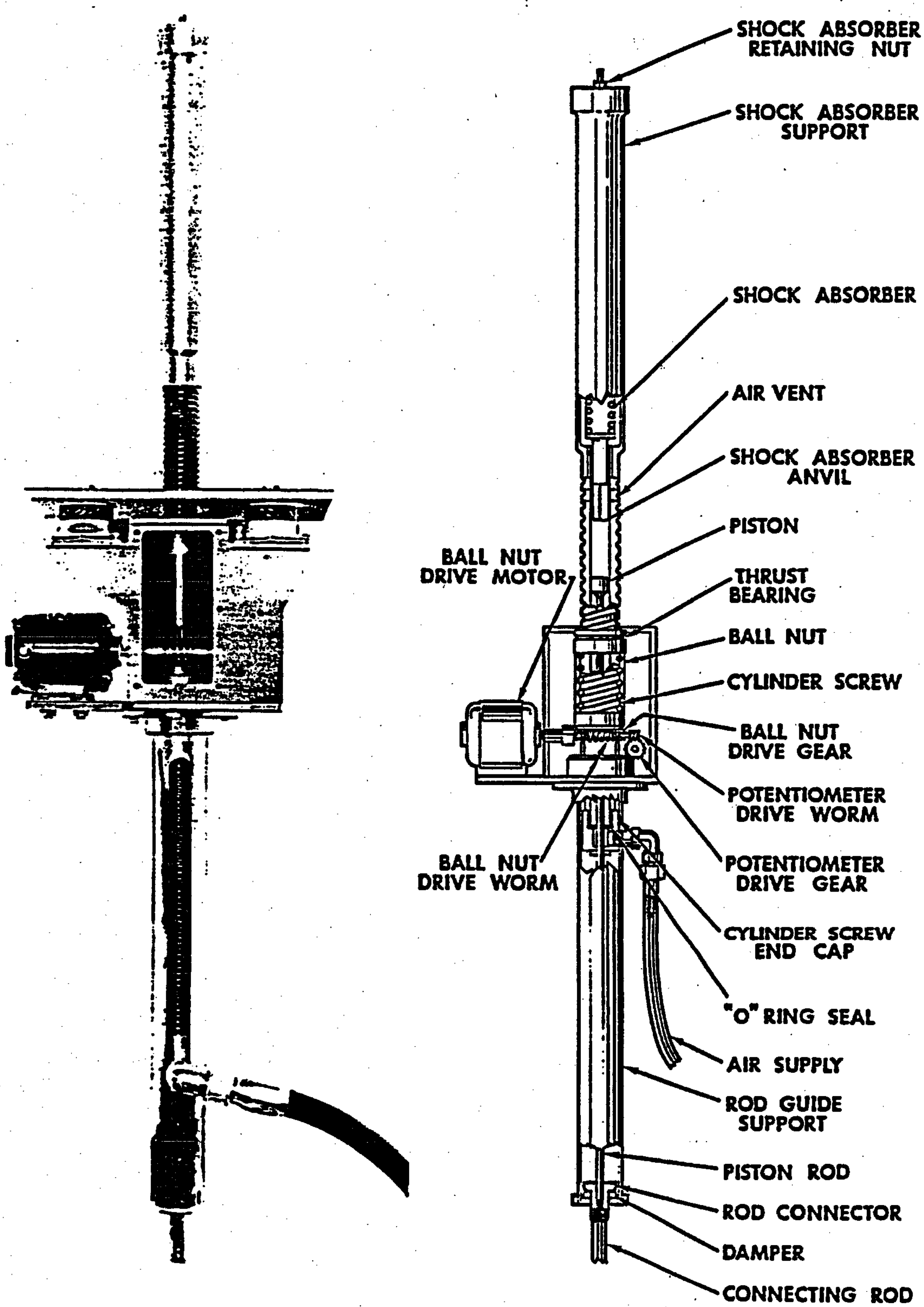

Figure 4-10 The Transient Rod Drive

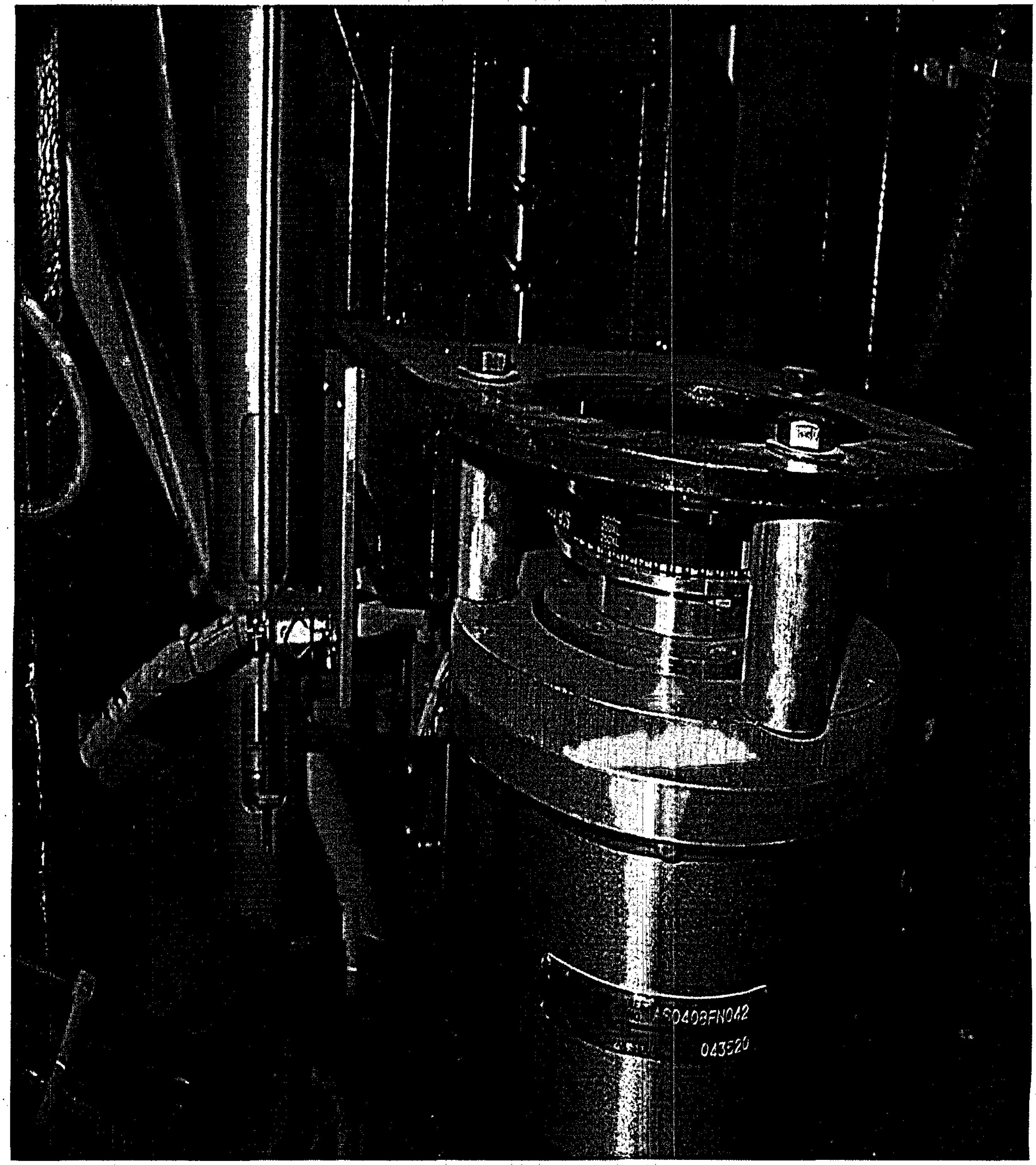

Figure 4-10a   The Transient Rod Drive

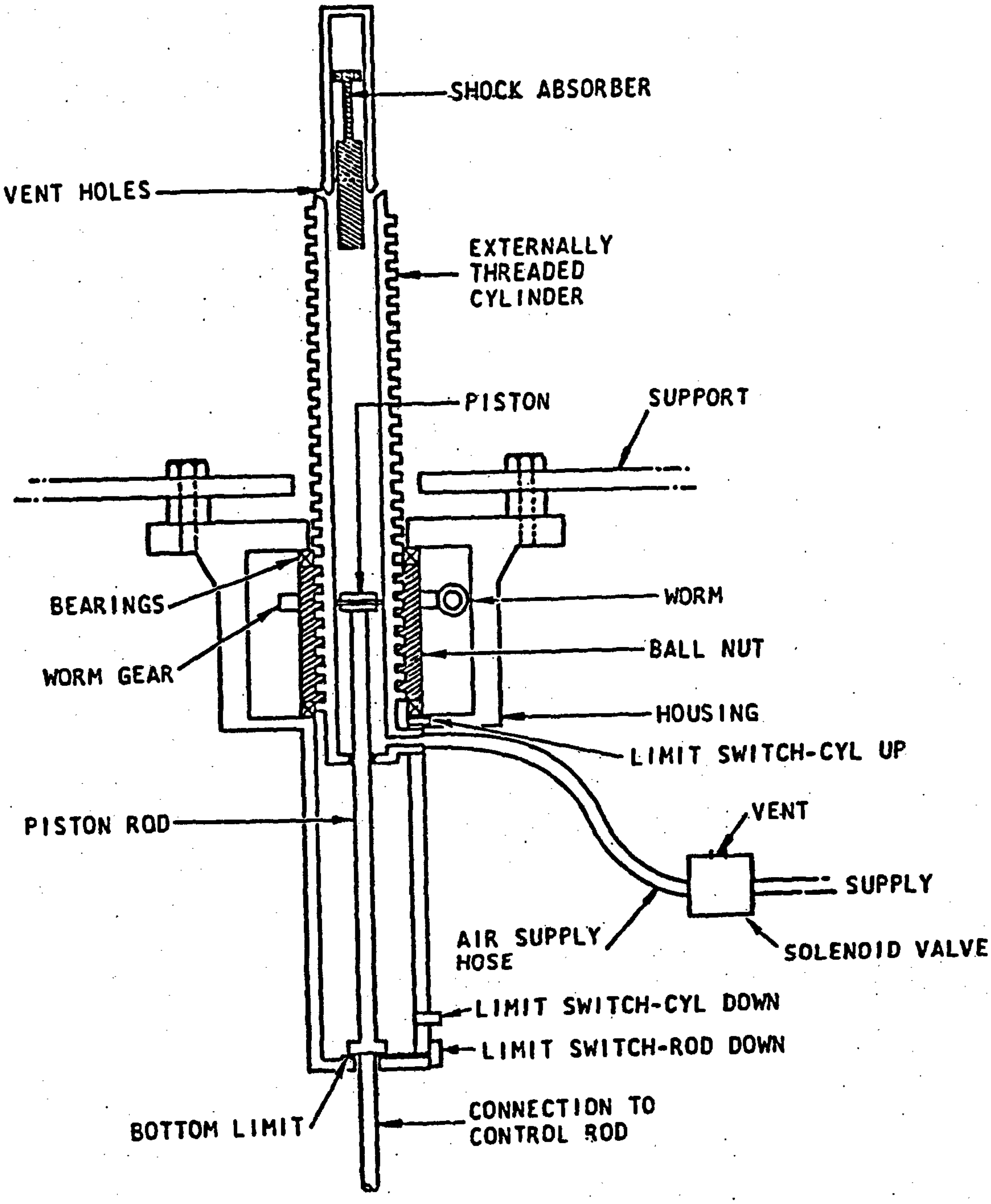

Figure 4-11 Transient Rod Drive Mechanism

Applying air to the transient rod cylinder by depressing the TRANSIENT ROD FIRE button prior to moving the cylinder allows the transient rod to be used as an ordinary control rod. Pre-positioning the transient rod cylinder and then applying air allows the transient rod to be used for square wave and pulse operation.

The electromechanical portion of the transient rod drive consists of a motor, a ball-nut drive assembly, and the externally threaded air cylinder. During electromechanical operation of the transient rod, the threaded section of the air cylinder acts as a screw in the ball-nut assembly. These threads engage a series of balls contained in a ball-nut assembly in the drive housing. The ball-nut assembly is in turn connected through a belt drive to an electric motor. Note that **Figure 10** shows the original General Atomics motor setup whereas **Figure 4-10a** shows the modification for the 1990 control system upgrade. The cylinder may be raised or lowered independently of the piston and control rod by means of the servo drive. Adjustment of the position of the cylinder controls the upper limit of the piston travel, and hence controls the amount of reactivity inserted for a pulse or square wave.

A system of limit switches similar to that used with the standard control rod drive is used to indicate the position of the air cylinder and the transient rod. Two of these switches, the Drive Up and Drive Down, are actuated by the cylinder. A third limit switch, the Rod Down switch, is actuated when the piston reaches its lower limit of travel.

The safety and transient rods are located in the C-ring of the reactor core while the shim and regulating rods are located in the D-ring (see **Figure 4-5**). Therefore, while the safety, shim and regulating rods are identical, the core location gives the safety rod greater reactivity worth. The worth of each control rod is measured on a frequency as required by the TS to ensure that the required shutdown margin is available to satisfy "single stuck rod criteria" and to provide an accurate means for determining the core excess reactivity, maximum reactivity, insertion rates, and the reactivity worth of experiments inserted into the core. Rod drive speeds both up and down and scram times for each control rod are measured on a frequency as required by the TS to ensure that no anticipated range of transients for the TRIGA is exceeded.

Scram logic and circuitry is discussed in section 7.2.3.1; interlocks and inhibits on rod withdrawal and trip and initiation systems are discussed in section 7.2.2.2. Scram insertion times are discussed in section 4.5.1.

4.2.3 Neutron Moderator and Reflector

In the TRIGA, a zirconium-hydride moderator is used with the uranium fuel to form a solid metal alloy fuel. Since the two components are alloyed, any sudden increase in power heats the fuel and moderator simultaneously. The higher temperature instantly makes the moderator less effective thus contributing a negative temperature coefficient in addition to the negative Doppler coefficient of the U-238 component of the fuel. Since the feedback occurs instantaneously, the effect is known as a "prompt negative temperature coefficient" and allows for an operation called pulsing the reactor. During a pulse, a control rod is ejected from a critical core (< 1 kW(th)) allowing the reactor power to go safely to powers of ~ 2000 MW(th), although for only milliseconds of time (time depends on the pulse magnitude) because of the prompt negative temperature coefficient.

While the pool water is the normal reflector for the TRIGA core, graphite reflector elements may occupy the grid positions not filled by fuel-moderator elements and other core components. The graphite reflector elements are canned in aluminum and have aluminum end fixtures and spacer blocks. These elements are of the same dimensions as the fuel-moderator elements, but are filled entirely with graphite. Each graphite reflector element weighs 2.8 pounds (1.27 kg) and is anodized after assembly. The spacer blocks have a blue anodized finish to make the graphite dummy elements easily distinguishable from fuel-moderator elements. When properly installed in the core, the tops of the triangular spacer blocks of the graphite elements are level with the top of the top grid plate. Since the graphite elements are designed for use as reflectors on the core periphery, any swelling that could occur would not inhibit control rod performance or have any other deleterious effect on the reactor core.

4.2.4 Neutron Startup Source

The start-up source used in the PSBR is a [redacted] americium-beryllium (Am-Be) neutron source doubly encapsulated in type 304L stainless steel. The source is contained in a water-filled aluminum source holder that has outside dimensions similar to standard fuel-moderator elements. The source holder can be positioned in any fuel element position in the core to provide neutron indication for reactor instrumentation for reactor startup. [redacted] a daily sensitivity check of the safety system's wide range neutron detection channel, as required by the TS.

Am-241 has a half-life of 433 years, and in the process of radioactive decay emits an alpha particle and gamma ray. When the alpha particle is absorbed by Be-9, a high energy 5.49 Mev neutron is emitted. The long half-life provides for a very stable neutron population for a cold reactor core. The source strength has been demonstrated to provide an adequate neutron induced signal for reactor startup. The power generated in the source at 1 MW(th) is less than 10 watts due to the fissioning of Am-241 (PSBR Amendment Request, April 10, 1986).

4.2.5 Core Support Structure

The reactor bridge is mounted on four wheels and can be moved north-south on rails that are bolted to the top surface of the pool walls. The reactor fuel grid plates are supported by a suspension tower. The tower is mounted to a trolley underneath the reactor bridge. The reactor tower is mounted to the trolley by means of a bearing assembly that allows the tower to be rotated. The trolley can be moved east-west. Movements of the bridge assembly north-south and the tower east-west are controlled by hand and the speed of the movement of each is limited by a high gear ratio hand wheel. Two vises clamp the bridge to the rails and the hand wheels are chained and padlocked so that the bridge assembly and tower cannot be moved during operation. Clamps also prohibit core rotation and east-west trolley movement during operations.

The suspension tower is a welded assembly of 2 inch x 2 inch x ¼ inch (5.08 cm x 5.08 cm x .64 cm) angle aluminum. The upper end of the tower is bolted to the reactor bridge trolley and the tower extends 22 feet and 8 inches (6.91 m) below the bridge floor. A 19 inch x 29 inch x 3 inch (48.3 cm x 73.7 cm x 7.6 cm) grid plate is bolted to the vertical angles of the bottom of the tower. Equipment supported by the suspension tower includes: the core assembly and grid plates, a central thimble, a GIC detector for the Power Range Monitor, a fission chamber for the Wide Range Monitor, other auxiliary power detectors, the N-16 diffuser system plumbing and spray nozzle, an external neutron source, a pneumatic transfer system terminus, vertical irradiation tubes, and core lights.

The grid plate arrangement is shown in **Figure 4-4**. The aluminum bottom grid plate supports the weight of the fuel and has fuel element locating holes over its entire surface; the holes are arranged in a hexagonal pattern. Additional holes with a diameter of ¼ inch (.64 cm) in the bottom grid plate provide a path for natural convective coolant flow. A 12 inch x 16 inch x 1 ¼ inch (30.5 cm x 40.6 cm x 3.2 cm) aluminum safety plate is suspended approximately 12 ¾ inches (~32.4 cm) below the bottom grid plate to prevent the control rods from dropping out of the core should their mechanical connections fail.

The aluminum top grid plate is 5/8 inch (1.59 cm) thick, and covers only a portion of the bottom grid plate area, so that experiments can be conveniently mounted on the bottom grid plate immediately adjacent to the active core. Holes approximately 1 ½ inch (~3.81 cm) in diameter in the top aluminum grid plate position the fuel elements and control rods. Small holes at various positions in the top grid plate permit insertion of wires and other small devices into the reactor for in-core measuring purposes. The following special in-core experimental facilities are available in the top grid plate:

- A central thimble 1.33 inch (3.38 cm) inside diameter
- A central removable section of top grid plate provides a 4.12 inch (10.47 cm) diameter hole (15.85 sq.inches or 102.36 sq.cm cross section).
- Two removable triangular sections of the top grid plate provide circular openings with 2.4 inch (6.1 cm) inside diameters.
- A large removable side section of the top grid plate.

Electrical power and control circuit wiring are supplied to the bridge via a floor trough by a cable arrangement that allows for reactor bridge movement. Four control rod drive motors, three area radiation monitors, the N-16 diffuser pump, a jib crane, and an accumulator tank for transient rod air, are all items supported by a bridge assembly which spans the pool as shown in **Figure 4-1.**

4.3 Reactor Pool

The reactor pool is approximately 30 feet (9.14 m) long, 14 feet (4.27 m) wide, and 24 feet (7.32 m) deep. The pool is constructed of steel reinforced concrete. [redacted]

The inside of the pool wall is coated with epoxy to form an epoxy pool liner. The pool is surrounded by earth fill with the exception of the south side. The neutron beam laboratory exists outside the south side of the pool at the elevation of the reactor core. Additional shielding is provided by 3 ½ feet (1.07 m) of high density concrete on the outside of the pool wall in this room.

The PSBR reactor pool has one advantage over most tank type facilities, in that the pool is partitioned by a divider wall, that can accommodate an aluminum gate. Thus, biological shielding can be provided for the reactor in either end of the pool in case of a pool leak, or a pool drain for maintenance or experimental facility modification. A rubber gasket on the gate faces the drained side of the pool, and pressure from the filled side of the pool provides pressure for a seal. The total pool volume is approximately 71,000 gallons (2.68×10^5 ℓ), with the large side holding about 41,000 gallons (1.55×10^5 ℓ) and the small side about 30,000 gallons (1.14×10^5 ℓ). The large volume of water provides a large drain time buffer for mitigating actions if a leak were to develop. [redacted]

Seven neutron beam ports penetrate the pool wall into the neutron beam laboratory to provide access to neutron beams (see section 10.2.1). [redacted] Four other pool wall penetrations exist, two located approximately 10 feet (3.05 m) above the pool floor to serve the pool recirculation loop and two located approximately 17 feet (5.18 m) above the pool floor for the heat exchanger.

Manually operated valves in the demineralizer room allow pool isolation from the recirculation system and primary heat exchanger system if a leak develops in those systems (see **figure 5-1**). The primary heat exchanger has additional pneumatically operated valves (when the pump is secured) to further isolate the pool water from the environment (secondary heat exchanger).

Some pool leaks occurred during early years of the facility's operation, but appropriate repairs to the concrete were made. The pool interior was also coated with an epoxy compound in the 1970s. No leaks have occurred since that time. Water data is taken daily, when the reactor is operated, to track any unusual losses of water beyond normal evaporative losses. All seven beam port flange connections in the interior pool had new flexitallic gaskets installed in 1997 or 1998 during pool drains for modifications to beam port #4 and beam port #7. Two separate pool level float switches initiate alarms to the DCC-X control computer.

The water table (see section 2.4) is well below the reactor pool, and therefore no external forces should be applied to the pool from this source.

PSBR standard operating procedures control positions in the pool where reactor operation is allowed to limit activation and damage to pool concrete, rebar, and divider wall channel iron. Additional pool wall shielding is provided around beam port #4 to protect the pool wall and limit activation.

[redacted]

There are TS concerning pool water level, pool water level alarm, pool conductivity, pool conductivity alarm, pool temperature, pool temperature alarm, and emergency-fill fire hose inspection. The justifications for these specifications are given in the bases of those TS in Chapter 14 (see sections 3.3.1, 3.3.2, 3.3.4, 3.3.5, 3.3.6 for LCOs and sections 4.3.1, 4.3.2, 4.3.3, 4.3.4 for surveillances).

4.4 Biological Shield

The principal biological shield for the 1 MW(th) PSBR TRIGA reactor is the pool water. Forty years of TRIGA reactor operation at its licensed steady state power of 1 MW(th) as well as during pulse operation indicates adequate biological shielding. (see section 11.1.5). The effectiveness of the biological shield is enhanced by a N-16 pump (see section 5.6), that diffuses the normal natural convective flow from the core and prolongs the time for the N-16 radiation to reach the pool surface.

No "hot spots" exterior to the reactor pool exist during reactor operation in any of the approved operating locations. Additional biological shielding on the neutron beam lab side of the pool wall was discussed in section 4.3. Except for the intended neutron beams exiting the beam ports, radiation levels in the neutron beam laboratory from the operation of the reactor are not measurable with standard radiation survey equipment.

Because of the reactor core's flexibility of movement discussed in section 4.2.5, a standard operating procedure exists for qualifying new operating positions in the reactor pool. This procedure considers such items as any unusual stresses placed on the control rods ability to scram, any radiation hazards introduced because of N-16 production, voids in biological shields or earth fill, radiation streaming or air activation in air-filled cavities or tubes, or activation of pool walls or other items in the pool.

Several sources of water are available for adding water to the pool to maintain the biological shield (see section 5.5).

4.5 Nuclear Design

4.5.1 Normal Operating Conditions

The 1 MW(th) steady-state (~ 2000 MW(th) pulse) TRIGA has the "inherent safety" that distinguishes TRIGA reactors, made possible by the prompt negative temperature coefficient of the reactor fuel. Approximately 65% of the prompt negative temperature coefficient of a TRIGA core comes from the “cell effect” or temperature dependent disadvantage factor, 15% from Doppler broadening of the U-238 resonances, and 20% from temperature dependent moderation and leakage associated with the water moderator/reflector.

TS 3.1.2, 3.1.3, 3.1.4, 3.2.6, and 3.2.2 (see chapter 14) limits are in effect for core excess, shutdown margin, transient rod worth, rod drop times, average reactivity insertion rate in manual mode, and maximum reactivity insertion in automatic mode. The bases of the TS provide the background or reason for the limit. The latest values at the time of the renewal submission for loading #52 are summarized in **Table 4-1**.

Figure 4-12 shows a typical TRIGA core loading (ldg 52). Core fuel arrangements are matched to experimental needs. Predictive computer analyses for safety implications are confirmed by experiment as appropriate. The only TS safety limit for the TRIGA is fuel temperature (see Chapter 14). The computer codes and confirming experiments as appropriate assure that the instrumented fuel element is in the location of MEPD (Maximum Elemental Power Density), and that the MEPD does not exceed 24.7 kW(th) per element for non-pulse operation assuming a maximum Normalized Power (NP) of 2.2. The code analysis also assures that the normalized power does not exceed 2.2 for any core loading with a maximum allowed TS pulse worth; the NP determines the peak pulse temperature for the maximum pulse. **Figure 4-13** shows the NP values for elements in core loading 52. The maximum NP is1.724 for the element in core position I-10, with a corresponding MEPD of 16.54 (see **Table 4-2**).

Following staff safety evaluation, permission to go to an unknown loading is approved by the facility Director. Once safety parameters are confirmed by procedure and measurement as appropriate, the loading description and number are recorded in the Master Core Loading Book and that loading becomes a "known loading". Permission to go to a "known loading" can be granted by the Director or Associate Director for Operations.

TABLE 4-1

Core Loading #52 Clean Critical 6/20/05

Control Rod	Total Rod Worth ($)	Worth Removed ($) at 50 watts	Worth Remaining (Core Excess) at 50 watts	Rod Drop Times (sec)	Average Reactivity Insertion Rate (cents/sec) Manual	Maximum Reactivity Insertion Rate (cents/sec) Auto
Transient	$2.88	$1.62	$1.26	0.658	6.5	6.8
Safety	$3.98	$2.32	$1.66	0.435	7.2	10.6
Shim	$2.93	$1.65	$1.28	0.458	9.0	13.7
Regulating	$3.01	$1.70	$1.31	0.442	9.45	14.1
Sum	$12.80	$7.29	$5.51			

Core Excess = $12.80 - $7.29 = $5.51 (TS limit is $7.00)
Shutdown Margin = $7.29 - $3.98 = $3.31 (TS limit is $0.25)
Transient Rod Worth = $2.88 (TS limit is $3.50)
Rod Drop - see above (TS limit is 1 second)
Manual Mode average reactivity insertion rate - see above (TS limit is 90 cents/sec for either the Safety, Shim or Regulating rods)
Automatic Mode maximum reactivity insertion rate - see above (TS limit is 90 cents/sec for the sum of the Safety, Shim and Regulating rods)

C25

[redacted]

[redacted]

C26

The TS are in effect for maximum normalized power (NP), maximum elemental power density (MEPD), and peak fuel temperature. These values and other operating parameters are summarized for loading #52 in **Tables 4-2 through 4-4.**

TABLE 4-2
Core Loading #52 1 MW(th) Operation 8/22/05

Control Rod	Total Rod Worth ($)	Worth Removed ($) at 50 watts	Worth Removed at 1 MW(th)	Reactivity Loss at 1 MW(th)
Transient	$2.88	$1.64	$2.47	
Safety	$3.98	$2.34	$3.46	
Shim	$2.93	$1.65	$2.52	
Regulating	$3.01	$1.71	$2.59	
Sum	$12.80	$7.34	$11.04	$3.70
Average Measured Peak Fuel Temperature at 1 MW(th) = 498 °C (TS limit is 650 °C)				
Maximum Predicted Normalized Power (NP) = 1.724 (TS limit is 2.2 if instrumented element is in the position of MEPD)				
Maximum Predicted Elemental Power Density (MEPD) = 16.54 (TS limit is 24.7) (based on 102 elements and 3 fueled control rod followers - equivalent to 104.25 elements)				

TABLE 4-3
Core Loading #52 Pulse Operation 8/17/05

Transient Rod Peak Worth	$2.88 (TS limit is $3.50)
Peak Power for Maximum Transient Rod Worth	1316 MW(th) (no TS limit)
Average Measured Peak Temperature for Maximum Transient Worth Pulse	453 °C measured

TABLE 4-4
Experimental Facility Worths (Core Loading #52 on 5/16/05, except as noted)

Experimental Facility	Reactivity Worth ($)
Fast Neutron Irradiator (FNI) - 4" side	- 31 cents
Fast Flux Irradiator (FFT)	- 15 cents
D_2O Tank	+ 46 cents
Rabbit I	- 7 cents
DT1/DT2 Irradiator vertical tubes (11/14/03 - ldg 51)	- 10 cents (both tubes)
2" x 6" rectangular vertical tube (bare)	- 22 cents
2" x 6" rectangular vertical tube (cadmium lined)	- 54 cents
1.5' x 1.5' Air filled Aluminum Box (7/25/95 - ldg 48)	- 80 cents

In 1995, in preparation for an experiment, a 1.5 feet x 1.5 feet (0.46 m x 0.46 m) air-filled aluminum box was constructed. The measured reactivity worth at the core face was - 80 cents. This reactivity worth bounds any possible reactivity changes due to flooding of any conceivable experiment facility. This worth is well below any limits on experiments in the TS, section 3.7.

4.5.2 Reactor Core Physics Parameters

The design and operating characteristics of the standard TRIGA cores are well known as is the inherent safety characteristic of this class of reactor.

Prompt Negative Temperature Coefficient	-1.4 E-4 $\Delta k/k/^{o}C$ (average)
ß effective	0.007
Prompt Neutron Lifetime	38 µsec

4.5.3 Operating Limits

TS 3.1.1, 3.1.2, 3.1.3, and 3.1.4 (Chapter 14), describe steady state and maximum power level limits, fuel temperature limits, excess reactivity limits, shutdown margin requirements, and pulse insertion limitations. The bases for these specifications provide the background or reason for the limits.

TS 3.1.5 describes core configuration limitations that describe the type of fuel and configuration allowed, and limits on MEPD and NP. The bases for these specifications provide the background or reason for the limits.

TS 3.2.1 describes the minimum operable control rods required. The basis for this specification provides the background or reason for the limit.

Section 4.0 of the TS provides the surveillance requirements to assure the above limits are met.

TS 2.0 describes the Safety Limit and Limiting Safety System Setting for the fuel temperature, with the basis providing justification for those limits.

Section 13.1.2 of this application discusses a $5 ramp analysis done for the AFRRI reactor by General Atomics. The conclusion was that this $5 ramp insertion was even less consequential than the reactivity accident analyzed in section 13.1.2.

4.6 Thermal-Hydraulic Design

The PSBR operates at 1 MW(th) steady state and is cooled by natural convection. Cooling water enters the core from the perimeter of the core region and through holes provided in the bottom grid plate. The triangular or fin shaped spacer blocks (depending on the vintage of the element) on the upper end of the fuel elements are used for positioning as well as providing a means for water to flow up through the core and out the top grid plate. Because of the natural convection cooling, the PSBR can be operated in a water-filled pool with no direct coupling between the core and the heat exchanger system. Since no connection to a forced circulation system is required, the reactor can be operated at any position in the pool. Many TRIGA reactors have been operated at power levels up to 1.5 MW(th) with natural convection cooling without any cooling problems noted.

5. REACTOR COOLANT SYSTEMS

5.1 Summary Description

The reactor pool water serves as the primary coolant and provides for natural thermal-convection cooling of the 1 MW(th) PSBR TRIGA reactor. It also serves as a biological shield, neutron moderator, neutron reflector and heat sink. The pool is open to the atmosphere of the reactor bay. The water is filtered and demineralized via the pool recirculation loop. Operating the reactor at 1 MW(th) can increase the water temperature of the 71,000 gallon (2.68×10^5 ℓ) pool several °C per hour. Therefore, a 1 MW(th) heat exchanger is available as the secondary coolant system to maintain the pool temperature in a range where evaporative losses are kept to a minimum and the integrity of the demineralizer system resins is not challenged.

5.2 Primary Coolant System

The 71,000 gallons of reactor pool water provide a very large heat sink for the 1 MW(th) TRIGA reactor. Natural thermal-convection cooling of the core along with the recirculation of the pool water at a rate of 40 gallons per minute (151 ℓ/m), provides some mixing of the pool water. Several hours of reactor operation are possible without any additional cooling by the heat exchanger (see section 5.3).

Two independent float switches located in the reactor pool send digital signals to the console (pool level low alarms) when the pool level drops to the point where less than 18.25 feet (5.56 m) of water is above the top of the bottom grid plate. TS 3.3.1 and 3.3.2 (Chapter 14) provide the requirements for the minimum water level required above the reactor core and specifies a pool level low alarm. The TS 4.3.4 specifies surveillance requirements for the pool level alarm. The bases of the TS provide the background or reason for the specifications.

A float switch located in the reactor pool, sends a digital signal to the console for a higher than normal pool level and a pool level high alarm is indicated on the console.

5.2.1 Transfer of Pool Water

Occasionally, for maintenance purposes, it is necessary to drain the reactor pool. In order to provide shielding for the reactor core, stored fuel elements, and experimental apparatus, only one half of the pool is drained at a time. The pool is divided into two parts by using the reactor bay overhead crane (see section 9.7.3) to insert an aluminum gate into the partial pool divider wall gate support structures.

The water from either half of the pool can be stored in a 48,000 gallon (1.82×10^5 ℓ) aluminum hold-up tank located behind the facility. [redacted]

[redacted]

[redacted]. However, to minimize the release of radioactivity and to conserve the demineralized pool water, the pool water is transferred to the storage tank when it is necessary to drain the pool. [redacted]

5.3 Secondary Coolant System

The PSBR heat exchanger (HX) limits the temperature of the PSBR pool water (see **Figure 5-2**). Maintaining lower pool temperatures decreases pool water evaporation losses and assures temperatures are maintained below 140° F (60° C) to prevent damage to the demineralizer resins.

The actual heat removal capacity of the heat exchanger system is dependent on such variables as flow rates, pool temperature, cooling water temperature, and the system's cleanliness. Since the source of cooling water is Thompson Pond, which is fed by a ~3 million gallons/day (~1.14×10^7 ℓ/day) spring, relatively small year round variations in cooling water temperature are noted (55° F ± 2° F or ~12.8°C).

The system is composed of two loops constructed primarily of aluminum with some stainless steel parts (see **Figure 5-2**). In the primary loop, pool water is pumped through the shell side of two heat exchangers connected in series. In the secondary loop, cooling water is pumped from Thompson Pond, a spring fed pond located approximately 650 yards (594 m) southeast of the PSBR, through the tube bundle side of the two heat exchangers and then to a storm sewer which returns the water to Thompson Pond. (Some of the effluent of the secondary side of the reactor heat exchanger is used by the Combustion Engineering Laboratory in an adjacent building but their use does not have a safety impact on the reactor.) The lower quality secondary side water is passed through the tube bundles since by removing the ends of the two heat exchangers, the tube bundles can be cleaned more easily than the shell side. Penn State has a Pennsylvania Department of Environmental Protection permit for secondary water discharge.

The heat exchanger inlet piping and outlet piping penetrate the west pool wall approximately 17 feet (5.2 m) above the pool floor (about 8 feet or 2.4 m above the pump room floor). The inlet pipe to the HX (approximately 6 feet or 1.8 m below the top of the pool water) is just south of the pool divider wall and the return pipe to the pool enters just north of the divider wall (approximately 6 feet or 1.8 m below the top of the pool water). The return pipe continues down a standpipe and terminates in a perforated pipe that returns the cooled water along the bottom width of the pool just north of the divider wall.

Two secondary pumps are located at the Thompson Pond pump house. Electrical breaker and valve manipulation at the pump house determines which pump is available for service. A phone line switch in the reactor demineralizer room controls the pump whose breaker is on.

When the system is operating, the primary side has a flow rate of approximately 400 gallons/min (1.51×10^3 ℓ/min and inlet pressure of approximately 16 psig, and the secondary side has a flow

[redacted]

rate of approximately 350 gallons/min (1.33×10^3 ℓ/min) and an outlet pressure of approximately 25 psig (notice these are points of minimum pressure difference in the system). The higher secondary than primary pressure prevents any release of reactor pool water to the environment if a leak would develop in the HX. When the primary heat exchanger pump is secured, two pneumatically operated valves close to isolate the primary side since if the secondary side is also off, the primary pressure due to the height of the pool is higher than the secondary pressure.

If a leak between the primary and secondary occurred while both were operating, the pool level would increase. A pool level high alarm is discussed in section 5.2. Periodic secondary side grab samples are taken to monitor for any radioactivity above background in the secondary water (see section 11.1.1.2).

A differential pressure transmitter outputs an analog signal to the console. If the secondary outlet pressure is not greater than the primary pressure by a preset amount or if power to the transmitter fails, then the console will indicate a status alarm message.

5.4 Primary Coolant Cleanup System

The 71,000 gallons (2.68×10^5 ℓ) of reactor pool water are filtered and demineralized by way of the pool recirculation loop. Cuno filters in the system contribute to pool water cleanliness by removing particulates greater than 5 microns in size. The mixed bed demineralizer maintains low pool water conductivity and an acidic pH to:

- provide conditions favorable to the stainless steel fuel and aluminum reactor components by minimizing corrosion of fuel and control rod cladding and essential water handling system components
- limit the concentration of particulate and dissolved contaminants that could be made radioactive by neutron irradiation
- maintain a high transparency of the water for safe observation of reactor equipment and activities

TS 3.3.5 discusses a conductivity limit, and TS 4.3.3 describes the frequency requirement for measurements (see Chapter 14). The bases of the TS provide the background or reason for the limit and frequency of measurements.

A PSBR procedure describes sampling frequency and action limits for the pool water radioactivity (see section 11.1.1.2). Pool conductivity, pH, and tritium are also measured periodically in a laboratory using a grab sample of pool water. There are no specific TS limits on gross radioactivity, tritium, or pH. 10 CFR 20 limits would apply for any liquid radioactive effluent released to the environment.

The ratio of the amounts of the demineralizer resins maintains an acidic pH for the pool water, an atmosphere that is favorable for the aluminum and stainless steel in the reactor pool. Pool water pH measurements are taken approximately weekly. The average pH reading over the three year period of October 2002 through September 2005 was 5.43, with a low of 5.12 and a high of 5.71 during the period.

The demineralizer system is designed for in-house re-generation, and a liquid evaporator is available to handle the liquid chemical waste from that process. Presently, rather than regenerate the expended resins, they are replaced with new resins. The resins are replaced about every 5 years. The spent resins are placed into 50 gallon (189 ℓ) drums, solidified, and disposed of as solid radioactive waste. The cuno filters are removed, dried in the evaporator building, and placed in 50 gallon (189 ℓ) drums for disposal as solid radioactive waste. Cuno filter and resin replacement is according to an approved procedure. Radiation levels are not significant and normal protective clothing is sufficient to guard against low level contamination.

Figure 5-1 shows the key components of the pool recirculation loop, most of which are aluminum or stainless steel. The water enters the system by way of a surface skimmer at the south end of the pool. The loop is composed of a coarse filter, a pump, a pressure switch, gauges, cuno filters, a flow switch, a conductivity-in cell, the demineralizer, a conductivity-out cell, and outlets just below the surface of the pool. The flow rate through the system is about 40 gallons per minute. The recirculation system piping enters and leaves the pool approximately 10' above the pool floor and enters the demineralizer room about one foot above its floor level.

Conductivity cells located at the demineralizer provide continuous conductivity-in and conductivity-out measurements (see **Figure 5-1**). Exceeding the conductivity-in alarm point setting on the analyzer provides a digital input for a console alarm. Pool water conductivity is typically on the order of 0.3 to 2.0 microsiemens/cm during most of the approximate 5 year lifetime of the resins. The resins are replaced in the range of 2.0 to 5.0 microsiemens/cm.

Since the recirculation system is for cleanliness and is not a forced cooling system, there are no critical pressures or flows. The system does have a pressure switch (with a delay timer to allow the pump to come up to operating pressure) that would secure the pump if pressure was lost, but this is only to protect the pump. If system flow is lost, an alarm is indicated on the console but this is for operator information and is not a critical alarm.

[redacted]

5.5 Primary Coolant Makeup Water System

The normal loss of primary coolant is by evaporation from the open pool. The evaporation rate can vary with relative humidity (a seasonal factor), temperature of air and water, air movement, etc. Net pool gains/losses are recorded as part of the daily reactor checkout, with values corrected for water added and temperature changes.

The methods for replenishing pool water (in order of preference) are as follows:

- The effluent from the bay air conditioner provides most of the pool make-up water during the warmer months and the air conditioner thus recycles to the pool some of the evaporative losses.
- In the cooler months, pool make-up water is from the liquid waste evaporator system (see section 9.7.2). Water produced from the operation of this system is stored in an underground 6000 gallon (2.27×10^4 ℓ) fiberglass processed water tank (see **Figure 9-3**) to be used for pool make-up. Metered water can be added from this tank through the reactor pool recirculation loop and demineralizer at a rate of about 20 gallons/min (76 ℓ/min).
- Water can be adding using metered university water through the reactor pool recirculation loop and demineralizer. This water can be added at a rate of about 15 to 35 gallons/min (57 to 132 ℓ/min).
- *Thompson Pond water can be added using the secondary heat exchanger pump using a fire hose connection at a rate of approximately 300 gallons/min (1.14×10^3 ℓ/min) (independent of site electricity)
- *University water can be added [redacted] at a rate of approximately 150 gallons/min (568 ℓ/min) (independent of site electricity)
- University water via fire hydrant via fire company pumper

*TS 3.3.4 states "A source of water of at least 100 GPM shall be available either from the University water supply or by diverting the heat exchanger secondary flow to the pool". The surveillance TS 4.3.1 states "The two dedicated fire hoses that provide supply water to the pool in an emergency shall be visually inspected for damage and wear annually, not to exceed 15 months".

5.6 Nitrogen-16 Control System

The major source of radiation in the reactor bay when the reactor is operating is the production of radioactive nitrogen-16 from the action of the reactor's fast neutrons on the oxygen in the pool water. The problem is minimized by a diffuser pump that prolongs the amount of time for the nitrogen-16 to reach the pool surface, allowing for decay of much of the nitrogen-16 that has a half-life of approximately 7 seconds. Since the N-16 is diffused into the bulk pool water for decay, significant N-16 levels are not noted in the recirculation system or the heat exchanger system piping in the demineralizer room. For a discussion of personnel exposures see section 11.1.5, Radiation Exposure Control and Dosimetry.

The submersible diffuser pump hangs from the underside of the reactor bridge about one foot from the reactor tower and is about four feet underwater. Pool water is taken into the bottom of the pump and then travels down a standpipe to approximately three feet above the top rear of the core where it is discharged through a four nozzle arrangement across the core, rear to front. The pump does not significantly affect the natural thermal-convection cooling of the core.

The N-16 pump is energized automatically by the console control computer, at approximately 200 kW(th). The reactor operator can also operate the pump manually via the console, but the operator cannot override the auto start feature.

5.7 Auxiliary Systems Using Primary Coolant

There are no auxiliary systems using primary coolant water. [redacted]

6.0 ENGINEERED SAFETY FEATURES

6.1 Summary Description

The building is constructed of concrete blocks, bricks, insulated steel and aluminum panels, structural steel, and re-enforced concrete and is in general, fireproof in nature. The reactor bay serves as a confinement designed to limit the exchange of effluents with the external environment through controlled or defined pathways. The reactor bay is kept at a negative pressure with respect to the atmosphere by one of two separate ventilation systems. During normal operations the reactor bay is exhausted by at least one of the two roof exhaust fans. When the evacuation alarm is actuated, the two roof exhaust fans are automatically secured and an emergency exhaust system is automatically actuated, whereby a negative pressure is maintained on the reactor bay and the effluent is exhausted through filters to a stack that exhausts approximately 24 feet (7.3 m) above ground level. The reactor bay meets the TS definition 1.1.8, "Confinement means an enclosure on the overall facility which controls the movement of air into and out through a controlled path".

6.2 Detailed Descriptions

6.2.1 Confinement

The ~70,000 feet^3 (1900 m^3) minimum volume reactor bay is maintained at a negative pressure with respect to the remainder of the building by one of two separate exhaust systems (see **Figure 6-1**). Fresh air to the room is supplied by leaks around doors, etc. Normal ventilation of the reactor bay is by the reactor bay facility exhaust system (FES). An emergency exhaust system (EES) is also available.

The FES function is to control air flow through the reactor bay to minimize worker radiation exposure and to release the reactor room air in a controlled manner (~3000 feet^3/min or 8.5 x 10^4 ℓ/min with both fans running) where dilution and diffusion of the effluent occurs before it comes into contact with the public. Argon-41 is the only radioactive gas of significance released during the normal operation of the reactor, and is the result of the action of fast neutrons on air in the reactor pool water and in experimental apparatus. See section 11.1.1.1 for typical Argon-41 annual releases and section 11.1.5 for a discussion of personnel exposures.

The EES creates sufficient negative pressure in the reactor bay so that any movement of radioactive material from the bay would be through the system filters. Air enters the EES through a screened opening [redacted] above the bay floor. The air then passes through a pre-filter, absolute filter, and carbon filter that are mounted in a housing [redacted]. The three horsepower exhaust fan (~3100 feet^3/min or ~9.1 x 10^4 ℓ/min with system dampers completely open) is also mounted there. Flow can be reduced through the system by adjusting the manual dampers. Filtered air exhausts into an 18 inch (46 cm) diameter PVC pipe and stack. The stack travels up the east outside wall of the reactor building and exhausts at a point above the reactor bay roof (~24 feet or ~7.3 m above ground level).

[redacted]

The most likely source of significant radioactivity would be failure of fuel element cladding. The EES is normally on standby in the automatic mode. Activation of the system occurs whenever the building evacuation alarm is initiated. The system can also be activated manually from the control panel in the Cobalt-60 facility entrance lobby.

The EES control panel in the Cobalt-60 facility entrance lobby shows the operational status of the EES system. The control panel consists of four differential pressure gauges, three of which show pressure drops across each of the filters. The fourth pressure gauge shows the velocity pressure in the stack. Also located on the control panel are two pilot lights; one indicates that the system is energized, the other indicates flow in the system (by means of a flow switch). A switch that allows the system to be manually activated is also on the panel. The control panel can be viewed with binoculars from outside the reactor building perimeter fence during an evacuation.

The three stage filter system is housed in a dust-tight containment. The purpose of the low-cost pre-filter is to filter atmospheric dust that would be deposited in the more expensive absolute filter. Thus, the lifetime of the absolute filter is extended. The high-efficiency absolute filter is needed to remove particulate radiation and has a removal efficiency of 99.9% for .3 micron-sized particles and 99.99% for one micron-sized particles. The carbon filter has a high efficiency for removing fission gases, most importantly the radioiodines.

Static tips are located upstream of the pre-filter, between the pre-filter and the absolute filter, between the absolute filter and the carbon filter, and downstream of the carbon filter. These static tips are connected to three of the differential pressure gauges by copper tubing. A stainless pitot tube mounted in the stack is connected to the fourth differential pressure gauge. As the EES is operated, both the efficiency and the pressure drop across the filters increase due to loading. The filters should be changed when the initial pressure drop (normal operating range for clean filters) has approximately doubled (removal range for spent filters), which is well before the maximum design pressure drop (flag setting) across the filter is exceeded (see **Table 6-1**). Periodic checks of the filter criteria are provided by a PSBR standard operating procedure.

Table 6-1
EES Filter Criteria

	Normal Operating Range (inches H_2O)	Removal Range (inches H_2O)	Flag Setting (inches H_2O)
Pre-filter	.07	.14 - .24	0.42
Absolute Filter	.7	1.4 – 1.5	1.65
Carbon Filter	.6	1.0 – 1.1	1.15
Stack	.2		0.52

The switch on the control panel has two operational modes, auto and manual. It is not possible to disable the system with this switch. Operating the system using the manual mode has no effect on the reactor's operation or any other system.

A Power Monitor box [redacted] has three red neon lights that are lit when there is three-phase AC power to the system. In the auto mode, when an evacuation is initiated, a red fan on light on the emergency exhaust system fan control box [redacted] is lit when the system has power.

Once the EES is energized, it takes ten to fifteen seconds for the EES flow to increase enough to activate the stack flow switch that turns on the red power-on light on the Cobalt-60 lobby control panel. Shortly thereafter, the air flow will stabilize at its normal rate (and the pressure drop gauges will stabilize). A console message "Emerg Ventilation Flow On" (also actuated by the flow switch in the stack) is the positive indication to the reactor operator that the system has flow. DCC-X (reactor console digital control computer discussed in Chapter 7) also disables the FES if the EES was activated by DCC-X; manually activating the EES does not disable the FES. When the louvers of the FES close there will be an "East and West Fans Off" message on DCC-X (this is also a reactor scram, see section 7.3.1.3). When the evacuation is cleared by the operator the EES returns to the auto mode and the FES will restart automatically for any fan(s) in operation prior to the evacuation and activation of the EES.

The TS describe the requirements for the confinement and for FES and EES system operability and periodic surveillance during reactor operation and fuel movement:

- TS 5.5a describes the confinement as designed to restrict leakage and describes the minimum volume
- TS 5.5b describes the FES and EES systems, and operability during normal and alarm conditions
- TS 3.4 addresses air passages (truck door and doorways) requirements to assure a negative reactor bay pressure, when the reactor is not secured or when fuel or a fueled experiment is being moved
- TS 3.5 describes requirements for FES operation and EES operability when the reactor is not secured or fuel or fueled experiments are being moved
- TS 4.4 describes that reactor bay doors must be locked or under surveillance by an authorized keyholder
- TS 4.5 indicates the surveillance frequencies to ensure the proper operation of the FES and the EES in controlling the releases of radioactive material to the uncontrolled environment.

The basis of the TS provides the background or reason for the above TS.

Section 13.1, Accident Analysis, gives a summary of projected radiological exposures from the MHA. This information indicates that even if the EES fails to operate during the MHA, doses to the public are still within 10 CFR 20 limits.

6.2.2 Containment

Not applicable for PSBR.

6.2.3 Emergency Core Cooling System

Not applicable for PSBR.

7.0 Instrumentation and Control Systems

7.1 Summary Description

The reactor console described in the following sections (see **Figure 7-1**) was designed to provide the PSBR with safe and reliable operation in four different modes; Manual, Automatic, Square Wave, or Pulse. This system was installed in 1991 as a replacement for the original General Atomic TRIGA console. Further hardware and software updates were made in 2004. The console utilizes a reliable hardwired analog system for safety related functions. All non-safety related functions are implemented with computer technology. In this use, a safety related function or system is defined as that system or function that must remain operational "... during and following a Design Basis Event (DBE) to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines"[15].

The console utilizes state-of-the-art digital equipment for reliability, allows flexibility for upgrades and can reduce life-cycle costs. The console's man-machine interface closely parallels that which the students see in industry. The console can reduce the duplication of instrumentation by allowing isolated data extraction from the reactor equipment for experimental and laboratory use. All safety functions are met with hardwired analog equipment, with those functions then duplicated in the software system.

Two independent systems are provided for the safety, protection, control and monitoring functions. The first, the reactor safety system (RSS), provides all of the SCRAM and operational interlock functions required by the TS. The RSS is completely hardwired without making use of any software programmable equipment or devices containing embedded microprocessors. The second system, the protection, control and monitoring system (PCMS), is fully computerized employing state-of-the-art digital control and monitoring features.

The PCMS consists of two computers, Digital Control Computer X (DCC-X) and its interface equipment and Digital Control Computer Z (DCC-Z). All of the non-safety related functions necessary for operation are provided by DCC-X. DCC-Z is strictly a monitoring computer in "read only" communication with DCC-X. DCC-Z performs data logging, data display and can broadcast data to a local area network. DCC-Z is not required for reactor operation.

The console (see Figure 7-1) houses both the RSS and PCMS with the exception of field transducers and actuator devices.

Safety in depth is a design basis of the PSBR console. **Figure 7-2** is a diagram of the RSS and PCMS. Analog electronics, with its established reliability, is used for the reactor safety system, RSS. The analog RSS is the safety related envelope within which the reactor operates. The PCMS computer, DCC-X, with its flexibility and versatility, provides a software protection envelope that continuously verifies the analog RSS with redundant and additional scrams and

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[redacted]

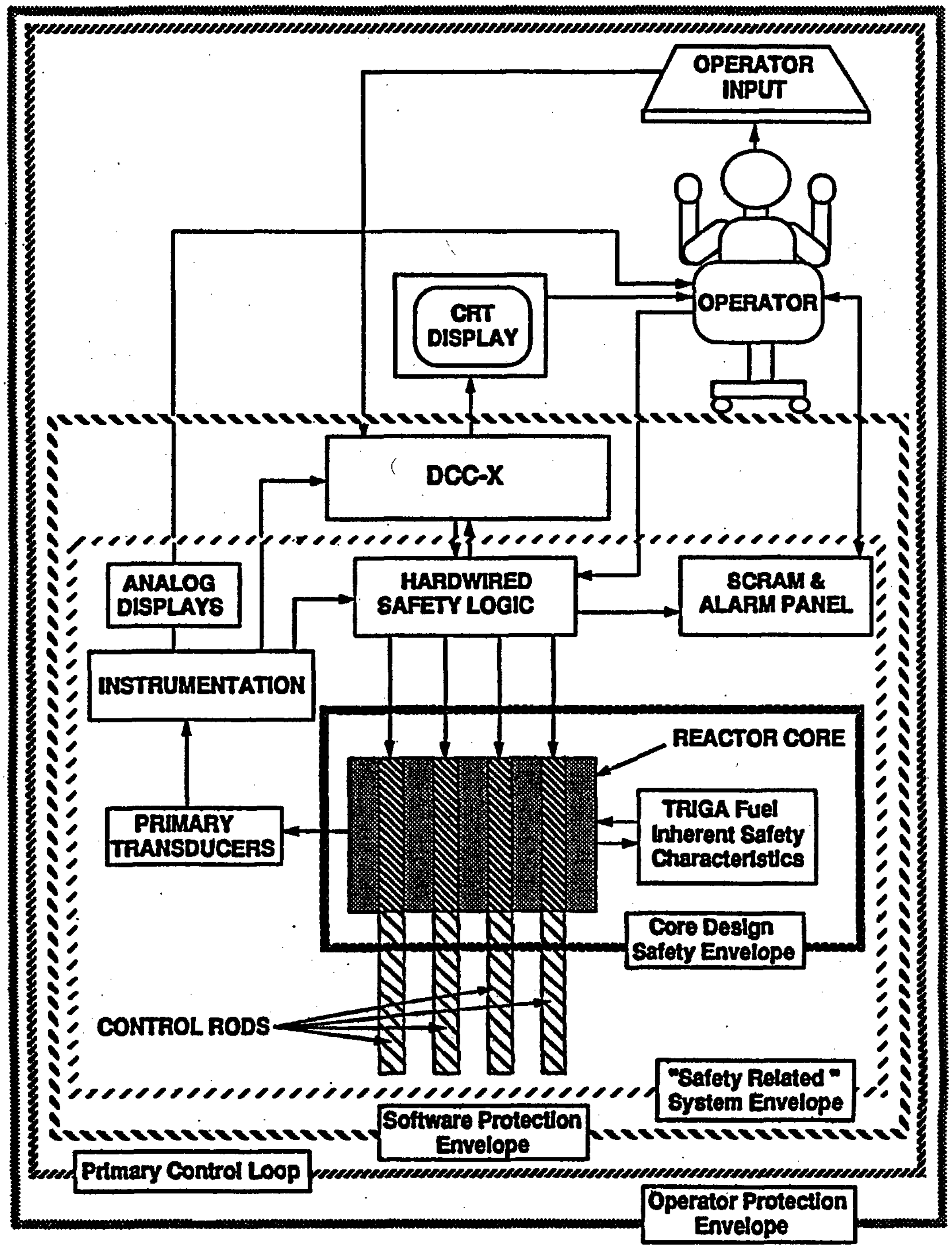


Figure 7-2 PSBR Safety, Protection and Control System

interlocks. Inside the analog safety related envelope is the core design safety envelope based on the inherent safety of the *TRIGA* fuel system. The final envelope of protection is licensed and highly trained operators using their educated judgment and properly written procedures. Any control loops operate outside the safety and protection envelope so that control cannot degrade reactor safety.

Indication of the reactor state is by the analog display devices that are wired directly from the analog reactor power and temperature instrumentation. In addition, the operator has consolidated information about the reactor state available from the LCD monitor parameter display of the DCC-X computer. The operator has the final check of the validity of the information by periodically comparing the analog display devices with the LCD monitor parameter display.

There are two power level instruments in the RSS and PCMS. The wide range power monitor uses a fission chamber and covers 10^{-10}% to 200% of 1 MW(th). The second detector, a gamma ionization chamber (GIC), is the input to the power range monitor with a range of 1 to 120% and to the pulse power monitor to a maximum of 2000 MW(th). In addition, there are two thermocouple (TC) inputs from an instrumented fuel element.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

The following are the basic design principles and philosophies used in configuring and designing the various systems.

(1) The RSS is separated from the PCMS through use of buffered devices and by physical separation to the extent possible within the console.
(2) The RSS is completely hardwired and does not contain any software programmable devices with embedded microprocessors for signal processing or actuation functions.
(3) The RSS logic is designed to fail safe on loss of power.
(4) Any enhancements, such as redundancy, to the safety functions or reactor protection functions are done through the PCMS via a DCC-X SCRAM input.
(5) The safety functions are designed to meet the single failure criterion for failures in the RSS crediting both the operable portions of the RSS and the PCMS to mitigate the failure consequences.
(6) The PCMS, consisting of the DCC-X and its interface equipment and DCC-Z, is designed to fail conservative through use of extensive self-tests and a watchdog.
(7) DCC-Z, the monitoring computer, does not perform any control actions and is buffered from the control computer, DCC-X, by use of one way data communications. All connections to external monitoring computer systems are via DCC-Z and hence these systems are also buffered from the PCMS DCC-X computer.
(8) DCC-X is designed to provide all reactor protection, control and monitoring functions necessary for safe operation.
(9) Where practical, design of human interfaces employ consistency in operation methodologies, equipment organization, labeling schemes, etc. to maintain an ergonomic interface for operation and maintenance.

7.2.2 Design Basis Requirements

7.2.2.1 Console Function Summary

The console provides the following functions:

- houses all equipment associated with the RSS and PCMS with exception of the field devices
- provides an ergonomic operator interface to control and display devices. Reference (6) was used as a human factors guide.
- serves as a desk with sufficient room for two operators
- provides EMI shielding and heat removal for the enclosed equipment

7.2.2.2 RSS Function Summary

The RSS performs the SCRAM and operational interlock functions.

<u>SCRAM Functions</u>
There are presently four control rods used in the PSBR that can be SCRAMed individually and all together by dedicated manual switches. A SCRAM of all four rods together shall occur automatically under the following conditions:

- Fuel Temperature High (LSSS).
- Reactor Power High (Fission Chamber).
- Reactor Power High (Gamma Ion Chamber).
- Loss of Detector Bias Voltage (Fission Chamber).
- Loss of Detector Bias Voltage (Gamma Ion Chamber).
- Pulse Timer timed-out.
- DCC-X Watchdog tripped.
- DCC-X SCRAM request.
- Manual SCRAM pushbutton.
- Reactor Operation Keyswitch Off.

<u>Interlock Functions</u>
The operational interlocks prevent driving of each rod and applying air to the transient rod under various circumstances. The intent of the interlocks is to prevent unintentional insertions of reactivity through improper operation of the controls governing the rod drives. The following are the interlock functions:

- prevent manual withdrawal of more than one rod when in manual or square wave modes of operation.
- prevent manual withdrawal of any rod when the fission chamber count rate is very low
- prevent movement of all rods except the transient rod when in pulse mode.
- prevent application of air to the transient if the drive is not fully down when in manual mode.
- prevent application of air to the transient rod in pulse mode if initial power is high.

All interlocks are duplicated and validated by DCC-X and alarm on failure. The reactor is SCRAMed if a failure is detected.

7.2.3 System Description

Three operational subsystems are provided to perform the necessary functions.

Reactor Safety System (RSS) is composed of the following hardware:

- A fission chamber, a pre-amp and a signal processor measure power over a 10 decade range with log of ffp, linear (linear on the PCMS LCD monitor displays, percent full power on a square root scale bargraph on the Wide Range Monitor front panel), and log-rate signal outputs.
- A gamma ion chamber with its amplifier provides linear power and pulse range power and a fuel temperature signal processor provides two fuel temperature signal outputs.
- A hardwired reactor safety system which uses relay logic to perform SCRAM and operational interlock functions.

Protection, Control and Monitoring System (PCMS)
PCMS availability is not of paramount concern and hence a single protection and control computer configuration is provided with a separate monitoring computer (see **figure 7-3**). The PCMS DCC-X control computer is designed with extensive self-testing and employs a fail conservative watchdog for high reliability against non-conservative failures.

The PCMS DCC-Z monitoring computer is buffered from the DCC-X control computer by receiving all signal data from the DCC-X computer via a one-way serial link. (A Cyclic Redundancy Check Code or CRC protocol is used to reduce errors in transmission). No failure in the monitoring computer can propagate back to the control computer. The monitoring computer also employs extensive self-testing to assure high reliability of information display. Bad data is highlighted through status annunciation.

The monitoring computer is provided with hardware and software for connection to a local area network (LAN). This allows remote computers on the LAN to be provided data from the buffered monitoring computer without the possibility of affecting PCMS computer operations.

The PCMS configuration is designed to make the most of the enhanced protection, control, and monitoring features provided by digital technology. Such features include graphic operator displays, an advanced control algorithm, historical data storage, and secure network communication of data to remote computers.

Console
Figure 7-1 shows the front view of the console. The console is composed of left wing, center, and right wing sections with the wings angled in to aid in reading their displays.

The RSS analog bargraph displays and testing controls are located in the left wing section. The center section contains the DCC-X and DCC-Z LCD monitor / keyboard interfaces and the rod control switches. The right wing section contains the hardwired alarm lights and infrequently used switches.

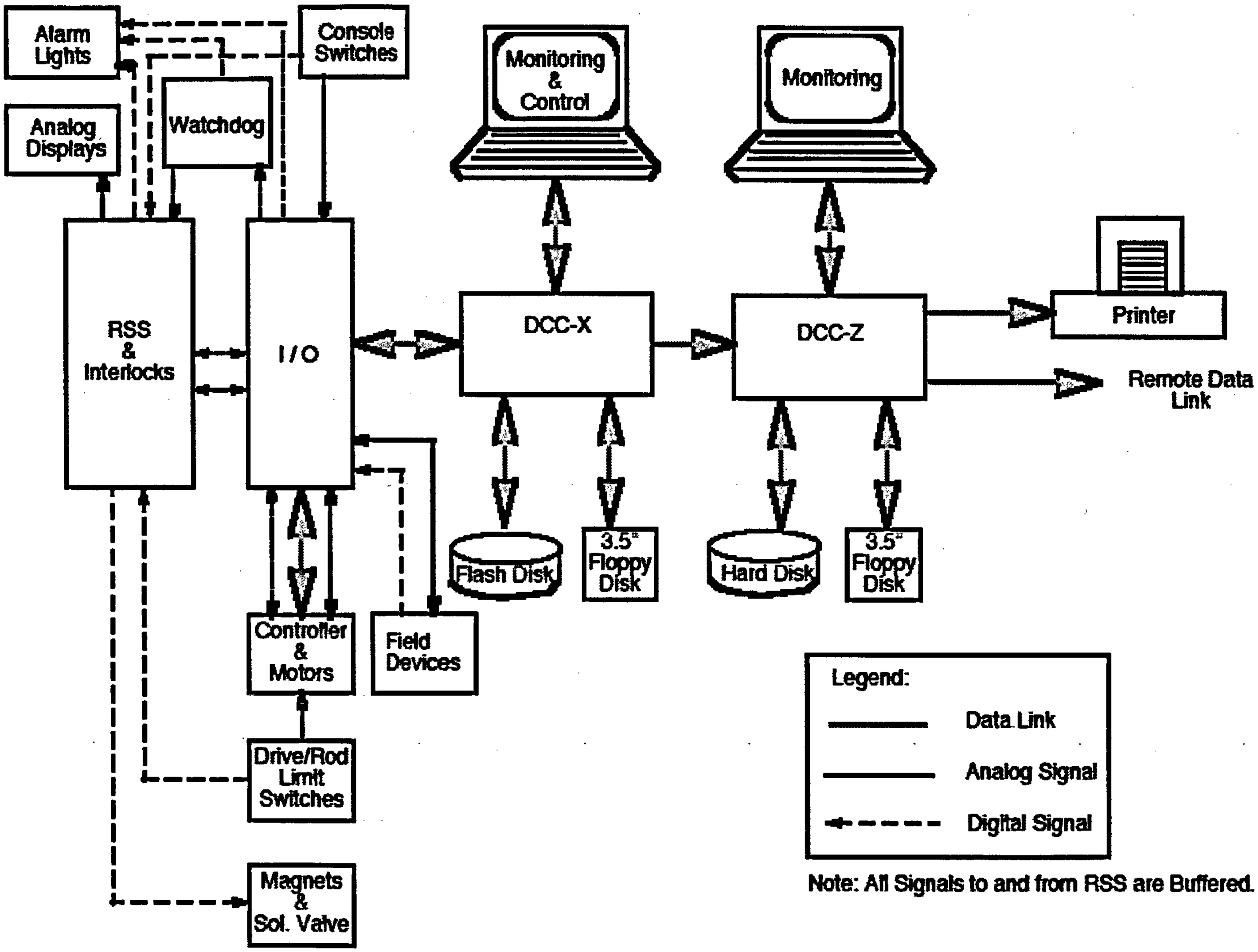


Figure 7-3 PCMS and Interfaces to Other Systems

Figure 7-4 shows the back view of the console and arrangement of the major components. Looking from the back, the three sets of racks contain the control and monitoring system equipment. The right set of racks contains the RSS equipment.

Two matching free standing control room racks contain the radiation monitoring equipment, neutron beam lab CCTV displays, and other equipment.

7.2.3.1 RSS Description

The RSS, which includes the instrumentation, is equipment provided by ***GAMMA-METRICS*** and is designed to provide to the operator:

- a measure of the flux level and rate at the wide range fission detector from source level (shutdown) to 200% of full power reactor operation.
- a measure of the pulse power at the gamma ion chamber from 0 to 2000 MW(th).
- a measure of the linear power at the gamma ion chamber from 0 to 120% of full power.
- a measure of the fuel temperature in degrees Celsius.
- the permission to operate control rods.
- all required safety trips and operational rod interlocks.

The equipment is designed to provide reliable neutron flux measurement from reactor shutdown to reactor full power level (10 decades). It is designed to measure neutron flux with the detector in a high gamma radiation and electrical noise environment, to measure the fuel temperature, and to provide permission and annunciators for rod movement. The RSS is entirely hardwired analog equipment and has no embedded microprocessors.

Figure 7-5 contains a functional block diagram of the RSS. The following paragraphs contain a general description at the block diagram level.

The RSS is divided into four functional subsystems:

- Wide Range Channel, consisting of the Wide Range Detector Assembly, Wide Range Amplifier, and Wide Range Monitor
- Power Range Channel, consisting of a gamma ion chamber, in-core thermocouples, and the Power Range Monitor
- Control and Alarm Subsystem, consisting of the SCRAM and Rod Control Switch Assembly, and SCRAM and Alarm Panel Assembly, and
- Power Distribution System consisting of AC power distribution panels and a DC power supply

<u>Wide Range Channel Description</u>
The wide range signals originate from the fission chamber of the Wide Range Detector Assembly located near the reactor core. This produces a series of pulses representing the power being generated in the reactor. These pulses are applied to the Wide Range Amplifier, located on the reactor bridge, which processes and amplifies the pulses; it then applies them to the Wide Range Monitor, installed in the Console Assembly. The Wide Range Monitor further processes the signal pulses to produce a front panel visual indication of the percent power (on a square root scale),

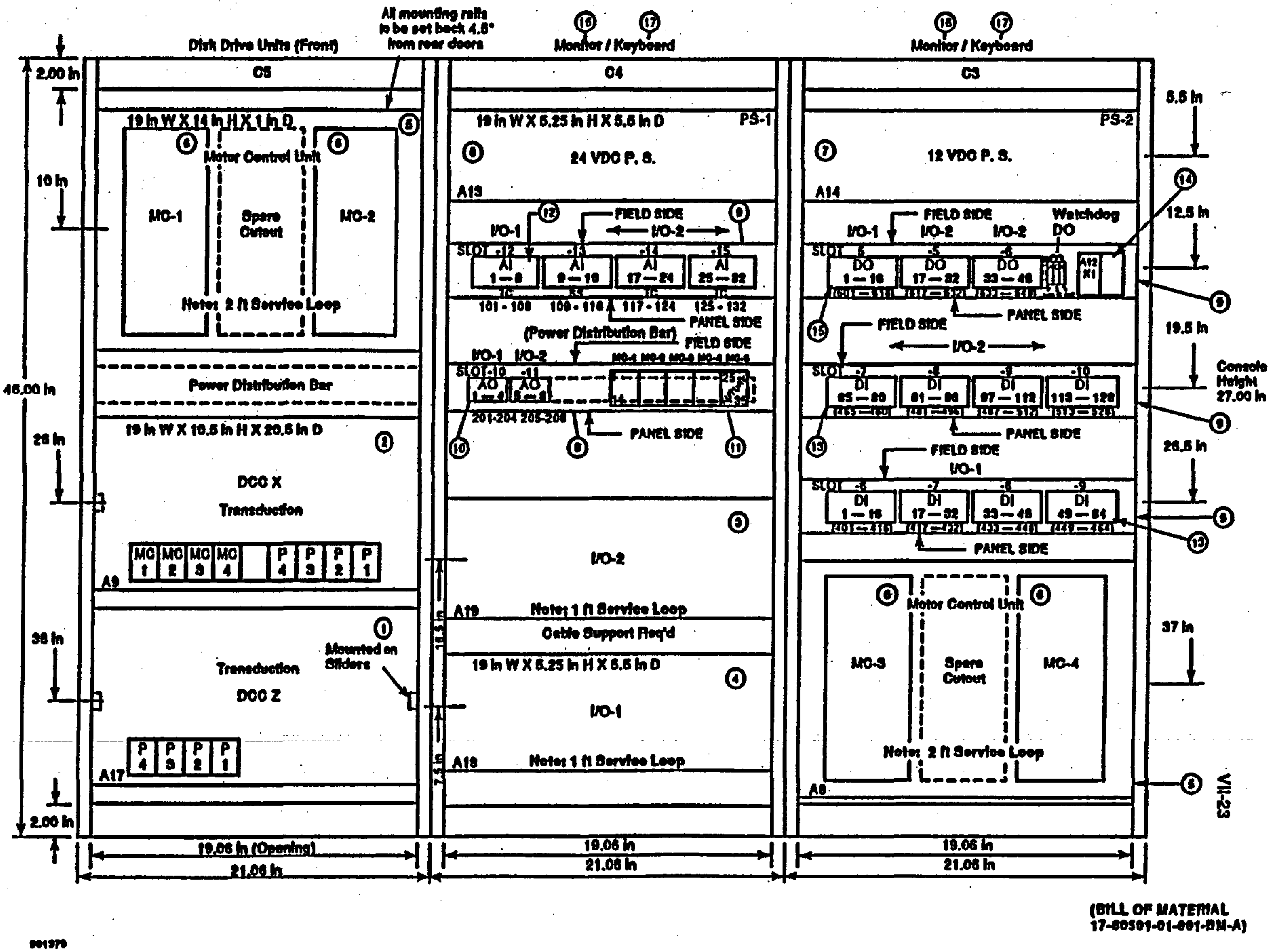
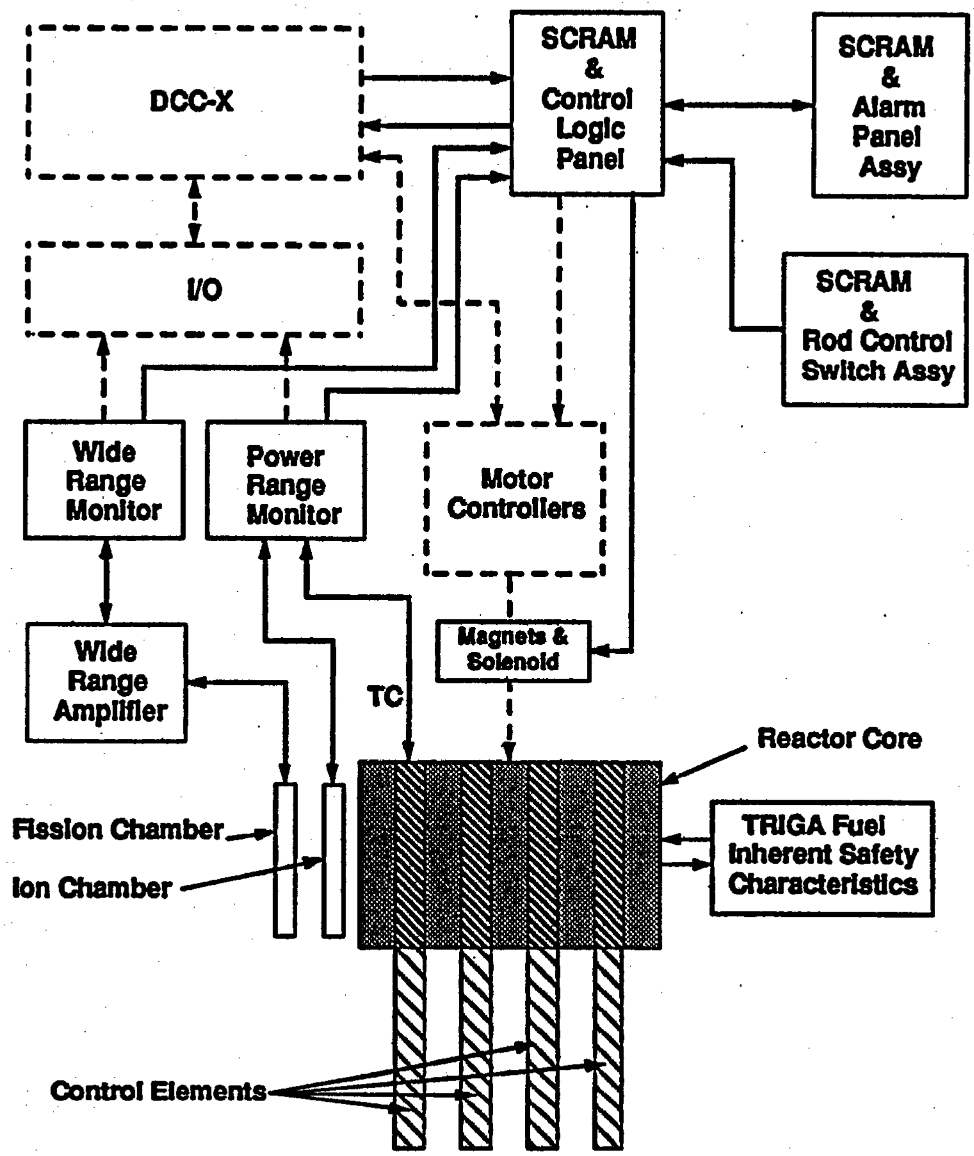


Figure 7-4 CMS Equipment Layout (console rear view)



Note: The dashed components are not part of the RSS

Figure 7-5 Functional Block Diagram of RSS fraction of full power (on a logarithmic scale, 10^0 corresponds to 1 MW(th)), and the rate of change of reactor power. The Wide Range Monitor has built in test and calibrate capability and provides safety trip and operational interlock signals to the control and alarm subsystem when preset trip level set-points are exceeded.

The Wide Range Monitor measures the number of pulses per unit time from the Wide Range Detector Assembly over the range from source level to the level where the error from the countrate loss due to coincident pulses becomes unacceptable. From about two decades below the upper end of the countrate range to full power, the Wide Range Monitor measures the mean square value of the time variant signal from the detector. This mean square value is proportional to the average rate of neutron pulses and is not dependent on the pulses being individually identifiable. It also provides good discrimination against alpha and gamma signals.

The derivative of the logarithm of reactor power provides a measurement that is proportional to the change in reactor power per unit time. The signals are displayed on the Rate bargraph in decades per minute.

The meters on the front panel of the Wide Range Monitor provide a display of the following signals:

Monitor Nomenclature	Display Signal
Power Range	Square Root Percent Power
Wide Range Log	Log Percent Power
Wide Range Rate	Rate of Change, DPM

Three dual LCD bargraphs in the Wide Range Monitor display the measured variables and the trip setpoints. The left bargraph shows the magnitude of the measured variable. The right bargraph shows a trip setpoint. Two trip set points are provided for each measured variable. The first setpoint is normally displayed. The second setpoint can be displayed by pressing the Trip 2 switch.

Output signals from the Wide Range Monitor are input to the digital PCMS, supplied by ***AECL Technologies*** in the Console Assembly. The digital PCMS provides redundant protection trip commands.

<u>Power Range Channel Description</u>
The Power Range Monitor receives a power signal from a gamma ion chamber to provide a linear power output of 0 to 120 percent power and a pulse power output of 0 to 2000 MW(th). It also provides fuel temperature displays in degrees Celsius, the signals come from two out of three available Type K in-core fuel thermocouples. There is a trip select switch on the front panel to select which of the two fuel temperatures will provide the trip signal to the SCRAM and Control Logic Assembly. The Power Range Monitor has built in test and calibrate capability and generates reactor safety trip signals to the control and alarm subsystem when preset trip level setpoints are exceeded.

The meters on the front panel of the Wide Range Monitor provide a display of the following signals:

Monitor Nomenclature	Display Signal
Power Range	Linear Percent Power
Fuel Temperature 1	No. 1, Degrees C
Fuel Temperature 2	No. 2, Degrees C

Three dual bargraphs in the Power Range Monitor display the measured variables and the trip setpoints. The left bargraph shows the magnitude of the measured variable. The right bargraph shows a trip set-point. Two trip set-points are provided for each measured variable. The first set-point is normally displayed. The second set-point can be displayed by pressing the Trip 2 switch.

Output signals from the Power Range Monitor are input to the digital PCMS supplied by ***AECL Technologies*** in the Console Assembly. The digital PCMS provides redundant protection trip commands.

<u>Control and Alarm Subsystem Description</u>
The Control and Alarm Subsystem consists of the SCRAM and Control Logic Assembly, SCRAM and Rod Control Switch Assembly, and SCRAM and Alarm Panel Assembly.

- <u>SCRAM and Control Logic Assembly Description</u>
  The SCRAM and Control Logic Assembly contains all the relays necessary for safety trips from the Wide and Power Range Channels, from manual SCRAMs and from automatic SCRAM functions provided by the PCMS. The SCRAM and Control Logic Assembly also contains relays necessary for reactor rod movement operation, both manually and automatically.

- <u>SCRAM and Rod Control Switch Assembly Description</u>
  The SCRAM and Rod Control Switch Assembly also contains the Manual SCRAM switch which removes power from the magnetic couplers on the Safety, Shim, and Regulating control rods and from the solenoid valve, dumping air from the Transient control rod. The SCRAM and Rod Control Switch Assembly also contains the rod control UP and DOWN manual control switches and the individual SCRAM switches for each control rod. The SCRAM and Rod Control Switch Assembly, contains a keylock switch which provides the functions of OFF, OPERATE, and RESET for reactor operation.

- <u>SCRAM and Alarm Panel Assembly Description</u>
  The SCRAM and Alarm Panel Assembly, contains annunciator lamps for display of the trip status of each control rod, for the DCC-X Watchdog SCRAM and for visual indication of trip status from individual spare inputs. The panel contains indicator lamps for display of EVACUATION INITIATED, HIGH POWER SCRAMs BYPASSED and of POWER ON. The panel also contains pushbutton switches for LAMP TEST, LOW COUNTRATE DEFEAT, and EVACUATION INITIATE.

Power Distribution System Description
The 24 Volt Power Supply Assembly contains the 24 Vdc source which powers all indicator lamps, all relays in the Scram and Control Logic Assembly and provides power to actuate the control rod electromagnets and to operate the transient rod air solenoid. The 24 Volt Power Supply Assembly also contains a relay that provides contact outputs for an external evacuation alarm.

7.2.3.2 PCMS Description

The PCMS consists of the following basic equipment:
- a control computer (DCC-X) system with hardware I/O and LCD monitor / keyboard operator interface.
- a monitoring computer (DCC-Z) system with a LCD monitor / keyboard operator interface, printer, hard disk and serial and LAN communications.
- an associated controller for each of the four control rod drive motors

Figure 7-3 shows the hardware configuration of the PCMS and interfaces to other systems. All signal interfaces to the PCMS are through the hardware I/O of DCC-X. DCC-Z receives all its signal data from DCC-X via a "one way" serial link using a broadcast type protocol. The "one way" serial link means that DCC-X cannot physically receive any communications from DCC-Z, therefore, DCC-X is buffered from the effects of failures in DCC-Z or connected systems. It is intended that reactor operation is permissible with the DCC-X and with or without DCC-Z.

7.2.3.3 Console Description

Basic summaries of the functions for DCC-X and DCC-Z are given in the sections 7.3.1 and 7.3.2.

7.2.4 System Performance Analysis

System design, design criteria, and design bases are discussed in the previous subparts of section 7.2 of this document. The bases for the relevant TS are also a reference. The present instrumentation and control system has been in operation since 1991. The 1991 system upgrade (with further upgrades in 2004) incorporated advancements in electronics and added software functions that meet or exceed the design criteria of the original General Atomics system. While no credit is taken in the TS for the PCMS safety redundancy, it does add an additional layer of protection on top of the RSS.

Limiting safety settings, limiting conditions for operation, surveillance requirements, and action statements concerning the instrumentation and control system are detailed in the TS, Chapter 14. The TS requirements are accomplished through use of various RSEC standard operating procedures and checks and calibration procedures. An administrative procedure identifies which RSEC procedure is in place to meet each of the TS requirements.

7.2.5 Conclusion

Operation during the term of the present license has shown the instrumentation and control system to be capable of performing all intended functions with excellent stability and reliability.

The instrumentation and control system adds an additional layer of protection above the inherent safety of the TRIGA fuel and safeguards the public by meeting the operability and surveillance requirements of the TS, that are in turn enforced by PSBR procedures.

7.3 Reactor Control System

7.3.1 DCC-X Functions

The PCMS is designed so that all TS required functions are performed by DCC-X without reliance on the DCC-Z computer system. In the case of failure of DCC-X, DCC-Z can be used to replace DCC-X and continue operation. The replacement can only be done by physically and electrically changing computers.

7.3.1.1 Modes of Operation

There are four modes of operation:

Manual
Rods are controlled manually with the up and down pushbuttons.

Auto
Various combinations of the rods are controlled automatically with manual control provided for the rods that are not being used for automatic control functions. Three combinations of rods are available for automatic control. The transient rod is never under automatic control.

- 1 Rod Auto -Regulating rod controlled with the shim and the safety "shimmed".
- 2 Rod Auto - Regulating and shim rods banked for automatic control and the safety rod "shimmed".
- 3 Rod Auto - Regulating , shim, and safety rods banked for automatic control and no "shimming" rod.

Square Wave
A square shaped power trend is initiated by firing the transient rod to provide a step increase in power. This is followed by auto control of power at a fixed set-point for a pre-selected time. The power trend is then automatically or manually terminated by a reactor SCRAM.

Pulse
A power pulse is produced by firing the transient rod to give a supercritical step increase of reactivity. All other rods are frozen in position. A pulse profile is developed due to the prompt negative temperature reactivity from the fuel as it heats up. The reactor is automatically scrammed after a preset time to cut off the tail of the pulse.

DCC-X controls the rod drive velocities for all manual and automatic control functions.

All of the operational interlocks and safety trips, that are required by the TS, are performed by the hardwired RSS. The PCMS computer, DCC-X, validates the operation of the RSS by performing the same logic as the RSS. If there is a failure of that validation a DCC-X SCRAM request is issued to the RSS, causing a SCRAM. This is redundancy beyond the original RSS and is not required by the TS. In addition, the PCMS initiates SCRAMs, by the DCC-X SCRAM request, for conditions other than reactor safety, i.e. radiation alarms. In these cases the functions initiated by the PCMS are not appropriately classified as part of the safety system so it is more correct to designate them as protection functions.

7.3.1.2 Reactor Stepback

A reactor stepback function is used to mitigate events that may lead to a RSS Trip or diminished control. The stepback is performed by driving all rods except the transient rod at high speed into the core. The stepback not only prevents operation, but also shuts down the reactor if it is already in operation.

The following conditions initiate a stepback:

- Fuel Temperature High
- Reactor Power High (Fission Chamber)
- Reactor Power High (Gamma Ion Chamber)
- Excessive Power Spread (between the Fission Chamber and the Gamma Ion Chamber)
- Reactor Operation Inhibit

A bypass is provided to allow testing the RSS SCRAMs without causing a stepback. The bypass is allowed only in manual mode and has an automatic time out feature.

The inhibit condition is imposed in various operational situations (see section 7.3.1.5).

7.3.1.3 Reactor SCRAMs

A DCC-X SCRAM request will be given to the RSS in the following conditions:

- Fuel Temperature #1 or #2 High
- Reactor Power High (Fission Chamber)
- Reactor Power High (Gamma Ion Chamber)
- Pulse Timer timed-out
- Radiation High from one of the following:
  - East Bay Monitor
  - West Bay Monitor
  - South Bay Monitor
  - East Air Monitor
  - West Air Monitor
  - Neutron Beam Lab Monitor
  - Co-60 Lab Monitor
- Emergency Evacuation Button
- Remote Scram Pushbutton (1 of 4)
- Reactor Bay Truck Door Open
- Both East and West Facility Exhaust Fans Off
- Interlock Validation Failure

- Rod Velocity Signal Failure
- Rod Motor Overspeed
- Square Wave Termination Request

To allow testing of the RSS logic, a testing bypass is provided for the DCC-X SCRAMs as specified for the stepback function (see 7.3.1.2).

7.3.1.4 Reactor Interlocks

All of the operational interlock logic in the RSS is validated by DCC-X. DCC-X monitors all of the inputs and outputs of the hardware logic and performs the identical logic in software. If a validation failure is detected, a SCRAM is requested.

The DCC-X control logic is designed to avoid signal outputs that are in violation with the RSS interlocks.

7.3.1.5 Facilities Systems Support

The following functions are performed for control and monitoring of various systems in the reactor facility:

<u>Emergency Evacuation</u>
An emergency evacuation is initiated on high radiation or manually from a switch on the console. Following initiation these listed actions occur:

- evacuation horn is energized
- facility exhaust system is secured
- emergency exhaust system is turned on
- alarm light on console is lit
- alarm is sent to university police

<u>Reactor Operation Inhibit</u>
Reactor operation is inhibited by initiating a reactor stepback when an inhibit condition exists. The inhibit conditions cover the following situations:

- keyswitch is off
- a radiation hazard from the neutron beam ports exists
- both east and west bay or air radiation trips are defeated
- pool temperature is high
- reactor bay truck door is open

<u>Manual Controls</u>
Manual control of the following devices is provided:

- east and west facility exhaust fans
- N-16 diffusion pump
- neutron beam lab CCTV camera and monitor
- rabbit system controls

Operating History Records

The following parameters are continually updated:

- integrated power
- total time that the reactor was critical
- total operating time (i.e. time that the keyswitch was in the operate position)

[redacted]

7.3.1.6 Alarms

Two types of alarms are provided, status alarms and state transition alarms. Status alarms are like traditional hardwired window alarms. Summary displays of alarm points are given to indicate the present state of these alarms. State transition alarms are chronological lists of alarm messages that are issued whenever an alarm point changes state.

7.3.1.7 Operator Interface

The operator interface is through the analog instrumentation displays, DCC-X LCD monitor display, the DCC-X keyboard, and hardwired control switches. All operator inputs are performed with the keyboard except for the following:

- manual SCRAM switches
- rod drive up and down switches
- operation keyswitch
- emergency evacuation switch
- low count rate interlock defeat switch (allows rod movement with a subcritical core)

Special operator displays are provided to give reactor status information as required for the various operator tasks.

The generic ***PROTROL*** maintenance display system is also provided with the following features:

- system utilities
- time trends
- bar charts
- message log
- tuning interface

7.3.1.8 Self Testing

The following standard ***PROTROL*** self tests are performed:

- control functions executed twice on startup
- I/O checks
- check watchdog contact open on startup before resetting the watchdog timer
- disk operation error handling
- serial port operation error handling

In addition to the standard checks, continuous analog feedback checks of the velocity signals to the motor controllers are performed.

7.3.2 DCC-Z Functions

DCC-Z provides redundant monitoring capability plus enhanced monitoring features not available on DCC-X. Reactor control functions are not available from DCC-Z.

DCC-Z contains the same operator displays as DCC-X, modified as necessary to exclude the control functions. The operator displays of pulse trends also include the capability to retrieve up to the 10 most recent historical pulse trends (only the most recent pulse is available on DCC-X).

DCC-Z has available the same standard ***PROTROL*** maintenance displays as DCC-X plus the additional historical trends. The message log includes all messages from DCC-X as well as DCC-Z.

DCC-Z transmits selected signals periodically onto the local area network for use by other machines connected to the LAN.

7.4 Reactor Safety System (RSS)

Tables 2a and 2b in the TS section 3.2.4 list the minimum safety circuits and the minimum operational interlocks for the PSBR.

The reactor safety system (RSS) shuts down the reactor (SCRAM) and inhibits movement of the rods when required by operational interlocks. The RSS is a completely hardwired system with no software programmable components or embedded microprocessors. All signal connections between the RSS and the PCMS are buffered through relays (for digital signals) or isolators (for analog signals).

A SCRAM is carried out by decoupling the control rods from their drive mechanisms allowing them to drop under gravity to their position of largest negative reactivity. The Regulating, Shim, and Safety rods are coupled to their drives by electromagnets. The transient rod is coupled to its drive by air pressure applied to its cylinder via a solenoid valve. The SCRAM logic opens relay contacts to de-energize the electromagnets and the solenoid valve.

The control rod operational interlocks are implemented through contacts to the motor controllers and transient rod solenoid valve logic. These signals are controlled through relay logic to prevent conditions that may lead to unintentional operation. The interlock logic is validated by DCC-X which will SCRAM the reactor on detecting a failure.

The RSS equipment occupies the entire left wing of the console (viewed from the front) and uses its own dedicated 120 Vac and 24 Vdc power supplies. Some of the switch inputs are located in other portions of the console for human factors reasons.

7.4.1 RSS Relay Logic Design

There are six separately fused circuits to divide the current load. The SCRAM and operational interlock functions are on separate fuses.

All lights on the alarm panel (right wing of the console) are driven from the RSS logic. A lamp test button tests all alarm lights. Two blocking diodes are incorporated in the lamp test circuit. One is to prevent the logic for individual lamps driving other lamps through the test circuit and the other provides added insurance in preventing lamp test operation from back feeding into other logic.

A number of trip functions in the RSS logic employ latches requiring manual action to reset. Many logic states are also latched in DCC-X software. The "reactor operate keyswitch" resets all latches in the RSS and DCC-X software (for RSS, "trip reset" buttons must be pressed on the Wide Range Monitor and/or Power Range Monitor to reset any latched RSS hardware trips before resetting the keyswitch).

7.4.2 SCRAM Logic

The SCRAM logic fails safe on loss of power or open circuit. The voting logic for redundant parameters (including the DCC-X SCRAM) is separated into two circuits so that any single short between two points in the system can affect only one of the redundant parameters. The arrangement of parameters into the two circuits is as follows:

<u>SCRAM Circuit #1</u>	<u>SCRAM Circuit #2</u>
DCC-X Watchdog Trip	DCC-X SCRAM
FC Power High	GIC Power High
FC Bias Voltage Low	GIC Bias Voltage Low
Manual SCRAM	Keyswitch Off
Fuel Temperature High	2 Spare SCRAMs
Pulse Timer SCRAM	
2 Spare SCRAMs	

The DCC-X watchdog trip is a combination of a digital output and two watchdog relay contacts, all wired in series ("OR" configuration for trip) for enhanced reliability (see **Figure 7-6**). The digital output allows the software to provide a quick trip signal independent of the watchdog relay. The watchdog relay covers software failures as well as hardware failures, but may take ≈ 2 seconds to issue the trip signal after a fault occurs.

The pulse timer SCRAM and watchdog trip employ latching logic so that the initiating condition is not needed to maintain the tripped state. The hardware latch for the watchdog eliminates dependence on the software for the latching function. DCC-X software must be re-started following a watchdog trip before it will resume updating the watchdog. The pulse timer latch ensures that once started, the timer will time out independent of the initiating signal from DCC-X or the keyswitch reset.

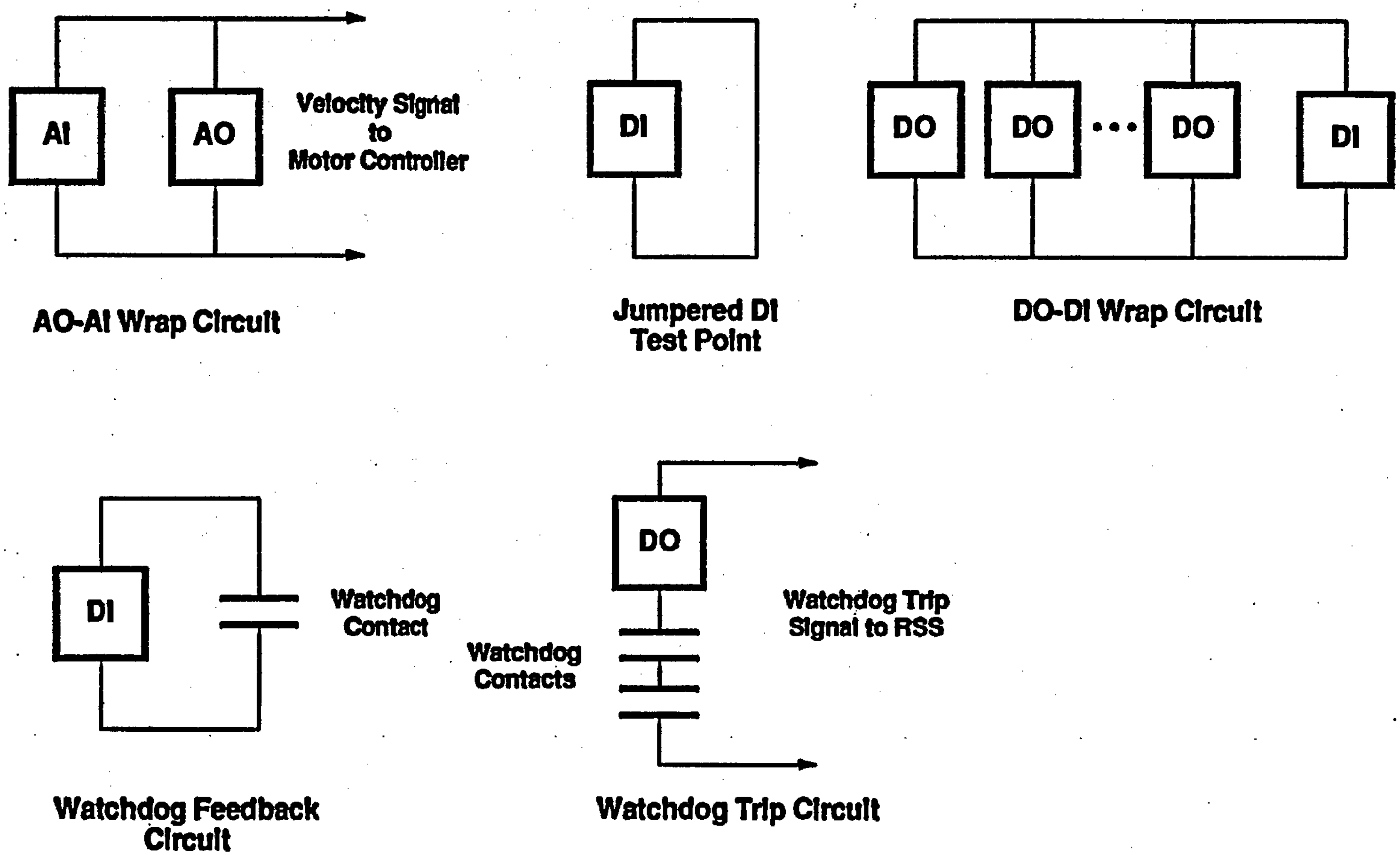


Figure 7-6 Watchdog and I/O Self Test Circuits

There is logic to bypass both high power SCRAMs in pulse mode. The bypass is performed by DCC-X which drives separate digital outputs to bypass each high power trip. A hardwired alarm light provides continuous indication of the bypass. DCC-X validates the bypass logic of both trips by reading back the bypass state through digital inputs. A failure generates a status alarm on the LCD monitor.

There are two fuel temperature measurement channels provided for the RSS logic; however only one is used for the RSS SCRAM. A latching switch on the front panel of the console (power range chassis) selects 1 of 2 temperature measurement channels.

The SCRAM final voting logic is similar for each rod. The voting is an OR gate of the circuit #1 and #2 SCRAMs, individual rod SCRAM button and the latching logic. The individual manual SCRAM button is wired into the final voting directly from the pushbutton contacts to maximize reliability. The rod SCRAM latch cannot be reset unless the initiating conditions have cleared.

The final SCRAM voting logic deenergizes the electromagnets for the Reg, Shim and Safety rods directly. For the transient rod, the final SCRAM voting provides a signal to the air permissive logic. A contact from the air permissive relay is wired in the solenoid valve circuit to execute the transient rod SCRAM.

The state of all SCRAM parameters and final voted logic is fed back to DCC-X via digital inputs for annunciation purposes. The hardwired alarm windows indicate the final SCRAMed state of the individual rods.

7.4.3 Transient Rod Air Interlock Logic

The transient rod air interlock logic drives the air permissive relay. The transient rod SCRAM overrides all other permissives by unconditionally de-energizing the permissive relay. The logic is designed to fail safe on loss of power. Contacts must close to allow application of air to the transient rod.

The permissive conditions for applying air are as follows:

- in pulse mode AND initial power low
- transient rod drive is at its bottom end of travel
- in square wave mode
- air has previously been applied (permissive latch)

The permissive latch is required to allow continued application of air following a pulse, and during manual driving of the transient rod. In both cases, the permissive conditions would otherwise be lost and the transient rod would drop. If air is initially applied in accordance with the permissives, continued application of air is allowed regardless of the state of the permissives.

DCC-X drives the contact requesting air to the transient rod. The above air permissive logic is also duplicated in software for the air request signal. In addition, the hardware logic is validated through a separate task in DCC-X that will SCRAM the reactor on validation failure.

7.4.4 Rod Drive Interlocks

The logic for the Regulating, Shim, and Safety rods is similar. The logic for the Transient rod differs slightly from the others.

The rod drive interlock logic is not totally designed to fail safe on loss of power since power must be applied to the motor controller digital inputs to perform the inhibit function. The logic is thus a combination of negative (open contact is true) and positive (closed contact is true) logic. The lower reliability of this arrangement is compensated by the interlock validation in DCC-X and by the use of redundant software interlocks for the demand velocity signal. The reactor is SCRAMed on interlock validation failure. The software interlocks in DCC-X make use of separate end-of-travel (EOT) switch contacts that are configured to fail safe on loss of power.

The rod drive pushbuttons provide independent contacts for the RSS interlock functions and manual drive via DCC-X software. Normally closed contacts are used for the interlock functions so that the interlocks fail safe. Normally open contacts are used for the inputs to DCC-X so that power supply failure will not give a spurious drive request.

For the Regulating, Shim, and Safety rods, up and down drive are inhibited under the following conditions:

Up Drive Interlocks:

- "up" EOT is reached (drive limit switch)
- watchdog trip
- more than one "up" pushbutton is pressed
- low count rate (and not defeated by pushbutton)
- short reactor period
- in pulse mode

Down Drive Interlocks:

- down EOT is reached (drive limit switch)
- in pulse mode

If more than one "up" pushbutton (third "up interlock" above) is pressed, the logic blocks manual up drive of all rods regardless of whether or not the rods are being used for automatic control. Rods being driven under automatic control are not affected.

Up and down drive inhibits for the transient rod are the same as above except there is no interlock on pulse mode, and the transient rod is not driven under automatic control.

Rod drive interlock validation is performed for the up drive interlocks of each rod. The pulse mode signal is also validated separately. A failure of any of these validations will result in a DCC-X SCRAM request.

7.5 Engineered Safety Features (ESFs) Actuation System

ESF actuation systems sense the need for and initiate the operation of ESF systems (1) to prevent or mitigate the consequences of damage to fission product barriers such as fuel, cladding, or fueled experiments caused by overpower or loss-of-cooling events, or (2) to gain control of any radioactive material released by accidents. The PSBR reactor does not have systems as described in item (1) because of the inherent safety of the TRIGA fuel. The PSBR does have a facility exhaust system (FES) and emergency exhaust system (EES) that would fall under category (2). The objective of these system's operations is to mitigate the consequences of the release of airborne radioactive materials.

Whenever the reactor is not secured, at least one of the two facility exhaust fans is in operation. Upon initiation of any alarms from the radiation monitors listed in section 7.7, or by pushing the evacuation initiate button on the reactor console, the PCMS secures the FES and actuates the EES. The FES and the EES systems are described in Chapter 6.2.1, Confinement. The effectiveness of the EES is described in Chapter 13.1.1, Maximum Hypothetical Accident. Sections 3.5, 4.5, and 5.5 of the TS describe the operational, surveillance, and design requirements, for the FES and the EES.

7.6 Control Console and Display Instruments

The console assembly is located in the control room. All of the equipment described below is located in the console assembly with the exception of the FC and GIC Detector Assemblies, the in core fuel temperature TCs, the Wide Range Amplifier and the field sensors. A window is provided between the control room and the reactor bay such that an operator seated at the console can observe personnel movement in the reactor bay. A CCTV system is also provided in the control room so that the operator can observe personnel movement in the neutron beam laboratory.

Internal communication systems and a commercial telephone are available to the reactor operator in the control room. The internal communication systems allow:

- two way conversation with anyone using the pneumatic transfer system sending station
- two way conversation between offices and offices within the building
- the use of a page system that has speakers in all parts of the building

7.6.1 PCMS Hardware Description

Figure 7-3 shows a block diagram of the PCMS hardware and interfaces to the other systems. The PCMS is composed of two computer subsystems called DCC-X and DCC-Z and the motor controllers for the four rod drives.

DCC-X is the monitoring and control computer that handles all control logic and the input/output (I/O) interface signals to the field, rod drive motor controllers, and RSS. The DCC-X computer system provides all the functions necessary for safe control and monitoring operation of the reactor.

DCC-Z is the monitoring computer receiving its signal information from DCC-X via a one-way serial data link. DCC-Z broadcasts signal information onto a local area network for use by other computer systems. DCC-Z buffers the other computer systems from DCC-X ensuring that failures in DCC-Z or the other systems cannot affect reactor operation.

7.6.1.1 Computers

The central element in each DCC is a Neatek model 406 AMD 586 SBC allowing use of the many special purpose plug-in cards available for this type of computer. The computer has a modular design and a passive back plane for easy "board swap" or module replacement repair. All cards including the processor are plugged into the passive ***AT*** bus back plane.

Each computer has a 1.44 MB 3.5" floppy disk drive and a 250 MB Zip Drive™ (for off-line use only). DCC-X has a 16 MB IDE flash disk and DCC-Z has a 20 GB Maxtor IDE hard disk. The large capacity disk in DCC-Z allows for storage of a significant amount of historical data at fast update rates.

Each computer system is equipped with an ***AT*** 101 enhanced style keyboard. The keyboard has a software programmable key repeat rate and delay. The repeat rate is slowed down by ***PROTROL*** software as a precaution to prevent inadvertent repeated operator inputs from the keyboard (e.g. book lying on key).

The following plug-in circuit boards are supplied with each computer (note: common board types between the two computers are the same manufacturer and model):

DCC-X:

- CPU board
- disk controller board
- display generator board
- serial port board
- I/O bus converter board

DCC-Z:

- CPU board
- disk controller board
- display generator board
- serial port board
- Local Area Network (LAN) board
- multi-function (serial/parallel port) board

The CPU board is running an AMD 586 processor at a clock frequency of 133 MHz. The board is supplied with 4 MBytes of RAM. This board is also equipped with an 80487 math coprocessor. The CPU board has six logic circuit layers implemented with CMOS circuitry and advanced low power SCHOTTKY gate arrays for low power consumption.

The disk controller board controls the hard disk, the Zip Drive™, and the floppy disk drive.

The display generator board drives the LCD monitor with a pixel resolution of 640 x 480 in 256 colors (similar to super VGA display generators). The advantage of the display generator board is that it has its own on board processor and firmware to generate and position the display images using high level software commands. This reduces the loading of the computer CPU which would be necessarily greater with VGA displays. It also has its own memory to store many different displays for instant retrieval. Each computer is supplied with a compatible LCD monitor.

The serial port board provides the multiple serial connections required and unloads the main CPU from the overhead of serial communications. The board is connected to a board mounted on the back of the DCC which contains 8 serial ports.

The I/O bus converter provides the interface to I/O equipment for DCC-X.

The multi-function board is used in DCC-Z for the parallel port connection to the printer (see reference [9]).

7.6.1.2 Input/Output Hardware

<u>Chassis Arrangement and Watchdog/Test Cards</u>
DCC-X is equipped with 2 chassis containing the following types of I/O:

- analog inputs (8 differential inputs per card)
- analog outputs (5 differential outputs per card)
- digital inputs (16 contact sense inputs per card)
- digital outputs (16 contact outputs per card)
- watchdog contact outputs

The I/O points provide the computer signal interfaces to the analog hardware equipment. Serial data links are also used for interfaces to digital equipment (e.g. motor controllers and DCC-Z). Usage of the I/O points can be summarized as follows:

- field interface signals to components throughout the reactor facility
- interface signals to the RSS, motor controllers and internal controls of the console
- digital and analog circuits output to input feedback (wrap around) self test of the I/O hardware (see figure 7-6)

The watchdog provides the function of testing both hardware and software integrity. The watchdog is composed of a relay and timer circuit that must be reset at a high enough frequency (period of 1.7 s) to maintain the relay energized. The software is scheduled to generate these pulses through the I/O hardware cyclically each time after passing all of the computer self-tests every 1.2 s. If a self-test fails or the software "freezes" for any reason, the watchdog will time out and its contacts will open.

Analog Signal I/O Cards
The analog inputs and outputs are all voltage signals using a 10 V range where possible to minimize the effect of signal noise pick-up. The following paragraphs describe the types of cards used to process the analog I/O signals.

Two types of analog input signal conditioning (gate) cards are used: transformer coupled and solid-state, each providing 8 differential voltage inputs. The card selection is based on meeting appropriate filtering, sampling rate, and surge withstand capability for the input signals.

The solid state gate card is unfiltered at its input and is used only for the gamma ion chamber pulse range signal. The pulse range signal is sampled at approximately 3000 samples per second during a pulse which is well within the card and ADC capability. This card can withstand a maximum of +/- 11 Vdc differential plus +/- 11 Vdc common mode on the input signals that should be adequate considering that the pulse range signal is only routed within the console.

The transformer coupled cards are used for all inputs other than the pulse range gamma ion chamber signal. All inputs have a double-pole filter (-6 db @ 14 Hz). The maximum sampling rate used for the transformer coupled signals is about 10 Hz. This card can withstand a maximum of +/- 25 Vdc differential plus +/- 400 Vdc common mode on the input signals.

The analog output cards are all the same type each having 4 differential bipolar voltage outputs with a fixed range of +/-10.24 V. Each output channel has a 12 bit digital to analog converter (DAC) providing maximum output current of +/- 5 mA (short circuit protected). The output signal can change at a maximum slew rate of 1 V/microsecond and has a setting time of 50 microseconds.

Digital Signal I/O Cards
The digital input card used provides for 16 optically isolated inputs. The card is jumper configured to sense the state of contacts wired across the input terminals with a closed contact being read as "true" in software. An external 24 Vdc power supply is used for contact sensing, resulting in a current of 4.4 mA through the contact when closed. A current level below 0.95 mA will read as an open contact. The 24 Vdc power is supplied to terminals on each I/O chassis which then supplies the digital input cards via the back plane. A maximum of 38 V is tolerable on the input terminals. The input signals are filtered with a 5 msec time constant.

The digital output card provides for 16 relay outputs. The relay contacts are mercury wetted having maximum ratings of 2 A, 200 V and 50 W. The contacts close within 2 msec. An external power supply of 12 Vdc is required for the relay coils. This is supplied through the front edge connectors of the cards.

Watchdog and I/O Self Test Circuits
PROTROL caters to a number of generic self-tests of the analog and digital I/O. These tests are defined in section 7.6.2.4. This section describes the hardware circuits required for the self-tests and the watchdog trip and feedback circuits. Typical circuits are shown in **figure 7-6.**

The watchdog trip circuit is wired to the RSS. This circuit is composed of a digital output and two contacts from the watchdog relay wired in series (contacts open to trip). The use of two watchdog contacts provides increased reliability. When the DCC-X software detects a failure requiring a watchdog trip, the watchdog is no longer updated and the digital output is opened. The digital output is opened immediately. The watchdog contacts will open after the timer delay. "Killing" the watchdog means that the digital output opens immediately and the watchdog timer is no longer reset. "Kicking" the watchdog means that the timer is reset. If the watchdog is not " kicked" before the watchdog timer "drops out" the watchdog contacts open causing a trip.

The analog wrap around circuit is used to test the analog velocity signal to the motor controllers; one wrap around circuit for each motor (see **figure 7-6**). When DCC-X starts up the wrap around circuits are tested by the CHK task before allowing DCC-X to go into controlling mode. Thereafter, the RRS task continually performs the velocity signal validation.

The first point of each digital input card is jumpered closed. This point is continually monitored by the CHK task to ensure that the point always reads in the closed state. Test failure results in a watchdog trip.

For the DO-DI wrap around test, the last point (contact) of each digital output card is wired in parallel to a digital input point (see **figure 7-6**). The CHK task continually toggles the digital outputs and checks for the correct response at the digital input. Test failure results in a watchdog trip.

The watchdog is continually tested for the proper state by the CHK task through the feedback circuit. Test failure results in a watchdog trip. If the test fails during start up, DCC-X will not be allowed to go into controlling mode.

7.6.1.3 Motors and Associated Controllers

The motors supplied for the rod drives are NSK *Megatorque* Motors, model AS0408 (flange mount), with mating EE style controller (interface option 05). The same motor/controller type is supplied for all drives, however the controller is configured differently for the transient rod. The motor has high torque and very smooth operation at low speeds.

The NSK *Megatorque* motors are servo controlled with selectable closed loop control modes of velocity, position, or torque (not used in this application). Position and velocity feedback are provided by a resolver on the motor shaft. The motor controller (or driver) supplies the required current to the 3 phases of the motor, performs the closed loop servo control and motion interlock functions, and handles the serial communications and I/O signal interfaces. All serial communications are with DCC-X over separate dedicated RS232C serial data links for each motor controller. Each motor's I/O interface handles an analog input signal for demanded velocity (from DCC-X), an analog output signal of measured velocity (to DCC-X), and various digital inputs for interlocking motor movement. The motor controller is very versatile, having the capability to easily switch between the various control modes "on the fly", provides the ability to input a velocity demand with an analog signal, and provides an analog output to monitor speed.

The NSK motors were picked for the following reasons:

- smooth operation at low speed
- limited maximum speed (1.5 or 4.5 rps)
- rugged design with long life expectancy
- versatile controller and signal interface
- reasonable comparative cost

The main negative feature of the NSK controller is that its digital inputs for interlocking motion require a closed contact to perform the interlocks. Redundant software interlocks on the demand signal have been provided to compensate so that loss of power will not violate an interlock. Software EOT interlocks are based on both fail-safe configured limit switches and the resolver position indication.

<u>Motor Control</u>
The motor controllers make use of the following signals for control of rod drive position and velocity (see reference [10] for a description of their functions):

Digital Contact Inputs:

- counter clockwise limit switch (CCLS)
- clockwise limit switch (CLS)
- home limit switch (HLS)
- home low speed limit switch (HLLS)
- emergency stop switch (EMST)
- machine ready (MRDY)
- servo on (SVON)

Analog Voltage Signals:

- velocity demand (input)
- measured velocity (output)

Serial Data Link:

- Controller to DCC-X:
    - motor position (resolver counts)
    - state of controller digital inputs: MRDY, SVON, CCLS
    - controller status:
        - drive ready (DRDY)
        - alarm 01 (AL01)
        - alarm 02 (AL02)
        - at home (HOME)

- DCC-X to Controller:
    - requests for the above data
    - commands to switch to velocity mode, position mode, or give a homing request

Before the motor provides torque, the MRDY and SVON inputs must both be true (closed contacts). These two inputs are wired directly to DCC-X digital outputs. Both DOs are set true if DCC-X is healthy, its watchdog is energized, and motor trouble is not detected. Otherwise, motor torque will be lost and rods connected to the rack and pinion type drives will likely fall. When torque is not applied the motors can be easily turned by hand, facilitating setup during commissioning.

Motor trouble is detected if any of the following conditions becomes true:

- a drive alarm occurs (AL01 or AL02)
- drive ready (DRDY) is false
- an error in serial link communications between the motor and DCC-X is detected

The interlocks listed below will provide torque to stop the motor, overriding all other controls (provided MRDY and SVON are true): NOTE: In all cases the input contact must remain closed to maintain the interlock.

- EMST - stops the motor while the input contact is closed. This input is wired to the terminal blocks for use during commissioning.
- CCLS - stops the motor from moving in the counter clockwise direction while the contact is closed. This input is controlled by RSS for the down drive disable function.
- CLS - stops the motor from moving in the clockwise direction while the contact is closed. This input is controlled by RSS for the up drive disable function.

Under normal circumstances, the motors perform servo control of either velocity or position. Velocity mode is used for all operations to position the rod. The velocity control set-point is provided by the analog voltage signal to the controller from DCC-X. Position mode is used only when the rods are stopped and when a homing operation is requested. This approach was used because it was found that some motors would drift very slowly when the rods are stopped in velocity mode. When holding the rods in position mode, the set-point is always the position value at the time of switching into position mode.

A homing operation is performed to move the drive down to a preset position relative to the home limit switch. This position is intended to be just above the bottom drive end of travel limit switch, yet close enough for the magnets to pick up the rod. For the transient rod, the home position is low enough so that the connected rod would have negligible reactivity insertion.

There are two reasons to home a rod:

- following a SCRAM to bring the drive into a position to allow reconnection of the rod when the SCRAM condition is reset. All rods home automatically following a SCRAM except for the transient rod.
- reset the resolver turns counter if power is lost to the controller. The resolver position measurement is only absolute over one turn.

To home a drive, the rod must first be SCRAMed and the drive must be above the HLS and HLLS switch positions. After the home request from DCC-X, the drive will move down at homing speed. When the HLLS contact closes, the drive will slow down to a creeping speed (both speeds are software tunable in the controller). Once the HLS contact opens the drive will continue moving at slow speed for a fixed homing offset (software tunable in the controller). When home is reached, the position measurement is set to zero counts in the controller and the HOME serial output flag goes true until the motor is moved from the home position. The home position value in DCC-X software can be set to any required value through tuning.

A homing request will be rejected under any of the following conditions:
- the rod is not SCRAMed first
- the CCLS input is true
- motor trouble is detected
- the drive is already homing

A homing operation will be aborted under any of the following conditions:
- the CCLS input goes true while homing
- the motor does not move
- motor trouble is detected

An analog voltage output signal from the controller provides a measurement of the motor velocity for monitoring purposes. This signal is used by DCC-X to SCRAM the reactor if the speed in the upwards direction is too high.

7.6.1.4 Power Supplies

All equipment in the console is powered from single phase 120 Vac. The AC power for the control and monitoring system equipment is distributed by two power bars, each containing 8 surge protected outlets.

7.6.1.5 I/O Assignment

The philosophy used for assignment of cards to chassis locations and signals to I/O points is to minimize the consequences of single card or chassis failures where possible. First consideration must be given to the signal requirements and the self check points. Wherever there is flexibility in choice of point assignment, signals that have some level of redundancy are chosen to be on separate cards/chassis if possible.

The cards are organized as shown in drawing -05-009-DD-A of reference [15]. The AO, DO, and DI cards are split between the two chassis, half in each. Chassis I/O-1 contains a bipolar ADC card and I/O-2 contains a unipolar ADC card that determines the splitting of AI cards between the chassis.

The following convention was adopted for the assignment of I/O points:
- The watchdog feedback DI is in I/O-1 providing maximal separation from the watchdog test card in I/O-2.
- The first point on each DI card is jumpered for self-tests.
- The last point on each DO card is wired into the DO-DI wrap circuit for self-testing (see figure 7-6).
- The DI point for the DO-DI wrap is on the last DI card of I/O-2. Being the farthest point in the daisy chain has a marginal reliability benefit.

- The AOs for rod velocity are in a separate chassis from the AIs used for AO-AI wrap testing (see figure 7-6). The AIs from the motor controllers for the overspeed trip are also on a separate chassis from the rod velocity AOs.
- The redundant signals for the DCC-X SCRAM function are on separate AI cards.
- The DIs associated with controlling and monitoring functions of the rods are grouped together on a separate card for each rod. Two rod DI cards are in one chassis and two are in the other chassis.
- The DIs used to monitor the status of redundant SCRAM parameters are in separate chassis.
- The DOs used for the following functions are in separate chassis:
  - bypass the redundant high power SCRAMs
  - control the two lines to police services
  - control the two Rabbits
  - control the two exhaust fans
- The DOs for control of each rod drive are grouped on separate cards where possible (there are 3 DO cards yet 4 motors).

7.6.2 *PROTROL* Generic Software Description

The ***PROTROL*** operating system, with the exception of a very small segment, is written entirely in Pascal. The control application programs are designed entirely in a very high-level control block language. These blocks are also written in Pascal, so there is a high degree of uniformity across the system.

Very intensive and extensive testing of the ***PROTROL*** operating system and service utilities (bar charts, trends, tuning) as well as the control language blocks has been performed to ensure the reliability and robustness of the system.

7.6.2.1 Control Language

A functional description of the block language is given in reference [11].
The major features of the control language used in ***PROTROL*** are:

- each control function or loop (e.g. reactor regulation or protection system trips) forms a separate unit within the computer system, in a structure analogous to using separate instrument racks for installing separate analog control systems. This provides "separation and independence" of different functions within the software. An added advantage is that testing of later changes or additions to the software is reduced in scope and very straight forward.
- there are provisions for separate control loops to send signals to one another, internally within the computer, without using physical I/O.
- control design is done in a "block" language, in which blocks correspond to conventional analog devices, or more advanced operations derived from AECL's experience in digital control.
- the "blocks" may input and output two kinds of control signals (with corresponding I/O for each) within the control programs: analog and boolean. The former corresponds to currents or voltages (e.g. 0..10V), and the latter to relay states (i.e. on/off or true/false).
- the signals are identified by self-documenting 12-character names or "tags".

- the set of "blocks" that can be used is very extensive: it includes all commonly used analog devices (e.g. PID, rate-limiter, etc.), plus a number of blocks with no simple analog world counterpart (e.g. SPREAD, SELECT, LEVEL-COMP, etc.).
- new or custom blocks can be added to the language for special applications, such as the MOTOR block created to provide the serial link interface to the NSK motor controllers.
- identical data structures are used in the control (DCC-X) and monitoring (DCC-Z) computers. This ensures that software signals have a common tag throughout the control and monitoring portions of the system.

7.6.2.2 The Operating System

The ***PROTROL*** operating system is a real-time, multi-tasking operating system.

The term "multi-tasking" means that a number of different functions can be performed, apparently (to the human) simultaneously. The kinds of functions performed include self-checks (diagnostics), control functions (e.g. reactor regulation) message display, tuning of set-points or gains, etc., and display on the LCD monitor the values of signals in bar chart or trend format. Each such function is performed by a separate "task", and the operating system kernel is responsible for scheduling each task (i.e. making it run at intervals) to meet system requirements.

The term "real time" means that the system runs identified critical tasks on time. Here "on time" means that, whether interrupt driven or periodically scheduled, the function is executed adequately fast to control the process to meet dynamic performance specifications (e.g. power overshoot). In practice, this means that all tasks are assigned a priority and the lower priority tasks are suspended when a high priority task "needs" to run.

For maximum software simplicity and predictability of response to all operating conditions, the ***PROTROL*** system is timer driven (sometimes this is described as "polling"). This means that hardware interrupts in the system, from the keyboard, disc drive, etc., are lower priority than the tasks performing self-diagnosis and control. Moreover, no process driven interrupts are implemented. This design philosophy ensures maximal determinism in system response (i.e. predictability of loading and response time), and therefore, maximal system robustness.

The task scheduling function (called the dispatcher) is run off the real time clock interrupt. The interrupt interval is programmed at system boot time to approximately 27 ms, so this is the system's fundamental period. All tasks run at intervals that are multiples of the fundamental period.

The multi-tasking operating system will support many individual and separate tasks, each operating at its own period and relative priority. Each task is self-contained, having its own database, block parameters, and access to the I/O.

The operating system schedules the execution of a number of tasks, representing the self-test (CHK), and TASK-1, . . . , TASK-n, which are the individual generic and application tasks. The data structures of each task are private to the task.

Interfaces to a task are via blocks in the control block language library:
- I/O blocks (AI, AO, DI, DO) to physical I/O hardware
- pseudo I/O blocks (SI, SO, BI, BO) to a shared data region called the ILDB

Each task does its own process I/O. In addition, there are a number of signals exchanged between tasks using the pseudo I/O.

7.6.2.3 Generic Tasks Running in the *PROTROL* System

As described in the previous section, the heart of the *PROTROL* system is the multi-tasking operating system. The operating system schedules a number of periodic tasks in accordance with their defined intervals and priority.

At the bottom of all *PROTROL* LCD monitor maintenance displays is a list in dark grey, of all the tasks in the system. As they are dispatched, the tasks labels are displayed in white (or underlined in white on some screens). By observing the continuous flashing of these indicators, the operator has an immediate view into the operating system's activity.

The generic tasks that are dispatched by the operating system are described below. The tasks exist on both DCC-X and DCC-Z unless noted (by "X" or "Z" beside task label).

LOAD	Calculates the percentage of time the computer is busy executing either generic or application software.
CHK	This task performs the self-diagnostic tests and periodically "kicks" the watchdog to keep it from signaling a SCRAM if all critical self-tests pass.
BCST X	Broadcasts signals to DCC-Z via the "one way" serial link.
HDS Z	Scans the tasks' data bases and stores selected signal values to disk at the one of 5 frequencies.
KBD	This task processes the keystrokes from the operator keyboard in accordance with the displayed menu in maintenance displays. It passes the keystrokes on to the appropriate task for processing in operator displays.
DSK	Handles all access to hard and floppy disk drives.
BAR	Barchart Display. This task displays and updates barcharts of up to 8 variables in a format like horizontal edgemeters. The sets of variables and update period are tunable on-line.
TRND	Trend Diagram. This task plots an image of up to 4 variables on a real-time high resolution display in a format similar to a (very fast moving) strip chart recorder. The sets of variables and update period are tunable on-line.
HTND Z	Retrieves the historical data stored by the HDS task.
HCPY	Responds to a hardcopy request to save a screen image and sends it to the printer (DCC-Z) or flash drive (DCC-X).
KLOK	Updates the clock display in the upper right corner of all displays.
MSG	The message task prints messages to a screen from a queue. The latest message is always displayed at the bottom of the screen. A maintenance display menu allows browsing through the last 250 messages.

STAT	This task periodically updates the maintenance display status of the operating system and tasks.
BRCV Z	Receives all signal data broadcast through the "one way" serial link from DCC-X to DCC-Z. It puts the data into the appropriate task record for use in DCC-Z.
BLN Z	Broadcasts signal data to the Local Area Network.
IDLE	This is a dummy task that is run when the CPU is ready, but no other task is required to run.

7.6.2.4 System Self-Checks and Defenses

The principles employed in the design of the system are:
- keep it simple and robust
- try to keep the DCC running for non-fatal conditions
- fail the DCC for fatal conditions (e.g. drop the watchdog)

The types of failures which could challenge the system are:
- failure of critical field sensors
- loss of power
- failure of communications with the process I/O
- failure of the arithmetic processing
- program corruption faults

<u>Defenses Against Loss of Field Sensor</u>
This type of failure is external to the DCC system, and the level of redundancy in the sensors and the actions to be taken are application specific, so defenses here are the responsibility of the control programs. The operating system does, however, provide assistance through the "quality" flag returned by each analog input block -- this flag warns the control program that the input is out of range.

<u>Defenses Against Loss of Power</u>
Loss of AC power will immediately drop the DCC's watchdog.

Partial loss of power (e.g. to the DI power supply or to an I/O chassis) is detected by the CHK program (which checks one point on each DI card, and "kills" the watchdog if a failure is detected) or through the "quality" flag returned to the control program by the AI block. I/O chassis failures (e.g. +5 V) that affect the operation of the test card, cause an immediate dropout of the watchdog.

<u>Defenses Against I/O Failure</u>
Several layers of defense are provided here, most of which are performed by the CHK program, and some of which are the responsibility of the control program(s).

The first level of defense is the periodic check that data sent to the I/O chassis is correctly received, and can be returned intact. This is done by writing data to, then reading the registers on the test card in each I/O chassis. Any detected failure is first re-tried once after a short delay; a second immediate failure results in dropping the watchdog (if it has not already dropped out since it had not been correctly "kicked").

The next layer of defense is wrap around tests of representative I/O points (see **figure 7-6**). One representative point from each DO card is wired in parallel with one DI point. These DOs are then driven in turn to test that they are accessible and functional. These DO-DI wrap around tests are done within the operating system by the CHK program.

Analog wrap tests are also performed to check critical AOs. During system startup, the critical AOs (i.e. rod velocity signals) are "borrowed" and tested by the CHK program. Thereafter, they are tested as part of the application logic.

The control designer maximizes the system availability and minimizes the effects of failure by assigning control signals for the control drives to separate cards when possible.

<u>Defenses Against Computational Faults</u>
The first line of defense is hardware integrity. This is verified on line every execution of the CHK program. This task (or program) tests every type of calculation performed by the floating-point processor. An error in computed multiply or square root, etc. results in "killing" the watchdog.

The second line of defense is software integrity. This implies that no module should output bad data nor fail because of bad data. Consequently, every module in the system that provides floating point outputs for other modules, calls a library routine to clamp values to within the "legal" range. Similarly every routine using data defends itself against illegal data. The "legal" range is defined so that no combination of "legal" values can produce a result that causes hardware problems.

<u>Defenses Against Program Corruption Faults</u>
The major software line of defense is the CHK program and the way it is executed. This is a periodic task, which must run on time or the watchdog will drop out since it is the only program that can "kick" the watchdog. Since CHK is a periodic program, it can only be executed provided that the dispatcher is working correctly. This means that virtually the whole used set of the CPU instructions must work correctly, and that large segments of RAM must also be uncorrupted.

The CHK program and the dispatcher together perform the following checks:

- they trip "stuck" functions that are taking too long to execute. Any stuck function is terminated and a message is issued. If it is non-critical, the rest of the system continues to operate. If it is a critical function the watchdog is "killed".
- they check that each control program executes the correct number of times between passes of CHK. The watchdog is "killed" if this condition is not satisfied.

7.6.2.5 DCC-X/DCC-Z Self-Tests and Robustness Functions

The following three features are incorporated into ***PROTROL*** to detect the level of faults or degradation of the computer and its' I/O:

- self-tests on start up
- self-tests while on line

- built-in protection functions

These features include many generic tests and functions for the various ***PROTROL*** computer configurations (e.g. single control computer, dual control computer, monitoring computer, etc.). This section lists those functions used in DCC-X and DCC-Z for the PSBR application.

The results of the self-tests are logically combined to drive the state of two flags which give the level of capability for control and monitoring "healthy" and "fit". Under normal circumstances, both flags are true. For DCC-X, the watchdog is kicked as long as the "fit" flag is true. If unfit, the watchdog will drop out resulting in a SCRAM and DCC-X will switch to "inactive" mode. If either flag is false, an alarm message is issued. DCC-Z follows the same logic except that it does not have a watchdog.

The states of the DCC resulting from the tests are indicated in the upper right corner of the LCD monitor displays. The possible states are as follows:

- Initializing - state when DCC is started up and performs initializing functions and initial self-tests
- Controlling (DCC-X) - state after successful initialization
- Monitoring (DCC-Z) - state after successful initialization
- Inactive - state after failing a self-test performed by the CHK task

The ***PROTROL*** tests configured for the DCC-X and DCC-Z are given below. The tests which affect fitness are indicated by [F]. Otherwise, only health is affected by the test result.

<u>Self-Tests on Start-Up</u>
The self-tests on start-up are listed below. If a DCC-X fitness test fails, it will switch from "initializing" to "inactive" (instead of "controlling") mode, and drop its watchdog before ever allowing control task I/O operations. Failure of a fitness test in DCC-Z will switch modes from "initializing" to "inactive".

- COMPUTER CHECKS
    - [F] Floating point processor test (X and Z)
    - [F] Serial link reception error (Z). A red banner will be placed across the DCC-Z LCD monitor to highlight serial link failure.
    - Serial link transmission error (X)
    - Disk failures (X and Z)

- SOFTWARE FUNCTION CHECKS (X AND Z)
    - [F] all critical tasks executed twice to completion at defined frequency
    - [F] memory allocation OK

- I/O CHECKS (X)
    - [F] Test for excessive I/O retrys
    - [F] DI cards jumpered point closed test
    - [F] DO-DI wrap test
    - [F] borrowed AO-AI wrap test

- [F] Watchdog/Test card communications test
- [F] Watchdog contact wrap test

<u>Self-Tests While On Line</u>
The on line self-tests are listed below. If a fitness test fails, the associated computer will switch to "inactive " mode and DCC-X will "kill" its watchdog. The mode change seals in until the computer is restarted for all cases except serial link communication failure on DCC-Z. DCC-Z will resume monitoring once proper communications is restored.

- COMPUTER CHECKS
  - [F] Floating point processor test (X and Z).
  - [F] Serial link reception error (Z). A red banner will be placed across the DCC-Z LCD monitor to highlight serial link failure.
  - Serial link transmission error (X).
  - Disk failures (X and Z).

- SOFTWARE FUNCTION CHECKS (X AND Z)
  - [F] all critical tasks execute to completion at defined frequency.
  - [F] memory allocation OK.
  - [F] manual shutdown request.

- I/O CHECKS (X)
  - [F] Test for excessive I/O retrys.
  - [F] DI cards jumpered point closed test.
  - [F] DO-DI wrap test.
  - [F] Watchdog/Test card communications test.
  - [F] Watchdog contact wrap test.

The built in protection functions are not tests per se, but rather functions in software that prevent anomalous system operation failure or provide early warning of software problems in the verification and validation phases of design. For example, ***PROTROL*** system software disables keys used by the disk operating system (DOS) that influence software operation. Other functions are those to prevent catastrophic failure if a software function becomes stuck, memory overflow occurs, or there are illegal uses of internal hardware (e.g. floating point processor). The protection functions provided for DCC-X and DCC-Z are listed below.

PROTECTION FUNCTIONS (X AND Z)
- keyboard typo-matic rate slowed down to minimum
- keyboard buffer size reduced to minimum for immediate recovery from string of anomalous key strokes
- DOS keyboard buffer overflow prevented to eliminate possibility of "beep" function
- DOS system keys disabled such as:
  - CRTL-ALT-DEL
  - CTRL-ALT-S
  - PRINT SCRN
  - SYSRQ

- CTRL BREAK/C
- PAUSE
- stuck functions shut down
- warning for sequence of events message buffer overflow
- floating point processor exceptions retried after processor reset or offending function shut down
- tasks shutdown if their remaining stack space is low
- display driver error buffer cleared frequently to prevent its use of RAM
- display changes delayed to ensure that the display driver command list buffer doesn't overflow flowing a quick sequence of display change requests

7.6.3 Application Software

The application software is composed of a number of independent tasks, listed below. The tasks concerned with control logic are ***PROTROL*** block language tasks. The other tasks are written in PASCAL code.

7.6.3.1 Block Language Tasks

RRS	This is the reactor regulating system task that performs functions related to controlling reactor power on DCC-X. On DCC-Z, this task contains only records of signals broadcast from the DCC-X RRS task via the "one way" serial link for display purposes (e.g. no control logic).
SSS	This is the safety support slow task. On DCC-X, it performs the slow portion of the RSS support functions (e.g. air interlock validation). On DCC-Z, it contains only records of signals broadcast from the DCC-X SSS task similar to the RRS task above.
SSF	This is the safety support fast task. On DCC-X, it performs the fast portion of the RSS support functions (e.g. DCC-X SCRAM and rod movement interlock validation). On DCC-Z, it contains only records of signals broadcast from the DCC-X SSF task similar to the RRS task above.
FAC	This task performs functions related to control and monitoring of the reactor facility (e.g. facility interlocks for reactor operation and police services alarms). On DCC-Z, it contains only records of signals broadcast from the DCC-X FAC task similar to the RRS task above.

7.6.3.2 Non-Block Language Tasks

OPR	This is the operator controls interface task. It processes all operator keyboard inputs for display navigation and functions that are controlled through menus. On DCC-X, it drives I/O directly related to keyboard input (e.g. manual control of bay exhaust fans).
DSP	This task handles updating of the LCD monitor displays.
PULS	This task runs on a periodic basis to control the digital output to request air to the transient rod. When a reactor pulse is requested this task will read and process the GIC pulse range analog input signal and generate both the pulse data file and display driver command list for display of the pulse. This task exists only on DCC-X and thus its internal variables cannot be accessed on DCC-Z.

In the ***PROTROL*** operating environment, tasks have the following attributes:

- independent scheduling frequency
- definable scheduling priority relative to other tasks
- independent module of code with controlled interfaces to other tasks
- critical/non-critical attribute to determine whether or not failure will drop the watchdog

The principles used for the task structure outlined above, are as follows:

- The independent safety system support functions are separated into different tasks from control system functions.
- Hardware I/O operations for safety support and control functions are performed independently in the associated task. This means that common I/O signals related to safety are <u>not</u> read from the hardware in one task and then passed to the other task in software (even though this would reduce CPU loading).
- Functions that do not have a high execution frequency requirement are placed into separate slower tasks so that CPU loading is not made unnecessarily high. For this reason there are both fast and slow safety support tasks (SSF & SSS) and control tasks (RRS & FAC).
- Related functions are grouped into common tasks to minimize task interfaces (even though they may have varying execution frequency requirements).
- Functions related to control of the operator interface are implemented in non-block language tasks allowing use of conditionally executed code. For example, dynamic portions of LCD monitor displays are updated only when needed (i.e. a change has occurred). Much of the code must execute only when a key is pressed. This minimizes CPU loading without peak loading concerns since the operator cannot physically generate requests fast enough to cause a high peak load. The operator interface has been slowed down to provide additional assurance.
- Special purpose functions where the block language is mostly unusable, are implemented in non-block language tasks (e.g. PULS).

The design details of the various application software tasks as they apply to DCC-X and DCC-Z are provided in the design manual (reference [14]).

7.7 Radiation Monitoring Systems

Table 7-1 lists the monitors and type of detectors, their ranges and alert and alarm settings, for the major radiation monitors that provide inputs to the PCMS. All except the air east and air west monitors have readout modules located in an instrumentation panel as shown in **Figure 7-7.** For the monitors in the panel (except the Reactor Bridge West High) an alert results in an amber warning light on the monitor and a status and an alarm message is issued by the PCMS; an alarm results in a red alarm light on the monitor, an alarm message on PCMS, and an evacuation initiation issued by the PCMS (see **Figure 7-8**). The air east and air west monitors have their own local bells and flashing red light alarms that can be observed by the control room operator. A current radiation reading can be read on bar or trend graphs on the PCMS for all the monitors listed in Table 7-1.

Table 7-1
Control Room *Alarmed Radiation Monitors

Monitor	Detector	Range	Setting**
Reactor Bridge East	G-M Tube	0.1 to 10^4 mR/hr	Alert: 50 mR/hr Alarm: 200 mR/hr
Reactor Bridge West	G-M Tube	0.1 to 10^4 mR/hr	Alert: 50 mR/hr Alarm: 200 mR/hr
Reactor Bay South	G-M Tube	0.1 to 10^4 mR/hr	Alert: 50 mR/hr Alarm: 200 mR/hr
Reactor Bridge West High	Ionization	1.0 to 10^4 R/hr	No alarm
Reactor Bay Air East	Thin End G-M Tube	10 to 10^5 c/m	Alert: 6000 c/m Alarm: 10000 c/m
Reactor Bay Air West	Thin End G-M Tube	10 to 10^5 c/m	Alert: 6000 c/m Alarm: 10000 c/m
Co-60 Bay	G-M Tube	0.1 to 10^4 mR/hr	Alert: 3 mR/hr Alarm: 6 mR/hr
Neutron Beam Laboratory	G-M Tube	0.1 to 10^4 mR/hr	Alert: 10 mR/hr Alarm: 20 mR/hr

*An analog output from the monitors is an input to the PCMS. When that signal exceeds the set-point a status alarm is issued, an alarm message is issued and an Evacuation is initiated (see section 7.3.1.5).
**Information only; setting is determined internally and established by PSBR procedure.

Figure 7-7 Instrumentation Panel

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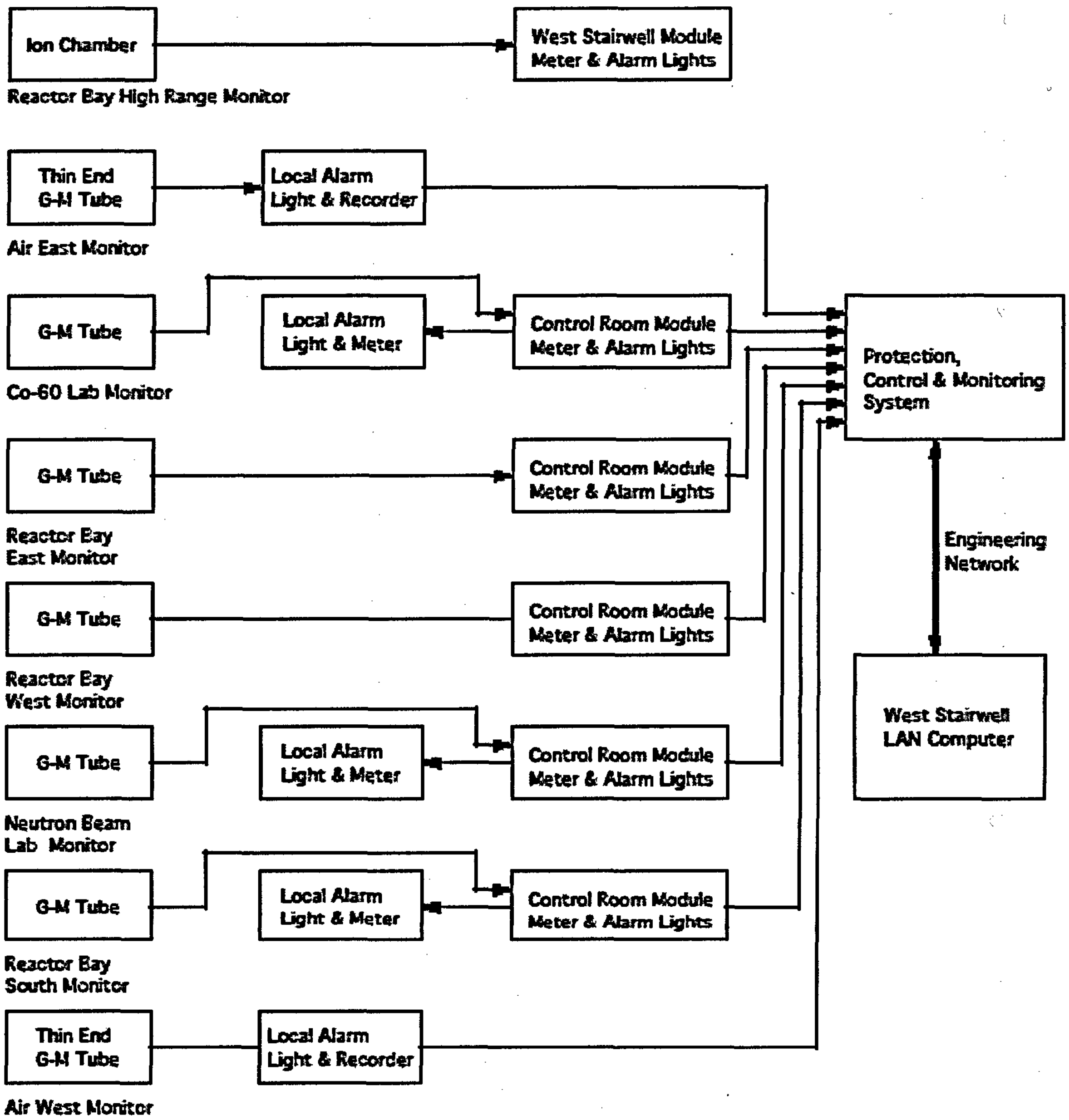

Figure 7-8 Radiation Monitoring System

7.8 Bibliography

[1] PSBR Console Specifications, Printed by PSU 88 Oct 27.

[2] DR-17-60501-001, Design Requirements, Reactor Safety System.

[3] DR-17-60501-002, Design Requirements, Control and Monitoring System.

[4] DR-17-60501-003, Design Requirements, Console Layout and Operator Interface.

[5] MM-17-60501-001, Maintenance Manual, PSBR CSS Upgrade, Control and Monitoring System.

[6] NUREG-0700, Guidelines for Control Room Design Reviews, 1981 September.

[7] MM-17-60501-004, Maintenance Manual, Gamma-Metrics Instruction Manual (#158), Reactor Control Console for Penn State University.

[8] OM-17-60501-001, Operating Manual, Control and Safety System.

[9] NEAT-406, Single Board Computer User's Guide 5x86 CPU Half Size with Flash Disk

[10] MM-17-60501-005, Maintenance Manual, NSK ***Megatorque*** Motor System User's Manual.

[11] RM-17-69020-001, Reference Manual, ***PROTROL*** Advanced Control Block Language: Control Designer's Reference Manual.

[12] TR-17-60501-001, Test Report, Reactor Safety System (Filled-Out copy of G-M Dwg #040264).

[13] MM-17-60501-003, Maintenance Manual, Computer Products I/O Hardware.

[14] DM-17-60501-001, Design Manual, Control and Safety System.

[15] Regulatory Guide 1.152, *Criteria for Programmable Digital System Software in Safety-Related Systems of Nuclear Power Plants*, U. S. Nuclear Regulatory Commission, Nov 1985.

TRADEMARKS

Trademark	Company
IBM, AT	International Business Machines Corporation
PROTROL	Atomic Energy of Canada Ltd.
Megatorque	Motornetics Corporation; subsidiary of Nippon Seiko K.K. (NSK)
TRIGA	General Atomics, San Diego, CA

GLOSSARY

ADC	Analog to Digital Converter
AI	Analog Input
AO	Analog Output
BIOS	Basic Input Output System
CANDU	Canada Deuterium Uranium Nuclear Power Reactors (PHWR)
CCTV	Closed Circuit Television
CPI	Computer Products Inc.
CRC	Cyclic Redundancy Check Code- This is a polynomial based error detection method typically used for network systems. This is a Consultative Committee on International Telephony and Telegraphy (CCITT) standard.
CSS	Control and Safety System
DAC	Digital to Analog Converter
DBE	Design Basis Event
dec	Decades - 10 folds of fraction full power
DEFPAR	Definable Parameter *(PROTROL* block 0 tuning)
DCC	Digital Control Computer. The DCC-X computer performs all required protection, control, and monitoring functions. The DCC-Z computer performs only monitoring and historical data collection. The DCC-Z computer is not required to be operational for safe operation of the reactor.
DI	Digital Input
DO	Digital Output
DOS	Disk Operating System
dps	Decades per Second
EES	Emergency Exhaust System
EOT	End of Travel
fac	Factor
FC	Fission Chamber
FES	Facility Exhaust System
ffp	Fraction Full Power
fps	Fractional Power (present value) per Second
frac	Fraction
GIC	Gamma lon Chamber
LED	Light Emitting Diode
I/O	Input/Output
ILDB	Interloop Data Base
LAN	Local Area Network
LN	Natural Logarithm
LSS	Limiting Safety System
PCMS	Protection, Control and Monitoring System
PSBR	Penn State Breazeale Reactor
PSU	Penn State University
RAM	Random Access Memory

RSS	Reactor Safety System
SBC	Single Board Computer
SCRAM	Reactor Safety Shutdown
TC	Thermocouple
TR	Transient Rod
Watchdog Circuit	A circuit consisting of a timer and a relay. The timer energizes the relay as long as it is reset prior to the expiration of the timing interval (see figure 7-7). If it is not reset within the timing interval, the relay will de-energize thereby causing a SCRAM. The term "kicked the watchdog" is slang for resetting the watchdog timer. The term "killed the watchdog" is slang for ceasing to reset the watchdog timer.
Wrap Around	Describes the situation where an output signal is fed back so that the output signal can be verified (see figure 7-7).

8. ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

Electrical power is supplied to the facility through a dedicated three phase transformer located inside the reactor site boundary fence (see **Figure 2-2**). The power is supplied by the Allegheny Power Company.

The normal building power supplies an Uninterruptable Power Supply (UPS), which supplies power to important equipment. A diesel generator starts automatically if building power fails, to supply the UPS and other equipment. The functions of UPS/diesel generator electrical power for the PSBR are mostly for operational convenience, assurance of equipment integrity by using an UPS as a power filter, radiation monitoring, reactor bay exhaust, fire alarm protection, and lighting for personnel safety for all conditions. A block diagram for the normal building electrical power system is shown in **Figure 8-1**.

Reactor shutdown is passive and fail-safe in that if normal, UPS, and diesel generator power are all lost, the control rods automatically fall into the core due to gravity shutting down the reactor. The major methods of adding water to the pool in case of a leak (secondary heat exchanger water from Thompson's Pond and university water) are not dependent on reactor building power. There are no limiting conditions for operations in the TS that require building power, UPS power, or diesel generator power when the reactor is secured.

An UPS system is maintained in the reactor bay and the devices listed in **Table 8-1** operate off of the UPS at all times unless the devices are intentionally switched to building power. The 5 kW UPS system contains an internal 4-battery bank that is kept in a charged state by normal building power (208VAC) via a built-in 10A charger. The battery bank in turn supplies power (240VAC), via a static inverter, to the following facility devices so that in the event of a power failure, the devices continue to operate normally (with the exception that the air monitor pumps are not supplied by UPS).

Table 8-1 Devices Powered by UPS

Reactor Bay East Rad Monitor	PA System including the evacuation alarm
Reactor Bay West Rad Monitor	Control Room Printer
Reactor Bay South Rad Monitor	[redacted]
Reactor Air East Rad Monitor	
Reactor Air West Rad Monitor	Rod Drive Motors and Rod Magnets
Neutron Beam Lab Rad Monitor	Emergency Light Above Console
Cobalt Bay Rad Monitor	Auxiliary Control Room Cabinet (includes spare wide range drawer and amp and anything else in the cabinet)
Cobalt Bay RA-1 Remote Meter	
Reactor Console	

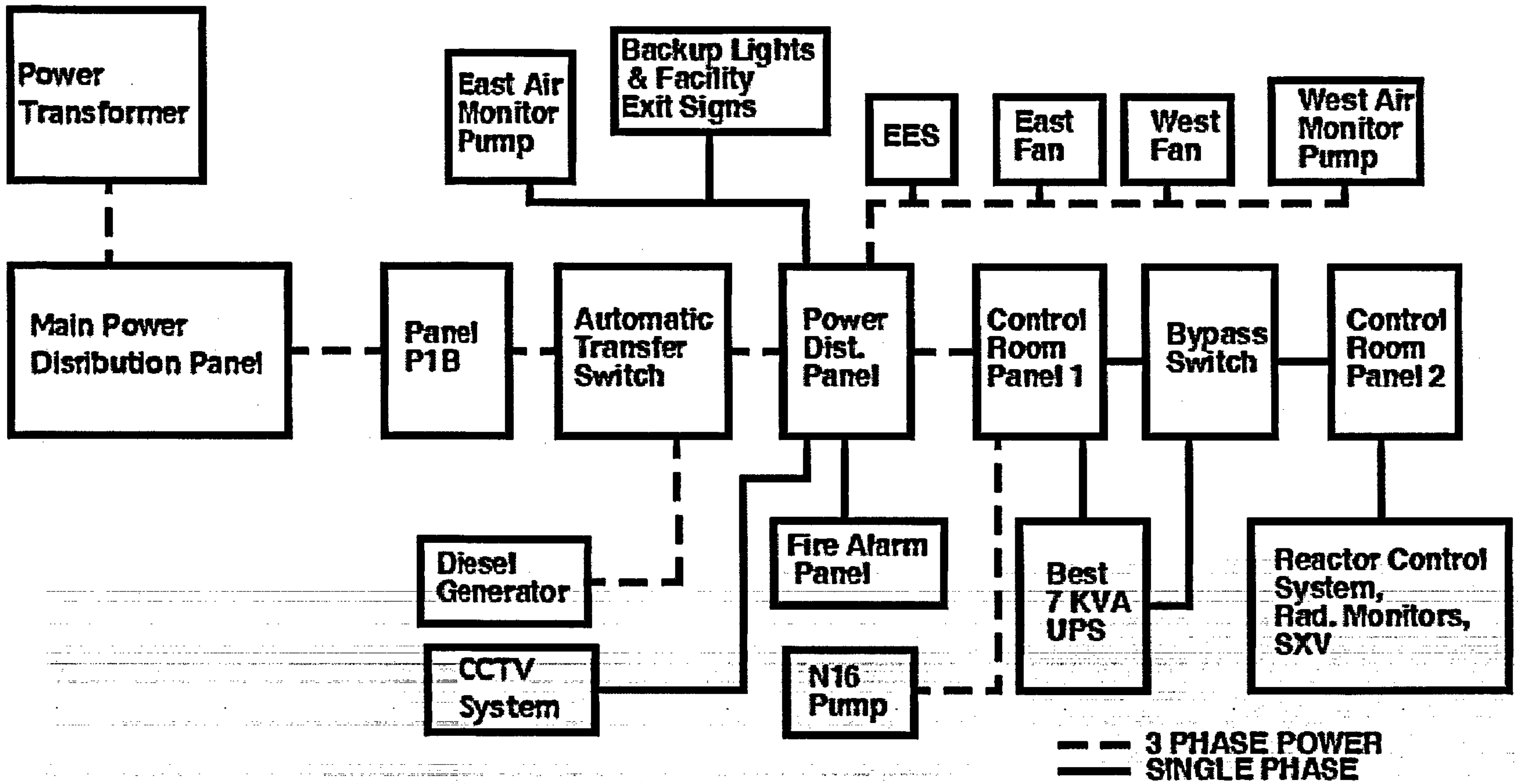

Figure 8-1 Building and Control System Power Distribution

On a loss of AC to the UPS, the UPS sends a "UPS-1 AC Supply Off" signal to the console. Also, an "UPS-1 Battery Low" message is sent to the console after 5 minutes of sustained building power loss to the UPS. When tested with new batteries, the system was capable of maintaining the load of 1.22kW (Input: 33A at 48VDC/ Output: 13.1A at 240VAC) in excess of one hour.

The diesel generator (16kW) is located outside of the Cobalt facility (north side) and starts upon failure of building power. The diesel generator supplies power to the devices in **Table 8-2.** No credit is taken or assumed for the diesel generator.

Table 8-2 Devices Powered by Diesel Generator

UPS System	Facility Fire Alarm System
Air Monitor Pumps	Facility Emergency Lighting System
Facility Exhaust System Fans	West Stairwell LAN Computer and Monitor
Emergency Exhaust System Fan	Co-60 Bay RA-1 Remote Siren and Beacon

8.2 Emergency Electrical Power Systems

There is no equipment at the PSBR that requires emergency power to maintain the reactor in a safe and secured condition. There are no limiting conditions for operations in the TS that require building power, UPS power, or diesel power when the reactor is secured.

If during a building power outage the diesel generator is available to power the emergency exhaust system (EES), the events discussed in Chapter 13, Accident Analyses, would be further mitigated. However, even if the EES is not available, exposures to the public are less than any limits in 10 CFR 20. It should be noted that TS 3.5.a (chapter 14) allows the EES to be out of service for periods of time less than 48 hours during maintenance or repair even when the reactor is not secured.

9. Auxilliary Systems

9.1 Heating, Ventilation, and Air Conditioning Systems

The air in the reactor bay and control room is heated and cooled by a dedicated reactor bay air conditioner. This unit recirculates, heats, cools, and dehumidifies reactor bay air as required, providing an acceptable environment for personnel and equipment. No air is interchanged with any other part of the building or outside of the building by this unit. Heating is supplemented by steam unit heaters as needed. The condensate from the reactor bay air conditioner can be piped into the reactor pool as makeup water to help compensate for pool water evaporation. The neutron beam laboratory has a separate air conditioner to provide cooling to that area; heat is supplied to the room by steam unit heaters located near the ceiling. No heating or cooling is provided for the demineralizer room. Steam for the heating system is supplied from University power plants located at the east and west ends of the campus. The reactor bay and neutron beam laboratory HVAC systems cannot operate in any way as to interfere with the reactor bay facility exhaust system or emergency exhaust system (see section 6.2.1). For the case of airborne radioactive materials, any effect the HVAC would have on the distribution and concentration of those materials would be confined to the reactor bay, and would be secondary to the effect of the exhaust systems (see section 6.2.1).

9.2 Handling and Storage of Reactor Fuel

[redacted]

Technical Specification 5.4 states that:

a. All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all conditions for moderation.
b. Irradiated fuel elements shall be stored in an array which shall permit sufficient natural convection cooling by water such that the fuel element temperature shall not reach the safety limit as defined in Section 2.1 of the TS.

The PSBR uses fuel storage racks based on a General Atomics design, to meet the 0.8 k_{eff} requirement. On file at the PSBR is a letter of March 1, 1966 from Fabian C. Foushee of General Atomics/General Dynamics, subject: "Storage of TRIGA Fuel Elements." Two methods are used to show that the storage is safe. The first method uses a criticality safety limit taken from a GA document GA-5402, "Criticality Safeguards Guide". This reference gives a very general limitation on the storage of well moderated U-235 as an average of 300 gm of U-235 per square foot of aspect area. Assuming in our case 12 wt% elements containing at most [redacted] of U-235 per element stored in the GA racks, than the concentration of fissile material [redacted]. This means that elements can be infinitely long and arranged in an infinite array and meet this safeguards requirement. The second method used was to calculate the k_{eff} of the element storage as an array one element thick and as any array two elements thick. The latter arrangement assumes that two racks [redacted] hanging front to back with no separation [redacted]. This arrangement is not physically possible because the front to back spacing can be a minimum [redacted] because of the rack design. Therefore these calculations are conservative. The results for 8.5wt% fuel as stated in the GA letter (Foushee, March 1, 1966) are as follows:

For 8.5 wt% elements	k_{eff}
Plane array one element thick	0.5096
Plane array two elements thick	0.7227

Calculations by Dan Hughes at the PSBR in November 1, 1994, indicate that by increasing the U-235 to 12 wt%, the only factor changed is the thermal utilization (f) which increases by 10.24%.

For 12 wt% elements	k_{eff}
Plane array one element thick	0.5618
Plane array two elements thick	0.7967

These results are not only conservative because the spacing of the racks back to front is assumed at 1.47 inches rather than the 2.5 inches provided by the racks, but the calculations use a homogeneous system rather than a lumped fuel system. In addition, this modification ignored the increased self-shielding of the higher loaded elements. Additional conservatism is added by the fact that the centerlines of the elements in the front and back rows of the storage rack have a 20 inch centerline to centerline vertical separation. Figure 9-1 shows the elevation view of the storage racks with regard to the front and back rows of the rack, [redacted]. The conclusions are that we can be confident that the GA storage racks [redacted] have a k_{eff} less than 0.8 as required by the TS.

[redacted]

[redacted]

The TS 3.4,3.5.b, and 4.4 (chapter 14) give limiting conditions for operations and surveillance requirements to assure the confinement is maintained whenever the reactor is not secured, or fuel or a fueled experiment with significant fission product inventory is being moved outside containers, systems, or storage areas. The bases of the TS serve as the background or reason for the TS requirement. Appropriate PSBR standard operating procedures enforce and reference the TS.

TS 4.1.3 (Chapter 14) gives surveillance requirements for inspection of fuel elements being placed in the core for the first time, periodic inspections while in use, and upon removal from service. There are no TS requirements for inspection of fuel in storage. PSBR administrative procedures require periodic inventory of reactor fuel.

The MHA discussed in Chapter 13, Accident Analyses, discusses the effects of a rupture of a fuel element in air.

No TRIGA fuel has been shipped from the facility to date. It is not expected that DOE would receive any of our spent fuel before 2012. See **Table 11-2** for PSBR Fuel Inventory.

9.3 Fire Protection Systems and Programs

The reactor building is constructed of concrete blocks, bricks, insulated steel and aluminum panels, structural steel, and re-enforced concrete and is in general, fireproof in nature. There is very little flammable material in the reactor bay.

The building (including the reactor bay and control room) is equipped with a comprehensive fire alarm system consisting of manual pull stations and smoke detectors. Smoke detector alarms indicate to a control room panel and a lobby entrance panel [redacted] Pull stations throughout the building assure quick personnel response and smoke detectors help assure early detection of a fire event. Automatic fuse activated sprinkler systems cover [redacted] (a sample preparation laboratory and shipping and receiving area for materials under the R-2 license) and the two facility Hot Cells. The sprinklers also alarm to the aforementioned panels [redacted] The hot cell sprinklers also sound an alarm bell on the hot cell loading dock. The comprehensive fire alarm system's reliability is maintained by documented periodic operability checks by the University Office of Physical Plant (OPP). The fire alarm system is powered by building power with available diesel generator power if needed (see section 8.1).

Fire extinguishers of either the CO_2 type or compressed air and water type are located at strategic locations throughout the building. Reliability is maintained by documented periodic checks by the OPP personnel. Fire fighting protection for all University buildings, including the reactor building, is provided by the Alpha Fire Company of State College. The firehouse is located approximately 1 1/2 miles (2.5 km) from the reactor building. A fire hydrant is located outside of the reactor site boundary fence, approximately 320 feet (98 m) from the southwest corner of the building.

PSBR Technical Specification 6.3, Operating Procedures, requires the facility to have a procedure for Fire or Explosion. A PSBR emergency procedure fulfills that requirement and provides guidance to the reactor staff for response to a fire alarm, and classifies all rooms in the building as to their potential fire hazard; locations of pull stations, smoke detectors, sprinkler systems, and fire extinguishers are also described. The PSBR Operator and Senior Operator Requalification Program, requires an annual oral exam on all emergency procedures. The PSBR Emergency Plan (EPP), Section 3.1, requires written agreements between the PSBR and the Alpha Fire Company of State College and this requirement is assured by a PSBR administrative procedure that requires periodic renewal of the letter.

The major radioactive inventory under the R-2 license would be the fission product inventory in the reactor fuel elements. No fire event is postulated that could cause damage to the reactor fuel. However under some circumstances a fire could be classified as an Alert or Unusual Event under the EPP (see sections 4.1 and 4.2 of the EPP).

Reactor shutdown is by means of four control rods, three of which are held out of the core by electromagnets. The other rod is held out by compressed air supplied through an electrically operated solenoid valve. The control rods are fail-safe in that failure (due to fire or otherwise) of electrical systems associated with the rods would cause the rods to fall into the core due to gravity. No other safety systems are required by the TS when the reactor is shutdown and no reactor fuel is being moved.

9.4 Communication Systems

The telephone system in the reactor building consists of phones in rooms and laboratories and the system's functionality is maintained for short periods of time by its own UPS system in case of a building power failure. Communication over the facilities Public Address (PA) System is possible using these phones. The PA system is powered by the main facility UPS and therefore the control room microphone should be available at all times. The building evacuation alarm that is initiated by DCC-X also operates over the PA system. Other phones independent of the telephone system are also available in the reactor control room and other areas of the building and operate on phone company voltage.

[redacted]

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

A PSBR administrative procedure identifies rooms or locations where radioactive materials in the facility are considered to be under the R-2 license. This is primarily reactor fuel, reactor core components and support structures, and other materials transported to and from specific facility areas designated in the PSBR procedure. These other materials may include customer samples awaiting shipment and transfer to customer licenses, or experimental apparatus used in the reactor or neutron beam lab that needs to be taken to other areas for research and development or maintenance and repair. The PSBR administrative procedure assures that rooms assigned for R-2 use have the necessary equipment to monitor the radioactivity.

Many samples made radioactive from exposure to reactor neutrons are transferred to the University's Broad Byproduct License upon removal from the pool. The University Isotope Committee (UIC) authorizes individuals to possess radioactive materials in certain quantities for specific purposes. Byproduct material from the reactor is only released to a person having a valid UIC authorization or to another NRC license.

Storage and use of all radioactive material at the PSBR, is monitored by the university's RPO, including material under the R-2 license.

9.6 Cover Gas Control in Closed Primary Coolant Systems

Not applicable to the PSBR.

9.7 Other Auxiliary Systems

9.7.1 Air Compressors

Compressed air for all facility needs is supplied by two air compressors (see **Figure 9-2**) located in the mechanical equipment room. This room is located under the facility machine shop. Normally, the large air compressor supplies air to the reactor transient rod line and to the line that goes to the remainder of the building, with the small compressor in a standby mode (if the large compressor fails to operate, the small compressor operates automatically as needed). Valves AC-1, AC-2, AC-3, and AC-4 are normally open (closing AC-1 or AC-2 would allow the small air compressor or large air compressor, respectively, to alone supply all transient rod and building air needs if the other compressor is taken out of service). Air from both compressors is treated by particulate filters, an oil filter, and a Hankison Refrigerated Type Compressed Air Dryer, all located in the mechanical equipment room. The large air compressor tank, the small air compressor tank, the Hankison dryer, and the filters in the lines in the mechanical equipment room are automatically relieved of moisture accumulation.

The large air compressor's operation is controlled by that compressor's air pressure switch. The compressor starts at ~95 psig and stops at ~115 psig. The small air compressor's operation is controlled by that compressor's air pressure switch. This compressor starts at ~80 psig and stops at ~105 psig. Since the small compressor starts at ~80 psig, it will only start if the large compressor fails to start at ~95 psig.

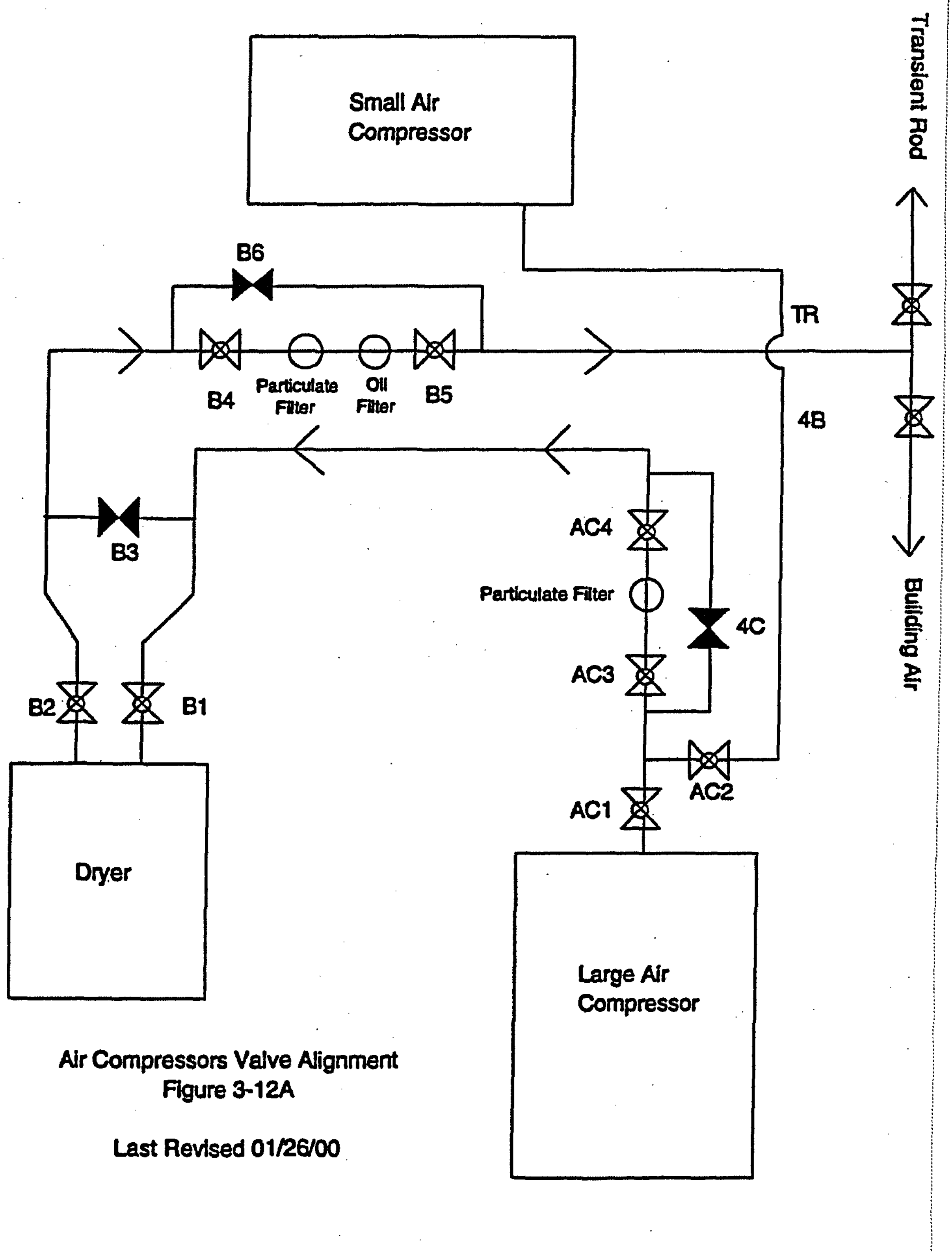

Figure 9-2 Air Compressors Valve Alignment

The building air line goes to a dryer in the machine shop (just outside the door to the demineralizer room), which removes grease and oil, and then the line branches to several building locations. Additional dryers are located in the system as appropriate. These additional dryers are drained to relieve moisture accumulation as per a preventative maintenance schedule. An alarm pressure switch in the building air line in the reactor demineralizer room, provides an input to DCC-X and a "Building Air Supply Pressure Low" message is indicated if air pressure drops to ~90 psig.

The transient rod air line runs from the mechanical equipment room to the air dryer on the reactor bridge, regulator (normal line pressure ~ 80 psig), alarm pressure switch, accumulator tank, solenoid valve and transient rod, in that order. The alarm pressure switch provides an input to DCC-X, and a "Tran Rod Air Supply Press Low" message is indicated if air pressure drops to ~60 psig.

9.7.2 Evaporator - Liquid Radioactive Waste Treatment

The purpose of the evaporator (see **Figure 9-3**) is to remove water from liquid radioactive waste so that only a small residue of solid radioactive materials remains. Since demineralizer resins are currently replaced when expended and not regenerated, the facility does not normally generate liquid waste in quantities that need evaporation. Currently city water is processed through the evaporator to provide a source of distilled makeup water for the reactor and Co-60 pools. On occasion, reactor pool water that remains in the 48,000 gallon (1.8×10^5 ℓ) hold-up tank following pool water transfers, is pumped to the evaporator building 4000 gallon (1.5×10^4 ℓ) floor tank and later evaporated for pool make-up water.

If liquid radioactive waste were produced from demineralizer regeneration or other source, it could be sent to either the 2000 gallon (7.6×10^3 ℓ) underground waste hold-up tank near the evaporator building or the 4000 gallon (1.5×10^4 ℓ) waste hold-up tank in the floor of the evaporator building. Liquid waste from either of these two waste hold-up tanks can be pumped to the evaporator feed tank by use of the transfer pump. From the feed tank the liquid waste is moved to the evaporator by the feed pump. A level control valve at the evaporator allows only enough of the flow in the feed loop to enter the evaporator to maintain a proper level. An eductor at the evaporator allows for some of the liquid waste in the evaporator to be returned to the feed loop (this allows for a more even mixture in the evaporator and the feed tank). The liquid waste in the evaporator is heated to the boiling point by very hot water passing through heating coils in the evaporator's bottom portion (boiling is aided by maintaining a vacuum in the evaporator). The resultant steam travels to the top portion of the evaporator where it is condensed (with cooling water passing through loops) and collected as distilled water. The distillate pump moves the distilled water to the distillate tank and from there it can flow (by gravity) to the 6000 gallon (2.3×10^4 ℓ) underground processed water tank until it is needed as pool make-up water. Waste residue from the evaporation process would be removed from the evaporator, further solidified as needed and disposed by the RPO.

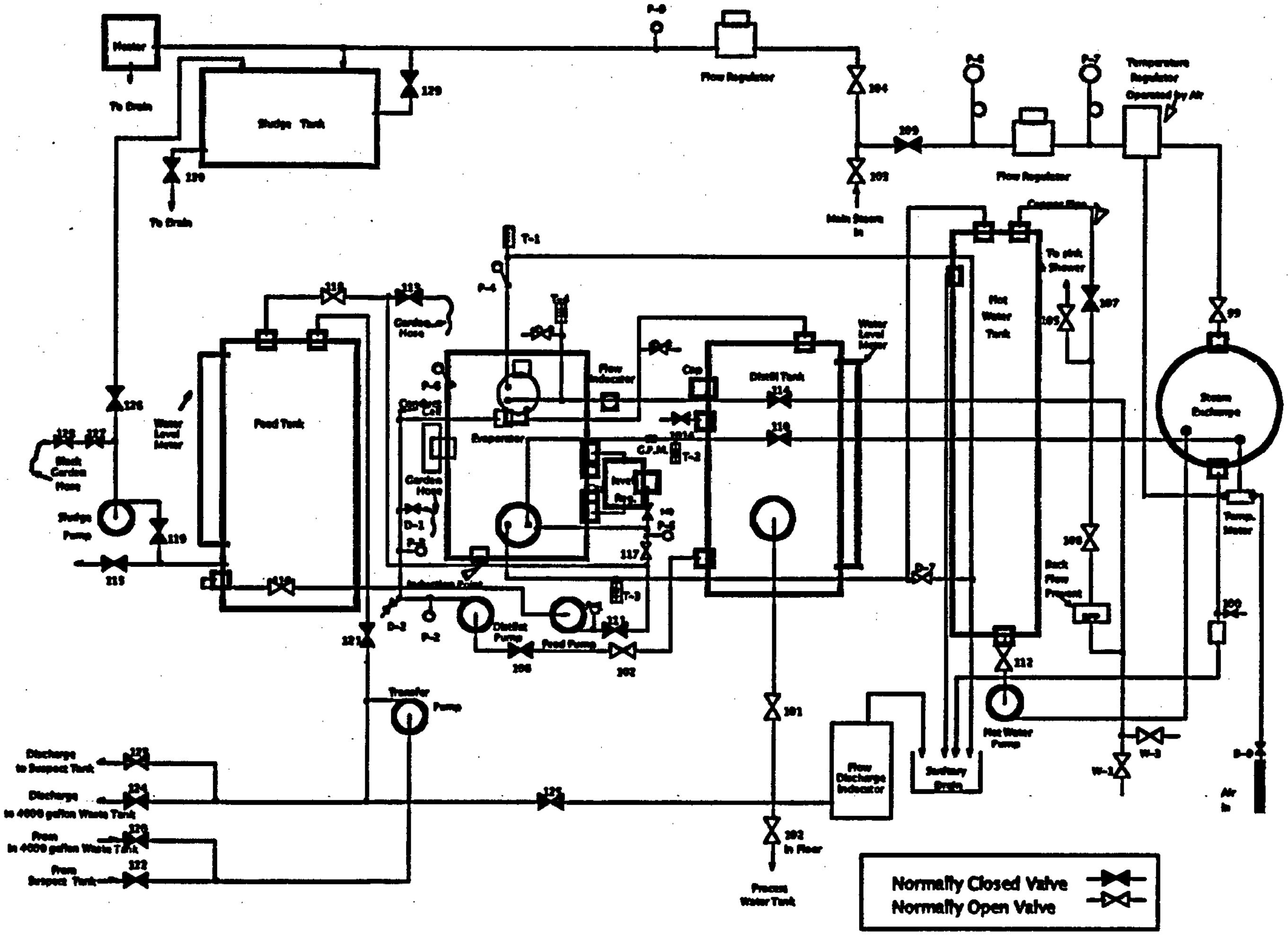

Figure 9-3 The PSBR Liquid Waste Evaporator System

9.7.3 Reactor Bay Overhead Crane

The reactor bay is equipped with a 3 ton (2722 kg) overhead crane supported by the building structure. The major function of the crane is to move experiments or experimental facilities or parts thereof such as the FFT and FNI shield plugs (see Section 10.2.4). It is also used to position the pool divider gate (see Section 4.3).

9.8 Bibliography

1. Letter of March 1, 1966 from Fabian C. Foushee of General Atomics/General Dynamics, subject: "Storage of TRIGA Fuel Elements."

2. GA Document GA-5402, "Criticality Safeguards Guide."

3. Interoffice Correspondence from Dan Hughes to Marc Voth, November 1, 1994, "GA Fuel Storage Racks."

10. EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The PSBR is used primarily for education, research and service. Neutron activation analysis, neutron radiography, neutron radioscopy, neutron transmission measurements, and neutron depth profile measurements are some of the current research and service activities. The current major experimental facilities are neutron beam ports coupled to the reactor through thermal columns, vertical tubes of various configurations (both in-core and in-reflector), two fast neutron irradiators, a central thimble, and an automatic sample transfer facility (rabbit system). The locations of present experimental facilities are shown in **Figure 10-1**.

Administrative control of experiments and experimental facilities is by PSBR standard operating procedures, which implement requirements of the TS. Standard operating procedures exist for experiment evaluation and approval, experiment encapsulation and irradiations, qualification of reactor pool operating positions, pneumatic transfer system (R1) operation, reactor operation using a beam port experimental facility, and reactor operation using fast neutron irradiation facilities. Experiment evaluation and approval includes applicability to the TS limits, including ALARA concerns and effluent release (mainly Ar-41), and 50.59 considerations.

10.2 Experimental Facilities

10.2.1 Beam Ports

In the south end of the reactor pool, seven beam ports penetrate through the pool wall as shown in **Figure 10-2** into the neutron beam laboratory (NBL) for dry irradiation purposes. The beam ports extend through the pool wall plus an additional high density concrete biological shield that provides a total wall thickness of five feet. Except for beam ports #4 and #7, the pool side of each beam port is currently sealed with a gasket and a blank flange to prevent water leakage. For beam port #4, a gasket seal is used to connect the beam port flange and the collimator that is an integral part of the D_2O tank. For beam port #7, a gasket seal is used to connect the beam port flange and the collimator tube that extends to the graphite box aside the D_2O tank.

Doors constructed of ¼ inch (.64 cm) stainless steel plate and poured full of lead (~ 4 inches or ~10.2 cm) are hung on wheel and track arrangements on the NBL wall for six of the seven beam ports (exception is beam port #7). The doors plus two stepped aluminum shells, 2 ½ feet (0.8 m) and 3 feet (0.9 m) long, which are poured full of high density concrete, can be used to provide shielding for six of the seven beam ports when they are not in use. The aluminum/concrete shields also reduce air movement to and from the beam tubes to minimize Ar-41 production and release. Beam Port #7 contains a permanent three-beam collimator (each beam 1 cm^2).

[redacted]

[redacted]

The approximate inside diameter measurements of each of the beam ports is as follows:

- Beam Ports 2, 4, and 6 are 7 inches (17.8 cm)
- Beam Ports 1 and 7 are 4 inches (10.2 cm)
- Beam Ports 3 and 5 are 3 inches (7.6 cm)

Elevations of the beam ports with respect to the pool floor are shown in **Figure 10-3**.

Beam Port #4 access is through a locked shield cave door, with access limited to senior reactor operators. The beam port #4 door and air actuated door controls are also both locked and under senior reactor operator control. Beam Port #7 beams terminate in a beam stop, with senior reactor operator key control of the area between the beam port exit and the beam stop.

Because of mechanical limitations (the D_2O tank, BP#4 collimator, and BP#7 collimator are fixed in position), the reactor cannot be moved closer than 27 inches (68.6 cm) to beam port #4. Presently, the reactor is coupled to beam port #4 by way of a D_2O tank. The reactor is operated tangentially to the D_2O tank to increase the beam's neutron to gamma ratio. From this position, reactor neutrons can also travel from the D_2O tank through a graphite box and down the beam port #7 collimator, where the neutrons are eventually attenuated by a beam stop.

Inflatable test plugs are available in the NBL that can be used for emergency repair in the event of a water leak through a beam port. Area radiation monitoring is provided by a G-M area monitor (see section 7.7). This monitor is equipped with a local alarm and readout as well as a remote alarm and readout in the reactor control room. Two remote manual scram buttons are located on the east and south walls of the NBL. A digital input is available to the PCMS to provide a reactor operation inhibit and DCC-X message "N-Beam Lab Radiation Hazard." The system to provide the digital input can be configured as necessary to provide an appropriate interlock and/or warning system as administratively determined.

10.2.2 D_2O Thermal Column

The present D_2O tank (see **Figure 10-2**) went into service in 1997. The tank is constructed of 6061-T6 aluminum and measures 24 inches (61 cm) in diameter and is 12 inches (30.5 cm) deep. Construction materials were selected to minimize eventual radioactive waste disposal problems. The tank is designed to optimize the thermal neutron beam intensity using beam port #4. The major use of the beam port #4 beam is for neutron radiography and neutron radioscopy.

A collimator tube that is an integral part of the D_2O tank, is bolted to the beam port #4 flange (using a water tight gasket). The tank is thus fixed in position. A shielding cover protects the collimator tube and is designed to withstand an impact [redacted] The tank is designed to operate tangentially to the reactor core to increase the neutron to gamma ratio.

[redacted]

A gamma shield is located at a 45 degree angle to the collimator tube to minimize reactor core gammas from entering the tube. Gamma shielding is provided for the pool wall by a 2 feet x 2 feet x 1 inch (.61 m x .61 m x 2.54 cm) lead shield around beam port #4. Gammas come directly from the core and are also induced by neutron absorption by the hydrogen in the pool water.

The tank was leak tested after fabrication and care was taken during installation to minimize and monitor stress on the beam port penetration. Studies show (G. C. Geisler, S. H. Levine, and I. B. McMaster, " A D_2O Thermal Column at the Breazeale Reactor," Penn State University Nuclear Engineering Department, Unpublished Paper with no Date) that D_2O leaking into the core should have the effect of decreasing core excess reactivity. Two effects occur when D_2O replaces H_2O in the TRIGA core. The lower neutron absorption cross section of D_2O causes a slight increase in the k_∞ of the core and the larger age and diffusion length of the D_2O causes an increase in the neutron leakage. Since the TRIGA is a small core with high leakage ($k_\infty = 1.44$), the leakage effect is predominant.

Tests run to qualify the new tank for use showed that operation against the tank has a minimal effect on the reactor. With the reactor against the tank (as compared to measurements made with the reactor in an open pool position) the worth of the safety rod increased 17 cents (about 3%) and fuel temperature (measured at 750 kW(th) by a 8.5 wt% instrumented element in the C-row) increased 14° C (about 5 %). Since the core neutron detector is away from the core to tank interface, any effect on detector calibration is minimized. Calculations show that the tank and support structure are not buoyant with the tank empty. The measured reactivity worth of the tank is approximately 60 cents positive, and this value will vary somewhat depending on the core loading and whether the rod worth curves used for the calculation were obtained in the open pool or against the tank.

The tank is filled or emptied by the use of two aluminum tubes from the tank to the surface of the pool. At the top of the two tubes is an expansion tank which is equipped with a level indicator, pressure gauge, a sampling port, a circulation pump used prior to collecting D_2O samples, and a pressure relief valve that would release to the reactor bay if tank pressure exceeded 13 psi. The D_2O was moved into the tank by pressurizing the storage drums (in which it was originally shipped) with inert gas to force the D_2O down the fill line, which enters the bottom of the tank. Removal of D_2O from the tank would be accomplished by applying gas pressure through the vent line, and allowing the D_2O to flow out through the fill line connected to the bottom of the tank back to the storage drums. A PSBR D_2O handling procedure describes in detail how to fill and empty the tank. Because of the ease with which tritium can be taken into the body by inhalation, ingestion, and absorption through the skin, the procedure requires the presence of the radiation protection office personnel when D2O is being transferred. To keep track of the tritium activity in the tank, a sample is taken from the tank each month and analyzed as per a PSBR procedure.

In August of 1998, facility modifications were made to open beam port #7 for use. A new aluminum extension tube attached by flange to beam port #7 was added. The tube has a shield to withstand impact. The extension tube features an internal collimator with three small collimator tubes of approximately 1 cm^2 each in areas to provide neutron beams for neutron transmission measurements of borated materials. The extension tube interfaces with the D_2O tank through a graphite interface box (approximately 9 inches x 24 inches or 22.9 cm x 61 cm) that was added to the back side of the D_2O tank. The interface box was needed to optimize the intersection of the aluminum extension tube with the D_2O tank. The graphite box is pressurized with air to prevent the entry of water (should the box leak) that would degrade the performance of the graphite. The box pressure is regulated in the range of 12 – 14 psig with a pressure release valve setting of ~15 psig. The graphite box features a thimble in which a rabbit system terminus could be placed in the future.

10.2.3 Vertical Tubes

Certain experiments such as electronic circuits cannot be submerged in water to be irradiated. For this type of irradiation, vertical tubes are available. A vertical tube is an air filled aluminum tube that extends from the reactor core level to above the reactor pool level. The vertical tubes are weighted so that they do not float and in some cases the lower ends are designed to plug into the bottom reactor core grid plate. In other cases, the tubes are supported by the instrument bridge.

Several sizes of round vertical tubes are available ranging from 1 ¼ inches (3.18 cm) up to 6 inches (15.2 cm) I.D. Other vertical tubes currently available have rectangular cross sections as large as 2 inches x 6 inches (5.1 cm x 15.3 cm). At times, tubes are specially made to accommodate a particular experiment. Presently, two 1 ¼ inch (3.17 cm) tubes (DT1 and DT2) are permanently located within the reactor core fuel region. All iradiation tubes have bends and/or plugs to reduce radiation streaming. The plugs also reduce air movement to and from the tubes to minimize Ar-41 production and releases.

A 1/2 ton (454 kg) hand operated jib crane is mounted on the floor of the reactor bridge. The jib crane is used to support shielding plugs used in vertical tubes. The shield plugs are lowered into the open ended vertical tubes and extended below the pool surface to minimize radiation dose rates above the tubes.

10.2.4 Fast Flux Tube (FFT) and Fast Neutron Irradiator (FNI)

The Fast Flux Tube (FFT) was placed into service in 1984. The FFT (**see figure 10-1**) uses an annular design for the dry irradiation tube and the surrounding lead, boron and cadmium shielding. The inside diameter is about 6 inches (15.3 cm). Originally designed for silicon wafer irradiations, the tube is now used for neutron irradiations for commercial, space, and defense applications. A massive shield plug (with a built-in motor to rotate samples) mitigates radiation streaming. The plug also reduces air movement to and from the tube to minimize Ar-41 production and releases.

The Fast Neutron Irradiator (FNI) (see **Figure 10-1**) was placed into service in 1997 and was designed to accommodate large size silicon wafers. A massive shield plug (with a built-in motor to rotate samples) mitigates radiation streaming. The plug also reduces air movement to and from the tube to minimize Ar-41 production and releases. The FNI has a 10 inch (25.4 cm) inside diameter annular dry irradiation tube, but the lead and the boron shielding are rectangular by design to provide a flat coupling face between the FNI and the reactor core, to minimize the water moderator effect. Provisions are made to allow for the expulsion of water from the enclosed aluminum cowling that surrounds the lead and boron shield to eliminate any water moderation within the shield. The tube is designed with a negative buoyancy under all conditions. One side of the FNI has 2 inches (5.1 cm) of lead while another side has 4 inches (10.2 cm) of lead. Therefore, positioning the reactor against one side or the other would vary the neutron-gamma characteristics in the tube.

The FNI reactivity worth for the 4 inch (10.2 cm) side was measured as - 31 cents for loading 52; for the initial measurements for loading 48 the 4 inch (10.2 cm) side was measured as –48 cents with the 2 inch (5.1 cm) side measured to be worth 5 to 10 cents less. The difference is believed to be because of the way the reactor couples to the two different sides of the tube. For loading 48, fuel temperature measurements made with a 8.5 wt% instrumented fuel element in the C-ring, indicated fuel temperatures were about 14% lower at 600 kW(th) with the reactor against the tube (same effect for 2 inch (5.1 cm) versus the 4 inch (10.2 cm) side). As part of the qualification of the tube, reactivity measurements were made that indicated that flooding the tube or cowling would not cause a measurable reactivity effect since the flooding effect is masked by the high neutron absorption of the outer boron shield.

The shield plugs are removed and inserted using the reactor bay overhead crane and PSBR standard operating procedures require the reactor to be moved away from the FNI and FFT when the shield plugs are removed to eliminate radiation streaming from the reactor core.

10.2.5 Central Thimble

The central thimble, located in the radial center of the core, provides space for the irradiation of samples at the point of maximum neutron flux. The thimble is an aluminum tube 1.5 inch (3.8 cm) O.D. and 1.33 inch (3.38 cm) I.D. It extends from the reactor bridge through the central hole of the removable hexagonal section of the top grid plate (to which it is attached), through the bottom grid plate, and is supported at its lower end by the safety plate situated below the bottom grid plate (see **Figure 4-6**).

The central thimble contains water since there are four ¼ inch (.64 cm) holes ¾ inch (1.91 cm) above the safety plate that allow entry of pool water. Thus, there is no streaming of neutrons or gammas to the reactor bridge. A cutaway section in the central thimble allows irradiated samples to be removed below the pool surface, minimizing personnel radiation exposure.

10.2.6 Pneumatic Transfer System I

The Pneumatic Transfer System I (see **Figure 10-4**) system provides a means of rapidly transferring samples between the laboratory wing of the facility and the reactor core.

The system is a closed loop design, with the major components being a core terminus, a laboratory terminus, and a blower and filter assembly, with connecting tubing between these units. Carbon dioxide (CO_2) is used as the working fluid to reduce production of radionuclides, mainly Ar-41. Four solenoid-operated valves control the fluid flow, whereby the system operates on a pressure differential drawing the "rabbit" into and out of the core by vacuum. See **Figure 4-5** for the core terminus location for core loading 52. This location could change in future core loadings.

Components of the system considered most prone to leakage and most difficult to seal are enclosed in a gas-tight steel container that also serves as a surge volume reservoir. The surge volume is essential to the operation of the closed loop system. The electrically operated switching valves and the system filter are enclosed in this containment. Over-pressure caused by any system malfunction, would be released through a "U" tube filled to a predetermined level so that if the system pressure exceeds two psig, the liquid is expelled from the "U" tube into the top of the receiver tank. With the "U" tube fluid expelled, the CO_2 working fluid would be released through an absolute filter and a charcoal filter, before being discharged to the atmosphere. Positioning and design of this tank is such that the liquid will flow back into the "U" tube, thus, resetting the over pressure release automatically after the system pressure is reduced to normal. The "U" tube is transparent and the fluid level can be observed by the reactor operator.

The CO_2working fluid is supplied from a high pressure cylinder through a standard two-stage regulator at about 20 psig to a fixed output regulator mounted on the containment box, that further reduces the pressure to a few inches of water. When a pressure sensor between these two regulators senses a pressure less than ~15 psig, a "Rabbit I Gas Pressure Low" message is indicated on DCC-X.

With a closed CO_2system, contamination with atmospheric air would lead to argon radioactivity in the system. The laboratory terminus is designed to minimize the entry of air into the system. The terminus is cylindrical and uses "O" ring seals on a sliding internal piston to minimize gas leakage.

Scheduled preventative maintenance as per PSBR auxiliary operating procedures, checks for system leaks, lubricates system valves, and checks absolute filter integrity.

The DCC-X computer of the PCMS allows operation of the pneumatic transfer system I from the keyboard when the operator controls screen is displayed. "Rabbit I Master" (on or off) controls the power (through the I/O) to the master relay that in turn permits the transfer fan to be operated by the "Rabbit I Fan" control. The master relay opens an electrically operated valve which supplies CO_2 to the system, and turns on a chart recorder that records the output of the radiation monitor. The audio is also controlled by the master relay which allows the operator to hear when a sample enters and leaves the core via microphones attached to the system tubing near the core

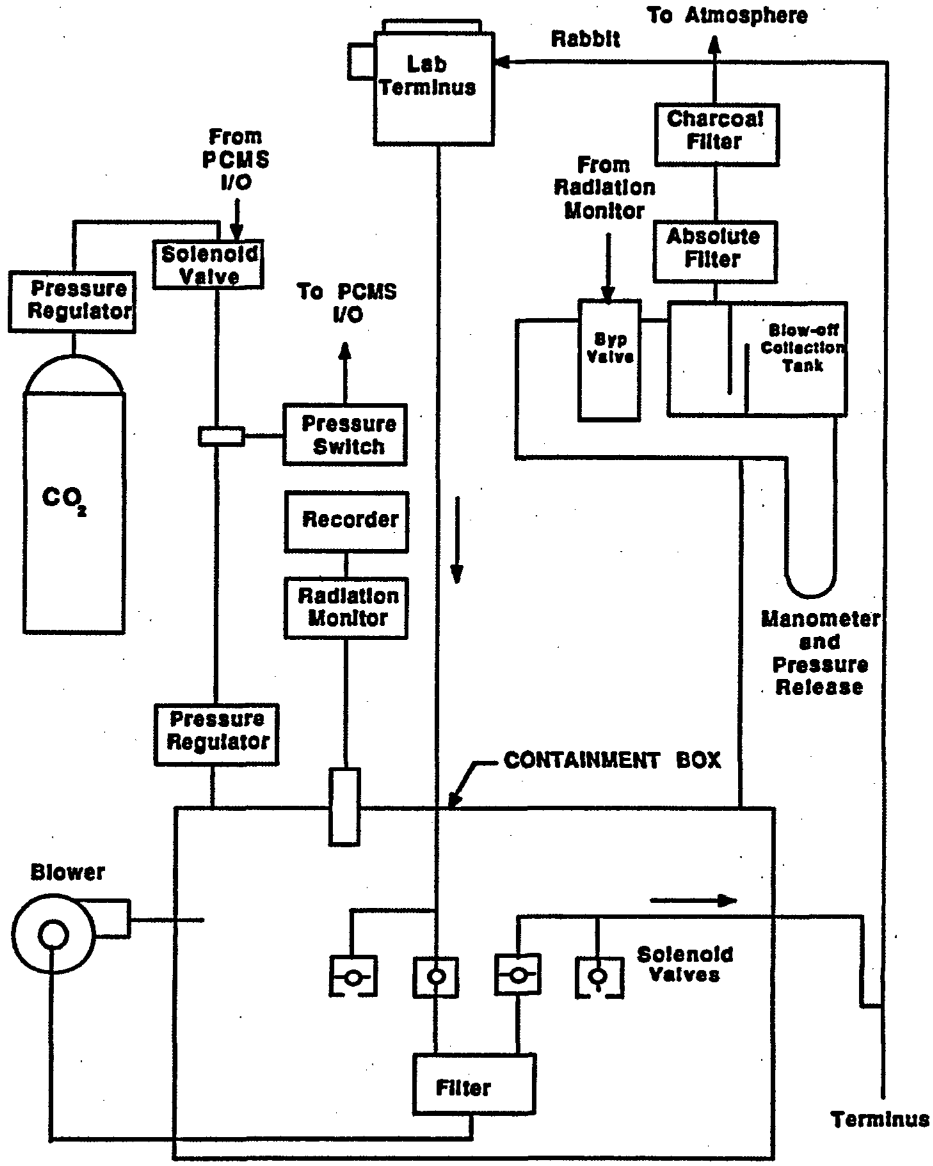

Figure 10-4 Pneumatic Transfer System I

terminus. A means of verbal communication between the experimenter and the reactor operator is available.

The experimenter in the laboratory selects either MANUAL or AUTO mode of operation. In AUTO mode, a preset timer controls the length of time the sample remains in the core.

A G-M tube in the containment box is connected to a radiation monitor located on the Rabbit I panel in the reactor bay. An alarm on this radiation monitor will send a signal to the PCMS that initiates the following action:

- prevents blower operation
- opens an electrically operated valve to bypass the pressure relief manometer so that any pressure in the system is relieved thus minimizing leakage to the building. A "Rabbit I Alarm Inhibit" control on DCC-X allows operation of the fan with a radiation alarm (but only as per PSBR standard operating procedures).

10.2.7 Instrument Bridge

The instrument bridge provides three-dimensional positioning of experiments in the vicinity of the reactor core. This bridge, which is mounted on wheels that roll on the same track on the pool wall as the reactor bridge, is constructed of steel I-beams. A tower constructed of aluminum tubing extends from this bridge down into the pool to below the level of the reactor core. The instrument bridge can be disassembled and removed from the pool by using the reactor bay overhead crane or reassembled on the other side of the reactor bridge if needed. A typical location of the instrument bridge is shown in **Figure 10-1**.

10.2.8 Hot Cells

Two hot cells are available at the PSBR for the safe handling of radioactive materials (see **Figure 1-3** for location). The cells are constructed of high density concrete 2 feet (0.61 m) thick and each cell has inside dimensions of 5 feet x7.5 feet x13 feet (1.5 m x 2.3 m x 3.9 m). Each cell has a 20.5 inches x 30.5 inches x 27 inches (52.1 cm x 77.5 cm x 68.6 cm) lead glass window and is designed to accommodate a 100 curie Co-60 source or its equivalent. Master-slave manipulators are provided in each of the cells along with a 0.5 ton (454 kg) remotely operated crane. A 3 ton (2722 kg) crane is available in the area to the rear of the hot cell concrete entry doors.

Access to the hot cells is by way of locked concrete doors or by overhead shield plugs that can be removed by a 3 ton (2722 Kg) crane in the reactor low-bay area. Permission for hot cell access is by PSBR procedures. Procedures may require an ALARA review, a radiation work permit, or other special requirements for hot cell entry.

10.2.9 Argon-41 Production Facility

This facility is designed to produce and ship to commercial customers radioactive Ar-41 gas, used as a tracer in refineries, chemical plants, etc.

Major components of the system are: an irradiation chamber; a control manifold that allows for system vacuum, irradiation chamber pressurization, transfer to shipping cylinders, and transfer of residual Ar-41 gas in the system to an evacuation chamber; and the evacuation chamber.

The entire system is first drawn to a vacuum. The irradiation chamber at the reactor core face is then filled with Ar-40 gas to a pressure of approximately 100 psig. Following irradiation, a portion of the radioactive Ar-41 gas is transferred to shielded shipping cylinders. Residual Ar-41 in the system is transferred to an evacuation chamber with some residual activity remaining in the manifold and system hoses.

10.2.10 Co-60 Irradiation Facility (for information only; not under R-2 license)

The Co-60 Irradiation Facility is a separate structure attached to the remainder of the PSBR by an entry way. A [redacted] (as of 1/1/05) gamma source composed of [redacted] is housed in a stainless steel lined pool which measures [redacted] (see **Figure 1-1** for location). A variety of vertical tubes is available for dry irradiations. Sample dose rates of approximately [redacted] are available for typical source arrangements. This pool has adequately shielded up to [redacted] of Co-60 in the past.

10.2.11 GammaCell 220 Excel Irradiator (for information only; not under R-2 license)

The GammaCell 220 Excel Irradiator was acquired in July of 2003 as a replacement for another similar irradiator. The source strength as of June 2003 was [redacted]. The maximum sample dose rate available as of July of 2004, was [redacted]. The Gamma Cell is currently located in [redacted] (see Figure 1-2).

10.3 Experiment Review

TS Section 3.7, Limitations of Experiments, provides the specifications that are intended to prevent damage to the reactor and minimize the release of radioactive materials in the event of an experiment failure. A PSBR procedure, SOP-5, Experiment Approval and Authorization, provides the administrative control to assure that the TS limits are not exceeded. SOP-5 has all the TS relevant items listed on its form for consideration by the reviewer and approver. Administrative control of who is authorized as a reviewer and approver is by way of SOP-5. Other PSBR procedures exist for experiment encapsulation and irradiation, pneumatic transfer system (R1) operation, and release of irradiated experiments; all have requirements that assist in controlling the production and release of radioactive materials. A standard operating procedure governs using a beam port experimental facility and gives guidance for the safe use of the neutron beam laboratory and describes any interlock system in operation. A standard operating procedure also exists for the use of the two fast neutron irradiators.

The SOP-5 review/approval could trigger any of the following:

- reactivity evaluations and/or measurements
- an Argon-41 production evaluation
- a 10 CFR 20 release limit review
- an ALARA review
- a Radiation Work Permit requirement
- a 50.59 review for safety and to assure the experiment falls within the confines of the MHA
- a review by the Penn State Reactor Safeguards Committee (PSRSC).

Methods for reactivity measurements are covered in SOP-5. Spreadsheets are used by the SOP-5 approver for Argon-41 production predictions and experiment activity predictions to evaluate experiments. PSBR procedures exist for ALARA reviews and use of radiation work permits to control the potential for personnel exposures. 50.59 reviews are conducted as required by SOP-5 or by an administrative "change" procedure. The authority and responsibility of the PSRSC is detailed in the PSRSC charter and operating procedure. SOP-5 also verifies that a valid University Isotopes Committee authorization exists to allow the receipt of radioisotopes by university users.

SOP-5 authorizations are supplemented by attachments and detailed experimental procedures as appropriate. If a 50.59 review is required for the experiment, an administrative procedure outlines the review and approval process that is used.

The procedures described above provide a comprehensive experiment review process to protect personnel and the general public.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

Radiation protection oversight of PSBR R-2 license activities is provided by the Radiation Protection Office (RPO) which is a part of the Department of Environmental Health and Safety (EHS). The RPO currently consists of the Manager of Radiation Protection, one associate health physicist, three health physics specialists, one assistant health physicist, plus administrative support. The manager reports administratively to the Director of EHS, who in turn reports through the Vice President for Physical Plant to the Senior Vice President for Finance. This is a different reporting chain than for the Reactor Director, who reports through the Dean of the College of Engineeering to the Vice President for Research and Dean of the Graduate School. An organization chart can be found in the TS, Chapter 14.

The Vice-President for Research and Dean of the Graduate School is the senior University official responsible to the Nuclear Regulatory Commission and the Commonwealth of Pennsylvania for the safe use of radioactive materials. The Vice-President for Research and Dean of the Graduate School appoints the University Isotopes Committee (UIC). The UIC oversees the use of radioactive materials within Federal, State, and University regulations and performs those functions required of a radiation safety committee as defined by Federal and State regulations and Penn State University's licenses to possess and use radioactive material.

Management and control responsibility for reactor facility generated radioactive waste is assigned to the reactor organization. The RPO provides independent oversight for monitoring, assessing, and limiting risks related to radiation sources. Penn State has established a policy of keeping radiation exposure as low as is reasonably achievable (ALARA) to experimenters, faculty, staff, students, and the general public.

Reactor management is responsible for having adequate procedures for the safety of operations that could result in radioactive releases to the environment or radioactive waste production from normal operations, maintenance, or accident conditions. The RPO is responsible through their own procedures for oversight of reactor activities to assure that radiation exposures and releases of radioactive materials are adequately controlled and for the ultimate handling and disposal of radioactive waste.

11.1.1 Radiation Sources

The radiation sources discussed in the following sections include the following:

- Airborne radiation sources
  - fission products from fuel failure
  - Ar-41 from neutron activation of air in solution in the reactor pool water, experiments, or experimental facilities
  - Radioactive material from an experiment failure
  - Tritium from evaporative pool water loss

- Liquid radiation sources
  - N-16 from the action of fast neutrons on oxygen in the pool water
  - Neutron activation of minerals, etc. present in the reactor pool water
  - Demineralizer resins
  - Isotope production in liquid samples
  - Tritium production in the pool water and in the D_2O tank
- Solid radiation sources
  - TRIGA reactor fuel (in service and spent)
  - Reactor core components and support structures, core radiation detectors, and startup source
  - Experimental facilities
  - Isotope production in solid samples
  - Neutron beams in the neutron beam laboratory
  - Radiation in the reactor bay from normal reactor operation

11.1.1.1 Airborne Radiation Sources

The reactor fuel and fission products are completely contained in the stainless steel clad TRIGA fuel elements, with releases to the environment only if the fuel cladding is breached. This possibility is one of the accidents discussed in Chapter 13 of this report in the analysis of the maximum hypothetical accident (MHA).

Gaseous effluent Ar-41 is released from dissolved air in the reactor pool water, air in dry irradiation tubes, air in neutron beam ports, and air leakage to and from the carbon-dioxide purged pneumatic sample transfer system.

The amount of Ar-41 released from the reactor pool is very dependent upon the operating power level and the length of time at power. The release per MWH is highest for extended high power runs and lowest for intermittent low power runs. The concentration of Ar-41 in the reactor bay and the bay exhaust was measured by the RPO during the summer of 1986. Measurements were made for conditions of low and high power runs simulating typical operating cycles. Based on these measurements, an annual release of between 558 mCi and 1693 mCi of Ar-41 is calculated for July 1, 2003 to June 30, 2004, resulting in an average concentration at ground level outside the reactor building that is 0.9 % to 2.7 % of the effluent concentration limit in Appendix B to 10 CFR 20.1001 - 20.2402. The concentration at ground level is estimated using only dilution by a 1 m/s wind into the lee of the 200 m^2 cross section of the reactor bay. These are the assumptions used in the MHA analysis.

During the report period, several irradiation tubes were used at high enough power levels and for long enough runs to produce significant amounts of Ar-41. The calculated annual production was 343 mCi. Since this production occurred in a stagnant volume of air confined by close fitting shield plugs, much of the Ar-41 decayed in place before being released to the reactor bay. The reported releases from dissolved air in the reactor pool are based on measurements made, in part, when a dry irradiation tube was in use at high power levels; some of the Ar-41 releases from the tubes are part of rather than in addition to the release figures quoted in the previous paragraph. Even if all of the 343 mCi were treated as a separate release, the percent of the Appendix B limit given in the previous paragraph would still be no more than 3.3 %.

Production and release of Ar-41 from reactor neutron beam ports was minimal. Beam port #7 has only three small (1 cm^2 area) collimation tubes exiting the port and any Ar-41 production in these small tubes is negligible. Beam port #4 has an aluminum cap installed inside the outer end of the beam tube to prevent air movement into or out of the tube as the beam port door is opened or closed. The estimated Ar-41 production in beam port #4 for all beam port operations is 68 mCi. With the aforementioned aluminum cap in place, it is assumed that this Ar-41 decayed in place. Radiation Protection Office air measurements have found no presence of Ar-41 during reactor operations with the beam port cap in place.

Any Ar-41 release from the pneumatic transfer system is insignificant since the system operates with CO-2 as the fill gas.

Argon-41 production is summarized in **Table 11-1** that shows the % of restricted and unrestricted 10CFR 20 release limits to be quite low.

Table 11-1
Argon-41 Production Values for Pool Production-Fiscal Year July 1, 2003 to June 30, 2004

	Restricted	Unrestricted
10 CFR 20 DAC limit	3 E -06 uCi/ml	-------
10 CFR 20 Effluent Concentration Limit	-------	1.0 E -8 uCi/ml
Calculated Concentration in Reactor Bay	4.2 E -08 uCi/ml	------
Calculated Concentration at Fenceline	------	2.7 E -10 uCi/ml
% of restricted area DAC	1.4%	------
% of Effluent Concentration Limit at the reactor fenceline at ground level	------	2.7 %**

* Table 11-1 values are based on 736 MWHRs of operation in the fiscal year July 1, 2003 to June 30, 2004. This year had the highest number of MWHRs in any of the ten fiscal years from July 1, 1995 through June 30, 2005.
** if it is assumed that all of the Argon-41 produced in irradiation tubes escaped, this number would increase to about 3.3%. Argon-41 produced in neutron beam ports is assumed to decay in place, and even if released would add a negligible amount to the total release.

The nearest residential area to the fenceline is a campus dormitory, located about 140 yards (128 meters) away. There are classroom and offices located about 100 yards (91 meters) from the fenceline.

Section 10.2.9 discusses the Ar-41 Production Facility. A PSBR experiment procedure approved by reactor management and RPO governs the use of the facility. The activities produced in this facility if released, fall within the bounds of exposures to the public considered in section 13.1.1 for the MHA.

Isotope production could involve irradiation of samples that could produce an aerosol, or other airborne release. Experimental approval considers sample container design and other containment (e.g. irradiation tubes with filters) to mitigate any potential release of material and to assure that no release would exceed Chapter-13 analysis or 10CFR 20 limits. If airborne releases result that are significant enough to activate the emergency exhaust system, the roughing, absolute, and charcoal filters would mitigate any release to the environment (except for inert gases).

The tritium production in the pool water is discussed in the next section, and appears in the reactor bay air as a byproduct of the pool water evaporation. The 2004-2005 fiscal year calculated release at the reactor facility fenceline for the airborne tritium, is less than 1% of the permissible concentration limit of 10CFR 20 for the public for tritium. Calculations are based on a predicted evaporation rate based on previous measurements and the measured concentration of tritium in the pool water. A dilution factor of 2E8 ml s^{-1} was used to calculate the unrestricted area concentration. This is from 200 m^2 (cross-section of the building) times 1 m s^{-1} (wind velocity). These are the assumptions used in the chapter 13 MHA analysis.

11.1.1.2 Liquid Radiation Sources

The main liquid radiation source is the N-16 produced in the reactor pool water by the action of fast neutrons on oxygen in the water. Another source of radiation would be the Argon-41 discussed in the previous section. Since the PSBR operates by natural convection cooling in a large pool, there is no significant transfer of radioactive Nitrogen-16 or Argon-41 through the recirculation or heat exchanger piping into the demineralizer room.

While the pool water is normally of a very high quality, prolonged reactor operation can produce Na-24 and other trace radioisotope activities that are collected by the reactor pool cuno filters and demineralizer. As long as pool water quality is high, radiation levels on the demineralizer are not significant. Tech Spec 3.3.5 prohibits reactor operation if the pool conductivity is greater than 5.0 micromhos/cm and Tech Spec 4.3.3 gives the surveillance requirements for monitoring the pool water conductivity. The demineralizer room is accessible to only authorized reactor staff, occupancy time is minimal, and time could be further restricted if unusual radiation levels were present.

RPO uses currently acceptable methods for resin disposal. Presently, at the end of life demineralizer resins are removed and placed into drums, allowed to decay, solidified and disposed of as solid waste; cuno filters are removed, air dried, and disposed of as solid waste. Handling of the filters and resins is done under a radiation work permit (RWP) and does not result in significant personnel exposure.

Isotope production can involve irradiation of liquid samples. Presently, a typical irradiation might involve a compound containing Bromine to produce radioactive Br-82 for use as a tracer in some industrial application (oil refinery, chemical plant, etc.). A typical activity might be several curies. Usually, an aluminum container with a swagelock fitting is used to contain the sample and the container is leak tested prior to irradiation. Maximum experimental radioactivity levels allowed from operation of the reactor are controlled by procedural review. In all cases, maximum activity produced is limited by 10CR 50.59 considerations that assure releases that could affect the public are bounded by the considerations in section 13.1.1. For university users, the activity at the time of release to the experimenter is also bounded by the UIC administration of Penn State's Broad Byproduct Material License. For industrial customers, the activity limit at the time of release is also bounded by that company's NRC or agreement state license. Experimental approval considers sample container design and other containment (i.e. irradiation tubes) to mitigate any potential release of material. If liquid radioactive material were to enter the pool water, the demineralizer would remove the material. Since the pool conductivity is very low, experience has shown that any introduction of a foreign material into the pool water is noted by an increase in conductivity. The conductivity alarm is discussed in section 5.4. A demineralizer room radiation monitor with a digital input to DCC-X (message is "Pump Room Radiation High") is also available to alert the staff to a significant increase in pool water activity levels.

The major source of tritium production is the result of neutron interaction with the deuterium in the D_2O tank, which contains about 25 gallons (95 liters) of D_2O. This tritium is contained in the aluminum D_2O tank. Activity in the tank is monitored by periodic grab samples taken from the tank, and the level in the tank is monitored during each reactor checkout (usually daily). Analysis of periodic grab samples of reactor pool water is done for tritium content to both assure no D_2O tank tritium has entered the pool water, and also to track the pool tritium activity as a result of neutron interaction with the small amount of deuterium naturally present in the pool water.

Occasionally when pool water is stored temporarily in the storage tank and then returned to the reactor pool, a few hundred to a few thousand gallons of pool water may remain in the bottom of the tank. This water is transferred to the evaporator building floor tank and then processed by the evaporator for pool make-up water. Insignificant amounts of solids are left behind as residue from the evaporative process and eventually these evaporator bottoms would be disposed of as solid radioactive waste as necessary.

PSBR auxiliary operating procedures describe sampling frequency and action limits for pool water. Pool water activity is monitored by analysis of periodic grab samples of pool water. Gross alpha and beta analysis would identify any unusual pool water activity introduced from any experiment failures, or from the solid sources (such as reactor fuel) discussed in the next section. Procedures dictate the frequency of the alpha and beta analysis and describe action if action limits are exceeded; management is notified and a gamma spectroscopy is performed to identify the source of the radioactivity. Since the quality of the pool water is very good, activation of minerals, etc. in the pool water is kept to a minimum.

A PSBR auxiliary operating procedure describes sampling frequency and action limits for secondary HX water. Periodic samples of secondary water are taken after the heat exchanger has been secured for a period of time set by procedure. The procedure dictates sampling frequency for gross alpha and beta activities, and describes action if action limits were to be exceeded;

reactor management would be notified and a gamma spectroscopy would be performed to identify the source of the radioactivity. The description of the HX system in section 5.3 indicates any leakage of pool water to the environment is unlikely. If any leakage of pool water through the secondary HX to the environment were to occur, concentration limits are well below any 10 CFR 20 limits for releases to the public.

Designated hot waste drains in the facility that could be recipients of liquid waste in case of a spill or experiment failure, are color coded and drain to either an underground 2000 gallon (7.6E3 liter) tank or a 4000 gallon (1.51E4 liter) tank in the floor of the evaporator building. Liquids in either tank can be processed by the evaporator (see section 9.7.2).

11.1.1.3 Solid Radiation Sources

The possible sources of radiation from solid radioactive sources are the irradiated TRIGA fuel elements in the current reactor core, spent irradiated reactor fuel elements, reactor core components and associated support structures, reactor core radiation detectors, reactor startup source, experimental facilities, and irradiated experiments. Aluminum of the appropriate quality is used wherever possible for construction of experiments and experimental facilities and core structure to minimize the radioactivity. Any activity with the above items that could lead to personnel exposure is dealt with by staff review and procedures including experimental approval, ALARA reviews, radiation work permits, and special procedures. As part of these reviews, design criteria specify materials used to minimize personnel exposure and eventual radioactive waste disposal.

The reactor core fuel elements produce dose rates that are several magnitudes higher than that from the other sources and are therefore the principle concern. The radiation exposure resultant from the fission products in the reactor fuel at shutdown is reduced to an insignificant level by the pool water shielding and presents no personnel hazard.

During normal reactor operations, low levels of radiation are present and measurable in the reactor bay and control room (see section 11.1.5 and **Table 11-3**). In recent years, the three most heavily used positions for reactor operations are the open pool, against the fast neutron irradiator (FNI), and against the D_2O tank (see **figure 10-1**). The Bay East and Bay West radiation monitors are mounted on the reactor bridge toward the pool walls; the reactor bridge is only accessible to personnel with reactor operator permission. Radiation levels as read on these monitors vary from ~5 mR/hr to ~35 mR/hr at 1 MW(th) full power operation depending on the reactor position. The Bay South monitor is mounted on top of the south pool wall and indicates an exposure rate likely to be received by an individual standing in that position. Exposure rates at that monitor vary from ~0.2 mR/hr to ~12 mR/hr at 1 MW(th) depending on the reactor position. Exposure rates at other accessible positions around the reactor pool where an individual would stand to look into the pool vary from essentially zero to 10 mR/hr at 1 MW(th) depending on reactor location. Radiation surveys are done for new operating positions, and additional postings, barriers, etc. are used as appropriate to limit personnel exposure. Reactor bay, control room, and personnel exposures are discussed in section 11.1.5.

[redacted]

[redacted]

Isotope production can involve irradiation of solid samples. Presently, a typical irradiation might involve a compound containing Bromine or Sodium to produce radioactive Br-82 or Na-24 for use as a tracer in some industrial application (oil refinery, chemical plant, etc.). A typical activity might be several curies. Usually, an aluminum container with a swagelock fitting is used to contain the sample and the container is leak tested prior to irradiation. Maximum experimental radioactivity levels allowed to be produced by operation of the reactor are controlled by procedural review. In all cases, maximum activity produced is limited by 10 CFR 50.59 considerations that assure releases that could affect the public are bounded by the considerations in section 13.1.1. For university users, the activity at the time of release to the experimenter is also bounded by the UIC administration of Penn State's Broad Byproduct Material License. For industrial customers, the activity limit at the time of release is also bounded by that company's NRC or agreement state license. If solid radioactive material were to enter the pool water, the demineralizer would remove the material. Since the pool conductivity is very low, experience has shown that any introduction of a foreign material into the pool water is noted by an increase in conductivity. The conductivity alarm is discussed in section 5.4. A demineralizer room radiation monitor with a digital input to DCC-X (message is "Pump Room Radiation High") is also available to alert the staff to a significant increase in pool water activity levels.

Significant radiation levels in the neutron beam lab are confined to within the cave area, where "high radiation areas" could exist for some experimental setups. Appropriate access control to the cave area is in place to meet requirements of 10CFR 20.1601.

The LOCA is discussed in section 13.1.3 of this document, [redacted] The LOCA as analyzed in 13.1.3 results in no fuel damage and release of radioactive fission products from the fuel.

11.1.2 Radiation Protection Program

The RPO maintains a university-wide Radiation Protection Program (RPP) and assures the program is commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10CFR 20.1101. This program covers activities under the R-2 license as well as other licenses such as SNM and Byproduct Material licenses. The RPP, for items not covered by the R-2 license, is documented by the relevant NRC and Pennsylvania license applications and conditions and the UIC Rules and Procedures. For tasks such as the conduct of meter calibrations, not specifically addressed in these documents, the RPO has established internal procedures (the PSBR also has its own procedures for meter calibrations, etc). Other internal RPO procedures provide checklists and instructions for actions performed by the RPO staff in support of Penn State's RPP (such as radiation and contamination surveys). The UIC is responsible for periodic (at least annually) reviews of the RPP content and implementation for items not included in the R-2 license.

The R-2 license portion of the RPP is detailed by a PSBR procedure. Reactor staff does an annual review of this procedure that describes what radioactive materials are authorized under the R-2 license, and lists all procedures that involve use of any radioactive material at the reactor facility. The annual review is to see that all procedures are current and work together to ensure compliance with the provisions of 10CFR 20.1101. Changes to PSBR procedures that are a part of the RPP are prepared by a reactor staff member, reviewed by the Manager of Radiation Protection (therefore maintaining involvement of the RPO), and approved by the facility Director. Periodic reviews required for such PSBR procedures follow the same chain of command.

The Manager of Radiation Protection acts as the Radiation Safety Officer (RSO) for the university and is responsible for:

- managing the RPP
- identifying radiation safety problems
- initiating, recommending, and/or providing corrective actions
- verifying implementation of corrective actions
- ensuring compliance with all applicable regulations

The RSO has been delegated this responsibility and authority by the Vice-President for Research and Dean of the Graduate School. The RSO has the authority to immediately stop any operation involving the use of radioactive material in which personnel health and safety may be compromised or may result in non-compliance with regulations. The RSO staff has the authority to enter any laboratory to carry out inspections and to determine compliance with licensees, regulations, or a user's authorization to use radioactive materials. The RSO reports to the Vice-President through the UIC.

The Penn State Reactor Safeguards Committee (PSRSC) Charter and Operating Procedure, states the committee shall exist to provide an independent review and audit (annual) function of the safety aspects of the PSBR facility operations. See sections 12.2.2, 12.2.3, and 12.2.4 of this document. The PSRSC charter requires one committee member to have health physics experience and one committee member either be a member of the UIC or designated by the

PSRSC chairman to be the liaison between the PSRSC and the UIC. The UIC liaison and the person having health physics experience can be the same person, and is usually the RSO.

Those who work with radioactive material or receive personal monitoring for the PSBR are required to first complete a training program conducted by the RPO. In addition to basic health physics principles, the instruction includes the explanation of regulations in 10CFR 19, 10CFR 20, 10CFR 21, and the rules and procedures for use of radioactive materials at Penn State. PSBR procedures establish levels of radiation protection orientation for facility visitors, students, and workers. The level of orientation is commensurate with potential radiation exposure. PSBR procedures require an annual review to assure all individuals receiving personnel monitoring are familiar with the orientation requirements. The facility reactor operator and senior operator requalification program requires training in the area of radiation control and safety during the two-year training cycle. As appropriate, this training is done in conjunction with the RPO.

PSBR procedures provide a means for reporting, documenting, and investigating events at the reactor including those involving a lack of proper radiation control. The ALARA program discussed in section 11.1.3 also provides a means to investigate radiation control events. PSBR procedures provide a means for "lessons learned" to be applied to facility operating procedures to improve future performance.

A PSBR RWP procedure establishes procedural controls for persons working with radioactive material. In general, RWP's are initiated as a requirement of another PSBR procedure and/or when a new experiment or operation is being conducted for the first time, and may involve an ALARA review. The RWP requires the involvement of the laboratory supervisor, the RPO staff, and PSBR management. If the experiment or operation becomes repetitive, a dedicated procedure may be developed in lieu of the RWP to incorporate the "lessons learned" from the RWP and/or ALARA reviews.

11.1.3 ALARA Program

The Pennsylvania State University's policy towards exposure to radiation is described by the ALARA program. It is the policy of Penn State University, as established by the UIC, that the release of radioactive material and the exposure of people to ionizing radiation be kept As Low As Reasonably Achievable (ALARA). The University's ALARA policy is based on the following three principles:

- Exposures of personnel to radiation or the releases of radioactive material to the environment may not exceed the limits in the federal and state regulations.
- Unplanned exposure of personnel or uncontrolled releases to the environment that exceed 10% of permissible limits will be investigated by the RPO of EHS to determine whether the exposures or releases were ALARA and whether action is required to limit future exposures of releases. Planned operations with estimated exposures or releases that exceed 10% of the permissible limits will be subject to an ALARA review by RPO prior to beginning the operation.
- Exposures and releases that do not exceed 10% of the permissible limits are low enough that no further consideration of ALARA is necessary. This policy does not limit each individual exposure to less than 10% of the NRC regulatory limit, but rather aims to

maintain all exposures ALARA. The university's ALARA policy is reiterated in a PSBR procedure, that describes how ALARA reviews at the reactor are initiated, how they are performed, and references numerous other PSBR procedures pertaining to issues of radiation protection.

11.1.4 Radiation Monitoring and Surveying

The reactor staff and the RPO have in place a RPP, the elements of which assure proper monitoring of solid, liquid, and air exposures to personnel as required by 10 CFR 20. The RPP is described in sections 11.1 and 11.1.2 of this document.

Adequate equipment exists to meet requirements of the TS and 10 CFR 20. Section 7.7 of this document includes the air and area radiation monitors required by the TS. In addition, a wide variety of portable ion chambers, "friskers", and personal monitors are available for personnel and laboratory radiation and contamination surveys by radiation workers. Other PSBR and RPO equipment includes an alpha-beta gas flow counter, gamma spectroscopy equipment, NaI equipment, and a liquid scintillation counter.

A calibration facility (traceable to NIST) is operated by the RPO and is used by reactor staff for calibrations of appropriate instruments using PSBR procedures. The instrument calibration program is described in facility procedures for area, air, and portable survey instruments. Calibration methods and frequency (usually annually) are described there. For TS required monitors, operability checks are a part of the reactor checkout (usually daily); another procedure requires functional checks of the instruments operation on a periodic basis (usually monthly) and checks instrument function in conjunction with other facility systems (reactor scram on high radiation) and checks proper operation of the Engineered Safety Features; Facility Exhaust System (FES) and Emergency Exhaust System (EES) discussed in section 7.5. Another procedure describes the proper checkout and use of survey instruments during radioactive sample release.

Personnel monitoring is discussed in section 11.1.5 and environmental monitoring is discussed in section 11.1.7.

11.1.5 Radiation Exposure Control and Dosimetry

The confinement with its FES and EES has been previously discussed in sections 6.2.1 and 7.5 of this document. During normal operation, the FES minimizes worker and visitor exposure to Ar-41 by providing constant air exchange to the reactor bay. The EES also minimizes public exposures because of its release of the reactor bay effluent at the reactor bay roof, where dilution can occur before any exposure to the public at ground level. Whenever the building evacuation alarm is initiated, the FES closes and the EES is activated. The effectiveness of this system during a MHA is discussed in section 13.1.1 of this document.

Designated hot waste drains in the facility that could be recipients of liquid waste in case of a spill or experiment failure go to hold tanks as described in section 11.1.1.2.

The primary control of high radiation areas is by the use of locked entryways for areas such as the cave in the neutron beam laboratory. [redacted] Direct surveillance and/or warning devices are sometimes used for short-term experiments where a high radiation area is present for short durations (e.g. reactor experiments requiring the N-16 pump to be off during reactor operations).

PSBR staff do not need respiratory equipment and have no NRC approved respiratory program. In the event respiratory protective equipment is needed under extreme emergency situations, the facility utilizes the resources of trained emergency responders who follow an approved respiratory protection program. [redacted]

The RPO administers the dosimetry program and maintains the necessary records. Personal monitoring is provided to individuals as required by 10CFR 20.1501 and 10CFR 20.1502. Personal dosimeters are also mounted in select locations within the facility to monitor those areas. Environmental dosimeters are used for external environmental monitoring. A commercial vendor is used for the personal monitoring and environmental monitoring program.

Area dosimetry monitors are located in the rear of the reactor control room, the reactor bay (just above the operator window), and in the neutron beam laboratory. The operating position for the reactor operator is below the reactor bay dosimeter. The control room dosimeter is located in the rear of the control room close to a public hallway. The control room dosimeter and reactor bay dosimeter indicate quarterly averages of 6.7 mRem and 33.3 mRem respectively, for the years 2000 to 2003 (*the year 2004 is an historical aberration from the average and reflects an experiment that involved operating the reactor without the N-16 pump on); see **Table 11-3.** Reactor operator duties have been typically shared by 8 to 15 operators per year, so the dose has been minimal to any one operator. The dosimeter in the rear of the control room indicates that the yearly dose has been very low to the most frequented hallway. The years 2001, 2003, and 2004 represent the highest MWHRs of operation of any years from 1986 to 2004. The recorded exposures for the control room and reactor bay reflect primarily contributions from reactor operation, N-16, and Ar-41. The neutron beam laboratory dosimeter (located near the east wall) reflects the hours of use and type of experiments conducted for that facility, which vary from year to year.

Table 11-3
Operations Areas Monitoring - Quarterly Exposures

YEAR	MWHRS of REACTOR OPERATION	CONTROL ROOM (mRem/qtr)	REACTOR BAY (mRem/qtr)	NEUTRON BEAM LAB (mRem/qtr)
2000	435	6.5	23.3	9.8
2001	656	7.0	40.0	28.8
2002	469	6.0	34.0	13.8
2003	674	7.3	36.0	24.3
2004	645	19.5*	91.3*	17.8

Occupational exposures of reactor operations personnel have historically been very low, seldom exceeding 0.5 Rem TEDE in a year and usually below 50 mRem/yr. Personnel exposures above 50 mRem/yr have not been from normal reactor operations but have usually involved neutron beam experiments or isotope production for commercial customers, in which received doses are in line with expectations. In the last few years, exposures to reactor staff have increased due to a significant increase in the amount of research and service work. The RPO tracks exposures and investigates if any unexpected exposures are indicated. The ALARA program requires an investigation if exposures to workers or members of the public exceed 10% of the federal limit. Exposure records are retained by RPO. Personnel dosimetry is issued quarterly for those for whom monitoring is required by 10CFR 20.1502.

While visitors and tour groups viewing the reactor by looking over the pool wall could be in a radiation field approaching 10 mR/hr, the viewing time is so minimal that doses are usually not measurable by electronic dosimeters set to read in the μR range.

RPO presents information to radioisotope users to assure they are aware of the pregnancy requirements in 10CFR 10.1208.

Adequate survey equipment and dosimetry is available during the full range of normal facility operations, potential accident conditions, and rescue and recovery. The RPO is prepared to do bioassays if needed.

11.1.6 Contamination Control

Contamination control is an element of the RPP and is accomplished in various ways. First, laboratory supervisors and radiation workers are responsible for conducting surveys in their own areas to minimize worker exposure and minimize the likelihood of contamination leaving the work area. Adequate radiation monitoring equipment is available in all areas of the reactor facility where radioactive materials are present. Secondly, the RPO conducts unannounced radiation and contamination surveys of radioactive material work areas and public areas. At the PSBR, the RPO contamination surveys are supplemented by more frequent surveys by the reactor staff, focusing on public areas.

All persons using radioactive materials at the university or persons who have personal monitoring to the RSEC, must successfully complete RPO training. The UIC may require additional training for those working under specific authorizations that present unusual radiation safety or regulatory problems. The training programs deals with exposure control and contamination control issues.

Adequate posting materials are available to post contaminated areas, equipment is available for contamination surveys and assessment of the magnitude of the problem, and decontamination control and cleanup materials are available in the reactor building and at the RPO. Since the reactor is in a dedicated building, area control as necessary is easily accomplished, and can be aided by a public address system.

Experiments or activities likely to generate significant contamination would be identified by the PSBR experimental review and approval or by review of the UIC authorization by the RPO. Additional controls would be specified as needed.

11.1.7 Environmental Monitoring

Environmental monitors are used and evaluated by a commercial vendor on a quarterly basis. Four areas around the reactor building are monitored, a nearby child care center (about 170 yards or 155 m away), a control location in State College (about 1 mile or 1.6 km) away, and a control location in Pleasant Gap (about 7 miles or 11.3 km) away from the PSBR. The years 2001, 2003, and 2004 represent the highest MWHRs of reactor operation of any years from 1986 to 2004. See **Table 11-4** for quarterly averages for the locations mentioned.

Table 11-4
Environmental Monitoring - Quarterly Averages (mRem/Qtr)

Year	MWHRS	North	South	East	West	Pleasant Gap	State College	Child Care
2000	435	5.3	3.2	5.8	3.1	2.2	-2.6	no data
2001	656	4	3.4	4.6	3.0	3.8	2.0	3.8
2002	469	10.0	8.9	10.2	8.4	8.5	6.5	10.6
2003	674	8.9	8.4	8.8	8.5	7.0	4.5	10.6
2004	645	8.6	9.8	8.5	8.4	6.9	3.9	8.5

Effluent concentrations are calculated annually for Ar-41 and tritium airborne releases (see section 11.1.1.1).

11.2 Radioactive Waste Management

University laboratory rules and policies for radioactive waste management exist to assure conformance with applicable regulations, in a manner that protects the health and safety of radiation workers, students, the public, and the environment.

All persons using radioactive materials at the university or who have personal monitoring for the RSEC, must successfully complete RPO training. The UIC may require additional training for those working under specific authorizations that present unusual radiation safety or regulatory problems. The training programs deal with the handling of radioactive waste.

11.2.1 Radioactive Waste Management Program

Radioactive waste management oversight is provided by the RPO that is a part of EHS. Daily waste disposal by the reactor staff is in the form of gloves, absorbent paper, laboratory wash, etc. which is placed in appropriate solid or liquid radioactive waste containers furnished by the RPO. From here the materials enter the normal waste stream from other campus laboratories. The RPO disposal of the waste is by storage for decay, release to the sanitary sewer (special cases only with written permission from the RPO and as per 10CFR 20.2003), and shipment to commercial burial sites.

Items such as demineralizer resins, experimental apparatus, etc. are handled on a case-by-case basis by the RPO. For highly radioactive items, radiation work permits and ALARA reviews may be required. PSBR procedures indicate that experiments should be designed to assure that production of unneeded activation products are minimized, to assure good ALARA practice and minimize radioactive waste disposal costs.

11.2.2 Radioactive Waste Control

Disposal of solid radioactive waste is controlled by the RPO who ensure that waste is properly stored at the EHS radioactive waste storage room in the Academic Projects Building, which is adjacent to the RSEC. RPO staff package the waste in accordance with DOT requirements for shipment to a LLRW broker for processing prior to final disposal. Reactor components or experimental fixtures that produce high radiation levels may be temporarily stored for decay in the reactor pool. The bulk of the volume of radioactive waste is contaminated paper, plastic, and small pieces of metal. Most of the activity comes from experimental fixtures that are no longer required.

Disposal of liquid waste to the Municipal Sanitary Sewer is controlled by the RPO, who ensure that releases comply with 10CFR 20.2003 and Penn State's ALARA policy. Small quantities of liquid radioactive waste are physically transferred to RPO for analysis and proper disposal. Large quantities are analyzed and then as appropriate released to the sanitary sewer following RPO approval.

Designated hot waste drains in the facility that could be recipients of liquid waste in case of a spill or experiment failure, are color coded and drain to either an underground 2000 gallon (7.6E3 liter) tank or a 4000 gallon (1.5E4 liter) tank in the floor of the evaporator building. Liquids in either tank can be processed by the evaporator (see section 9.7.2).

11.2.3 Release of Radioactive Waste

Effluent concentrations are calculated annually for Ar-41 and tritium airborne releases (see section 11.1.1.1). Also see sections 11.2, 11.2.1, and 11.2.2 above.

11.3 Bibliography

Standard ANSI/ANS-15.11-1993, "Radiation Protection at Research Reactor Facilities", American Nuclear Society, July 23, 1993 ANSI Approval

10 CFR 20, Standards for Protection Against Radiation

"Rules and Procedures for the use of Radioactive Material at the Pennsylvania State University" by The University Isotopes Committee, Spring 2001

12. CONDUCT OF OPERATIONS

12.1 Organization

An adequate organization exists to have management:

- that are knowledgeable about the TS to operate a safe facility
- are responsible for complying with regulations and license conditions
- that implement a meaningful radiation protection program that will protect the health and safety of the public

12.1.1 Structure

The Director of the Penn State Breazeale Reactor (PSBR) reports to the Dean of the College of Engineering, who for matters related to reactor operations and the R-2 license, reports to the Senior Vice President for Research and Dean of the Graduate School, who is the responsible individual for the R-2 license. The directorship of the PSBR is a subset of broader responsibilities assigned to the Director of the Radiation Science and Engineering Center. An organization chart is presented in section 6 of the PSBR TS that is chapter 14 of this document. Section 6.6.2.b of the TS, requires a written report to the NRC within 30 days, for changes in the facility organization involving level 1 and level 2 personnel.

The Associate Director for Operations reports to the reactor Director. The licensed SROs and ROs report to the Associate Director for Operations for matters pertaining to reactor operations.

The Radiation Protection Office (RPO) has a reporting chain independent of the PSBR. The Manager of Radiation Protection reports to the Director of Environmental Health and Safety (EHS), who reports to the Vice President for Physical Plant, who reports to the Senior Vice President for Finance and Business/Treasurer.

The Penn State Reactor Safeguards Committee is appointed by the Dean of the College of Engineering , acting for the Senior Vice President for Research and Dean of the Graduate School. The committee reports to the Dean of the College of Engineering.

Numerous PSBR procedures assure that all operations involving the PSBR are conducted in compliance with existing local, state, and federal regulations. The reactor is operated within the limits established by the operating license and the TS. An ALARA program (as low as reasonably achievable) is in effect to minimize radiation exposures to the public, the staff, and the environment.

12.1.2 Responsibility

Responsibility for the safe operation of the PSBR is within the chain of command shown in the organization chart (TS, section 6, Chapter 14). Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, are responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the R-2 operating license and the TS.

12.1.3 Staffing

Section 6.1.3 of the PSBR TS (chapter 14) gives the staffing requirements when the reactor is not secured and defines events needing the presence of a Senior Reactor Operator. These TS requirements fulfill the requirements of 10CFR 50.54 (I, j, k, l and m(1)). PSBR procedures have personnel requirements for staffing requirements that meet or exceed the TS requirements.

12.1.4 Selection and Training of Personnel

Section 6.1.4 of the PSBR TS (chapter 14) states : "The selection, training, and requalification of operations personnel shall meet or exceed the requirements of all applicable regulations and the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6. Section 6.1.1 of the TS describes the minimum education qualifications for the Director, again based on ANSI 15.4.

A reactor operator training program outlines the initial training. Site specific training is similar for all trainees, with some variation in other aspects of the training (e.g. reactor physics) depending on previous experience.

The competence of the PSBR operators and senior operators is maintained by requalification as defined in the PSBR Operator and Senior Operator Requalification Program. This program meets the requirements of ANSI 15.4 and10 CFR 55 as it applies to research reactors.

Competency of licensed individuals to respond appropriately to the Emergency Plan is maintained by an annual review of the Emergency Plan for the reactor staff by the Emergency Director, and by participation of licensed individuals in drills required by the Emergency Plan.

Training on 10 CFR Part 55 is done during the two-year cycle of the operator requalification program. [redacted], individuals are made aware of the posted NRC Form 3 as required by 10 CFR Part 19. Part 19 review is also done annually for the reactor staff.

12.1.5 Radiation Safety

The RPO staff provides advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection. The organizational separation of the radiation protection staff from the reactor staff is discussed in section 12.1.1. Radiation safety aspects of reactor operation fall under the purview of the university Radiation Protection Program (RPP), and the radiation safety staff is granted the authority to interdict or terminate safety-related activities. The RPP is discussed in detail in chapter 11 of this document.

12.2 Review and Audit Activities

The Penn State Reactor Safeguards Committee, an independent group with technically experienced members from both within and outside the University, advises the Director on all matters or policy pertaining to safety. Members are appointed by the Dean of the College of Engineering, acting for the Senior Vice President for Research and Dean of the Graduate School.

12.2.1 Composition and Qualifications

The Penn State Reactor Safeguards Committee Charter and Operating Procedure, outlines the composition and qualifications of the committee members. This charter meets or exceeds the requirements of the TS section 6.2.1 (chapter 14).

12.2.2 Charter and Rules

The Penn State Reactor Safeguards Committee Charter and Operating Procedure, outlines the committee rules. This charter meets or exceeds the requirements of TS section 6.2.2 (chapter 14).

12.2.3 Review Function

The Penn State Reactor Safeguards Committee Charter and Operating Procedure, outlines the committee's review function. This charter meets or exceeds the requirements of the TS section 6.2.3 (chapter 14).

12.2.4 Audit Function

The Penn State Reactor Safeguards Committee Charter and Operating Procedure, outlines the committee's audit function. This charter meets or exceeds the requirements of the TS section 6.2.4 (chapter 14).

12.3 Procedures

The practical application of the philosophy of safe reactor operation is augmented by detailed written procedures. These procedures govern the activity of all staff personnel, experimenters, and visitors while in the PSBR. These procedures fall into five general categories, Emergency Procedures (fire, civil disorder, loss of pool water, etc.); Standard Operating Procedures (operation of the reactor, instrumentation checkout, fuel handling, etc.); Special Procedures (security systems, cooling systems, etc.); Administrative Policies (personnel requirements for operation, facility keys, reactor safeguards committee procedures, facility access, etc.). and Auxiliary Operating Procedures (pool water makeup and demineralizer operation, water collection and analysis, etc.).

The number and scope of PSBR procedures significantly exceed the requirements of the PSBR TS, section 6.3 (see chapter 14). In addition, experiment specific procedures are developed as needed .

All PSBR procedures are approved by the Director. Permanent procedures are reviewed periodically and the review is documented.

12.4 Required Actions

The PSBR TRIGA reactor has only one safety limit, which is on fuel temperature (TS 2.1 in chapter 14). The TS 6.5.1, Action To Be Taken in the Event the Safety Limit is Exceeded, requires reactor shutdown, facility management notification, PSBR Safeguards Committee chairman notification, and NRC notification as per the TS 6.6.2, Special Reports.

The TS section 1.1.34, Reportable Occurrence, lists reportable events. The TS section 6.5.2, Action to be Taken in the Event of a Reportable Occurrence, discusses the actions to be taken: reactor shutdown, Director notification, level 1 notification, PSBR Safeguards Committee chairman notification, and NRC notification as per section 6.6.2, Special Reports.

A PSBR administrative procedure also provides guidance on the above events and provides further guidance on other events that may be reportable to the NRC, such as any violation of any limits in 10 CFR 20, or discovery of an operation or system failure reportable under 10 CFR 21.

12.5 Reports

The PSBR TS section 6.6.1 (see chapter 14), Operating Reports, lists the required contents of the annual operating report from the PSBR to the NRC.

The TS section 6.6.2.a discusses how to file special reports for violation of safety limits, release of radioactivity from the site above allowed limits, and a reportable occurrence. The TS section 6.6.2.b requires written special reports for permanent changes in facility management at levels 1 and 2, and for significant changes in the transient or accident analysis as described in the SAR.

12.6 Records

The PSBR TS section 6.7 (chapter 14), Records, lists the records required and their retention period for three categories:

- Section 6.7.1, Records to be Retained for at Least Five Years
- Section 6.7.2, Records to be Retained for at Least One Training Cycle
- Section 6.7.3, Records to be Retained for the Life of the Reactor Facility

Records maintained and retention periods for those records meet or exceed the above TS requirements.

12.7 Emergency Planning

See Section 12.13, references 3, 4, and 5.

12.8 Security Planning

See Section 12.13, references 6 and 7.

12.9 Quality Assurance

Since the PSBR is an existing facility, section 50.34(a)(7) of 10 CFR 20 for applicants for construction permits to have a quality assurance program is not applicable. The section 50.34(b)(6)(ii) requirement for a description in the SAR of managerial and administrative controls to be used to ensure safe operation, is satisfied in sections 12.1, 12.1.1, and 12.1.2 of this SAR.

As applicable for existing facilities, the guidance provided by ANSI-15.8-1995, Quality Assurance Program Requirements for Research Reactors, is incorporated into many PSBR procedures. While existing facilities are not required to prepare quality assurance documentation for the as-built facility, the available as-built records for the PSBR are stored in a fire-proof cabinet. Replacements, modifications, and changes to safety-related items are made under a formal PSBR administrative procedure, with steps for management approval (and Reactor Safeguards Committee review and NRC approval if appropriate) and for design basis review of structures, systems, or components. This formal safety evaluation review satisfies 50.59 requirements to assure the SAR or TS are not compromised, and also reviews the change in regards to the Physical Security Plan and Emergency Plan. A change procedure includes a detailed work package with detailed drawings and procedures as necessary, functional testing requirements, procedure development or update, other log and document updates, and licensed operating staff training.

The guidance of ANSI-15.8-1995 for existing facilities is addressed for the following topics as follows:

12.9.1 Organization

The organization structure is defined by the TS (chapter 14).

12.9.2 Quality Assurance Program

As discussed in the following sections, PSBR operating procedures incorporate appropriate elements of the ANSI-15.8-1995 guidance.

12.9.3 Performance Monitoring

PSBR procedures provide for the investigation and documenting of events.

12.9.4 Operator Experience

See Section 12.13, references 1 and 2.

12.9.5 Operating Conditions

Pre-operations checklists and procedures are used to determine readiness to operate (e.g. reactor checkout). Numerous Check and Calibrations Procedures (CCPs) are used to periodically monitor for abnormal conditions or adverse trends. Operating conditions are documented in an operations log, for which guidelines for log book entries are detailed by procedure. Numerous procedures call for operators to involve management as appropriate.

12.9.6 Operational Authority

Various procedures, especially the reactor operating procedure, discusses conduct of operations for licensed personnel, including a protocol for shift turnover. Formal log book sign-ins and briefings are part of the turnover.

12.9.7 Control Areas

The reactor operating procedure defines operator's responsibilities and control room protocol. Personnel requirements for reactor operation are defined by procedure.

12.9.8 Ancilliary Duties

The reactor operating procedure discusses the conduct of operations for licensed personnel, requiring independence of the RO and SRO from the experimenter to limit operating staff distractions.

12.9.9 Emergency Communications

[redacted]

12.9.10 Configuration Control

A PSBR administrative procedure applies to any proposed electrical, mechanical, or software change, or a change in a method of operation of facility equipment that could affect a system or structure associated with the reactor.

The numerous facility water handling drawings show valves with their normally open or closed positions.

Detailed PSBR procedures exist for checks, calibrations, maintenance, or repair on reactor instrumentation and radiation monitoring equipment and to assure their proper return to service.

12.9.11 Lockouts and Tagouts

PSBR procedures provide requirements for Do Not Operate Tag-Outs, Administrative Tag-Outs, Equipment Tag-Outs, and Energy Isolating Device Tag-Outs. This tag-out system is used when there is potential for equipment damage or personnel injury during equipment operation, maintenance, inspection, modification activities, or from inadvertent activation of equipment. Many PSBR procedures require the use of tagouts.

12.9.12 Test and Inspection

PSBR procedures call for calibrations, tests, or inspections following testing, maintenance, or modifications.

12.9.13 Operating Procedures

The TS section 6.3 (chapter 14), Operating Procedures, provides a list of required facility procedures. Additional PSBR procedures supplement the required procedures.
PSBR procedures are approved by reactor management and normally the writer or reviewer is not the same person as the approver. Procedures are controlled and reviewed on a periodic basis as per procedure, to assure they are technically correct and that the wording is clear and concise. The facility policy on the use of procedures is stated in the Introduction to the operating procedures. A controlled copy of all operations procedures is available in the control room.

12.9.14 Operator Aid Postings

Postings are a part of an approved procedure. The periodic procedure review process discussed in 12.9.13 above assures that postings are current.

12.9.15 Equipment Labeling

Valves, electrical wiring, electrical panels, and electric breakers are labeled as appropriate.

12.9.16 Quality Records

A records system exists for the categorizing and storing the records produced by the procedures discussed in the above sections.

12.10 Operator Training and Requalification

The NRC approved Operator and Senior Operator Requalification Program, meets the part 50.54(I-1) of 10 CFR requirement to have an operator requalification program. The purpose of the program is to assure all operators and senior operators maintain competence. The program meets the requirement of 55.59(c)(1) for the program period not to exceed 24 months. The plan includes information on: schedule; exemptions for persons administering exams; lecture topics and drills; biennial written exams; annual oral exams on emergency procedures; annual operating exams; pass-fail exam criteria; procedures for an operator to qualify to return to duty following an exam failure; on-the-job-training; inactive operators return to duty; evaluation and training of operators; requalification documentation and records; requalification document review and audit; and medical certification.

The PSBR Technical Specification 6.2.4, Audit (chapter 14), specifies that the annual audit of facility operations for the Penn State Reactor Safeguards Committee include an audit of the requalification program for the operating staff (at least every other calendar year not to exceed 24 months).

12.11 Startup Plan

Not applicable to the PSBR. The reactor facility has existed since 1955 and the current TRIGA reactor has been in operation since 1965. No significant modifications are being made at the time of the re-license submittal.

12.12 Environmental Reports

The PSBR falls under a categorical exclusion as described in 10CFR 51.22, and therefore no environmental report is necessary.

On January 23, 1974, the AEC staff concluded in a memorandum addressed to D. Skovholt and signed by D. R. Miller, "that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 MW(th) or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities."

Since this Safety Analysis Report is written in support of extending the license expiration date for an additional 20 years, no changes in land and water use are contemplated. Emissions of radioactive materials or other effluents will not change as a result of extending the license term.

12.13 References

1) Standard ANSI/ANS-15.4-1988, Selection and Training of Personnel for Research Reactors, American Nuclear Society, June 9, 1988 ANSI Approval, reaffirmed July 12, 1999

2) PSBR Administrative Procedure, AP-3, Operator and Senior Reactor Operator Requalification, June 26, 1997

3) Standard ANSI/ANS-15.16-1982, Emergency Planning for Research Reactors, American Nuclear Society, October 11, 1982 ANSI Approval, reaffirmed May 3, 2000

4) NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors", USNRC, October 1983

5) Penn State Breazeale Reactor (PSBR) Emergency Prepardness Plan, Revision 4, September 21, 2000

6) Regulatory Guide 5.59, Revision 1, "Standard Format and Content for A Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance, US Nuclear Regulatory Commission, February 1983

7) The Physical Security Plan for The Pennsylvania State University Breazeale Reactor, Revised in its entirety and reprinted on June 11, 1990, revised page 1-1 on March 10, 1992

8) ANSI-15.8-1995, "Quality Assurance Program Requirements for Research Reactors", American Nuclear Society, September 12, 1995 ANSI Approval

13. ACCIDENT ANALYSES

The Penn State Breazeale TRIGA Reactor (PSBR) was initially loaded with 8.5 wt% U-$ZrH_{1.65}$ TRIGA fuel[a] in December 1965.[26] The reactor core was operated with good performance with this fuel from 1965 through the early 1970's. It was then decided to strive to reduce fuel costs for the supplier, the Department of Energy (DOE), by achieving higher fuel burnup through an increased uranium concentration in the fuel. Fuel management studies at the PSBR performed in 1972[1,2] showed, by analysis and experiment, that replacing some of the 8.5 wt% fuel with 12 wt% U-$ZrH_{1.65}$ TRIGA fuel would achieve a better fuel utilization and a substantially lower fuel cost. The basic "in-out" fuel management method was selected as it would provide the necessary excess core reactivity to achieve a longer fuel burnup. This "in-out" method would start with 12 wt% U fuel being placed in some fuel locations in the center most ring, the B ring. The remainder of the core would be 8.5 wt% fuel. As the fuel was consumed, the partially burned 12 wt% fuel would be moved further out sequentially to the C and D rings while removing the 8.5 wt% fuel in those locations. New 12 wt% fuel would be fed into the B ring to replace fuel moved to outer fuel rings. In a given fuel location unburned 12 wt% fuel will produce a greater power density than the 8.5 wt% fuel by approximately 35%. Thus, increased power density results in higher fuel temperatures which were studied[3-7] analytically and experimentally to avoid exceeding safe operating conditions. The calculations agreed closely with the experimental data. Since 1972 the PSBR has been refueled with 12 wt% fuel.

On July 13, 1972 six 12 wt% fuel elements were placed in the B-ring replacing 8.5 wt% fuel which was moved to outer rings. This increased the core k_{eff}[5] to the level required for a larger fuel burnup and also increased the maximum measured fuel temperature to slightly over 400 °C, well below the safety limit of 1150 °C. The maximum radial power peaking factor was about 2.0 and the core reactivity increased by 1.624% Δk/[redacted] ($2.32). [redacted]

In 1985 (Core Loading 38) a higher steady state maximum fuel temperature was observed (in an unburned instrumented fuel assembly of 12 wt%U, I-15) compared to previous similar instrumented (ie. I-13) fuel elements located in the same core position. The reason for this temperature increase is due to an increased fuel to fuel cladding gap. This was verified by comparing the peak fuel temperature of two similar instrumented fuel elements in the same core

[a] From this point on in Chapter 13, fuel refers to U-$ZrH_{1.65}$ TRIGA fuel with 20% nominal enrichment and zirconium to hydrogen atom ratio of 1 to 1.65 nominal.

position subjected to the same sized pulse. The peak fuel temperature during a pulse of the two instrumented fuel elements were nearly the same whereas the steady state maximum fuel temperature was higher in the newer instrumented instrumented fuel element (I-15). Since the reactor pulse can be considered as an adiabatic process (the reactor pulse is << than the thermal time constant of the fuel), there is no instant heat transfer and the conductance of the gap between fuel and clad is immaterial. However under steady state conditions, the maximum fuel element temperature is a function of the gap conductance. Therefore with a larger gap, the fuel temperature will increase. In the case of I-15, there existed a larger fuel to fuel cladding gap prior to use than with I-13 after use.

It also has been found that the measured steady state maximum fuel temperature increases when the fuel element experiences increasing sizes of reactor pulses.[25,26] Further use of the I-15 instrumented fuel element in larger pulses showed an increased steady state maximum fuel temperature. It is believed that the larger pulses produced a permanent strain in the fuel cladding due to the fuel thermal expansion. At the lower average steady state fuel temperature the fuel expansion is less than that occurring during the pulse, thus creating a fuel to fuel cladding gap. A new fuel management strategy has been developed to manage the sustained steady state fuel temperature.[27]

The principal computer programs used to perform the calculations are PSU-LEOPARD[8], EXTERIMINATOR-2[9], MCRAC[10], and SCRAM[11]. PSU-LEOPARD incorporates the standard LEOPARD[12] computer program as originally received and adds additional subroutines. LEOPARD and PSU-LEOPARD calculate the group constants of the core as a function of burnup.

MCRAC is an automatic, multi-cycle, two-dimensional depletion code that gives the power distribution, k_{eff}, and isotopic inventory of the core at each burn-up step. It is based on the flux and k_{eff} calculation performed by EXTERMINATOR-2, a multi-group two-dimensional diffusion theory code.

SCRAM is a multi-cycle depletion code created specifically for TRIGA reactors and adapted to the PSBR lattice design. It uses analytical equations to compute the power distribution, k_{eff}, and isotopic inventory for each cycle. The analytical equations are based on diffusion theory and the empirically fitted constants are derived using the PSU-LEOPARD, EXTERMINATOR-2, and MCRAC codes. In general, the calculations give good agreement with the measured power distributions and neutron fluxes.[1,2,7] Any equivalent codes can be used as long as they are properly benchmarked.

The fact that one can calculate the power distribution with good accuracy is important to calculating the safety margin in PSBR operation. The calculations identify the fuel element having the maximum elemental power density, MEPD, in the core, and thus the one which will produce the highest fuel temperatures. This is true for both steady state and pulse operations.

The experimental and analytical studies which have been performed to show the safety margin in the operation of the PSBR are described in this section. In particular, the maximum measured fuel temperatures during steady state operation and pulse operation are mathematically related in a unique way to allow predictions of their values during the PSBR operation. All predictions indicate that the design and construction of the PSBR is such that the safety limit of the fuel will not be exceeded during steady state operation. Further, any pulse temperature of too large a magnitude can be prevented by reviewing the steady state temperature measurements prior to pulsing a large excess reactivity into the core as part of administrative control. This is also true for abnormal operating conditions.

13.1 Accident-Initiating Events and Scenarios

The following accidents are analyzed:
1. The loss of coolant accident.
2. The design basis accident which includes cladding rupture.
3. A reactivity accident.

The results of these analyses demonstrate that the reactor can continue to be operated safely within bounds of this safety analysis and the regulatory limits.

The following discussion in this section (subsections A and B) provides background information prior to the discussion of the accidents in section 13.1.1, 13.1.2 and 13.1.3.

A. TRIGA Fuel Temperature Analysis of the Penn State Breazeale Reactor

There are two limiting conditions for establishing maximum allowed fuel temperatures. First, when the reactor is operating in the pool of water, the fuel temperature safety limit is 1150 °C. Under these conditions, the fuel cladding temperature is less than 500 °C and the cladding will not be ruptured by the internal hydrogen pressure.[13] During a loss of coolant accident (LOCA), the fuel is not covered with water and must be air cooled. Secondly, when the fuel is air cooled, the cladding temperature will go above 500 °C, where the strength of the cladding decreases. Below a fuel temperature of 950 °C the hydrogen pressure will not rupture the cladding when the fuel and the cladding are the same temperature.[13] Under these conditions, the fuel temperature safety limit is 950 °C.

Flux gradients across the fuel produce uneven temperature distributions. Pulsing a TRIGA fuel element to high power densities produces sudden expansion and contraction. During the rapid expansion phase, a large temperature gradient in the radial direction can cause uneven axial expansion producing a transverse bend. Experience with the TRIGA fuel elements has shown that they can receive thousands of pulses without being damaged provided their temperature limits are not exceeded. TRIGA fuel elements are considered damaged and no longer useable if their cladding has been ruptured or their dimensions change to where the transverse bend exceeds 0.125 inches over the length of the cladding or the length increases 0.125 inches.

The temperature profile in a single TRIGA fuel element is a function of its fuel and fission product distribution and is different for pulse operation compared to steady state operation. During steady state operation, the maximum fuel temperature is at the central fuel-zirconium rod interface. Since the thermocouple is placed near this interface, the measured fuel temperature is close to the maximum fuel temperature[4] (the maximum fuel temperature has been analytically determined to be no more than 5% more than the measured fuel temperature). During a pulse, the maximum fuel temperature is near the fuel-cladding interface and the measured fuel temperature is no less than 60-65% of the maximum fuel temperature.[4] To know what limits to place on operation, it is important to understand the TRIGA fuel temperature distribution in a fuel element during steady state and pulse operation and to relate the measured fuel temperature to the maximum fuel temperature.

An instrumented TRIGA fuel element is built with three thermocouples placed 0.0226 ft radially from the center, but spaced vertically 1 inch apart. The middle thermocouple is in the midplane of the fuel region of the TRIGA fuel element. The thermocouple measures the fuel temperature at a specific point within the fuel element which is not the maximum fuel temperature for pulse operation. During a pulse, the temperature distribution is the same as that of the volumetric thermal source strength, $q'''(\underline{r})$, so that the peak fuel temperature is near the fuel cladding interface. This is due to self-shielding within the fuel. As a result, the measured fuel temperature can be significantly lower than the maximum fuel temperature. On the other hand, the peak fuel temperature during steady state operation is at the inner boundary of the fuel; thus, the measured fuel temperature is slightly less than the maximum fuel temperature. The measured fuel temperature in an 8.5 wt% U fuel element is closer to the maximum temperature than it is in a 12 wt% U fuel element because the self-shielding of the 12 wt% U fuel U is greater than the 8.5 wt% U fuel thereby producing a $q'''(\underline{r})$ with a steeper gradient.

The temperature distribution within the fuel can be calculated from a knowledge of fuel geometry, heat transport parameters, and $q'''(\underline{r})$ as shown by Haag and Levine.[3] The volumetric thermal source strength in a fuel element is a function of the core power, the core configuration, and the element's position within the core. For any core configuration, $q'''(\underline{r})$ can be determined by neutronic analysis and then used to determine the peak temperature during a pulse or during steady state operation. The steady state fuel temperature is determined by the boundary conditions at the cladding water interface and the value of $q'''(\underline{r})$. Studies performed by Haag and Levine have shown that subcooled boiling takes place in the PSBR when the TRIGA core exceeds 200 kW. This helps limit the temperature rise of the cladding surface temperature, t_c, because when boiling occurs t_c increases proportional to approximately $(q'')^{0.33}$.[14] Hence, the heat flux, q'', must increase by a factor of 8 to increase the difference between t_c and the water saturation temperature by a factor of 2.

It is important to recognize that the $q'''(\underline{r})$ produced in a fuel element for a particular core configuration is the heat source that establishes the fuel temperature for both steady state operation and pulse operation. Hence, there is a direct relation between the measured fuel temperature at steady state and during pulse operation for the same core configuration and fuel element. The fuel temperature measured in the fuel element having the highest power density in the core during steady state operation can be used to determine the maximum fuel temperature in the core during a pulse. This relationship is described in this section.

(1) <u>Steady State Analyses</u>

Standard heat transport calculations are used to analyze the steady state fuel temperatures for the PSBR.

Let

$q_j'''(\underline{r})$ = volumetric thermal source strength at position $\underline{r}$ within the j^{th} region.

P_j = power generated by the j^{th} fuel element.

Then

$$P_j = \int_V q_j'''(\underline{r})dV, \tag{1}$$

where the integral is over the fuel volume, V, of the j^{th} fuel element. The average power, $\overline{P}$, produced by a fuel element in the core, and the normalized power for the j^{th} fuel element, NP_j, are related to P_j by the expression

$$P_j = \overline{P}NP_j, \tag{2}$$

A core of N_C fuel elements producing a total power of Q can be expressed as

$$Q = N_C\overline{P} = N_C\overline{\overline{q}}'''V, \tag{3a}$$

and

$$\overline{P} = \frac{Q}{N_C} = \overline{\overline{q}}'''V, \tag{3b}$$

where $\overline{\overline{q}}'''$ is the volumetric thermal source strength averaged over all fuel in the core, and N_C is the number of fuel elements in the core.

It can be immediately observed that operating at 1 MW(th), implies that for a core with $N_C = 90$, then $\overline{P} = 0.0111$ MW(th), and for a core with $N_C = 100$, then $\overline{P} = 0.010$ MW(th).

Using Goodwin's[15] measured $q_j'''(r,z)$ as

$$q_j'''(r,z) = \left(A_o + B_o r^2\right)\overline{q}_j'''(z), \tag{4}$$

where for the j^{th} fuel element

$$\int_V q_j'''(r,z)dV = \overline{\overline{q}}_j'''V, \tag{5}$$

z is the axial position along the j^{th} fuel element, and A_o and B_o are constants. A thermocouple is located at the fuel midplane [redacted] Thus, at the fuel midplane,

$$q_j'''(r[redacted]) = q_{oj}'''(r) = \bar{q}_{oj}'''(A_o + B_o r^2) \tag{6}$$

and

$$\bar{q}_{oj}''' = f_a \bar{\bar{q}}_j''' = f_a NP_j \bar{\bar{q}}''' = f_a NP_j \frac{\bar{P}}{V}, \tag{7}$$

where f_a is the axial peaking factor. The following definitions are used for the j^{th} fuel element in these equations:

$q_j'''(r, z)$ = point volumetric thermal source strength,

$\bar{q}_j'''$ = volumetric thermal source strength averaged over the radial direction ,

$\bar{\bar{q}}_j'''$ = volumetric thermal source strength averaged over the fuel volume.

When the j^{th} subscript is missing, $\bar{q}$ '''and $\bar{\bar{q}}$ ''' refer to the fuel element producing an average power in the core.

For the j^{th} fuel element

$$P_j = \bar{\bar{q}}_j'''V, \tag{8a}$$

and

$$\bar{\bar{q}}''' = \frac{\bar{P}}{V}, \tag{8b}$$

Equation (8a) can be written, using Equation (2), in the following form:

$$P_j = NP_j \bar{\bar{q}}'''V \tag{9}$$

The temperature rise between the fuel and the cladding at the fuel element midplane during steady state operation is directly dependent on $q_{oj}'''(r)$. Dropping the j subscript for convenience but remaining in the fuel midplane

$$\frac{1}{r}\frac{1}{dr}\left[k_f r\frac{dt}{dr}\right] = -\bar{q}_o'''(A_o + B_o r^2), r_z \le r \le R, \tag{10}$$

where

r_z = radius of Zr rod in the center of the fuel rod,

R = radius of the fuel rod.

Integrating Equation (10) gives

$$t(r) - t_s = \frac{\bar{q}_o'''}{2k_f}\left[\frac{A_o}{2}(R^2 - r^2) - A_o r_z^2 \ln\frac{R}{r} + \frac{B_o}{8}(R^4 - r^4) - \frac{B_o r_z^4}{2}\ln\frac{R}{r}\right], \tag{11}$$

All parameters in Equation (11) are described and given in **Table 13-1**. Substituting the values for the parameters into Equation (11) gives, for the thermocouple temperature, t_{tc},

$$t_{tc} - t_s = 6.039\text{x}10^{-5}\bar{q}_o''', \tag{12a}$$

Substituting Equation (7) into Equation (12a) and returning to the j^{th} fuel element gives

$$(t_{tc} - t_s)_j = 6.039\text{x}10^{-5}\frac{f_a\bar{P}}{V}NP_j, \tag{12b}$$

Using Q = 1 MW(th) in Equation (3b),

$$\bar{P} = \frac{1}{N_C}3.412\text{x}10^6\frac{\text{BTU}}{\text{hr}}, \tag{13}$$

Table 13-1

Parameters for the 12 wt% U-ZrH TRIGA Fuel Elements
(Enriched to less than 20% ^{235}U)

Thermocouple radius, R_{tc}	[redacted] †
Fuel mean radius, R	[redacted] †
Zr rod radius, r_z	[redacted] †
Cladding thickness, C	[redacted]
Fuel element radius, R + C	[redacted] †
Conductivity cladding, k_c	9.5 Btu/hr ft^2 °F
Conductivity fuel, k_f	10.5 Btu/hr ft^2 °F
Core average volumetric thermal source strength, $\bar{q}'''$	2.49×10^8 $\frac{\text{Btu}}{\text{hr ft}^3}$
A_o (12 wt% U fuel new) (Reference 4)	0.6534
B_o (12 wt% U fuel new) (Reference 4)	202 ft^{-2}
Number of elements, N_c (Loading 36)	[redacted]
Number of elements, N_c (Loading 45T)	[redacted]
Axial peaking factor, f_a	1.25*
Prompt temperature coefficient., α (Ref. 19)	-1.4×10^{-4} δk/k/°C
C_f (Loading 36) Correction Factor	0.98**

* This value is used for analyses subsequent to the original SAR since it is a more typical value used for TRIGA reactors.

** See the description of this factor in the text immediately after Equation (15).

† These are nominal values and may vary with the manufacturers specification. It is also assumed that the fuel - cladding gap is zero (0) which in general is not true.

The volume of fuel in a TRIGA fuel element is

$$V = \pi(R^2 - r_z^2)\,H,$$

where H is the fuel height. Using [redacted] the value from Table 13-1 for the other parameters

V = [redacted] (14)

Substituting Equations (13) and (14) into Equation (8b) gives

$$\bar{q}''' = C_f \frac{2.49 \times 10^8}{N_C} \frac{\text{BTU}}{\text{hrft}^3}, \qquad (15)$$

where C_f is a correction factor for setting the power instrument to read 1 MW(th) when the actual power is reduced by the factor $(1-C_f)$. This reduction is to provide an extra margin of safety to compensate for an uncertainty in calibration. What value is used for C_f depends on how the equations are being used. If they are being used to predict fuel temperature from a given NP, a more conservative $C_f = 1$ should be used. If the equations are being used to determine NP from measured fuel temperature, a $C_f \leq 1$ may be used depending on the confidence the experimenter has in the accuracy of the power calibration.

Equation (12b) can now be written as:

$$(t_{tc} - t_s)_j = 1.5\text{x}10^4 C_f \frac{f_a NP_j}{N_C} \text{ °F}, \tag{16}$$

$$(t_{tc} - t_s)_j = 8.33\text{x}10^3 C_f \frac{f_a NP_j}{N_C} \text{ °C},$$

The measured fuel temperature, t_{tc}, depends on the temperature at the cladding surface in the fuel midplane, t_c. Because of subcooled boiling above 200 kW, this temperature rises very slowly. The Δt is proportional to $(q''')^{0.33}$, where Δt is the difference between t_c and the coolant saturation temperature.[14] As a result, it is assumed that the surface cladding is superheated by a fixed Δt degrees and thus at 1 MW(th), t_c = 140 °C. This should be correct within ± 10 °C at 1 MW(th) for all NP_j's greater than 1 and less than 3. We may write

$$t_{tcj} = (t_{tc} - t_s)_j + (t_s - t_c)_j + t_c, \tag{17}$$

where the first term on the right hand side of Equation (17) is evaluated using Equation (16). The second term, the temperature change between the fuel cladding and the surface of the fuel rod, is derived assuming a gap, g, between the cladding and the fuel. Solution of the standard heat equation gives for

$$(t_s - t_c)_j = (t_s - t_g)_j + (t_g - t_c)_j, \tag{18a}$$

the following equation:

$$(t_s - t_c)_j = \frac{q_{oj}'}{2\pi k_g} \ln\frac{R+g}{R} + \frac{q_{oj}'}{2\pi k_c} \ln\frac{R+g+C}{R+g}, \tag{18b}$$

where

t_s = temperature of the fuel at the fuel cladding interface
t_g = temperature of the cladding facing the gap
k_g = conductivity of the gap
k_c = conductivity of the cladding
C = thickness of the cladding
g = thickness of the gap
q_{oj}' = linear heat generation rate

and the other parameters are as previously defined.

Thus

$$(t_s - t_g)_j = \frac{q_{oj}'}{2\pi k_g} \ln \frac{R+g}{R}, \tag{19a}$$

and

$$(t_g - t_c)_j = \frac{q_{oj}'}{2\pi k_c} \ln \frac{R+g+C}{R+g} \cong \frac{q_{oj}'}{2\pi k_c} \ln \frac{R+C}{R}, \tag{19b}$$

Equation (19a) will be evaluated experimentally as described later, whereas Equation (19b), the temperature drop across the cladding, can be evaluated from the physical values of the parameters.

By definition

$$q_{oj}' = \pi(R^2 - r_z^2)\bar{q}_{oj}''' = \pi(R^2 - r_z^2) f_a N P_j \bar{\bar{q}}''', \tag{20}$$

Assuming an average core temperature drop across the gap, $\overline{\Delta t}_g$, Equation (19a) becomes

$$(t_s - t_g)_j = C_f f_a N P_j \overline{\Delta t}_g, \tag{21a}$$

where

$$\overline{\Delta t}_g = \frac{R^2 - r_z^2}{2k_g} \bar{\bar{q}}''' \ln \frac{R+g}{R}. \tag{21b}$$

Also, Equation (19b) becomes, using Equation (20)

$$(t_g - t_c)_j = \frac{R^2 - r_z^2}{2k_c} f_a N P_j \bar{\bar{q}}''' \ln \frac{R+C}{R}. \tag{21c}$$

Using the values of **Table 13-1**, Equation (21c) reduces to

$$(t_g - t_c)_j = 1.248\text{x}10^3 C_f \frac{f_a N P_j}{N_C} \ ^\circ\text{F}, \tag{21d}$$

$$(t_g - t_c)_j = 6.936\text{x}10^2 C_f \frac{f_a N P_j}{N_C} \ ^\circ\text{C}.$$

Substituting Equations (21a) and (21d) into Equation (18a), and using the result in Equation (17), it follows that

$$t_{tcj} = \left(\frac{9.024\text{x}10^3}{N_C} + \overline{\Delta t_g} \right) C_f f_a N P_j + 140 \ ^\circ\text{C}. \tag{22}$$

Equation (22) is used to calibrate an instrumented 12 wt% U fuel element to provide a measured fuel temperature, t_{tc}, during steady state operation.

(2) Pulsing Characteristics of the PSBR

The temperature distribution in a TRIGA fuel element during a pulse has the same distribution as expressed in Equation (4) up to 89% of the fuel radius.(15) It has been found that adiabatic conditions hold up to 0.07 sec. during which time the maximum fuel temperature is reached.(15) Using the values of **Table 13-1**, it is found(4) that the maximum fuel temperature during a pulse is 1.6 times that measured by the thermocouple. Thus, during the pulse, the shape of the temperature distribution in a fuel element remains constant, but the magnitude quickly rises. What we are concerned with here is the maximum fuel temperatures reached during the pulse. To prevent confusion, we use the term highest maximum fuel temperature to refer to the highest temperatures reached at any point within the fuel element during the pulse. The highest maximum fuel temperature is thus the maximum fuel temperature reached during a pulse and must remain below 1150 °C. However, the highest measured fuel temperature is 1/1.6 or 0.625 times the highest maximum fuel temperature which corresponds to a measured fuel temperature of 720 °C. Thus, setting the Limiting Safety System Setting (LSSS) at 650 °C, corresponds to a maximum fuel temperature of 1040 °C. The LSS scram will have no effect on the maximum fuel temperature reached during a pulse because the instrumentation time lag allows the peak to be reached before a scram can occur. The maximum fuel temperature reached during a pulse must be limited by the magnitude of the prompt excess reactivity insertion (δk_p) and/or the $q'''(r)$ produced in a fuel element for a particular core configuration.

A semi-empirical equation, Equation (29), is developed using the definition of the prompt temperature coefficient. The large negative prompt temperature coefficient, α, provides the TRIGA core with its pulsing capability. When excess k_{eff}, $\delta k_{ex} = k_{eff} - 1$, is inserted into the reactor, the reactor will go on a prompt period, provided

$$\delta k_p = \delta k_{ex} - \beta \quad , \tag{23}$$

is positive, i.e., $\delta k_p > 0$. β is the effective delayed neutron fraction (0.007).

Let:

$\overline{\delta t}_p$ = maximum fuel temperature rise averaged over the total core fuel volume for a pulse.

$\overline{\delta t}_{poj}$ = maximum fuel temperature rise averaged over the radius of the j^{th} fuel element at its midplane.

α = prompt temperature coefficient of reactivity of the TRIGA core.

The prompt temperature coefficient is defined as:

$$\alpha = -\frac{\delta k_p}{\overline{\delta t}_{pp}}, \tag{24a}$$

or

$$\overline{\delta t}_{pp} = -\frac{\delta k_p}{\alpha}, \tag{24b}$$

where

δk_p = prompt excess reactivity insertion.

$\overline{\delta t}_{pp}$ = average maximum rise in core fuel temperature due to the prompt excess reactivity insertion.

Equation (24b) does not include the average core temperature rise due to a pulse insertion of $1 excess reactivity, δt_{p1}. Thus, the total average fuel temperature rise during a pulse, $\overline{\delta t}_p$, is:

$$\overline{\delta t}_p = \overline{\delta t}_{pp} + \overline{\delta t}_{p1}. \tag{25a}$$

For the j^{th} fuel element, its corresponding temperature increase in the midplane is

$$\overline{\delta t}_{poj} = f_a NP_j \overline{\delta t}_p, \tag{25b}$$

or

$$\overline{\delta t}_{poj} = f_a NP_j\left(-\frac{\delta k_p}{\alpha}\right) + f_a NP_j \overline{\delta t}_{p1}. \tag{25c}$$

Initially the core fuel temperature is that of the pool water, T_O, and during the pulse an adiabatic increase in temperature, $\delta t_{poj}(r)$, is assumed. Hence,

$$\delta t_{poj}(r) = \overline{\delta t}_{poj}(A_o + B_o r^2). \qquad (26a)$$

Equation (26a) expresses the maximum temperature rise above room temperature for the j^{th} fuel element as a function of fuel radius. For convenience

$$\delta t_{poj}(r) = \overline{\delta t}_{poj} f(r), \qquad (26b)$$

where

$$f(r) = A_o + B_o r^2 \qquad (26c)$$

The temperature in the midplane of the fuel element at any r position, $t_{poj}(r)$ can be expressed as:

$$t_{poj}(r) = \overline{\delta t}_{poj} f(r) + T_o. \qquad (27a)$$

Let

t_{poj} = maximum pulse temperature measured by the thermocouple in the j^{th} fuel element.

Then

$$t_{poj} = \overline{\delta t}_{poj} f(r_{tc}) + T_O. \qquad (27b)$$

Using the values of Table 13-1

$$T_{poj} = \text{[redacted]}\ \overline{\delta t}_{poj} + T_o. \qquad (27c)$$

Substituting Equation (25c) into Equation (27c)

$$t_{poj} = \text{[redacted]}\left[f_a NP_j\left(-\frac{\delta k_p}{\alpha}\right) + f_a NP_j \overline{\delta t}_{p1}\right] + T_O \qquad (28)$$

Equation (28) is used as the basis for developing the semi-empirical equation, Equation (29), to fit the actual pulse data as a function of NP_j, i.e.,

$$t_{poj} = K_{14} f_a NP_j\left(-\frac{\delta k_p}{\alpha}\right) + f_a NP_j \overline{\delta t}_{po} + T_O, \qquad (29)$$

where K_{14} and $\overline{\delta t}_{po}$ are empirical constants to be determined experimentally. The experimental data may also be represented by:

$$t_{poj} - T_o = M_j \delta k_p + b_j, \tag{30}$$

where

$$M_j = \frac{K_{14} f_a NP_j}{-\alpha}, \tag{31a}$$

and

$$b_j = f_a NP_j \overline{\delta t}_{po} \tag{31b}$$

NP_j and $\overline{\delta t}_{po}$ are determined by fitting the experimental data to Equation (30) where NP_j, f_a, and α are known. In Equation (30), $M_j \ \delta k_p$ represents the temperature rise during a pulse due to the prompt excess reactivity (δk_p) insertion and b_j represents the corresponding temperature rise due to the \$1 excess reactivity insertion.

(3) TRIGA Experiment to Measure Fuel Temperatures

Using the analyses of the previous sections, a calibration was made to determine fuel temperatures for steady state and pulse modes of operation. This section describes the calibration techniques.

A series of fuel temperature measurements were made using the 12 wt% U fuel instrumented fuel elements in core configuration loading 36 as shown in **Figure 13-1**. One instrumented fuel element, I-13, had been in the core since September 1977 (in the B ring, the position of MEPD) and the other, I-14, had never been used. The first series of measurements was taken with I-13 in the G-8 core position and I-14 in the G-10 core position. The core position of a fuel element is identified in **Figure 13-1** by a letter for the vertical axis position and a number for the horizontal axis position. The instrumented fuel elements have been numbered sequentially with an I prefix. After rotating both fuel elements at 0.5 MW(th) steady state operation to obtain maximum temperature readings, the reactor power was increased in steps to 0.7 MW(th), 0.9 MW(th), and 1 MW(th). The actual values of the power were 0.98 (for loading 36) of that read on the recorder because the readout on the linear recorder was adjusted to read 0.4 MW(th) when the actual power, as determined by a thermal power calibration, was approximately 390 kW (see the description of C_f immediately after Equation (15)).

After completing the series of steady state runs, the reactor was pulsed sequentially with 2, 2.25, 2.50, and 2.75 dollar pulses. Before each pulse, the reactor was made subcritical to allow the temperature to reach equilibrium.

The above experiment was then repeated to determine reproducibility and measure the effect of the gap in I-14 created by the 4 pulses. The above measurements were again repeated with I-13 positioned in G-10 and I-14 positioned in G-8 to again study the reproducibility of the data and obtain another measurement on the temperature drop across the fuel cladding gap, $\overline{\Delta t}_g$.

[redacted]

The steady state and pulse measurements were again repeated, first with I-14 in F-10 and then with I-14 in H-11; I-13 was in position in G-8 for both sets of these measurements.

The data for all measurements (done prior to 1986) are summarized in **Table 13-2**. Both the chart recorder and the meter were used to measure the fuel temperature of I-13 as shown in **Table 13-2**. The chart recorder was connected to the thermocouple at the midplane of the fuel element, whereas, the meter was connected to the thermocouple located 1" below. The banked control rods during a pulse causes the position of the highest power density in the fuel element to be displaced slightly downward from the midplane of the fuel. This causes the meter readings to be approximately 24 °C higher than those read on the chart recorder.

(4) <u>Evaluation of $\overline{\Delta t_g}$ for Fuel Element I-14</u>

The unused TRIGA 12 wt% U fuel instrumented fuel element, I-14, has been placed in the core configuration of Figure 13-1, Core Loading 36, and experiments performed to evaluate Equation (22). It is assumed that before pulsing the instrumented fuel element, I-14, it had a $\overline{\Delta t_g}$ equal to 0 as the fuel would be in contact with the cladding. When the fuel element is first pulsed, the cladding is stretched introducing a gap which increases the $\overline{\Delta t_g}$. After a number of pulses of the same size (i.e. \$2.50), $\overline{\Delta t_g}$ reaches a maximum value and does not increase with further pulsing.[4]

The increase in $\overline{\Delta t_g}$ after pulsing I-14 several times is now determined by comparing the steady state temperatures for the same condition after each set of pulses. It has been found that measured fuel temperatures will increase further due to an increase in $\overline{\Delta t_g}$ when at some later time pulses of a larger size are performed (i.e. \$3.00).[25, 26]

At 1 MW(th), I-14 measured t_{tc} = 372 °C in Core Loading 36 and position G-10 before any pulsing began. (Note that for work that was done for the original SAR $C_f = 0.98$ and $f_a = 1.35$ and for subsequent analyses 1 and 1.25 are used respectively. The former because of the reasons stated immediately after Equation (15) and the latter because the 1.25 is a more typical value used as a design specification for axial peaking for TRIGA fuel (31)) Using Equation (22),

$$372 = 0.98\frac{9.02x10^3}{94.6}x1.35NP_j + 140,$$

it follows that

$$NP_j = \frac{372-140}{128x0.98} = 184.$$

After 4 pulses

$$t_{tc} = 418\ ^\circ C,$$

and thus using Equation 22 again while holding NP_j constant at 1.84,

$$418 = 0.98(95.39 + \overline{\Delta t_g})(1.35)(1.84) + 140,$$

or

$$\overline{\Delta t_g} = 19\ ^\circ C.$$

Table 13-2

Fuel Temperature Measurement Data for Loading 36
$T_0 = 21\ ^\circ C$

Fuel Element	Core Position	ttc (°C) SS 1 MW(th)	tpo (°C) Recorder/Meter Pulse \$2.00	Pulse \$2.25	Pulse \$2.50	Pulse \$2.75
I-13	G-8	412	333/379	392/421	436/467	478/509
I-13	G-8	411	---	---	---	---
I-13	G-8	411	343/381	387/421	431/461	478/511
I-13	G-8	411	350/381	389/419	435/466	478/511
I-14	G-8	455	389	427	468	517
I-14	G-8	466	395	434	482	518
I-13	G-10	381	323/333	359/371	399/412	430/453
I-13	G-10	382	311/332	357/373	400/416	439/453
I-14	G-10	372	339	375	415	456
I-14	G-10	418	---	---	---	---
I-14	G-10	450	348	391	425	466
I-14	G-11	433	342	373	411	449

I-13 is a 12 wt% U fuel element burned to 2.2 Megawatt days
I-14 is a fresh 12 wt% U fuel element

These data and analysis show the initial (few pulses) increase in the temperature across the gap. Further data has shown that the increase in $\overline{\Delta t_g}$ diminishes to zero with successive pulses (see Table 13-2, lines 9 and 10). Element I-14 was then moved from position G-10 to position G-8 in the B-ring where the NP_j is different from that in G-10. Assuming $\overline{\Delta t_g} = 19\ ^\circ C$ and using $t_{tc} = 445\ ^\circ C$ as measured in its new position at 1 MW(th), it follows that

$$445 = 0.98\ (114.4)\ 1.35\ NP_j + 140.$$

Therefore,

$$NP_j = 2.02.$$

After 8 pulses, the t_{tc} for element I-14 was measured again in position G-8. This time t_{tc} was 466 °C at 1 MW(th). Therefore, using Equation (22) while holding NP_j constant at 2.02,

$$466 = (96.4 + \overline{\Delta t_g})(1.35)(2.02)0.98 + 140,$$

we find

$$\overline{\Delta t_g} = 26.6\ ^{\circ}C.$$

It can be observed that after 8 pulses, $\overline{\Delta t_g} = 26.6\ ^{\circ}C$. Past studies have shown that additional pulses do not alter the $\overline{\Delta t_g}$ significantly.[25] For I-14, it is assumed that after many more pulses of the same size (i.e. $2.50), the $\overline{\Delta t_g}$ increase will be 1 °C, hence we use

$$\overline{\Delta t_g} = 27.6\ ^{\circ}C,$$

and Equation (22) becomes for Loading 36 at Q = 1 MW(th)

$$(t_{tc})_j = 163 C_f NP_j + 140. \qquad (32)$$

Equation (32) can now be used to determine the NP_j for I-14 anywhere in Core Loading 36 at 1 MW(th) power as long as larger sized pulses are not performed resulting in an increase in $\overline{\Delta t_g}$.[25] To generalize Equation (32) for any core configuration and similar fuel element design specifications, it is only necessary to account for N_C. If this is done, Equation (32) becomes

$$(t_{tc})_j = \frac{1.57 \times 10^4}{N_C} C_f NP_j + 140. \qquad (33)$$

The steady state data of **Table 13-2** has been evaluated using Equation (33) and the results are tabulated in **Table 13-3**. The t_{tc} for the G-10 position was not measured after all pulsing had ceased and, therefore, is not listed in **Table 13-3**.

Table 13-3

Evaluation of NP_j's from I-14 Data

Core Position	ttc (°C)	NPj Steady State Eq. (33)	NPj (Ave) Pulse Eq. (34)
G-8	467	2.01	2.07
F-10	450	1.90	1.85
H-11	433	1.80	1.78
G-10		1.84	1.80

Table 13-4

Pulse Parameter Characteristics of Fuel Element I-14

Core Position	NP_j	$M_j \times 10^4$ °C/δk/k	$M_j/NP_j \times 10^4$ °C/δk/k	b_j °C	b_j/NP_j °C
G-10	1.84	2.23	1.21	162	88
G-8	2.01	2.44	1.21	197	98
G-8	2.01	2.34	1.17	210	104
F-10	1.90	2.25	1.18	170	89
H-11	1.80	2.04	1.13	178	99

Equation (33) is used to evaluate the measured fuel temperature during steady state operation. During steady state operation, the measured fuel temperature is close to the maximum fuel temperature (within 5%). In this case, the LSSS of 650 °C is extremely conservative because under steady state conditions, the maximum fuel temperature is no greater than 682 °C (650 °C + 5%) and thus, is well below the safety limit fuel temperature of 1150 °C. For loading 36, an upper limit for the measured maximum fuel temperature can be determined by setting NP = 2.2. Extensive calculations have been performed[1, 2, 5, 7, 27] to study the maximum power distribution produced by different core configurations with fresh 12 wt% U fuel in the B-ring and the other core configurations containing a mixture of both 12 wt% U fuel and 8.5 wt% U fuel. Future maximum steady state measured fuel temperature will be below the 650 °C LSSS.

(5) Evaluation of the Pulse Data for Fuel Element I-14

Each series of pulse data using I-14 is fitted to a straight line to determine M_j and b_j of Equation (30). **Table 13-4** summarizes the results, wherein the data in the last column show that b_j / NP_j is constant within ± 8% for the different core positions. The constants K_{14} and $\overline{\delta t_{po}}$ are determined to be 1.22 and 71 respectively using Equation (31). Substituting these values and those from **Table 13-1** into Equation (29) the result is

$$t_{poj}=1.177\mathrm{x}10^4 NP_j \delta k_p+95.8NP_j+T_0. \tag{34}$$

Equation (34) is now used to evaluate NP_j for the various core positions. The results shown in **Table 13-5** give consistent values for NP_j using different pulse magnitudes. This validates Equation (34). The value of NP_j as obtained from steady state Equation (33) is in good agreement with the corresponding value of NP_j obtained using the pulse Equation (34). The highest measured fuel temperatures in fuel element I-14 are compared to that using Equation (34) in **Figure 13-2**. It can be observed that the measured temperatures are in good agreement with Equation (34).

The Penn State in-core fuel management codes were employed to determine the power distribution and NP_j's for Core Loading 36. In a recent Ph.D. thesis,[16] the group constants of the individual fuel elements were evaluated as a function of their burnup using the SCRAM code. The SCRAM code provides a simple but reasonably accurate method of depleting the PSBR core as has occurred since December 1965. These constants were input into the EXTERMINATOR-2 code to obtain the NP_j's for Core Loading 36. The NP_j's for G-9 and H-10 with new 12 wt% U fuel varied between 2.03 and 2.10. This is to be compared with the measured values of 2.01 steady state and 2.07 by pulse. In general, the steady state and pulse NP's agree with ± 3%.

It is now possible to eliminate the NP_j from Equation (32) and Equation (34) to give for I-14

$$t_{poj} - T_o = \left(72.2\delta k_p + 0.588\right)\frac{t_{tcj} - 140}{C_f}\ ^\circ\mathrm{C}. \tag{35}$$

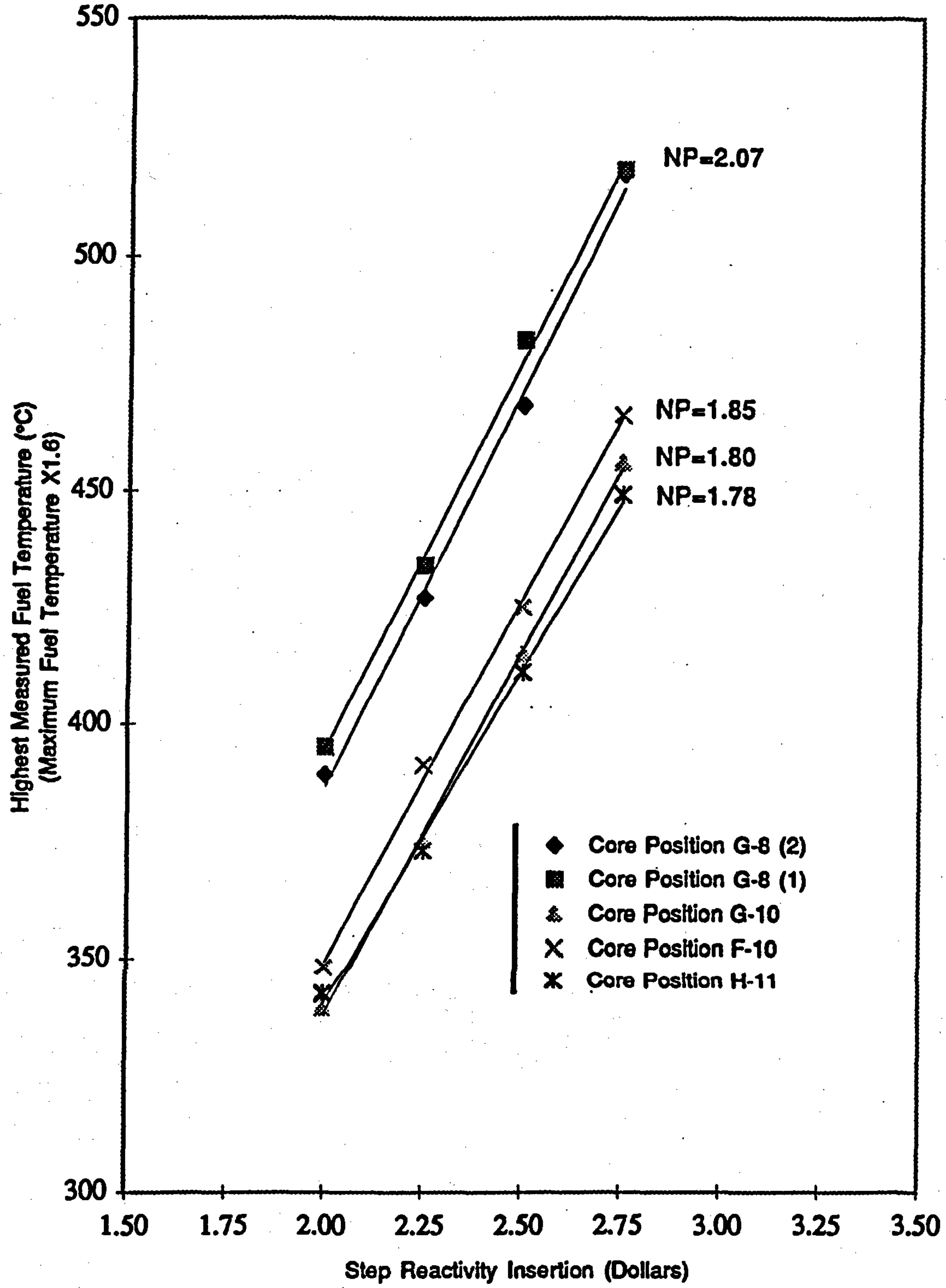

Figure 13-2 Comparing Highest Measured Fuel Temperatures During a Pulse with EQ(34) for Fuel Element I-14

Equation (35) is an equation developed for fuel element I-14. It can be used anywhere in the core to predetermine the highest measured fuel temperature as a function of pulse prompt excess reactivity insertion (δk_p). It will require using the I-14 measured temperature when operating at 1 MW(th) with I-14 in the same core position. For Core Loading 36 and using a maximum value of NP = 2.2 and corresponding t_{tc} = 499 °C, see Equation (32), the maximum value for t_{poj} is 684 °C for a pulse reactivity insertion of \$3.50 assuming T_o = 21 °C and C_f = 1.0. This corresponds to the maximum fuel temperature of 1095 °C which is below the safety limit of 1150 °C for fuel damage.

Thus, in the future, Equation (28) can be used to evaluate and predict t_{poj}. This requires placing I-14 in the hottest spot in the core and running at 1 MW(th) to evaluate t_{tc}. Then starting with a \$2 pulse, verify Equation (28) and predict t_{poj} for the high values of δk_p. The t_{poj} related to a \$2 pulse will be more than 100 °C below t_{poj} for a \$2.75 pulse and even much lower than that for a \$3.50 pulse. Hence, these initial pulses will produce maximum measured fuel temperatures well below 700 °C and allow determining the maximum fuel temperature attainable at the maximum allowable reactivity insertion pulses. A 700 °C fuel temperature measured by the thermocouple in a 12 wt% U fuel element corresponds to a maximum fuel temperature of 1120 °C. This is below the maximum allowed 1150 °C.

(6) <u>Evaluation of the Fuel Element I-13 Temperature Data (Pulse and Steady State)</u>

Table 13-2 shows the temperature data taken for I-13 and I-14. As expected, a review of these data shows that the measured temperature of I-13 during 1 MW(th) steady state operation, t_{tc}, is significantly lower than the t_{tc} of I-14 for the same core positions. On the other hand, the I-13 measured pulse temperature data is not significantly lower than that of I-14 for the corresponding conditions. This is because the A_o and B_o constants of Equation (14) are not the same for I-13 and I-14. The depletion of the outer rim of fuel in I-13 during burnup, in addition to the burnup of ^{235}U in all of the fuel element, lowers the self-shielding of thermal neutrons. As a result, the q'''(r) distribution for I-13 is much flatter than that of I-14. Using Equation (4) for I-13, and setting

$$A_o = 0.9267$$

$$B_o = 40$$

yields the results of the I-13 data as shown in **Table 13-6**. The equivalent of Equation (33) and Equation (34) for I-13 are Equation (36) and Equation (37) respectively.

$$(t_{tc})_j = \frac{1.68x10^4}{N_c} C_f NP_j + 140 \qquad (36)$$

$$t_{poj} = 1.475 \text{ x } 10^4 NP_j \delta k_p + 80 NP_j + T_o. \qquad (37)$$

Table 13-5

Table of NP_j Determined for I-14 Using Pulse Data in Eq. (34)

δk_{ex} Dollars	δk_p Dollars	Core Positions				
		G-10	G-8	G-8	F-10	H-11
2.00	1.00	1.79	2.06	2.10	1.84	1.8
2.25	1.25	1.79	2.04	2.08	1.86	1.78
2.5	1.50	1.79	2.04	2.10	1.85	1.78
2.75	1.75	1.81	2.07	2.07	1.86	1.78
Ave NP_j		1.80	2.05	2.09	1.85	1.78

It can be observed that the agreement between the pulse data and steady state data for determining NP_j is not as good as that for I-14. This is due to the approximations made in deriving Equations (33) and (34), namely,

a. The $A_o + B_o r^2$ shape of $q'''(r)$ approximates the excess burnup of U-235 at the perimeter of the U-ZrH fuel in I-13.

b. The $\overline{\Delta t}_g$ for I-13 is probably different from that of I-14.

However, temperatures measured by I-13 are consistent for the purposes of monitoring the core fuel temperatures.

(7) Conclusion (Temperature Analysis)

A major conclusion of this section (based upon the present fuel specifications) is that an unused instrumented 12 wt% U fuel element can be calibrated and used to monitor the maximum fuel temperatures in the core. Once calibrated, the fuel element will only be used to measure maximum fuel temperatures in new core configurations. For steady state operations a measured fuel temperature of 650 °C results in a maximum fuel temperature well below 1150 °C. Under these conditions, the measured fuel temperature is close to the maximum fuel temperature. For pulse operation, a measured 700 °C fuel temperature corresponds to a maximum fuel temperature of 1120 °C which is below 1150 °C. The safety limit shall not be exceeded during pulse or steady state operation.

Once a fuel element has been depleted, its maximum steady state temperature decreases as long as the gap between the fuel and the cladding remains the same. As experience has shown, pulsing at higher levels will increase the fuel/cladding gap and the maximum steady state temperature may increase before it starts to decrease. The fuel temperatures measured during steady state operation with a depleted fuel element are related to the fuel temperature of a new fuel element by a simple ratio. Hence, this ratio can be used to assess the maximum fuel temperature during steady state in a new fuel element. The maximum fuel temperatures measured with a depleted fuel element during a pulse are close to that in a new fuel element. The preferential depletion of the periphery of the fuel element causes the power distribution and hence, the temperature distribution during a pulse, to be flatter than that of a new fuel element. Thus, the measured fuel temperature in a depleted fuel element corresponds to a lower maximum fuel temperature. It is also closer to the average fuel temperature. The core average fuel temperature rise for a given δk_p insertion is the same for all cores. The lower NP for a depleted fuel element accounts for its being closer to the average core fuel temperature.

Table 13-6

Evaluation of NPj's from I-13 Data
I-13 (1.2 MWD Depleted)

Core Position	t_{tc} (°C)	NPj Steady State Equation (36)	NPj(Ave) Pulse Equation (37)
G-8	411	1.56	1.62
G-8	382	1.39	1.54

For comparison, see I-14 data in **Table 13-3.**

B. Evaluation of the Limiting Safety System Setting (LSSS)

The limiting safety system setting is a measured fuel temperature of 650 °C as defined in the TS.

If the core power were at 1.15 MW(th) (15% over power) steady state, the measured fuel temperature in the B-ring using extrapolated experimental data for, **Figure 13-3**, Core Loading 47[b] with 95.5 elements would be 650 °C, (using the 12 wt% U fuel element, I-15, which had been pulsed at the $3 level 20 times). The maximum temperature will be slightly higher, but the fuel temperature near the cladding will be approximately half this temperature. The extrapolated 650 °C fuel temperature is close to the maximum fuel temperature (within approximately 5%) due to the radial temperature distribution. A sudden insertion of reactivity with power at 1.15 MW(th), close to but less than $1, into the core will initially increase the reactor power exponentially at a period faster than one second. Using a negative temperature coefficient of 1 x 10^{-4} δk/°C,[c] the increase in average core fuel temperature is less than,

$$\frac{0.007\delta k/k}{1\mathrm{x}10^{-4}\delta k/k\ {}^\circ C} = 70^\circ C$$

and for an NP = 2.2 and f_a = 1.25, the maximum fuel temperature increase is 193 °C (2.2 x 1.25 x 70 °C = 193 °C). Adding this increased fuel temperature in the hottest fuel element to the 650 °C steady state temperature results in 843 °C, much less than the safety limit of 1150 °C. For this to occur at power levels above the power level scram setpoint will require that both power level scrams fail. The temperature scram will be initiated when the measured temperature exceeds its set point. The equilibrium temperature of 843 °C will be achieved at least within two to three periods (seconds) after reactivity insertion. A control rod drop time less than one second assures an early decrease in reactivity and fuel temperature. At this point, the control rods moving into the core will begin to decrease the reactor power in less than a second after the scram. Control rods are checked annually to assure their rod drop time is less than one second. The kinetics of the reactor cause the reactor power to decrease as soon as the control rods move a few inches into the core. Thus, the maximum fuel temperature will remain well below 1150 °C since the measured fuel temperature is close to the maximum fuel temperature for these quasi-static conditions.

The maximum allowed pulse reactivity of $3.50 is established to prevent the fuel temperature from exceeding the safety limit of 1150 °C. A $3.50 pulse, the maximum measured temperature starting from pool ambient temperature, using Equation (34) and NP = 2.2, is 684 °C. This corresponds to a maximum fuel temperature of 1095 °C. The temperature scram will not lower the maximum fuel temperature attained during a pulse once the pulse is initiated; however, it does protect the core from high temperatures during steady state operation. The core average fuel temperature is independent of core size for a given δk_p insertion, therefore the maximum fuel temperature attained during a pulse for an NP = 2.2 is also independent of core size for a given δk_p insertion.

[b]Core Loading 47 is considered an extreme loading relative to steady state measured peak fuel temperatures.
[c]The temperature coefficient during a fast period is slightly less than the prompt temperature coefficient.

[redacted]

13.1.1 Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) is an assumption that a fuel element cladding ruptures in an air cooled core releasing volatile fission products to the reactor bay. The MHA is defined as a postulated accident with potential consequences greater than those from any event that can be mechanistically postulated. The assumptions create conditions far more severe than is actually possible. Nevertheless, the accident is bounded by the regulatory limits.

The potential hazards associated with the MHA are related to the escape of fission products from the ruptured cladding of a fuel element into the reactor bay and then from the bay to the environment. The fission product buildup in a fuel element is a function of the fuel element power history and sustained measured steady state fuel temperature. It is assumed in these calculations that the fuel element is a 12 wt% U-ZrH fuel element operated in the core position of highest power density. The core is assumed to operate continuously at 1 MW(th) throughout its life. A review of the operating history of the 12 wt% U-ZrH fuel element I-15 shows that when operating at 1 MW(th) it reached a maximum measured fuel temperature of ≈ 600 °C with Core Loading 47, however 650 °C is used as the bounding sustained measured steady state fuel temperature used to calculate the release fraction. Since all of the volatile fission products reach their maximum after a few weeks of operation, a value of MEPD = 24.7 kW is used to compute the fission product activity, A_{fp}, in a fuel element. It is assumed that the rupture occurs when the reactor is just completing continuous 1 MW(th) operation. The saturated activity of the core, R, for one fission product (f_p) nuclide is

$$R = 1\text{MW} \times Y_D \frac{3.1\times10^{16}}{3.7\times10^{10}} \text{curies} \tag{37a}$$

where,

Y_D	=	fission product cumulative yield
1 MW(th)	=	3.1×10^{16} fission/sec
1 curie	=	3.7×10^{10} disintegration/sec

The fission product activity in the core is an accumulation of f_p activity from the previous weeks operation provided the half life is greater than approximately a day. The contribution to the f_p activity at the time of the rupture is as follows:

$R\left(1-e^{-\lambda T_1}\right)$ Activity produced during the week the rupture occurs,

$R\left(1-e^{-\lambda T_1}\right)e^{-\lambda(T_1+T_2)}$ Activity produced during the week before the rupture occurs,

and

$R\left(1-e^{-\lambda T_1}\right)e^{-\lambda n(T_1+T_2)}$ Activity produced during the n^{th} week before the rupture occurs.

The total activity of this f_p in the core at the time of the rupture is the sum of the above activities. It can easily be shown that the core activity is

$$C_o = R \frac{1 - e^{-\lambda T_1}}{1 - e^{-\lambda(T_1 + T_2)}} \tag{37b}$$

and,

$$A_{fp} = \frac{NP}{N_c} R \frac{1 - e^{-\lambda T_1}}{1 - e^{-\lambda(T_1 + T_2)}} \tag{37c}$$

where in the case of continuous operation at 1 MW(th),

T_1	=	number of hours operated in one week =168 hours
T_2	=	number of hours not operated in one week = 0 hours
$T_1 + T_2$	=	number of hours in one week = 168 hours

[redacted] For the 15 years from 1967-81, the PSBR operated an average of 15 MWhr/wk. For the 13 years from 1981-94, the PSBR was operated an average of less than 5.5 MWhr/wk. (Update Note: For the 19 years from 1986-2004, the PSBR operated an average of 7.2 MWhr/wk, with a high of 13.5 MWhr/wk in 2003). Continuous operation at 1 MW(th), therefore, establishes a large safety factor for this calculation, but allows for future increase of operating time.

Tables 13-7 and 13-8 list the activity of each of the important gaseous fission products. The fission product yields were taken from the Katcoff, et.al., report.[20] These yields include direct production plus precursor decay from fission products. Only a fraction of these fission products escape from the fuel element into the reactor bay. The fraction that escapes from the fuel element is called the release fraction, f_r.

The fission product release fraction, determined experimentally at General Atomic, is a function of the sustained fuel temperature.[21] The only fission products that escape are those that have diffused into the gap between the fuel and cladding during the operation of the reactor. Therefore, the release fraction in accident conditions is characteristic of the sustained normal operating temperature and not the temperature during an accident transient.

A review of the operating history of the instrumented fuel element I-13 in the PSBR B-ring shows that its maximum measured temperature during steady state operation at 1 MW(th) was less than 460 °C. After the first year of operation, its measured temperature at 1 MW(th) dropped to approximately 400 °C. During this period, its burnup was approximately 0.65 MWD. This temperature drop occurs because as burnup increases, the ^{235}U core inventory decreases with a corresponding drop in NP and temperature.

Operation after one year lowers the maximum measured fuel temperature in the B-ring to 400 °C or less. However, the experience mentioned above with I-15 in Core Loading 47, which is considered as an extreme (having a large peak to average temperature) core loading, indicates that measured fuel temperatures as high as 600 °C are possible. Thus, it is conservative to use a maximum measured fuel temperature of 650 °C, to compute the release fraction. Operation using the mixture of 12 and 8.5 wt% U fuel elements will only be allowed by the TS if the MEPD is ≤ 24.7 kW and the maximum measured fuel temperature, of an instrumented fuel element in the position of MEPD, is 650 °C. Thus the initial assumptions and the limits on operation resulting from this accident analysis are:

1. A maximum power operating history of continuous 1 MW(th) operation.
2. A maximum sustained measured steady state fuel element temperature of 650 °C.
3. A MEPD of 24.7 kW.

Experiments demonstrated at General Atomic[21] generated release fraction data for the U-ZrH fuel under various conditions. Below 400 °C the release fraction, f_r, is a constant, 1.5×10^{-5}, and above 400 °C the following equation is used:

$$f_r = 1.5x10^{-5} + 3.6x10^{3} \exp\left(\frac{-1.34x10^{4}}{T_o}\right) \quad (38)$$

where,

T_o is the fuel temperature in Kelvin.

For low temperature results, i.e., below 400 °C, the release fraction for a typical $U\text{-}ZrH_x$ fuel element is constant, independent of operating history or details of operating temperatures. Averaging Equation (38) over the volume and temperature profile of the fuel element gives a release fraction of 3.1×10^{-4} using maximum measured fuel temperature of 650 °C. Applying the release fraction of 3.1×10^{-4} to a single element operating at a MEPD of 24.7 kW yields, for each fission product activity in **Table 13-7** and **13-8**, a release to the reactor bay of C_o curies. The bay concentration, C_b(Ci/ml) is based on a minimum free air bay volume of 1900 m^3. An immediate and complete mixing is assumed to occur.

The concentration in the unrestricted area (outside the reactor building) is obtained by dividing the activity release rate through the emergency exhaust system by the dilution rate. The release rate for the emergency exhaust system is equal to the flow rate (3100 cfm or 1.46×10^{6} ml/sec) times the bay concentration C_b. The dilution rate is the wind velocity (1 m/s) times the cross-section area of the building (200 m^2) or 2×10^{8} ml/sec. Thus, at the instant the fuel element cladding ruptures, the maximum concentration in the unrestricted area, C_u(Ci/ml) is:

$C_u = C_b \times 7.30 \times 10^{-3}$ Ci/ml.

In the reactor bay (restricted area), the concentration in air of airborne activity decreases with time because of radioactive decay and the removal of air by the emergency exhaust system.

[redacted]

[redacted]

Consider a model in which activity, released from the single fuel element, instantly mixes completely with air in the bay. The resulting uniform concentration is C_b. We have, then,

$$C_T = C_b e^{-(\lambda_d + \lambda_r)T} \tag{39}$$

where,

C_T = bay air concentration of activity at elapsed time t after release from the single fuel element (Ci/ml)

λ_d = decay constant (sec^{-1})

λ_r = ventilation rate constant (sec^{-1})

$$\lambda_r = \frac{\text{ventilationrate}}{\text{bayvolume}} = \frac{1.46 m^3 sec^{-1}}{1900 m^3} = 7.7 x 10^{-4} sec^{-1}.$$

The exposure to an airborne concentration is equal to the integral of C_T over the period T, labeled IC_T.

$$IC_T = \frac{C_b}{\lambda_d + \lambda_r}\left(1 - e^{-(\lambda_d + \lambda_r)T}\right). \tag{40}$$

The reactor bay exposure in DAC-hours (derived air concentration hours) is equal to IC_T divided by the DAC value from 10 CFR Part 20 Appendix B. The same technique is used to determine the exposure in effluent limit concentration hours for the unrestricted area. The exposures determined for the reactor bay and the unrestricted area are then compared to the limits of 2000 DAC-hours for the reactor bay and 8760 effluent limit concentration hours for the unrestricted area, respectively, to arrive at the percent of the annual limit.

[redacted] Release of the activity from a fuel element over an extended period of time would reduce the dose because of the decay of short half-life radioisotopes before release.

[redacted]

[redacted]

[redacted] Since credit is not taken for the emergency exhaust system filtration in determination of the consequences of this accident, a Technical Specification requiring filtration as a limiting condition for operation is not required.

The above calculations assumed an MEPD of 24.7 kW and a maximum measured fuel temperature of 650 °C. Using the higher temperature increases the release fraction and the higher power density increases the fission product inventory. In addition, it was assumed that there are no mitigating circumstances such as fission products plating out on surfaces or dissolving in the pool water. It was further assumed that the rupture occurs to the element with the highest fuel temperature, the highest power density, and occurs immediately after continuous 1 MW(th) operation. Therefore using these parameters provide very conservative (hypothetical worst case) results.

The MHA creates conditions far more severe than are actually possible. Therefore, the fact that the MHA consequences are within the 10 CFR Part 20 limits outside the reactor bay in the unrestricted area shows that the PSBR's operation is safe to the public. [redacted]

13.1.2 Insertion of Excess Reactivity

In this accident, it is assumed that the reactor is taken to a 1.15 MW(th) power level with the transient rod inserted in the core and then the reactor is pulsed with a \$3 reactivity insertion. This accident requires a breakdown in the PSBR Standard Operating Procedures, the overpower scrams, and a failure of the interlocks.

When the core is operating at 1.15 MW(th), its total reactivity has been reduced by more than \$4 depending on the average core fuel temperature. The measured fuel temperature in the B-ring using extrapolated experimental data for Core Loading 47[d] with 95.5 elements would be 650 °C in the 12 wt% U fuel element, I-15. The maximum temperature will be slightly higher, but the fuel temperature near the cladding will be approximately half this temperature.

The maximum allowed core reactivity of \$7 leaves less than \$3 available for pulsing (actually the reactivity loss for Core Loading 47 would be ≈ \$4.75 at 1.15 MW(th) leaving only \$2.25 for a pulse).[e] Should a \$2.25 pulse occur while the reactor is at 1.15 MW(th), the measured fuel temperature will rise from 650 °C to 1030 °C as calculated using the maximum measured \$2.25 pulse temperature[e] for I-15 and Core Loading 47. In this case, when the core is pulsed from an initial power of 1.15 MW(th), the maximum fuel temperature is the measured fuel temperature.

[d] Core Loading 47 is considered an extrema loading relative to steady state measured peak fuel temperatures.
[e] These data are based on that obtained in 1994 & 1995.

This is because the temperature rise during a pulse has a different radial shape than that attained during steady state operation. During a pulse, the increase in fuel temperature is a maximum near the edge of the fuel. Superimposing this shape of the fuel temperature on that attained at a steady state power of 1.15 MW(th) produces, at the end of the pulse, a relatively flat radial temperature distribution at approximately 1030 °C. However, since the negative temperature coefficient acts immediately as the transient rod moves upward, the final maximum fuel temperature will be less than 1030 °C. If the steady state fuel temperature is higher for a particular loading the reactivity loss due to temperature feedback is greater and the reactivity available for a pulse is subsequently less. Effectively the core excess reactivity limit also limits the maximum temperature of the fuel in this accident independent of the initial steady state fuel temperature. Therefore with the limit on the core excess reactivity the initial steady state core temperature and the maximum reactivity available for the accidental pulse work against each other (raising one lowers the other and conversely) to limit the final maximum fuel temperature to less than 1030 °C.

Administratively, if the reactor is operating above 900 kW all four control rods must be balanced. This implies that the transient rod would not be available for the $2.25 pulse. In addition, the Technical Specification required pulse interlock (section 3.2.4), which prevents initiation of a pulse when reactor power is greater than 1 kW, will prevent the postulated accident. Also the Technical Specification required high power scrams will prevent operation at 1.15 MW(th). The result of the reactivity accident are peak fuel temperatures less than the safety limit of 1150 °C.

A $5.00 ramp analysis was performed for AFRRI by General Atomics.[24] It indicated that even with a reactivity addition rate of $2.50/second (averaged over the full rod travel) the safety limit (1150 °C) was not reached. The rod withdrawal was terminated with a high power scram less than 1 second into the event. A reactivity of $1.86 was added after criticality was achieved and before the SCRAM occurred. The maximum power in the transient was 330 MW(th) with a maximum fuel temperature of 330 °C. Compared to the above analyzed reactivity accident this excursion is inconsequential. It should be noted that the amount of reactivity available in the ramping rods does not impact on the final result as long as the reactivity addition rate does not exceed the $2.50/second rate and the SCRAM time is not significantly longer than that analyzed.

13.1.3 Loss of Coolant

The PSBR pool contains 71,000 gallons of water. [redacted] A series of alarms will occur as the water level drops more than 26 cm below reference pool full level. [redacted] If the reactor is operating at 1.0 MW(th), a low pool level alarm will alert the operator who is required by administratively approved procedure to shut down the reactor. There exists a moveable gate that can be used to isolate either side of the pool after the leak is noticed.

[redacted]

If the reactor is operating when the leak occurs the reactor operator will shut down the reactor upon receipt of the low pool level alarm. Within two minutes after the shut down the maximum fuel temperature will drop more than 350 °C.[17] Three minutes after the shut down the maximum fuel temperature is within 20 °C of the water temperature. [17]

[redacted]

As soon as the water falls below the reactor core, the fuel temperature will begin to rise, because the natural air convection cooling is less effective than water. The rate of rise of the fuel temperature will depend on the previous operating history of the reactor, and the effectiveness of natural air convection to cool the fuel elements. The time it takes for the water to fall below the bottom of the core once LOCA occurs is θ_S. The time it takes once air cooling begins until the fuel temperature reaches its maximum temperature is θ_e. Thus, the total time, t, starting when the LOCA occurs until the fuel reaches its maximum temperature, is the sum of θ_S and θ_e.

General Atomic conducted a set of LOCA experiments for TRIGA reactors.[18] In these experiments dummy TRIGA fuel elements were electrically heated in a grid to determine the rate of temperature increase of TRIGA fuel elements when cooled by natural air convection. The dummy fuel elements were wound with resistance wire to simulate a cosine distribution similar to that produced in the core. The standard TRIGA grid-plate assembly pitch for a circular (non-hexagonal) core was used with a seven element assembly to mock-up the central portion of a standard core. The LOCA experiments were more conservative for two reasons.[1] The PSBR does not have a central fuel element in the core to block the air flow in the hottest part of the core.[2] In addition, the hexagonal pitch of the PSBR is less likely to produce hot spots on the cladding. When the core is uncovered, the central part of the PSBR core will allow more efficient cooling of the fuel elements in the B-ring increasing the safety factor associated with these calculations.

[redacted]

To calculate the time θ_e, it is necessary to review GA's results as summarized in **Figure 13-4 and 13-5. Figure 13-4** shows that with a constant cosine shape power input of 267 watts, it takes approximately $\theta_e = 300$ minutes before the maximum fuel temperature reaches the equilibrium temperature, T_{eqfuel}, of 600 °C. The maximum fuel temperature attainable, i.e., T_{eqfuel} as a function of the source power in

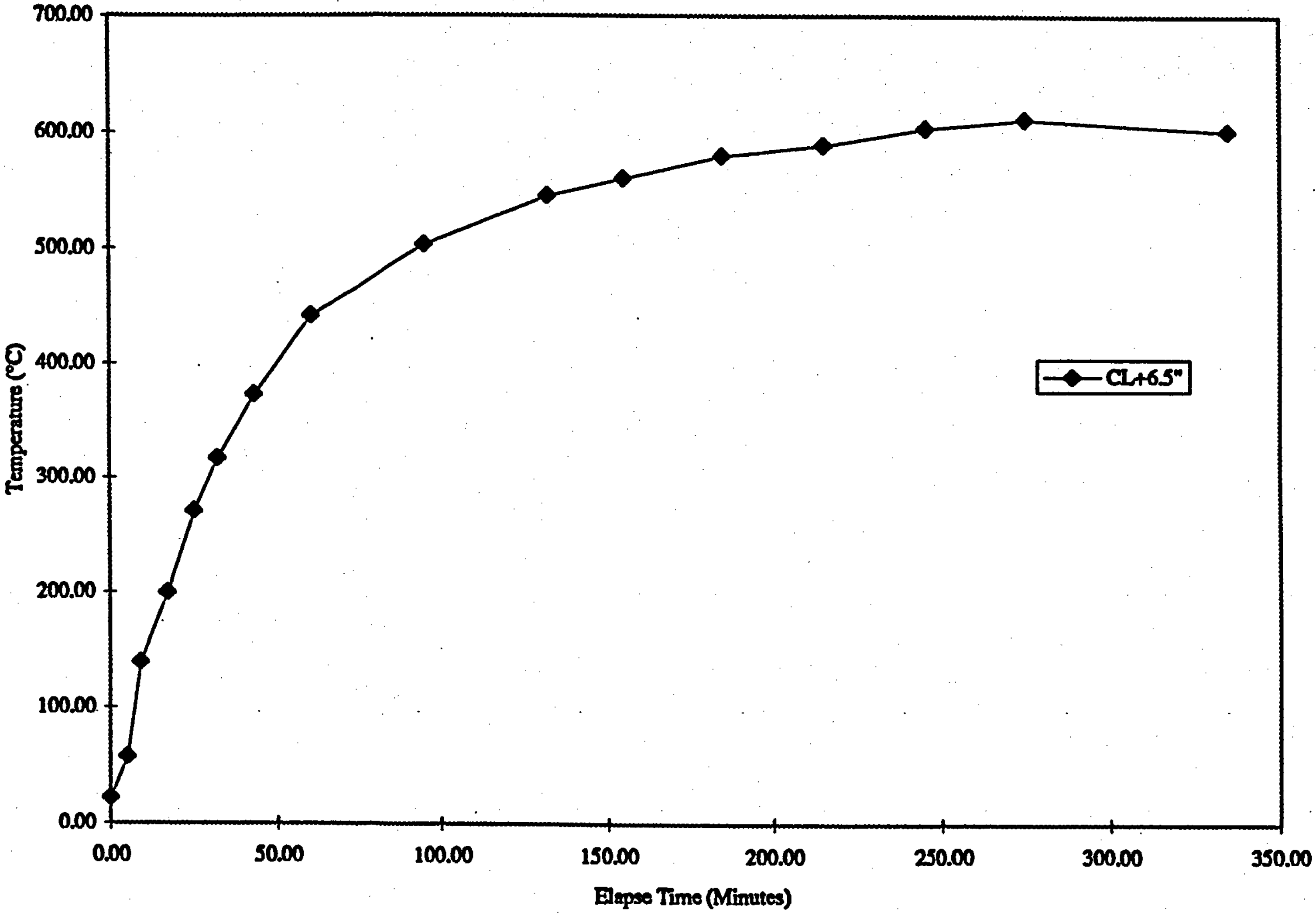

Figure 13-4 The Time Dependence of Air Cooled Fuel Body for Center Element with 267 W Input (Data from Reference 18)

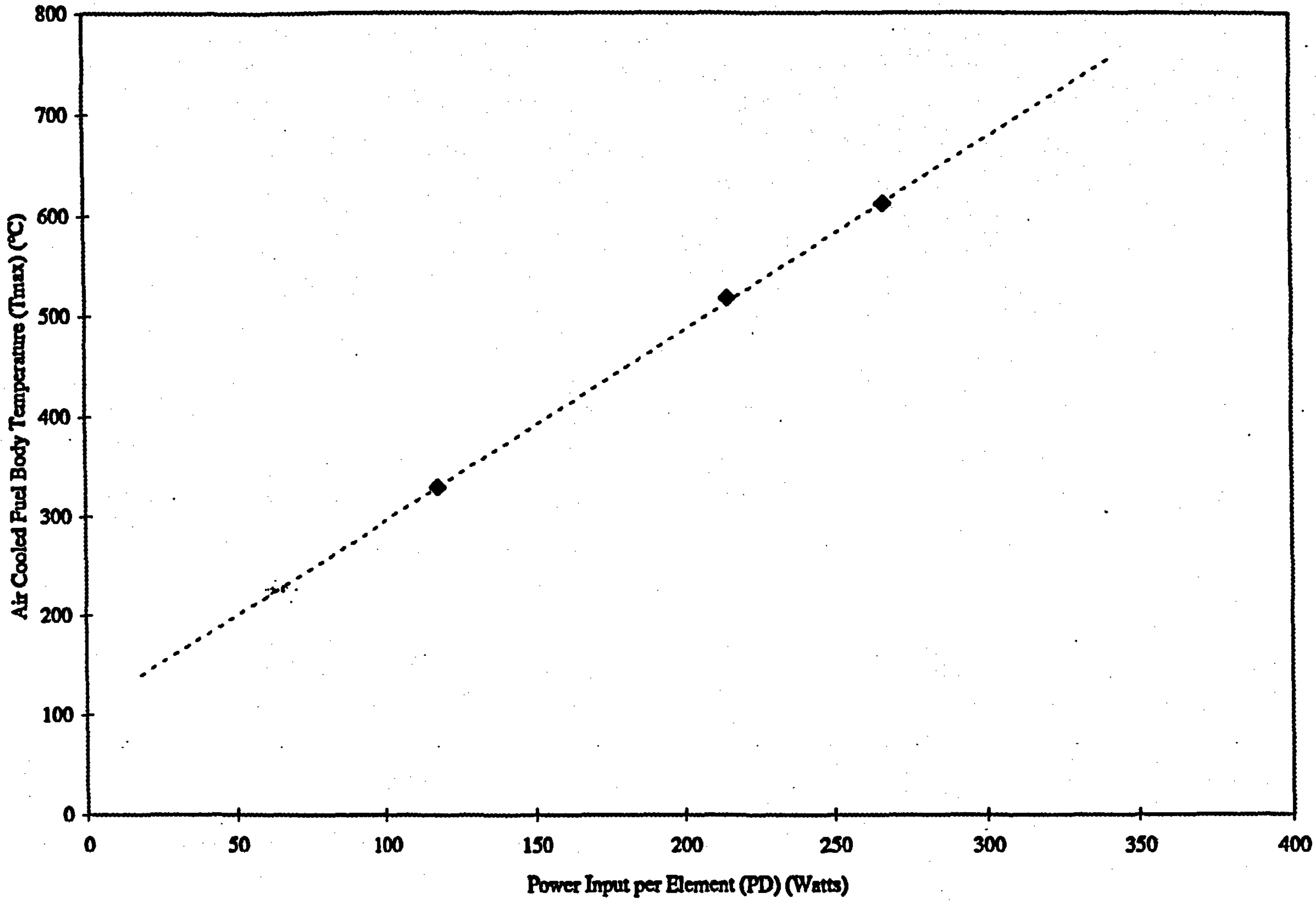

Figure 13-5 Summary of Equilibrium Data for LOCA Simulation Showing the Fuel Element Cladding Temperature versus Power Input to the Element for all Seven Dummy Elements Heated with the Same Power Input (Data from Reference 18)

watts, is given in **Figure 13-5**. The thermal time constant of the fuel after a LOCA is approximately the same for all values of decay power. Thus, it will take 300 minutes once air cooling begins before the fuel temperature reaches T_{eqfuel}. The fitted equation for T_{eqfuel} as a function of θ_e and T_{max} (which in the case of **Figure 13-4** is 600 °C) is as follows:

$$T_{eqfuel} = T_{max} (-8.063 \times 10^{-3} + 2.193 \times 10^{-2}\theta_e - 2.263 \times 10^{-4}\theta_e^2 + 1.193 \times 10^{-6}\theta_e^3 - 3.031 \times 10^{-9}\theta_e^4 + 2.929 \times 10^{-12}\theta_e^5),\ 0<\theta_e<300 \text{ minutes.} \quad (41)$$

Before the maximum fuel temperature reached during a LOCA is determined, the core operation history and maximum fuel element powers must be established. As shown in Figure **13-5**, the maximum fuel temperature reached during a LOCA is directly related to the decay power and the decay power is a function of the pre-LOCA reactor operating history.

The following assumptions are conservative and are used to determine the decay power. The PSBR is licensed for a maximum steady state power of 1 MW(th) and normally operates on a 40 hr/wk schedule. Even when the reactor operates on a two shift schedule for reactor operator instruction or for laboratory experiments, the average power per 40 hr week is much less than 1 MW(th). For the 15 years from 1967-81, the PSBR was operated an average of 15 MW hr/wk. For the 13 years from 1981-94, the PSBR was operated an average of less than 5.5 MW hr/wk. (Update Note: For the 19 years from 1986-2004, the PSBR operated an average of 7.2 MW hr/wk, with a high of 13.5 MW hr/wk in 2003.) Assuming the PSBR operates for 40 hours at 1 MW(th) during the week establishes an upper limit for its operation. However, to cover any future increase in operational activities, it will be assumed that the reactor is operated continuously for one week at 1 MW(th). With this assumption, the fission product decay power can be determined in the following manner.

El-Wakil[14] gives the following equation for decay power:

$$\frac{P_s}{P_o} = \left[0.1(\theta_s + 10)^{-0.2} - 0.087(\theta_s + 2x10^7)^{-0.2}\right] - \left[0.1(\theta_s + \theta_o + 10)^{-0.2} - 0.087(\theta_s + \theta_o + 2x10^7)^{-0.2}\right] \quad (42)$$

where,

P_S = the power after shutdown produced by the fission product decay,

P_O = the steady state power before LOCA, i.e. 1 MW(th),

θ_S = the time after LOCA initiation [redacted],

θ_O = the time of operation at power before LOCA, i.e., 168 hr. x 3600 sec/hr = 6.05×10^5 sec.

Using these values, at time θ_S after LOCA initiation or at the beginning of air convection cooling:

$$\frac{P_s}{P_o} = \text{[redacted]},$$

or if we assume a maximum elemental power density, MEPD, of 24.7 kW, the maximum power due to fission product decay, P_D at time θ_S, is

$$P_D = \frac{P_s 1000kW}{N_p P_o} \quad (42a)$$

$$= \text{[redacted]} \text{ x } 24.7 \text{ kW/fuel element}$$

$$= \text{[redacted]}$$

Assuming the core is uncovered and reaches equilibrium fuel temperature [redacted] after a LOCA, the hottest fuel element in the core will have less than [redacted] of power which will continue to decay.

Figure 13-5 shows the equilibrium fuel cladding temperature as a function of fuel element power input. The data of **Figure 13-5** have been fit with a straight line equation, i.e.,

$$\text{[redacted]}_{max} \text{[redacted]}, \quad (43)$$

where P_D is the decay power (watts) producing temperature in °C.

[redacted] $_{max}$ [redacted].

This shows, that once air cooling begins and if the average decay power remains constant at [redacted] for approximately 4 hours the maximum fuel temperature is 860 °C. However, the decay power does not remain constant but decreases exponentially. By combining Equations (41), (42), (42a), and (43) an equation of T_{eqfuel} as a function of time after the LOCA initiation is obtained. This equation indicates that the fuel reaches a maximum temperature of 468 °C approximately [redacted] after LOCA initiation, with a θ_S [redacted] and then continues to decrease (see **Figure 13-6**). If θ_S is [redacted] the fuel reaches a maximum temperature of 480 °C approximately [redacted] after LOCA and then continues to decrease. For the most severely conceived LOCA for the PSBR, the maximum fuel temperature remains well below the 950 °C limit throughout the LOCA. It should also be stated that the experimental data of **Figure 13-4** shows that because of the long time it takes to heat up a TRIGA fuel element, i.e. θ_e, once air cooling begins, the time θ_S is less critical.

The LOCA for a TRIGA core may also be analyzed analytically instead of by the results of experiments used above. General Atomic used one of their own two dimensional transient-heat transport computer codes to calculate the TRIGA fuel element temperature after the loss of pool water.(19) It was assumed that the reactor was operating for an infinite period of time. Their results are plotted in **Figure 13-7** for several cooling or delay times showing maximum fuel temperatures in

[redacted]

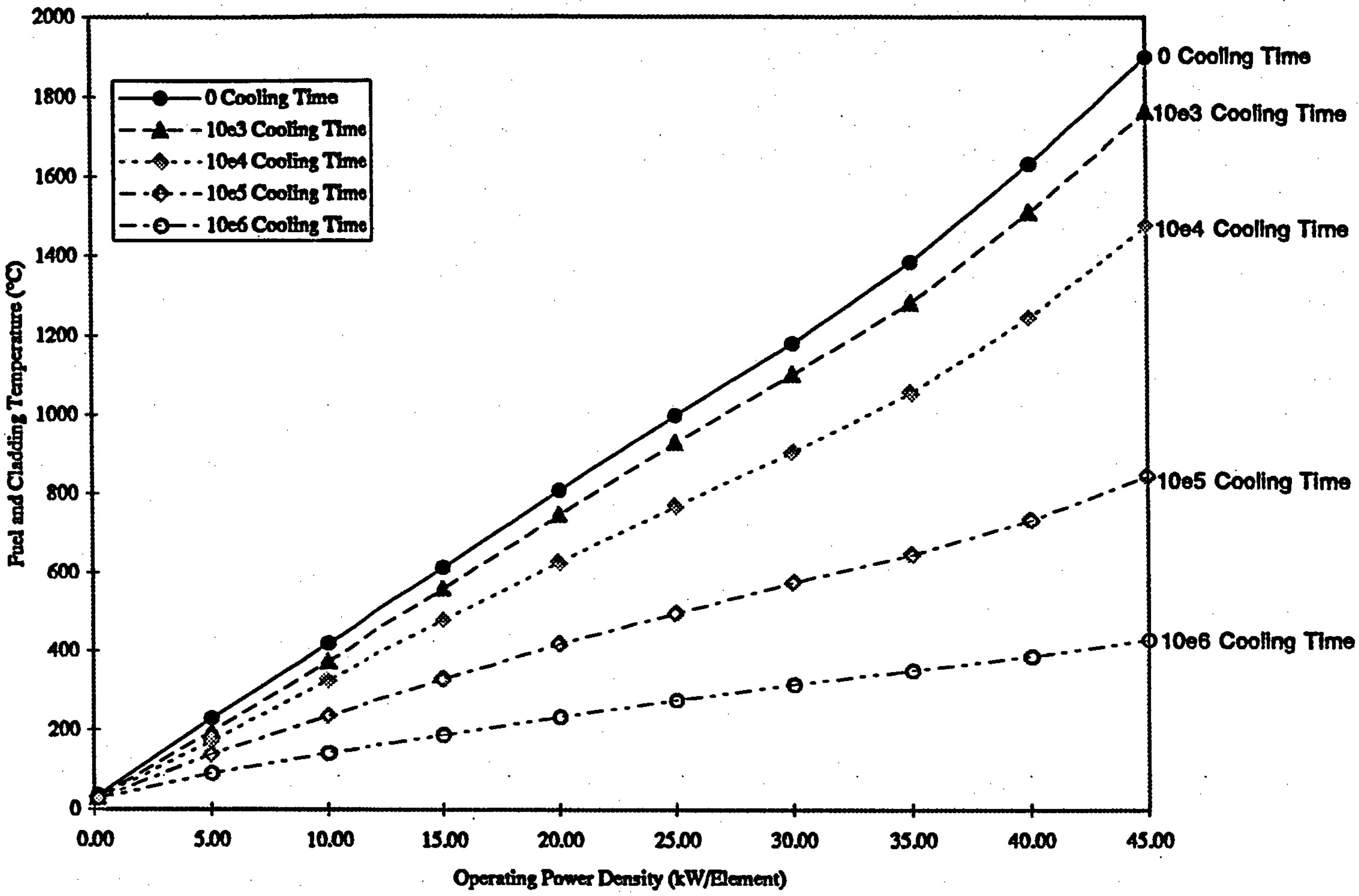

Figure 13-7 Maximum Fuel Temperature Versus Power Density After LOCA for Various Cooling Times Between Reactor Shutdown and LOCA Initiation (Data from Reference 19)

the TRIGA fuel element as a function of its operating power. It can be observed that a fuel element having approximately 24.7 kW before the LOCA, will attain < 950 °C maximum temperature, [redacted] after the LOCA.

The 950 °C maximum fuel temperature is important when analyzing the TRIGA core for a LOCA. During a LOCA, the fuel element is uncovered producing cladding temperatures greater than 500 °C. Under these conditions, if 950 °C fuel temperature and 950 °C cladding temperature is reached or exceeded, the TRIGA cladding could be ruptured.[13, 30] Below a fuel temperature of 950 °C the cladding remains intact. The strength of the fuel element cladding is a function of its temperature. The yield strength of the stainless steel cladding (assuming that the cladding design specifications of the TRIGA fuel elements do not change between procurements from the as-tested cladding) under LOCA conditions (heated in air), is shown in **Figure 13-8.** Also shown in **Figure 13-8** is the cladding stress produced by any gas in the gap. This gas pressure consists of the hydrogen gas pressure plus the pressure of the volatile fission products plus the pressure of the trapped air. It can be observed that the cladding stress equals the cladding yield strength at approximately 950 °C. This is different from the safety limit (1150 °C) which is the case when the cladding is in water and cladding temperatures remain below 500 °C.

The following results are applicable to the PSBR experiencing a LOCA:

1. The maximum temperature that the PSBR TRIGA fuel element can have during a LOCA without damage to the cladding is 950 °C. As long as the 950 °C temperature limit is not exceeded, there will be no stress sufficient to rupture the cladding thereby allowing the escape of fission products.

2. Reviewing the complete history of the TRIGA cores at Penn State, the PSBR has never been operated for 40 hours at 1 MW(th) during a week with the 12 wt% U fuel. In addition, no core configuration is permitted to operate at a power level such that the MEPD is greater than 24.7 kW. Assuming continuous operation at 1 MW(th) with the 12 wt% U fuel producing 24.7 kW, a LOCA will not cause the fuel element to heat up to 950 °C under any condition.

In conclusion, a LOCA with the PSBR will not result in damaged fuel and, thus, fission product containment within the fuel is assured.

13.1.4 Loss of Coolant Flow

Not applicable to the PSU TRIGA that is cooled by natural convective flow.

13.1.5 Mishandling or Malfunction of Fuel

Any fission product release caused by mishandling or malfunction of fuel would be bounded by the MHA analysis in section 13.1.1, that assumes a fuel element cladding ruptures in an air cooled core releasing volatile fission products. The method by which this occurs is not described.

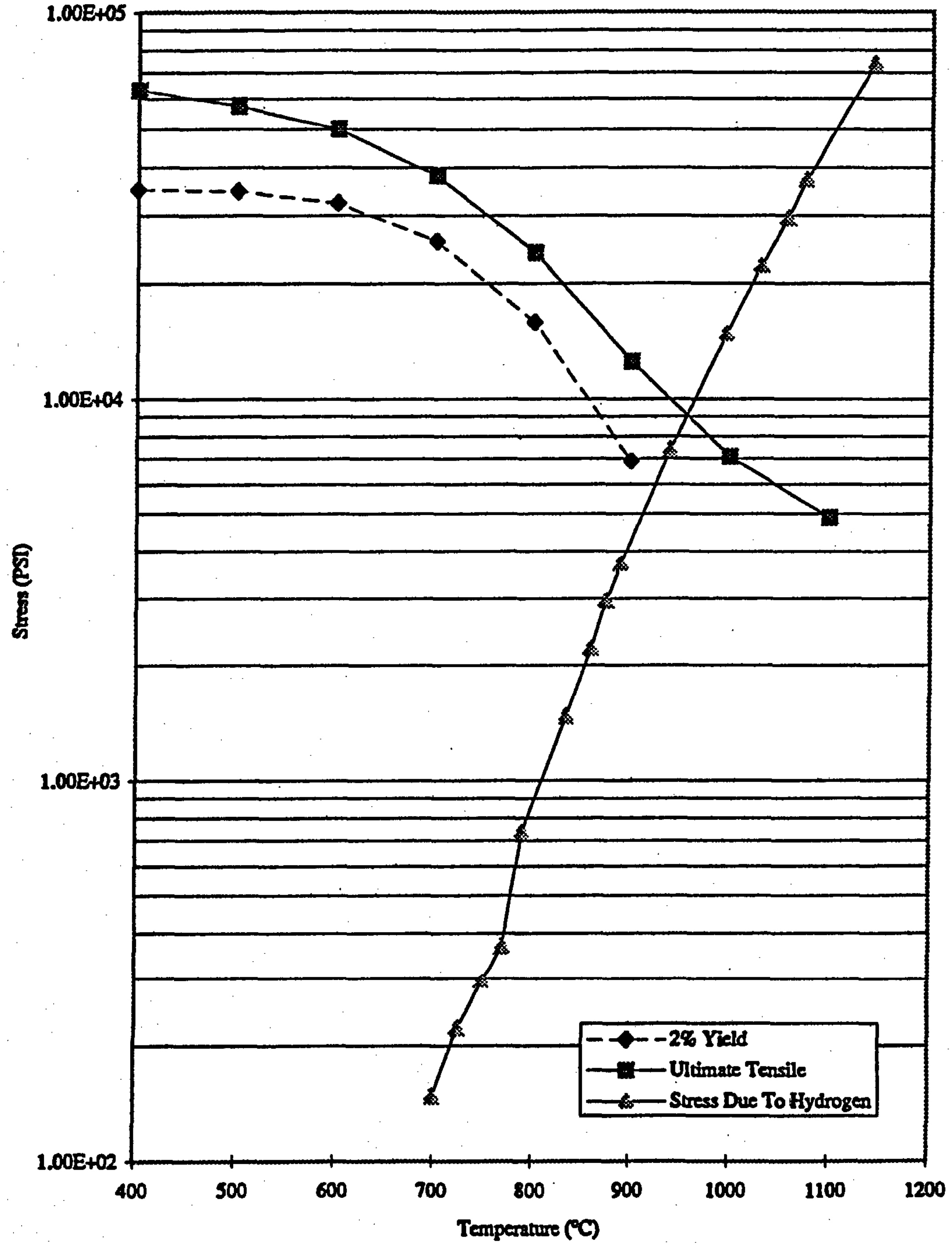

Figure 13-8 Strength and Applied Stress as a Function of Temperature, U-ZrH1.65 Fuel With Fuel and Cladding the Same Temperature (Data from Reference 13)

The PSBR redesigned the fuel handling tool many years ago to improve on the original General Atomic design following the dropping of several fuel elements. No elements have been dropped using the re-designed tool. All elements that have been dropped are removed from service.

The PSBR facility has no history of leaking fuel elements.

13.1.6 Experiment Malfunction

There are no current experiments being conducted whose malfunction or failure could result in a significant accident scenario that would in any way approach the MHA. Section 10.3 of this document discusses the experimental review process at the PSBR to assure proper experiment reviews, including 50.59 reviews when applicable.

13.1.7 Loss of Normal Electrical Power

There is no accident scenario initiated as a result of the loss of on-site electrical power. See Chapter 8 for a description of electrical power for the facility.

13.1.8 External Events

No accident initiators have been identified that could cause damage to the reactor core.

13.1.9 Mishandling or Malfunction of Equipment

Since the TRIGA reactor is designed to safely handle a large reactivity insertion associated with a pulse, no operator errors, Reactor Safety System (RSS), or Protection, Control or Monitoring System (PCMS) failures would initiate an accident beyond that analyzed in the MHA.

[redacted]

13.2 Accident Analysis and Determination of Consequences

The only accident that has consequences is the Maximum Hypothetical Accident as discussed in section 13.1.1. Analysis and consequences are discussed there.

13.3 Summary and Conclusions

There are two limits that, if not exceeded, will prevent rupture of the cladding of a TRIGA fuel element. They are:

1. Limit the fuel temperature to a maximum 1150 °C when the cladding temperature remains below 500 °C, i.e., when the fuel is covered with water.

2. Limit the fuel temperature to a maximum 950 °C when the cladding temperature is the same as the fuel temperature i.e., as with an air cooled core after a LOCA.

The TS for the PSBR are established to prevent reaching these two limits. The 1150 °C temperature limit is not reached as the fuel temperatures are limited during pulse mode operations. Equation (34) provides a direct method for determining the maximum fuel temperature based on the measured fuel temperature during a pulse. Using this equation the following limits are established:

1. The maximum allowed reactivity insertion for the pulse mode and the maximum allowed worth of the pulse rod is \$3.50. A sudden insertion of \$3.50 excess reactivity results in a maximum peak fuel temperature of 1095 °C and a measured peak fuel temperature of 684 °C if the NP $\leq$ 2.2.

2. With any core loading the maximum radial peaking factor, called the normalized power, NP, in the SAR is 2.2 if the transient rod worth is \$3.50. This ensures that a pulse with the full travel of the maximum allowed transient rod worth will not cause the fuel temperature in any fuel element to exceed the safety limit of 1150 °C. If the maximum allowed pulse is less than \$3.50 for any given core loading (i. e. the pulse can be limited by the worth of the transient rod, by the core excess, or administratively) the maximum NP can be increased as long as a calculation by an accepted method (documented in an administratively approved procedure) is done to show that the safety limit is not exceeded with the allowed pulse and NP. The limits shall be either physical or administrative or both.

3. The maximum allowed excess reactivity of the core is \$7. Thus, when the core is operating at 1.15 MW(th) steady state, a maximum of \$2.25 of excess reactivity is available for pulsing, a minimum of \$4.75 of excess reactivity is needed to reach 1.15 MW(th). Based on core loading 47 and I-15 (an extrapolated measured fuel temperature of 650 °C) at 1.15 MW(th) prior with a pulse insertion of \$2.25 (measured fuel temperature of 380 °C) the temperature would equal 1030 °C. However the Technical Specification required interlock prevents pulse initiation when the power is above 1 kW and the Technical Specification required high power SCRAMs prevent operation above 1.1 MW(th).

4. Core configuration limitations are also established to prevent a fuel element from producing too much power relative to the other fuel elements. The maximum elemental power density, MEPD, allowed is 24.7 kW. If core size and or NP leads to a MEPD greater than 24.7 kW when the reactor power is 1 MW(th) the maximum allowed reactor power must be administratively reduced to reduce the MEPD to 24.7 kW or less. The maximum allowed reactor power to maintain the MEPD less than 24.7 kW for a given core configuration shall be determined by calculation by an accepted method (documented in an administratively approved procedure).

   Limits set for steady state operation prevent the maximum fuel temperature reaching 1150 °C. Limits imposed here prevent the fuel temperature during a LOCA from reaching 950 °C. If operated at 1 MW(th) continuously, a single fuel element could operate at its maximum power level of 24.7 kW and still not have its fuel temperature reach 950 °C during any conceived LOCA. In addition, the 24.7 kW MEPD and the maximum measured fuel temperature of 650 °C during steady state operation limit the release of fission products such that the consequenses are within the 10 CFR Part 20 limits if the cladding ruptures.

The MHA analyzes the effect of a fuel element cladding rupture in air after continuous operation at 1 MW(th). In addition, the reactor was assumed to have operated continuously during the previous year. [redacted]

In conclusion, the analyses described in this section shows that under no possible accident conditions will the regulations in either 10 CFR Part 20 or 10 CFR Part 100 be violated. Thus, the PSBR can be operated safely within the regulatory limits.

13.4 Bibliography

1. Naughton, W. F., Cenko, M. J., Levine, S. H., and Witzig, W. F., "TRIGA Core Management Model, "Nucl Technology, vol 23, p. 256 (Sept 1974).

2. Naughton, W. F., Cenko, M. J., Levine, S. H., and Witzig, W. F., "Increasing TRIGA Fuel Lifetime with 12 wt% U TRIGA Fuel," TOC-5, TRIGA Owner's Conference III (February 1974).

3. Haag, J. A., and Levine, S. H., "Thermal Analysis of The Pennsylvania State University Breazeale Nuclear Reactor," Nucl. Technology, vol. 19, p.6 (July 1973).

4. Levine, S. H., Geisler, G. C., and Totenbier, R. E., "Temperature Behavior of 12 wt% U TRIGA Fuel," TOC-5, TRIGA Owner's Conference III (February 1974).

5. Levine, S. H., Totenbier, R. E., (Penn State Univ.) and Ahmad T. Ali (PPAT Ismail-Malysia), "Fourteen Years of Fuel Management of the Penn State TRIGA Breazeale Reactor (PSBR)," ANS Trans, vol. 33 (November 1979).

6. Levine, S. H., and H. Ocampo, "The k∞-Meter Concept Verified via Subcritical/Critical TRIGA Experiments, " Proceedings of the International Symposium on the Use and Development of Low and Medium Flux Research Reactors," MIT, Cambridge, MA (October 1983).

7. Kim, S. S., and S. H. Levine, "Verifying the Asymmetric Multiple Position Neutron Source (AMPNS) Method Using the TRIGA Reactor," Ninth TRIGA User's Conference, Anaheim, CA (March 1984).

8. Levine, S. H., "Module 5 - In-core Fuel Management," Nuclear Fuel Cycle Educational Module Series, N. D. Eckhoff, gen. Ed., Kansas State University (July 1980).

9. Fowler, T. B., et.al., "EXTERMINATOR-II: A FORTRAN IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions, "ORNL-4078, Oak Ridge National Laboratory (April 1967).

10. Huang, H. Y. and S. H. Levine, "An Automated Multiple-Cycle PWR Fuel Management Code," ANS Trans. (November 1978).

11. Cenko, M. J., "Comparison of PSBR Operation's History with the TRIGA Core Management Model, "M. S. Thesis, The Pennsylvania State University (1972).

12. Barry, R. F., "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for IBM-7094," WCAP-3269-26, Westinghouse Electric Corporation (September 1963).

13. Simnad, M. T., F. C. Foushie, and G. B. West, "Fuel elements for Pulsed Reactors," GA Report E-117-393 (January 1975A).

14. El-Wakil, M. M., "Nuclear Heat Transport," ANS (May 1978).

15. Goodwin, W. A., "The Measurement of Radial Power Distribution in a TRIGA Fuel Element During Reactor Excursion," Ph.D. Thesis, University of Illinois (1967).

16. Kim, S. S., "Development of an Asymmetric Multiple Position Neutron Source (AMPNS) Method for Monitoring the Criticality of the Degraded Reactor Core," Ph.D. Thesis, The Pennsylvania State University (1984).

17. PSBR Log Book 37, page 265 (November 21, 1984).

18. Shoptaugh, J. R., Jr., "Simulated Loss-of-Coolant Accident for TRIGA Reactors," GA-6596 (August 1965).

19. West, G. B., "Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor," GA-9064 (January 5, 1970).

20. Katcoff and Seymour, Nucleonics, vol. 18, p. 201 (November 1960).

21. Foushee and R. H. Peters "Summary of TRIGA Fuel Fission Product Release Experiments," GULF-EES-A10801 (September 1971).

22. Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, App. I."

23. International Commission on Radiation Protection Report #23.

24. General Atomics, *Analysis of 5 Dollar Ramp Insertion over 2 Second Interval in AFRRI TRIGA™ Reactor,* General Atomics Publication of work performed for Armed Forces Radiobiological Research Institute, Bethesda, Maryland, April 1988.

25. Coffer, C. O., Dee, J. B. Shoptaugh, Jr., J. R., West, G.B., and Whittemore, W. L., "Characteristics of Large Reactivity Insertions in a High Performance TRIGA U-Zr-H Core," GA-6216, General Atomics (April 12, 1965).

26. W. F. Naughton, "Core Management Program to Optimize Fuel Utilization in TRIGA Research Reactors," Ph.D. Thesis, Nuclear Engineering Department, The Pennsylvania State University, University Park, Pennsylvania (September 1972).

27. D. Hughes, P. Boyle, and S. H. Levine, "A New Management Plan for the Penn State TRIGA Reactor with Supporting Experiments and Calculations," Unpublished paper of the Nuclear Engineering Department, The Pennsylvania State University, University Park, Pennsylvania (March 1996).

28. NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," Docket No. 50-163, U. S. Nuclear Regulatory Commission (August 1987).

29. M. T. Simnad, G. B. West, J. D. Randall, W. J. Richards, and D. Stahl, "Interpretation of Damage to the Flip Fuel During Operation of the Nuclear Science Center Reactor at Texas A&M University," GA-A16613, General Atomics (December 1981)

30. M. T. Simnad, "The U-ZrHx Alloy: Its Properties and Use in TRIGA Fuel," GA E-117-833, General Atomics (February 1980)

31. Stationary Neutron Radiography System Final Safety Analysis Report prepared by Argonne National Laboratory, McClellan Air Force Base (January 1992).

14. TECHNICAL SPECIFICATIONS

It is proposed that the existing TS through amendment #37 be considered as part of the license renewal application. There is one proposed change to the TS. In definition 1.1.34.g., the word "containment" is changed to "confinement". The present wording is that from ANSI/ANS-15.1-1990. The proposed change reflects the fact that the PSBR has a confinement but not a containment.

14.1 Biblography

Standard ANSI/ANS-15.1-1990, "The Development of TS for Research Reactors", American Nuclear Society, Reaffirmed December 14, 1999

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

Not applicable to PSBR since it is an existing facility and no facility changes are addressed in this application for license extension.

15.2 Financial Ability to Operate a Non-Power Reactor

The PSBR, the principal facility within the Penn State Radiation Science and Engineering Center (RSEC), is administered for The Pennsylvania State University (PSU) by the College of Engineering (COE). The RSEC operations and maintenance (O&M) expenditures for the 2004/2005 fiscal year were approximately $1.4M. The O&M expenditures over the preceding past five years have varied between $1.1M and $1.7M. For the 2004/2005 fiscal year approximately $1.1M was associated with salaries and student support, the remaining $0.3M was for supplies, materials, and expenses. O&M expenditures for the future are expected to be consistent with the experience of recent years.

The COE provides about $600K per year and the remainder of the expenditures is funded by income derived from the conduct of the RSEC missions of education, research, and service. Fringe benefits, university overhead, utilities, routine building maintenance, and other infrastructure expenditures as well as radiological protection/health physics services (including normal waste disposal costs) through the Radiation Protection Office are provided by PSU and do not show as part of the RSEC budget or expenditures. Funding and support are expected to continue in this manner in the future. The commitment of PSU to the continuing funding and support of the PSBR is amply demonstrated by the submittal of this application for license extension.

The O&M expenditures over the past several years have included about $1M in facility improvements including: shop expansion, storage building construction, lobby expansion, reactor console computer upgrade, gamma spectroscopy systems and sample changer, classroom renovations, and a number of other miscellaneous items. The physical improvements are a strong indication of the commitment to continuing safe, viable, and productive operation of the PSBR.

Support for continuing research improvements is also provided by the DOE University Research Instrumentation (URI) grant program and by the DOE Innovations in Nuclear Infrastructure and Education (INIE) program. The DOE funding is highly desirable but is not required for continuing safe operation.

The fifty years of operation of the PSBR has amply demonstrated that the combination of direct and indirect funding support from the PSU coupled with income derived from education, research, and service work is not only adequate for supporting safe day-to-day operation and maintenance but also capable of providing for growth in facilities, staff, and usage.

15.3 Financial Ability to Decommission the Facility

As required by 10 CFR Part 50.75, the University has on file with the NRC a Penn State University Decommissioning Funding Plan for all University facilities operating under NRC licenses, including the reactor R-2 license. The most recent plan is dated October 2003. The PSBR meets the financial tests and self-guarantee for decommissioning in accordance with 10CFR30, Appendix E. Funding levels will be reviewed and adjusted as appropriate every three years.

15.4 References

Penn State University Decommissioning Funding Plan for Radiation Research Facilities, December 2003, prepared by Radiation Safety Officers Eric J. Boeldt and Kenneth L. Miller.

Letter from Pamela J. Henderson, Chief, Nuclear Materials Safety Branch I, Division of Nuclear Materials Safety, Nuclear Regulatory Commission to Eva Pell, Penn State Vice President for Research and Dean of the Graduate School, subject: "The Pennsylvania State University, Review of Financial Assurance Submittal, Control No. 134216."

Self Guarantee dated December 17, 2003, submitted with Decommissioning Funding Plan dated December 2003.

Certification of Financial Assurance dated April 21, 2004, from Kenneth S. Babe, Penn State Corporate Controller, to U. S. NRC Region I.

16. OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

There are no components in use at the PSBR that have had prior use at any other facility or organization.

16.2 Medical Use of Non-Power Reactors

The PSBR is not engaged in any activities using special nuclear materials for medical therapy.

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